C-E Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Tel. 203/688-1911
Telex: 99297

POWER SYSTEMS

March 10, 1983
LD-83-019

Docket No.: STN 50-470F

Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: CESSAR-F SER Confirmatory Item 16

Reference: Letter, R. L. Tedesco to A. E. Scherer, dated January 15, 1982

Dear Mr. Eisenhut:

The reference requested that Combustion Engineering (C-E) perform an analysis of a locked reactor coolant pump shaft for System 80™ assuming a turbine trip with coincident loss of offsite power and the worst single failure of a safety system active component. It was also stated that the low population zone dose acceptance criteria should be three hundred (300) REM, consistent with 10 CFR 100 guidelines.

The new analysis was performed and informally submitted to the staff reviewer last year. We have been subsequently informed that the informal submittal is acceptable and have been requested to formally submit the material on the docket. Enclosed please find, therefore, the amended CESSAR-F Section 15.3.3, "Single Reactor Coolant Pump Rotor Seizure With Loss of Offsite Power". The enclosure will be incorporated into CESSAR-F via the next amendment. The revised section shows that the CESSAR-F locked rotor analysis is conservative with respect to the acceptance criteria. As a result, we believe that this submittal should resolve Confirmatory Item 16 of the CESSAR-F SER.

If I can be of any further assistance in this matter, please contact either myself or Mr. G. A. Davis of my staff at (203) 688-1911, extension 2803.

Very truly yours,

COMBUSTION ENGINEERING, INC.,

A. E. Scherer
Director
Nuclear Licensing

AES:las
Enclosure

E003

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Chapter 15

15.3.2 FLOW CONTROLLER MALFUNCTION CAUSING FLOW COASTDOWN

This event is categorized as a Boiling Water Reactor event in SRP 15.3.2 and, therefore, will not be analyzed.

15.3.3 SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER

15.3.3.1 Identification of Event and Causes

A single reactor coolant pump rotor seizure can be caused by seizure of the upper or lower thrust-journal bearings. Loss of offsite power subsequent to turbine/generator trip may be caused by a complete loss of the external electrical grid triggered by the turbine/generator trip. The loss of offsite power causes a loss of power to the start-up transformers which prevents the plant electrical loads from being transferred to them from the unit auxiliary transformers. Therefore, the onsite loads will lose power and the plant will experience a simultaneous loss of feedwater flow, condenser inoperability, and a coastdown of all reactor coolant pumps. Approximately 12 seconds after the loss of offsite power occurs the diesel generators start providing power to the two plant 4.16 kV safety buses. No credit is taken for restoration of offsite power prior to initiation of shutdown cooling.

For decreasing reactor coolant flow events, the major parameter of concern is the minimum hot channel DNBR. This parameter establishes whether a fuel design limit has been violated and, thus, whether fuel damage could be anticipated. Those factors which cause a decrease in local DNBR are:

a. increasing coolant temperature
b. decreasing coolant pressure
c. increasing local heat flux (including radial and axial power distribution effects)
d. decreasing coolant flow

For the single reactor coolant pump rotor seizure event, the minimum DNBR occurs during the first one to four seconds of the transient, and the reactor is tripped by the CPCs on the approach to the DNBR limit. Therefore, any single failure which would result in a lower DNBR during the transient would have to affect at least one of the above parameters during the first one to four seconds of the event.

The single failures that have been postulated are listed in Table 15.0-6. Most of these failures affect the secondary system, and during the first one to four seconds they do not affect the primary system parameters which determine the DNBR. The only failures which could affect the RCS behavior during this interval are (1) a loss of normal AC, (2) a failure of the pressurizer level control system, (3) a failure of the pressurizer pressure control system, and (4) a failure of the reactor regulating system. The loss

of normal AC power, which is assumed, results in loss of power to the reactor coolant pumps, the condensate pumps, the circulating water pumps, the pressurizer pressure and level control systems, the reactor regulating system, and the feedwater control system.

Loss of function of the condensate and circulating water pumps and the feedwater control system initially affect only the secondary system and, thus, do not affect DNBR in the first one to four seconds of the transient. Loss of power to the reactor regulating system and pressurizer level and pressure control systems renders those systems inoperable. This inoperability will have no significant impact on DNBR during the first one to four seconds. Loss of power to the reactor coolant pumps is the only potentially significant failure with regard to DNBR which results from a loss of AC. However, as a result of a three second delay between the time of turbine trip and the time of loss of offsite power, there is no effect on minimum DNBR.

Failure of the pressurizer level control, pressure control, or reactor regulating systems cannot appreciably affect any of the four factors which determine DNBR during the first one to four seconds of the event. Thus, none of the single failures listed in Table 15.0-6 will result in a more adverse transient minimum DNBR than that predicted for the single reactor coolant pump rotor seizure event.

The assumed loss of AC renders the steam bypass control system inoperable as a result of the loss of the circulating water pumps. This results in the secondary system energy being released to the atmosphere by the secondary safety valves (prior to operator action) and the Atmospheric Dump Valves (ADV) after operator action is assumed. Operator action is assumed at 1800 seconds into the transient. At this time the operator begins a controlled cooldown of the plant.

A single active failure of an ADV to close is assumed at 1800 seconds. The stuck open ADV causes the eventual dryout of the affected steam generator which results in all of the iodine contained in this steam generator being released to atmosphere. Thus, this failure in combination with the loss of offsite power maximizes the radiological consequences of the single reactor coolant pump rotor seizure event. None of the other single failures listed in Table 15.0-6 in combination with a loss of AC will yield more severe radiological consequences.

15.3.3.2 Sequence of Events and System Operation

Table 15.3.3-1 presents a chronological list and time of system actions which occur following the single reactor coolant pump rotor seizure event. Refer to Table 15.3.3-1 while reading this and the following section. The success paths referenced are those given on the Sequence of Events Diagram (SED), Figure 15.3.3-1. This figure, together with Figure 15.0-1, which contains a glossary of SED symbols and acronyms, may be used to trace the actuation and interaction of the systems used to mitigate the consequences of this event.

The timings in Table 15.3.3-1 may be used to determine when, after event initiation, each action occurs.

The loss of offsite power is assumed to occur due to grid instability. A three second delay between the time of turbine trip and the time of loss of offsite power is conservatively assumed in the analysis, based on the discussion presented in the report submitted via Combustion Engineering Letter LD-82-040, A. E. Scherer to D. G. Eisenhut, dated March 31, 1982.

The sequence presented demonstrates that the operator can cool the plant down to cold shutdown during the event. If offsite power can be restored, then the operator may elect instead to stabilize the plant at a mode other than cold shutdown. All actions required to stabilize the plant and perform the required repairs are not described here.

The sequence of events and systems operations described below represents the way in which the plant was assumed to respond to the event initiator. Many plant responses are possible. However, certain responses are limiting with respect to the acceptance guidelines for this section. Of the limiting responses, the most likely one to be followed was selected.

Table 15.3.3-2 contains a matrix which describes the extent to which normally operating plant systems are assumed to function during the transient.

