

A Preliminary Assessment of Core Melt  
Probability In Cold Shutdown  
Following a Postulated LOCA  
at the Sequoyah Nuclear Plant

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## 1. INTRODUCTION

This report presents the results of an analysis to conservatively estimate the probability of a core melt accident in cold shutdown (Mode 5) initiated by a postulated loss of coolant accident (LOCA) at the Sequoyah Nuclear Plant. The terms "cold shutdown" and "Mode 5" are synonymous and, at Sequoyah, refer to the plant state when: (a) average reactor coolant temperature is less than or equal to 200°F, (b) the reactor is shutdown with  $K_{eff}$  less than 0.99, (c) percent rated core thermal power, excluding decay heat, is zero, and (d) the reactor vessel head bolts are fully tensioned (Ref. 1).

Event trees were used to define the Sequoyah plant response to small, medium and large LOCAs. Simplified fault trees further defined the equipment failures and operator errors that were associated with each event in the event trees. Using data mostly from WASH-1400 (Ref. 2), the fault and event trees were quantified by means of hand calculations.

A total of 20 cases were analyzed with varying assumptions regarding the LOCA initiating event (safe shutdown earthquake or operator error), time of LOCA initiation following reactor shutdown, LOCA size, availability of offsite power during a 100-hour period following the LOCA, and maintenance status. With no significant maintenance in progress at the time of LOCA initiation, the probability of core melt in cold shutdown for these 20 cases was estimated to be in the range from  $3.96 \times 10^{-5}$  to  $1.14 \times 10^{-7}$  per reactor-year. If maintenance affecting one electrical or cooling water support system train is in progress, the probability of core melt for the 20 cases increases and is estimated to be the range from  $7.53 \times 10^{-5}$  to  $8.46 \times 10^{-6}$  per reactor-year. In contrast, an overall core melt probability of  $6 \times 10^{-5}$  per reactor-year was reported in WASH-1400 (Ref. 2). It could be noted, however, that the estimates of core melt probability presented in this report are based on a number of assumptions and simplifications which, taken as a group, should yield very conservative results. The estimates of core melt probability in Mode 5 should therefore be considered as upper limits. A more detailed analy-

sis would most likely indicate that the probability of core melt in Mode 5 is significantly lower.

## 2. TRANSITION TO COLD SHUTDOWN FROM POWER OPERATION

Cold shutdown (Mode 5) is usually a transitional plant operating state between power operations (Mode 1) and refueling (Mode 6). A representative plot of reactor coolant average temperature versus time during a shutdown and cooldown is shown in Figure 2-1 (from Ref. 3).

Following reactor shutdown, initial cooldown and depressurization of the reactor coolant system (RCS) is accomplished by using the steam generators to transfer heat to the main condenser (e.g., by dumping steam) and pressurizer spray and heaters to control RCS pressure. Four hours after reactor shutdown and initiation of cooldown, the Sequoyah RCS nominally will be at 350°F and 425 psig (Ref. 4). At this point the residual heat removal (RHR) system can be placed in operation to continue the cooldown.

At Sequoyah, there is an administrative cooldown limit of 50°F per hour using the RHR system, although a maximum cooldown rate of 100°F per hour is permitted by the Technical Specifications (Ref. 1). A design basis for the RHR system is to cool the RCS from 350°F to 140° in 16 hours (e.g., 20 hours after reactor shutdown). Considering the administrative cooldown limit and the need for a period of time to accomplish the transition to RHR cooling, it is likely that the Sequoyah plant could be in Mode 5 as soon as eight to ten hours after reactor shutdown.

As a point of comparison to other pressurized water reactor (PWR) plants, Figure 2-2 (from Ref. 3) and Figure 2-3 (from Ref. 5) respectively illustrate the average RCS temperature during cooldown of a B&W and a C-E plant. From these figures, it is estimated that six hours after reactor shutdown is the earliest time at which a PWR plant will enter Mode 5.

It should be noted that cold shutdown may also be a steady-state plant condition during maintenance, testing, inspection or repair activities that cannot be conducted during other plant operating modes. Such activities include reactor coolant pressure boundary repairs (e.g., steam generator repair or replacement, pipe crack repairs) which may require the extended maintenance of a cold shutdown condition.

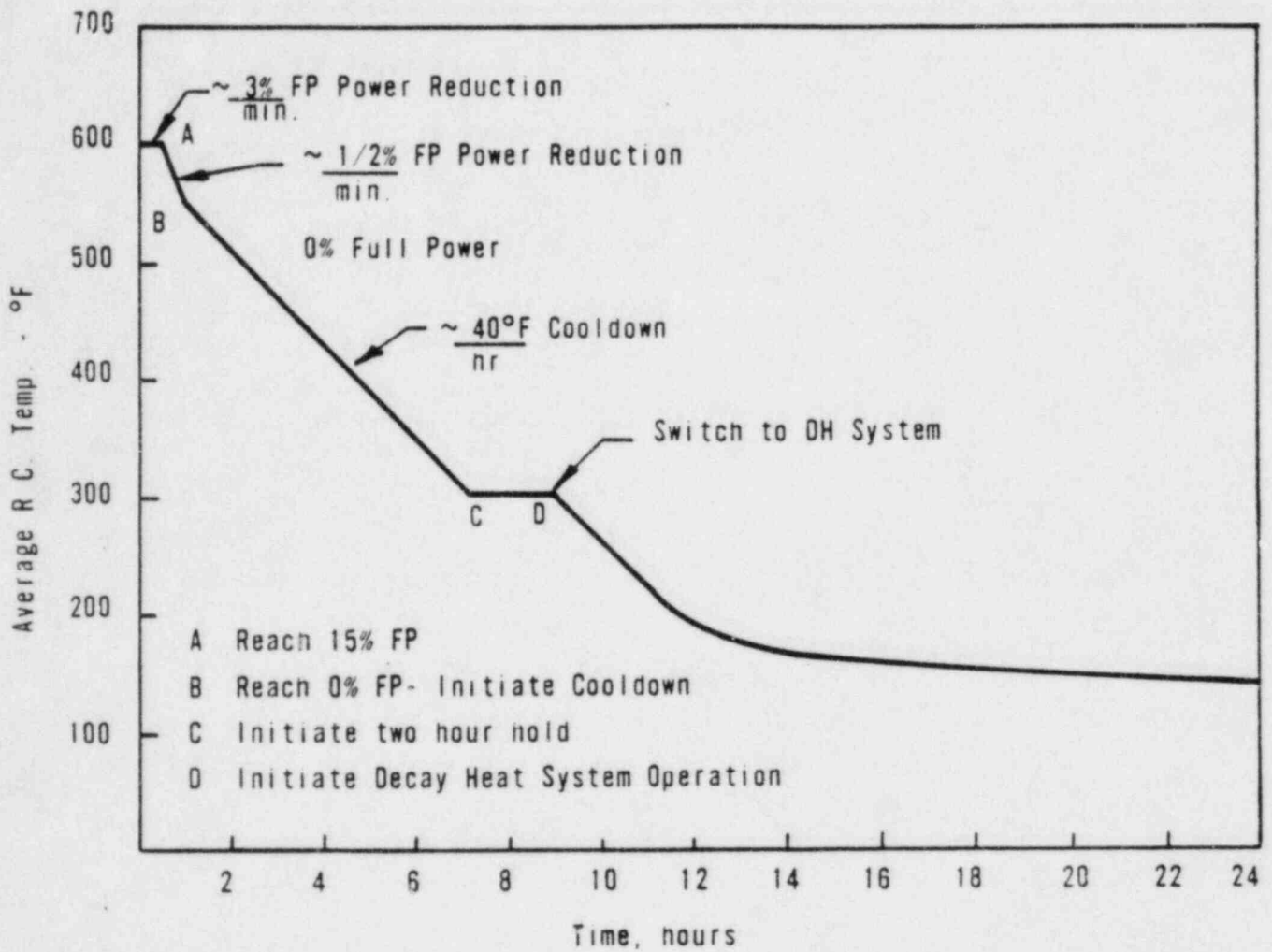


Figure 2-1. Reactor Coolant Temperature vs. Time Plot for Typical Shutdown From Full Power and Cooldown (from B-SAR-205).

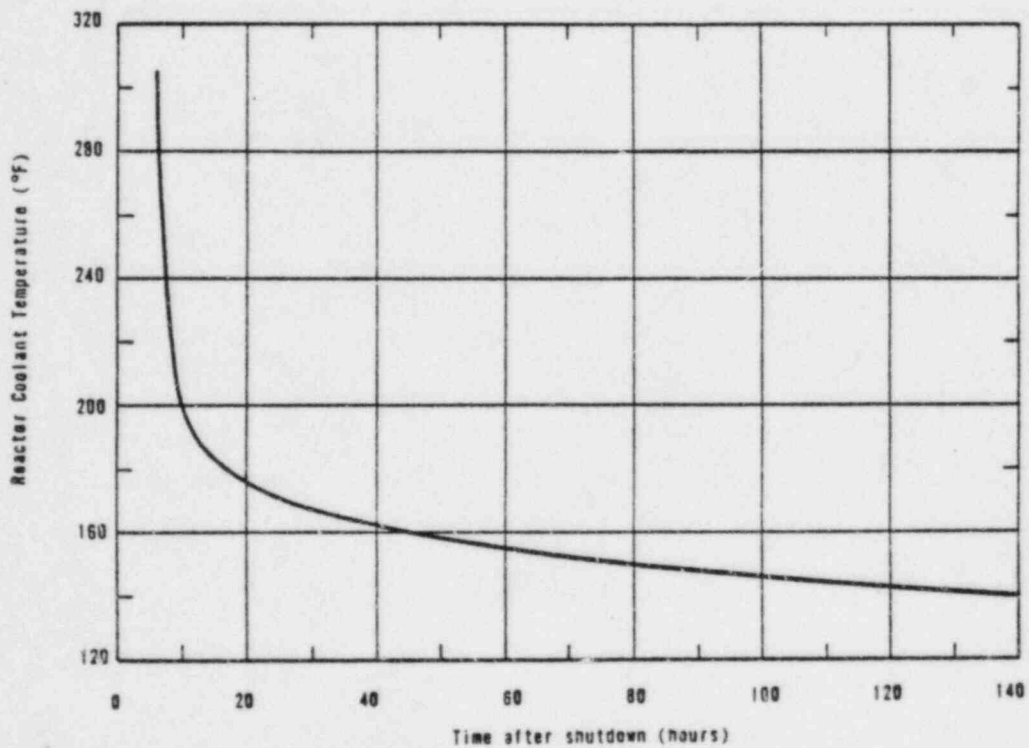
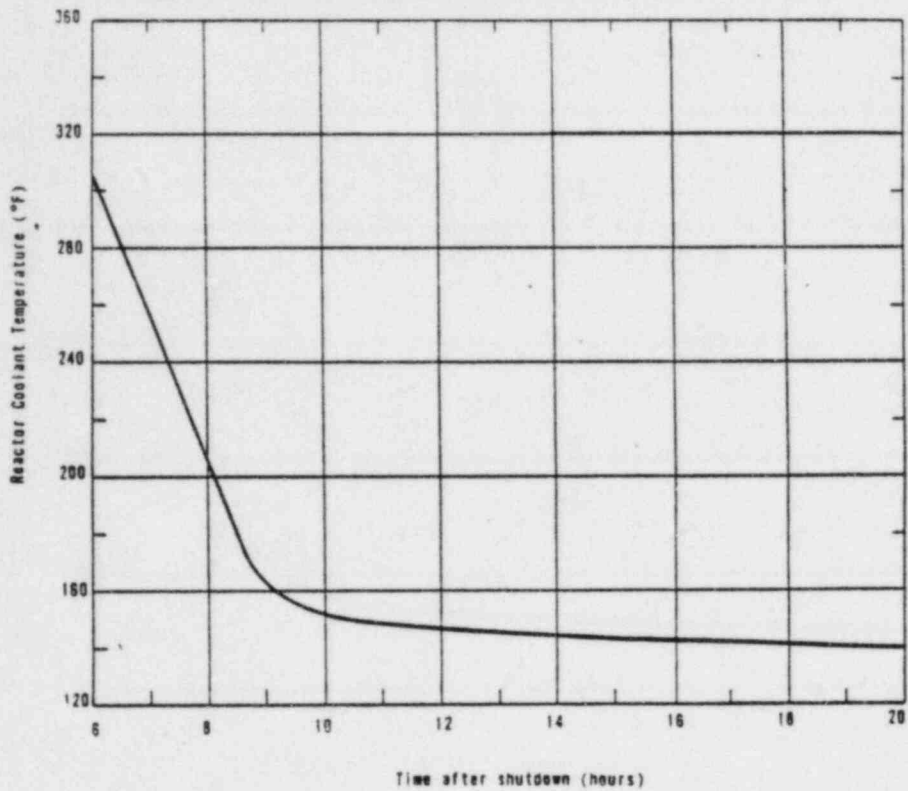


Figure 2-2. B & W Reactor Coolant Temperature Following Reactor Shutdown With Two RHR Loops in Operation (top curve) and With One RHR Loop in Operation (bottom curve). (From B-SAR-205)



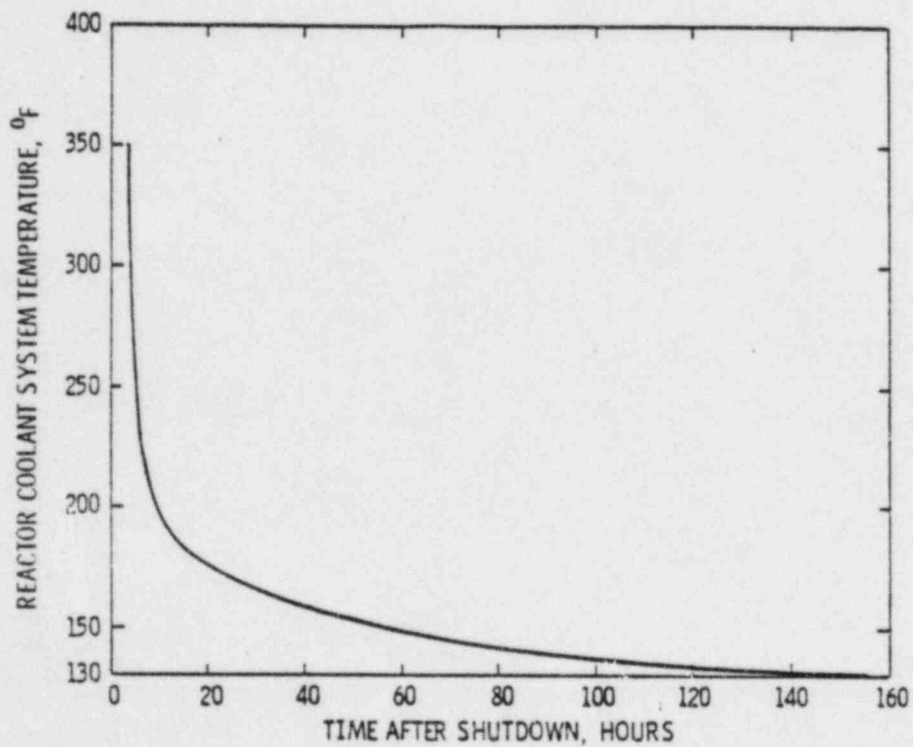
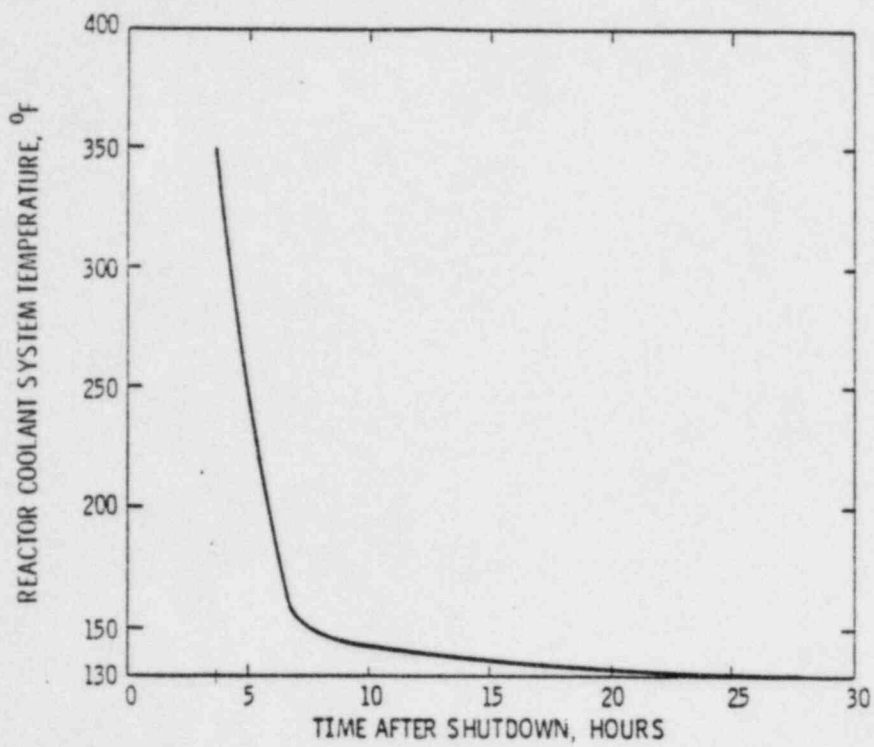


Figure 2-3. C-E Reactor Coolant Temperature Following Reactor Shutdown With Two RHR Loops in Operation (top curve) and With One RHR Loop in Operation (bottom curve). (From San Onofre 2 & 3 FSAR)

### 3. POTENTIAL PWR SAFETY CONCERNS RELATED TO A LOCA OCCURRING DURING COLD SHUTDOWN

In comparison to power operation, cold shutdown is a relatively benign plant operating mode. The reactor core is already shutdown, decay heat is less than one percent of full power and the stored energy in the RCS has been largely dissipated during cooldown and depressurization. There are, however, a variety of safety concerns that may be associated with a LOCA that is initiated while the plant is in cold shutdown. Included are the following:

- Dependence on the operator for initiation of protective actions,
- Core decay heat removal and coolant inventory control requirements following LOCA,
- RCS and RHR overpressure protection requirements during response to some LOCAs with high-head coolant injection pumps,
- Criticality control requirements if extended makeup is provided to the RCS from an inadequately borated water source,
- Relaxed containment integrity requirements in cold shutdown.

This section describes potential LOCA sources during cold shutdown and provides background information related to each of the potential safety concerns listed above.

#### 3.1 DESCRIPTION OF POTENTIAL MODE 5 LOCAs

Potential Mode 5 LOCA sources are summarized in Table 3-1. The random pipe break LOCAs listed in this table are also potential LOCA sources in other modes of plant operation. Most of the LOCAs due to system mis-

alignment by an operator are unique to periods when: (a) the RHR system is in operation, or (b) personnel are performing testing or maintenance inside containment. A LOCA due to stuck-open RHR shutdown cooling suction line safety valve can only occur when the RHR system is aligned for shutdown core cooling. A LOCA could also be induced by an unmitigated pressure transient and failure of overpressure protection systems. In Mode 5, upper limits for operating pressure are established by the RHR system design pressure (600 psig) and by RCS temperature and pressure limits which will be discussed in detail in Section 3.4. When RCS cooldown or heatup is not in progress, the RHR system pressure limit is more restrictive than the RCS temperature-pressure limits down to an average coolant temperature of approximately 125°F.

For the purpose of analysis, Table 3-2 defines three Mode 5 LOCA categories: large, medium and small. The flow rate selected as the dividing point between medium and large LOCAs is related to the capacity of the high-head coolant injection pumps at RHR system design pressure. This flow rate would be approximately equal to the respective pump runout flow rates listed in Table 4-1 (e.g., 550 to 650 gpm). If coolant loss from the RCS at 600 psig would be equal to or greater than the capacity of the respective high-head pump, the RCS cannot be pressurized above RHR system design pressure and further degradation of RCS pressure boundary integrity due to overpressurization would not be expected. If, on the other hand, the coolant loss from the RCS at 600 psig is significantly less than the capacity the respective high-head pump, the RCS could be pressurized to a point where a balance is achieved between pump capacity (decreases as RCS pressure increases) and coolant loss via the LOCA (increases as RCS pressure increases). The potential may exist for further RCS pressure boundary degradation due to overpressurization. This subject is discussed in more detail in Section 3.4.

The flow rate selected as the dividing point between medium and small LOCAs is related to the decay heat rate expected following extended operation of the reactor at full power. See Section 3.3 for details regarding Sequoyah decay heat generation rates.

To gain some insight into the break sizes that fall into each LOCA category, calculations were performed to estimate the leak rate from various size pipe breaks when the RCS was at 600 psig. The code listing and detailed results of the calculations are included in Appendix B. A summary of these results are listed in Table 3-3 and are illustrated graphically in

Figure 3-1. For each size pipe break, two different leak rates are listed in Table 3-3. The first value is for a guillotine break that occurs one foot down a smaller diameter pipe that is connected to RCS loop piping. For this postulated LOCA geometry, kinetic or "entrance" losses have a greater impact on leak rate than frictional losses. Choke flow conditions were assumed not to exist thus, the calculated flow rates are the maximum possible flows. The second value in Table 3-3 is for a guillotine break that occurs 100 feet down a straight, smaller diameter pipe that is connected to RCS loop piping. As can be seen in Table 3-3, frictional losses due to flow in the long smaller diameter pipe can have a significant effect on a LOCA leak rate. The effect is greatest in the smallest diameter piping where the leak rate is a factor of 3 to 4 less than the leak rate from a one foot stub pipe.

For a guillotine break one foot from the interface with the RCS loop piping, the three Mode 5 LOCA categories would include the following piping:

- Large LOCA - pipe diameter > 1 inch
- Medium LOCA - 0.5 inch < pipe diameter  $\leq$  1 inch
- Small LOCA - pipe diameter  $\leq$  0.5 inch

Most of the piping listed in Table 3-1 would therefore be considered as potential large LOCA sources in Mode 5.

### 3.2 ROLE OF THE OPERATOR IN MITIGATING A LOCA OCCURRING DURING COLD SHUTDOWN

Unlike power operation, there are few protective actions that will occur automatically following a LOCA in cold shutdown. It must also be noted, however, that few protective actions are required to maintain the plant in a safe condition following a LOCA in cold shutdown, and considerable time is usually available to accomplish necessary manual or remote-manual protective actions.

Required operator actions are included in the event trees and fault trees presented in Section 5 and Appendix A, respectively. Time lines are included in Section 5, and results of calculations are summarized in this section to provide insight into the time available for necessary operator

actions following a LOCA in cold shutdown. It is assumed that the operator has the capability to detect basic symptoms of a LOCA in Mode 5 (e.g., decreasing or zero pressurizer level and RCS pressure) and to initiate a timely response.

### 3.3 DECAy HEAT AND COOLANT INVENTORY MAKEUP REQUIREMENTS

As discussed in Section 2, six hours following reactor shutdown is the earliest time that a PWR would be expected to reach a cold shutdown condition. Following extended full power operation, core decay heat generation rate six hours after reactor shutdown would be approximately one percent of rated core thermal power. For the Sequoyah nuclear plant, this equates to about 34 MWt. At 720 hours (30 days) after reactor shutdown, decay heat generation rate would be approximately 5 MWt. Table 3-4 provides a summary of the estimated decay heat generation rate following extended full power operation of the Sequoyah nuclear plant. Both fission product and heavy element decay contribute to the total decay heat rate listed in Table 3-4 and plotted in Figure 3-2.

Following plant cooldown and establishment of a relatively constant RCS temperature in cold shutdown, all sensible heat from the reactor coolant and RCS and core internal structures has been transferred to the ultimate heat sink. Decay heat production continues as described above. The integrated heat transferred by an RHR system to the ultimate heat sink following shutdown of a 3800 MWt PWR is shown in Figure 3-3 (from Ref. 3). This figure illustrates that, for shutdowns as short as a few days, the majority of the core decay heat is produced after cold shutdown conditions have been established.

Having an estimate of decay heat generation rate, calculations were performed to estimate the time it would take to boil down to the core mid-plane following a large LOCA which initially drops coolant level to the elevation of the RCS hot leg nozzles. Approximate water volumes in the reactor vessel are shown in Figure 3-4. To drop coolant level to the core mid-plane, about 64,650 pounds of water must boil off, requiring  $64.7 \times 10^6$  BTU assuming an initial water temperature of 200°F. Assuming that all decay heat is transferred into the reactor coolant, Table 3-5 lists times to boil down to the core mid-plane as a function of time following reactor shutdown.

Under the stated assumptions, the times in Table 3-5 represent minimum times for uncovering the core to the mid-plane. Actual times may be longer.

The calculated times in Table 3-5 are not particularly sensitive to initial coolant temperature. If coolant temperature was initially 140°F, the time to boil down to the core mid-plane would only be increased by about 15 percent.

This study assumes that the onset of core melt will occur after coolant level has dropped to the core mid-plane. The times listed in Table 3-5 will be used to as an estimate of the time available for the operator to initiate protective actions to prevent core melt.

Also included in Table 3-5 is an estimate of the makeup rate necessary to match coolant boil off. If at least this makeup rate can be provided following a LOCA, reactor coolant level can be stabilized or increased. A single centrifugal charging or safety injection pump can provide adequate makeup flow at pressures in excess of RCS design pressure in Mode 5. In addition, a single RHR pump can provide the required makeup flow when RCS pressure remains below RHR pump shutoff head (about 195 psig, see Table 4-1). This study does not take credit for the reciprocating charging pump because its makeup capacity (98 gpm, see Table 4-1) is less than the makeup requirements listed in Table 3-5 during the first two days following a shutdown.

#### 3.4 RCS AND RHR OVERPRESSURE PROTECTION REQUIREMENTS

A variety of transients initiated in cold shutdown can cause an RCS pressure excursion. Table 3-6 lists the types of transients that are usually considered in the design of the overpressure protection features of the RHR system. Rates of RCS pressure rise for these transients are listed in Table 3-6 and are illustrated graphically in Figure 3-5 (from Ref. 3). These values apply to a 3800 Mwt B&W standard plant, but serve to illustrate that high-head coolant injection pumps are capable of rapidly increasing RCS pressure during cold shutdown.

As discussed previously, RHR system design pressure (600 psig) and the RCS temperature-pressure limits establish the upper limit for RCS pressure during cold shutdown. The Sequoyah RCS cooldown temperature-pressure limitations are shown in Figure 3-6 (from Ref. 1). It is likely that coolant injection in response to a LOCA in Mode 5 will cause a cooldown of the RCS.

Allowable combinations of RCS pressure and temperature for specific cooldown rates are below and to the right of the limit lines in Figure 3-6. As is evident in this figure, pressure limits become more restrictive as the rate of RCS cooldown increases. A temperature-pressure limit curve for RCS heatup is shown in Figure 3-7 (from Ref. 1). These graphs define limits to assure prevention of nonductile failure.

RHR system and RCS overpressure protection is provided by a single relief valve on the shutdown cooling suction line from the RCS. This valve is capable of relieving the combined capacity of all charging pumps at the relief valve setpoint pressure of approximately 600 psig. Only one centrifugal charging pump is normally operating during cold shutdown. Further overpressure protection for the RHR system is provided by the redundant shutdown cooling suction line isolation valves which close automatically if RCS pressure increases above 600 psig. Once an isolation valve is closed, the RHR suction safety valve is isolated from the RCS, which now is protected only by power-operated relief valves (PORVs, 2350 psig setpoint) and pressurizer safety valves (2485 psig setpoint). These setpoints are far in excess of the limits imposed by the temperature-pressure curves in Figures 3-6 and 3-7. The operator could, however, remotely open a PORV to provide RCS overpressure protection in this case.

Medium and small LOCAs during cold shutdown have been defined in Table 3-2. For these types of LOCAs, a centrifugal charging pump or a safety injection pump can provide makeup at a rate that exceeds the leak rate at an RCS pressure of 600 psig (e.g., 550 to 650 gpm). The high-head pumps can eventually restore RCS coolant inventory, and pressurize the RCS to a point where makeup and leak rates become equal. Maximum RCS pressure will be related to: (a) performance of the RHR overpressure protection features and operator actions related to overpressure protection, (b) the respective pump characteristic curve (see Section 4), and (c) the LOCA pipe break characteristics. With successive failures of overpressure protection features, it is possible that RCS pressure could be driven above the applicable temperature-pressure limit curve during response to a medium or small LOCA with a high-head coolant injection pump.

For this study, it will be assumed that an unmitigatable breach will result following overpressurization of the RCS and/or RHR system during response to a medium or small LOCA with a high-head coolant injection pump. A core melt is then assumed to be inevitable.

### 3.5 CRITICALITY CONTROL REQUIREMENTS DURING LOCA RESPONSE

Prior to the LOCA, the reactor coolant is adequately borated to maintain  $K_{eff}$  less than 0.99 (shutdown margin greater than 1.0 percent delta k/k), as required by the Technical Specifications (Ref. 1). Nominal Mode 5 RCS boron concentration is in the range from 945 ppm (all rods inserted) to 1,016 ppm (BOL, one rod stuck out). Boron reactivity worth in cold shutdown is approximately 1.0 percent delta k/k per 70 ppm boron.

It should be evident that extended RCS makeup from an inadequately borated (or unborated) water source could potentially lead to boron dilution and an inadvertent criticality following a Mode 5 LOCA. In this study, it is assumed that reactor coolant makeup is provided from the refueling water storage tank (2,000 ppm boron), or is provided from some alternative water source such as the primary water storage tank (0 ppm boron) only after proper boration via the chemical and volume control system (CVCS) boric acid blender. It is estimated that the continuous makeup rate from the latter water source is limited to 150 gallons per minute because of the need to provide adequate boration. Coordinated manual actions may be necessary to maintain an adequate supply of concentrated boric acid to meet long-term boration requirements.

Other borated water sources such as the cold leg accumulators and the upper head injection accumulator (all approximately 2000 ppm boron) could be aligned to provide limited makeup to the RCS. These water sources are not modeled in the event trees in Section 5 because they have only a very limited capability to prevent a core melt.

As discussed previously, it is likely that coolant injection in response to a LOCA in Mode 5 will cause a cooldown of the RCS. In the reactor coolant temperature range of concern (e.g., less than 200°F). The moderator temperature coefficient of reactivity is numerically much smaller than during power operation. With a coolant boron concentration of 1000 ppm, the moderator temperature coefficient would be approximately  $-0.25 \times 10^{-4} \Delta\rho/^\circ\text{F}$  (see Figure 3-8). If  $K_{eff}$  were initially 0.99, an RCS cooldown from 200°F to 100°F would add  $0.25 \times 10^{-2} \Delta\rho$  reactivity, resulting in a final  $K_{eff}$  of 0.9925. A significant cooldown of the RCS will therefore only affect the margin of subcriticality of the core, and should not result in the core becoming critical.



### 3.6 CONTAINMENT ISOLATION REQUIREMENTS DURING MODE 5

There are no Technical Specification requirements that containment isolation be maintained at Sequoyah during cold shutdown. A preliminary survey of several PWRs, conducted as part of this study, suggests that it is common practice for containment equipment hatches to be open and both airlock doors open simultaneously (e.g., the normal interlock on the doors is bypassed) during cold shutdown to facilitate the movement of personnel and equipment into and out of containment. The initial opening of an airlock or equipment hatch does not occur immediately upon reaching cold shutdown, but rather at some later time associated with planned activities inside containment. A number of plants leave the equipment hatch open whenever that are in Mode 5 to simplify access requirements, and in some cases, to help maintain habitable containment temperatures. Other plants, including Sequoyah, open the equipment hatch only when it is necessary to move large equipment that cannot be brought through a personnel airlock. All plants in this preliminary survey maintained at least one airlock open in Mode 5.

Equipment hatches are not necessarily designed for rapid closure. Some plants, including Sequoyah, have equipment hatches that open inward and are moved away on tracks with the aid of a dedicated winching system. It may be possible to reclose such hatches in 30 to 60 minutes. At least one plant has an equipment hatch that opens outward and is removed to a nearby laydown area with the aid of a crane. At this plant, it was estimated that four to eight hours would be required to reclose the hatch. When an equipment hatch is reclosed in an emergency, there would be no means to verify the leak tightness of the hatch seal (e.g., a containment leak test could not be conducted). The value assumed for containment leak rate following restoration of containment integrity may therefore be uncertain.

Temporary service lines (e.g., welding cables) may occasionally be run through a personnel airlock. In spite of this type of obstruction, it appears reasonably certain that airlocks can be reclosed much more quickly than equipment hatches. Quick disconnect fittings may be used on some service lines, thereby providing a capability to rapidly reclose at least one airlock door.

It is conservative to assume that the probability of having a non-isolated containment in Mode 5 is one (e.g., containment isolation requirements will be relaxed). After a Mode 5 LOCA, the time available to restore

containment isolation before the onset of core melt will be a function of the time after reactor shutdown and the actual core power history. Minimum times for protective actions by the operator are discussed in Section 3.3. At Sequoyah, an open airlock could very likely be reclosed when required. As stated previously, the Sequoyah equipment hatch is only opened when required to move large equipment. The equipment hatch design appears to provide a relatively rapid reclosure capability, however, the time required to reclose the hatch (e.g., 30 to 60 minutes) is comparable to the minimum time available for operator protective actions during the interval from eight to 35 hours after reactor shutdown (e.g., 39 to 62 minutes). Significantly greater time is available if the operator is successful in establishing an effective short-term coolant injection capability (see Section 5.2).

This study did not include containment response in the LOCA event trees in Section 5; therefore, no assumptions were made in the analysis regarding the initial containment isolation status or the capability to restore containment isolation.

Table 3-1. Potential LOCA Sources in Mode 5.

1. Random pipe break LOCA due to seismic event
  - a) RCS loop piping (27.5" to 31")
  - b) Piping connected to RCS\*
    - RHR shutdown cooling suction line (14")
    - Pressurizer surge line (14")
    - ECCS cold leg injection lines (4x10")
    - Upper head injection accumulator lines (4x8")
    - RHR and/or safety injection hot leg injection lines (4x6")
    - Pressurizer safety valve lines (3x6")
    - Pressurizer power-operated relief valve (PORV) header (6")
    - Pressurizer spray lines (2x4")
    - Individual PORV lines (2x3")
    - CVCS normal charging line (3")
    - CVCS alternate charging line (3")
    - CVCS high pressure letdown line (3")
    - RCS loop flow/temperature sensor return line to RCP suction (4x2")
    - RCS loop flow/temperature sensor supply line (8x2")
    - RCS loop low point drains to RC drain tank (4x2")
    - Pressurizer auxiliary spray supply line (2")
    - Charging pump safety injection lines (4x1-1/2")
    - CVCS excess letdown line (1")
    - Miscellaneous vent, drain, sample lines and sensor fittings (3/4")
2. LOCA due to system misalignment by operator
  - a) Inadvertent alignment of RHR pump to RHR containment spray line (8")
  - b) Inadvertent alignment of RHR pump to RWST return line (8")
  - c) Inadvertent opening of PORV (3")
  - d) Inadvertent alignment of RCS to RC drain tank (2" to 3/4")
  - e) Inadvertent opening of miscellaneous vent, drain or sample lines or sensor fittings (3/4")
3. LOCA due to design RCS pressure transient and stuck-open RHR safety valve (3")
  - a) Loss of RHR heat removal capability
  - b) CVCS makeup valves stuck fully open
  - c) Low pressure letdown valve fails shut when RCS is solid, with continued operation of charging pumps
  - d) Pressurizer heaters inadvertently energized when RCS is solid
  - e) Inadvertent safety injection pump startup
4. LOCA due to unmitigated pressure transient and pressurization of RCS well above the pressure/temperature limit curve, causing RHR pipe breach or reactor vessel failure.
  - a) Inadvertent alignment of upper head injection accumulator to RCS
  - b) Inadvertent pressurization of RCS using charging or safety injection pumps

Notes: \* The RCS and connected piping are illustrated in Figure 4-4.

Table 3-2. Definition of Mode 5 LOCA Size Categories.

LOCA Category	Definition
Large LOCA	Rate of coolant loss from the LOCA exceeds the capacity of a single centrifugal charging pump or safety injection pump when the RCS is assumed to be at RHR system design pressure. <sup>(a)</sup> Primary safety concern is maintaining adequate core coolant inventory. The high-head pumps cannot repressurize the RCS significantly above RHR system design pressure.
Medium LOCA	Rate of coolant loss from the LOCA is less than the capacity of a single centrifugal charging pump or safety injection pump, when the RCS is assumed to be at RHR system design pressure <sup>(a)</sup> , but the rate of coolant loss is sufficient to remove all core decay heat <sup>(b)</sup> . Operation of the RHR system in the RHR mode is not required. Overpressurization of the RCS and the RHR system by the high-head pumps is a potential safety concern.
Small LOCA	Coolant loss at a rate that is insufficient to remove all core decay heat <sup>(b, c)</sup> . Heat removal from the RCS, in addition to that provided by the LOCA itself, is required. Overpressurization of the RCS and the RHR by the high-head pumps is a potential safety concern.

- Notes:
- (a) About 550 to 650 gpm at approximately 600 psig RCS pressure.
  - (b) About 160 gpm boil-off rate at 20 hours after shutdown (assuming RCS boiling at atmospheric pressure with no makeup), decreasing as a function to time.
  - (c) There may be a time following reactor shutdown beyond which no LOCA size fits this definition (e.g., decay heat generation rate has decreased to such a level that an additional core heat removal capability, beyond the LOCA itself, is no longer required).

Table 3-3. Estimated LOCA Leak Rate at 600 psig RCS Pressure.

Actual Pipe Inside Diameter (inches)	Leak Rate in gpm, as a Function of Length of Pipe Between RCS Loop Piping and the Break Location	
	1 foot	100 feet
0.5	146	33
0.75	345	91
1	628	186
1.5	1,442	499
2	2,579	995
3	5,797	2,582
4	10,250	5,024
6	22,990	12,680
8	40,950	24,280
10	63,380	39,870
14	100,300	75,990

Table 3-4. Estimated Sequoyah Nuclear Plant Decay Heat Rate.

Time After Shutdown (hours)	Decay Heat Rate ( $10^6$ BTU/hr)*			Decay Heat Power (P) As a Fraction of Design Power ( $P_0$ )** $P/P_0$
	Fission Product	Heavy Element	Total	
1	162.91	20.73	183.64	$1.58 \times 10^{-2}$
2	129.10	17.91	147.01	$1.26 \times 10^{-2}$
3	115.20	17.25	132.45	$1.14 \times 10^{-2}$
4	106.08	16.97	123.05	$1.06 \times 10^{-2}$
5	98.82	16.75	115.57	$9.93 \times 10^{-3}$
6	92.71	16.53	109.24	$9.38 \times 10^{-3}$
7	87.51	16.33	103.84	$8.92 \times 10^{-3}$
8	83.06	16.14	99.20	$8.52 \times 10^{-3}$
9	79.24	15.94	95.18	$8.18 \times 10^{-3}$
10	75.96	15.74	91.70	$7.88 \times 10^{-3}$
11	73.13	15.56	88.69	$7.62 \times 10^{-3}$
12	70.66	15.36	86.02	$7.39 \times 10^{-3}$
20	-	-	74.80***	$6.43 \times 10^{-3}$
24	56.23	13.26	69.49	$5.97 \times 10^{-3}$
48	46.99	9.87	56.86	$4.88 \times 10^{-3}$
72	41.61	7.36	48.97	$4.21 \times 10^{-3}$
96	37.89	5.47	43.36	$3.73 \times 10^{-3}$
120	35.24	4.08	39.32	$3.38 \times 10^{-3}$
144	33.25	3.04	36.29	$3.12 \times 10^{-3}$
168	31.71	2.26	33.97	$2.92 \times 10^{-3}$
192	30.46	1.68	32.14	$2.76 \times 10^{-3}$
216	29.40	1.26	30.66	$2.63 \times 10^{-3}$
240	28.48	0.93	29.41	$2.53 \times 10^{-3}$
400	23.95	0.13	24.08	$2.07 \times 10^{-3}$
500	21.84	0.03	21.87	$1.88 \times 10^{-3}$
600	20.06	0.01	20.07	$1.72 \times 10^{-3}$
700	18.57	0.0	18.57	$1.60 \times 10^{-3}$
720	18.30	0.0	18.30	$1.57 \times 10^{-3}$

Notes: \* Adapted from B-SAR-205, Tables 9.4-2 and 9.4-3 assuming 10,000 hours of reactor operation at the Sequoyah Nuclear Plant design power level of 3411 Mwt.

