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October 24, 1985
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File No.: G9.17

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Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Loss of FW Transient; Decay Heat Curves

Dear Mr. Knighton:

The attachment enclosed provides changes to the South Texas Project (STP) FSAR Chapter 15 (Sections 15.0.9, 15.2.7, Table 15.2-1) and Chapter 10 (Section 10.4.9). These changes will be incorporated in the STP FSAR in a future amendment and reflect assumptions made in the evaluation of the Loss of Normal Feedwater Transient. These assumptions include the use of the 1979 version of ANSI/ANS 5.1-1979 for decay heat generation and that auxiliary feedwater is delivered by one auxiliary feedwater pump to one steam generator.

If you should have any questions on this matter, please contact Mr. M. E. Powell at (713) 993-1328.

Very truly yours,

M. R. Wisenburg
Manager, Nuclear Licensing

JSP/yd

Attachment: Revised FSAR Chapters 10 and 15

L1/NRC/W

B001
1/1

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Revised 9/25/85

of the postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Reference 15.0-9 discusses the quality assurance program which has been implemented to 43
assure that the NSSS will satisfactorily perform its assigned safety functions. The incorporation of these features in the NSSS, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0-6 will be available for mitigation of the events discussed in Chapter 15. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI N18.2-1973 is utilized. The design of safety-related systems (includ- 43
ing protection systems) is consistent with IEEE 379-1972 and Regulatory Guide 1.53 in the application of the single failure criterion.

In the analysis of the Chapter 15 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case. The pressurizer heaters are not assumed to be energized during any of the Chapter 15 events.

15.0.9 Residual Decay Heat

<u>Total Residual Heat</u>

Residual heat in a subcritical core is calculated for the loss-of-coolant accident (LOCA) per the requirements of Appendix K of 10CFR50.46 (Ref. 15.0-1), as described in References 15.0-2 and 15.0-3. These requirements include assuming infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the same models are used except that fission product decay energy is based on core average exposure at the end of the equilibrium cycle.

except where noted in the text,

15.0.10 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular very specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the RCS pipe rupture (Section 15.6), are summarized in their respective accident analyses sections. The codes used in the analyses of each transient have been listed in Table 15.0-2.

15.0.10.1 <u>FACTRAN</u>. FACTRAN calculates the transient temperature distribution in a cross section of a metal clad uranium dioxide fuel rod and the transient heat flux at the surface of the clad using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which exhibits the following features simultaneously:

1. A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.

2. Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation.

relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot standby condition. 43

4. The standby diesel generators, started on loss of voltage on the plant emergency buses, begin to supply plant vital loads.

A loss of normal feedwater is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

Reactor trip on low-low water level in any steam generator provides protection for a loss of normal feedwater.

The AFWS is started automatically as discussed in Section 15.2.6.1. The steam-driven auxiliary feedwater pump utilizes steam from the secondary system and exhausts to the atmosphere. The motor driven auxiliary feedwater pumps are supplied power from the standby diesel generators. The pumps take suction directly from the auxiliary feedwater storage tank for delivery to the steam generators. 43

Upon loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and removal of residual heat is maintained by natural circulation in the reactor coolant loops. The analysis presented in Section 15.2.6 demonstrates the natural circulation capability of the RCS.

A loss of normal feedwater caused by a loss of offsite power is the most limiting Condition II event in the decrease in secondary heat removal category. There , a full analysis of the system transient is presented below to show that following a loss of normal feedwater, the AFW system is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS or loss of water from the reactor core, and returning the plant to a safe condition.

15.2.7.2 Analysis of Effects and Consequences.

Method of Analysis

A detailed analysis using the LOFTRAN Code (Reference 15.2.3) is performed in order to obtain the plant transient following a loss of normal feedwater. The simulation describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generator and feedwater system. The digital program computes pertinent variables including the steam generator water level, pressurizer water level, and reactor coolant average temperature. 43

Assumptions made in the analysis are:

1. The plant is initially operating at 102 percent of the nominal NSSS design rating.

INSERT I

2. A conservative core residual heat generation is based upon long-term operation at the initial power level preceding the trip.

3. A heat transfer coefficient in the steam generator is associated with RCS natural circulation.

INSERT I

2. Core residual heat generation is based on the 1979 version of ANS 5.1 (Reference 15.2-6). ANSI/ANS 5.1-1979 is a conservative representation of the decay energy release rates.

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4. Reactor trip occurs on steam generator low-low water level. No credit is taken for immediate release of the control rod drive mechanisms caused by a loss of offsite power. 43

5. The worst single failure in the AFW occurs (auxiliary feedwater pump fails to start). 43

5 ~~6~~. Auxiliary feedwater is delivered by one ~~two~~ auxiliary feed water pump~~s~~ to one ~~two~~ steam generator~~s~~. This assumption is more severe than the single failure assumption of loss of one AFW pump.

6 ~~7~~. Secondary system steam relief is achieved through the steam generator safety valves.

7 ~~8~~. The initial reactor coolant average temperature is 4.5 ~~4.0~~°F lower than the nominal value since this assumption results in a greater expansion of the RCS water during the transient and, thus, in a higher water level in the pressurizer at the time of maximum insurge.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the RTS and ESF (e.g., the AFW) in removing long-term decay heat and preventing excessive heatup of the RCS with possible resultant RCS overpressurization or loss of RCS water.

As such, the assumptions used in this analysis are designed to minimize the energy removal capability of the system and to maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion, as noted in the assumptions listed above. 43

One such assumption is the loss of offsite power. This assumption results in coolant flow decay down to natural circulation conditions and a corresponding reduction in the steam generator heat transfer coefficient. Following a loss of offsite power, the first few seconds of a loss of normal feedwater transient will be virtually identical to the transient response (including DNBR and neutron flux versus time) presented in Section 15.3.2 for the complete loss of forced reactor coolant flow.