Table 15.3.3-3 contains a matrix which describes the extent to which safety systems are assumed to function during the transient.

The success paths in the sequence of events diagram, Figure 15.3.3-1, are as follows:

Reactivity Control:

Following seizure of a reactor coolant pump shaft, the core flow rapidly decreases to the value that would occur with only three reactor coolant pumps operating. The rapid reduction in primary coolant flow rate causes an increase in the average coolant temperature in the core, a corresponding reduction in the margin to DNB, and an increase in the primary system pressure. A low DNBR reactor trip is generated by the Core Protection Calculators. The trip conservatively assumes the largest possible delay time for the sensor delay, calculation period, CEDM dead time, and the CEDM coil delay time (see Chapter 7). The CEAs begin to drop into the core at 1.25 seconds inserting negative reactivity.

Prior to initiating or during manual cooldown the operator adjusts the boron concentration to insure that a proper negative reactivity shutdown margin is achieved. This is accomplished by using the HPSI pumps which also replace RCS volume shrinkage. The operator may also borate using the charging pumps by manually loading them on the diesel generators and then aligning them to the Refueling Water Tank (RWT), the source of borated water.

Reactor Heat Removal:

The reactor heat removal takes place by means of natural circulation in the reactor coolant system following the coastdown of the undamaged reactor coolant pumps. The steam generator provides primary to secondary heat transfer.

The Shutdown Cooling System (SCS) is manually actuated when the RCS temperature and pressure have been reduced to the shutdown cooling entry conditions of 350°F and 400 psia, respectively. This system provides sufficient cooling to bring the RCS to cold shutdown.

Secondary System Integrity:

The CEDM bus undervoltage relays, sensing the interruption of power on the CEDM power supply buses, generate a turbine trip signal. This results in closure of the turbine stop valves. The external grid which the plant is feeding is assumed to collapse as a result of turbine/generator trip. The loss of offsite power causes a loss of power to the start-up transformers which prevents the plant electrical loads from being transferred to them from the unit auxiliary transformers. Therefore, the onsite loads will lose power and the plant experiences a simultaneous loss of feedwater flow, condenser inoperability, and a coastdown of all reactor coolant pumps. The pressure in both steam generators increases resulting in the opening of the Main Steam Safety Valves (MSSVs), which prevents secondary pressure from exceeding safety limits. The MSSVs close when the secondary pressure drops. Water level in each of the steam generators begins decreasing immediately after the loss of main feedwater flow and an emergency feedwater actuation signal is generated on low steam generator water level. The Emergency Feedwater Actuation System (EFAS) setpoint is first reached in the steam generator in the unaffected loop. This leads to the start-up of the emergency feedwater pumps. The primary source of the emergency feedwater is the condensate storage tank. The capacity of the storage tank is 300,000 gallons, which is sufficient feedwater to maintain the plant at hot standby for 8 hours. The condensate storage tank is provided with an atmospheric vent to maintain atmospheric pressure inside the tank. The maximum condensate radioactivity concentration is 0.1 μCi/lbm (2.2×10^{-4} μCi/gm) dose equivalent I-131.

After 30 minutes the operator initiates cooldown of the RCS by using the atmospheric dump valves and the Auxiliary Feedwater System (AFWS). Once the dump valves are opened, one valve is assumed to remain stuck open. This results in the eventual generation of a Main Steam Isolation Signal (MSIS) on low steam generator pressure. Once the Main Steam Isolation Valves (MSIVs) are closed, this prevents further blowdown of the unaffected steam generator. Auxiliary feedwater is automatically terminated to the affected steam generator as a result of a high steam generator differential pressure signal. The affected generator is then allowed to dry out. Cooldown is continued by the operator by utilizing the atmospheric dump valves of the unaffected steam generator together with the Auxiliary Feedwater System. This process

continues until shutdown cooling entry conditions are reached. The AFWS may be a separate system or may be one emergency feedwater pump designated as "auxiliary" and intended for normal startup and shutdown of the plant. The operator may let the ESFAS regulate the feedwater flow by issuing and withdrawing EFAS-1 and/or EFAS-2 signals down to cold shutdown entry conditions. See applicant's FSAR for details of the Emergency and/or Auxiliary Feedwater Systems.

Primary System Integrity:

The pressurizer assists in the control of the RCS pressure and volume changes during the transient by compensating for the initial expansion of the RCS fluid. The combination of the loss of primary system heat sink (turbine stop valves close) with the reduction of reactor coolant flow results in an increase in RCS pressure.

During cooldown, the operator may control RCS pressure and pressurizer level by turning on the HPSI pumps and throttling the HPSI discharge valves to control the rate of change of RCS pressure. The operator may also control RCS pressure and pressurizer level via manual actuation and control of the charging pumps and related auxiliary spray. When the RCS pressure has been reduced to approximately 650 psia, the operator will vent or drain the safety injection tanks to reduce their pressure and will isolate them.

Restoration of AC Power:

A low voltage on the 4.16 kV safety buses generates an undervoltage signal which starts the diesel generators. The non-safety buses are automatically separated from the safety buses and all loads are shed (except 4890 V load centers). After each diesel generator set has attained operating voltage and frequency, its output breaker closes connecting it to its safety bus. ESF equipment is then loaded in sequence on to this bus.

Radioactive Effluent Control:

Containment Isolation Actuation Signal (CIAS) isolates various systems to reduce or terminate radioactive releases. CIAS actuates primary, secondary, and containment isolation equipment. Other actions may be initiated by BOP systems. See Applicant's FSAR for details.

Spent Fuel Heat Removal:

Spent Fuel Pool (SFP) cooling is terminated on the loss of normal power to the ESF loads. Spent fuel heat removal is continuously accomplished by utilizing the heat capacity of the SFP water. Pool cooling is restored by manually loading the SFP cooling pumps onto the diesel generators and by aligning the SFP heat exchangers to receive essential cooling water.

15.3.3.3 Analysis of Effects and Consequences

15.3.3.3.1 Core and System Performance

A. Mathematical Model

The NSSS response to a single reactor coolant pump rotor seizure with loss of offsite power resulting from turbine trip was simulated using the CESEC-III computer program described in Reference 27 of Section 15.0. The initial DNBR was calculated using the TORC computer code (see Section 15.0.3) which uses the CE-1 CHF correlation described in Reference 19 of Section 15.0.

B. Input and Parameters and Initial Conditions

The ranges of initial conditions considered are given in Section 15.0. Table 15.3.3-4 gives the initial conditions used in this analysis. The rationale for selecting the values of the initial conditions which have a first order effect on the analysis follows. Using the highest core power maximizes the RCS heatup, which is the driving force of the secondary steam release. A high core inlet temperature was chosen to minimize the degree of subcooling in the core. This also results in higher steam generator pressures and, thus, quicker opening of the secondary safety valves, which is more adverse from a radiological standpoint. The reactor coolant flow was chosen to be its minimum value of 146.1×10^6 lbm/hr. This low flow was chosen in combination with the other conditions mentioned above since this will allow operation with a low radial peaking factor. The use of a low radial peaking factor maximizes the amount of fuel pins which may experience DNB. The primary system pressure was chosen to be compatible with the other initial conditions. The most positive moderator temperature coefficient and the minimum available scram CEA worth tend to maximize the heat flux after a reactor trip occurs, increasing the RCS heat-up. The operator initiation of plant cooldown at 30 minutes with the subsequent failure of an atmospheric dump valve to close maximizes the offsite doses.