\*\* Sequoyah design power,  $P_0$ , is  $11,641.7 \times 10^6$  BTU/hr (Sequoyah FSAR Table 4.4-1).

\*\*\* From Sequoyah FSAR, Table 5.5-8.

Table 3-5. Estimates of Time to Boil Down to Core Mid-plane Following LOCA and Makeup Rate to Match Boil-off.

Time After Shutdown (hours)	Decay Heat Rate <sup>(a)</sup> (10 <sup>6</sup> BTU/hr)	Estimated Time to Boil Down to Core Mid-plane <sup>(b)</sup> (minutes)	Estimated Makeup Rate Required to Match Boil-off <sup>(c)</sup> (gpm)
6	109.24	36	197
7	103.84	37	187
8	99.20	39	179
9	95.18	41	171
10	91.70	42	165
11	88.69	44	160
12	86.02	45	155
20	74.80	52	135
24	69.49	56	125
35	62.50 <sup>(d)</sup>	62	113
48	56.86	68	102
72	48.97	79	88
96	43.36	90	78
120	39.32	99	71
144	36.29	107	65
168	33.97	114	61
192	32.14	121	58
400	24.08	161	43
720	18.30	212	33

- Notes: (a) From Table 3-4.  
 (b) Assumes reactor coolant is initially at 200°F, and boils at atmospheric pressure.  
 (c) Assumes reactor coolant is at 212°F, boiling at atmospheric pressure, and makeup water is initially at 70°F.  
 (d) Estimated from Figure 3-2.

Table 3-6. Pressure Rise and Relief Valve Capacity\*.

INCIDENT DESCRIPTION	RATE OF PRESS. RISE, PSI/MIN	REQUIRED RELIEF CAPACITY, GPM**
Loss of DHR System Cooling	12	225
Makeup Control Valve Fails Full Open	36	524
All Pressurizer Heaters Energized	5	1435
High Pressure Injection (HPI) actuation (all three HPI pumps operate)	162	2000
<u>EQUILIBRIUM PRESSURE, PSIG</u>		
Core Flood Tank Outlet Valve Opened	<u>At Pressurizer</u>	<u>At DHR Pump Suction</u>
Initial Pressurizer Pressure at Midpoint of Band	432	455
Initial Pressurizer Pressure at High Point of Band	474	497

Notes: \* From B-SAR-205

\*\* At a setpoint of 455 psig, the minimum required capacity is 2000 gpm at 10% accumulation. This relief valve will prevent the DHR system design pressure from being exceeded by more than 10% during the worst incident concurrent with the DHR pump operating at any developed head up to and including shutoff head. Each of the dual DHR suction lines contains a relief valve sized for the full relief flow rate in the event that the DHR system is being operated on one letdown line only.



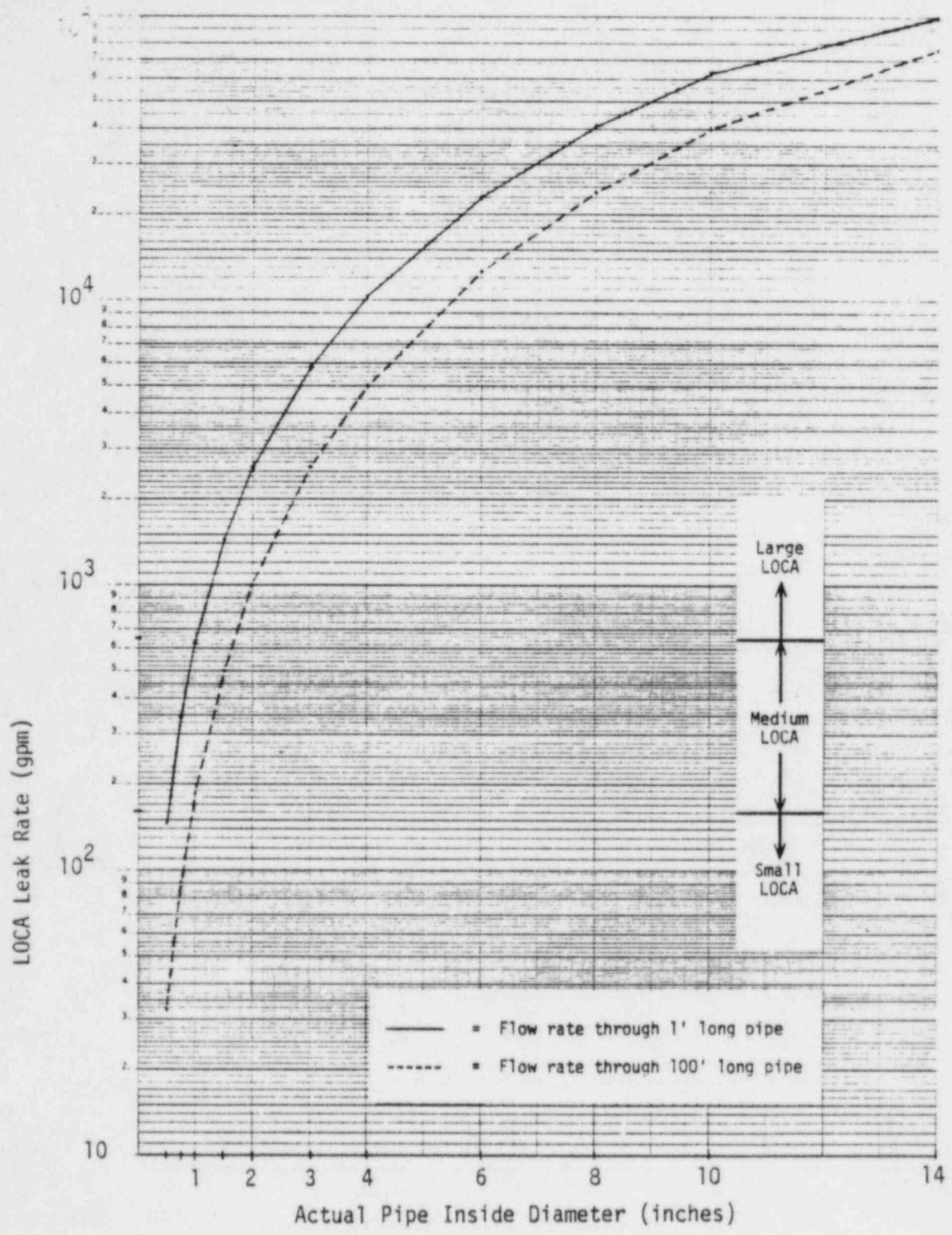


Figure 3-1. LOCA Leak Rate at 600 psig RCS pressure as a Function of Pipe Diameter.

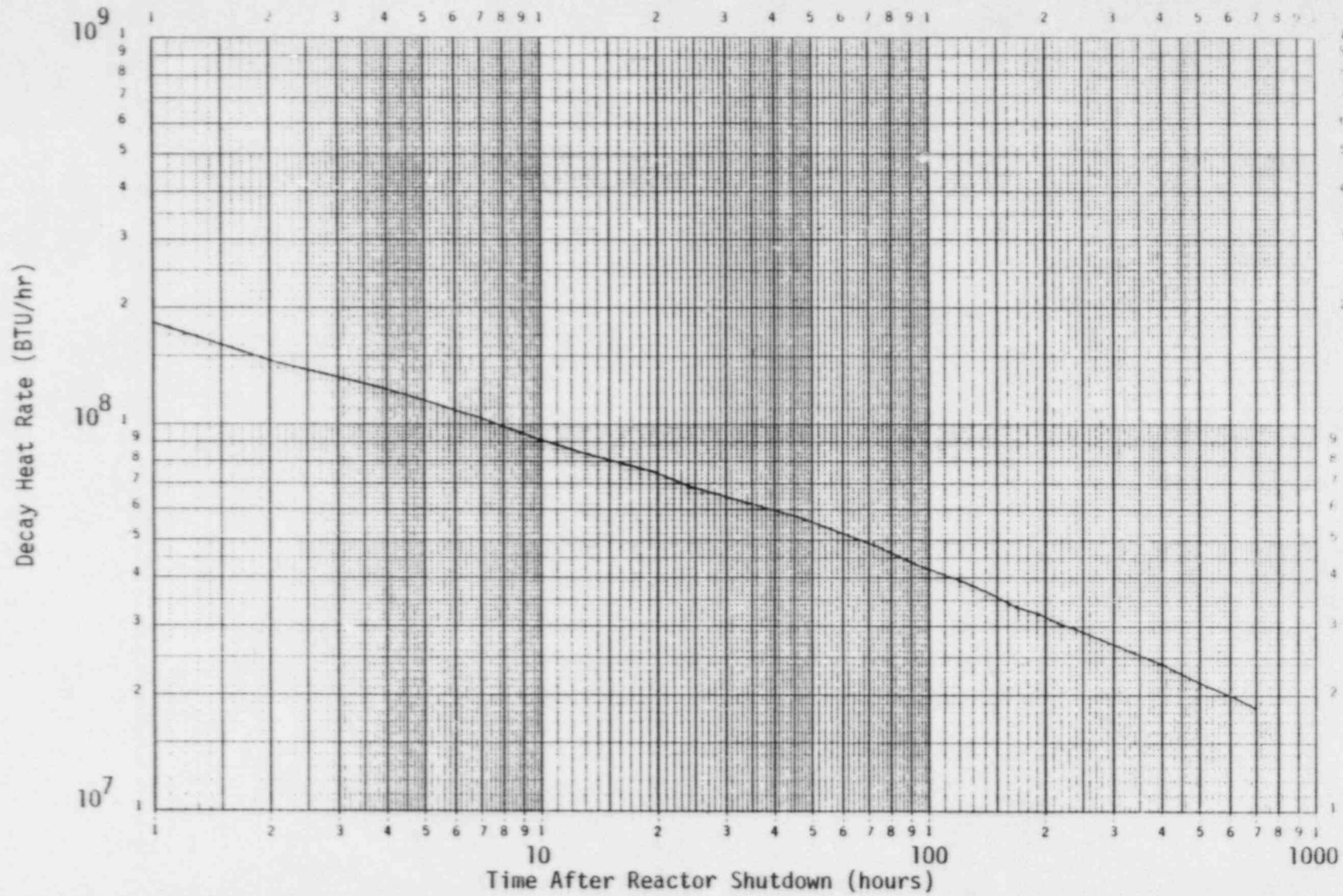


Figure 3-2. Estimated Sequoyah Nuclear Plant Decay Heat Rate.

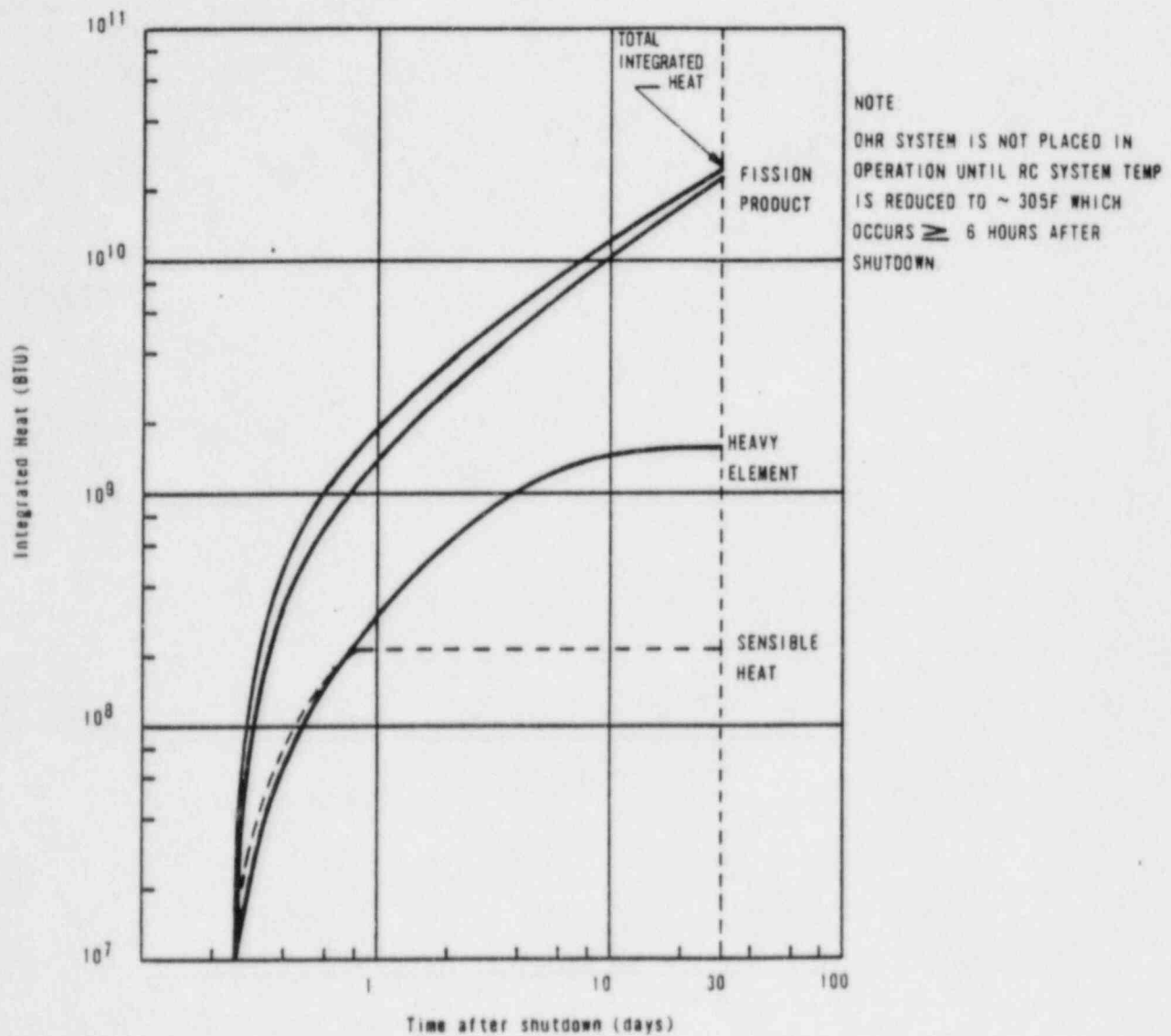
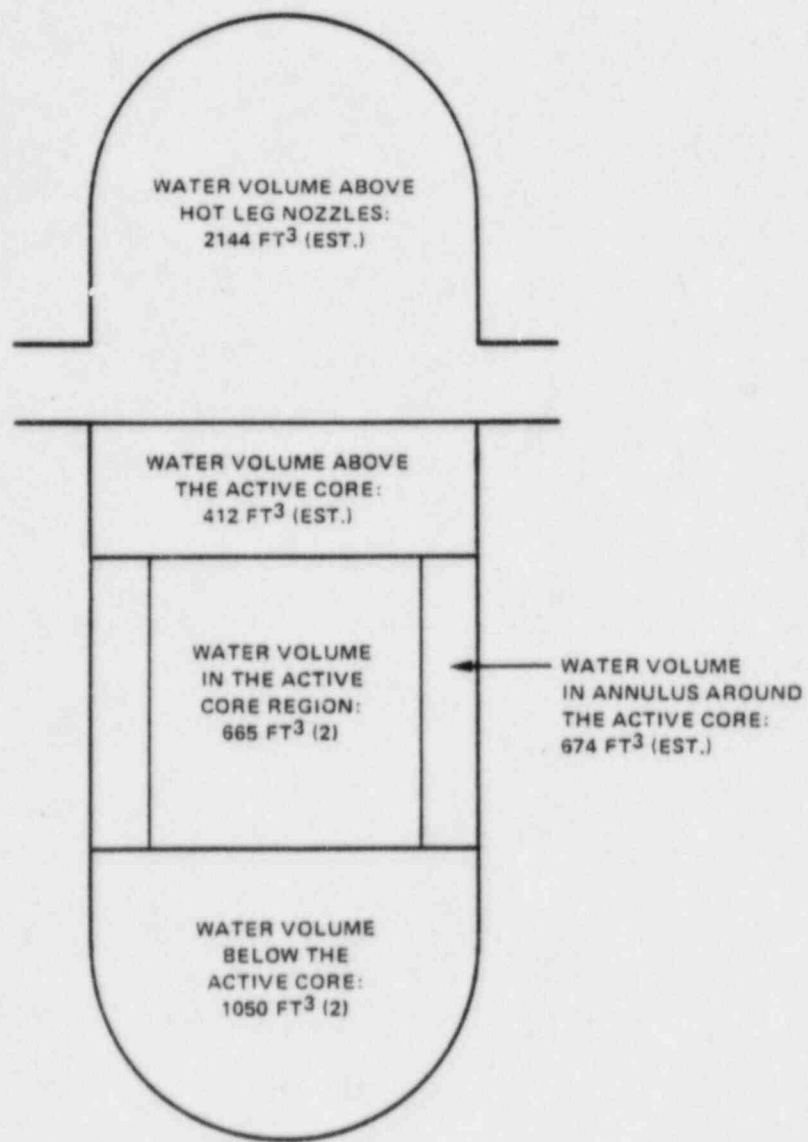


Figure 3-3. Integrated Heat Transferred by a Residual Heat Removal System to the Ultimate Heat Sink Following Shutdown of a 3800 MWt PWR (from B-SAR-205).



NOTES:

(1) NOT TO SCALE

(2) FROM D.C. COOK FSAR, TABLE 3.2.1.1. ALSO NOTE THAT TOTAL WATER VOLUME OF REACTOR VESSEL WITH CORE AND INTERVALS IN PLACE IS 4945 FT<sup>3</sup>

Figure 3-4. Approximation of Water Volumes in The Reactor Vessel<sup>(1)</sup>.

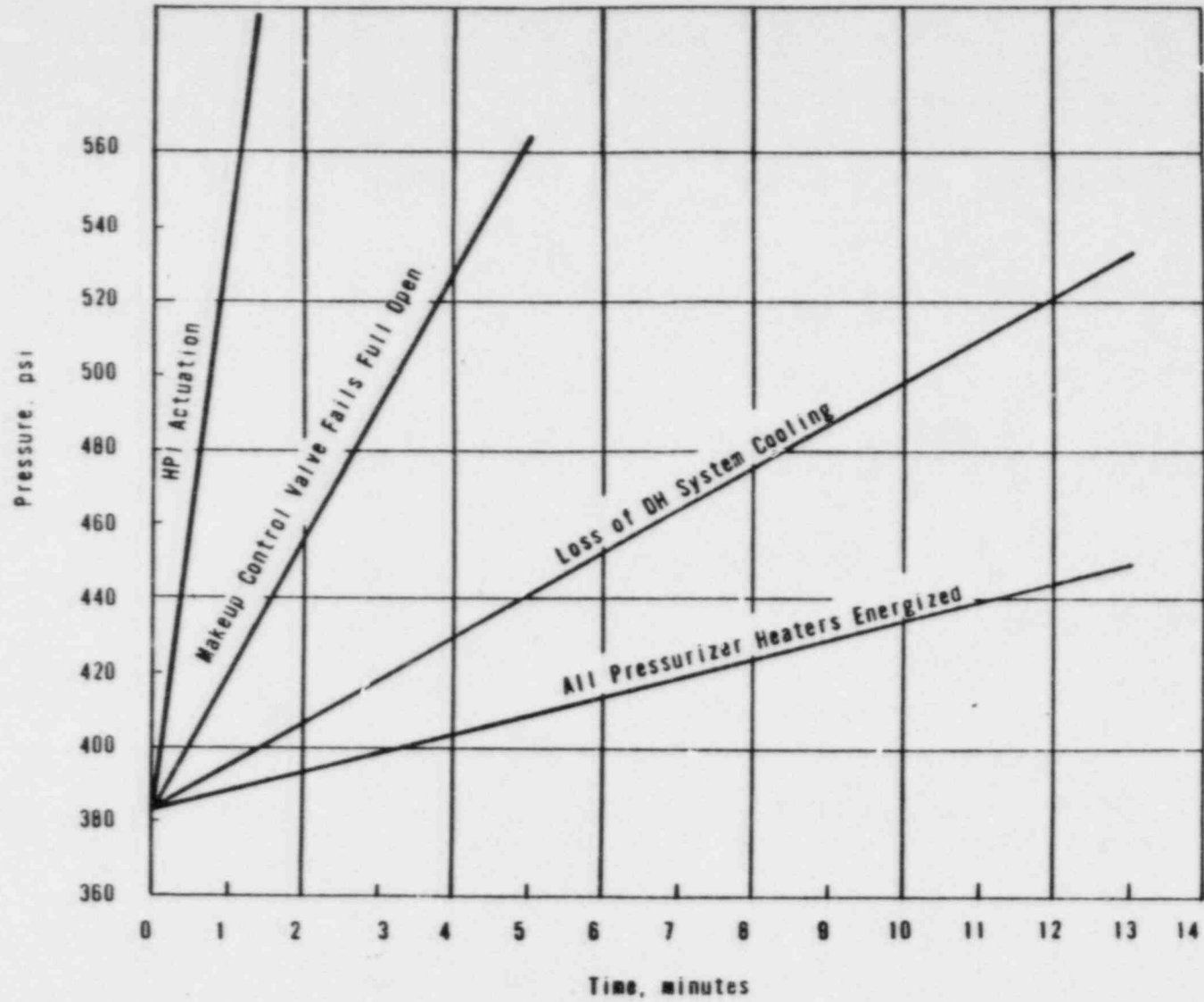


Figure 3-5. Pressure Vs Time for DHR Overpressure Transients.  
(From B-SAR-205)

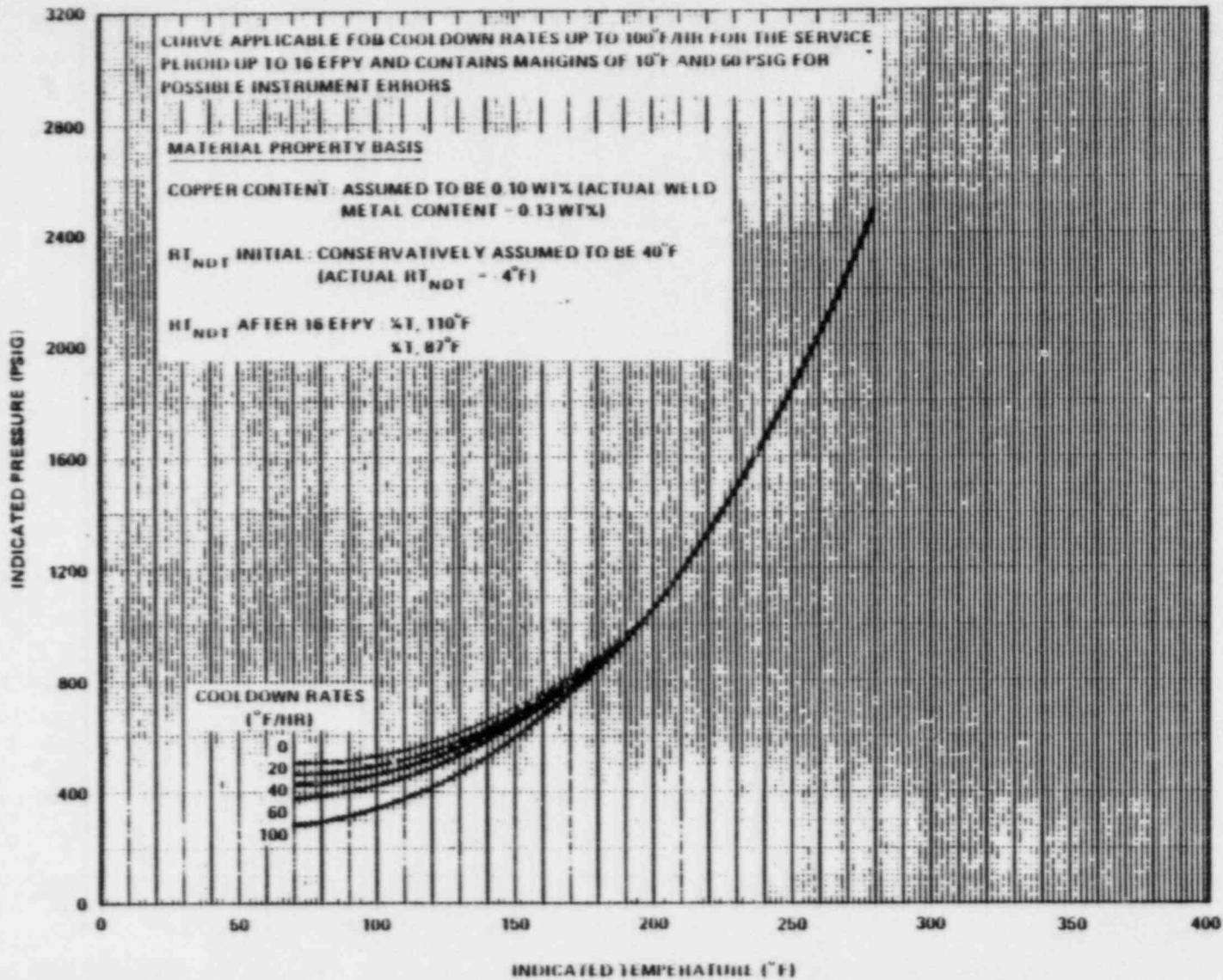


Figure 3-6. Seabrook Unit 2 Reactor Coolant System Cooldown Limitations Applicable up to 16 EFPY (From NUREG-0789).

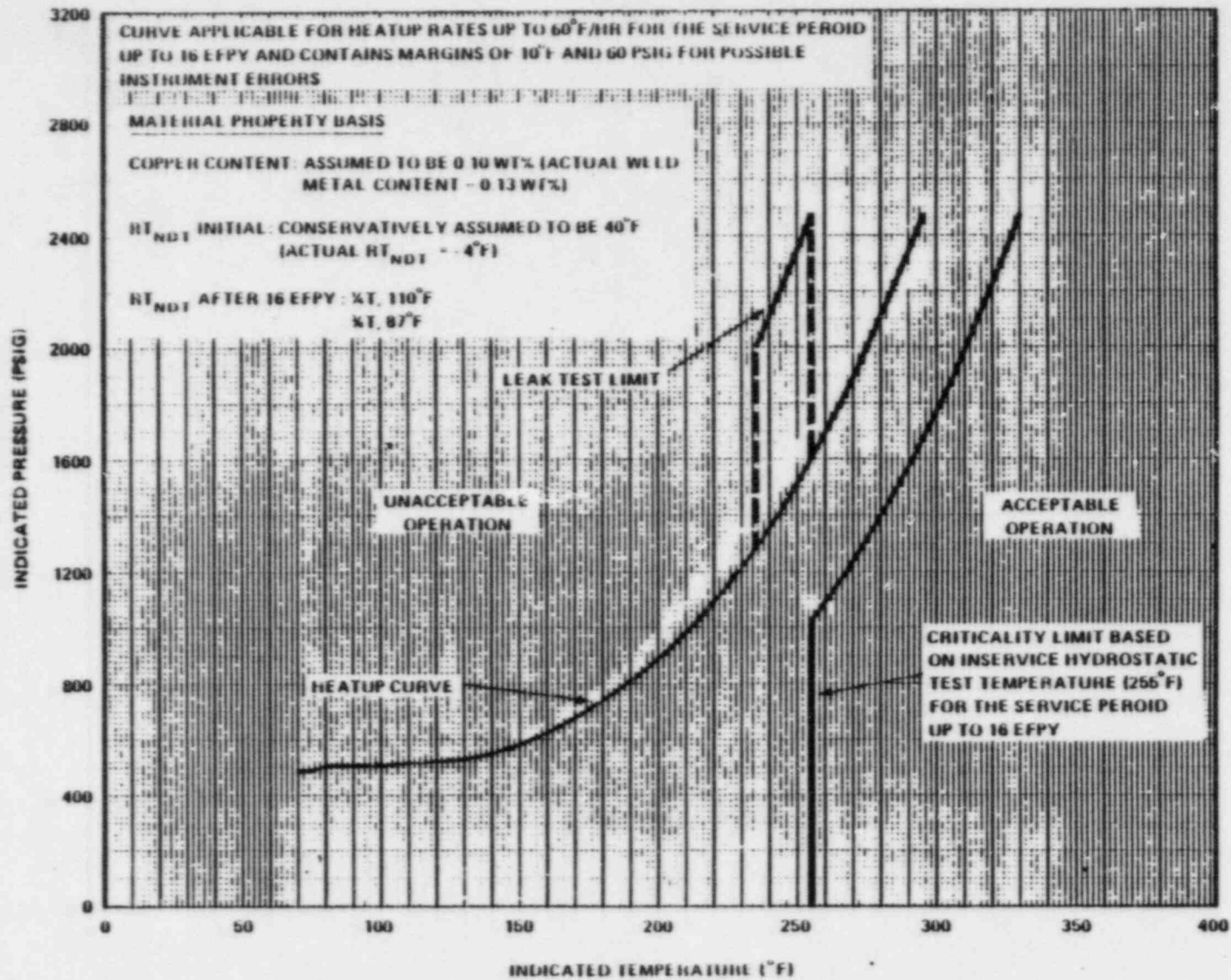


Figure 3-7. Sequoyah Unit 2 Reactor Coolant System Heatup Limitations Applicable up to 16 EFPY (From NUREG-0789).

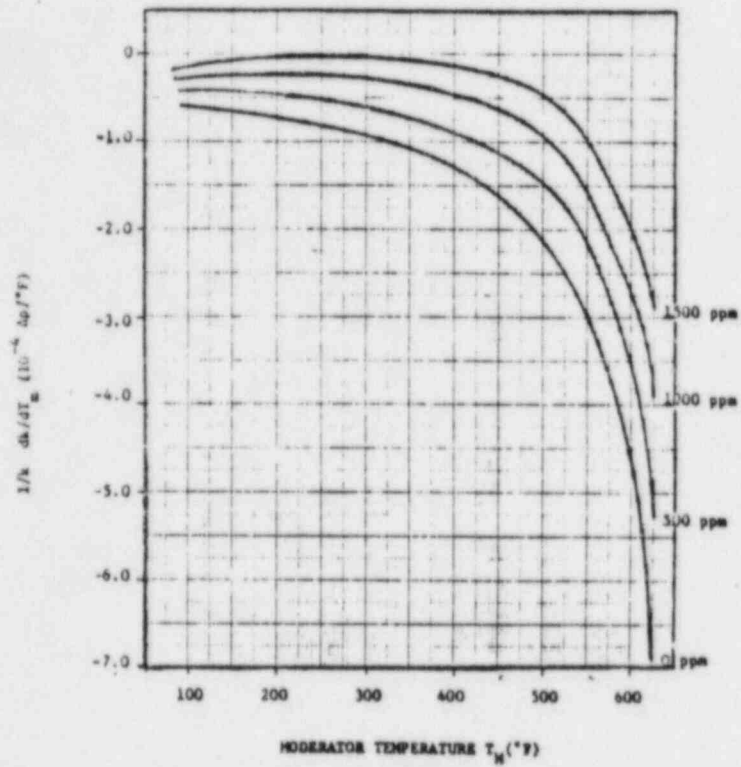


Figure 3-8. Moderator Temperature Coefficient Versus Moderator Temperature, BOL, All Control Rods Inserted (from D. C. Cook FSAR).





#### 4. DESCRIPTION OF SYSTEMS DURING COLD SHUTDOWN

This section provides a brief description of the following systems at the Sequoyah nuclear plant that are related to LOCA initiation and response in cold shutdown:

- Reactor coolant system (RCS),
- Residual heat removal (RHR) system,
- Chemical and volume control system (CVCS),
- High pressure safety injection (HPSI) system,
- Class 1E AC electric power system,
- Component cooling water (CCW) system,
- Essential raw cooling water (ERCW) system.

The first four systems will be called "front-line" systems because of their direct role in LOCA response. The last three systems will be called "support systems" because they are required for the operation of the front-line systems.

Although not modeled in the event trees in Section 5, the cold leg accumulators and the upper head injection (UHI) accumulator are also briefly described in this section.

##### 4.1 FRONT-LINE SYSTEMS

A summary of design data for pumps capable of providing reactor coolant makeup is presented in Table 4-1. Characteristic curves for the centrifugal charging, residual heat removal and safety injection pumps are shown in Figures 4-1 to 4-3, respectively. Design and operating data for potential reactor coolant makeup water sources is presented in Table 4-2.

#### 4.1.1 Reactor Coolant System

The Sequoyah nuclear plant has a four-loop RCS as shown in Figure 4-4. All interfacing piping larger than one inch in diameter appears in this diagram. Actual line sizes are listed in Table 3-1.

In cold shutdown, the RCS can be maintained with a steam bubble in the pressurizer or it can be placed in a water solid condition. In either case, the CVCS controls RCS pressure, and makeup is provided by one centrifugal charging pump. The general practice at Sequoyah is to maintain a steam bubble in the pressurizer when possible.

#### 4.1.2 Residual Heat Removal System

The RHR system at Sequoyah is shown in Figure 4-5 in its shutdown cooling alignment. This system has two loops which share a common shutdown cooling suction line. Two independent return paths to the RCS exist via motor-operated valves 63-93 and 63-94. During cold shutdown, letdown from the RCS to the CVCS is accomplished via a low pressure letdown flow path on the outlet side of the RHR heat exchangers.

During shutdown cooling, the RHR system is protected against overpressurization by relief valve 74-505 on the shutdown cooling suction line. This relief valve is sized to relieve the combined flow of all charging pumps at the relief valve setpoint of approximately 600 psig. During cold shutdown, only one centrifugal charging pump is in operation. Further overpressure protection is provided by the series shutdown cooling suction isolation valves 74-1 and 74-2 which close automatically if pressure exceeds 600 psig.

In addition to its shutdown cooling function, the RHR system also can perform coolant injection and recirculation functions as part of the emergency core cooling system (ECCS). The RHR pumps can be aligned to provide low pressure coolant injection from the refueling water storage tank (RWST) by isolating the normal shutdown cooling suction path (e.g., by closing valve 74-1 or 74-2) and opening the RWST isolation valve 63-1. If the shutdown cooling suction path is not isolated and if RCS pressure is higher than the static head from the RWST, the RWST suction check valve 63-505 will seat, preventing the RHR pumps from taking a suction on the RWST (Ref. 6).

The RHR pumps can also be aligned to provide low pressure recirculation from the containment active sump to the RCS. Valves 63-1, 63-72 and 63-73 are interlocked to prevent the simultaneous alignment of an RHR pump to

the RWST and to the containment sump. The switchover from injection to recirculation during cold shutdown would be accomplished manually when RWST level reaches a low-level alarm setpoint.

Due to Sequoyah containment geometry, it is necessary to add a considerable amount of water from the RWST to ensure that the active containment sump (bottom at elevation 667.0 feet, curb top at approximately 680.5 feet) is flooded for all LOCAs and that an RHR recirculation flow path can be established. There is also a pit sump (bottom at elevation 658.3 feet) beneath the reactor vessel. Some break locations such as a break at a reactor vessel hot or cold leg nozzle may result in flooding of the pit sump before any water reaches the containment floor at the 679.78 foot elevation and spills over an approximately eight-inch curb into the active sump.

Rough calculations indicate that about 100,000 to 120,000 gallons of water will flood the pit sump. It was estimated by NRC inspectors that 105,000 gallons would fill the Sequoyah active sump and flood the containment floor at the 679.78 foot elevation to a depth of about 18 inches (10 inches above the sump curb) if little or no water leaked to the pit sump. It is therefore estimated that a minimum of 155,000 to 175,000 gallons of water must be dumped into containment to ensure that the active sump will be flooded and that an RHR recirculation flow path is available.

A large LOCA will rapidly dump approximately 68,000 gallons (9091 ft<sup>3</sup>) to containment if the RCS was being maintained with a steam bubble in the pressurizer. Slightly more would be dumped if the RCS was in a water solid condition.

Note that it may be possible to establish an unorthodox flow path between the RWST and the suction of an RHR pump by: (a) aligning a containment spray pump to the RWST by opening valve 72-21 or 72-22, (b) bypassing valve interlock circuitry, (c) simultaneously aligning the selected containment spray pump to the containment sump suction header by opening valve 72-20 or 72-23, and (d) closing the RHR pump suction isolation valves 74-3 and 74-21. The RWST is then connected to the sump suction header via the containment spray system piping and it should then be possible to draw a suction on the RWST with the RHR pumps.

#### 4.1.3 Chemical and Volume Control System

The Sequoyah CVCS is illustrated in Figure 4-6. During cold shutdown, the high pressure letdown path used during power operation remains in operation and a low pressure letdown path from the RHR system is established. The maximum letdown flow rate is approximately 120 to 150 gpm. The normal charging return path to the RCS is also used during cold shutdown. Only a single centrifugal charging pump is operating.

The control mode for the CVCS depends on whether the RCS is maintained with a steam bubble in the pressurizer or is placed in a water solid condition. With a bubble in the pressurizer, the low pressure letdown valve establishes a fixed backpressure, and the pressurizer level controller automatically positions a makeup control valve to modulate makeup flow rate as necessary to control pressurizer level. In this mode of operation, makeup flow rate would automatically increase as pressurizer level decreased during a LOCA. Small LOCAs may be adequately controlled and automatic response would be limited only by the ability to provide extended makeup to the volume control tank (VCT) which would commence when VCT level drops to a low-level setpoint. It is estimated that a long-term makeup rate of 150 gpm can be established from the primary water storage tank (PWST) and properly boric acid via the boric acid blender. Automatic makeup from this source is limited by the capacity of the boric acid storage tanks. If VCT level cannot be adequately maintained from the normal makeup source and level drops to a low-low level setpoint, the charging pump suctions will be automatically shifted to the RWST by opening valve 62-135 or 62-136.

If the RCS is maintained in a water solid condition, the low pressure letdown valve is again set to establish a fixed back pressure, and the makeup control valve is set for a fixed makeup flow rate. During a LOCA, the makeup flow rate would remain constant, and the RCS would rapidly depressurize. The low pressure letdown valve would automatically close in an attempt to maintain RCS pressure. Makeup to the VCT would be provided automatically as described before.

The centrifugal charging pumps are part of the ECCS and can be aligned to a high pressure coolant injection flow path by opening valve 63-39 or 63-40 and valve 63-25 and 63-26. This flow path feeds a single header which branches to inject into all four RCS cold legs.

#### 4.1.4 High Pressure Safety Injection System

The HPSI system at Sequoyah is illustrated in Figure 4-7. The system has two loops which share a common suction line from the RWST. During cold shutdown, the circuit breakers for the HPSI pumps are racked-out to render the pumps inoperable.

#### 4.1.5 Cold Leg Accumulators

Four cold leg accumulators are provided, one for each of the four RCS cold legs. A typical cold leg accumulator is shown in Figure 4-8. During cold shutdown, the motor-starter for the motor-operated accumulator isolation valve is racked-out to render the valve incapable of remote operation.

#### 4.1.6 Upper Head Injection Accumulator

The UHI injection system injects directly into the reactor vessel head, as shown in Figure 4-9. During cold shutdown, the redundant hydraulic isolation valves in the injection lines are closed, and are locked by means of motor-operated gags.

When actuated, the UHI accumulator isolation valves remain open until the water accumulator reaches a low-level setpoint of approximately 3366 gallons (450 ft<sup>3</sup>). The isolation valves close automatically to prevent injecting pressurized nitrogen into the RCS.

### 4.2 SUPPORT SYSTEMS

A summary of support system requirements of major components is presented in Table 4-3.

#### 4.2.1 Class 1E AC Electric Power System

The Sequoyah Class 1E AC electric power system consists of two independent load groups or divisions as shown in Figure 4-10. Offsite power is the normal power source during cold shutdown. Following a loss of offsite power, each load group can be supplied from its own standby diesel generator. All 6900 AC Class 1E loads are listed a Figure 4-10. Only the 480 VAC loads directly related to LOCA mitigation in cold shutdown are included in that figure.

#### 4.2.2 Component Cooling Water and Essential Raw Cooling Water Systems

Together, the CCW and ERCW systems establish the heat transfer paths from components requiring cooling to the ultimate heat sink. The CCW and ERCW systems at Sequoyah are shown in Figures 4-11 and 4-12, respectively. As illustrated, these systems serve both Units 1 and 2 at the Sequoyah site.