If offsite power were not lost for this incident, the reactor coolant flow would remain at its normal value, and the reactor would trip via the low-low steam generator water level trip. The DNBR never falls below the value at the start of the transient. The reactor coolant pumps may be manually tripped at some later time to reduce heat addition to the RCS and prevent filling the pressurizer. 43 43

An additional assumption made for the loss of normal feedwater evaluation is that the pressurizer power-operated relief valves (PORV's) are assumed to function normally. Operation of the ~~valves~~ PORV's maintains peak RCS pressure ~~at or~~ below the actuation setpoint (~~2350~~ 2500 psia) of the pressurizer safety valves throughout the transient.

If these valves were assumed not to function, the coolant system pressure during the transient would rise to the actuation point of the pressurizer safety valves ~~(2500 psia)~~. The increased RCS pressure, however, results in less expansion of the coolant and, hence, more margin to the point where water relief from the pressurizer would occur. Plant characteristics and initial conditions are further discussed in Section 15.0.3.

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A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-12. 2 Q211 .6

Plant systems and equipment which are available to mitigate the effects of a loss of normal feedwater accident are discussed in Section 15.0.8 and listed in Table 15.0-6. Normal reactor control systems are not required to function. Pressurizer power-operated relief valves are assumed to function in order to provide a more limiting transient, as described above. The RTS is required to function following a loss of normal feedwater as analyzed here. The AFW system is required to deliver a minimum auxiliary feedwater flow rate. In the long-term after automatic actuation of the AFW system, feedwater addition is manually controlled to maintain proper steam generator water level. No single active failure will prevent operation of any system required to function. A discussion of ATWT considerations is presented in Reference 15.2-2.

Results

Figures ~~15.2-9~~ 15.2-9A through 15.2-9D and 15.2-10 show the significant plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of the steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. Within one minute following the low-low water level signal, at least ~~two~~ one auxiliary feedwater train~~s~~ ~~are~~ is delivering flow automatically, reducing the rate of water level decrease. 43

The capacity of the auxiliary feedwater pump~~s~~ is such that the water level in the steam generator~~s~~ being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS relief or safety valves. From Figure~~s~~ ~~15.2-9 and~~ 15.2-10 it can be seen that at no time is the tubesheet uncovered in the steam generator~~s~~ receiving auxiliary feedwater flow. ~~and that at no time is there water relief from the pressurizer.~~ INSERT II.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown on Figures ~~15.2-9~~ 15.2-9A through 15.2-9D and 15.2-10, the plant approaches a stabilized condition following reactor trip and auxiliary feedwater initiation. Plant procedures may be followed to further cool down the plant.

15.2.7.3 Radiological Consequences. The steam release and resulting radiological consequences from this transient would be the same as that for the loss of offsite power, and, similarly, radiological consequences resulting from this transient are less severe than the steam line break accident.

15.2.7.4 Conclusions. Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves, and the water level in ~~all~~ the steam generator~~s~~ receiving auxiliary feedwater is maintained above the tubesheet~~s~~. The radiological consequences of this event are not limiting.

INSERT II

The analysis also indicates that at no time is there water relief from the pressurizer; Figure 15.2-9D shows that the peak water volume in the pressurizer is less than 2100 ft^3 which is the filled pressurizer volume.

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TABLE 15.2-1 (Cont'd)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)	
	Rods begin to drop	6.4	
	Minimum DNBR occurs	(1)	
	Initiation of steam release from steam generator safety valves	8.0	18
	Peak pressurizer pressure occurs	8.0	
Loss of Normal Feedwater Flow	Main feedwater flow stops	0.0	
	Low-low steam generator water level trip	~~22.9~~ 20.8	43
	Rods begin to drop	~~24.9~~ 22.8	
	Reactor coolant pumps begin to ~~cast~~ coast down(2)	~~24.9~~ 24.8	
	One ~~Two~~ steam generator~~s~~ begins to receive auxiliary ~~feed from two~~ feedwater from its associated auxiliary feedwater pump~~s~~	~~82.9~~ 80.8	43
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~~~2150~~ ~2800	

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TABLE 15.2-1 (Cont'd)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)	
	Peak water ~~level~~ volume in pressurizer occurs	~~2196.0~~ 2816.0	
Feedwater System Pipe Break			
1. With Offsite Power Available	Main feedline rupture occurs	10	
	Low-low steam generator water level reactor trip setpoint reached in affected steam generator	27	43
	Rods begin to drop	29	
	Auxiliary feedwater is delivered to intact steam generators	87	
	Low steamline pressure setpoint reached in affected steam generator	652	32 Q211.74
	All main steam isolation valves close	660	43
	Pressurizer power-operated relief valve setpoint reached	788	43
	Steam generator safety valve setpoint reached in intact steam generator receiving auxiliary feedwater	960	
	Pressurizer water relief begins	2544	