During this event two sources of radioactivity contribute to the offsite doses: the initial activity in the steam generator and the activity associated with the assumed one gallon per minute steam generator tube leak. The initial secondary activity is assumed to be at 0.1 μCi/gm dose equivalent I-131. The initial activity assumed to be present in the reactor coolant leaking through the steam generator tubes is 4.6 μCi/gm (see Table 15.3-5).

C. Results

The dynamic behavior of important NSSS parameters following a single reactor coolant pump rotor seizure with a loss of offsite power is presented on Figures 15.3.3-2 to 15.3.3-10. Table 15.3.3-1 summarizes the significant results of the event. Refer to Table 15.3.3-1 while reading this section.

The single reactor coolant pump rotor seizure event results in a flow coast down in the affected loop, a consequent reduction in flow through the core, an increase in the average coolant temperature in the core, a corresponding reduction in the margin to DNB, and an increase in the primary system pressure. A low DNBR reactor trip is generated by the core protection calculators. The reactor trip causes a turbine trip signal to occur. The CEAs begin to drop into the core at 1.25 seconds. At this time the turbine/generator trips. Three seconds later the loss of offsite power occurs. The remaining RCPs do not begin their normal coastdown until after the loss of offsite power. However, there is a slight decrease in RCP flow during the three seconds immediately after turbine trip and prior to the loss of offsite power due to decreasing pump speed caused by frequency degradation (approximately 1 Hertz/second) of the electrical grid. The loss of offsite power also causes a loss of main feedwater and condenser inoperability. The turbine trip with the SBCS and the condenser unavailable leads to a rapid buildup in secondary system pressure and temperature. This increase in pressure is shown in Figure 15.2.2-8. The opening of the MSSVs limits this pressure increase. The maximum secondary system pressure is 1347 psia which is less than 110% of design pressure.

The increasing temperature of the secondary system leads to a reduction of the primary to secondary heat transfer. Concurrently, the failed reactor coolant pump and the three reactor coolant pumps coasting down (Figure 25.3.3-7) result in RCS flow which further reduces the heat transfer capability of the RCS. This decrease in heat removal from the RCS leads to an increase in the core coolant temperatures as shown in Figure 15.3.3-5. The core coolant temperatures peak shortly after the time of reactor trip.

The increase in RCS temperature leads to an increase in RCS pressure, as shown in Figure 15.3.3-4, caused by the thermal expansion of the RCS fluid. The RCS pressure reaches a maximum value of 2387 psia at 4.2 seconds which is less than 110% of design pressure. After this time, the RCS pressure decreases rapidly due to the declining core heat flux (see Figure 25.5.5-3), in combination with the opening of the MSSVs. Opening of the MSSVs limits the peak temperature and pressure of the secondary system. The MSSVs cycle until the emergency feedwater begins entering the steam generators. Emergency feedwater begins entering the steam generator in the unaffected loop at 263 seconds, thus, enhancing the RCS cooldown and the subsequent reduction in pressure.

During the first few seconds of the transient, the combination of decreasing flow rate, and increasing RCS temperatures results in a decrease in the fuel pins' DNBR. The transient minimum DNBR of 0.967 occurs at 1.4 seconds as indicated in Table 15.3.3-1. Figure 15.3.3-9 shows the variation of the minimum DNBR with time. The negative CEA reactivity inserted after reactor trip causes a rapid power and heat flux decrease which causes DNBR to increase again. For this event no more than 0.85 percent of the fuel pins are calculated to experience DNB. All fuel pins which experience DNB are conservatively assumed to fail.

The offsite doses for this event result from steam released through the Main Steam Safety Valves (MSSVs) and Atmospheric Dump Valves (ADVs).

At 30 minutes, the operator is assumed to use the ADVs to begin cooldown. At this time, one atmospheric dump valve is assumed to stick open. This leads to the generation of an MSIS and eventual termination of emergency feedwater flow to the affected steam generator. Once the affected steam generator has blown dry, the operator may continue a controlled cooldown via the atmospheric dump valves of the unaffected steam generator. Additionally, the operator will be feeding auxiliary feedwater flow to the unaffected steam generator. Table 15.3.3-1 shows the integrated steam release from the MSSVs and the ADVs. The radiological release produced by the transient results in a 277 rem two hour thyroid inhalation dose at the exclusion area boundary. The two hour thyroid inhalation dose at the exclusion area boundary is shown in Table 15.3.3-7.

15.3.3.3.2 Radiological Consequences

A. Physical Model

To evaluate the consequences of the single reactor coolant pump rotor seizure with a loss of offsite power event, it is assumed that the condenser is not available for the entirety of the transient. For the first thirty minutes of the event, the cooldown is performed via the main steam safety valves. Afterwards, an atmospheric dump valve is assumed to stick open once the valve(s) are actuated by the operator in an attempt to initiate a controlled cooldown. After MSIS occurs and the affected generator has blown dry, the operator may then proceed to initiate a controlled cooldown via the atmospheric dump valves of the unaffected steam generator and the auxiliary feedwater system.

B. Assumptions, Parameters, and Calculational Methods

The major assumptions, parameters, and calculational methods used to evaluate the radiological consequences of the single reactor coolant pump rotor seizure are presented in Tables 15.3.3-5 and 15.3.3-6. Additional clarification is provided as follows:

1. The Reactor Coolant System (RCS) equilibrium activity is based on long term operation at 108% of the ultimate core power level of 3800 MWt (3800 MWt x 1.08 = 4100 MWt) with 1% failed fuel. Refer to Table 11.1.1-2 for the isotopic distribution of RCS activity.

   The RCS activity was calculated to determine the total amount of activity transmitted into the secondary system during the duration of the accident due to a 1 gal/min primary to secondary leak. The primary to secondary leakage of 1 gal/min (technical specification limit) is assumed to continue to the steam generators for the entire event. The fluid density is assumed to be constant at its initial value of 45 lbm/ft^3. The activity in the fuel clad gap is 10% of the iodines and 10% of the noble

gases accumulated in the fuel at the end of core life, assuming continuous fuel power operation. All of the activity in the fuel gap for fuel rods that are calculated to experience DNB is assumed to be uniformly mixed with the reactor coolant. This assumption is consistent with Regulatory Guide 1.77.

2. The steam generator equilibrium activity is assumed to be 0.1 μCi/gm dose equivalent Iodine-131 (I-131) prior to the accident. This is the technical specification limit for steam generator activity.

3. Offsite power is not available. At 1800 seconds the operator attempts to take control of the plant using the atmospheric dump valves. One atmospheric dump valve is assumed to stick open at this time for the remainder of the transient.