A simplified model of a CCW and ERCW loop is shown in Figure 4-13. Although not an accurate model of the Sequoyah systems, this is likely to be a conservative model because it lacks much of the component redundancy and all of the built-in cross-connections actually found at Sequoyah.

#### 4.3 INSTRUMENTATION RELATED TO LOCA DETECTION IN MODE 5

Table 4-4 lists instrumentation that may be of use in detecting a LOCA occurring in Mode 5.

#### 4.4 TECHNICAL SPECIFICATION REQUIREMENTS IN MODE 5

Table 4-5 lists the limiting conditions for operation and surveillance requirements in NUREG-0789 (Ref. 1) that are applicable during Mode 5.

Table 4-1. Design Data for Pumps Capable of Providing Reactor Coolant Makeup.

	Centrifugal Charging Pump	Reciprocating Charging Pump	Residual Heat Removal Pump	Safety Injection Pump	Boric Acid Transfer Pump
Number (per unit)	2	1	2	2	2
Type	Horizontal Centrifugal	Variable Speed, Positive Dis-Placement	Vertical Centrifugal	Horizontal Centrifugal	Horizontal Centrifugal
Design Flow (gpm)	150	98	3000	425	75
Design Head (ft.)	5800	5800	375	2500	235
Runout Flow (gpm)*	550	-	5500	650	
Head at Runout Flow (ft.)*	1400	-	250	1500	
Shutoff Head (ft.)	5800	-	450	3500	
(psid)**	2514	-	195	1517	
Normal Status in Mode 5	One pump operating, one pump not used but operable. Circuit breaker for latter pump not racked out, but control switch tagged.	Pump not used, but operable. Circuit breaker not racked out, but control switch tagged.	Operable, with 2, 1 or 0 operating	Inoperable. Circuit breaker for pump racked out.	Available for normal boration or emergency boration as necessary, pumping to charging pump suction.

\* From RESAR-3S, Westinghouse Reference Safety Analysis Report, Docket STN-50545

\*\* Conversion is 1 foot water at 60°F = 0.4335 psi



Table 4-2. Design Data for Reactor Coolant Makeup Water Sources.

	Refueling Water Storage Tank	Primary Water Storage Tank	Boric Acid Storage Tank	Cold Leg Accumulators	Upper Head Injection Accumulator
Number (per unit)	1	1	1 + 1 (shared)	4	1
Type	Atmospheric	Atmospheric	Atmospheric	Pressurized	Pressurized
Nominal Water Volume (gal)*	370,000 to 375,000	175,000 (est)	6,542	7,857 to 8,071	13,500 to 13,850 (1,805 to 1,851 ft <sup>3</sup> )
Minimum Water Volume in Mode 5 (gal)**	35,443	---	2,175	---	---
Operating Pressure (psig)	---	---	---	385 to 447	1,185 to 1,285
Boron Concentration (ppm)	2,000***	0	20,000 to 22,500	1,900 to 2,100	1,900 to 2,100
Normal Status in Mode 5	Available as a water source for RHR, charging, safety injection and containment spray pumps.	Available as a water source for the chemical and volume control system (CVCS). Boration may be required.	Available for boration of CVCS makeup water from primary water storage tank.	Inoperable. Single motor isolation valve in each discharge line to RCS cold leg shut and circuit breaker for valve motor racked out.	Inoperable. Redundant hydraulic valves in each injection line shut and locked with gag. motor-operated gag.

\* Modes 1 to 3 (all) and Mode 4 (RWST and BAST) as per NUREG-0789

\*\* NUREG-0789, Limiting Condition for Operation 3.1.2.5

\*\*\*2,000 to 2,100 ppm in Modes 1 to 4

Table 4-3. Summary of Major Support System Requirements.

Component	AC Electric Power Load Group	Component Cooling Water		Essential Raw Cooling Water Loop
		Header	Heat Exchanger	
RHR Pump 1A-A	1A	1A	A	
RHR Heat Exchanger 1A	-	1A	A	
Centrifugal Charging Pump 1A-A	1A	1A	A	
SI Pump 1A-A	1A	1A	A	
CCW Pump 1A-A	1A	1A	A	
CCW Pump C-S (Alt)	1A(alt)	1B	C	
CCW Heat Exchanger A	-	1A	-	1B or 2A
ERCW Pump J-A	1A	-	-	1A or 2A
ERCW Pump K-A (Unit 2 Pump)	2A	-	-	1A or 2A
ERCW Pump Q-A	1A	-	-	2A or 1A
ERCW Pump R-A (Unit 2 Pump)	2A	-	-	2A or 1A
Diesel Generator 1A-A	1A	-	-	1A or 2B
RHR Pump 1B-B	1B	1B	C	
RHR Heat Exchanger 1B	-	1B	C	
Centrifugal Charging Pump 1B-B	1B	1B	C	
Reciprocating Charging Pump	1B	-	-	
SI Pump 1B-B	1B	1B	C	
CCW Pump 1B-B	1B	1A	A	
CCW Pump C-S	1B(norm)	1B	C	
CCW Heat Exchanger C	-	1B	-	1A or 2B
ERCW Pump L-B	1B	-	-	1B or 2B
ERCW Pump M-B (Unit 2 Pump)	2B	-	-	1B or 2B
ERCW Pump N-B	1B	-	-	2B or 1B
ERCW Pump P-B (Unit 2 Pump)	2B	-	-	2B or 1B
Diesel Generator 1B-B	1B	-	-	1B or 2A

Table 4-4. Summary of Plant Variables That Could Provide Indication of a LOCA.

Variable	Response During LOCA	Remarks
Pressurizer level	Decreasing level, or instrument pegged low.	1 cold calibrated and 3 hot calibrated channels. Low level alarm.
RCS pressure, wide range	Decreasing pressure, rapid decrease if RCS was initially water solid.	2 channels. No alarm when pressure decreases (already below low pressure alarm setpoint when in Mode 5).
RHR loop A/B flow rate	Flow erratic or drops to zero if pumps cavitate or become airbound.	Low flow alarm on "miniflow" pump recirculation line.
CVCS letdown flow rate	Flow decreases or drops to zero as low pressure letdown valve attempts to control RCS pressure.	No alarm
CVCS makeup flow rate	Increasing flow to control pressurizer level if steam bubble was maintained in pressurizer.	No alarm
CVCS Volume Control Tank (VCT) level	Decreasing level when letdown flow decreases, and makeup flow constant or increasing.	Low level alarm automatically initiates makeup, low-low level alarm shifts charging pump suction to RWST.
VCT makeup flow rate	Primary water and boric acid flow indicated when VCT level drops to low level setpoint.	Individual instruments monitor primary water and boric acid makeup flow rates. No alarm.
Remote position indication for selected valves	Unexpected valve position indicated.	To determine if a flow diversion path has been established due to operator error. Multiple valves served by common audible alarm with no reflash capability.
RWST level	Decreasing if charging pumps automatically realigned on low-low VCT level. Increasing if operator error establishes a return flow path from RHR system to the RWST.	Low level alarm
Containment pocket (or pit) sump level	Alarm if pocket sump flooded.	Part of RCS leak detection system. Single level sensor alarms in Incore Instrument Room. Not required in Mode 5*.
Reactor building floor water level	Alarm if containment floor flooded.	Sensor about 6 inches above floor level.
RHR pump room sump level	Alarm if flooding occurs in RHR pump room.	Extended operation of an airbound RHR pump following a LOCA may lead to pump mechanical seal failure and pump room flooding.
CVCS pump room sump level	Alarm if flooding occurs in CVCS pump room.	
Containment temperature	Increasing	Available on plant computer.
Containment atmosphere particulate activity	Increasing	Part associated with vent isolation operable in Mode 5.
Containment atmosphere gaseous radioactivity	Increasing	Part associated with vent isolation operable in Mode 5.
Containment purge exhaust radioactivity	Increasing	Not required in Mode 5**.

\*See NUREG-0789, LCO 3.4.6.1

\*\*See NUREG-0789, LCO 3.3.3.1

Table 4-5. Sequoyah Technical Specifications Applicable in Mode 5.

System	Limiting Conditions For Operation (LCO)	Surveillance Requirements	Remarks
Reactivity Control Systems	3.1.1.2	4.1.1.2	Shutdown margin
	3.1.2.1	4.1.2.1	Boration system flow paths
	3.1.2.3	4.1.2.3	Charging pumps
	3.1.2.5	4.1.2.5	Borated water sources
	3.1.3.3	4.1.3.3	Rod position indication
Instrumentation	3.3.1	4.3.1	Source range instrumentation
	3.3.3.1	4.3.3.1	Radiation monitoring
	3.3.3.3	4.3.3.3	Seismic instrumentation
	3.3.3.4	4.3.3.4	Meteorological instrumentation
	3.3.3.6	4.3.3.6	Chlorine detection system
	3.3.3.8	4.3.3.8	Fire protection instrumentation
	3.3.3.9	4.3.3.9	Rad. liquid effluent monitors
	3.3.3.10	4.3.3.10	Rad. gas effluent monitors
Reactor Coolant System	3.4.1.4	4.4.1.4	Number of RHR loops
	3.4.2	4.4.2	Number of safety valves
	3.4.8	4.4.8	Coolant specific activity
	3.4.9.1	4.4.9.1	RCS temperature-pressure limits
	3.4.9.2	4.4.9.2	Pressurizer temperature and delta T limits
	3.4.10	4.4.10	Structural integrity
Plant Systems	3.7.2	4.7.2	Steam generator minimum temperature-pressure limits
	3.7.7	4.7.7	Control room emerg. HVAC
	3.7.9	4.7.9	Some snubbers
	3.7.10	4.7.10	Sealed sources
	3.7.11	4.7.11	Fire suppression system
	3.7.12	4.7.12	Fire barrier penetrations
Electric Power Systems	3.8.1.2	4.8.1.2	AC power
	3.8.2.2	4.8.2.2	AC distribution
	3.8.2.4	4.8.2.4	DC distribution
	3.8.3.2	4.8.3.2	MOV thermal overload protection
Special Test Exception	3.10.5	4.10.5	Rod position indication
Radioactive Effluents	3.11.1	4.11.1	Liquid effluents
	3.11.2	4.11.2	Gaseous effluents
	3.11.3	4.11.3	Solids
	3.11.4	4.11.4	Total dose
Radiological Environmental Monitoring	3.12.1	4.12.1	Monitoring program
	3.12.2	4.12.2	Land use census
	3.12.3	4.12.3	Interlaboratory comparison program

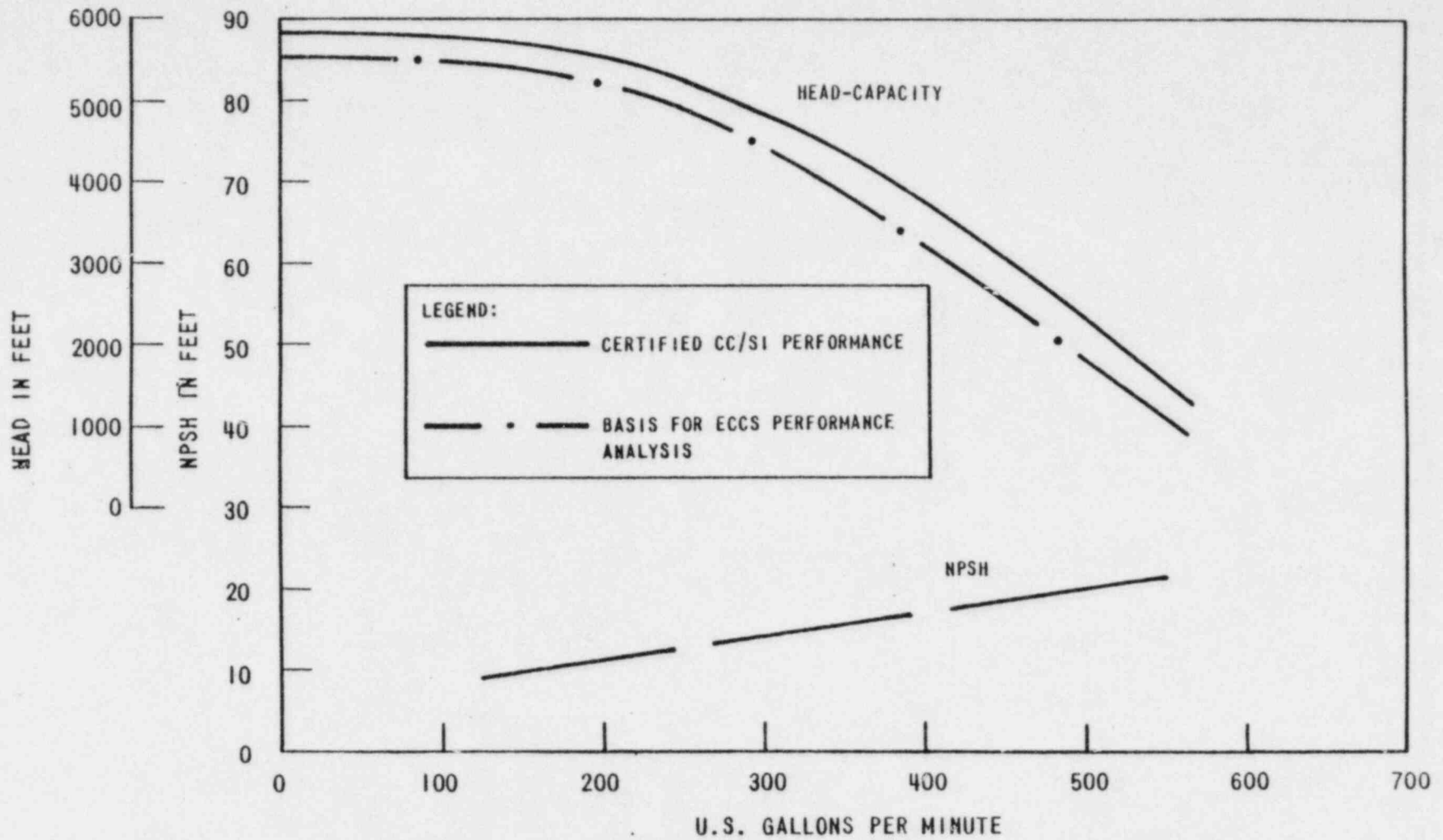


Figure 4-1. NPSH and Head Capacity Curves for Charging Pumps (From Sequoyah FSAR).

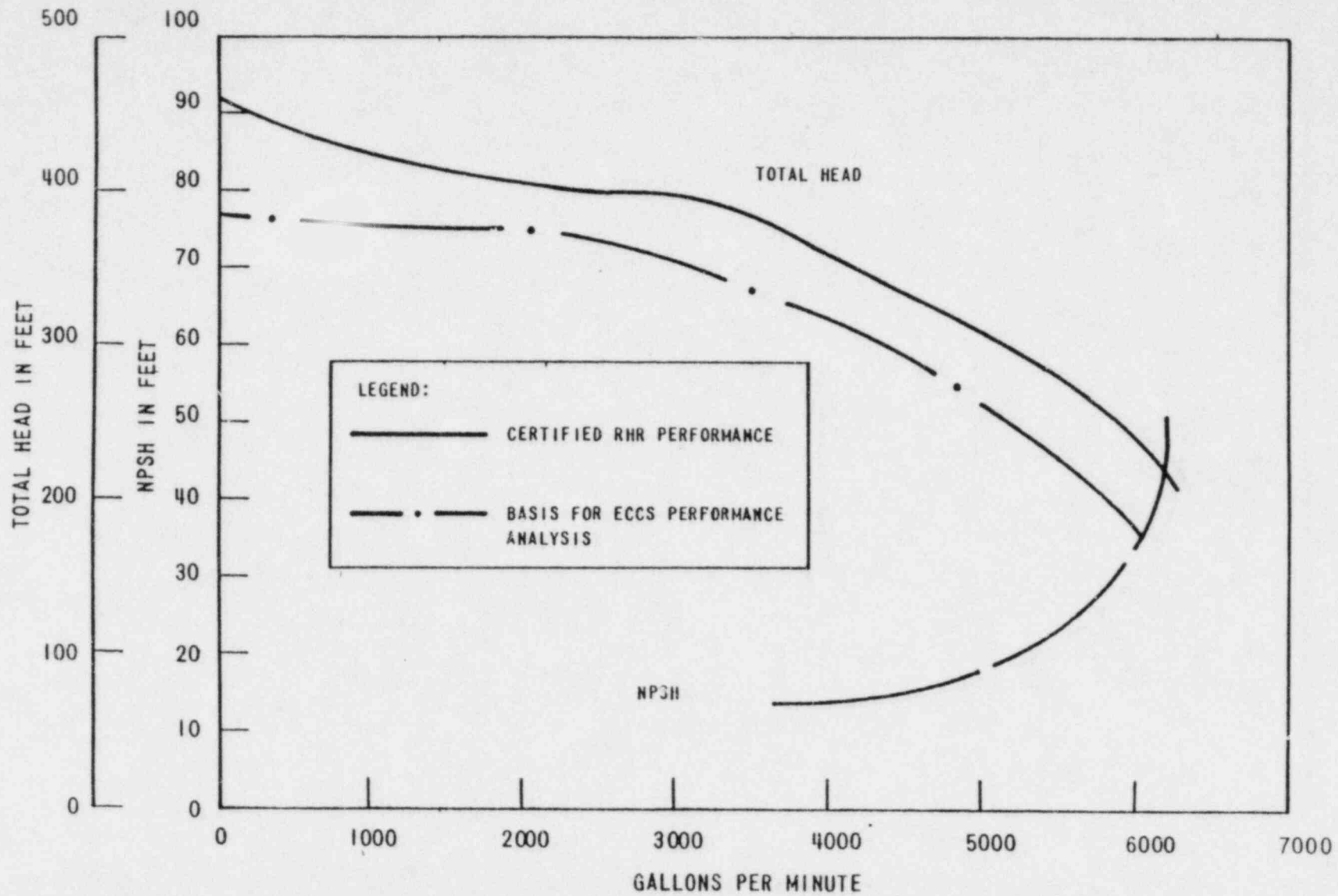


Figure 4-2. NPSH and Head Capacity Curves for RHR Pumps (from Sequoyah FSAR).

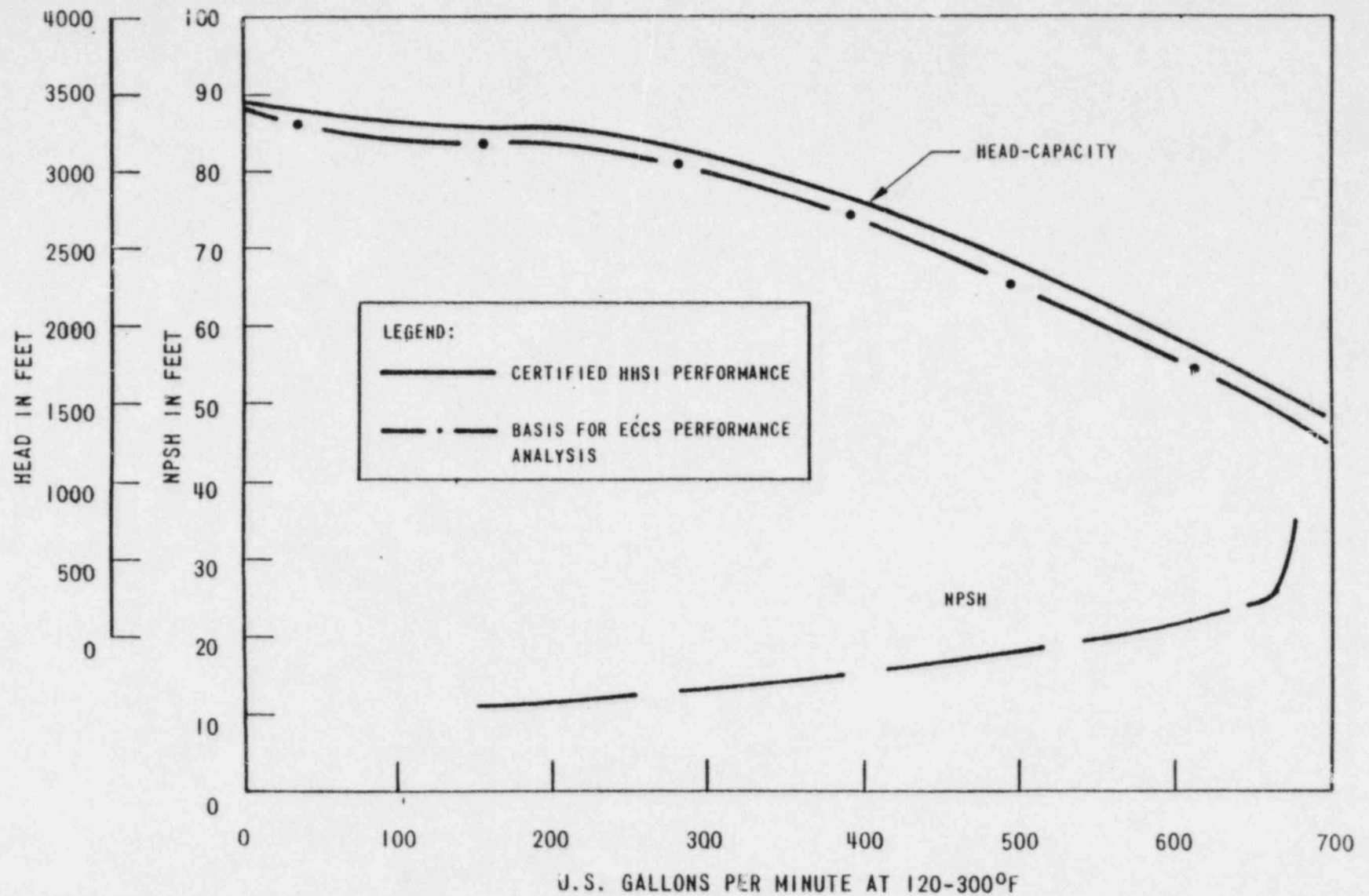


Figure 4-3. NPSH and Head Capacity Curves for Safety Injection Pumps (from Sequoyah FSAR).

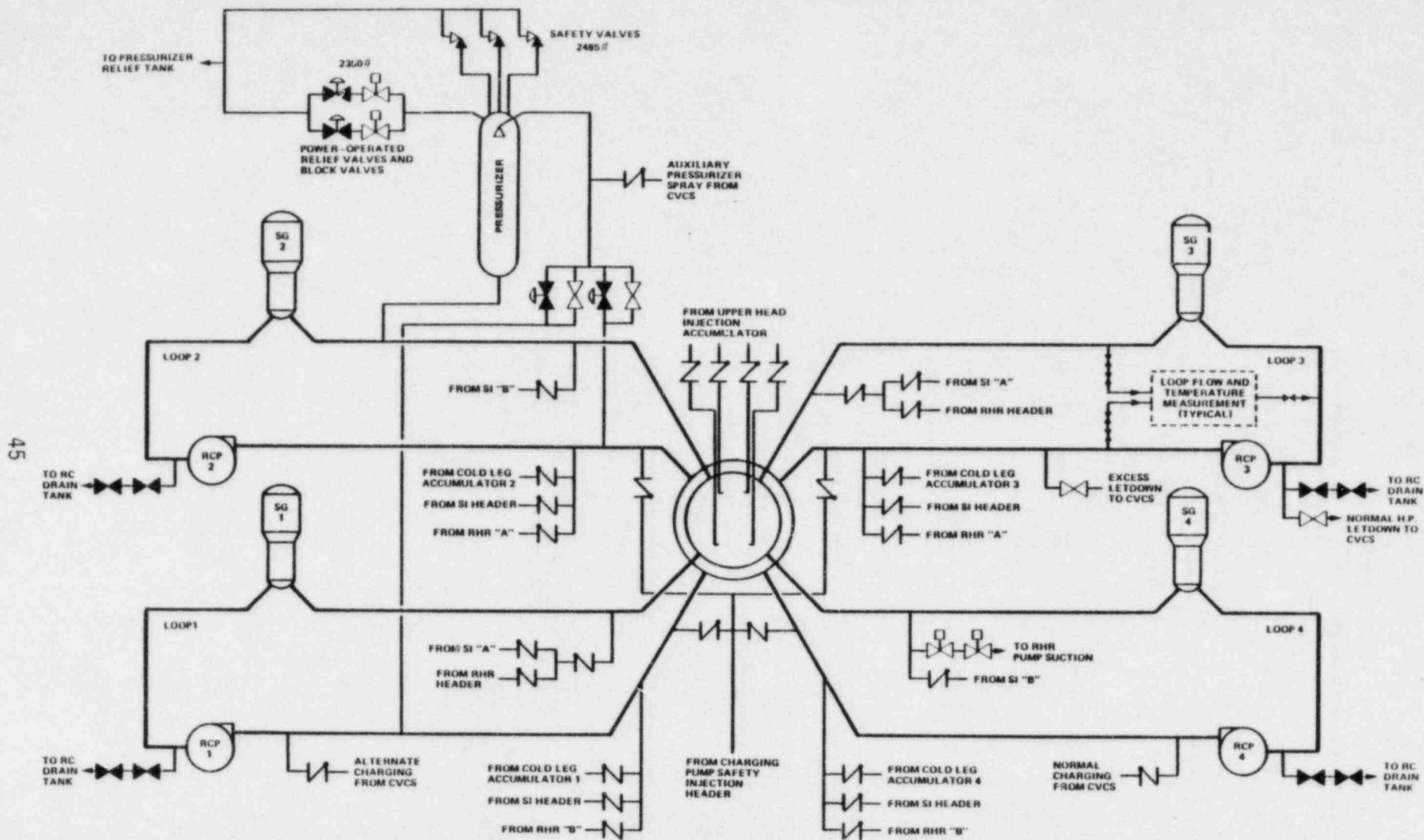


Figure 4-4. Sequoyah Reactor Coolant System and Interfacing Piping.



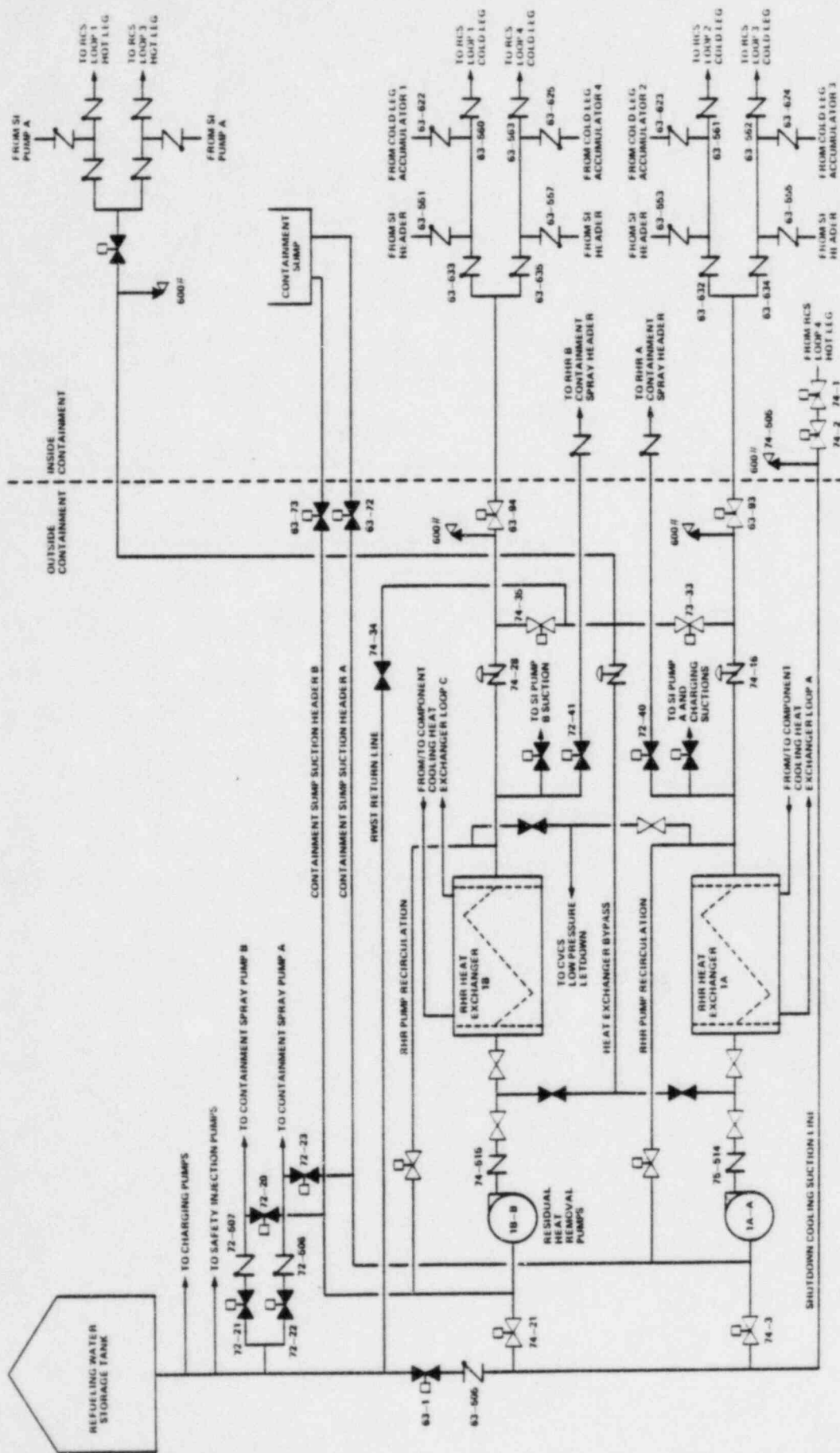


Figure 4-5. Sequoyah Residual Heat Removal System.

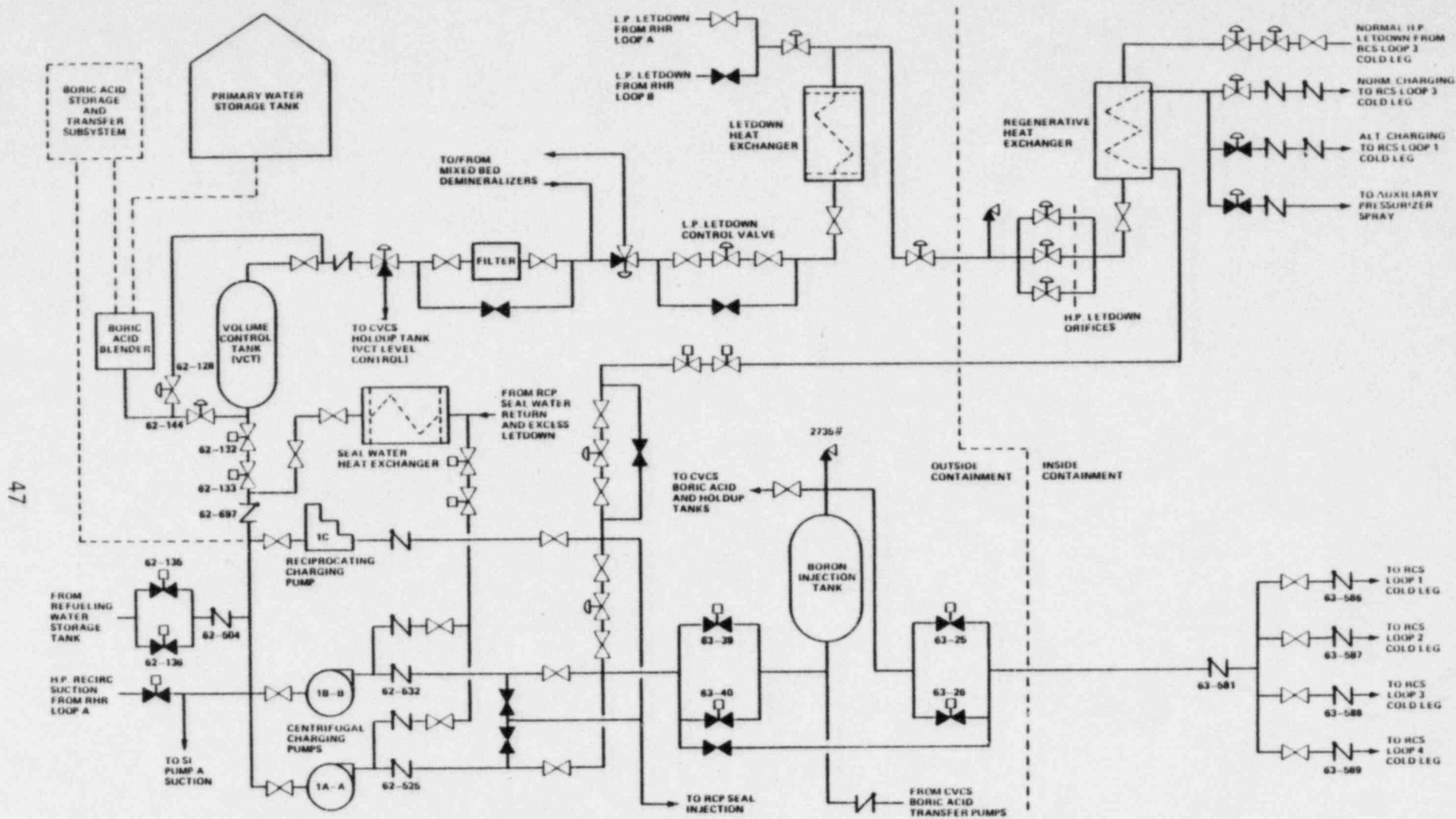


Figure 4-6. Sequoyah Chemical and Volume Control System.

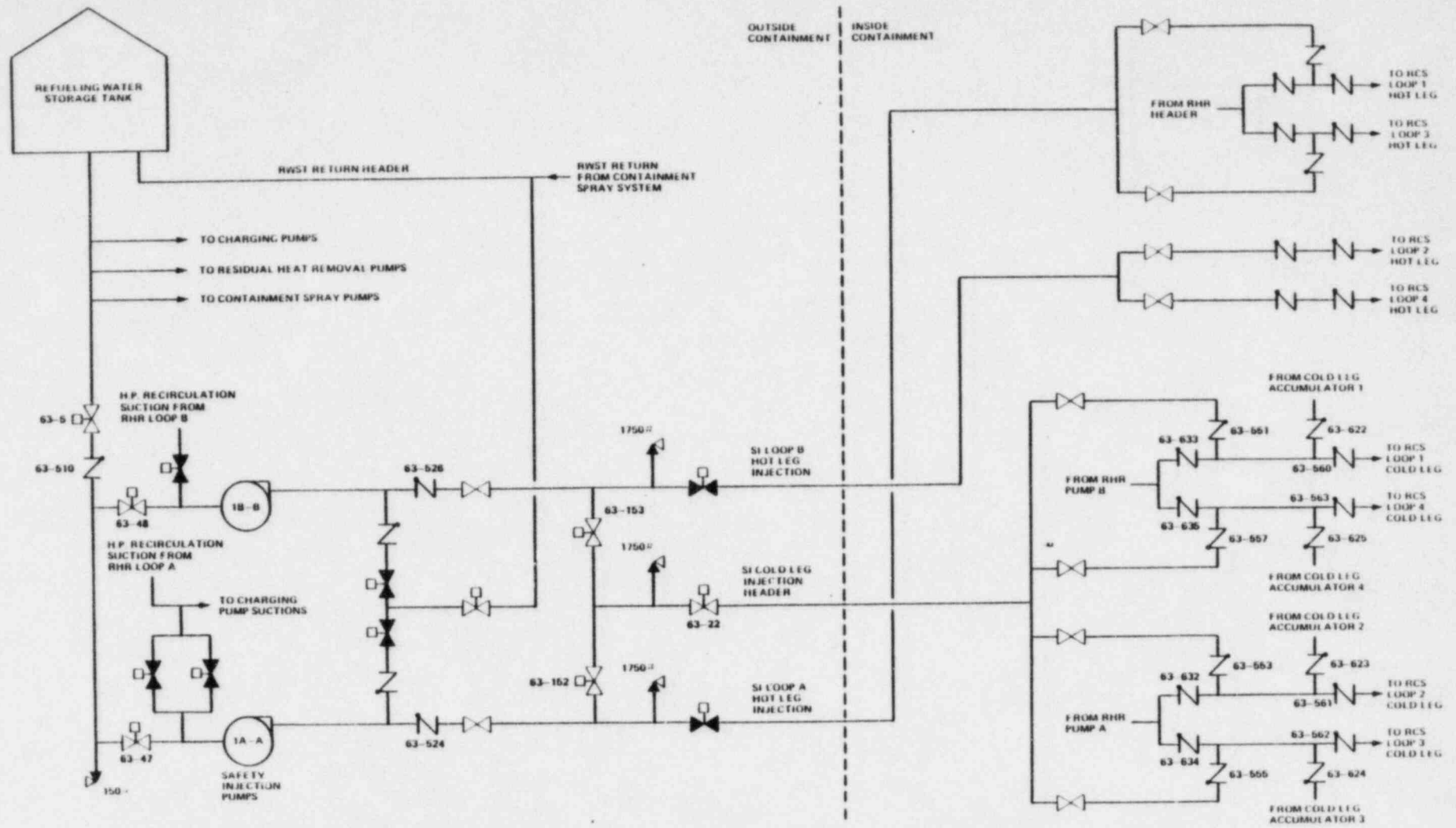


Figure 4-7. Sequoyah High Pressure Safety Injection System.

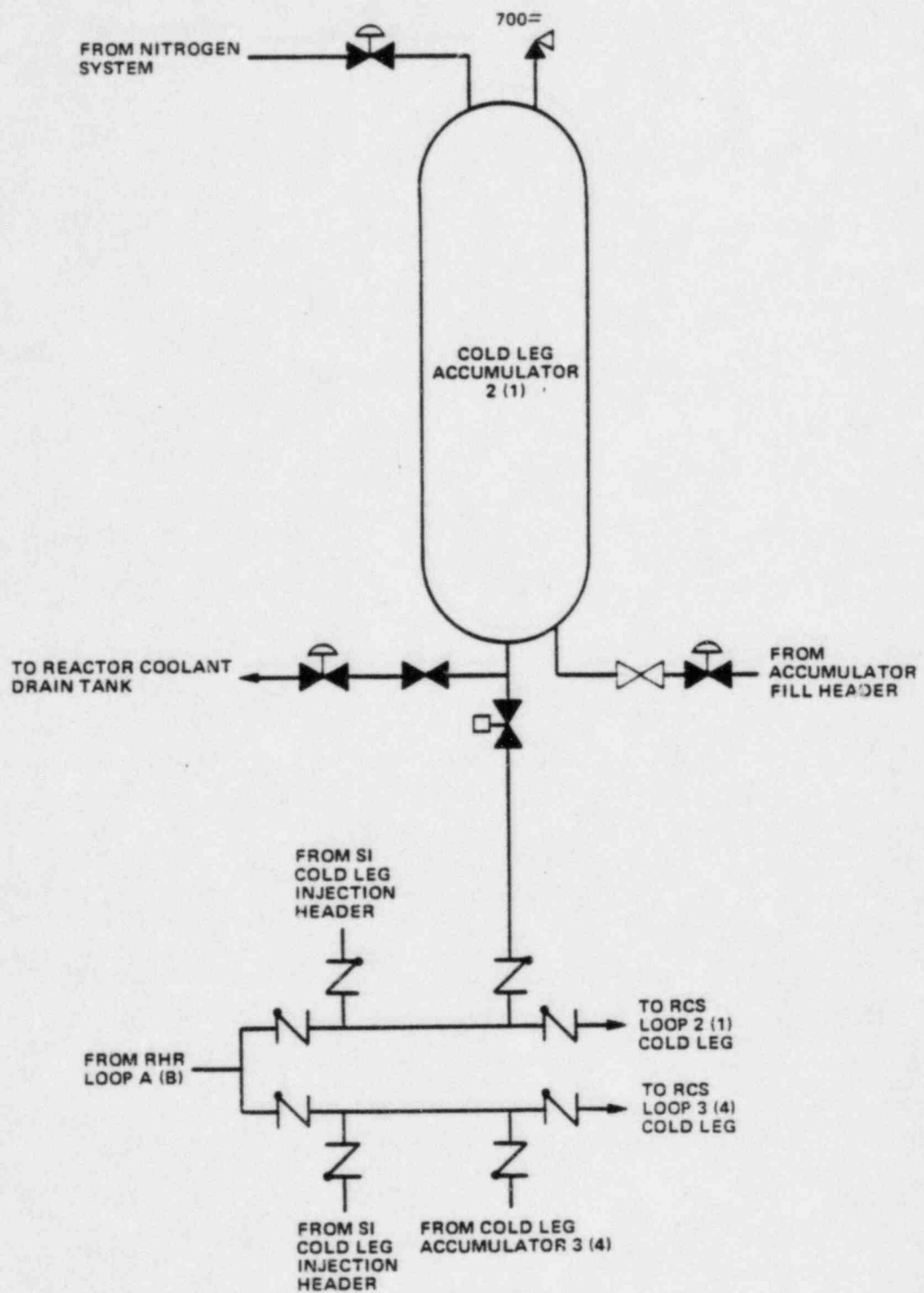


Figure 4-8. Typical Cold Leg Accumulator.

