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3.8 STEAM AND POWER CONVERSION SYSTEMS

Applicability: Applies to the operating status of the steam and power conversion systems.

Objective: To define conditions of the steam-relieving capacity.

Specification:

1. When the reactor coolant of a nuclear unit is heated above 350°F, the following conditions must be met:

   a. TWELVE (12) of its steam generator safety valves shall be operable (except for testing).

   b. Its main steam stop valves shall be operable and capable of closing in 5 seconds or less.

   c. System piping, interlocks and valves directly associated with the related components in TS 3.8.1 a, b shall be operable.

2. The iodine-131 activity on the secondary side of a steam generator shall not exceed 0.67 μCi/gm.

3. With the reactor coolant system above 350°F, if any of above specifications cannot be met within 48 hours, the reactor shall be shutdown and the reactor coolant temperature reduced below 350°F. Specification 3.0.1 applies.

TABLE 3.16-1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No. Unit 3	Valve No. Unit 4	Maximum (a)(b) Allowable Leakage
High-Head Safety Injection			5.0 gpm
Loop A, hot leg	3-874A	4-874A	5.0 gpm
cold leg	3-875A	4-875A	5.0 gpm
cold leg	3-873A	4-873A	5.0 gpm
Loop B, hot leg	3-874B	4-874B	5.0 gpm
cold leg	3-875B	4-875B	5.0 gpm
cold leg	3-873B	4-873B	5.0 gpm
Loop C, cold leg	3-875C	4-875C	5.0 gpm
cold leg	3-873C	4-873C	5.0 gpm
Residual Heat Removal			
Loop A, cold leg	3-876A	4-876A	5.0 gpm
		4-876E	5.0 gpm
Loop B, cold leg	3-876B	4-876B	5.0 gpm
	3-876D	4-876D	5.0 gpm
Loop C, cold leg	3-876C	4-876C	5.0 gpm
	3-876E		5.0 gpm

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Minimum differential test pressure shall not be less than 150 psid.

3.18 AUXILIARY FEEDWATER SYSTEM

3.18.1 Two independent auxiliary feedwater trains as specified in Table 3.18-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

1) With one of the two required independent auxiliary feedwater trains inoperable, either restore the inoperable train to an OPERABLE status within 72 hours, or place the affected unit(s) in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2) With both required auxiliary feedwater trains inoperable, within 1 hour either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train, or place the affected unit(s) in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

TABLE 3.18-1

AUXILIARY FEEDWATER SYSTEM OPERABILITY

Unit	Train	Steam Supply Flowpath	Pump	Discharge Water Flowpaths(3)
3	1	SG 3C via MOV-3-1405 or SG 3B via MOV-3-1404(1)	A or C(2)	SG 3A via CV-3-2816 SG 3B via CV-3-2817 SG 3C via CV-3-2818
3	2	SG 3A via MOV-3-1403 or SG 3B via MOV-3-1404(1)	B or C(2)	SG 3A via CV-3-2831 SG 3B via CV-3-2832 SG 3C via CV-3-2833
4	1	SG 4C via MOV-4-1405 or SG 4B via MOV-4-1404(1)	A or C(2)	SG 4A via CV-4-2816 SG 4B via CV-4-2817 SG 4C via CV-4-2818
4	2	SG 4A via MOV-4-1403 or SG 4B via MOV-4-1404(1)	B or C(2)	SG 4A via CV-4-2831 SG 4B via CV-4-2832 SG 4C via CV-4-2833

NOTES

(1) Steam admission valves MOV-3-1404 and MOV-4-1404 can be aligned to either train to restore operability in the event MOV-3-1403 or MOV-3-1405, or MOV-4-1403 or MOV-4-1405 are inoperable.

(2) During single and two unit operation, one pump is required to be operable in each train. The standby pump "C" can be aligned to either train to restore operability in the event one of the required pumps is inoperable.

(3) One flow control valve in each train for each unit can be inoperable for a period not to exceed 72 hours. The ACTION required for a single train inoperable shall be followed.

3.19 CONDENSATE STORAGE TANKS

3.19.1 The Condensate Storage Tanks shall be OPERABLE with a contained water volume of at least 185,000 gallons of water as follows:

3.19.1.1 Single Unit Prior to Escalating into Mode 3

a) ONE water supply from either Condensate Storage Tank including flowpath piping and valves.