4. Credit is assumed for emergency feedwater flow. For the fluid leaked from primary to secondary, iodine is assumed to released to the atmosphere with a partition coefficient of 1.0.

5. No credit for radioactive decay in transit to dose point is assumed.

6. The atmospheric dispersion factors used in this analysis, which are based on meteorological conditions assumed present during the course of the accident, are calculated according to the model described in subsection 2.3.4. The 5% level X/Q presented in Table 2.3-1 was used.

7. The mathematical model used to analyze the activity released during the course of the accident is described in Section 15.0.

8. Table 15.3.3-6 presents the integrated mass release from the secondary safety valves and the total primary to secondary leakage.

9. Calculated secondary mass releases are presented in Table 15.3.3-5.

C. Identification of Uncertainties and Conservatisms in the Evaluation of the Results

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of the single reactor coolant pump rotor seizure with a loss of offsite power are as follows:

1. The RCS equilibrium activity is based on 1% failed fuel, which is greater by a factor of two to eight than that normally observed in past PWR operation.

2. The steam generator equilibrium activity for the affected steam generator is assumed to be equal to the technical specification limit (0.1 μCi/gm dose equivalent I-131). This specific activity is greater by a factor of approximately 1300 than the normal expected steam generator activity (refer to Table 11.1.8-1).

3. The primary to secondary leakage of 1 gal/min (technical specification limit) is conservative because operation with a 1 gal/min primary to secondary leak is not expected (the expected leakage rate is equal to 20 gal/day).

4. The meteorological conditions assumed to be present at the site during the course of the accident are based on 5% level X/Qs. Meteorological conditions will be less severe 95% of the time. This results in the poorest values of atmospheric dispersion calculated for the EAB or LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the EAB or LPZ outer boundary.

5. The assumption of no operator action for 1800 seconds (30 minutes) is a conservative assumption.

15.3.3.4 Conclusions

The maximum RCS and steam generator pressures due to a single reactor coolant pump rotor seizure in combination with loss of offsite power following generator trip event remain less than 110% of their design values. Only a small fraction of the fuel pins experience DNB and are conservatively assumed to fail. The two hour thyroid dose is within 10CFR100 guidelines.

Table 15.3.3-1

(Sheet 1 of 2)

SEQUENCE OF EVENTS FOR THE SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

Time (sec.)	Event	Setpoint or Value	Total Integrated Safety Valve Flow (lbm)	Success Path
0.0	Seizure of a Single Reactor Coolant Pump	—	—	—
0.76	Low DNBR Trip Signal Generated, projected	1.19	—	Reactivity Control
1.25	CEAs Begin to Drop Into the Core	—	—	Reactivity Control
1.25	Turbine Trip/Generator Trip	—	—	—
1.4	Minimum Transient DNBR	0.967	—	—
4.1	Main Steam Safety Valves Open, Unaffected Loop, psia	1280	—	Secondary System Integrity
4.2	Maximum RCS Pressure, psia	2387	—	—
4.25	Loss of Offsite Power Occurs	—	—	—
4.5	Main Steam Safety Valves Open, Affected Loop, psia	1280	—	Secondary System Integrity
6.8	Maximum Steam Generator Pressure, Unaffected Loop, psia	1347	3,492	—
7.4	Maximum Steam Generator Pressure, Affected Loop, psia	1340	5,451	—

Table 15.3.3-1 (Continued)

(Sheet 2 of 2)

SEQUENCE OF EVENTS FOR THE SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

Time (Sec.)	Event	Setpoint or Value	Total Integrated Safety Valve Flow (lbm)	Success Path
218	Low Water Level EFAS Setpoint Reached in the Steam Generator, Unaffected Loop, percent of wide range	20	85,679	Secondary System Integrity
263	Emergency Feedwater Begins Entering Steam Generator, Unaffected Loop, lbm/sec	119	91,407	Secondary System Integrity
697	Low Water Level EFAS Setpoint Reached in the Steam Generator, Affected Loop, percent of wide range	20	115,189	Secondary System Integrity
	Emergency Feedwater Begins Entering the Steam Generator, Affected Loop, lbm/sec	119		
821	Steam Generator Safety Valves Close, Affected and Unaffected Loop, psia	1218	120,398	Secondary System Integrity
1800	Atmospheric Dump Valves Opened to Initiate Plant Cooldown, °F/hour. One Atmospheric Dump Valve Sticks Open	-100.0	120,398	Secondary System Integrity
7200	Total Steam Release to Atmosphere, lbm	—	1,128,293	—

TABLE 15.3.3-2 (sheet 1 of 2)

DISPOSITION OF NORMALLY OPERATING SYSTEMS
FOR THE SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE
WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

SYSTEM	NORMAL AUTOMATIC MODE THROUGH-OUT TRANSIENT	MANUAL MODE THROUGH-OUT TRANSIENT	NORMAL AUTOMATIC MODE INOPERATIVE ON LOSS OF A.C.	MANUAL MODE INOPERATIVE ON LOSS OF A.C.	SINGLE-FAILURE ASSUMED WITHIN SYSTEM	ASSOCIATED NOTES
1. Main Feedwater Control System			✓			
2. Main Feedwater Pump Turbine Control System*			✓			
3. Turbine-Generator Control System*	✓					
4. Steam Bypass Control System				✓		
5. Pressurizer Pressure Control System				✓		
6. Pressurizer Level Control System				✓		
7. Control Element Drive Mechanism Control System	✓					
8. Reactor Regulating System				✓		
9. Core Operating Limit Supervisory System			✓			
10. Reactor Coolant Pumps				✓		1
11. Chemical and Volume Control system				✓		
12. Secondary Chemistry Control System*				✓		
13. Condenser Evacuation System*				✓		
14. Turbine Gland Sealing System*				✓		
15. Nuclear Cooling Water System*				✓		
16. Turbine Cooling Water System*				✓		
17. Plant Cooling Water System*				✓		
18. Condensate Storage Facilities*				✓		
19. Circulating Water System*				✓		
20. Spent Fuel Pool Cooling and Clean-Up System*				✓		
21. Non-Class 1E (Non-ESF) A.C. Power*				✓		
22. Class 1E (ESF) A.C. Power*	✓					
23. Non-Class 1E D.C. Power *		✓				
24. Class 1E D.C. Power*		✓				

*Balance-of-Plant Systems

TABLE 15.3.3-2 (Cont'd.) (Sheet 2 of 2)

DISPOSITION OF NORMALLY OPERATING SYSTEMS
FOR THE SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE
WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

SYSTEM	NORMAL AUTOMATIC MODE THROUGH-OUT TRANSIENT	MANUAL MODE THROUGH-OUT TRANSIENT	NORMAL AUTOMATIC MODE INOPERATIVE ON LOSS OF A.C.	MANUAL MODE INOPERATIVE ON LOSS OF A.C.	SINGLE-FAILURE ASSUMED WITHIN SYSTEM	ASSOCIATED NOTES
23. Non-Class 1E D.C. Power*		✓				
24. Class 1E D.C. Power*		✓				