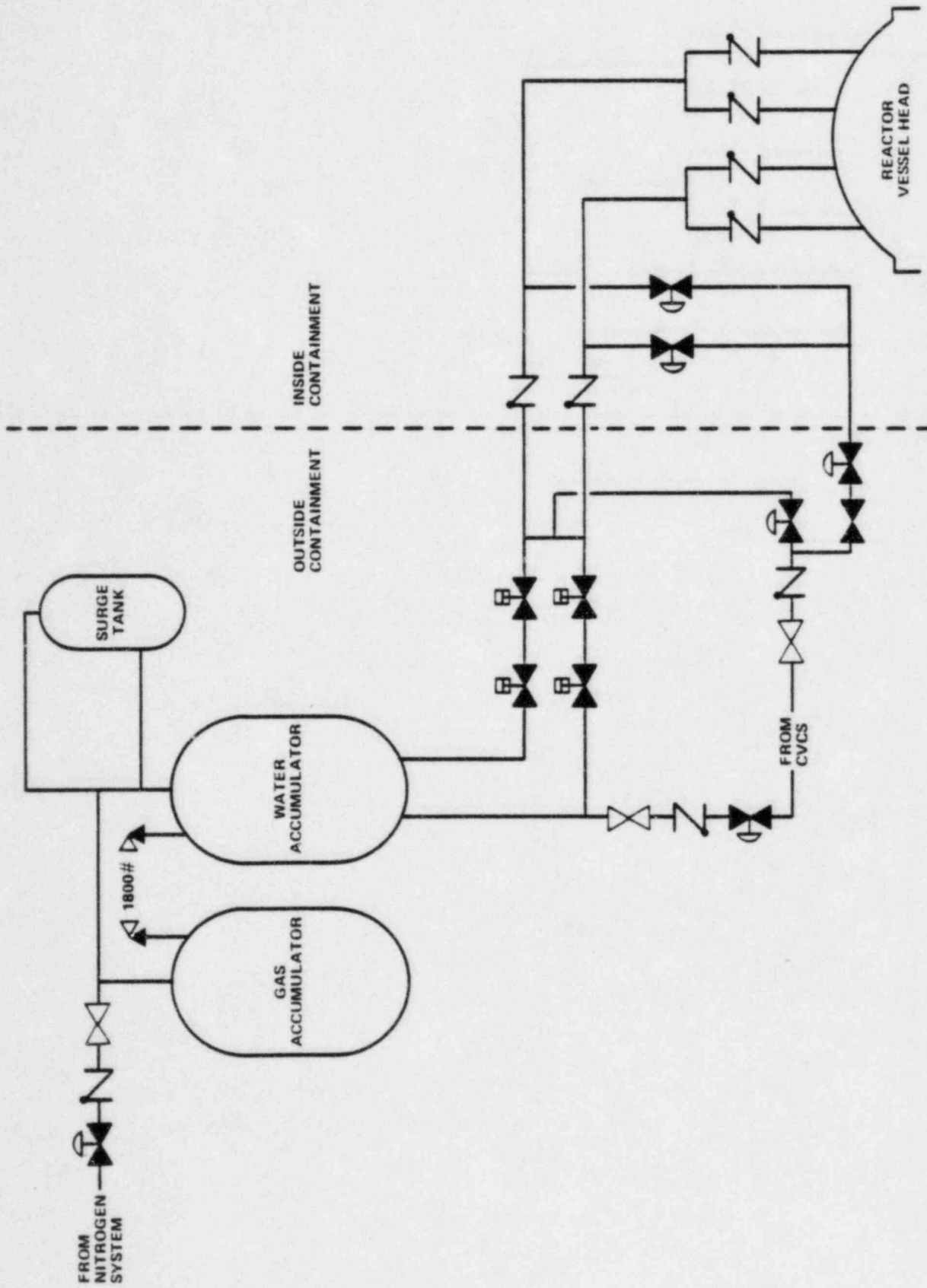
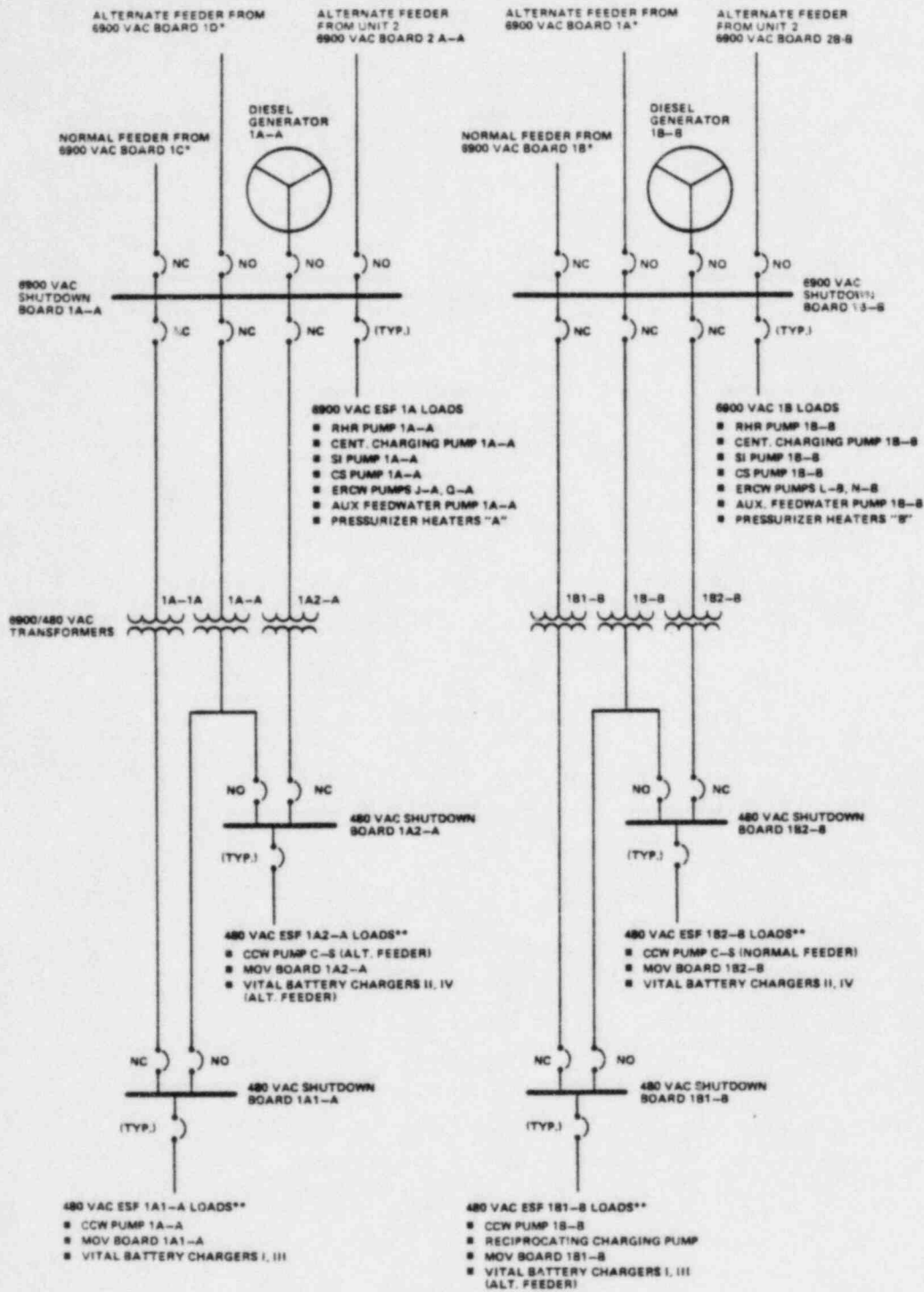


Figure 4-9. Upper Head Injection System.



- NOTES:
- \* 9900 VAC BOARD 1A, 1B, 1C AND 1D CAN RECEIVE POWER FROM THE UNIT 1 MAIN TURBINE GENERATOR OR ONE OF TWO STARTUP TRANSFORMERS
  - \*\* ONLY SELECTED 480 VAC LOADS ARE LISTED
  - † ABBREVIATIONS USED IN THIS FIGURE ARE AS FOLLOWS:
- |      |   |                             |
|------|---|-----------------------------|
| ESF  | = | ENGINEERED SAFETY FEATURE   |
| RHR  | = | RESIDUAL HEAT REMOVAL       |
| SI   | = | SAFETY INJECTION            |
| CS   | = | CONTAINMENT SPRAY           |
| ERCW | = | ESSENTIAL RAW COOLING WATER |
| CCW  | = | COMPONENT COOLING WATER     |
| MOV  | = | MOTOR-OPERATED VALVE        |
| NO   | = | NORMALLY OPEN               |
| NC   | = | NORMALLY CLOSED             |

Figure 4-10. Sequoyah Class 1E AC Electric Power System.

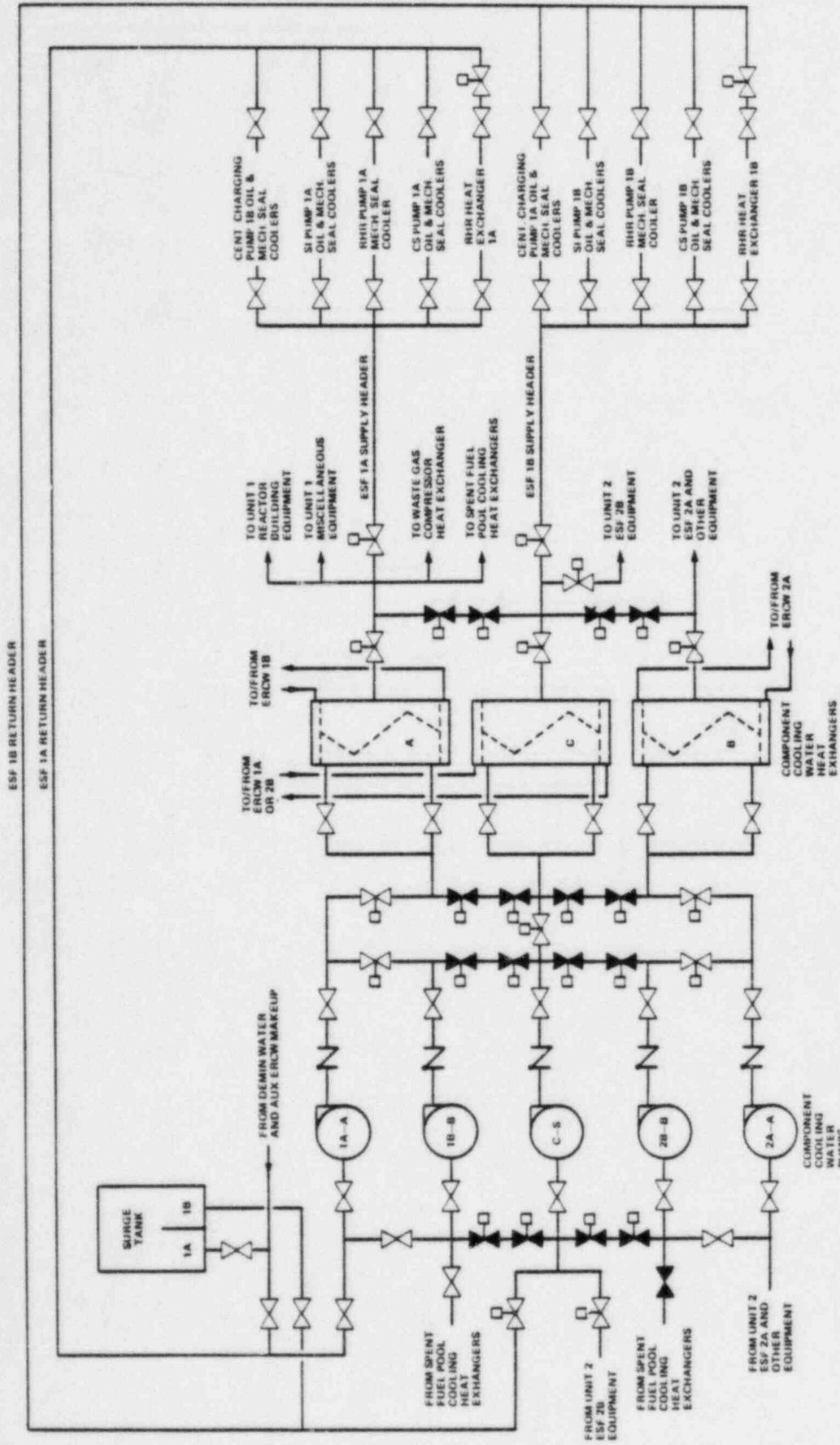


Figure 4-11. Sequoyah Component Cooling Water System.

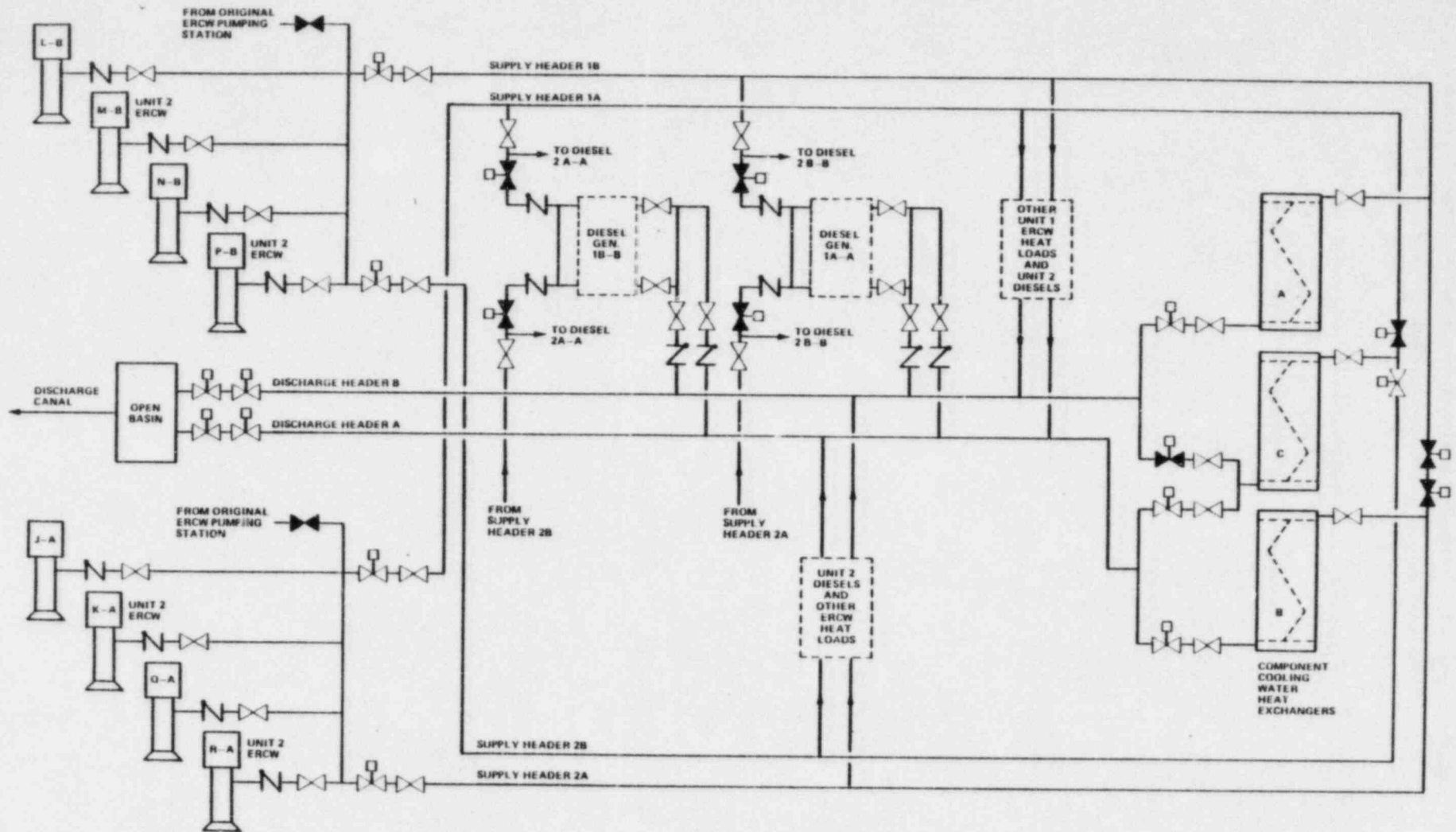


Figure 4-12. Sequoyah Essential Raw Cooling Water System.



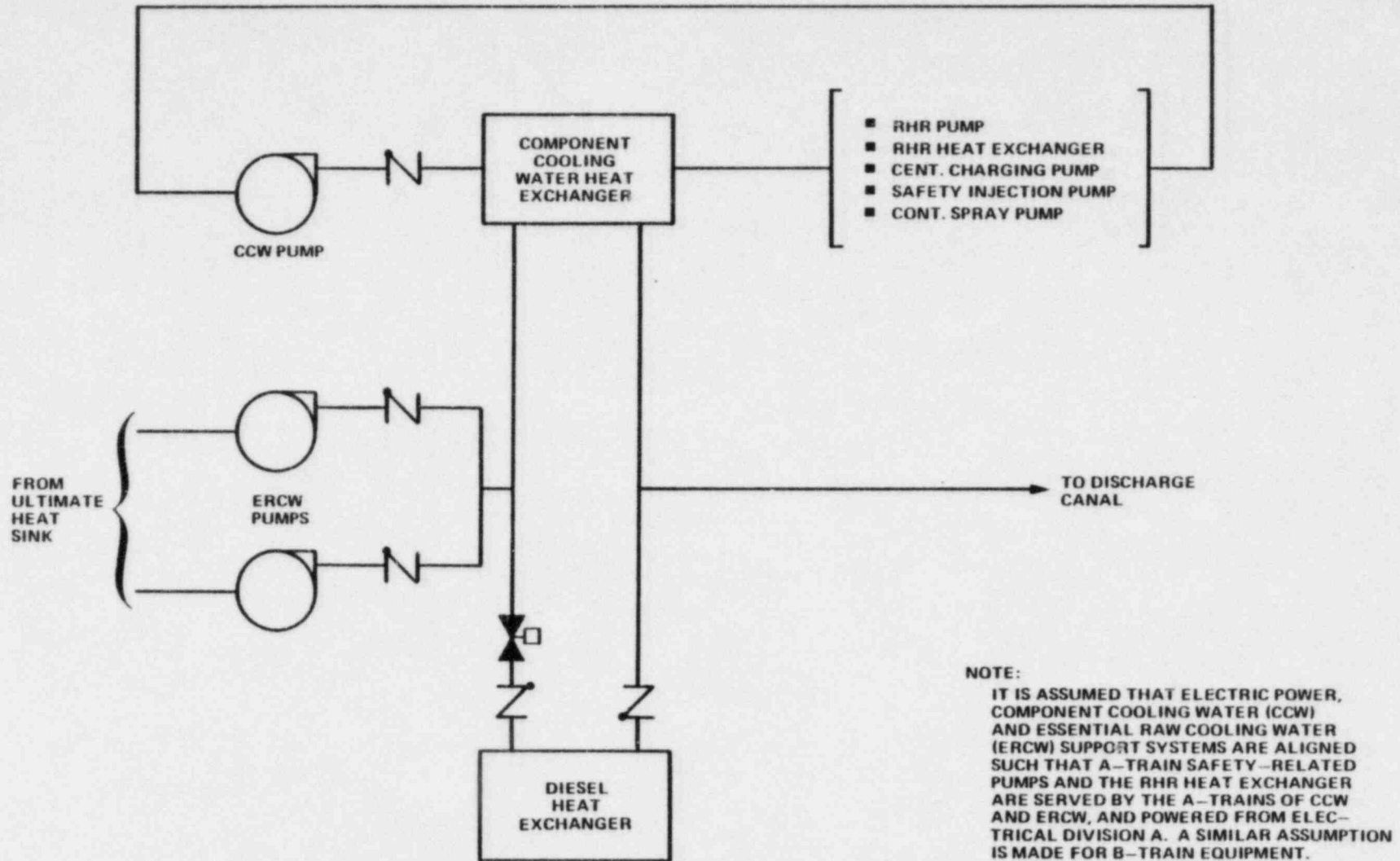


Figure 4-13. Simplified Model of Sequoyah CCW and ERCW Cooling Loop.

## 5. EVENT TREES FOR RESPONSE TO MODE 5 LOCAs

Event trees were developed to describe the Sequoyah plant response to postulated LOCAs. The following four LOCA event trees are presented in this section:

- Large LOCA L<sub>2</sub> (2 RHR pumps initially operating)
- Large LOCA L<sub>1</sub> (1 RHR pump initially operating)
- Medium LOCA
- Small LOCA

In these event trees, a successful event sequence (e.g., no core melt) is one in which an adequate long-term core cooling capability can be established following a successful short-term response. Long-term core cooling can be provided either by establishing an RHR recirculation flow path between the containment active sump and the RCS, or by providing continuous makeup to the RCS at a rate that at least equals the RCS coolant boiloff rate.

In the event trees, multi-mode systems such as the RHR system and the CVCS are modeled as multiple events including: (a) an event to represent the portion of the system that is common to all operating modes, and (b) additional events as necessary to represent the components associated with each unique operating mode of the system.

Within each event tree, individual events are arranged under the functional groupings listed in Table 5-1. Containment response to the LOCA is not included in the event trees.

### 5.1 EVENT TREE GENERAL ASSUMPTIONS

The following general assumptions were made in the development of the cold shutdown LOCA event trees.

- An operator may commit an error which initiates a LOCA. During

response to a given LOCA, operator errors of omission are considered (e.g., operator fails to take a necessary action) as well as some operator errors of commission (e.g., operator creates a diversion path in a response system).

- Information required for the operator to make timely decisions regarding LOCA response is available from control room instrumentation. In NRC Inspection and Enforcement Report No. 50-327/81-07 (Ref. 6) it was indicated that the Sequoyah plant operators reacted rapidly to the basic symptoms of a Mode 5 LOCA (e.g., decreasing or zero RCS pressure and pressurizer level). It was not necessary for a diagnosis of the cause of the LOCA to be made before responding appropriately to the symptoms of the LOCA.
- If the LOCA is initiated by operator error (e.g., valve misalignment), no subsequent operator action is taken to terminate the LOCA (e.g., by restoring proper valve lineup). In the NRC I&E report cited above (Ref. 6), it was indicated that the plant operators did not diagnose the cause of the loss of reactor coolant and the incorrect valve lineup was corrected only after "the auxiliary operator returned to the control room and reported that he had opened the RHR spray valve." Therefore, no credit is taken for operator diagnosis and termination of the cause of a LOCA.
- Only water that is adequately borated will be used for reactor coolant makeup. Boron dilution events caused by makeup with inadequately borated or unborated water were not considered.
- Only one centrifugal charging, RHR or HPSI pump is necessary for adequate coolant makeup following a LOCA.
- Operator is assumed to utilize injection systems in the following order: charging pump, RHR pump, HPSI pump.
- The refueling water storage tank (RWST) is a finite water supply with an available inventory of about 350,000 gallons upon entering

Mode 5. Replenishment of the RWST from alternate water sources is not considered.

- Charging, RHR and safety injection pumps can supply water from the RWST to the RCS at their respective runout flow rates following a large LOCA.
- The primary water storage tank (PWST) and the boric acid storage tank (BAST) together constitute an essentially infinite water source assuming alternate water sources can be aligned to replenish the PWST, and batches of boric acid can be produced manually to replenish the BAST.
- Nominal continuous makeup rate of properly borated water to the CVCS volume control tank (VCT) from the PWST and BAST is 150 gpm.
- No credit is taken for the limited amount of water in the cold leg and UHI accumulators.
- The LOCA diverts flow from one of the coolant injection paths available to the system being used for coolant makeup.
- If the RHR suction safety valve and isolation valves fail to operate and the RHR system is pressurized above its design pressure, system will fail.
- If the reactor coolant system is pressurized significantly above the RCS minimum pressure and temperature curve limits, an unmitigatable reactor vessel failure may occur.
- Containment recirculation sump failure is not considered as a potential contributor to failure of the recirculation mode of the RHR system.

## 5.2 LARGE LOCA L<sub>2</sub> EVENT TREE

The large LOCA L<sub>2</sub> event tree is shown in Figure 5-1. Two RHR pumps are assumed to be operating prior to the LOCA. RHR pump suction will be rapidly lost during the LOCA, causing these pumps to cavitate and become airbound. It is assumed that an airbound RHR pump will eventually fail if not secured by the operator. Additional information on pump failure when airbound is presented in Reference 7.

A representative time line for plant response following a large LOCA initiated 20 hours after shutdown is shown in Figure 5-2. If no operator action is taken, reactor coolant will boil down to the core mid-plane in approximately 52 minutes (see Section 3.3). Coolant injection systems can provide makeup from the RWST for a variety of times, based on the runout flow rate of the respective pump. An RHR pump can empty the RWST in about one hour. An SI pump and a centrifugal charging pump take about 9 and 11 hours, respectively, to empty the RWST. When the RWST is empty, the RHR system must be aligned for containment sump recirculation if core melt is to be prevented.

The normal CVCS makeup source (e.g., the PWST and BAST) is also available, but at a maximum continuous makeup rate of 150 gpm. If it is assumed that the charging pumps are aligned to the ECCS injection path via the boron injection tank (see Section 4.1), one of four coolant injection paths may be affected by the LOCA, resulting in only 113 gpm reaching the reactor vessel. This makeup rate can match the coolant boil-off rate at approximately 35 hours after reactor shutdown. If the RWST can serve as a coolant makeup source until 35 hours after reactor shutdown, the following two protection options are available to the operator: (a) align the RHR system for containment sump recirculation, or (b) provide long-term, properly borated makeup to the RCS from the PWST and BAST. A time line illustrating the time constraints associated with successful LOCA mitigation using the normal CVCS water source is shown in Figure 5-3.

Use of the normal CVCS return path was not considered because it represents a less effective makeup capability than the ECCS injection path. At best, the operator could be assured that coolant from only one of two makeup paths was reaching the reactor vessel following a LOCA (e.g., the LOCA directly affects one injection path). In this case, the actual coolant makeup rate would be 75 gpm.

Although not modeled in the event tree, a potential role for the cold leg and UHI accumulators can be seen in Figure 5-3. Each accumulator contains sufficient water volume to reflood the reactor vessel from the core mid-plane to the hot leg nozzles. Two cold leg accumulators could be dumped simultaneously to ensure that water from at least one reached the core. The UHI accumulator could be depressurized and dumped directly into the reactor vessel. Altogether, these accumulator could provide approximately three hours of core cooling during the period from 20 to 35 hours after reactor shutdown (assuming the operator knew the optimum time to dump each accumulator). By adding this three hours of core cooling to the time lines in Figure 5-3, a large LOCA occurring three hours earlier could be successfully mitigated using the normal CVCS water source. Alternatively, core melt could be delayed for three hours. In the latter case, core melt would not be prevented, but additional time is available to restore containment isolation and to implement the site emergency plan.

### 5.3 LARGE LOCA $L_1$ EVENT TREE

The large LOCA  $L_1$  event tree is shown in Figure 5-4. One RHR pump is assumed to be operating prior to the LOCA. If the operating RHR pump becomes airborne and fails, the idle RHR pump is available and can be aligned for coolant injection or containment sump recirculation. Other aspects of this event tree are comparable to the large LOCA  $L_2$  event tree described previously.

### 5.4 MEDIUM LOCA EVENT TREE

The medium LOCA event tree is shown in Figure 5-5. This event tree introduces additional events related to RHR and RCS overpressurization when a high-head pump (e.g., centrifugal charging or HPSI pump) is used for coolant injection. The high-head pumps are capable of providing makeup at a rate greater than the LOCA leak rate at approximately 600 psig RCS pressure. Ultimately, these pumps could restore RCS coolant inventory and pressurize the RHR and/or the RCS above their respective pressure limits if the operator fails to control pressure and the RHR overpressure protection features fail. RHR pipe rupture or reactor vessel failure may result from this overpressurization.

## 5.5 SMALL LOCA EVENT TREE

The small LOCA event tree is shown in Figure 5-6. This event tree includes the RHR and RCS overpressure protection events found in the medium LOCA event tree and introduces additional events related to core heat removal. At 600 psig RCS pressure, the small LOCA does not carry away all core heat. Without additional core heat removal, the RCS will gradually heat up and pressurize above RHR design pressure until the leak rate increases enough to establish an equilibrium RCS temperature and pressure.

Table 5-1. Grouping of Events in the Cold Shutdown LOCA Event Trees.

Event Tree	Functional Grouping of Events
Large LOCA, L <sub>1</sub> and L <sub>2</sub>	<ul style="list-style-type: none"> <li>● Equipment Protection (RHR Pumps)</li> <li>● Short-term Coolant Injection</li> <li>● Long-term Core Cooling</li> </ul>
Medium LOCA, M	<ul style="list-style-type: none"> <li>● Short-term Coolant Injection</li> <li>● RHR Overpressure Protection</li> <li>● RCS Overpressure Protection</li> <li>● Long-term Core Cooling</li> </ul>
Small LOCA, S	<ul style="list-style-type: none"> <li>● Normal Makeup</li> <li>● Short-term Coolant Injection</li> <li>● RHR Overpressure Protection</li> <li>● Core Heat Removal (also performs RCS overpressure protection function)</li> <li>● Long-term Core Cooling</li> </ul>



LARGE LOCA 2 RHR LOOPS INITIALLY OPERATING	EQUIPMENT PROTECTION		SHORT-TERM COOLANT INJECTION				LONG-TERM CORE COOLING		SEQUENCE No.	SEQUENCE	CORE STATUS
	OPERATING RHR PUMP SECURED	CVCS/ CHARGING COMMON SYSTEM	CHARGING ALIGNMENT FOR RHRST INJECTION	RHR COMMON SYSTEM	RHR ALIGNMENT FOR RHRST INJECTION	RHRST INJECTION VIA SI PUMPS	RHR ALIGNMENT FOR RHRST SECIRC	CVCS ALIGNMENT FOR NORMAL MAKEUP			
L <sub>2</sub>	A	B	C	D	E	F	G	H			

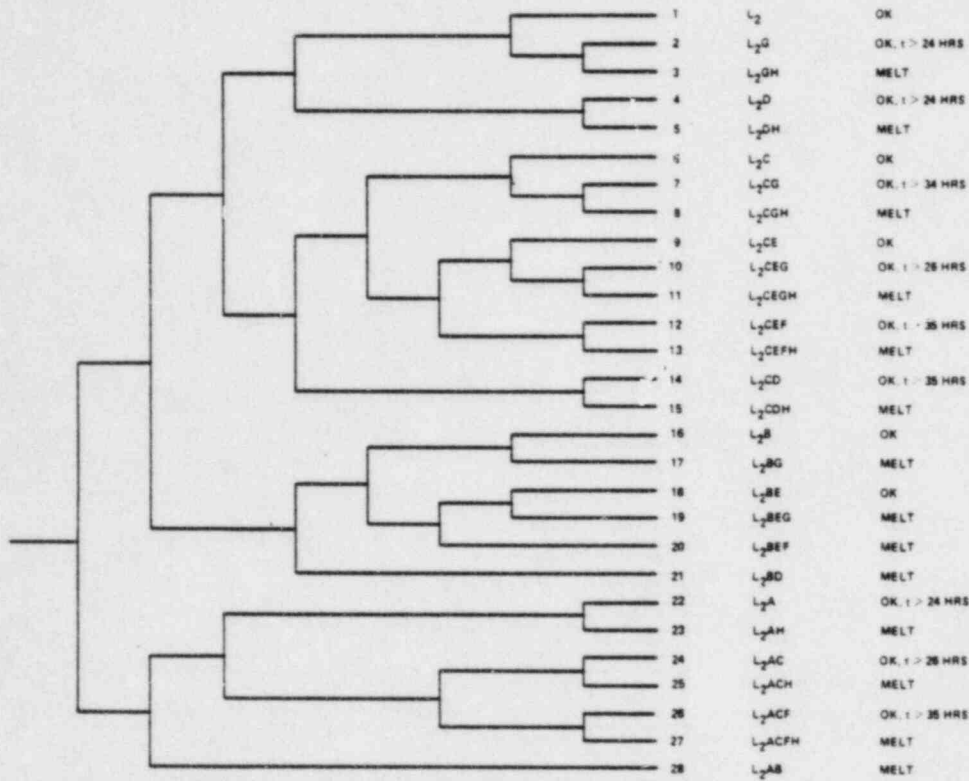


Figure 5-1. Large LOCA L<sub>2</sub> Event Tree.

OPERATOR LOCA RESPONSE ACTIONS

1. NO INITIAL OPERATOR RESPONSE
2. OPERATOR ALIGNS CHARGING PUMP TO INJECT FROM RWST (~ 550 gpm)
3. OPERATOR ALIGNS RHR PUMP TO INJECT FROM RWST (~ 5500 gpm)
4. OPERATOR RESTORES AND ALIGNS SI PUMP TO INJECT FROM RWST (~ 650 gpm)
5. OPERATOR ALIGNS RHR FOR LOW PRESSURE RECIRCULATION
6. OPERATOR FAILS TO ESTABLISH RECIRCULATION FLOW PATH WHEN RWST IS EMPTIED

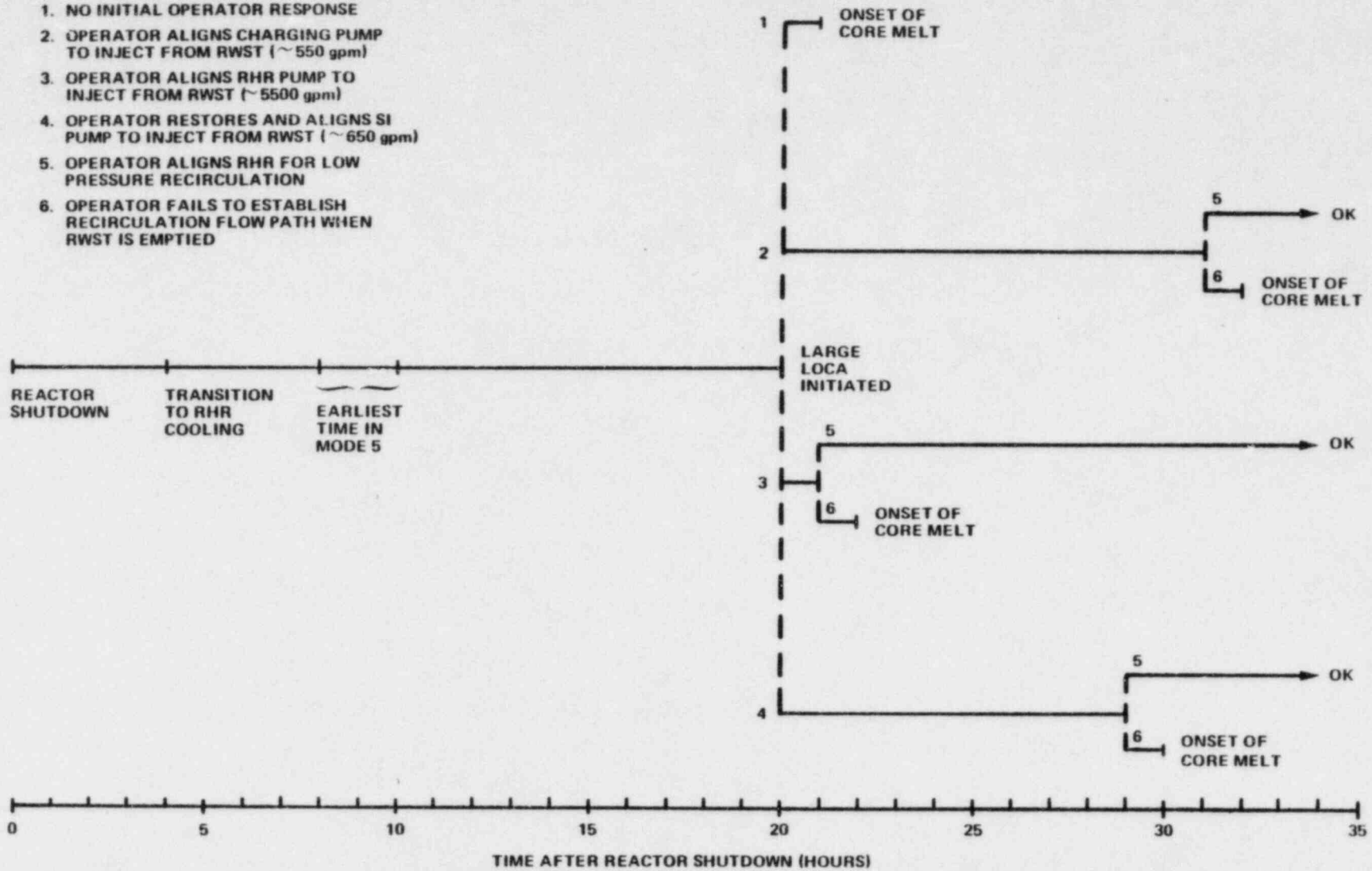


Figure 5-2. Time Line for Plant Response Following Large LOCA Initiated at 20 Hours After Reactor Shutdown.

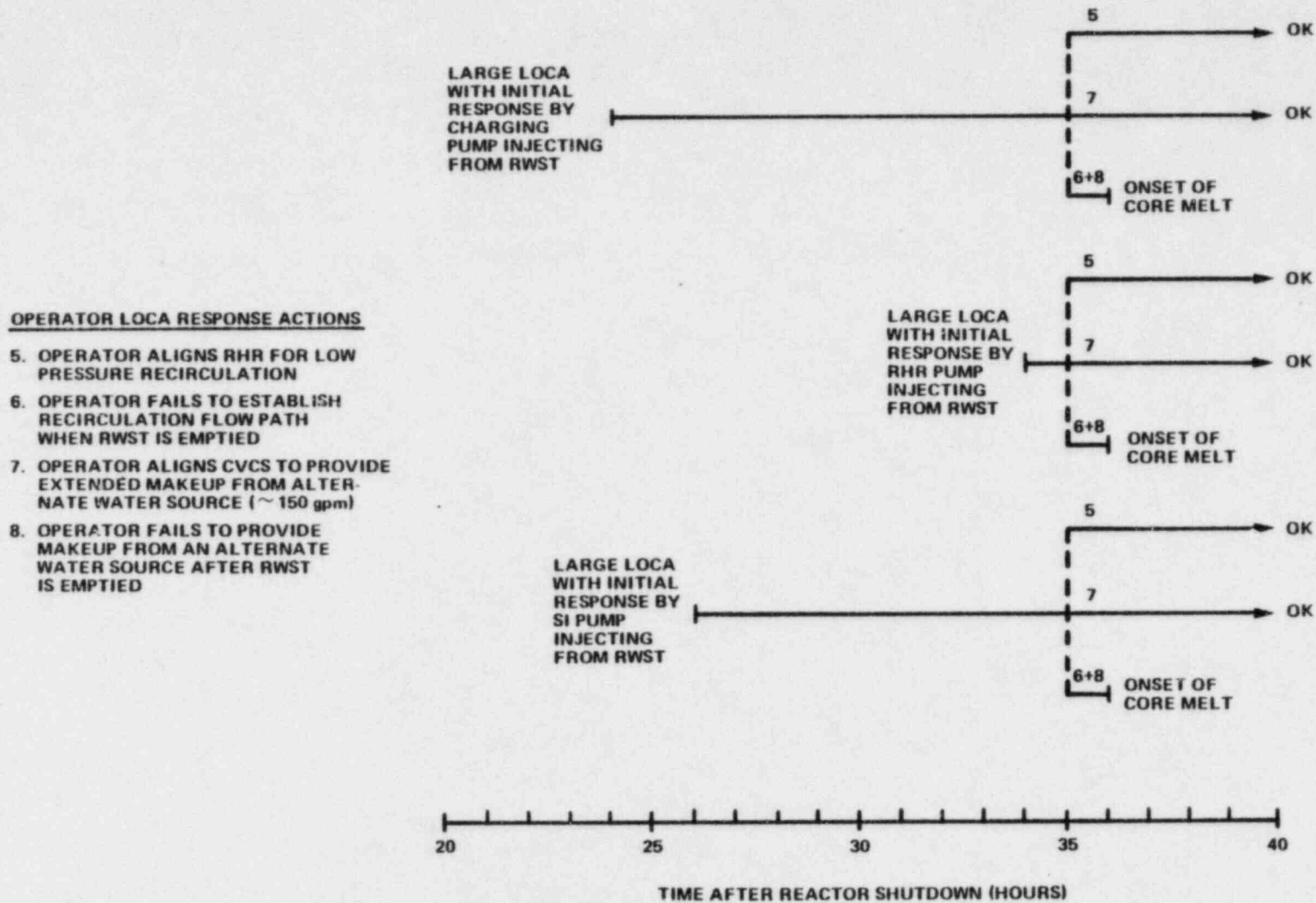


Figure 5-3. Time Line Illustrating Time Constraints Associated With Successful LOCA Mitigation With a 150 gpm Alternate Water Source.