3.19.1.2 Second Unit Prior to Escalating into Mode 3

a) ONE water supply from each unit corresponding Condensate Storage Tank including flowpath piping and valves.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

Single Unit at or Above Mode 3

1) With one water supply from a Condensate Storage Tank inoperable, within 4 hours, either realign the other Condensate Storage Tank containing the required water volume to the suction of the Auxiliary Feedwater pumps or restore the inoperable water supply to OPERABLE status or be in HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2) With both water supplies from the Condensate Storage Tanks inoperable, within 4 hours restore the water supply from either Condensate Storage Tank to Operable status or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Both Units at or Above Mode 3

1) With one water supply from a Condensate Storage Tank inoperable, restore the inoperable water supply to OPERABLE status within 4 hours or place one unit in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. Refer to Single Unit Operation ACTION for single unit at or above MODE 3.

2) With both water supplies from the Condensate Storage Tanks inoperable within 1 hour restore one water supply from a Condensate Storage Tank to OPERABLE status or place one unit in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. If unable to restore at least one water supply from a Condensate Storage Tank to OPERABLE status within 4 hours from initial declaration of inoperability, the second unit shall be placed in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

4.21 CONDENSATE STORAGE TANKS

4.21.1 The Condensate Storage Tanks shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limit when the tank is the supply source for the auxiliary feedwater pumps.

B3.8 BASES FOR LIMITING CONDITIONS FOR OPERATION, STEAM AND POWER CONVERSION SYSTEMS

The limit on secondary coolant iodine-131 specific activity is based on a postulated release of secondary coolant equivalent to the contents of three steam generators to the atmosphere due to a net load rejection. The limiting dose for this case would result from radioactive iodine in the secondary coolant. I-131 is the dominant isotope because of its low MPC in air and because the other shorter lived iodine isotopes cannot build up to significant concentrations in the secondary coolant under the limits of primary system leak rate and activity. One tenth of the iodine in the secondary coolant is assumed to reach the site boundary making allowance for plate-out and retention in water droplets. The inhalation thyroid dose at the site boundary is then;

$$\text{Dose (Rem)} = C \cdot V \cdot B \cdot DCF \cdot X/Q \cdot 0.1$$

Where: C = secondary coolant I-131 specific activity

= 1.34 curies/m^3 (μCi/cc) or 0.67 Ci/m^3, each unit

V = equivalent secondary coolant volume released = 214 m^3

B = breathing rate = 3.47×10^{-4} m^3 sec.

X/Q = atmospheric dispersion parameter = 1.54×10^{-4} sec/m^3

0.1 = equivalent fraction of activity released

DCF = dose conversion factor, Rem/ci

The resultant thyroid dose is less than 1.5 Rem.

In the unlikely event of complete loss of electrical power to the nuclear units, decay heat removal will be assured by the availability of the steam-driven auxiliary feedwater pumps and steam discharge via the steam generator safety valves and PORVs.[1]

(1) FSAR - Section 10.3

B3.18 BASES - AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power. Steam can be supplied to the pump turbines from either or both units through redundant steam headers. Two D.C. motor operated valves and one A.C. motor operated valve on each unit isolate the three main steam lines from these headers. Both the D.C. and A.C. motor operated valves are powered from safety related sources. Auxiliary feedwater can be supplied through redundant lines to the safety related portions of the main feedwater lines to each of the steam generators. Air operated fail closed flow control valves are provided to modulate the flow to each steam generator. Each steam driven auxiliary feedwater pump has sufficient capacity for single and two unit operation to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

B3.19 BASES - CONDENSATE STORAGE TANKS

There are two (2) seismically designed 250,000 gallon condensate storage tanks. A minimum of 185,000 gallons is maintained in each tank. The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the Reactor Coolant System at HOT STANDBY conditions for approximately 23 hours or maintain the Reactor Coolant System at HOT STANDBY conditions for 15 hours and then cool down the Reactor Coolant System to below 350°F at which point the Residual Heat Removal System may be placed into operation.