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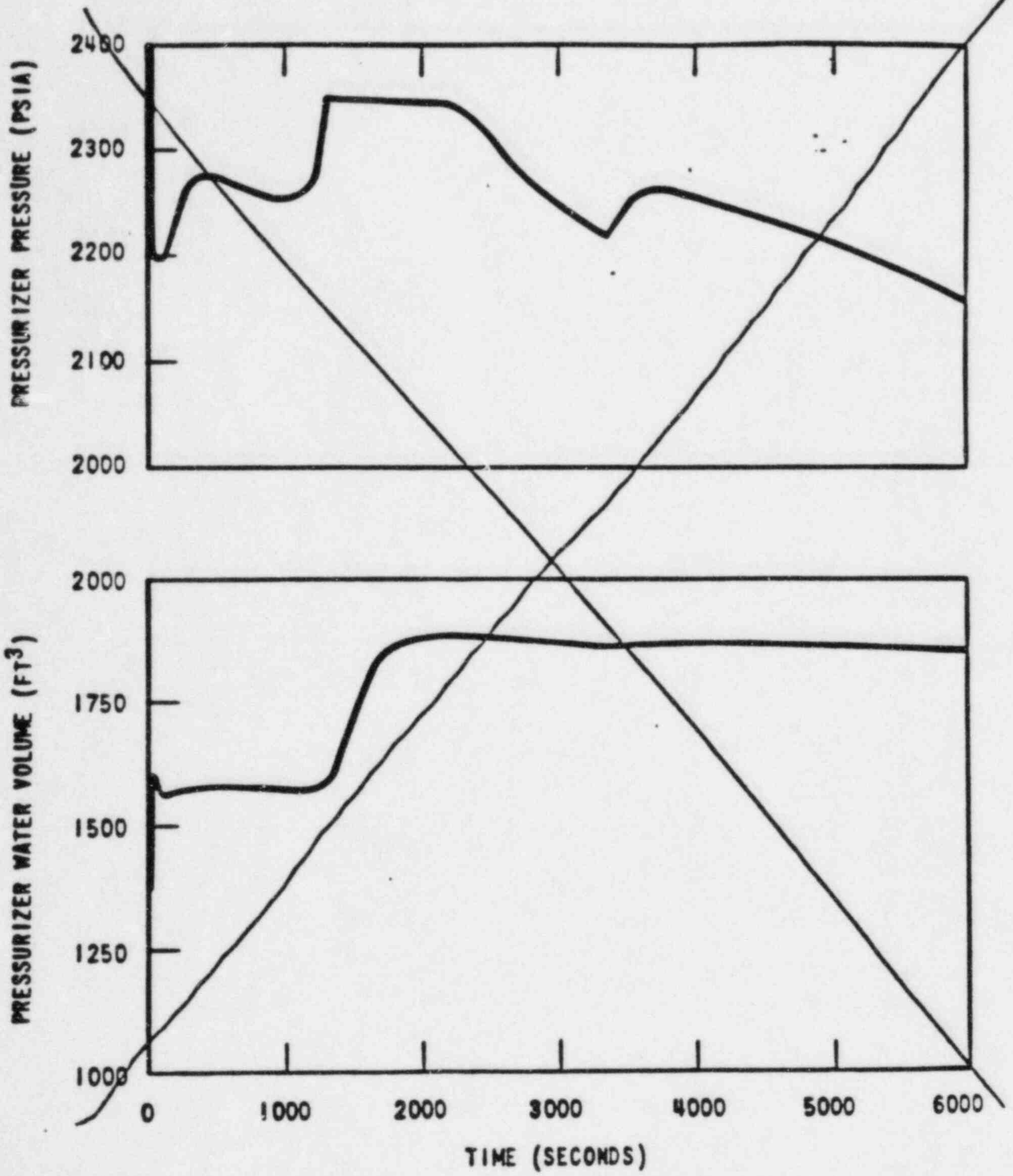

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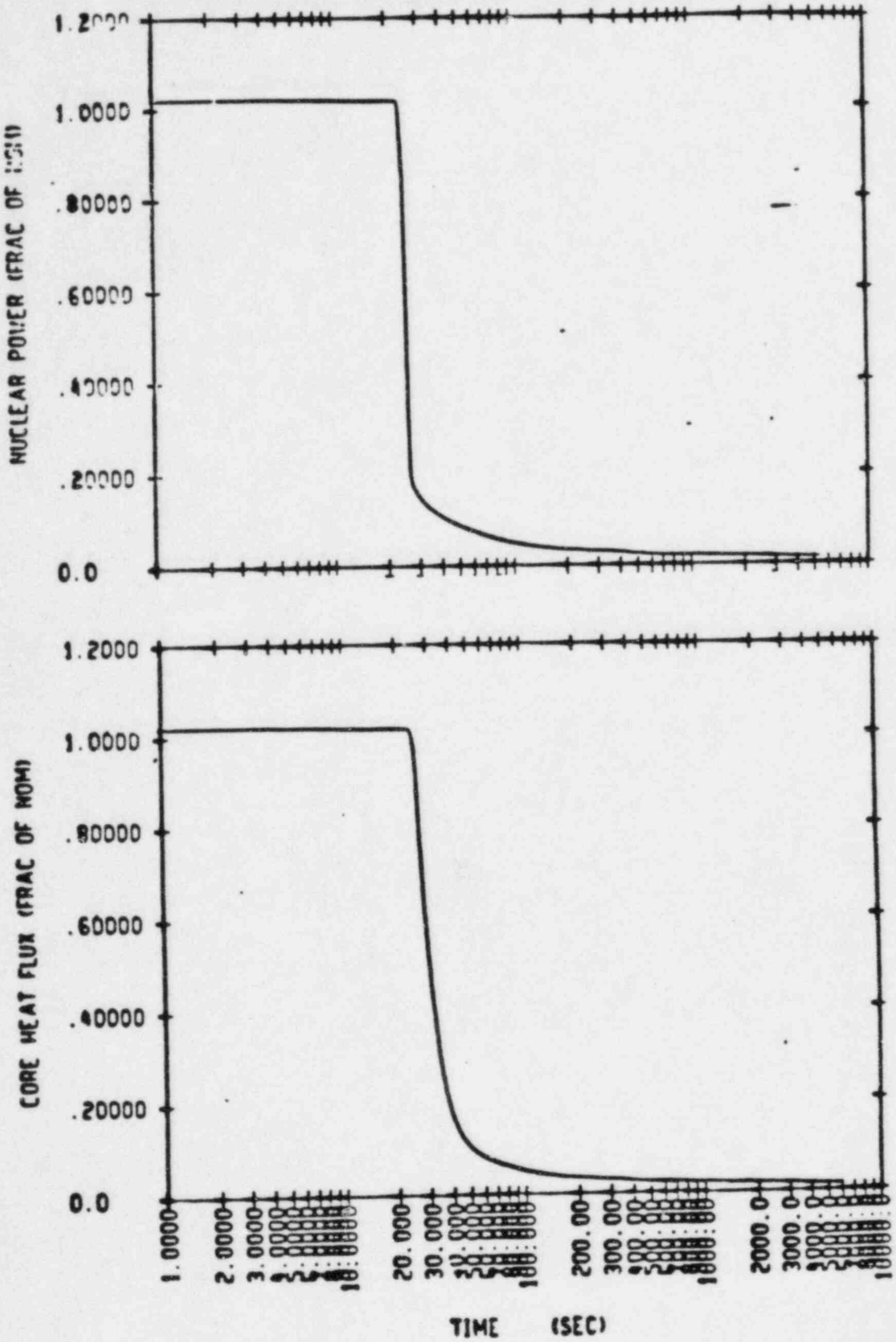
Pressurizer Pressure and Water Volume Transients for Loss of Normal Feedwater