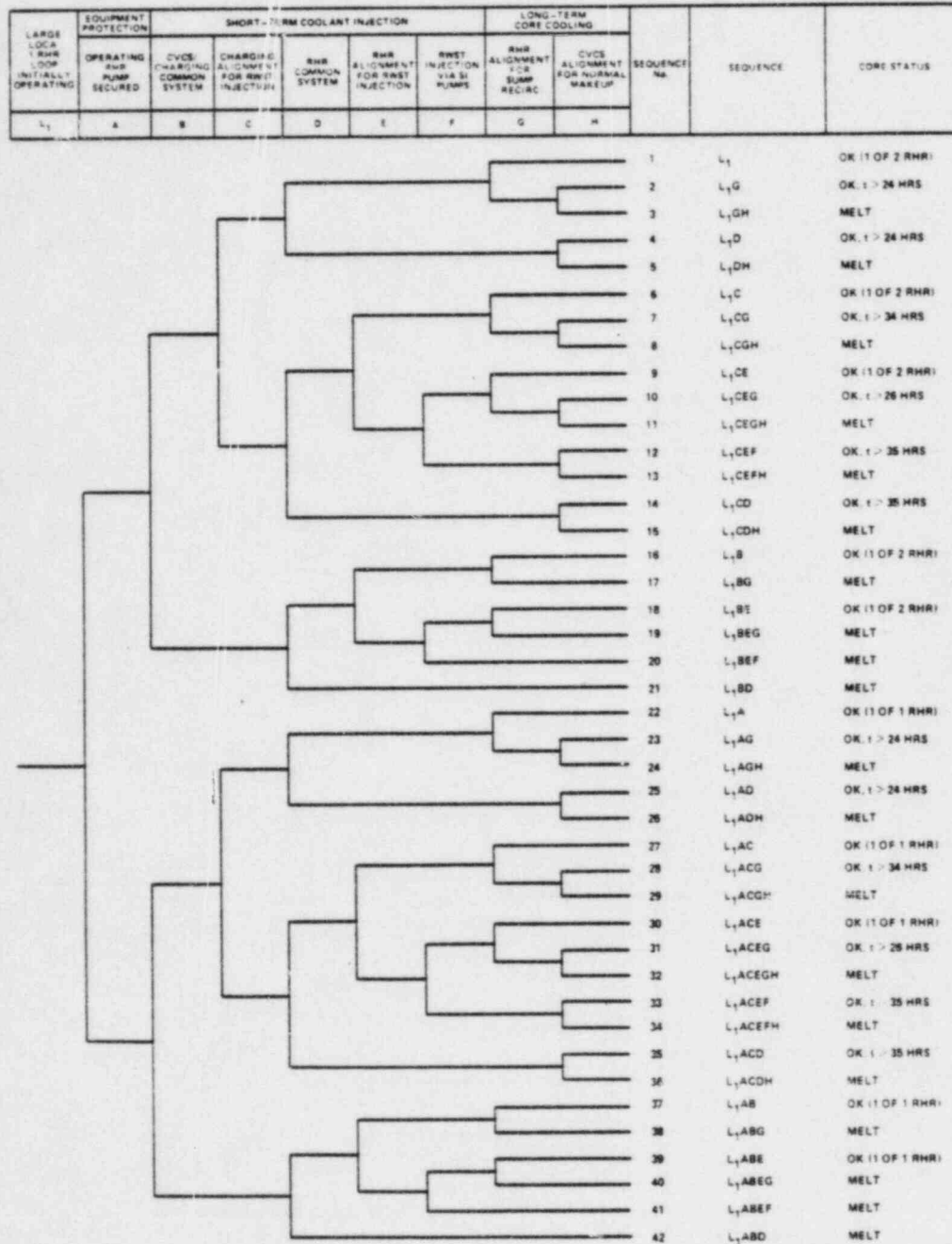


Figure 5-4. Large LOCA L<sub>1</sub> Event Tree.

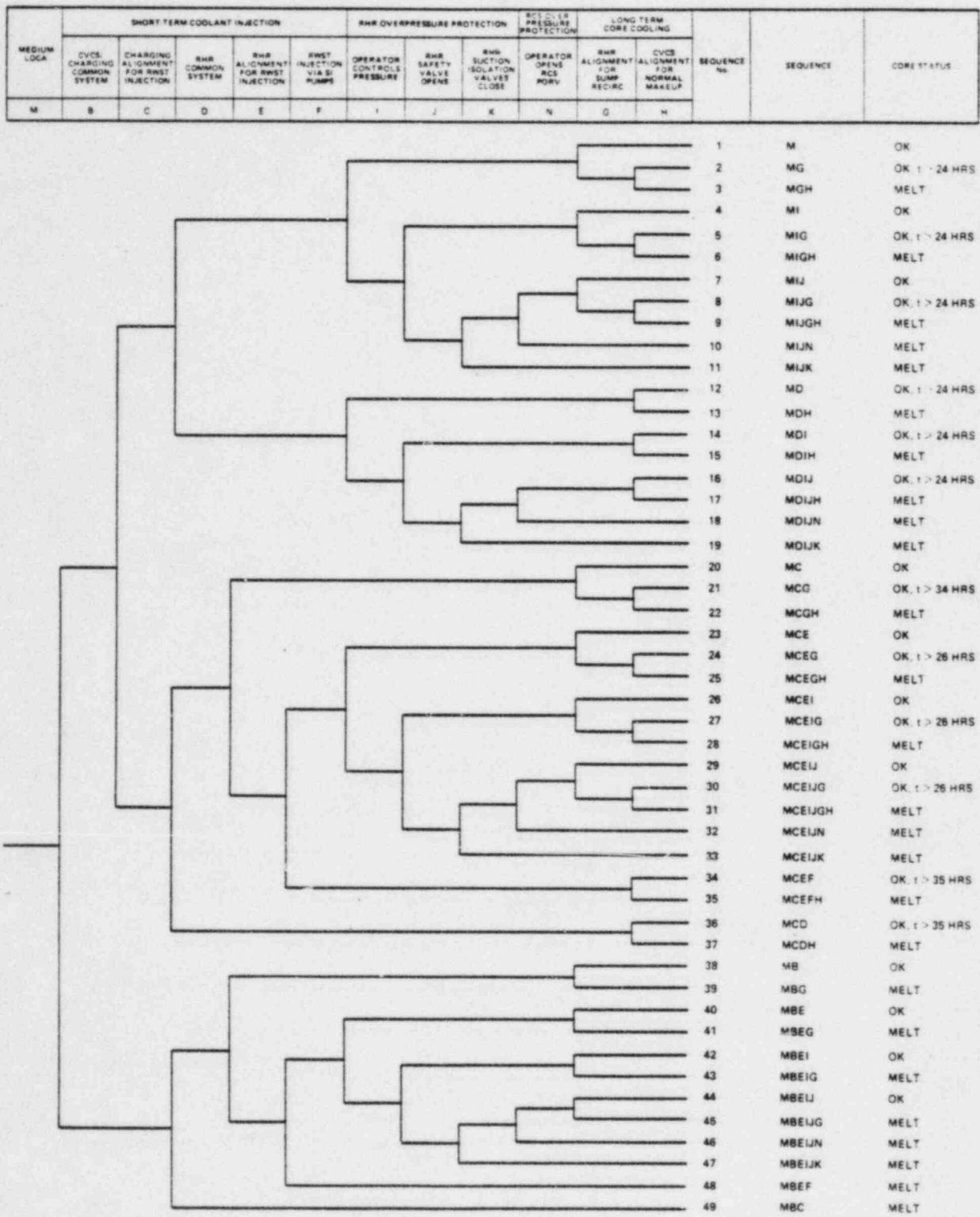


Figure 5-5. Medium LOCA Event Tree.

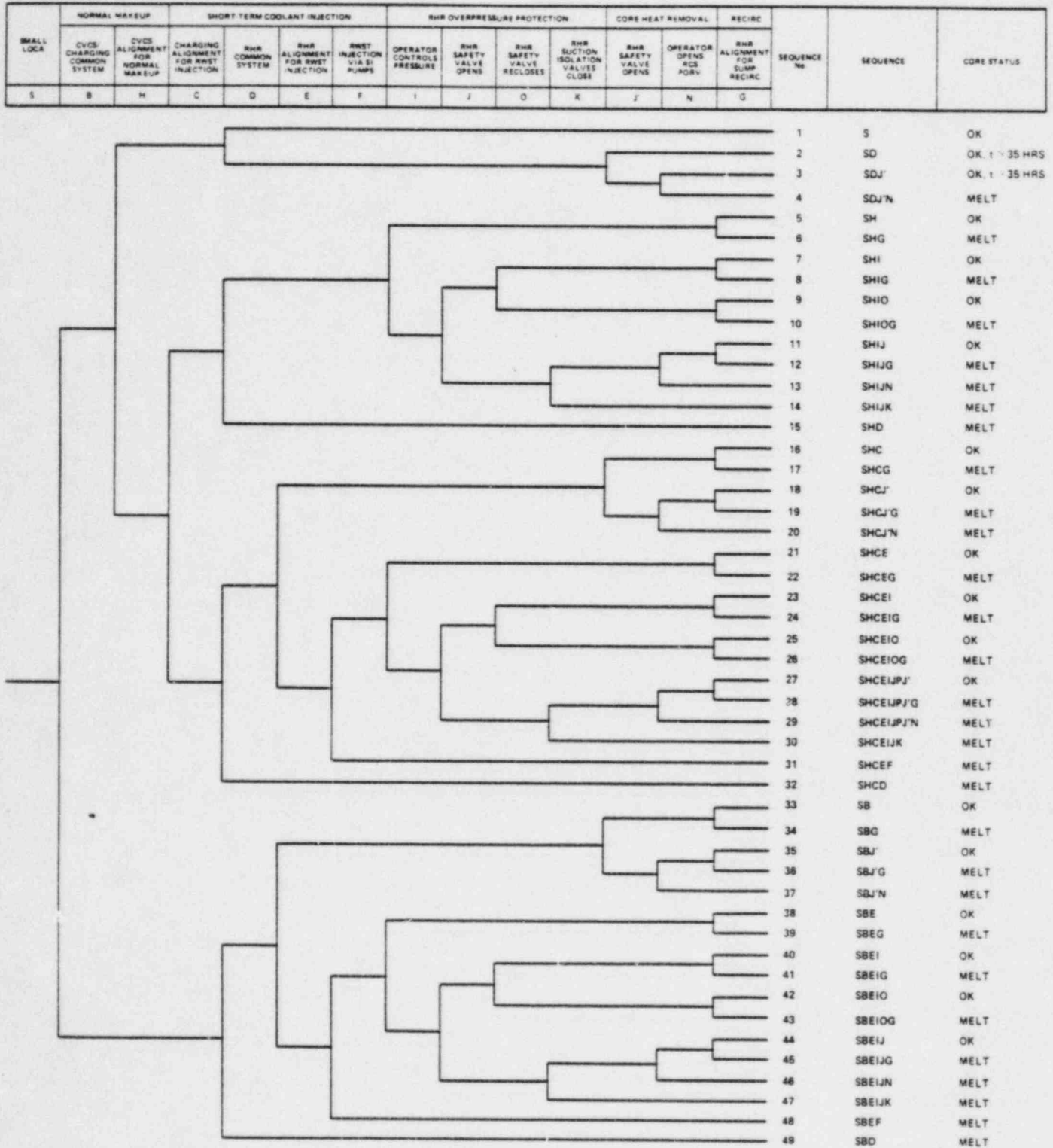


Figure 5-6. Small LOCA Event Tree.



## 6. FAULT TREES FOR RESPONSE TO MODE 5 LOCAs

Simplified fault trees were used to define the equipment failures and operator errors associated with each event in the event trees described in Section 5. The fault trees for each event are included in Appendix A. Assumptions and simplifications made in the construction of these fault trees are listed below:

- Manual valves which need not change position during LOCA response are not modeled.
- Support systems (e.g., electric power CCW and ERCW) are not included in the fault trees. Support system failure probabilities are estimated separately and these results are weighed in the computation of event sequence probabilities (see Section 7).
- The normal CVCS coolant return path to the RCS is not modeled.
- The RHR and HPSI hot leg injection paths are not modeled.
- Normally open valves which must fail closed are not modeled.
- Check valve failure to open is modeled any time flow has been interrupted in a pipe.
- Random pipe or tank rupture is not considered in the fault trees.
- Effects of ventilation system failure on long-term operation of safety-related equipment is not considered.





## 7. QUANTIFICATION OF EVENT TREES

### 7.1 DATA FOR QUANTIFICATION

The data used in quantifying the fault and event trees is listed in Table 7-1. The component, offsite power and operator error data were abstracted from WASH-1400 (Ref. 2). The safe shutdown earthquake (SSE) data was derived from a BNL-NUREG informal report (Ref. 7). Cooling water and electrical system reliability was estimated using WASH-1400 component data and a simplified model of these Sequoyah support systems.

### 7.2 ASSUMPTIONS REGARDING QUANTIFICATION

The following assumptions were made in the quantification of the fault and event trees:

- Plant conditions in cold shutdown do not significantly affect equipment failure probabilities,
- In the event of a safe shutdown earthquake (SSE), a pipe break occurs with a probability of 1.0.
- A normally-closed, motor-operated valve (MOV) outside containment can be treated as a manual valve because of the time available for manual actions and the general availability of manual operating features on MOVs,
- Operator practice at Sequoyah of manually seating safety-related MOVs following valve closure does not affect valve failure-to-open probability,
- Sequoyah CCW and ERCW system failure probability can be estimated based on the simplified system model described in Section 4.2.

These estimated failure probabilities appear in Table 7-1.

- The failure probability for the onsite Class 1E electric power system is dominated by the battery, diesel generator, and diesel ERCW failure probabilities. Failure probability of a Class 1E electrical division can be estimated based on a simplified model which includes only the above elements. Estimated failure probabilities for a Class 1E AC electrical division are included in Table 7-1.
- Operator errors are ranked into two categories (ommission, commision) using engineering judgement as to their likelihood.
- Fault tree truncation is done during the quantification process to eliminate events which do not apply to the sequence being evaluated.
- Event sequences are weighted by power availability probabilities and initiating event probabilities.
- Success probabilities were not included in the event sequence quantification.

### 7.3 DESCRIPTION OF CALCULATIONS

The calculations associated with determining the probability of core melt in cold shutdown began by developing the Boolean algebraic expressions for each of the fault trees shown in Appendix A. These expressions contain all the events that could be involved in the analysis. In order to reflect changes in sequence assumptions, these expressions must be truncated to eliminate those events which are no longer involved due to specific sequence assumptions. For example, the fault tree shown in Figure A-2 for event A in the event trees deals with the securing of the RHR pumps following a large LOCA. This tree is either left intact or is truncated based on the initial assumption regarding the number of operating RHR pumps (e.g., two or one). The external events in the tree are simply flags to indicate this truncation

process. Truncation also occurs due to assumptions dealing with power availability and the occurrence of loss of offsite power.

Support system availability was used as a partitioning function in the solution process. The support system was assumed to be at "full support" (both electrical load groups and cooling water loops available), "half support" (one of two load groups and cooling water loops available), or "no support" (both electrical load groups and cooling water loops unavailable). The fault tree expressions were truncated and solved under each of these assumptions. Fault trees containing pumps were also solved for various lengths of time of pump operation. In this way a table of fault tree quantitative solutions was developed using the data in Table 7-1 for varying assumptions dealing with support system availability and event tree sequences (See Table 7-2).

Using the tabulated fault tree results, each event tree sequence was appropriately solved by combining the system event failure probabilities for each failed system event in the sequence. This was done under the assumptions of full and half support system availability. Once all sequences in an event tree were quantized, the values associated with sequences leading to melt were summed for each support system availability assumption. These values were then weighted by the appropriate probability of being in that particular support system availability situation. These weighted probabilities were summed along with the probability of no support systems available to give the probability of a melt given the initiating event. The probability of the initiating event is then multiplied by the probability of a melt, given the initiating event, to find the resulting probability of core melt.

This process of quantification was done for a number of cases dealing with initiating events, LOCA size, maintenance considerations, and length of loss of offsite power. The computations involved were done by hand and thus were done as simply as possible. Details of the quantification are presented in Section 7.4.

#### 7.4 QUANTIFICATION OF CORE MELT PROBABILITY

Using the data in Table 7-1, the fault trees in Appendix A were quantified under varying assumptions dealing with their usage in the event trees. Table 7-2 gives the quantification results for the individual fault

trees. Note that these values do not include support system failure probabilities. Thus failures of electric power or cooling water are not covered probabilistically in the numerical results in Table 7-2. Event A has been evaluated assuming a common mode failure of the operator rather than independent failures of the operator to secure multiple RHR pumps.

The event trees for large and medium LOCAs were solved for various initiating event assumptions using the intermediate results from Table 7-2. Tables 7-3 to 7-7 present tabular summaries of the individual sequence probabilities for the following cases:

- Large LOCA L<sub>2</sub> initiated by an SSE (Table 7-3),
- Large LOCA L<sub>1</sub> initiated by an SSE (Table 7-4),
- Medium LOCA initiated by an SSE (Table 7-5),
- Large LOCA L<sub>2</sub> initiated by operator error with no loss of offsite power (Table 7-6),
- Large LOCA L<sub>2</sub> initiated by operator error with a one-hour loss of offsite power (Table 7-7).

The small LOCA event tree was not quantified because there may be conditions when this LOCA category does not exist (e.g., when decay heat levels are low, see Section 3).

The tables give conservative event sequence probabilities since they only include multiples of event failure probabilities and do not include event success probabilities. Also, note that each sequence that is quantified is labeled with a "P" or an "M". The P stands for "possible melt", depending on the time between reactor shutdown and LOCA initiation. The M stands for "melt" regardless of time of LOCA initiation. Two values are given for each sequence; a "full support" probability and a "half support" probability. These values indicate the likelihood of an event sequence given either full electric power and component cooling support or only a single load group supplying electric power and component cooling functions. These probabilities do not include support system failure probability but only consider random failures of the system components or operator error included in the fault trees in Appendix A.

The summary values at the bottom of Tables 7-3 to 7-7 indicate the sum of the appropriate individual sequences in the P or M category. The M

summary result is simply the sum of the M labeled sequences. The P summary result is the sum of the P labeled sequences along with all M labeled sequences which do not have event H as part of the sequence from a success or failure standpoint.

Table 7-8 shows the results of the quantification process for 20 different cases when support system availability likelihood and initiating event probabilities are included. These values are found by the following method:

$$P_i(\text{melt}) = P_i(\text{initiating event}) \times \left\{ \begin{aligned} &P_i(\text{full support}) \times P_i(\text{melt}|\text{full support}) + \\ &P_i(\text{half support}) \times P_i(\text{melt}|\text{half support}) + \\ &P_i(\text{no support}) \end{aligned} \right\}$$

where:  $P_i$  is a probability function related to the  $i^{\text{th}}$  case, and is in units of per reactor-year.

$P_i(\text{initiating event})$  is found in Table 7-1 (e.g., an SSE during cold shutdown or an operator error of commission).

$P_i(\text{full support})$ ,  $P_i(\text{half support})$ , and  $P_i(\text{no support})$  are found in Table 7-9.

$P_i(\text{melt}|\text{full support})$  and  $P_i(\text{melt}|\text{half support})$  is found in one of Table 7-3 to 7-7 depending on the case.

To illustrate how the core melt probabilities in Table 7-8 were computed, consider the following example for Case 1, no maintenance. From Table 7-1 we get an initiating event probability for an SSE of  $2.0 \times 10^{-4}$ . From Table 7-3 ( $L_2$  LOCA initiated by an SSE) for the "P" case of no time constraint, we get values of  $1.32 \times 10^{-2}$  and  $2.27 \times 10^{-2}$  for full and half support core melt probabilities, respectively. From Table 7-9 we use the "LOSP one hour-no maintenance" case to get values of 0.91926, 0.07714, and 0.00360 for full, half, and no support system availability likelihoods. Substituting these values into the previous equation yields:

$$\begin{aligned}
P_1(\text{melt}) &= (2.0 \times 10^{-4}) \times \left\{ (.91926) \times (1.32 \times 10^{-2}) + \right. \\
&\quad \left. (.07714) \times (2.27 \times 10^{-2}) + (.00360) \right\} \\
&= (2.0 \times 10^{-4}) \times \left\{ (.01213) + (.00175) + (.00360) \right\} \\
&= (2.0 \times 10^{-4}) \times (.01748) \\
&= 3.50 \times 10^{-6}
\end{aligned}$$

Other cases are computed in a similar fashion. Maintenance case computations used the data in the lower half of Table 7-9 and only used the half support values in Tables 7-3 to 7-7. These cases assume an entire support train is unavailable due to maintenance (e.g., a diesel generator or a component cooling water loop is unavailable because of maintenance). Note that the unavailability of front-line systems due to maintenance is not modeled.

A total of 40 core melt probabilities are listed in Table 7-8 (20 cases, each with and without maintenance on support systems). No attempt has been made to probabilistically combine these cases to get an overall probability of core melt.

Some points of interest dealing with this analysis are the following:

- The probability of overpressurization of the RCS or RHR in a medium LOCA is of low probability ( $3 \times 10^{-9}$  in sequences 10 and 11 from Table 7-5). These sequences do not treat a potential common mode operator error which might increase these sequence probabilities to  $3 \times 10^{-4}$ . If a common mode operator error were assumed, four separate operator actions would be grouped into one event. Such an assumption would increase the medium LOCA probabilities listed in Table 7-5.
- The SSE induced LOCA initiating event probability is only the probability of an SSE and does not include the likelihood of a resulting pipe rupture.
- The operator error induced LOCA initiating event (e.g., due to system misalignment) assumes there is no operator recovery which restores system alignment.

## 7.5 ANALYSIS OF RESULTS

From a review of Table 7-8, the following general observations are made:

- A large LOCA initiated while two RHR pumps are running is the "worst case" LOCA. Lower probabilities of core melt are associated with large LOCAs initiated while one RHR pump is running, and with medium LOCAs.
- Unavailability of one support system train due to maintenance increases core melt probability by as much as an order of magnitude.
- Ability of the CVCS to provide adequate makeup from the normal makeup water source (primary water storage tank plus boration as necessary) is a function of time after shutdown. Core melt probabilities are reduced as much as 50 percent when this normal makeup water source is adequate (e.g., greater than 24 to 35 hours after shutdown).
- Long-term loss of offsite power has a significant impact on core melt probability. If offsite power is assumed to be unavailable for 100 hours, core melt probability is estimated to be more than an order of magnitude higher than the case of a one hour loss of offsite power

## 7.6 TREATMENT OF COMMON MODE FAILURE

Common mode failures of equipment or operator actions were addressed in this analysis. However, the amount of effort spent on this area was minimal. Those obvious areas of common mode failure were handled appropriately as they arose. For example, the electric power system and other component support systems were treated as potential sources of common mode failure of entire trains of systems. In addition, the operator was assumed to have not secured both RHR pumps in appropriate event sequences if he failed to secure one pump. These were the only common mode failures reflected in the results.



Diesel generator common mode failure (unable to start either diesel) was not modeled but this would only impact the results dealing with short losses of offsite power. Sequences dealing with lengthy loss of offsite power are dominated by diesel running failure.

The cases dealing with medium LOCAs and concern about overpressurizing the RCS do not treat the operator as making a common mode failure when attempting to control RCS pressure through one of three different mechanisms. Currently these sequences are of the order  $3 \times 10^{-9}$  in probability. A common mode treatment of operator error in these sequences would increase these sequence probabilities to approximately  $3 \times 10^{-4}$  which would impact the medium LOCA result.

Common mode failure of equipment in the same location or environment was not addressed due to the limitations of the scope of this study.

## 7.7 UNCERTAINTY IN ANALYSIS

No sensitivity analyses or uncertainty analyses were done in this study. Point estimates of component, event, and operator failure probabilities were used with no error band consideration. Conservatism was built into some of the data to reflect uncertainty, however, most of the data used were taken directly from the WASH-1400 data base.

Although no uncertainty analysis was done, the study did suggest those inputs to which the results were most sensitive or which were felt to be very conservative. One area of conservatism arises from the assumption that failure to secure an operating RHR pump during a cold shutdown large LOCA would lead to total failure of the pump with probability 1.0. This is not accurate but an accurate estimate of the probability of failure of an airbound pump was not available. This assumption does impact the results and the probability of core melt would likely decrease if more accurate data were available.

Another conservatism was the assumption that the SSE initiating event leads to a LOCA with probability 1.0. Again, no data dealing with pipe failure rate under cold shutdown SSE conditions were readily available. This probably adds a significant conservatism to the calculated probability of core melt. With accurate data, it is possible that the calculated core melt probability could be reduced by orders of magnitude, depending on the actual likelihood of pipe rupture.

The operator error initiating event assumed that no likelihood exists for terminating a LOCA created by system misalignment. If the operator has the capability to diagnose, identify, and correct the cause of this type of LOCA, the probability of core melt in the related sequences would be reduced.

Operator error in general is handled crudely and may be conservative or not. It is not readily known which way the error in this model would drive the results.

Another area of significance in the results is the assumed failure rate for an operating diesel generator. This rate dominates the SSE sequences for 100 hour loss of offsite power cases. It even begins to dominate for the 10 hour LOSP cases. Any changes in this rate would impact results for those cases mentioned.

Table 7-1. Failure Rate Data for Components.

Component/System	Description*	Failure Type*	Failure Probability
Check valve	Failure to open	D	$1.0 \times 10^{-4}$
Valve outside containment	Closed for long periods	D	$2.0 \times 10^{-4}$
	Closing or closed for short time and opening	D	$1.0 \times 10^{-4}$
Valve inside containment	MOV opening	D	$1.1 \times 10^{-3}$
	MOV closing	D	$1.0 \times 10^{-3}$
	NV opening	D	$4.0 \times 10^{-4}$
	NV closing	D	$3.0 \times 10^{-4}$
Relief valve	Fail to open	D	$3.0 \times 10^{-2}$
	Fail to close	D	$1.0 \times 10^{-2}$
Pump	Fail to start	D	$1.0 \times 10^{-3}$
	Fail to run	O	$3.0 \times 10^{-5}/\text{hr}$
Essential raw cooling water (ERCW) loop	100 hr. operation, pump & check valve in parallel, serving single ERCW loop	-	$1.68 \times 10^{-5}$
Component cooling water (CCW) loop	100 hr. operation, pump & check valve serving single CCW loop	-	$4.10 \times 10^{-3}$
Onsite Class 1E AC power division	Diesel starting & running for specified time, plus battery & ERCW with 2 check valves, 1 MOV		
	1 hour	-	$3.63 \times 10^{-2}$
	10 hours	-	$6.20 \times 10^{-2}$
	100 hours	-	$2.84 \times 10^{-1}$
Offsite power		O	$2.0 \times 10^{-5}/\text{hr}$
Operator error	Commission	-	$3.0 \times 10^{-3}$
	Omission	-	$1.0 \times 10^{-2}$
Instrumentation	Automatic valve closure instrument channel	D	$1.0 \times 10^{-2}$
SSE during cold shutdown	20% time per year in cold shutdown	-	$2.0 \times 10^{-4}$

\* Abbreviations used in this table include the following:

- MOV = motor-operated valve
- NV = pneumatic or hydraulic valve
- SSE = safe shutdown earthquake
- D = demand failure
- O = operating failure

Table 7-2. Fault Tree Quantifications Failure Probability Results.

Event	Special Assumptions	Hours to Failure*	Full Support	Half Support +
A	1 pump at start of event	D	$1.10 \times 10^{-2}$	
	2 pumps at start - both not secured	D	$1.00 \times 10^{-2}\Psi$	
	2 pumps at start - only one secured	D	$2.00 \times 10^{-3}$	
B	No LOSP	D	$5.00 \times 10^{-4}$	$1.05 \times 10^{-3}$
	LOSP	D	$5.02 \times 10^{-4}$	$1.60 \times 10^{-3}$
C		D	$1.01 \times 10^{-2}$	
D	No LOSP	D	$4.76 \times 10^{-5}$	$6.93 \times 10^{-3}$
	LOSP	D	$5.62 \times 10^{-5}$	$7.50 \times 10^{-3}$
E		D	$3.10 \times 10^{-3}$	
F		D	$1.07 \times 10^{-2}$	$1.18 \times 10^{-2}$
		9	$1.07 \times 10^{-2}$	$1.21 \times 10^{-2}$
G	RHR used to inject in sequence	D	$3.10 \times 10^{-3}$	$3.10 \times 10^{-3}$
	RHR did not inject in sequence	D	$3.00 \times 10^{-3}$	$4.00 \times 10^{-3}$
H	CVCS used to inject in sequence	89	$5.45 \times 10^{-4}$	
		90-91	$5.46 \times 10^{-4}$	
		99	$5.48 \times 10^{-4}$	
	CVCS did not inject in sequence	89	$2.45 \times 10^{-4}$	
		90-91	$2.46 \times 10^{-4}$	
		99	$2.48 \times 10^{-4}$	
I		D	$1.00 \times 10^{-4}$	
J		D	$3.00 \times 10^{-3}$	
K		D	$1.01 \times 10^{-4}$	$1.10 \times 10^{-3}$
N		D	$1.00 \times 10^{-2}$	$1.04 \times 10^{-2}$
O		D	$1.00 \times 10^{-2}$	

\* D indicates a failure to start or demand failure

+ Blanks in this column indicate no difference between half & full support

Ψ Common mode failure of operator

Table 7-3. Quantification of Large LOCA L<sub>2</sub> Event Tree (SSE Initiating Event).

LOCA L <sub>2</sub> EVENT SEQUENCE (1)	EVENT STATUS (2)								TIME REQUIREMENTS FOR SYSTEM OPERATION (3)				CONSEQUENCE (4)	SEQUENCE PROBABILITY (5)	
	RHR SECURED	CVCS COMMON	CVCS RWST	RHR COMMON	RHR RWST	SI RWST	RHR RECIRC.	CVCS NORM. M/U	CVCS	RHR	SI	NORM. M/U		FULL SUPPORT	HALF SUPPORT
	A	B	C	D	E	F	G	H							
1	+	+	+	+			+		11	89					
2	+	+	+	+			-	+	100			89	P	3.00 x 10 <sup>-3</sup>	4.00 x 10 <sup>-3</sup>
3	+	+	+	+			-	-	100			<u>89</u>	M	1.64 x 10 <sup>-6</sup>	2.18 x 10 <sup>-6</sup>
4	+	+	+	-				+	100	D		89	P	7.05 x 10 <sup>-5</sup>	8.42 x 10 <sup>-3</sup>
5	+	+	+	-				-	100	D		<u>89</u>	M	3.84 x 10 <sup>-8</sup>	4.59 x 10 <sup>-6</sup>
6	+	+	-	+	+		+			100					
7	+	+	-	+	+		-	+	99	1		99	P	3.13 x 10 <sup>-5</sup>	3.13 x 10 <sup>-5</sup>
8	+	+	-	+	+		-	-	99	1		<u>99</u>	M		
9	+	+	-	+	-	+	+			91	9				
10	+	+	-	+	-	+	-	+	91		9	91	P	9.39 x 10 <sup>-8</sup>	1.25 x 10 <sup>-7</sup>
11	+	+	-	+	-	+	-	-	91		9	<u>91</u>	M	2.31 x 10 <sup>-11</sup>	3.08 x 10 <sup>-11</sup>
12	+	+	-	+	-	-		+	100		D	100	P	3.35 x 10 <sup>-7</sup>	3.69 x 10 <sup>-7</sup>
13	+	+	-	+	-	-		-	100		D	<u>100</u>	M		
14	+	+	-	-				+	100	D		100	P	7.12 x 10 <sup>-7</sup>	8.50 x 10 <sup>-5</sup>
15	+	+	-	-				-	100	D		<u>100</u>	M		
16	+	-		+	+		+		D	100					
17	+	-		+	+		-		D	1			M	1.56 x 10 <sup>-6</sup>	4.96 x 10 <sup>-6</sup>
18	+	-		+	-	+	+		D	91	9				
19	+	-		+	-	+	-		D		9		M	4.67 x 10 <sup>-9</sup>	1.98 x 10 <sup>-8</sup>
20	+	-		+	-	-			D		<u>9</u>		M	1.67 x 10 <sup>-8</sup>	6.00 x 10 <sup>-8</sup>
21	+	-		-					D	D			M	3.54 x 10 <sup>-8</sup>	1.35 x 10 <sup>-5</sup>
22	-	+	+					+	100			89	P	1.00 x 10 <sup>-2</sup>	1.00 x 10 <sup>-2</sup>
23	-	+	+					-	100			<u>89</u>	M	5.45 x 10 <sup>-6</sup>	5.45 x 10 <sup>-6</sup>
24	-	+	-			+		+	100		9	91	P	1.01 x 10 <sup>-4</sup>	1.01 x 10 <sup>-4</sup>
25	-	+	-			+		-	100		9	<u>91</u>	M	2.48 x 10 <sup>-8</sup>	2.48 x 10 <sup>-8</sup>
26	-	+	-			-		+	100		D	100	P	1.08 x 10 <sup>-6</sup>	1.19 x 10 <sup>-6</sup>
27	-	+	-			-		-	100		D	<u>100</u>	M		
28	-	-							D				M	5.04 x 10 <sup>-6</sup>	1.60 x 10 <sup>-5</sup>
													M	4.72 x 10 <sup>-5</sup>	1.65 x 10 <sup>-4</sup>
													P	1.32 x 10 <sup>-2</sup>	2.27 x 10 <sup>-2</sup>

Table 7-3. Quantification of Large LOCA L<sub>2</sub> Event Tree  
(SSE Initiating Event) (Continued).

- Notes:
- (1) Event sequence numbers correspond to the sequence numbers in the event tree.
  - (2) Event status is indicated as "+" (system success), "-" (system failure), or blank (system does not appear in the particular sequence).
  - (3) Listed time requirements, in hours; were used in computing equipment failure probability over a 100 hour interval following LOCA. Underlined times indicate system failure within the specified interval. The code "D" indicates a demand failure of the system in question.
  - (4) Consequences are listed as "M" (core melt), or as "P" (core melt, subject to specified time constraints).
  - (5) Probabilities are per year. "Full Support" probabilities are values assuming no failure occurs in support systems (electric power, component cooling water and essential raw cooling water). "Half Support" probabilities are values which include the probability of failure of one electrical division and/or one cooling water train.

Table 7-4. Quantification of Large LOCA L<sub>1</sub> Event Tree (SSE Initiating Event)\*.

LOCA L <sub>1</sub> EVENT SEQUENCE (1)	EVENT STATUS (2)								TIME REQUIREMENTS FOR SYSTEM OPERATION (3)					CONSEQUENCE (4)	SEQUENCE PROBABILITY (5)	
	RHR SECURED	CVCS COMMON	CVCS RWST	RHR COMMON	RHR RWST	SI RWST	RHR RECIRC.	CVCS NORM. M/U	CVCS	1 RHR	2 RHR	SI	NORM. M/U		FULL SUPPORT	HALF SUPPORT
	A	B	C	D	E	F	G	H								
1	+	+	+	+			+		11		89					
2	+	+	+	+			-	+	100				89	P	3.00 x 10 <sup>-3</sup>	4.00 x 10 <sup>-3</sup>
3	+	+	+	+				-	100				89	M	1.64 x 10 <sup>-6</sup>	2.18 x 10 <sup>-6</sup>
4	+	+	+	-					100		D		89	P	5.62 x 10 <sup>-5</sup>	7.50 x 10 <sup>-3</sup>
5	+	+	+	-					100		D		89	M	3.06 x 10 <sup>-8</sup>	4.09 x 10 <sup>-6</sup>
6	+	+	-	+	+		+				100					
7	+	+	-	+	+		-	+	99		1		99	P	3.13 x 10 <sup>-5</sup>	3.13 x 10 <sup>-5</sup>
8	+	+	-	+	+		-	-	99		1		99	M		
9	+	+	-	+	-	+	+				91	9				
10	+	+	-	+	-	+	-	+	91			9	91	P	9.39 x 10 <sup>-8</sup>	1.25 x 10 <sup>-7</sup>
11	+	+	-	+	-	+	-	-	91			9	91	M	2.31 x 10 <sup>-11</sup>	3.08 x 10 <sup>-11</sup>
12	+	+	-	+	-	-		+	100			D	100	P	3.35 x 10 <sup>-7</sup>	3.69 x 10 <sup>-7</sup>
13	+	+	-	+	-	-		-	100			D	100	M		
14	+	+	-	-				+	100		D		100	P	5.68 x 10 <sup>-7</sup>	7.58 x 10 <sup>-5</sup>
15	+	+	-	-				-	100		D		100	M		
16	+	-		+	+		+		D		100					
17	+	-		+	+		-		D		1			M	1.56 x 10 <sup>-6</sup>	4.96 x 10 <sup>-6</sup>
18	+	-		+	-	+	+		D		91	9				
19	+	-		+	-	+	-		D			9		M	4.67 x 10 <sup>-9</sup>	1.98 x 10 <sup>-8</sup>
20	+	-		+	-	-			D			9		M	1.67 x 10 <sup>-8</sup>	6.00 x 10 <sup>-8</sup>
21	+	-		-					D		D			M	2.82 x 10 <sup>-8</sup>	1.20 x 10 <sup>-5</sup>
22	-	+	+	+			+		11	89						
23	-	+	+	+			-	+	100				89	P	3.30 x 10 <sup>-5</sup>	4.40 x 10 <sup>-5</sup>
24	-	+	+	+			-	-	100				89	M	1.80 x 10 <sup>-8</sup>	2.40 x 10 <sup>-8</sup>
25	-	+	+	-				+	100	D			89	P	8.25 x 10 <sup>-5</sup>	8.25 x 10 <sup>-5</sup>
26	-	+	+	-				-	100	D			89	M	4.50 x 10 <sup>-8</sup>	4.50 x 10 <sup>-8</sup>
27	-	+	-	+	+		+			100						
28	-	+	-	+	+		-	+	99	1			99	P	3.44 x 10 <sup>-7</sup>	3.44 x 10 <sup>-7</sup>
29	-	+	-	+	+		-	-	99	1			99	M		
30	-	+	-	+	-	+	+			91		9				

Table 7-4. Quantification of Large LOCA L<sub>1</sub> Event Tree (SSE Initiating Event) (Continued).