Note:1 A Failure in this System is the event initiator

*Balance-of-Plant Systems

TABLE 15.3.3-3

UTILIZATION OF SAFETY SYSTEMS
FOR THE SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE
WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

SYSTEM	ACTUATED AND REQUIRED	ACTUATED BUT NOT REQUIRED	SAFETY GRADE BACK-UP TO NON-SAFETY GRADE SYSTEM	SINGLE-FAILURE ASSUMED WITHIN SYSTEM (SEE NOTES)	ASSOCIATED NOTES
1. Reactor Protection System	✓				
2. DNBR/LPD Calculator	✓				
3. Engineered Safety Features Actuation Systems	✓		✓		
4. Supplementary Protection System					1
5. Reactor Trip Switch Gear	✓				
6. Main Steam Safety Valves*	✓				
7. Primary Safety Valves					
8. Main Steam Isolation System*	✓		✓		
9. Emergency Feedwater System*	✓		✓		
10. Safety Injection System	✓				
11. Shutdown Cooling System	✓				2
12. Atmospheric Dump Valve System*	✓			✓	2,3
13. Containment Isolation System*		✓			
14. Containment Spray System*					
15. Iodine Removal System*					
16. Containment Combustible Gas Control System*					
17. Diesel Generators and Support Systems*	✓				
18. Component (Essential) Cooling Water System*	✓				
19. Station Service Water System*	✓				

Notes:

1. Safety grade back-up to a safety grade system.
2. Manually actuated during normal cooldown
3. One atmospheric dump valve is assumed to stick open.

*Balance-of-Plant Systems -

Table 15.3.3-4

ASSUMED INITIAL CONDITIONS FOR THE SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

Parameter	Value
Core Power Level, MWt	3876
Core Inlet Coolant Temperature, °F	580
Reactor Coolant System Pressure, psia	2257
Steam Generator Pressure, psia	1221
Core Mass Flow, 10^6 lbm/hr	146.1
Maximum Radial Power Peaking Factor	1.40
Maximum Axial Power Peak	1.47
Core Minimum DNBR	1.48
Doppler Coefficient Multiplier	0.85
CEA Worth on Trip, 10^{-2} $\Delta\rho$ (most reactive CEA stuck)	-10.0
Moderator Temperature Coefficient	0.0

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

Parameters	Value
A. Data and Assumptions Used to Evaluate the Radioactive Source Term	
a. Power Level, Mwt	4200
b. Burnup	2 year
c. Percent of Fuel Calculated to Experience DNB, %	0.85
d. Reactor Coolant Activity Before Event (based on 4100 MWt)	4.6 μCi/gm Table 11.1.1-2
e. Secondary System Activity Before Event	Section 15.0.4
f. Primary System Liquid Inventory, lbm	525,600
g. Steam Generator Inventory	
- Liquid, lbm per steam generator	167,075
- Steam, lbm per steam generator	14,863
B. Data and Assumptions Used to Estimate Activity Released from the Secondary System	
a. Primary to Secondary Leak Rate, gpm	1.0 (total)
b. Total Mass Release Through the Main Steam Safety Valves and Atmospheric Dump Valves (2 hours)	1,128,293

TABLE 14.3.3-5 (Continued)
(Sheet 2 of 3)

PARAMETERS USED IN EVALUATING THE
RADIOLOGICAL CONSEQUENCES OF A SINGLE REACTOR COOLANT PUMP ROTOR
SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

Parameters	Value
C. Reactor Coolant System Activity After Event, Ci	
Isotope	
I-131	1.039 (+5)
I-132	1.519 (+5)
I-133	2.097 (+5)
I-134	2.265 (+5)
I-135	1.954 (+5)
Kr-85M	2.621 (+4)
Kr-85	8.315 (+2)
Kr-87	4.806 (+4)
Kr-88	6.867 (+4)
Xe-131M	7.320 (+2)
Xe-133	2.105 (+5)
Xe-135	3.767 (+4)
Xe-138	1.679 (+5)
d. Percent of Core Fission Products Assumed Release to Reactor Coolant	Refer to Section 15.3.3.3.2B
e. Iodine Partition Coefficient for the Unaffected Steam Generator	100.0
f. Iodine Partition Coefficient for the Affected Steam Generator	1.0
g. Credit for Radioactive Decay in Transit to Dose Point	No
h. Loss of Offsite Power	Yes
D. Dispersion Data	
1. Distance to Exclusion Area Boundary, m	500

TABLE 15.3.3-6

SECONDARY SYSTEM MASS RELEASE
TO THE ATMOSPHERE FOR THE SINGLE REACTOR COOLANT PUMP ROTOR
SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP EVENT

Time (sec.)	Integrated Safety Valve Flow (lbm)	Integrated Primary to Secondary Leakage (gallons)
0.0	0.0	0.00
2.0	0.0	0.03
3.0	0.0	0.05
5.0	480	0.08
10.0	13,884	0.17
20.0	36,677	0.33
40.0	53,354	0.67
60.0	54,812	1.00
80.0	62,536	1.33
100.0	66,282	1.67
150.0	73,248	2.50
200.0	83,047	3.33
300.0	95,560	5.00
500.0	108,710	8.33
821.0*	120.398	13.68
1800.0**	120,398	30.00