Figure 15.2-9



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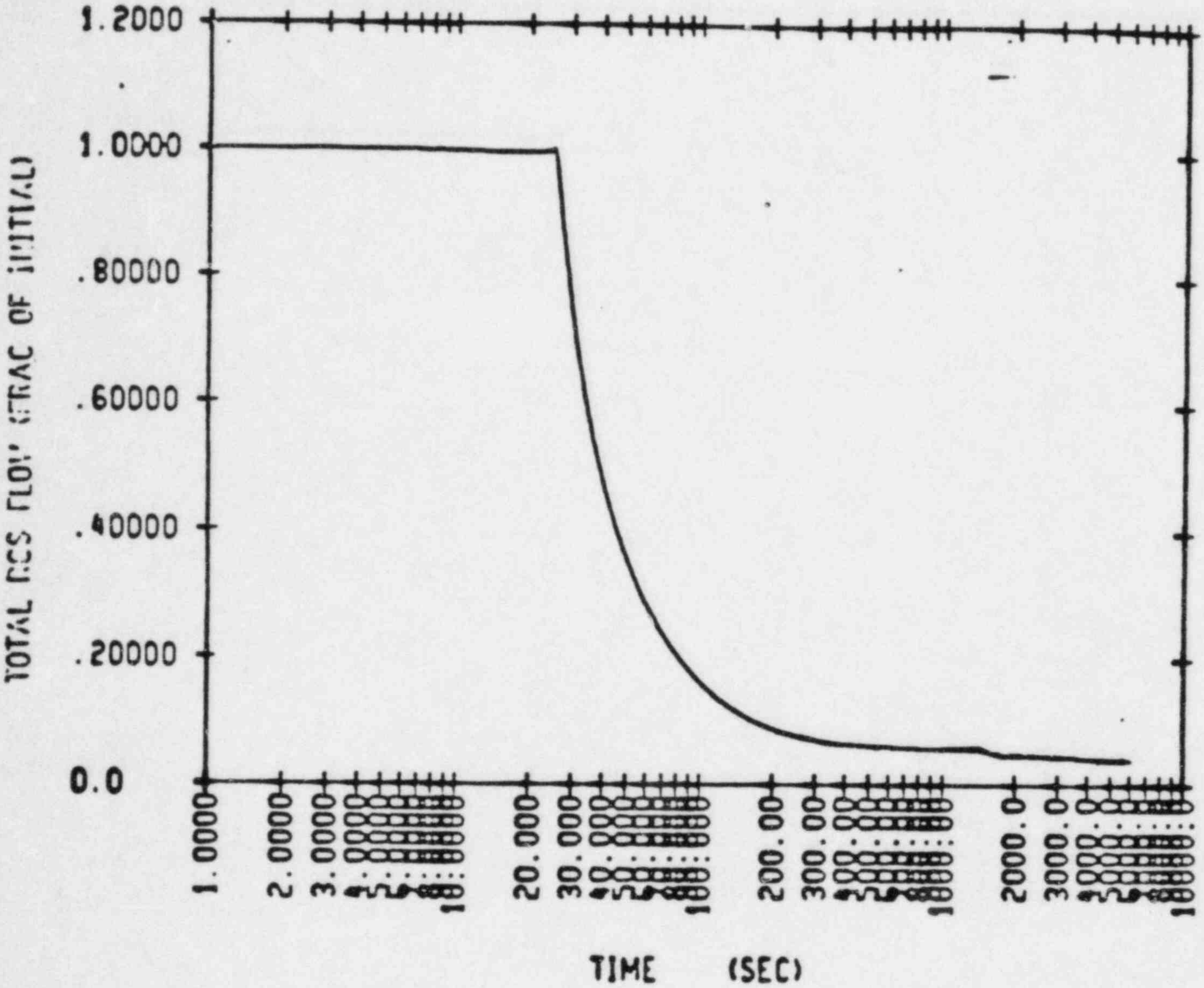
~~[illegible]~~ Nuclear Power and Core Heat Flux Transients