LOCA L <sub>1</sub> EVENT SEQUENCE (1)	EVENT STATUS (2)								TIME REQUIREMENTS FOR SYSTEM OPERATION (3)					CONSEQUENCE (4)	SEQUENCE PROBABILITY (5)	
	RHR SECURED	CVCS COMMON	CVCS RWST	RHR COMMON	RHR RWST	SI RWST	RHR RECIRC.	CVCS NORM. M/U	CVCS	1 RHR	2 RHR	SI	NORM. M/U		FULL SUPPORT	HALF SUPPORT
	A	B	C	D	E	F	G	H								
31	-	+	-	+	-	+	-	+	91			9	91	P	1.03 x 10 <sup>-9</sup>	1.38 x 10 <sup>-9</sup>
32	-	+	-	+	-	+	-	-	91			9	91	M	2.54 x 10 <sup>-13</sup>	3.39 x 10 <sup>-13</sup>
33	-	+	-	+	-	-	-	+	100			D	100	P	3.69 x 10 <sup>-9</sup>	4.06 x 10 <sup>-9</sup>
34	-	+	-	+	-	-	-	-	100			D	100	M		
35	-	+	-	-	-	-	-	+	100	D			100	P	8.33 x 10 <sup>-7</sup>	8.33 x 10 <sup>-7</sup>
36	-	+	-	-	-	-	-	-	100	D			100	M		
37	-	-	-	+	+	-	+	-	D	100						
38	-	-	-	+	+	-	-	-	D	1				M	1.71 x 10 <sup>-8</sup>	5.46 x 10 <sup>-8</sup>
39	-	-	-	+	-	+	+	-	D	91		9				
40	-	-	-	+	-	+	-	-	D			9		M	5.14 x 10 <sup>-11</sup>	2.18 x 10 <sup>-10</sup>
41	-	-	-	+	-	-	-	-	D			9		M	1.83 x 10 <sup>-10</sup>	6.60 x 10 <sup>-10</sup>
42	-	-	-	-	-	-	-	-	D	D				M	4.14 x 10 <sup>-8</sup>	1.32 x 10 <sup>-7</sup>
														M	3.68 x 10 <sup>-5</sup>	1.43 x 10 <sup>-4</sup>
														P	3.21 x 10 <sup>-3</sup>	1.18 x 10 <sup>-2</sup>

\*See Notes, Table 7-3.



Table 7-5. Quantification of Medium LOCA Event Tree (SSE Initiating Event)\*.

LOCA # EVENT SEQUENCE (1)	EVENT STATUS (2)											TIME REQUIREMENTS FOR SYSTEM OPERATION (3)				CONSEQUENCE (4)	SEQUENCE PROBABILITY (5)	
	CVCS COMMON	CVCS RWST	RHR COMMON	RHR RWST	SI RWST	OPER. PRESS.	RHR SW OPEN	RHR ISOL.	OPER. PORV OPEN	RHR RECTIC.	CVCS NORM. M/U	CVCS	RHR	SI	NORM. M/U		FULL SUPPORT	HALF SUPPORT
	B	C	D	E	F	I	J	K	N	O	H							
1	+	+	+			+				+		11	89					
2	+	+	+			+						100			89	P	3.00 x 10 <sup>-3</sup>	
3	+	+	+			+						100			89	M	1.64 x 10 <sup>-6</sup>	
4	+	+	+			-	+			+		11	89					
5	+	+	+			-	+			-	+	100			89	P	3.00 x 10 <sup>-7</sup>	
6	+	+	+			-	+			-	-	100			89	M	1.64 x 10 <sup>-10</sup>	
7	+	+	+			-	-	+	+	+		11	89					
8	+	+	+			-	-	+	+	-	+	100			89	P	9.00 x 10 <sup>-10</sup>	
9	+	+	+			-	-	+	+	-	-	100			89	M	4.92 x 10 <sup>-13</sup>	
10	+	+	+			-	-	+	-			11				M	3.00 x 10 <sup>-9</sup>	
11	+	+	+			-	-	-				11				M	3.03 x 10 <sup>-11</sup>	
12	+	+	-			+				+		100	D		89	P	5.62 x 10 <sup>-5</sup>	
13	+	+	-			+				-		100	D		89	M	3.06 x 10 <sup>-8</sup>	
14	+	+	-			-	+			+		100	D		89	P	5.62 x 10 <sup>-9</sup>	
15	+	+	-			-	+			-		100	D		89	M	3.06 x 10 <sup>-12</sup>	
16	+	+	-			-	-	+	+	+		100	D		89	P	1.68 x 10 <sup>-12</sup>	
17	+	+	-			-	-	+	+	-		100	D		89	M	9.18 x 10 <sup>-15</sup>	
18	+	+	-			-	-	+	-			11	D			M	1.69 x 10 <sup>-13</sup>	
19	+	+	-			-	-	-				11	D			M	1.70 x 10 <sup>-15</sup>	
20	+	-	+	+						+			100					
21	+	-	+	+						-	+	99	1		99	P	3.13 x 10 <sup>-5</sup>	
22	+	-	+	+						-	-	99	1		99	M		3.13 x 10 <sup>-5</sup>
23	+	-	+	-	+	+				+			91	9				
24	+	-	+	-	+	+				-	+	91		9	91	P	9.39 x 10 <sup>-8</sup>	
25	+	-	+	-	+	+				-	-	91		9	91	M	2.31 x 10 <sup>-11</sup>	
26	+	-	+	-	+	-	+			+			91	9				
27	+	-	+	-	+	-	+			-	+	91		9	91	P	9.39 x 10 <sup>-12</sup>	
28	+	-	+	-	+	-	+			-	-	91		9	91	M	2.31 x 10 <sup>-15</sup>	
29	+	-	+	-	+	-	-	+	+	+			91	9				
30	+	-	+	-	+	-	-	+	+	-	+	91		9	91	P	2.82 x 10 <sup>-14</sup>	

Table 7-5. Quantification of Medium LOCA Event Tree  
(SSE Initiating Event) (Continued).

LOCA #: EVENT SEQUENCE (1)	EVENT STATUS (2)											TIME REQUIREMENTS (3) FOR SYSTEM OPERATION				CONSEQUENCE (4)	SEQUENCE PROBABILITY (5)	
	CVCS COMMON	CVCS RWST	RHR COMMON	RHR RWST	SI RWST	OPER. PRESS.	RHR SV OPEN	RHR ISOL.	OPER. PORV OPEN	RHR RECIRC.	CVCS NORM. M/U	CVCS	RHR	SI	NORM. M/U		FULL SUPPORT	HALF SUPPORT
	B	C	D	E	F	I	J	K	N	G	H							
31	+	-	+	-	+	-	-	+	+	-	-	91		9	91	M	$6.93 \times 10^{-18}$	$9.24 \times 10^{-18}$
32	+	-	+	-	+	-	-	+	-	-	-			9		M	$9.39 \times 10^{-14}$	$9.77 \times 10^{-14}$
33	+	-	+	-	+	-	-	-	-	-	-			9		M	$9.49 \times 10^{-16}$	$1.03 \times 10^{-14}$
34	+	-	+	-	-	-	-	-	-	-	+	100		D	100	P	$3.35 \times 10^{-7}$	$3.69 \times 10^{-7}$
35	+	-	+	-	-	-	-	-	-	-	-	100		D	100	M		
36	+	-	-	-	-	-	-	-	-	-	+	100	D		100	P	$5.68 \times 10^{-7}$	$7.58 \times 10^{-5}$
37	+	-	-	-	-	-	-	-	-	-	-	100	D		100	M		
38	-	-	+	+	-	-	-	-	-	-	+	D	100					
39	-	-	+	+	-	-	-	-	-	-	-	D	1			M	$1.56 \times 10^{-6}$	$4.96 \times 10^{-6}$
40	-	-	+	-	+	+	-	-	-	-	+	D	91	9				
41	-	-	+	-	+	+	-	-	-	-	-	D		9		M	$4.67 \times 10^{-9}$	$1.98 \times 10^{-8}$
42	-	-	+	-	+	-	+	-	-	-	+	D	91	9				
43	-	-	+	-	+	-	+	-	-	-	-	D		9		M	$4.67 \times 10^{-13}$	$1.98 \times 10^{-12}$
44	-	-	+	-	+	-	-	+	+	+	+	D	91	9				
45	-	-	+	-	+	-	-	+	+	-	-	D		9		M	$1.40 \times 10^{-15}$	$5.94 \times 10^{-15}$
46	-	-	+	-	+	-	-	+	-	-	-	D		9		M	$4.66 \times 10^{-15}$	$1.55 \times 10^{-14}$
47	-	-	+	-	+	-	-	-	-	-	-	D		9		M	$4.72 \times 10^{-17}$	$1.64 \times 10^{-15}$
48	-	-	+	-	-	-	-	-	-	-	-	D		<u>9</u>		M	$1.67 \times 10^{-8}$	$6.00 \times 10^{-8}$
49	-	-	-	-	-	-	-	-	-	-	-	D	D			M	$2.82 \times 10^{-8}$	$1.20 \times 10^{-5}$
																M	$3.55 \times 10^{-5}$	$1.31 \times 10^{-4}$
																P	$3.09 \times 10^{-3}$	$1.16 \times 10^{-2}$

\*See Notes, Table 7-3.

Table 7-6. Quantification of Large LOCA L<sub>2</sub> Event Tree (Operator Error Initiating Event With No LOSP)\*.

LOCA L <sub>2</sub> EVENT SEQUENCE (1)	EVENT STATUS (2)								TIME REQUIREMENTS FOR SYSTEM OPERATION (3)				CONSEQUENCE (4)	SEQUENCE PROBABILITY (5)	
	RHR SECURED	CVCS COMMON	CVCS RWST	RHR COMMON	RHR RWST	SI RWST	RHR RECIRC.	CVCS NORM. M/U	CVCS	RHR	SI	NORM. M/U		NO LOSP, FULL SUPPORT	NO LOSP, HALF SUPPORT
	A	B	C	D	E	F	G	H							
1	+	+	+	+			+		11	89					
2	+	+	+	+				+	100			89	P	3.00 x 10 <sup>-3</sup>	4.00 x 10 <sup>-3</sup>
3	+	+	+	+					100			89	M	1.64 x 10 <sup>-6</sup>	2.18 x 10 <sup>-6</sup>
4	+	+	+	-				+	100	D		89	P	4.76 x 10 <sup>-5</sup>	6.93 x 10 <sup>-3</sup>
5	+	+	+	-				-	100	D		89	M	2.59 x 10 <sup>-8</sup>	3.78 x 10 <sup>-6</sup>
6	+	+	-	+	+			+		100					
7	+	+	-	+	+			-	99	1		99	P	3.13 x 10 <sup>-5</sup>	3.13 x 10 <sup>-5</sup>
8	+	+	-	+	+			-	99	1		99	M	7.76 x 10 <sup>-9</sup>	7.76 x 10 <sup>-9</sup>
9	+	+	-	+	-	+	+			91	9				
10	+	+	-	+	-	+	-	+	91		9	91	P	9.39 x 10 <sup>-8</sup>	1.25 x 10 <sup>-7</sup>
11	+	+	-	+	-	+	-	-	91		9	91	M	2.31 x 10 <sup>-11</sup>	3.08 x 10 <sup>-11</sup>
12	+	+	-	+	-	-		+	100	D	100		P	3.35 x 10 <sup>-7</sup>	3.69 x 10 <sup>-7</sup>
13	+	+	-	+	-	-		-	100	D	100		M	8.31 x 10 <sup>-11</sup>	9.15 x 10 <sup>-11</sup>
14	+	+	-	-				+	100	D		100	P	4.81 x 10 <sup>-7</sup>	7.00 x 10 <sup>-5</sup>
15	+	+	-	-				-	100	D		100	M	1.19 x 10 <sup>-10</sup>	1.74 x 10 <sup>-8</sup>
16	+	-		+	+			+	D	100					
17	+	-		+	+			-	D	1			M	1.55 x 10 <sup>-6</sup>	3.26 x 10 <sup>-6</sup>
18	+	-		+	-	+	+		D	91	9				
19	+	-		+	-	+	-		D		9		M	4.65 x 10 <sup>-9</sup>	1.30 x 10 <sup>-8</sup>
20	+	-		+	-	-			D		9		M	1.66 x 10 <sup>-8</sup>	3.94 x 10 <sup>-8</sup>
21	+	-		-					D	D			M	2.38 x 10 <sup>-8</sup>	7.28 x 10 <sup>-6</sup>
22	-	+	+					+	100			89	P+		
23	-	+	+					-	100			89	M o		
24	-	+	-			+	+		100		9	91	P+		
25	-	+	-			+	-		100		9	91	M o		
26	-	+	-			-	+		100	D	100		P+	1.00 x 10 <sup>-2</sup>	1.00 x 10 <sup>-2</sup>
27	-	+	-			-	-		100	D	100		M o	2.48 x 10 <sup>-6</sup>	2.48 x 10 <sup>-6</sup>
28	-	-							D				M	5.00 x 10 <sup>-6</sup>	1.05 x 10 <sup>-5</sup>
													M	1.07 x 10 <sup>-5</sup>	2.96 x 10 <sup>-5</sup>
													P	1.31 x 10 <sup>-2</sup>	2.10 x 10 <sup>-2</sup>

\*See Notes, Table 7-3.

+Combined due to initiating event impact.

oCombined due to initiating event impact.

Table 7-7. Quantification of Large LOCA L<sub>2</sub> Event Tree (Operator Error Initiating Event Plus 1 Hour LOSP).

LOCA L <sub>2</sub> EVENT SEQUENCE (1)	EVENT STATUS (2)								TIME REQUIREMENTS FOR SYSTEM OPERATION (3)				CONSEQUENCE (4)	SEQUENCE PROBABILITY (5)	
	RHR SECURED	CVCS COMMON	CVCS RWST	RHR COMMON	RHR RWST	SI RWST	RHR RECTRC.	CVCS NORM. M/U	CVCS	RHR	SI	NORM. M/U		1 HR. LOSP FULL SUPPORT	1 HR. LOSP HALF SUPPORT
	A	B	C	D	E	F	G	H							
1	+	+	+	+			+		11	89					
2	+	+	+	+			-	+	100			89	P	3.00 x 10 <sup>-3</sup>	4.00 x 10 <sup>-3</sup>
3	+	+	+	+			-	-	100			89	M	1.64 x 10 <sup>-6</sup>	2.18 x 10 <sup>-6</sup>
4	+	+	+	-				+	100	D		89	P	7.05 x 10 <sup>-5</sup>	8.42 x 10 <sup>-3</sup>
5	+	+	+	-				-	100	D		89	M	3.84 x 10 <sup>-8</sup>	4.59 x 10 <sup>-6</sup>
6	+	+	-	+	+		+			100					
7	+	+	-	+	+		-	+	99	1		99	P	3.13 x 10 <sup>-5</sup>	3.13 x 10 <sup>-5</sup>
8	+	+	-	+	+		-	-	99	1		99	M	7.76 x 10 <sup>-9</sup>	7.76 x 10 <sup>-9</sup>
9	+	+	-	+	-	+	+			91	9				
10	+	+	-	+	-	+	-	+	91		9	91	P	9.39 x 10 <sup>-8</sup>	1.25 x 10 <sup>-7</sup>
11	+	+	-	+	-	+	-	-	91		9	91	M	2.31 x 10 <sup>-11</sup>	3.08 x 10 <sup>-11</sup>
12	+	+	-	+	-	-		+	100	D	100		P	3.35 x 10 <sup>-7</sup>	3.69 x 10 <sup>-7</sup>
13	+	+	-	+	-	-		-	100	D	100		M	8.31 x 10 <sup>-11</sup>	9.15 x 10 <sup>-11</sup>
14	+	+	-	-				+	100	D		100	P	7.12 x 10 <sup>-7</sup>	8.50 x 10 <sup>-5</sup>
15	+	+	-	-				-	100	D		100	M	1.77 x 10 <sup>-10</sup>	2.11 x 10 <sup>-8</sup>
16	+	-		+	+		+		D	100					
17	+	-		+	+		-		D	1			M	1.56 x 10 <sup>-6</sup>	4.96 x 10 <sup>-6</sup>
18	+	-		+	-	+	+		D	91	9				
19	+	-		+	-	+	-		D		9		M	4.67 x 10 <sup>-9</sup>	1.98 x 10 <sup>-8</sup>
20	+	-		+	-	-			D		9		M	1.67 x 10 <sup>-8</sup>	6.00 x 10 <sup>-8</sup>
21	+	-		-					D	D			M	3.54 x 10 <sup>-8</sup>	1.35 x 10 <sup>-5</sup>
22	-	+	+					+	100			89	P+		
23	-	+	+					-	100			89	M o		
24	-	+	-			+		+	100		9	91	P+		
25	-	+	-			+		-	100		9	91	M o		
26	-	+	-			-		+	100	D	100		P+	1.00 x 10 <sup>-2</sup>	1.00 x 10 <sup>-2</sup>
27	-	+	-			-		-	100	D	100		M o	2.48 x 10 <sup>-6</sup>	2.48 x 10 <sup>-6</sup>
28	-	-							D				M	5.04 x 10 <sup>-6</sup>	1.60 x 10 <sup>-5</sup>
													M	1.08 x 10 <sup>-5</sup>	4.36 x 10 <sup>-5</sup>
													P	1.31 x 10 <sup>-2</sup>	2.26 x 10 <sup>-2</sup>

\*See Notes, Table 7-3.

+Combined due to initiating event impact.

oCombined due to initiating event impact.

Table 7-8. Summary of Calculations to Estimate Core Melt Probability During a 100 Hour Period Following a LOCA Initiated in Cold Shutdown (Mode 5).

Case	LOCA Initiating Event (1)	Added Time (2)		Loss of Offsite Power				LOCA Type (3)		Estimated Probability Per Reactor-Year of Core Melt in Mode 5	
		Yes	No	1 hr.	10 hr.	100 hr.	L <sub>2</sub>	L <sub>1</sub>	M	No Maintenance (4)	With Maintenance (5)
1	/	/	/	/	/	/	/	/	/	3.50 x 10 <sup>-6</sup>	1.28 x 10 <sup>-5</sup>
2	/	/	/	/	/	/	/	/	/	1.49 x 10 <sup>-6</sup>	1.07 x 10 <sup>-5</sup>
3	/	/	/	/	/	/	/	/	/	1.47 x 10 <sup>-6</sup>	1.07 x 10 <sup>-5</sup>
4	/	/	/	/	/	/	/	/	/	4.08 x 10 <sup>-6</sup>	1.77 x 10 <sup>-5</sup>
5	/	/	/	/	/	/	/	/	/	2.08 x 10 <sup>-6</sup>	1.57 x 10 <sup>-5</sup>
6	/	/	/	/	/	/	/	/	/	2.05 x 10 <sup>-6</sup>	1.57 x 10 <sup>-5</sup>
7	/	/	/	/	/	/	/	/	/	1.96 x 10 <sup>-5</sup>	6.06 x 10 <sup>-5</sup>
8	/	/	/	/	/	/	/	/	/	1.78 x 10 <sup>-5</sup>	5.91 x 10 <sup>-5</sup>
9	/	/	/	/	/	/	/	/	/	1.77 x 10 <sup>-5</sup>	5.91 x 10 <sup>-5</sup>
10	/	/	/	/	/	/	/	/	/	7.31 x 10 <sup>-7</sup>	8.47 x 10 <sup>-6</sup>
11	/	/	/	/	/	/	/	/	/	7.29 x 10 <sup>-7</sup>	8.46 x 10 <sup>-6</sup>
12	/	/	/	/	/	/	/	/	/	7.29 x 10 <sup>-7</sup>	8.46 x 10 <sup>-6</sup>
13	/	/	/	/	/	/	/	/	/	1.24 x 10 <sup>-6</sup>	1.35 x 10 <sup>-5</sup>
14	/	/	/	/	/	/	/	/	/	1.24 x 10 <sup>-6</sup>	1.35 x 10 <sup>-5</sup>
15	/	/	/	/	/	/	/	/	/	1.24 x 10 <sup>-6</sup>	1.35 x 10 <sup>-5</sup>
16	/	/	/	/	/	/	/	/	/	1.65 x 10 <sup>-5</sup>	5.74 x 10 <sup>-5</sup>
17	/	/	/	/	/	/	/	/	/	1.65 x 10 <sup>-5</sup>	5.74 x 10 <sup>-5</sup>
18	/	/	/	/	/	/	/	/	/	1.65 x 10 <sup>-5</sup>	5.74 x 10 <sup>-5</sup>
19	/	/	/	/	/	/	/	/	/	3.96 x 10 <sup>-5</sup>	7.53 x 10 <sup>-5</sup>
20	/	/	/	/	/	/	/	/	/	1.14 x 10 <sup>-7</sup>	1.27 x 10 <sup>-5</sup>

Table 7-8. Summary of Calculations to Estimate Core Melt Probability During a 100 Hour Period Following a LOCA Initiated in Cold Shutdown (Mode 5) (Continued).

- Notes:
- (1) Calculations were performed for LOCAs initiated by a safe shutdown earthquake (SSE) or by operator misalignment of valves capable of diverting water from the reactor coolant system during Mode 5 operations.
  - (2) Some event sequences were potential success paths only after a specified time in Mode 5 following shutdown. These paths were nonsuccess paths during the initial period of Mode 5 operation. When the added time constraint was considered (e.g., time after shutdown greater than limiting time constraint), core melt probabilities were slightly reduced by the availability of an additional effective makeup water source.
  - (3)  $L_2$  = large LOCA with two RHR pumps initially operating.  
 $L_1$  = large LOCA with one RHR pump initially operating.  
M = medium LOCA.
  - (4) The numerical results listed in this column apply when no equipment is out of service for maintenance. Only random failures affect the availability of support systems.
  - (5) The numerical results listed in this column apply when one support system train is unavailable because of maintenance (e.g., one diesel generator being overhauled, one CCW heat exchanger being inspected). Restoration of the unavailable support system train following the LOCA is not considered.

Table 7-9. Probability of Support System Availability.

Assumptions	Full Support	Half Support	No Support
No LOSP - No Maintenance	.99178	.00820	.00002
LOSP 1 hr - No Maintenance	.91926	.07714	.00360**
LOSP 10 hr - No Maintenance	.87104	.12283	.00613**
LOSP 100 hr - No Maintenance	.50844	.40922	.08234
No LOSP - Maintenance*	--	.99588	.00412
LOSP 1 hr - Maintenance*	--	.95783	.04217
LOSP 10 hr - Maintenance*	--	.93246	.06754
LOSP 100 hr - Maintenance*	--	.71305	.28695

\* Maintenance affecting one support system train is assumed to be in progress (e.g., one diesel generator being overhauled, one CCW heat exchanger being inspected). The affected support system train is assumed to remain unavailable throughout the 100 hour post-LOCA period.

\*\*Conservative values due to combination of probability of 2nd loss of offsite power with the no support probability.

## 8. SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

### 8.1 SUMMARY AND CONCLUSIONS

During cold shutdown, there are few automatic actions associated with response to a LOCA. The operator must, therefore, take a more active role in LOCA detection and mitigation than during power operation. There is considerable time available, however, for the operator to take required protective actions.

The estimated probability of core melt following a postulated LOCA during cold shutdown was evaluated for 20 different cases with varying assumptions regarding the LOCA initiating event (safe shutdown earthquake or operator error), time of LOCA initiation following reactor shutdown, LOCA size, availability of offsite power during a 100-hour period following the LOCA and maintenance status. With no maintenance in progress, the probability of core melt for those 20 cases was estimated to be in the range of  $3.96 \times 10^{-5}$  to  $1.14 \times 10^{-7}$  per reactor-year. If maintenance affecting one support system train is in progress (e.g., one diesel generator or component cooling water loop is out of service) the probability of core melt for the 20 cases increases and is estimated to be in the range from  $7.53 \times 10^{-5}$  to  $8.46 \times 10^{-6}$  per reactor-year. In contrast, an overall core melt probability of  $6 \times 10^{-5}$  was reported in WASH-1400 (Ref. 2). The estimates of core melt probability following a postulated LOCA during cold shutdown are believed to be very conservative and should be interpreted as upper limits, however, these values are nontrivial in comparison to the WASH-1400 results.

The large LOCA was the "worst case" LOCA identified in this study. Major contributors to core melt probability include the following:

- Initiating event probability
- Probability of operator error during response
- Probability of failure of an airbound RHR pump
- Diesel generator reliability during long-term unavailability of offsite power



Of these, only operator error during response may have been treated in a nonconservative manner. The large conservatism in the treatment of the other items, particularly initiating event probabilities, tend to suggest that the overall results are very conservative. A more detailed analysis would most likely indicate that the probability of core melt in Mode 5 is significantly lower, perhaps by an order of magnitude or more.

## 8.2 RECOMMENDATIONS

Based on this preliminary assessment of core melt probability in cold shutdown following a postulated LOCA, several operating practices and design changes are recommended for consideration as potential means for reducing core melt probability.

### 8.2.1 Operating Practices

- Secure Operating RHR Pumps  
Operating RHR pumps should be immediately secured following a LOCA in which pressurizer level indication is lost. This action is intended to minimize the probability of RHR pump failure due to cavitation or airbinding. This operator action is not unlike the current practice of manually securing any operating main coolant pumps when required RCS temperature and pressure conditions cannot be satisfied.
  
- Align Centrifugal Charging Pump Suction to RWST  
The operating centrifugal charging pump suction should be realigned to the RWST to provide necessary reactor coolant makeup. Under the assumptions made in the analysis, there is a time dependency associated with obtaining adequate makeup from the normal makeup water source (primary water storage tank, with boration as necessary). There is no time dependency associated with the RWST, and it supplies highly borated water, therefore, the RWST should be the primary water source for LOCA response.

- **Align Centrifugal Charging Pump Discharge to ECCS Injection Path**  
The normal CVCS return path to the RCS is a single, or at most, a dual path. A LOCA in RCS loops 1 or 3 could divert a large portion of the makeup flow (as much as 100 percent). By aligning the centrifugal charging pump to the ECCS injection path, four parallel injection paths to the RCS are provided. A LOCA would be expected to affect only one of these paths, thus a maximum of 25 percent of the makeup water would be diverted from the reactor vessel.
  
- **Use Only a Single High-Head Pump For Coolant Makeup**  
The large and medium LOCA definitions in Table 3-2 are based on the flowrate of a single high-head makeup pump at 600 psig (RHR system design pressure). Use of more than one high-head pump for makeup is not necessary for adequate core cooling. In addition, the range of break sizes falling into the medium LOCA category will increase if more than one high-head pump is providing makeup. It is likely that the probability of overpressurizing the RCS will increase.
  
- **Use Only Makeup Water That is Adequately Borated**  
Boron dilution accidents were not considered in this analysis, however, such accidents may be credible if extended use is made of unborated water (e.g., fire water, condensate, etc.).
  
- **Restore Containment Isolation**  
If an equipment hatch is open, it may take a considerable period of time to restore containment isolation. It is therefore important that efforts to restore containment isolation be initiated in a timely manner to limit potential offsite radiological consequences. A preferred operating practice would be to have the equipment hatch open only when required to move equipment that cannot pass through the more easily isolated personnel airlocks. This operating practice is implemented by several utilities, and should aid in minimizing the probability that the containment could not be reisolated following a LOCA in Mode 5.

### 8.2.2 Design Changes

- **RHR Pumps Designed For Airbound Operation**

In Reference 7, it was noted that RHR pumps are not currently required to be designed for operation under no-suction (e.g., airbound) conditions. It was further noted in Reference 7 that it would be desirable to design RHR pumps "so that they will not fail because of airbinding, with this ability verified by tests." With this design capability, it would not be necessary to initiate a manual protective shutdown of the RHR pumps following a LOCA in which pressurizer level indication is lost. The large LOCA event trees in Figures 5-1 and 5-4 would be simplified by the deletion of Event A.
  
- **Reactor Vessel Wide-Range Level Indication**

A wide-range reactor vessel level indication capability would provide useful information to the operator and would aid in making optimum use of available water resources, including the cold leg and UHI accumulators. It is possible that this monitoring capability would allow the LOCA source to be identified, or at least localized to one RCS loop. This capability would be particularly important if the RHR system could not be aligned for RHR sump recirculation, and core cooling had to be provided by the CVCS in a feed-only mode. This type of level instrument could be isolated during normal reactor operation if necessary, and only placed in service at some suitable time during the RCS cooldown sequence.
  
- **Redundant RHR Suction Line Safety Valves**

The Sequoyah RHR system has a single RHR suction line with a single safety valve to protect the RHR system suction piping against overpressurization. Redundant suction safety valves could reduce the failure probability of Event J, and the subsequent challenge of the RHR suction isolation valves and the pressurizer PORVs for RHR and RCS overpressure protection.

- Power-Operated Relief Valve Variable Setpoint Capability

When the RHR suction line isolation valves are shut in Mode 5, the RCS is left without an overpressure protection system that is designed to limit RCS pressure rise commensurate with the applicable temperature-pressure limits (see Figures 3-6 and 3-7). Main coolant pump seal leakage and any coolant letdown via the CVCS high pressure letdown path may aid in limiting RCS pressure under this condition, however, their effect is uncertain.

One approach for providing additional RCS overpressure protection in Mode 5 is to add a dual-setpoint capability to the control circuitry for the pressurizer power-operated relief valves (PORVs). During power operation, these valves would be set to lift at their normal setpoint of 2350 psig. At some suitable point in the RCS cooldown sequence, the higher-pressure setpoint of the PORV control circuit could be bypassed and the lower-pressure setpoint activated. The normal higher-pressure setpoint would be restored during the subsequent startup sequence. This design feature could reduce the failure probability of Event N in the medium and small LOCA trees by adding an automatic PORV actuation capability in parallel with the existing remote-manual capability.



## 9. REFERENCES

1. NUREG-0789, "Technical Specifications, Sequoyah Nuclear Plant, Unit 2, Docket 50-328," U.S. Nuclear Regulatory Commission, September 1981.
2. WASH-1400, "Reactor Safety Study," Appendix III, U.S. Nuclear Regulatory Commission, October 1975.
3. B-SAR-205, "Babcock & Wilcox Standard Nuclear Steam Supply System," Docket STN-50-561, Babcock & Wilcox Company.
4. Sequoyah Nuclear Plant FSAR, Dockets 50-327, 50-328.
5. San Onofre Nuclear Generating Station, Units 2 and 3 FSAR, Dockets 50-361, 50-362.
6. NRC Inspection and Enforcement Report No. 50-327/81-07, "Containment Spray Event of February 11, 1981, TVA Sequoyah Nuclear Plant."
7. Baslik, A. J. and Bari, R.A., "Risk Reduction From Safety-Grade Means of Reaching and Maintaining Cold Shutdown," BNL-NUREG Informal Report, Brookhaven National Laboratory, January 1981.



## APPENDIX A - FAULT TREES

This appendix contains the fault trees associated with the large, medium and small LOCA event trees described in Section 5. Basic fault tree symbols are shown in Figure A-1. Fault trees for the following events are included in the remainder of this appendix:

- Event A: Failure to secure operating RHR pumps (Figure A-2)
- Event B: Failure of 2-of-2 CVCS pump common (Figure A-3)
- Event C: Failure of CVCS alignment to RWST (Figure A-4)
- Event D: Failure of 2-of-2 RHR pump common (Figure A-5)
- Event E: Failure of RHR alignment to RWST (Figure A-6)
- Event F: Failure of 2-of-2 SI pump injection (Figure A-7)
- Event G: Failure of RHR alignment to recirculate (Figure A-8)
- Event H: Failure of CVCS normal makeup (Figure A-9)
- Event I: Failure of operator to control pressure (Figure A-10)
- Event J: Failure of RHR relief valve to lift (Figure A-10)
- Event K: Failure of RHR isolation alignment (Figure A-11)
- Event N: Failure of operator PORV actuation (Figure A-12)
- Event O: Failure of RHR relief valve to close (Figure A-12)



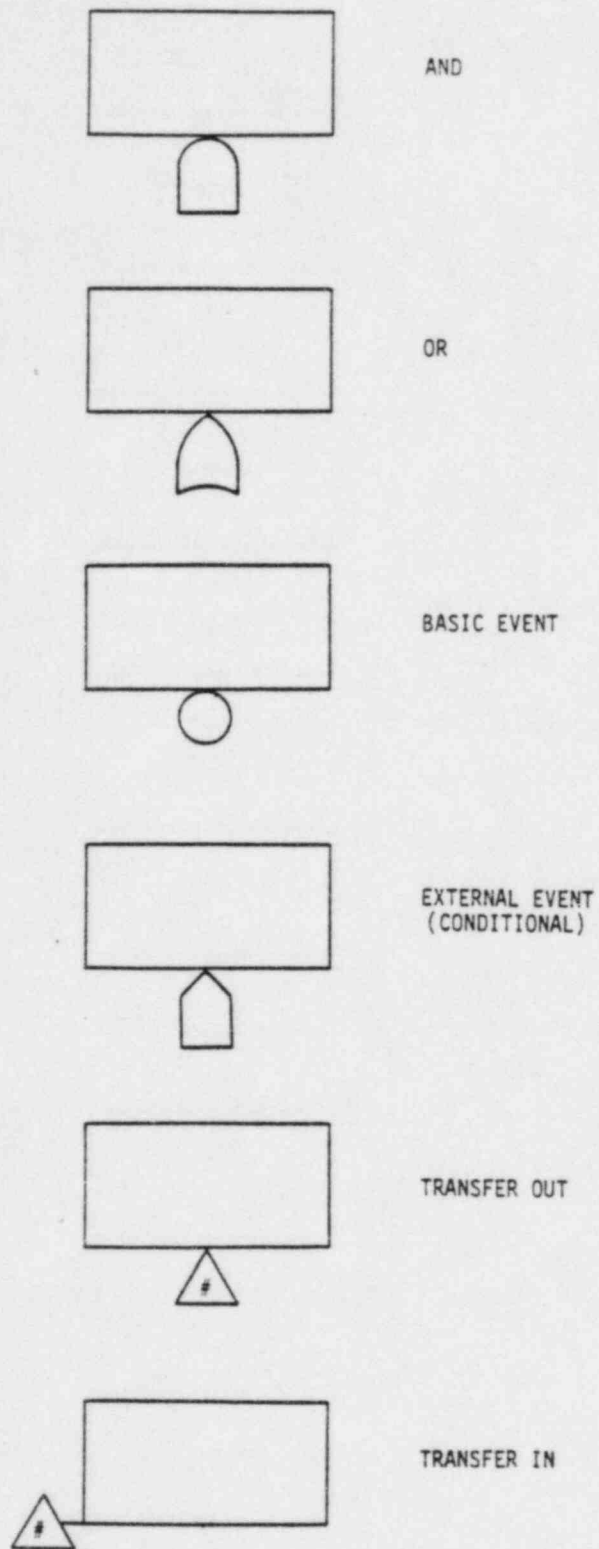


Figure A-1. Basic Fault Tree Symbols

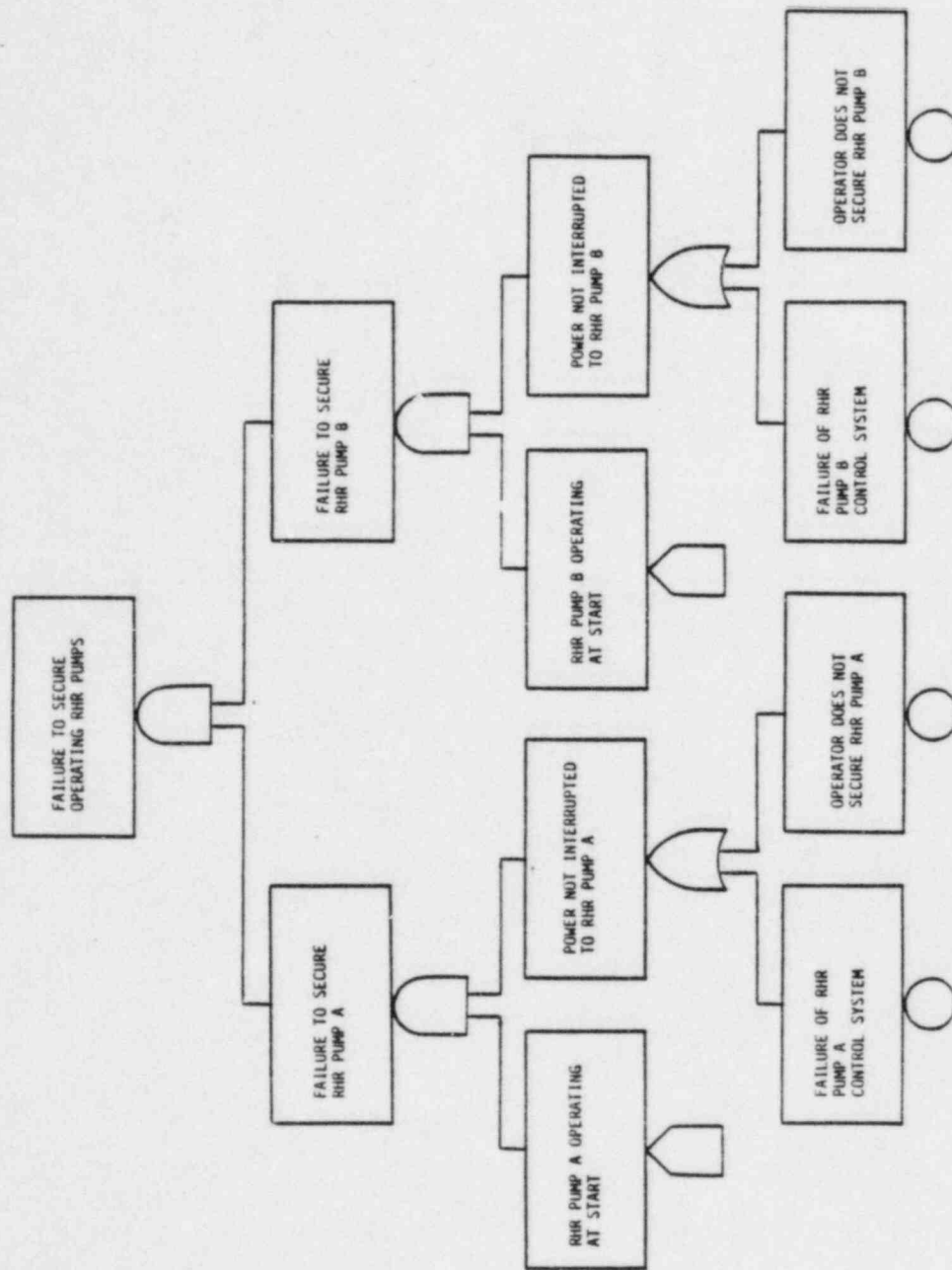
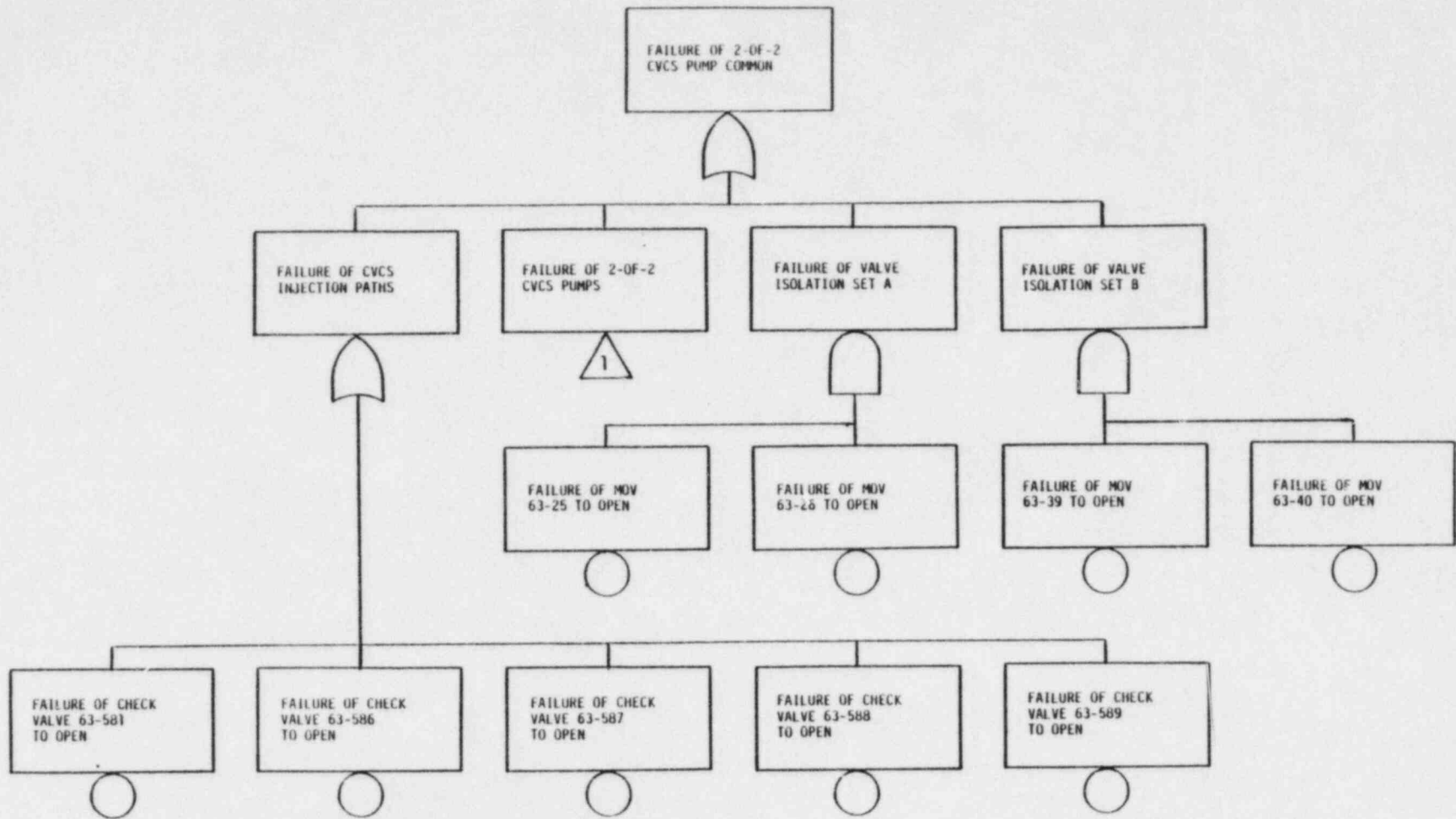


Figure A-2. Fault Tree for Event A.