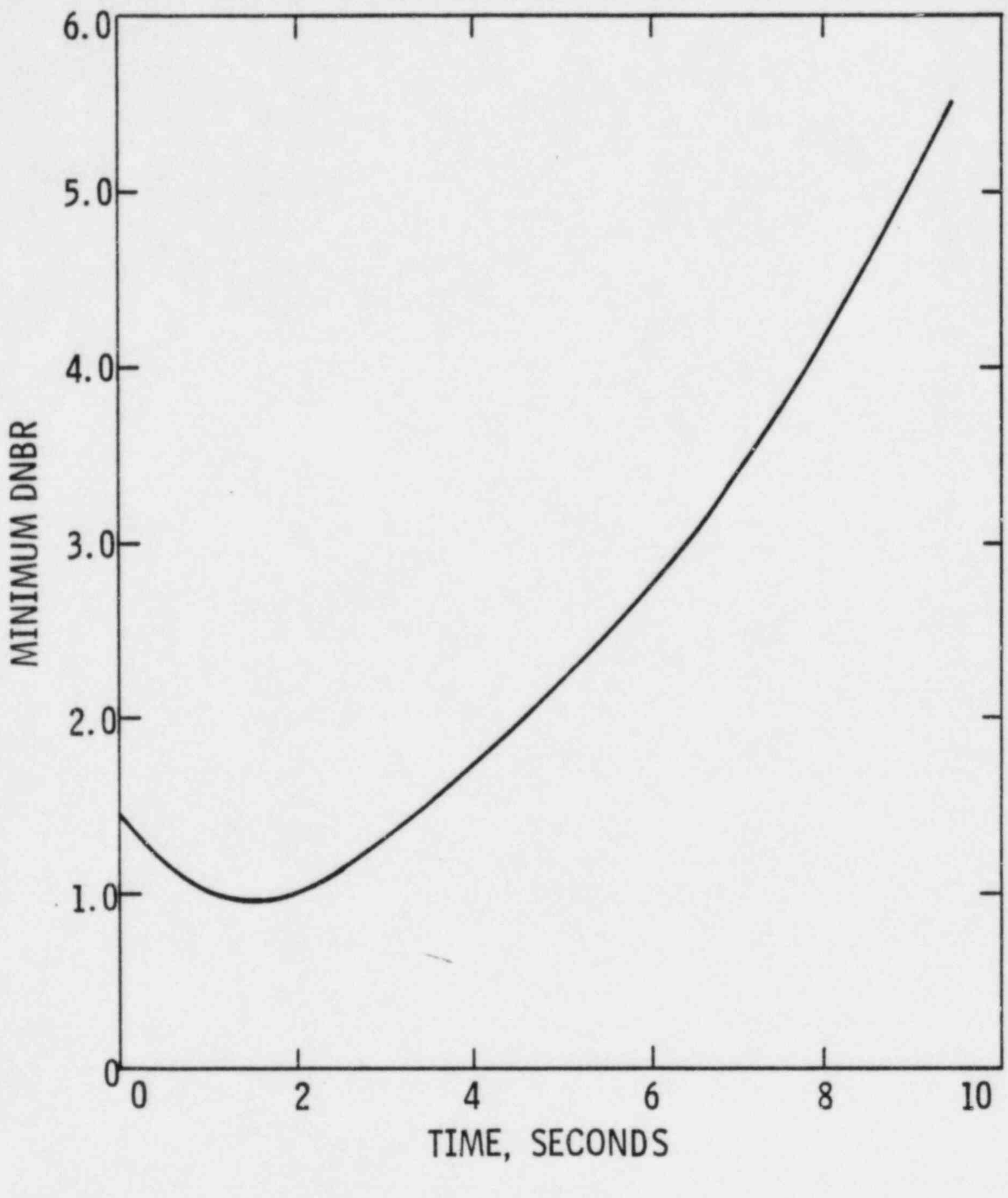
* Main steam safety valves close.