for Loss of Normal Feedwater

Figure 15.2 9A



Total Reactor Coolant System Flow Transient

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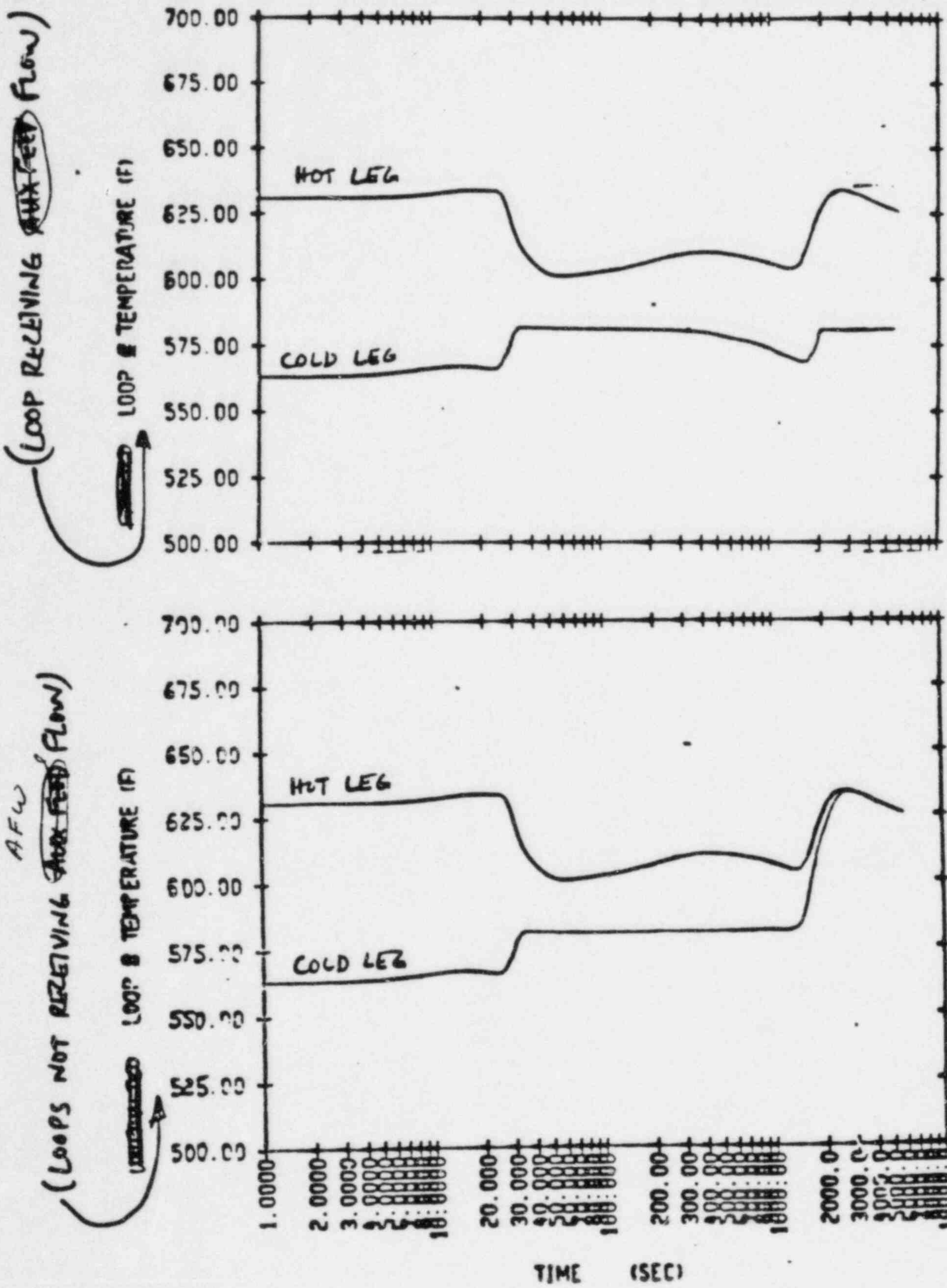
~~Pressurizer Pressure and Water Volume Transients~~
for Loss of Normal Feedwater

Figure 15.2 8B



Loop Temperature Transient

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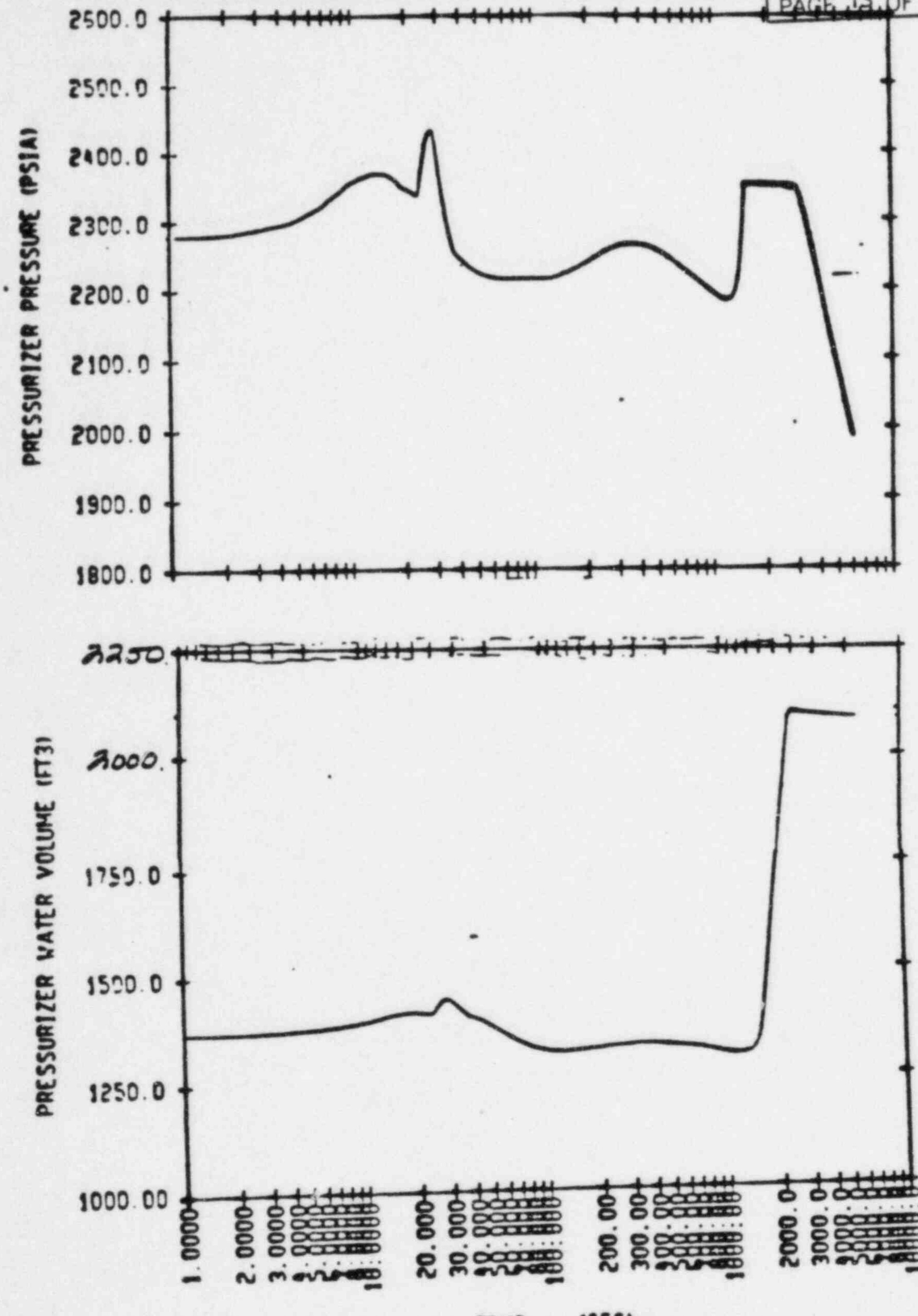