A4

Figure A-3. Fault Tree for Event B.

A5

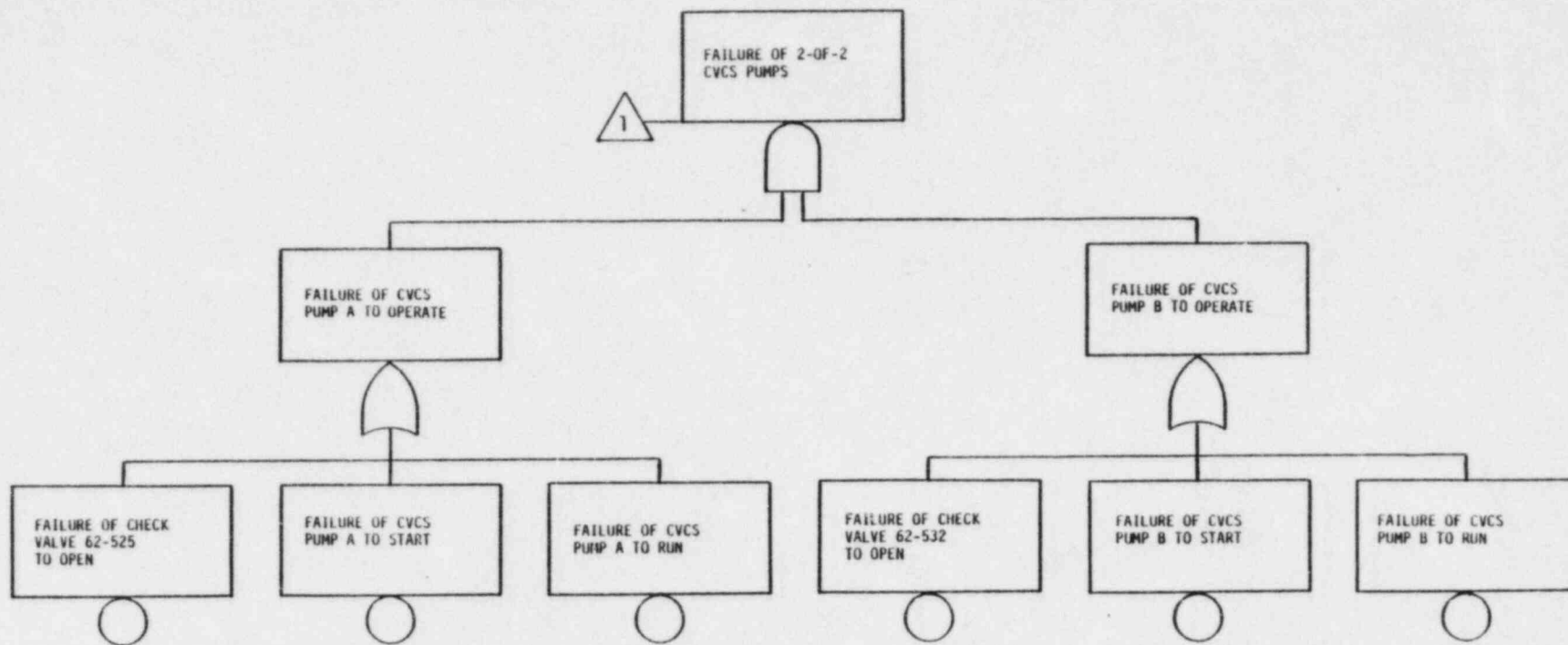


Figure A-3. Fault Tree for Event B (Continued).

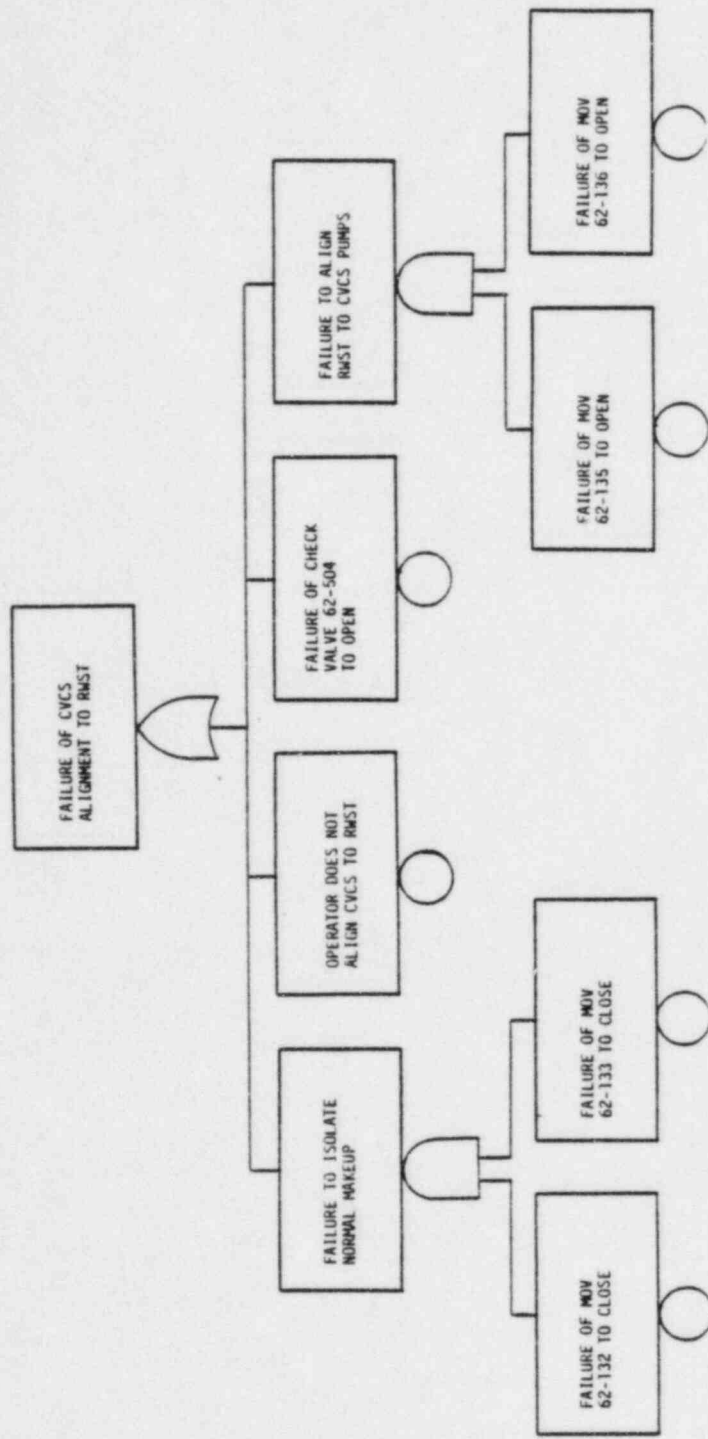


Figure A-4. Fault Tree for Event C.

A7

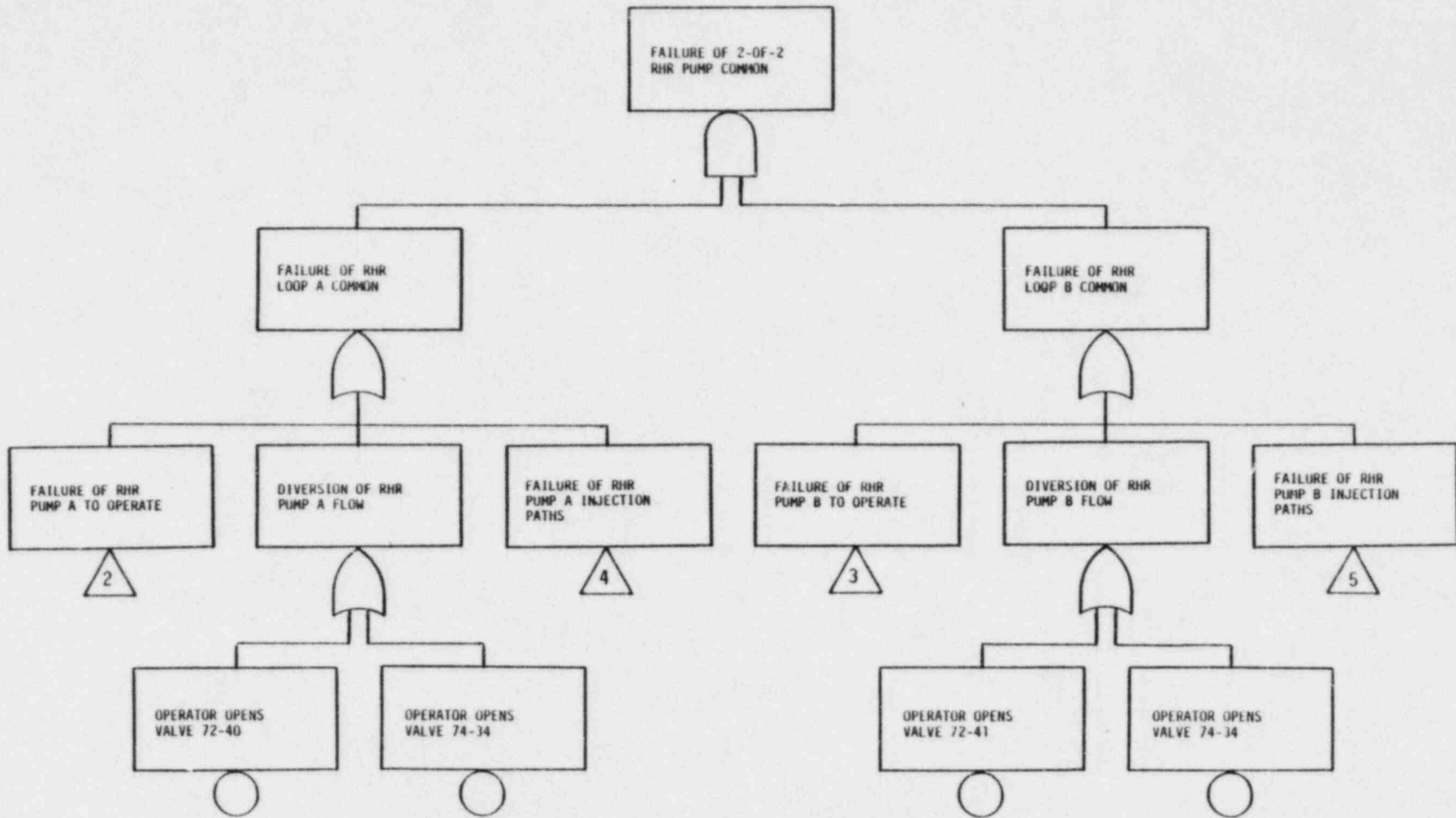


Figure A-5. Fault Tree for Event D.

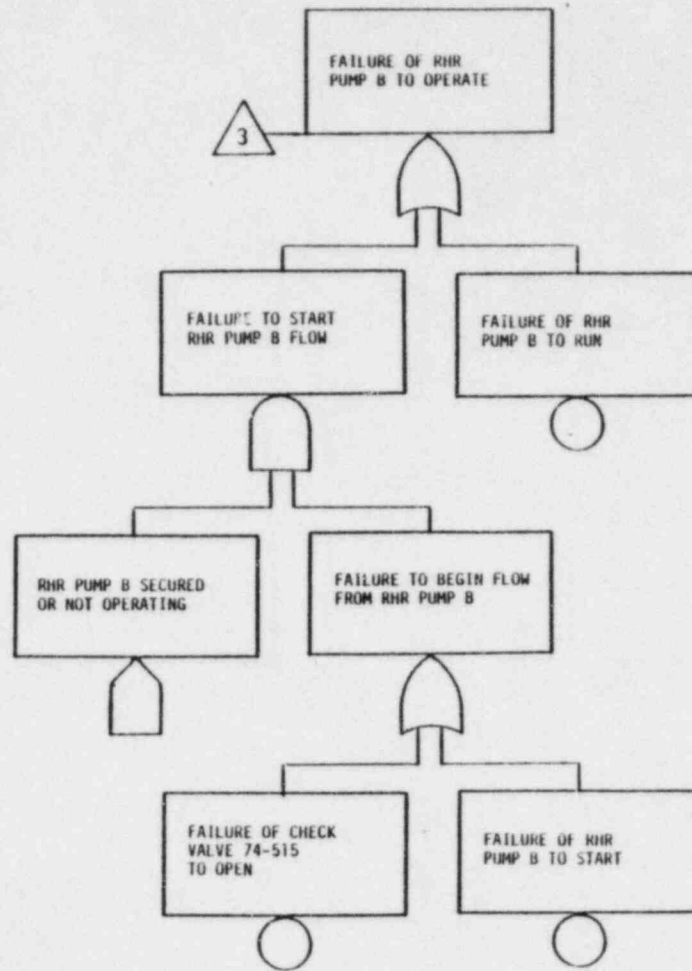
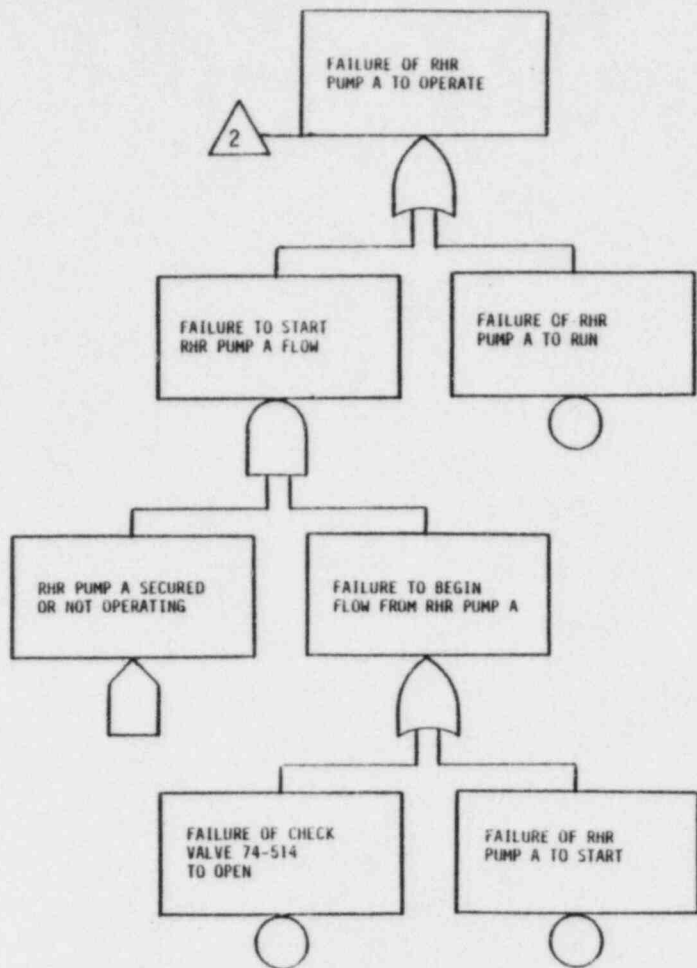


Figure A-5. Fault Tree for Event D (Continued).

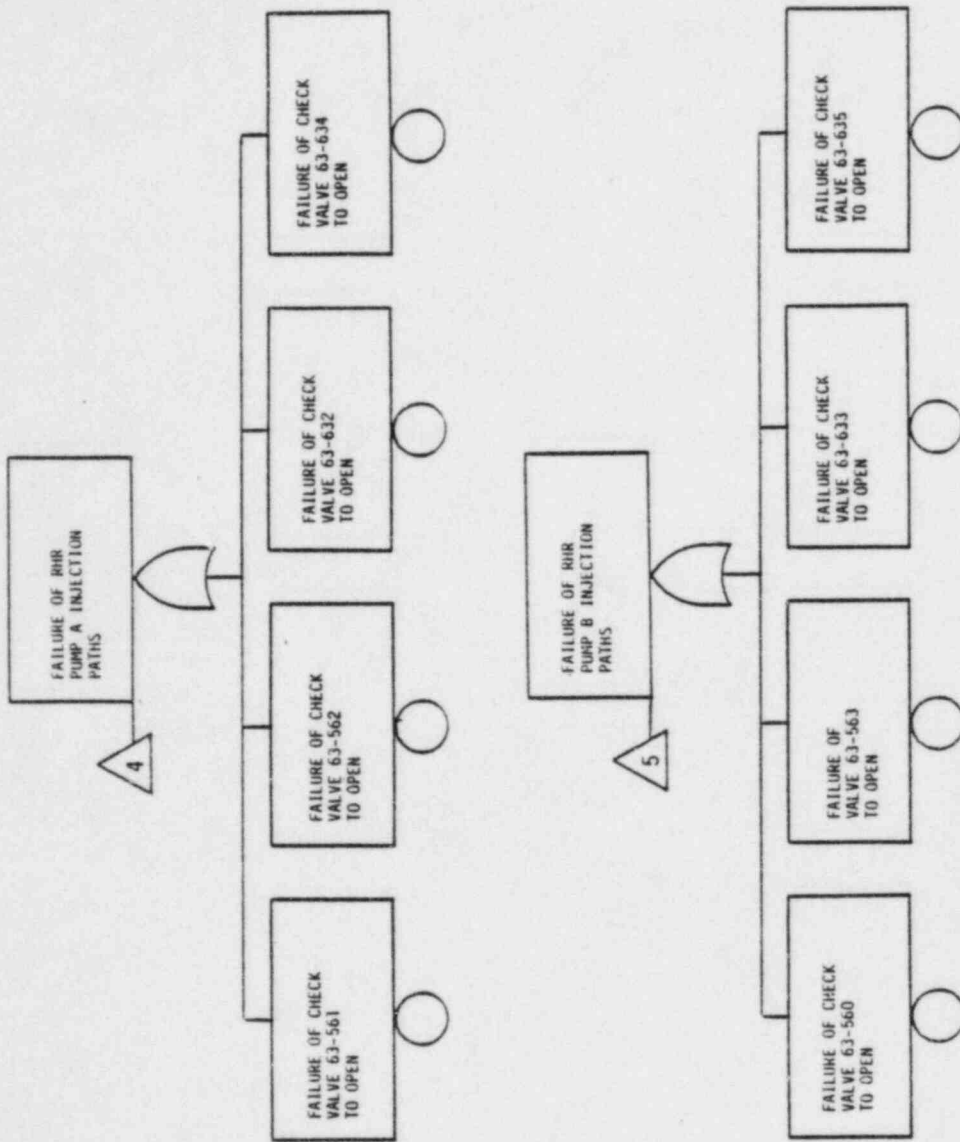


Figure A-5. Fault Tree for Event D (Continued).



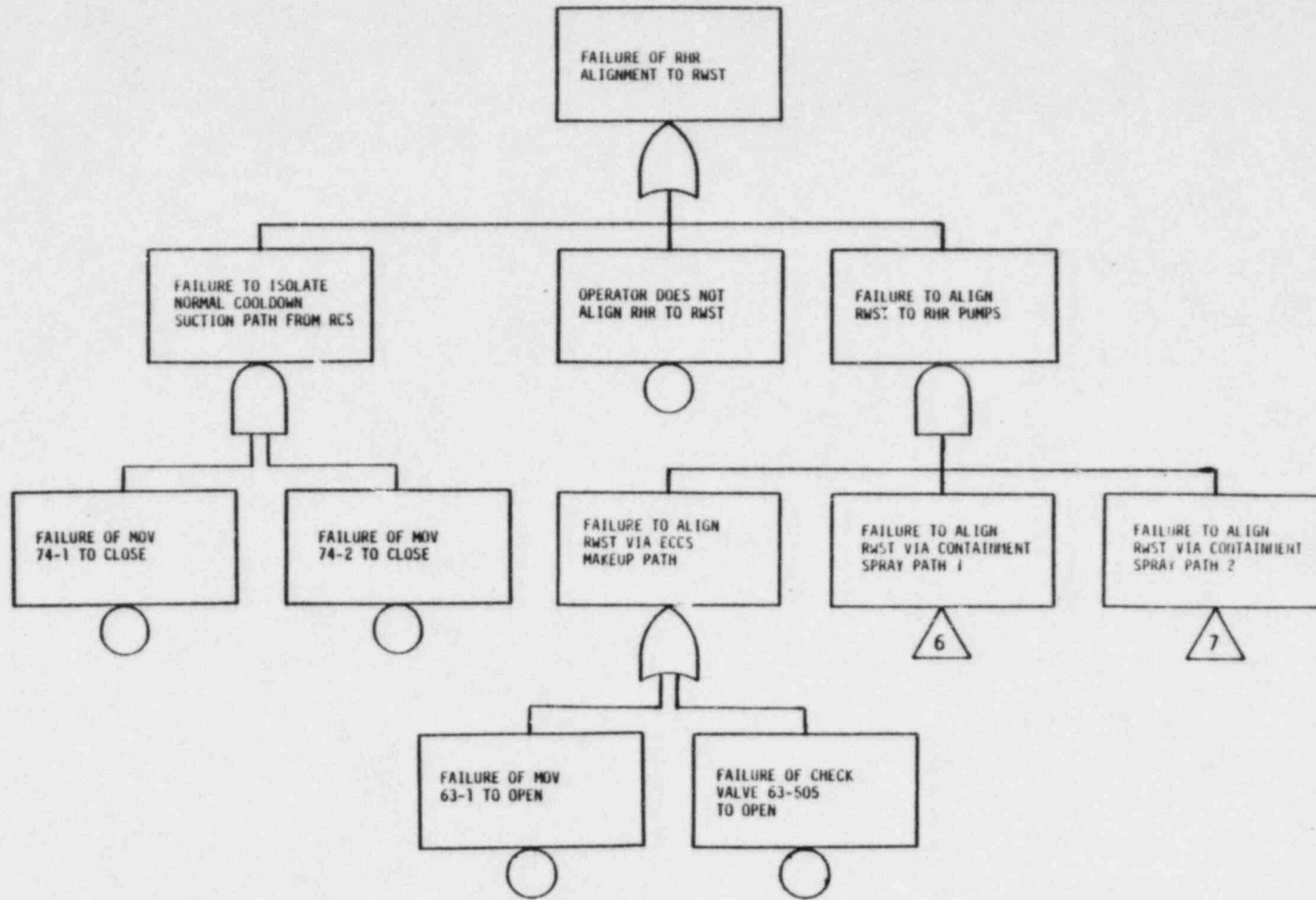


Figure A-6. Fault Tree for Event E.

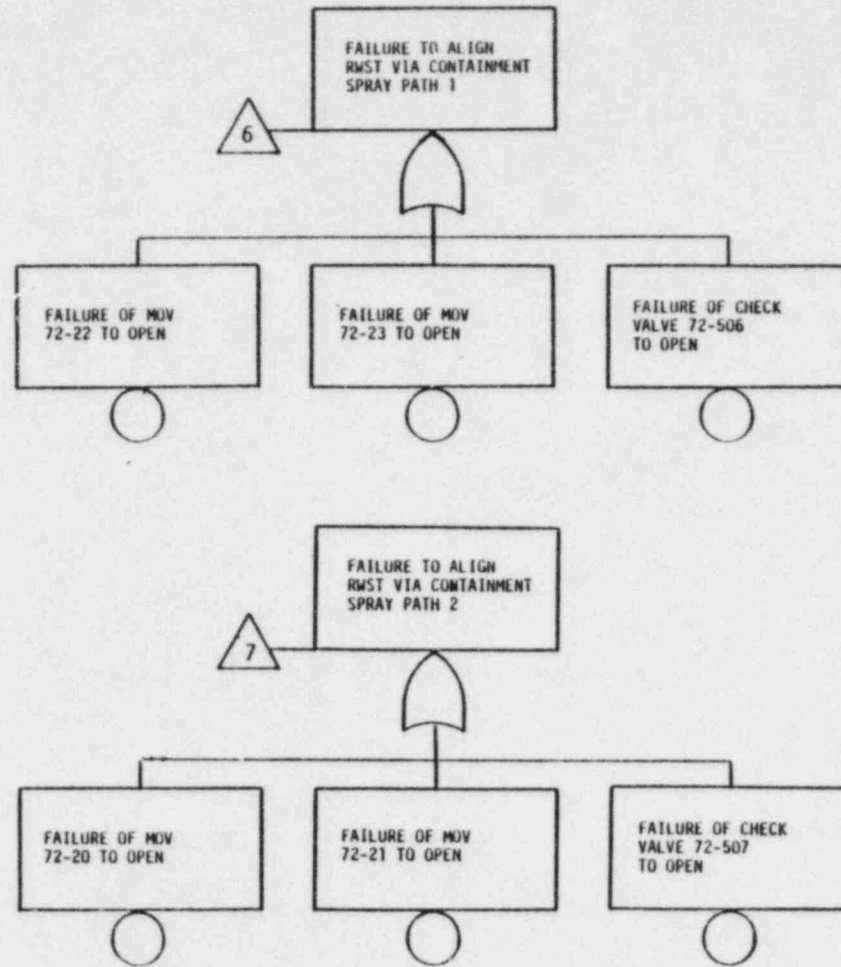


Figure A-6. Fault Tree for Event E (Continued).

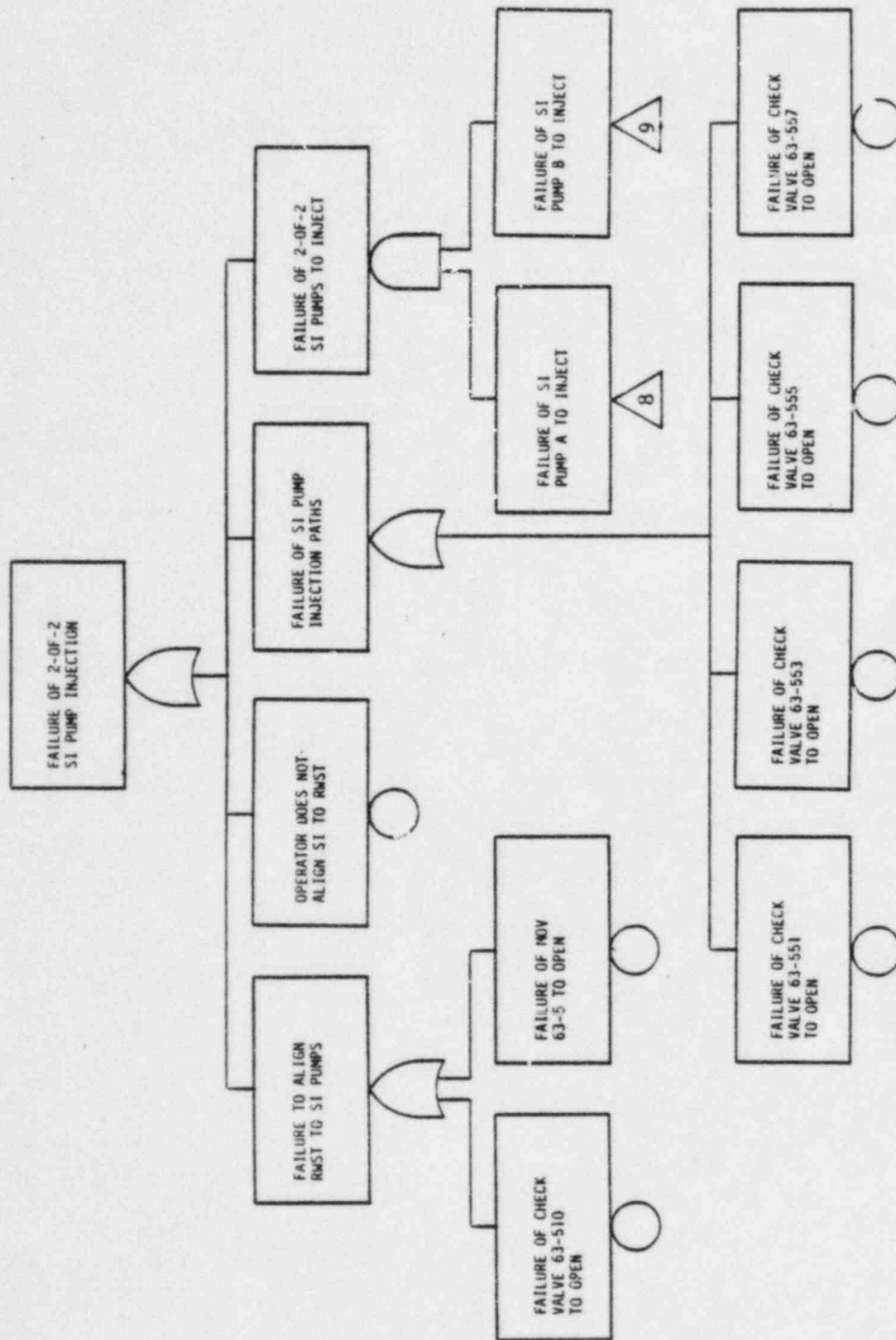


Figure A-7. Fault Tree for Event F.

A13

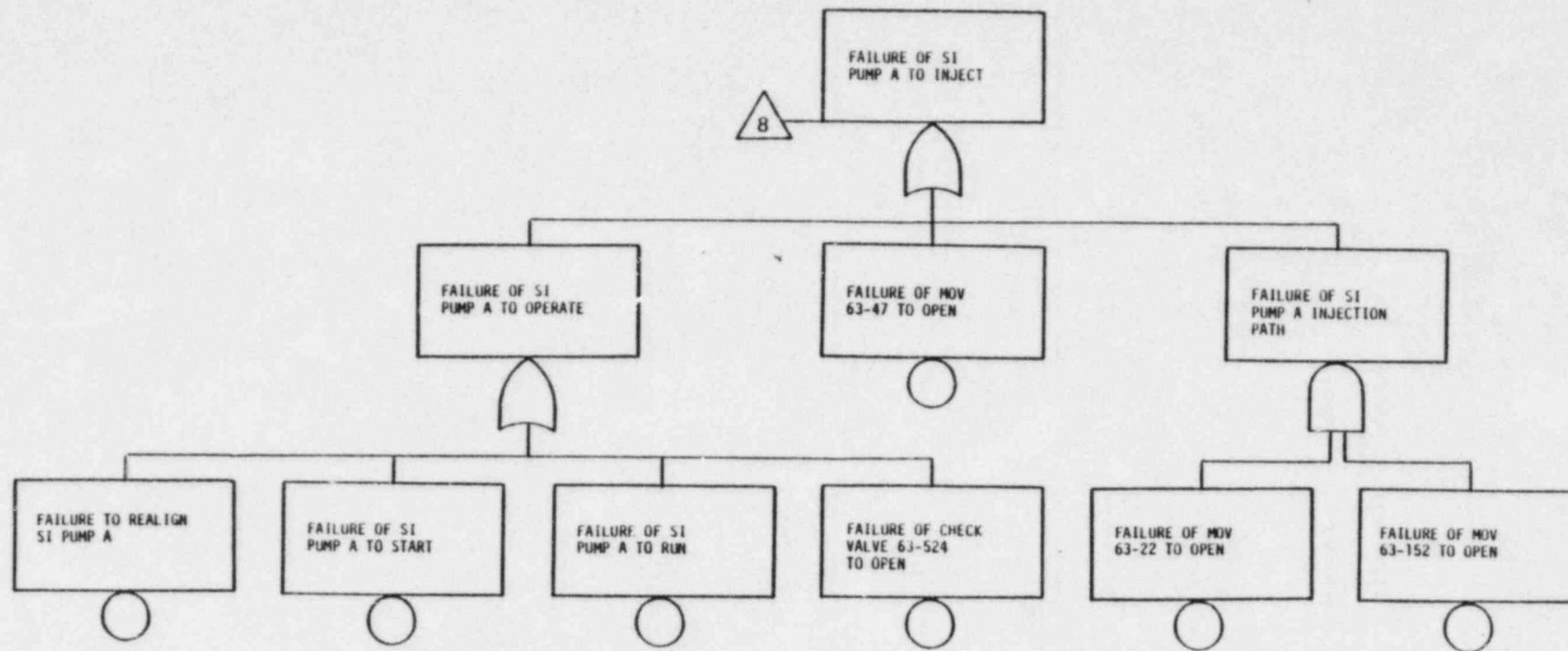


Figure A-7. Fault Tree for Event F (Continued).

A14

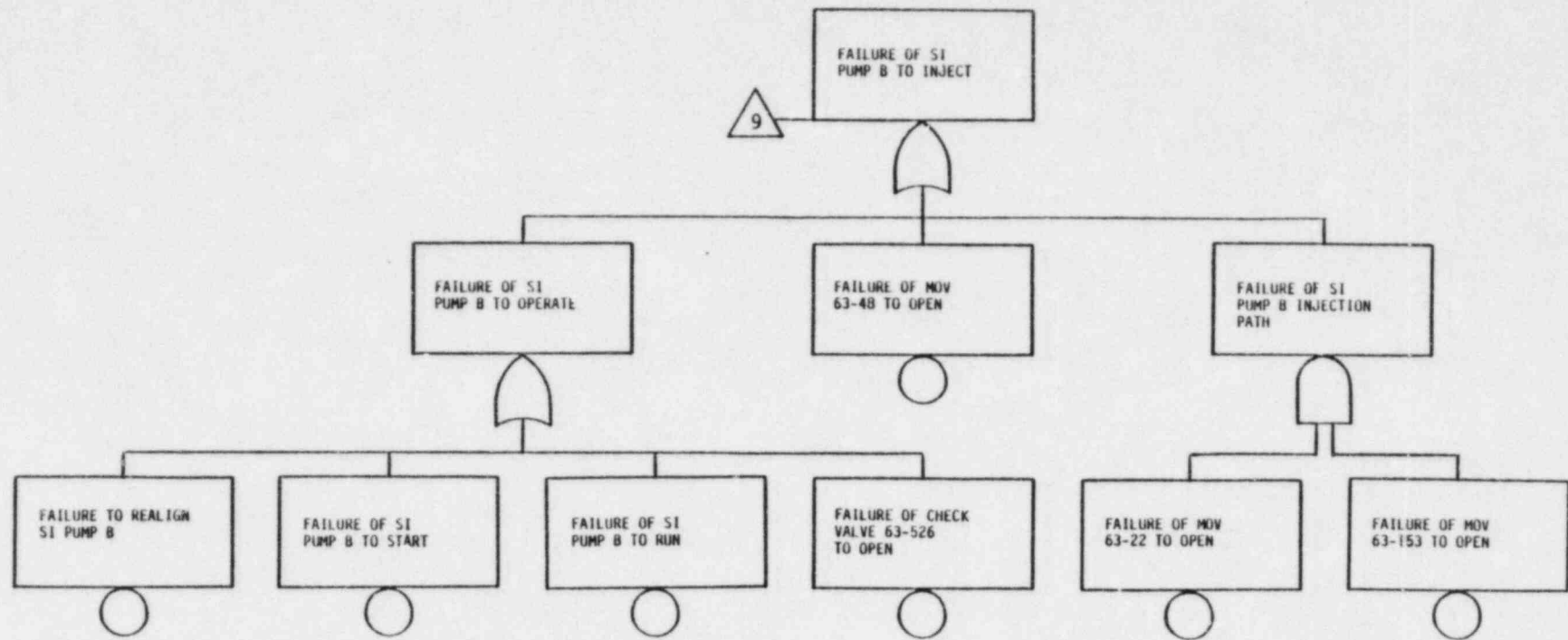


Figure A-7. Fault Tree for Event F (Continued).

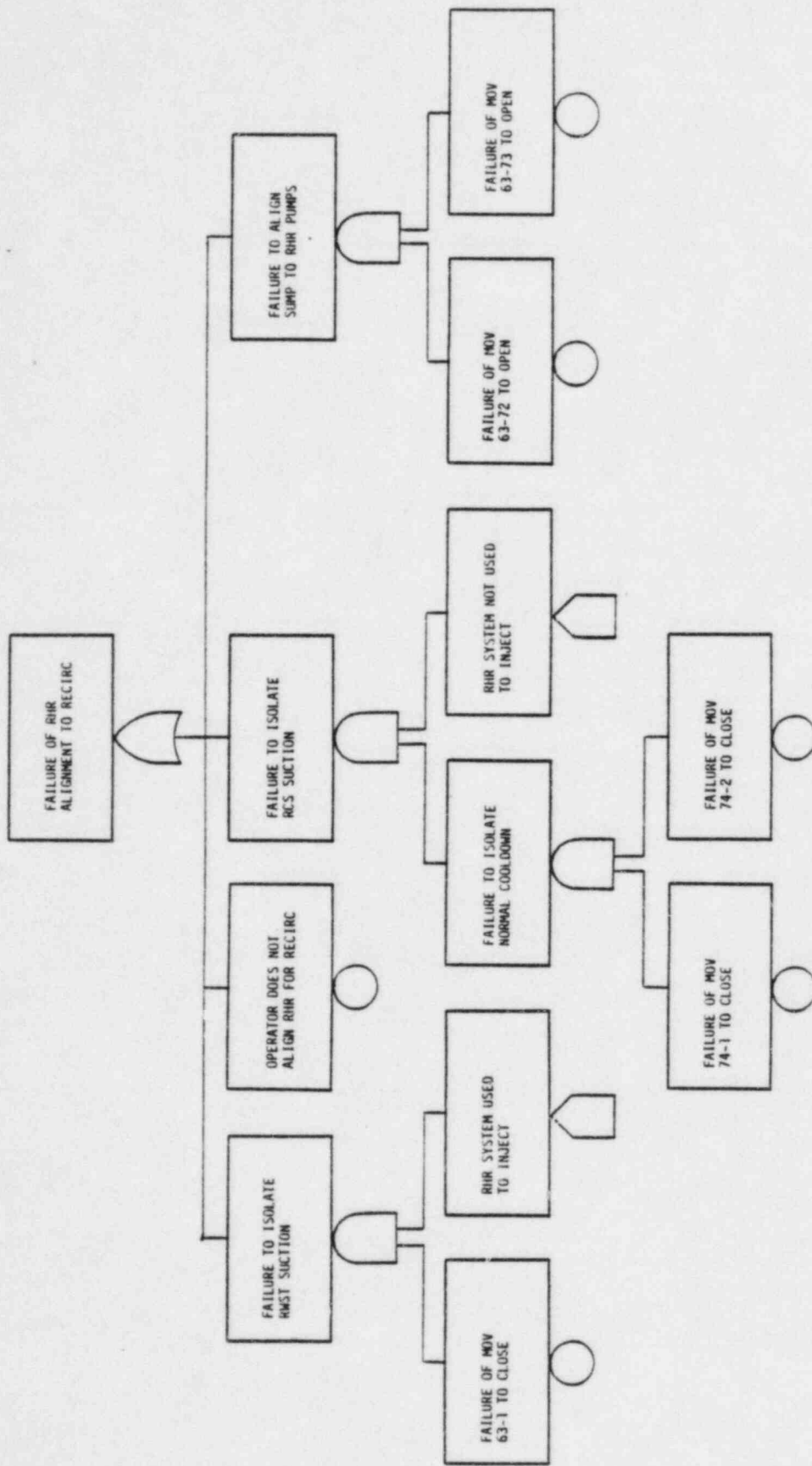


Figure A-8. Fault Tree for Event G.