ATTACHMENT I

SAFETY AND NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Description of Amendment Request:

Page 3.8-1

The proposed amendment would delete the Specifications for the Auxiliary Feedwater (AFW) System and the Condensate Storage Tank (CST) in current Technical Specification 3.8, Steam and Power Conversion Systems. Requirements for the AFW System and CST will be included in new Technical Specifications 3.18 and 3.19.

Pages 3.18-1, 3.18-2, 3.19-1

The proposed amendment would add Technical Specification 3.18, Auxiliary Feedwater System, and 3.19, Condensate Storage Tank. These proposed Specifications provide explicit limiting conditions for operation (LCO), applicability requirements, and ACTION requirements for operation of the AFW System and CST. The format (i.e., LCO, applicability, action requirements) is that of NUREG-0452, Standard Technical Specifications for Westinghouse Pressurized Water Reactors (WSTS), although the requirements in the proposed Specifications differ from the WSTS because of the uniqueness of the Turkey Point Plant AFW System design (i.e., shared system, three turbine driven pumps, etc.).

Proposed Specification 3.18 would differ from the current Technical Specification 3.8 as follows:

1) Table 3.18-1 defines the number of independent auxiliary feedwater pumps and their associated flowpaths (steam and water) required to be operable for single and two unit operation.

2) The proposed Specification (LCO) requires that two of the three turbine driven AFW pumps be operable for both single and two unit operation. A single AFW pump is sized to provide adequate flow to satisfy the minimum AFW flow requirements for two unit operation. A recent Westinghouse reanalysis of the Loss of Non-Emergency AC Power to the Plant Auxiliaries event is attached. A second operable pump would satisfy the single active failure criterion. Although all three AFW pumps would normally be operable and aligned to the AFW system, as is required by the current Specification for two unit operation, the proposed Specification (LCO) is consistent with the current design basis and safety analyses, would permit additional operational flexibility (reducing heatup/cooldown transients on the units), and is consistent with 10 CFR 50.36(c)(2) which states that LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

3) The applicability of the proposed AFW specification is Modes 1, 2, 3, as defined in the Technical Specifications. This change differs from the current requirements in that the action requirements are applicable in all specified modes, whereas, under the current Technical Specification action is only specified to be taken when a limiting condition is not met during power operation, although the AFW System is required to be operable when the reactor coolant temperature is above 350°F. Modes for AFW operation are not specified in the current Technical Specifications.

4) The ACTION requirements in the proposed AFW Specification are consistent with the current Specification except for the following. The proposed Specification would allow one discharge water flowpath (i.e., a flow control valve)to be inoperable in both trains for a period not to exceed 72 hours and allow one train to be inoperable in both units for a 72 hour period. In both cases, the AFW System will provide the minimum required flow through the remaining four operable flowpaths, or through the remaining operable train in each unit, respectively.

Proposed Specification 3.19 would differ from the current Technical Specification 3.8 as follows:

1) The proposed ACTION requirements are more restrictive in that they require action to be taken within 4 hours (consistent with the WSTS) as opposed to 48 hours in the current Specification.

Page 4.21-1

The proposed amendment would add Technical Specification 4.21, Condensate Storage Tank. This specification provides a surveillance requirement to demonstrate the CST operable by verifying at least once per 12 hours that the water volume in the CST is within its limits when the CST is the supply source for the AFW pumps. There is no similar requirement in the current Specifications.

Pages B3.8-1, B3.18-1, B3.19-1

The proposed amendment would add separate bases (B3.18 and B3.19) for the AFW system and the CST. The Bases for the Steam and Power Conversion Systems, B3.8, would be modified accordingly to delete reference to the AFW System and CST.

Page 3.16-2

In Table 3.16-1, the valve numbers for HHSI Loop C Cold Leg and RHR Loop B Cold Leg shown as 3-875B and 3-876A would be corrected to read 3-875C and 3-876B, respectively, to reflect the correct valve numbers.

Basis for No Significant Hazards Consideration Determination:

The Commission has provided standards for determining whether a significant hazards consideration exists 10 CFR 50.92(c)]. A proposed amendment to an operating license for the facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

Operation of Turkey Point Units 3 and 4 in accordance with the proposed amendments would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   Technical Specification 3.18 and Table 3.18.1 define the number of independent AFW pumps and their associated flowpaths (steam and water) required to be operable for single and two unit operation. Operation of the system in accordance with this Specification would ensure that adequate core and RCP heat removal is available to prevent water relief out the pressurizer relief or safety valves. This is the basis for the current Technical Specification and consistent with the FSAR safety analyses.