** Operator begins cooldown utilizing the atmospheric dump valves. One atmospheric dump valve is assumed to stick open.

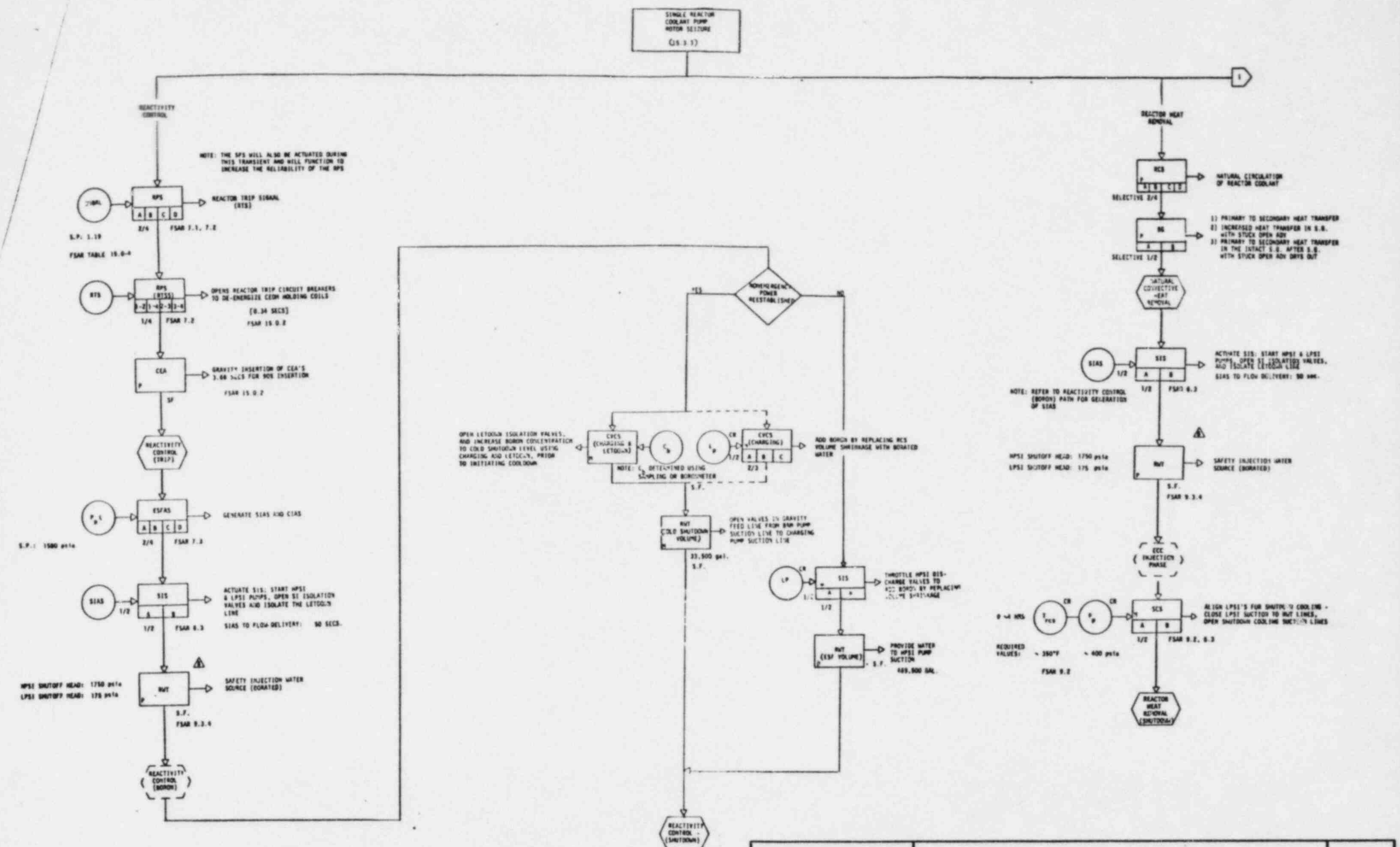
TABLE 15.3.3-7

RADIOLOGICAL CONSEQUENCES OF A POSTULATED SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

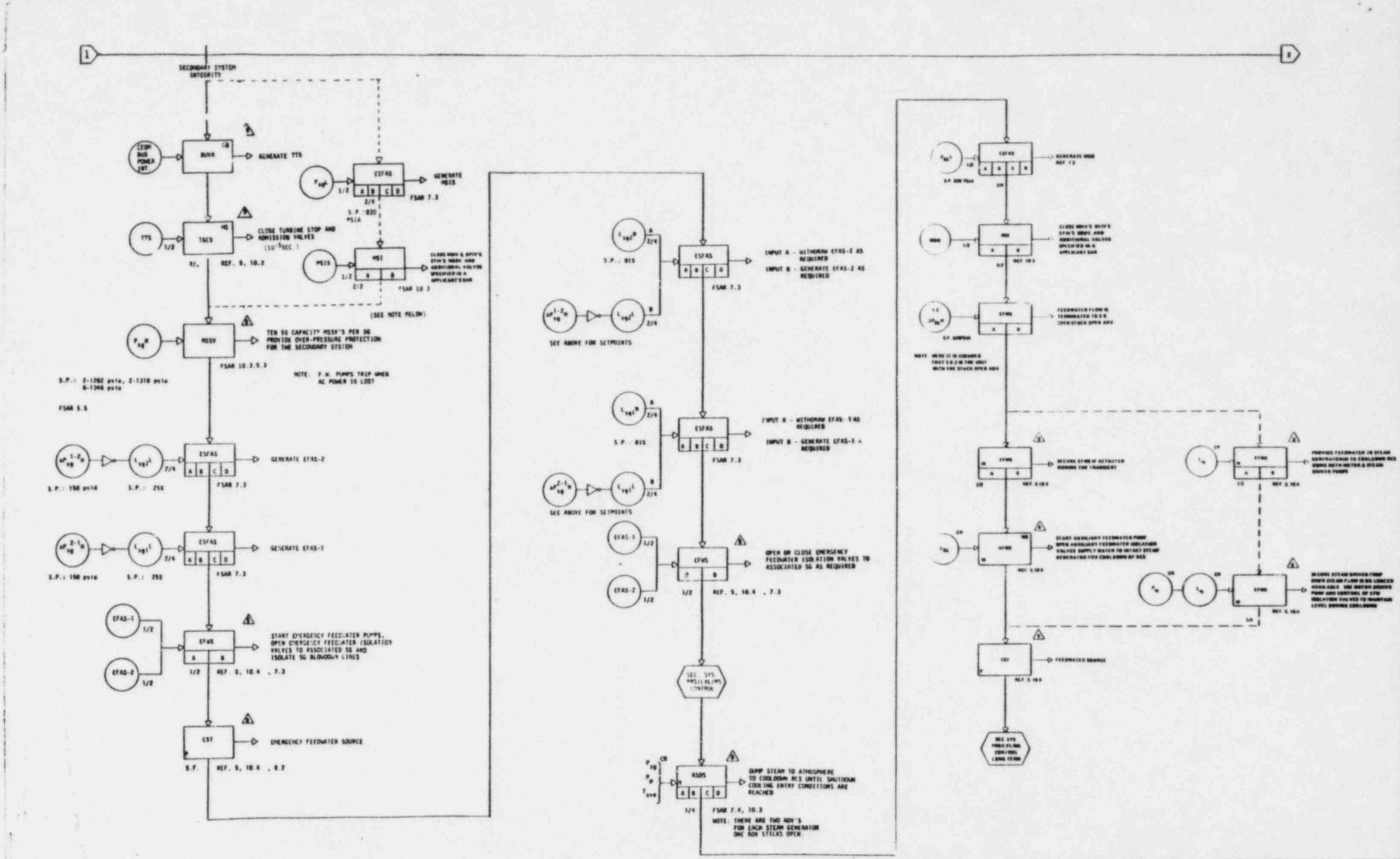
Result	From Secondary System Steam Releases
Exclusion Area Boundary Dose (0-2 hours), rem:	
Thyroid	277.0



C - E SYSTEM 80	SINGLE RCP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP MINIMUM DNBR vs TIME	Figure 15.3.3-9



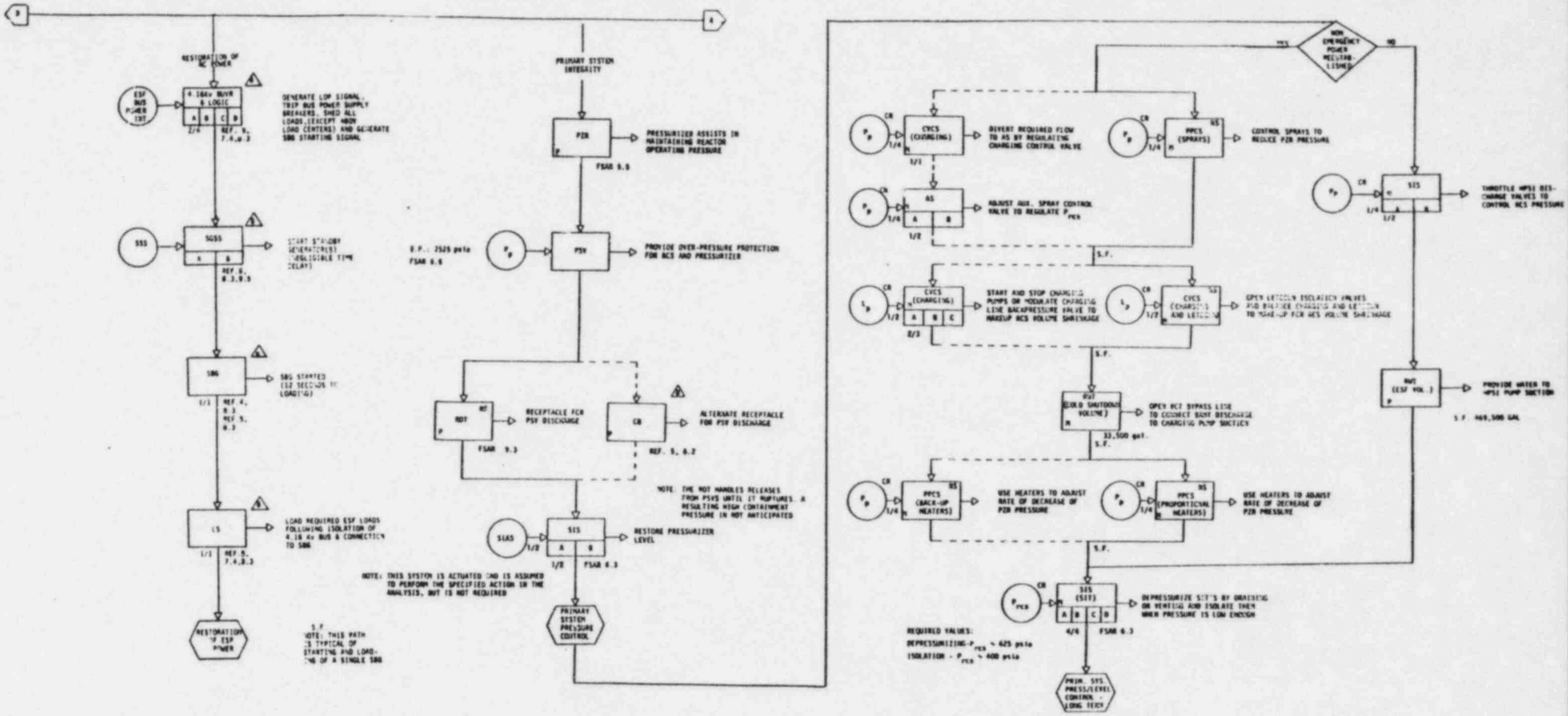
C - E SYSTEM 80	SEQUENCE OF EVENTS DIAGRAM FOR SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP	Figure 15.3.3-1A