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Pressurizer Pressure and Water Volume Transients for Loss of Normal Feedwater

Figure 15.2-8 D

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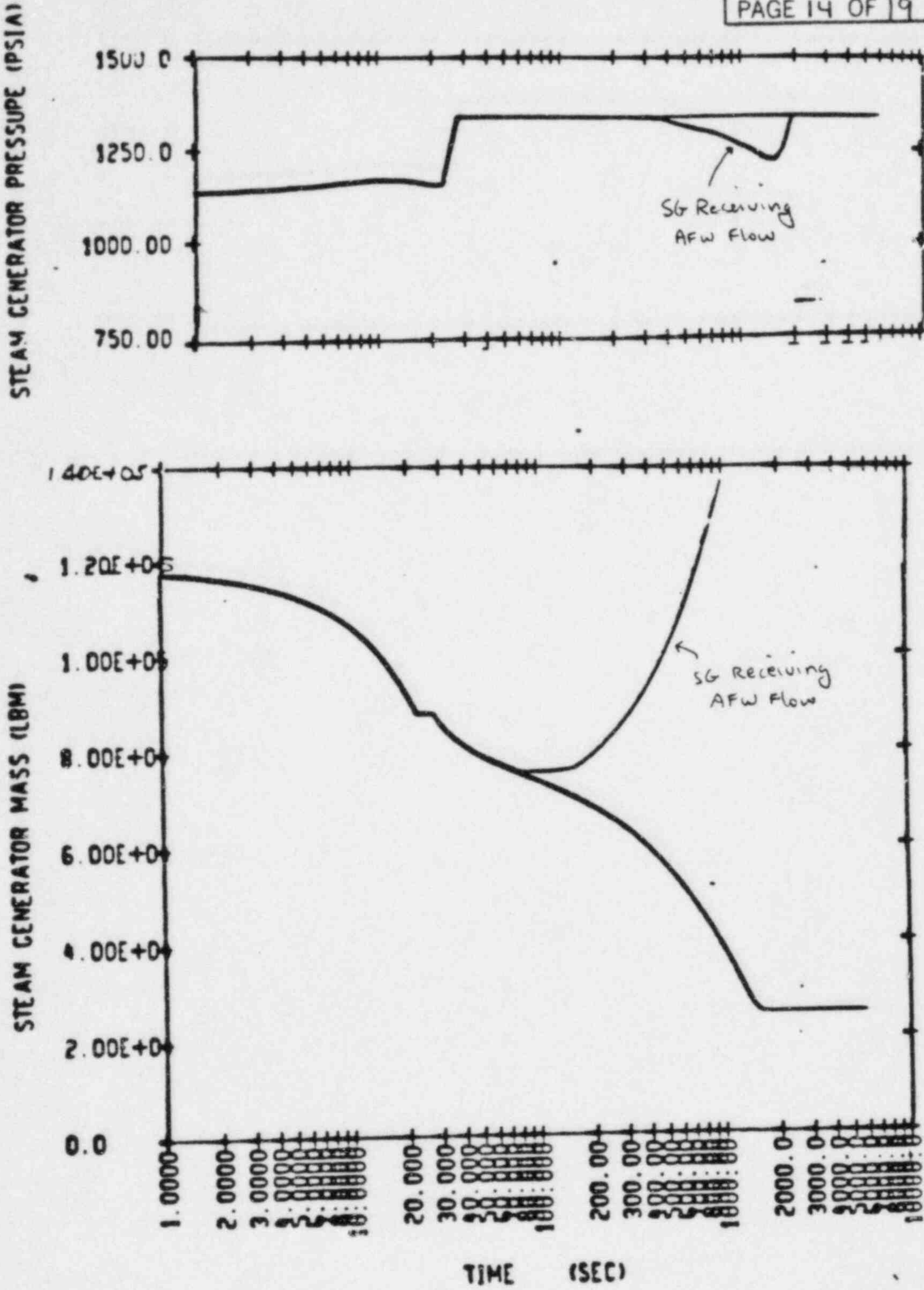