   The requirements for CST operation in proposed Technical Specification 3.19 are as restrictive or more restrictive than the requirements in current Technical Specification 3.8.

   The addition of Specification 4.21 to verify operability of the CSTs further ensures that the limiting conditions for operation for the CSTs will be met.

   The changes to Table 3.16-1 would correct valve designations. No changes to the systems were made.

Based on the above, operation in accordance with the proposed changes would not involve an increase in the probability or consequences of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

The operation of the AFW System and CSTs is not significantly different from that allowed by the current Technical Specifications, and since the conclusions of the safety analyses remain valid (i.e., adequate core and reactor coolant pump heat removal is available), operation in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Involve a significant reduction in a margin of safety.

As noted in response to (1) and (2) above, the operation of the AFW System and CSTs is permitted by the proposed Technical Specification is not significantly different from that allowed by the current Technical Specifications. Adequate heat removal capability is available to remove core and RCP heat and to prevent water relief out the pressurizer relief or safety valves, insuring that the integrity of the core and RCS is not compromised. Also, the addition of CST surveillance requirements further ensures that the LCO for the CSTs will be met. Thus, operation in accordance with the proposed changes will not involve a significant reduction in a margin of safety.

Based on the above discussion, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated, or create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

Therefore, operation of the facility in accordance with the proposed amendment would pose no threat to the public health and safety, and would not involve a significant hazards consideration.

14.1.12 LOSS OF NON-EMERGENCY A-C POWER TO THE PLANT AUXILIARIES

14.1.12.1 Identification of Causes and Accident Description

A complete loss of non-emergency AC power may result in the loss of all power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite AC distribution system.

This transient is more severe than the turbine trip event because for this case the decrease in heat removal by the secondary system is accompanied by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip:

(1) upon reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal; or (2) due to loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of AC power with turbine and reactor trips, the sequence described below will occur:

1. Plant vital instruments are supplied from emergency DC power sources.

2. As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for steam dump. If the steam flow rate through the power relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

3. As the no load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

4. Both emergency diesel generators will start on loss of voltage on both the 4160 volt buses of either unit. At the same time, these buses will be isolated from their normal supply and their motor feed breakers will be opened. Motor control center tie breakers will open, separating the vital loads from the others. At this time, the generator breakers will close. All further operations required to pick up the emergency loads will be done manually by the operator. These operations will be done automatically in a sequential manner only if there has been a coincident safety injection signal in the same unit.

The following provides the necessary protection against a loss of AC power:

1. Reactor trip on

   a. Low-low water level in any steam generator
   b. Steam flow-feedwater flow mismatch coincident with low water level in any steam generator

2. Three turbine-driven auxiliary feedwater pumps (shared by units 3 & 4) are started on any of the following:

   a. Low-low water level in any steam generator
   b. Any safety injection signal
   c. Loss of offsite power
   d. Loss of 4 kV bus
   e. Trip of all main feedwater pumps
   f. Manual actuation

The steam driven auxiliary feedwater pumps are started upon the loss of normal feedwater supply. The turbine utilizes steam from the main steam line to drive the feedwater pump to deliver makeup water to the steam generators. The turbine driver exhausts the steam to the atmosphere. The pumps take suction directly from the condensate storage tanks for delivery to the steam generators.

The steam-driven auxiliary feedwater pump can be tested at any time by admitting steam to the turbine driver. The auxiliary feedwater control valves and main steam power relief valves can be operationally tested whenever the unit is at hot shutdown and the remaining valves in the system are operationally tested when the turbine driver and pump are tested.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops.

A loss of AC power event, as described above, is a more limiting event than the turbine-trip initiated decrease in secondary heat removal without loss of AC power. However, a loss of AC power to the plant auxiliaries as postulated above could result in a loss of normal feedwater if the condensate pumps lose their power supply.