C-E SYSTEM 80

SEQUENCE OF EVENTS DIAGRAM FOR SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

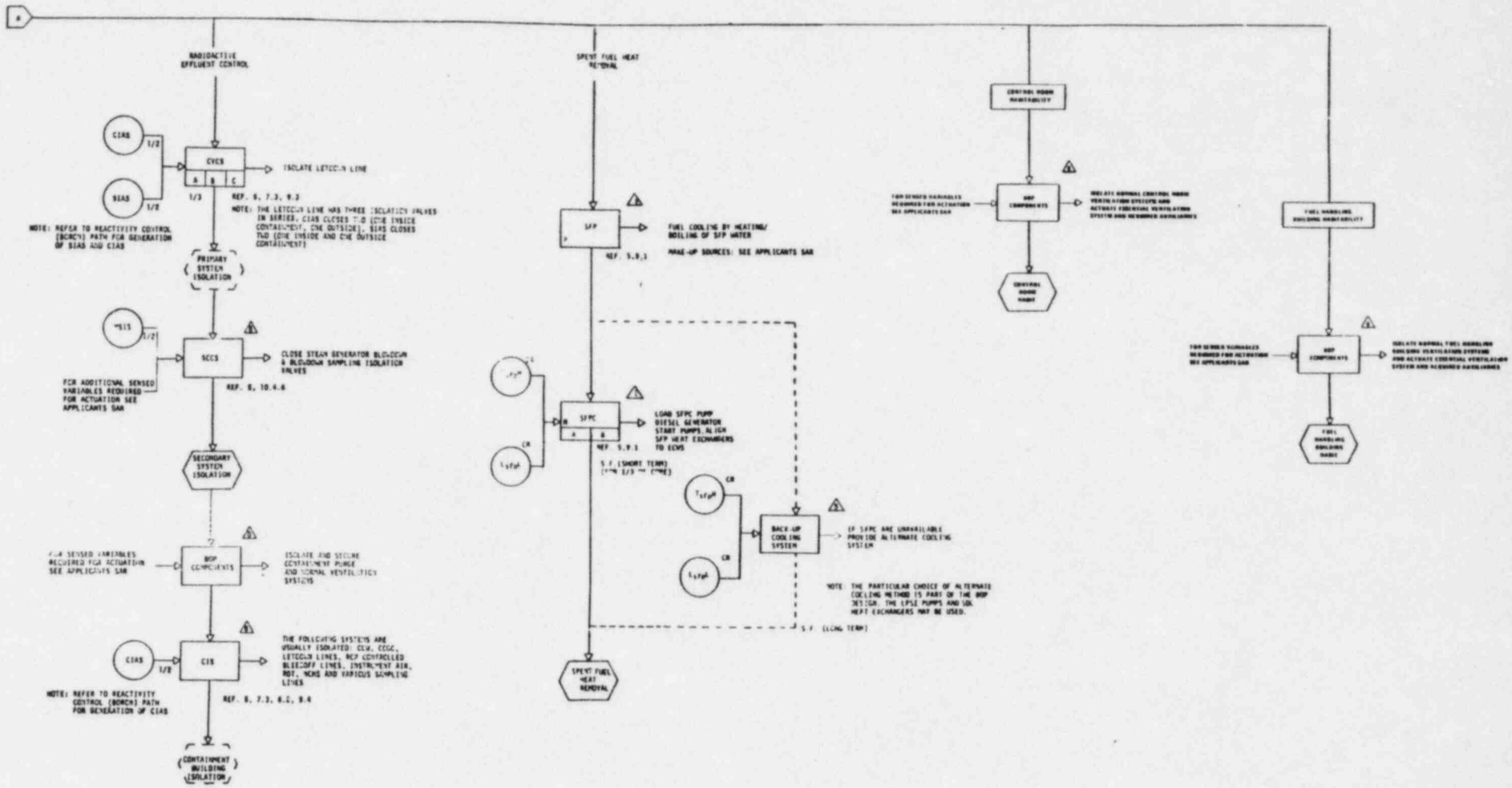
Figu
15.3.



C - E SYSTEM 80

SEQUENCE OF EVENTS DIAGRAM FOR SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

Figure 15.3.3-1C



C-E SYSTEM 80	SEQUENCE OF EVENTS DIAGRAM FOR SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP	Figure 15.3.3-1D

15.3.4 REACTOR COOLANT PUMP SHAFT BREAK WITH LOSS OF OFFSITE POWER

15.3.4.1 Identification of Event and Causes

A single reactor coolant pump sheared shaft could be caused by mechanical failure of the pump shaft. This is assumed to result from a manufacturing defect in the shaft. Loss of offsite power following turbine/generator trip may be caused by a complete loss of the external electrical grid triggered by the turbine/generator trip.

15.3.4.2 Sequence of Events and Systems Operation

The sequence of events and systems operations is similar to that for the reactor coolant pump rotor seizure event, Section 15.3.3. The difference is that for the shaft break event, the reactor is tripped on differential pressure across either steam generator, whereas, for the pump rotor seizure event, the reactor is tripped by the CPC on a low projected DNBR condition.

The flow coastdown for a Rotor Seizure (RS) event is faster than the coastdown for a Shaft Break (SB) event. For a shaft break, the rotor is still capable of rotating, thereby offering less resistance to flow during the rapid flow decrease. This results in a less severe coast down for the shaft break event than for the rotor seizure event. The SB trip time is 1.2 seconds; the RS trip time is 0.91 seconds. Despite the later trip time, the slower SB coastdown results in a higher minimum DNBR and less fuel failure for SB than for LR.

For both RS and SB, three seconds after turbine trip a Loss of Offsite Power (LOP) was assumed. Both RS and SB reach the same 3-pump asymptotic flow before their respective LOPs and do not result in decreasing DNBR after LOP. The RS plus LOP minimum DNBR is lower and fuel failure higher than those for SB plus LOP.

15.3.4.3 Analysis of Effects and Consequences

15.3.4.3.1 Core and System Performance

The analysis of effects and consequences for this event is similar to that for the reactor coolant pump rotor seizure event, Section 15.3.3. The SB coastdown is slower and trip is later than those of the rotor seizure event. The SB plus LOP event produces a higher minimum DNBR and less radiological release than RS plus LOP.

15.3.4.3.2 Radiological Consequences

The radiological consequences due to steam release from the secondary system would be less than the consequences of the RS event as described in Section 15.3.3. Thus, the two hour thyroid inhalation dose for the SB with loss of offsite power event is less than 277.0 rem.

15.3.4.4 Conclusion

The conclusion from the SB event is that this event would be no more adverse than the RS event. For both events, the total number of fuel pins calculated in DNB, and which are conservatively assumed to fail, is less than 0.85%. The resultant radiological consequences, which are given in Table 15.3.3-7, are within the guidelines of 10CFR100.