Steam Generator Pressure and Mass Transients

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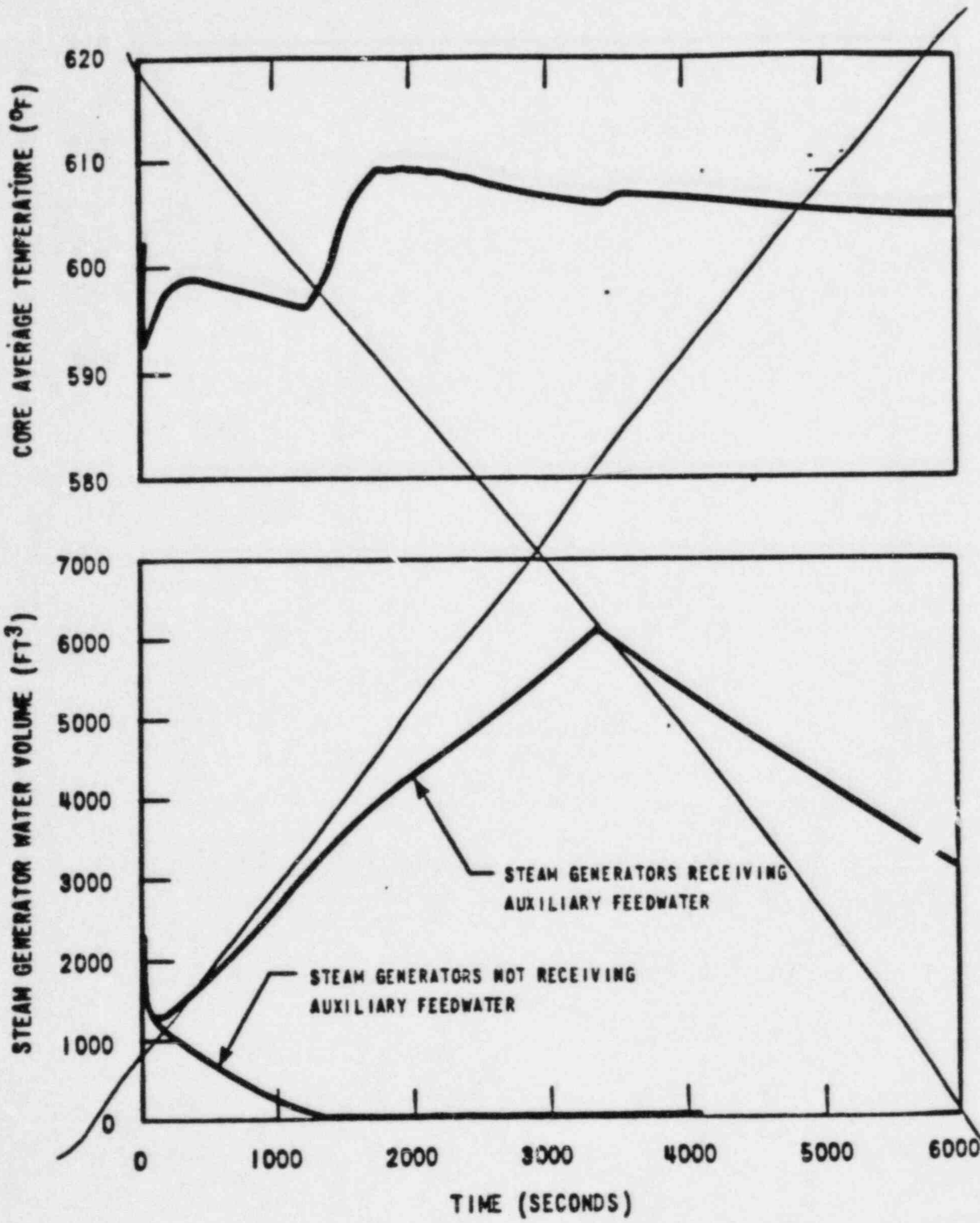
~~Core Average Temperature Transient and Steam Generator~~
~~Water Volume Transient~~ for Loss of Normal Feedwater

Figure 15.2-10



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Core Average Temperature Transient and Steam Generator Water Volume Transient for Loss of Normal Feedwater

Figure 15.2-10

valves, flash tank level, and flash tank steam discharge valves from the control room. The SG blowdown and sample Containment isolation valves are closed automatically by the signals initiating the start of the AFW System. (See Section 10.4.9.5 for a discussion of AFW control.) The SG blowdown inlet flow control valves are controlled automatically by flow transmitters and controllers which maintain the blowdown flowrate from each SG to the flash tank. 39

Flash tank level is maintained by controlling the flash tank condensate drain to the condenser with a flow override to prevent excessive flow through the blowdown demineralizers when the blowdown flowrate is increased above normal. Flash tank pressure is maintained by controlling flash tank steam flow to the FW heater 13, with a provision of bypassing this steam to the condenser if the heater is out of service and also on turbine trip. Blowdown water temperature to the demineralizers is regulated by a control valve in the cooling water outlet line from the SG blowdown regenerative heat exchanger. A blowdown flash tank safety relief valve provides overpressure protection. 39

Blowdown flowrate from each SG, blowdown flash tank pressure and temperature, and steam and liquid flowrate from the flash tank are displayed in the control room. High and low water levels in the flash tank are alarmed on a control room annunciator. High blowdown flowrate and high and low flash tank pressures are displayed on the plant computer and on an annunciator. 39

High blowdown water temperature at the SG blowdown regenerative heat exchanger outlet is alarmed in the control room. This high temperature also terminates the blowdown water flow to the mixed-bed demineralizers. On high level in the flash tank, the control valve at the flash tank outlet line (which goes directly to the condenser, bypassing the HXs and the demineralizers) is modulated to maintain proper flash tank water level. Local pressure gauges are furnished throughout the system, and a level gage is installed on the blowdown flash tank. 39

10.4.8.5 Tests and Inspections. Periodic tests and recalibration will be performed on flow, pressure, and temperature indicators. The system isolation valves will be periodically tested to check operability in accordance with ASME B&PV Code, Section XI. In addition, periodic inspection and preventive maintenance will be conducted on components as required. Valving and system arrangement will be such as to make all components available for inspection. Active components are so designed that they can be tested during plant operation. 39 39

10.4.9 Auxiliary Feedwater System

10.4.9.1 Design Bases. The function of the AFW System is to supply FW to the secondary side of the SGs whenever the normal FW supply is not available. Causes and analyses for conditions which require the use of the AFW System, including loss of coolant from small breaks, are discussed in Chapter 15.