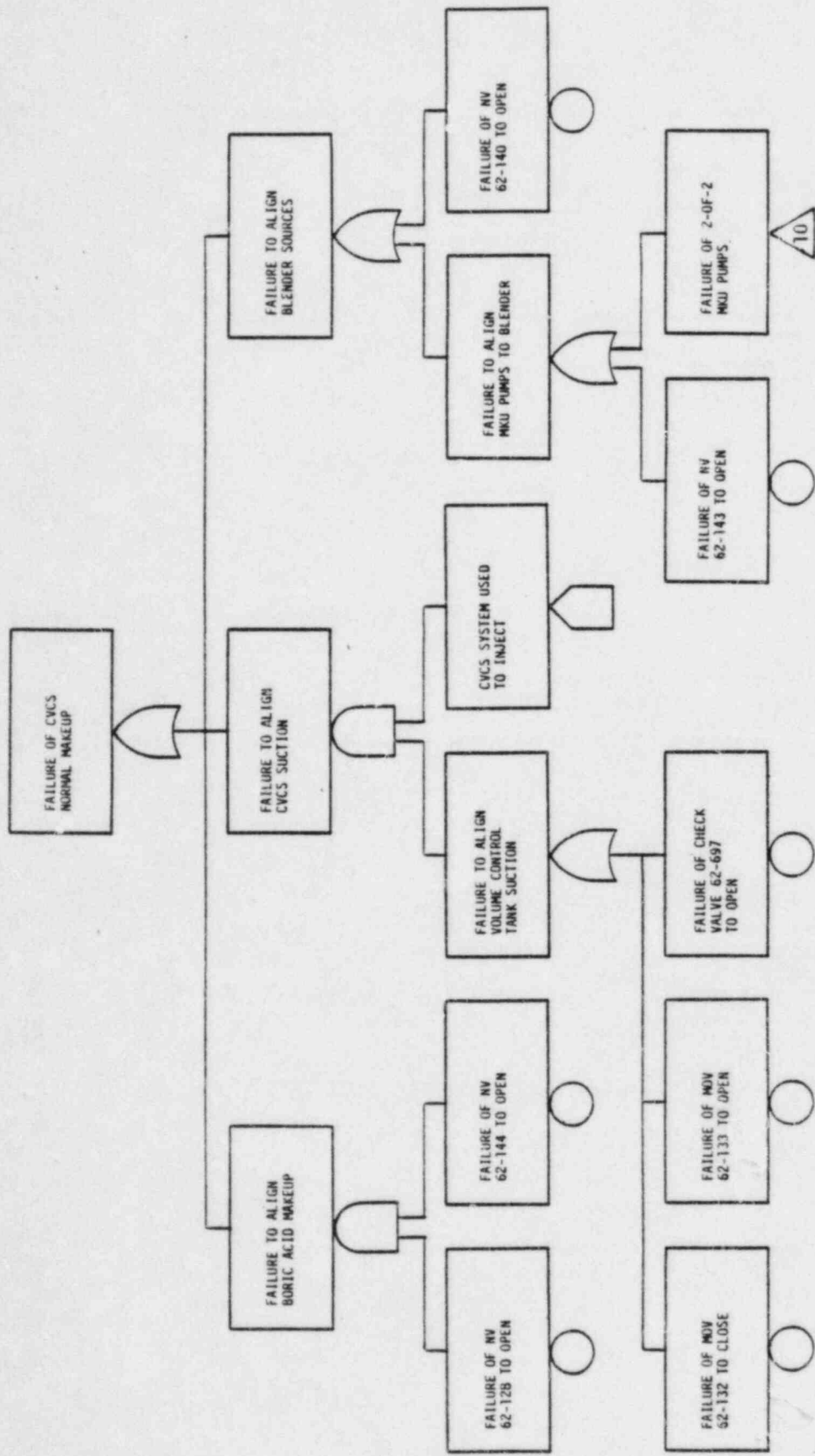


Figure A-9. Fault Tree for Event H.

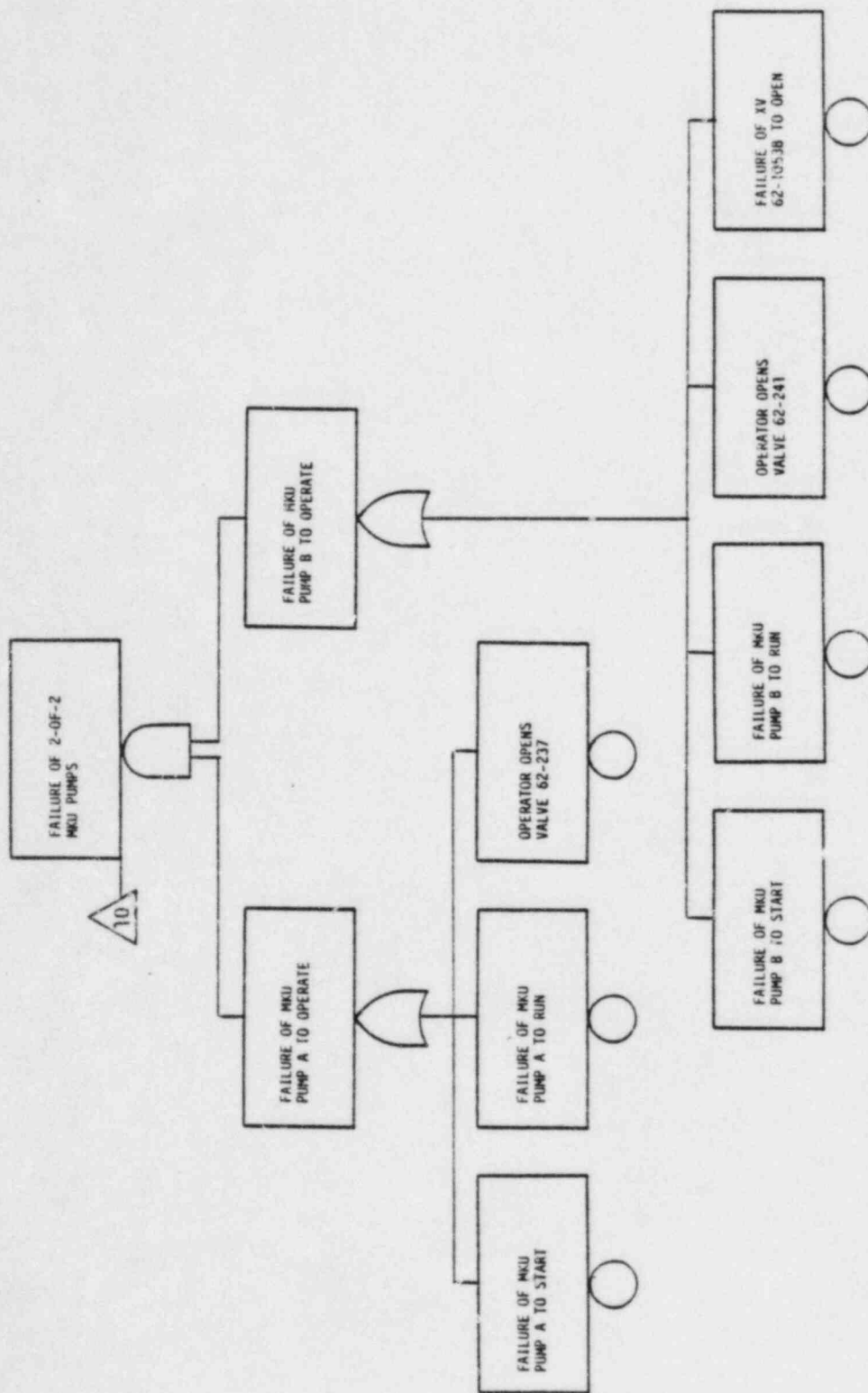


Figure A-9. Fault Tree for Event H (Continued).



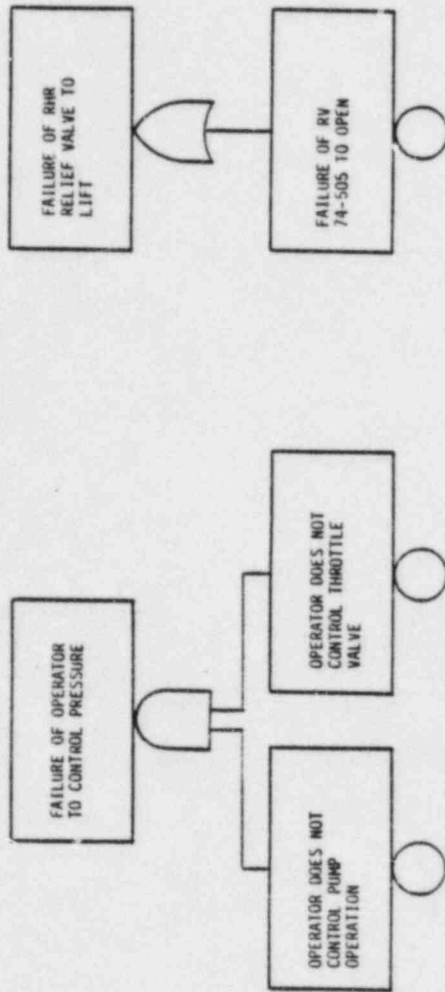


Figure A-10. Fault Trees for Events I and J.

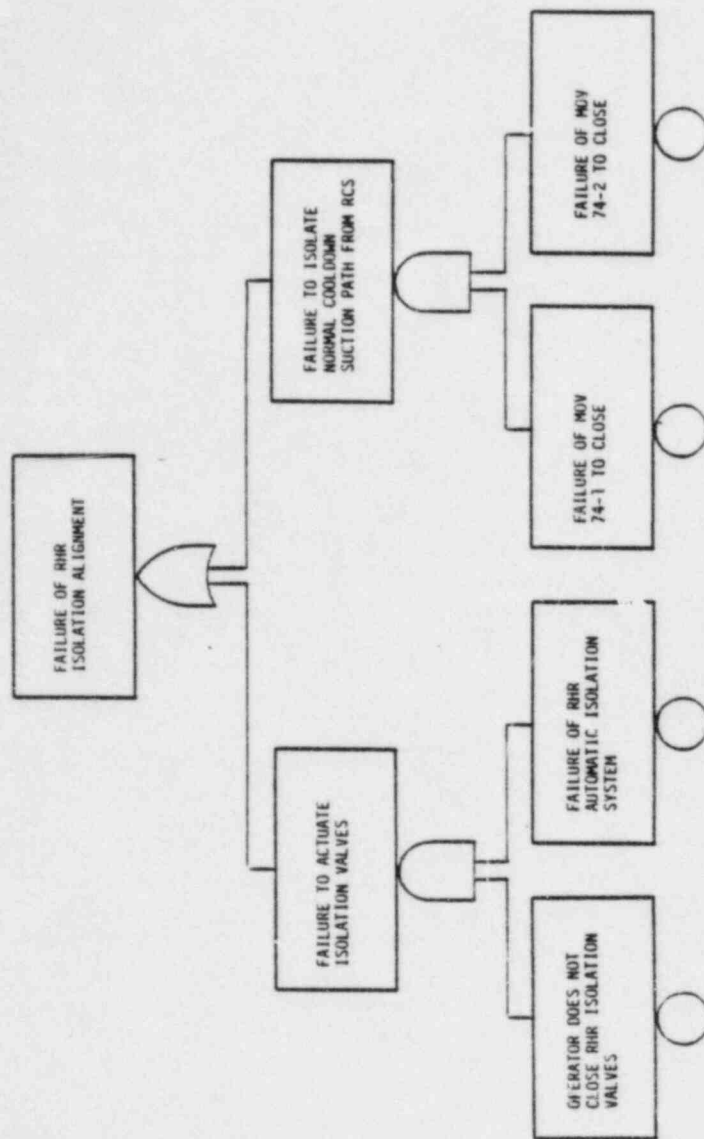


Figure A-11. Fault Tree for Event K.

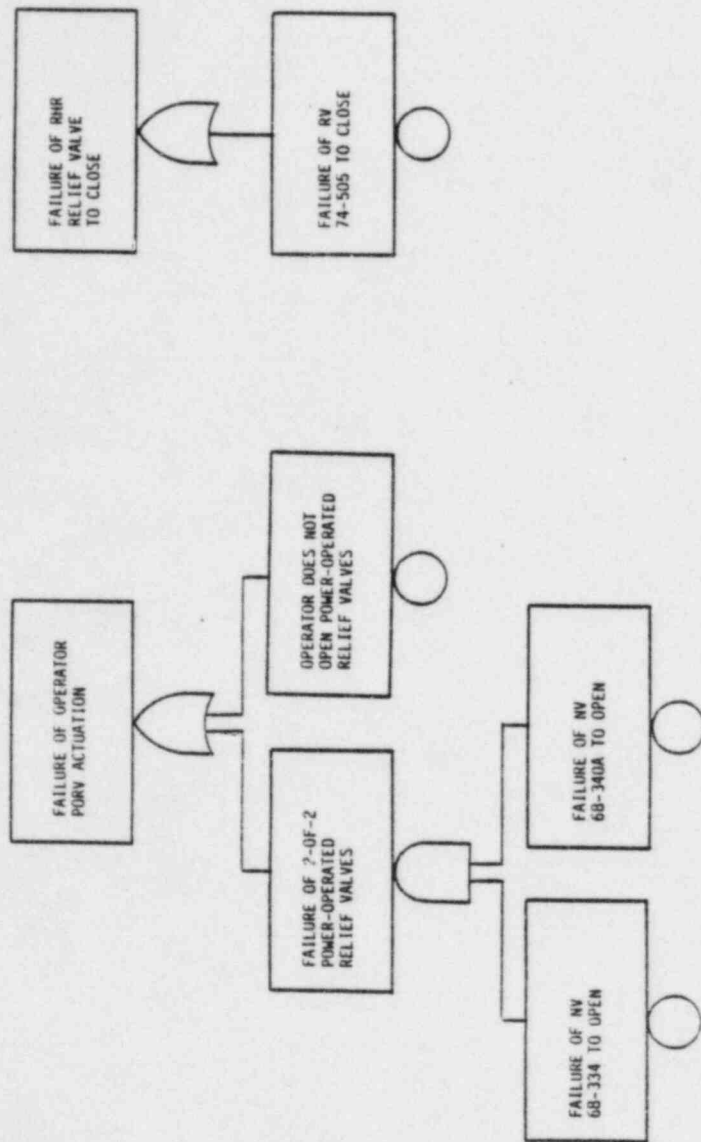


Figure A-12. Fault Trees for Events N and O.

## APPENDIX B - ESTIMATION OF LOCA LEAK RATES

This appendix contains the code listing and the detailed results of the calculated leak rates from various size pipe breaks postulated to occur in Mode 5. The basic geometry of the problem can be visualized as a large diameter RCS loop pipe (e.g., at least 27.5 inches in diameter) with a smaller diameter connecting pipe (e.g., 0.5 to 14 inches in diameter) that has a guillotine break either one foot or 100 feet from the interface with the RCS loop piping. Initially, the RCS is assumed to be at 600 psig (RHR system design pressure) and 200°F (upper limit for Mode 5). Additional assumptions include the following:

- o All pipes are hydraulically smooth
- o The geometry of the RCS loop pipe and the connecting smaller diameter pipe can be considered as an ordinary entrance.
- o Choke flow conditions do not exist

The code listing appears in Table B-1. This calculational routine is based on the mechanical energy balance equation as presented in Bennett and Meyers<sup>(1)</sup> and Perry (2). The results of the leak rate calculations for eleven different pipe sizes (0.5, 0.75, 1.0, 1.5, 2, 3, 4, 6, 8, 10, and 14 inch actual pipe inside diameter) are found in Table B-2. The calculated leak rates are summarized graphically in Section 3.

---

(1) Bennet, C. O. and Meyers, J. E., Momentum, Heat and Mass Transfer, McGraw Hill, 1962.

(2) Perry, R. H., Chilton, C. H., and Kirkpatrick, S. D. (editors), Chemical Engineering Handbook 5th Edition, McGraw Hill, 1973.

Table B-1  
Code Listing for Estimating  
Mode 5 LOCA Leak Rates

```

00010 REAL*4 LWORK,KIN,FC,LTERM
00020 DIMENSION A(4),C(6)
00030 DATA GC,KSTOP/32.17,20/
00040 DATA C/0.9356,-0.91934,-0.5389,0.4053,8.22E-04
00050 1,-258.2/
00060 DATA A/-0.1367,0.3687,-0.04206,0.001828/
00070 FRICT(X1,X2)=C(1)+C(2)*X1+C(3)*X2+C(4)*X1*X2
00080 1+C(5)*X1*X1+C(6)*X2*X2
00090 ADDX(X)=A(1)+X*(A(2)+X*(A(3)+X*A(4)))
00100 OPEN(UNIT=32,DIALOG='DSKD:PIPES.OUT')
00110 IOU=32
00120 C
00130 C THIS CODE CALCULATES THE FLUID VELOCITY THROUGH
00140 C A PIPE GIVEN SOME INITIAL PRESSURE AND ZERO
00150 C VELOCITY. THE EXIT OF THE PIPE IS AT ZERO PSIG.
00160 C AN ORDINARY ENTRANCE IS ASSUMED. PIPE WALL ROUGHNESS,
00170 C E/D, CAN RANGE FROM 0. TO 0.0025
00180 C THE MECHANICAL ENERGY BALANCE EQUATION IS USED AS
00190 C DESCRIBED IN: MOMENTUM, HEAT AND MASS TRANSFER
00200 C BY C.O. BENNETT & J.E. MYERS, (1962),
00210 C PAGES 154 THROUGH 169. POLYNOMIAL EXPRESSIONS
00220 C FOR THE FRICTION FACTOR (PAGE 166) AS A FUNCTION
00230 C OF REYNOLDS NUMBER AND PIPE ROUGHNESS, AND
00240 C EQUIVALENT LENGTH AS A FUNCTION OF INSIDE PIPE
00250 C DIAMETER WERE FITTED TO SOME DATA AND WRITTEN
00260 C AS FUNCTIONS AT THE BEGINNING OF THE CODE.
00270 C
00280 C JUNE 10, 1982
00290 C
00300 TYPE 10
00310 10 FORMAT(1X,'ENTER MAIN PIPE PRESSURE, PSIG')
00320 ACCEPT 20, PSIG
00330 20 FORMAT(F10.0)
00340 TYPE 30
00350 30 FORMAT(1X,'ENTER PIPE LENGTH, INCHES')
00360 ACCEPT 20, STUBIN
00370 TYPE 40
00380 40 FORMAT(1X,'ENTER PIPE DIAMETER, INCHES')
00390 ACCEPT 20, STUDD
00400 TYPE 50
00410 50 FORMAT(1X,'ENTER PIPE WALL E/D')
00420 ACCEPT 20, ED
00430 TYPE 60
00440 60 FORMAT(1X,'ENTER WATER VISCOSITY, CP')
00450 ACCEPT 20, CP
00460 TYPE 70
00470 70 FORMAT(1X,'ENTER WATER DENSITY, GM/CC')
00480 ACCEPT 20, DEN
00490 C
00500 C ECHO THE INPUT
00510 C
00520 80 TYPE 90, PSIG
00530 90 FORMAT(/,1X,'1. MAIN PRESSURE = ',F10.3,' PSIG')
00540 TYPE 100, STUBIN
00550 100 FORMAT(1X,'2. PIPE LENGTH = ',F7.2,' INCHES')
00560 TYPE 110, STUDD
00570 110 FORMAT(1X,'3. PIPE DIAMETER = ',F6.2,' INCHES')
00580 TYPE 120, ED
00590 120 FORMAT(1X,'4. PIPE WALL E/D = ',F8.5)
00600 TYPE 130, CP
00610 130 FORMAT(1X,'5. WATER VISCOSITY = ',F6.2,' CP')
00620 TYPE 140, DEN
00630 140 FORMAT(1X,'6. WATER DENSITY = ',F6.2,' GM/CC')
00640 TYPE 150
00650 150 FORMAT(/,1X,'WANT TO CHANGE ANY?')
00660 ACCEPT 160, ANS

```

Table B-1 (Continued)

```

00670 150  FORMAT(A1)
00680      IF(ANS.EQ.'N') GO TO 250
00690      TYPE 170
00700 170  FORMAT(/,1X,'ENTER THE LINE NUMBER TO BE CHANGED')
00710      ACCEPT 180, LINE
00720 180  FORMAT(I1)
00730      GO TO (190,200,210,220,230,240) LINE
00740 190  TYPE 10
00750      ACCEPT 20, PSIG
00760      GO TO 80
00770 200  TYPE 30
00780      ACCEPT 20, STUBIN
00790      GO TO 80
00800 210  TYPE 40
00810      ACCEPT 20, STUBD
00820      GO TO 80
00830 220  TYPE 50
00840      ACCEPT 20, ED
00850      GO TO 80
00860 230  TYPE 60
00870      ACCEPT 20, CP
00880      GO TO 80
00890 240  TYPE 70
00900      ACCEPT 20, DEN
00910      GO TO 80
00920  C
00930  C      NOW START THE CALCULATION
00940  C
00950 250  VIS=(6.72E-04)*CP
00960      DENSTY=62.4*DEN
00970      DP=144.*PSIG
00980      DIA=STUBD/12.
00990      AREA=0.785*DIA*DIA
01000      GTERM=448.86*AREA
01010  C
01020  C      CALCULATE EXTRA LENGTH DUE TO ENTRANCE.
01030  C
01040      XLENTH=ADDX(STUBD)
01050      XLENTH=10.**XLENTH
01060      TOTALX=XLENTH+STUBIN/12.
01070      GCD=GC*DIA
01080  C
01090  C      GUESS A VELOCITY = 10 FT/SEC
01100  C
01110      V=10.
01120      V2=V*V
01130      RETERM=DIA*DENSTY/VIS
01140      RE=V*RETERM
01150  C
01160  C      GET FANNING FRICTION FACTOR. SEE PAGE 166
01170  C      OF BENNETT AND MYERS.
01180  C
01190      REL=ALOG10(RE)
01200      F=FRICT(REL,ED)
01210      LTERM=2.*TOTALX/GCD
01220      LWORK=F*V2*LTERM
01230      GC2=2.*GC
01240      KINETC=V2/GC2
01250      PCALC=DENSTY*(LWORK+KINETC)
01260  C
01270  C      BRACKET THE VELOCITY.
01280  C
01290      KOUNT=0
01300      IDIR=1
01310      TEST=DP-PCALC
01320      IF(TEST.GT.0.) IDIR=2

```

Table B-1 (Continued)

```

01330 C
01340 C      IDIR=1 MEANS DP-PCALC < 0, DECREASE V.
01350 C      IDIR=2 MEANS DP-PCALC > 0, INCREASE V.
01360 C
01370 260 VOLD=V
01380 GO TO (270,280) IDIR
01390 270 V=V/2.
01400 GO TO 290
01410 280 V=2.*V
01420 290 V2=V*V
01430 RE=V*RETERM
01440 REL=ALOG10(RE)
01450 F=FRICT(REL,ED)
01460 LWORK=F*V2*LTERM
01470 KINETC=V2/CC2
01480 PCALC=DENSTY*(LWORK+KINETC)
01490 TEST=DP-PCALC
01500 IF(TEST.LT.0..AND.IDIR.EQ.2) GO TO 300
01510 IF(TEST.GT.0..AND.IDIR.EQ.1) GO TO 300
01520 KOUNT=KOUNT+1
01530 IF(KOUNT.GT.KSTOP) GO TO 400
01540 GO TO 260
01550 C
01560 C      NOW DO TRIAL AND ERROR TO GET THE VELOCITY.
01570 C
01580 300 KOUNT=0
01590 IF(V.LT.VOLD) GO TO 310
01600 VHIGH=V
01610 VLOW=VOLD
01620 GO TO 320
01630 310 VHIGH=VOLD
01640 VLOW=V
01650 320 V=(VHIGH+VLOW)/2.
01660 V2=V*V
01670 RE=V*RETERM
01680 REL=ALOG10(RE)
01690 F=FRICT(REL,ED)
01700 LWORK=F*V2*LTERM
01710 KINETC=V2/CC2
01720 PCALC=DENSTY*(LWORK+KINETC)
01730 TEST=DP-PCALC
01740 IF(TEST.GT.0.) GO TO 330
01750 VHIGH=V
01760 GO TO 340
01770 330 VLOW=V
01780 340 KOUNT=KOUNT+1
01790 IF(KOUNT.LT.KSTOP) GO TO 320
01800 GPM=GTERM*V
01810 TYPE 350, GPM
01820 350 FORMAT(/,1X,'FLOW RATE = ',1PE10.3,' GALLONS/MINUTE')
01830 C
01840 C      WRITE IT ALL OUT.
01850 C
01860 PCALC=PCALC/144.
01870 WRITE (100,360) PSIG,PCALC
01880 360 FORMAT(/,5X,'SPECIFIED MAIN HEADER PRESSURE = '
01890 1,F8.2,' , CALCULATED = ',F8.2,' PSIG')
01900 WRITE (100,370) CP,VIS
01910 370 FORMAT(5X,'WATER VISCOSITY = ',F5.2,' CENTIPOISE
01920 1, OR ',1PE10.3,' LBS(M)/(FT*SEC)')
01930 WRITE (100,380) STUBIN,STUBD
01940 380 FORMAT(5X,'PIPE LENGTH = ',F6.1,' INCHES, DIAMETER
01950 1 = ',F6.2,' INCHES')
01960 WRITE (100,390) ED
01970 390 FORMAT(5X,'PIPE WALL ROUGHNESS E/D = ',F8.3)
01980 WRITE (100,400) DEN,DENSTY

```

Table B-1 (Continued)

```

01990 400  FORMAT(3X,'WATER DENSITY = '.F6.4,' CM/CC. OR = '
02000 1.1PE10.3.' LES(M)/FT**3')
02010  WRITE (10U,410) V,GPM
02020 410  FORMAT(5X,'WATER VELOCITY = '.1PE10.3,' FT/SEC
02030 1, OR = '.1PE10.3,' GALLONS/MINUTE')
02040  WRITE (10U,420) AREA
02050 420  FORMAT(5X,'PIPE FLOW AREA = '.1PE10.3,' FT**2')
02060  WRITE (10U,430) XLENTH,TOTALX
02070 430  FORMAT(5X,'EXTRA LENGTH DUE TO ENTRANCE = '.F6.1
02080 1,' FT, TOTAL LENGTH = '.1PE10.3,' FT')
02090  WRITE (10U,440) RE,F
02100 440  FORMAT(5X,'REYNOLDS NUMBER = '.1PE10.3,', FRICTION
02110 1 FACTOR = '.0PF8.5)
02120  WRITE (10U,450) KINETC,LWORK
02130 450  FORMAT(5X,'KINETIC TERM = '.1PE10.3,', LOST WORK
02140 1 TERM = '.1PE10.3,' LES(F)*FT/LBS(M)')
02150  TYPE 460
02160 460  FORMAT(/,1X,'DO IT AGAIN?')
02170  ACCEPT 470, ANS
02180 470  FORMAT(A1)
02190  IF(ANS.EQ.'Y') GO TO 80
02200  GO TO 500
02210 480  TYPE 490
02220 490  FORMAT(/,1X,'NO BRACKET')
02230 500  CONTINUE
02240  END

```



Table B-2

Output Data Summaries For Mode 5  
LOCA Leak Rate Calculations

Pipe Diameter 0.5 Inches

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.018E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 12.0 INCHES, DIAMETER = 0.50 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 2.337E+02 FT/SEC, OR = 1.460E+02 GALLONS/MINUTE  
 PIPE FLOW AREA = 1.869E-03 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 1.1 FT, TOTAL LENGTH = 2.090E+00 FT  
 REYNOLDS NUMBER = 2.968E+06, FRICTION FACTOR = 0.00311  
 KINETIC TERM = 3.077E+02, LOST WORK TERM = 3.307E+02 LBS(F)\*FT/LBS(M)

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 1200.0 INCHES, DIAMETER = 0.50 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 3.322E+01 FT/SEC, OR = 3.255E+01 GALLONS/MINUTE  
 PIPE FLOW AREA = 1.869E-03 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 1.1 FT, TOTAL LENGTH = 1.911E+02 FT  
 REYNOLDS NUMBER = 6.607E+05, FRICTION FACTOR = 0.00326  
 KINETIC TERM = 4.402E+01, LOST WORK TERM = 1.394E+03 LBS(F)\*FT/LBS(M)

Pipe Diameter 0.75 Inches

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 12.0 INCHES, DIAMETER = 0.75 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 2.506E+02 FT/SEC, OR = 3.450E+02 GALLONS/MINUTE  
 PIPE FLOW AREA = 3.066E-03 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 1.3 FT, TOTAL LENGTH = 2.309E+00 FT  
 REYNOLDS NUMBER = 4.668E+06, FRICTION FACTOR = 0.00320  
 KINETIC TERM = 9.764E+02, LOST WORK TERM = 4.620E+02 LBS(F)\*FT/LBS(M)

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 1200.0 INCHES, DIAMETER = 0.75 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 6.607E+01 FT/SEC, OR = 9.093E+01 GALLONS/MINUTE  
 PIPE FLOW AREA = 3.066E-03 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 1.3 FT, TOTAL LENGTH = 1.013E+02 FT  
 REYNOLDS NUMBER = 1.230E+06, FRICTION FACTOR = 0.00312  
 KINETIC TERM = 6.784E+01, LOST WORK TERM = 1.371E+03 LBS(F)\*FT/LBS(M)

Table B-2 (Continued)

Pipe Diameter 1 Inch

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 12.0 INCHES, DIAMETER = 1.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 2.567E+02 FT/SEC, OR = 6.280E+02 GALLONS/MINUTE  
 PIPE FLOW AREA = 5.451E-03 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 1.6 FT, TOTAL LENGTH = 2.555E+00 FT  
 REYNOLDS NUMBER = 6.373E+06, FRICTION FACTOR = 0.00330  
 KINETIC TERM = 1.024E+03, LOST WORK TERM = 4.145E+02 LBS(F)\*FT/LBS(M)

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 1200.0 INCHES, DIAMETER = 1.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 7.598E+01 FT/SEC, OR = 1.859E+02 GALLONS/MINUTE  
 PIPE FLOW AREA = 5.451E-03 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 1.6 FT, TOTAL LENGTH = 1.016E+02 FT  
 REYNOLDS NUMBER = 1.387E+06, FRICTION FACTOR = 0.00308  
 KINETIC TERM = 8.973E+01, LOST WORK TERM = 1.349E+03 LBS(F)\*FT/LBS(M)

Pipe Diameter 1.5 Inches

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 12.0 INCHES, DIAMETER = 1.50 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 2.619E+02 FT/SEC, OR = 1.442E+03 GALLONS/MINUTE  
 PIPE FLOW AREA = 1.227E-02 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 2.1 FT, TOTAL LENGTH = 3.128E+00 FT  
 REYNOLDS NUMBER = 9.756E+06, FRICTION FACTOR = 0.00349  
 KINETIC TERM = 1.066E+03, LOST WORK TERM = 3.720E+02 LBS(F)\*FT/LBS(M)

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 1200.0 INCHES, DIAMETER = 1.50 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 9.078E+01 FT/SEC, OR = 4.998E+02 GALLONS/MINUTE  
 PIPE FLOW AREA = 1.227E-02 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 2.1 FT, TOTAL LENGTH = 1.021E+02 FT  
 REYNOLDS NUMBER = 3.381E+06, FRICTION FACTOR = 0.00313  
 KINETIC TERM = 1.281E+02, LOST WORK TERM = 1.310E+03 LBS(F)\*FT/LBS(M)

Table B-2 (Continued)

Pipe Diameter 2 inches

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 12.0 INCHES, DIAMETER = 2.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 2.635E+02 FT/SEC, OR = 2.579E+03 GALLONS/MINUTE  
 PIPE FLOW AREA = 2.181E-02 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 2.8 FT, TOTAL LENGTH = 3.800E+00 FT  
 REYNOLDS NUMBER = 1.309E+07, FRICTION FACTOR = 0.00365  
 KINETIC TERM = 1.080E+03, LOST WORK TERM = 3.589E+02 LBS(F)\*FT/LBS(M)

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 1200.0 INCHES, DIAMETER = 2.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 1.017E+02 FT/SEC, OR = 9.951E+02 GALLONS/MINUTE  
 PIPE FLOW AREA = 2.181E-02 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 2.8 FT, TOTAL LENGTH = 1.028E+02 FT  
 REYNOLDS NUMBER = 5.049E+06, FRICTION FACTOR = 0.00322  
 KINETIC TERM = 1.607E+02, LOST WORK TERM = 1.278E+03 LBS(F)\*FT/LBS(M)

Pipe Diameter 3 Inches

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 12.0 INCHES, DIAMETER = 3.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 2.632E+02 FT/SEC, OR = 5.797E+03 GALLONS/MINUTE  
 PIPE FLOW AREA = 4.906E-02 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 4.4 FT, TOTAL LENGTH = 5.367E+00 FT  
 REYNOLDS NUMBER = 1.961E+07, FRICTION FACTOR = 0.00391  
 KINETIC TERM = 1.077E+03, LOST WORK TERM = 3.616E+02 LBS(F)\*FT/LBS(M)

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 1200.0 INCHES, DIAMETER = 3.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 1.173E+02 FT/SEC, OR = 2.582E+03 GALLONS/MINUTE  
 PIPE FLOW AREA = 4.906E-02 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 4.4 FT, TOTAL LENGTH = 1.044E+02 FT  
 REYNOLDS NUMBER = 8.734E+06, FRICTION FACTOR = 0.00343  
 KINETIC TERM = 2.137E+02, LOST WORK TERM = 1.225E+03 LBS(F)\*FT/LBS(M)

Table B-2 (Continued)

Pipe Diameter 4 Inches

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 12.0 INCHES, DIAMETER = 4.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 2.619E+02 FT/SEC, OR = 1.025E+04 GALLONS/MINUTE  
 PIPE FLOW AREA = 8.722E-02 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 6.1 FT, TOTAL LENGTH = 7.053E+00 FT  
 REYNOLDS NUMBER = 2.601E+07, FRICTION FACTOR = 0.00412  
 KINETIC TERM = 1.066E+03, LOST WORK TERM = 3.723E+02 LBS(F)\*FT/LBS(M)

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 1200.0 INCHES, DIAMETER = 4.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 1.283E+02 FT/SEC, OR = 5.024E+03 GALLONS/MINUTE  
 PIPE FLOW AREA = 8.722E-02 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 6.1 FT, TOTAL LENGTH = 1.061E+02 FT  
 REYNOLDS NUMBER = 1.274E+07, FRICTION FACTOR = 6.00363  
 KINETIC TERM = 2.559E+02, LOST WORK TERM = 1.182E+03 LBS(F)\*FT/LBS(M)

Pipe Diameter 6 Inches

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 12.0 INCHES, DIAMETER = 6.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 2.609E+02 FT/SEC, OR = 2.299E+04 GALLONS/MINUTE  
 PIPE FLOW AREA = 1.962E-01 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 9.0 FT, TOTAL LENGTH = 1.004E+01 FT  
 REYNOLDS NUMBER = 3.887E+07, FRICTION FACTOR = 0.00447  
 KINETIC TERM = 1.058E+03, LOST WORK TERM = 3.802E+02 LBS(F)\*FT/LBS(M)

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 1200.0 INCHES, DIAMETER = 6.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 1.439E+02 FT/SEC, OR = 1.268E+04 GALLONS/MINUTE  
 PIPE FLOW AREA = 1.962E-01 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 9.0 FT, TOTAL LENGTH = 1.090E+02 FT  
 REYNOLDS NUMBER = 2.144E+07, FRICTION FACTOR = 0.00398  
 KINETIC TERM = 3.220E+02, LOST WORK TERM = 1.116E+03 LBS(F)\*FT/LBS(M)

Table B-2 (Continued)

Pipe Diameter 8 Inches

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 12.0 INCHES, DIAMETER = 8.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 2.615E+02 FT/SEC, OR = 4.095E+04 GALLONS/MINUTE  
 PIPE FLOW AREA = 3.489E-01 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 11.4 FT, TOTAL LENGTH = 1.240E+01 FT  
 REYNOLDS NUMBER = 3.193E+07, FRICTION FACTOR = 0.00473  
 KINETIC TERM = 1.063E+03, LOST WORK TERM = 3.759E+02 LBS(F)\*FT/LBS(M)

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 1200.0 INCHES, DIAMETER = 8.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 1.550E+02 FT/SEC, OR = 2.428E+04 GALLONS/MINUTE  
 PIPE FLOW AREA = 3.489E-01 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 11.4 FT, TOTAL LENGTH = 1.114E+02 FT  
 REYNOLDS NUMBER = 3.079E+07, FRICTION FACTOR = 0.00426  
 KINETIC TERM = 3.736E+02, LOST WORK TERM = 1.065E+03 LBS(F)\*FT/LBS(M)

Pipe Diameter 10 Inches

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 12.0 INCHES, DIAMETER = 10.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 2.590E+02 FT/SEC, OR = 6.338E+04 GALLONS/MINUTE  
 PIPE FLOW AREA = 5.451E-01 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 14.9 FT, TOTAL LENGTH = 1.587E+01 FT  
 REYNOLDS NUMBER = 6.432E+07, FRICTION FACTOR = 0.00498  
 KINETIC TERM = 1.043E+03, LOST WORK TERM = 3.955E+02 LBS(F)\*FT/LBS(M)

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
 WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
 PIPE LENGTH = 1200.0 INCHES, DIAMETER = 10.00 INCHES  
 PIPE WALL ROUGHNESS E/D = 0.00000  
 WATER DENSITY = .9626 GM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
 WATER VELOCITY = 1.629E+02 FT/SEC, OR = 3.987E+04 GALLONS/MINUTE  
 PIPE FLOW AREA = 5.451E-01 FT\*\*2  
 EXTRA LENGTH DUE TO ENTRANCE = 14.9 FT, TOTAL LENGTH = 1.149E+02 FT  
 REYNOLDS NUMBER = 4.045E+07, FRICTION FACTOR = 0.00431  
 KINETIC TERM = 4.126E+02, LOST WORK TERM = 1.026E+03 LBS(F)\*FT/LBS(M)

Table B-2 (Continued)

Pipe Diameter 14 Inches

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
PIPE LENGTH = 12.0 INCHES, DIAMETER = 14.00 INCHES  
PIPE WALL ROUGHNESS E/D = 0.00000  
WATER DENSITY = .9626 CM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
WATER VELOCITY = 2.091E+02 FT/SEC, OR = 1.003E+05 GALLONS/MINUTE  
PIPE FLOW AREA = 1.068E+00 FT\*\*2  
EXTRA LENGTH DUE TO ENTRANCE = 62.7 FT, TOTAL LENGTH = 6.372E+01 FT  
REYNOLDS NUMBER = 7.268E+07, FRICTION FACTOR = 0.00511  
KINETIC TERM = 6.794E+02, LOST WORK TERM = 7.590E+02 LBS(F)\*FT/LBS(M)

SPECIFIED MAIN HEADER PRESSURE = 600.00, CALCULATED = 600.00 PSIG  
WATER VISCOSITY = 0.30 CENTIPOISE, OR 2.016E-04 LBS(M)/(FT\*SEC)  
PIPE LENGTH = 1200.0 INCHES, DIAMETER = 14.00 INCHES  
PIPE WALL ROUGHNESS E/D = 0.00000  
WATER DENSITY = .9626 CM/CC, OR = 6.007E+01 LBS(M)/FT\*\*3  
WATER VELOCITY = 1.585E+02 FT/SEC, OR = 7.599E+04 GALLONS/MINUTE  
PIPE FLOW AREA = 1.068E+00 FT\*\*2  
EXTRA LENGTH DUE TO ENTRANCE = 62.7 FT, TOTAL LENGTH = 1.627E+02 FT  
REYNOLDS NUMBER = 5.568E+07, FRICTION FACTOR = 0.00481  
KINETIC TERM = 3.902E+02, LOST WORK TERM = 1.048E+03 LBS(F)\*FT/LBS(M)