The AFW System is designed to perform the following safety functions: 39

1. Supply the SGs with water required for decay heat removal. 31

∴ three AFW motor driven pumps are automatically actuated following a

2. Start and deliver design flow automatically following any incident causing loss of FW. Under any condition, the AFWS is capable of starting and operating unattended for at least 10 minutes.

3. Function within a SG pressure range from approximately 100 psia up to a pressure equivalent to the lowest set SG safety-valve relief pressure plus accumulation (1,338 psia). The lower value corresponds to the point at which the Residual Heat Removal System (RHRS) can be operated for continuing cooldown. 46

4. Function under the following conditions: loss of main FW; various environmental occurrences; a main FW line break or a MS line break; with or without offsite power available considering at the same time any single failure.

5. Supply FW in the unlikely event the control room must be evacuated.

6. Be tested during normal plant operation.

7. Meet safety class (refer to the AFWS, piping diagram, Figure 10.4.9-1, for SC 2 and SC 3 divisions) and seismic Category I requirements as defined in Section 3.2. 39

Although

The AFWS is designed to deliver 550 gal/min within one minute of automatic 31
initiation to at least one SG after a feedwater line rupture or a steam line 39
break. The AFWS is designed to deliver 550 gal/min within one minute of 31
automatic initiation to each of at least two SGs after a loss of FW accident. 39

loss of main feedwater, a

The ~~AFWS is designed to deliver 550 gal/min within one minute of automatic initiation to at least two SGs after~~ loss of offsite power (LOOP). The motor 45
driven AFW pumps are automatically started by the load sequencers, though when
the pumps are started they are in a recirculation mode, and no flow will enter 39
the SGs until a SG low-low water level or safety injection (SI) signal 45
initiates flow.

The AFWS is designed to prevent the possibility of hydraulic instability (i.e., water hammer) by incorporation of the following:

1. A separate nozzle is provided for the introduction of AFW to the SG. (This AFW nozzle does not incorporate a feed ring or feed preheater design.)

2. The length of horizontal piping immediately upstream of the AFW nozzle is minimized. 39

3. The AFW inlet piping within the SG is designed to be self venting.

4. The outlet of the AFW nozzle is designed to be below the normal SG water level.

The combination of the above prevents the formation of steam voids in the inlet piping which is susceptible to condensation upon the introduction of AFW.

, analysis of the above transients has shown that a minimum AFW flow of 515 gal/min is acceptable.

The fourth pump is a horizontal, centrifugal, multistage, noncondensing steam turbine-driven unit which supplies FW to the fourth SG. A steam line connection is taken from the SC 2 section of one MS line upstream of the MS isolation valve (see Figure 10.3-1). The AFW steam line is provided with a remote manual containment isolation valve. The turbine discharge steam exhausts 39
directly to the atmosphere.

Each SG is supplied by a separate AFW train. Normally closed, fail-closed 39
cross-connections are provided between the four trains to permit flow from any pump to any SG.

Each of the four pumps is provided with a minimum-flow automatic recirculation 31
system. The recirculation flow returns to the upper section of the AFST.

Each pump recirculation line is designed to SC 3 requirements inside the isolation valve cubicle (IVC). 39 The recirculation lines from the IVC to the AFST are designed to NNS class requirements. Water losses through credible failures of recirculation lines are included in the storage tank inventory requirements.

The AFW line to each SG, one per pump, is provided with a remote manual containment isolation valve (see Section 6.2.4). 39 Each line connects directly to the upper shell of the SG.

The AFW pumps are located in a seismic Category I building and are physically 39
separated from each other by their placement in individual compartments. These compartments are designed to preclude coincident damage to redundant equipment in the event of a postulated pipe rupture, equipment failure, or 39
missile generation.

Figures 1.2-21 and 1.2-25 show the AFWS component arrangements. The AFW steam 39
supply pipe to the AFW turbine is routed directly to the turbine pump compartment, located immediately beneath the MS line piping. This piping is routed such that it does not penetrate any of the AFW motor-driven pump compartments.

10.4.9.3 Safety Evaluation. The AFWS is designed to seismic Category I requirements and will withstand a single failure and still perform its design requirements. The loss of one motor-driven pump or the turbine-driven pump will not limit the design safety function of the system. In the event that the makeup water to the AFST is lost, the minimum quantity of water within the 31
AFST is sufficient for a safe shutdown of the reactor. Therefore, failure of any one AFW component will not preclude safe shutdown of the reactor. To demonstrate the capability to meet the single-failure criterion, a component failure mode and effects analysis is presented in Table 10.4-3. In addition the AFWS has been analyzed to determine its reliability and the results of the 39
analysis are provided in Appendix 10A ~~(Later)~~. The system is SC 3 from the AFST (Figure 9.2.6-2) up to the containment isolation valves. The steam line to the AFW pump turbine is SC 2 to the isolation valve and SC 3 to the turbine. The isolation valves and piping from the containment isolation 45
valves to the SG are SC 2 (see Figure 10.4.9-1). 39

INSERT III

INSERT
III

The loss of feedwater transient has been analyzed with one auxiliary feedwater pump delivering water to one effective steam generator. This represents the very conservative case of one pump unavailable due to maintenance and the A train and D train pumps not starting automatically due to failure of the A train initiation signal.