Following the reactor coolant pump coastdown caused by the loss of AC power, the natural circulation capability of the RCS will remove residual and decay heat from the core, aided by auxiliary feedwater in the secondary system. An analysis is presented here to show that the natural circulation flow in the RCS following a loss of AC power event is sufficient to remove residual heat from the core.

Turkey Point Units 3 and 4 share common electrical and auxiliary feedwater systems. Thus, a loss of non-emergency AC power to the plant auxiliaries could simultaneously affect both units. The auxiliary feedwater system would then be required to provide flow to both units.

The worst single failure in the auxiliary feedwater system could result in availability of only one of the three auxiliary feedwater pumps. Flow from this pump could be as low as 125 gpm to one of the units until the operator takes action from the control board to realign the flow split to the units.

The analysis is performed for one unit, representing the worst case of the two units.

14.1.12.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN Code (Reference 1) is performed to obtain the plant transient following a station blackout. The simulation describes the plant thermal kinetics, RCS including the natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator mass, pressurizer water level, and reactor coolant average temperature.

Assumptions made in the analysis are:

1. The plant is initially operating at 102 percent of the Engineered Safety Features design rating, 2307.4 MWt.

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

3. A heat transfer coefficient in the steam generator associated with RCS natural circulation, following the reactor coolant pump coastdown.

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.

5. The worst single failure occurs in the auxiliary feedwater system. This results in the availability of one auxiliary feedwater pump supplying 125 gpm to three steam generators three minutes following a start signal on low-low steam generator water level. At ten minutes following reactor trip the operator takes action from the control board to increase the flow to 230 gpm delivered to three steam generators.

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

7. The initial reactor coolant average temperature is 4°F higher than the nominal value, and initial pressurizer pressure is 30 psi higher than nominal.

8. The pressurizer power-operated relief valves and pressurizer spray system are assumed to operate normally. This results in a conservative transient with respect to peak pressurizer water level. If these control systems did not operate the pressurizer safety valves would maintain peak RCS pressure at or below the actuation setpoint (2500 psia) throughout the transient.

9. A control rod drop time to dashpot of 2.4 seconds was assumed, consistent with optimized fuel assemblies (OFA).

The assumptions used in the analysis are essentially identical to the loss of normal feedwater flow incident (Section 14.1.11) except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip.

Results

The transient response of the RCS following a loss of AC power is shown in Figures 14.1.12-1 and 14.1.12-2. The calculated sequence of events for this accident is listed in Table 14.1.12-2.

The first few seconds after the loss of power to the reactor coolant pumps will closely resemble a simulation of the complete loss of flow incident i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. The natural circulation flow was calculated using an analytical method based on the conditions of equilibrium flow and maximum loop flow impedence. The model has given results within 15% of the measured flow values obtained during natural circulation tests conducted at the Yankee-Rowe plant and has also been confirmed at San Onofre and Connecticut Yankee. The natural circulation flow ratio as a function of reactor power is given in Table 14.1.12-1.

14.1.12.3 Conclusions

Analysis of the natural circulation capability of the Reactor Coolant System has demonstrated that sufficient heat removal capability exists following RCP coastdown to prevent fuel or clad damage.

14.1.12.4 References

1. Burnett, T. W. T., et al, "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

2. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

TABLE 14.1.12-1

NATURAL CIRCULATION REACTOR COOLANT FLOW
VS. REACTOR POWER

Reactor Power % Full Power	Reactor Coolant Flow % Nominal Flow
3.5	4.6
3.0	4.3
2.5	4.0
2.0	3.7
1.5	3.3
1.0	2.9

TABLE 14.1.12-2

TIME SEQUENCE OF EVENTS FOR LOSS OF NON-EMERGENCY AC POWER

Event	Time (sec)
Main feedwater flow stops	10
Low-low steam generator water level trip	64
Rods begin to drop	66
Reactor coolant pumps begin to coastdown	68
Flow from one turbine driven auxiliary feedwater pump is started	244
Operator realigns system to increase auxiliary feedwater flow to 230 gpm	664
Feedwater lines are purged and cold auxiliary feedwater is delivered to three steam generators	1080
Peak water level in pressurizer occurs	3720
Core decay heat decreases to auxiliary feedwater heat removal capacity	~4000

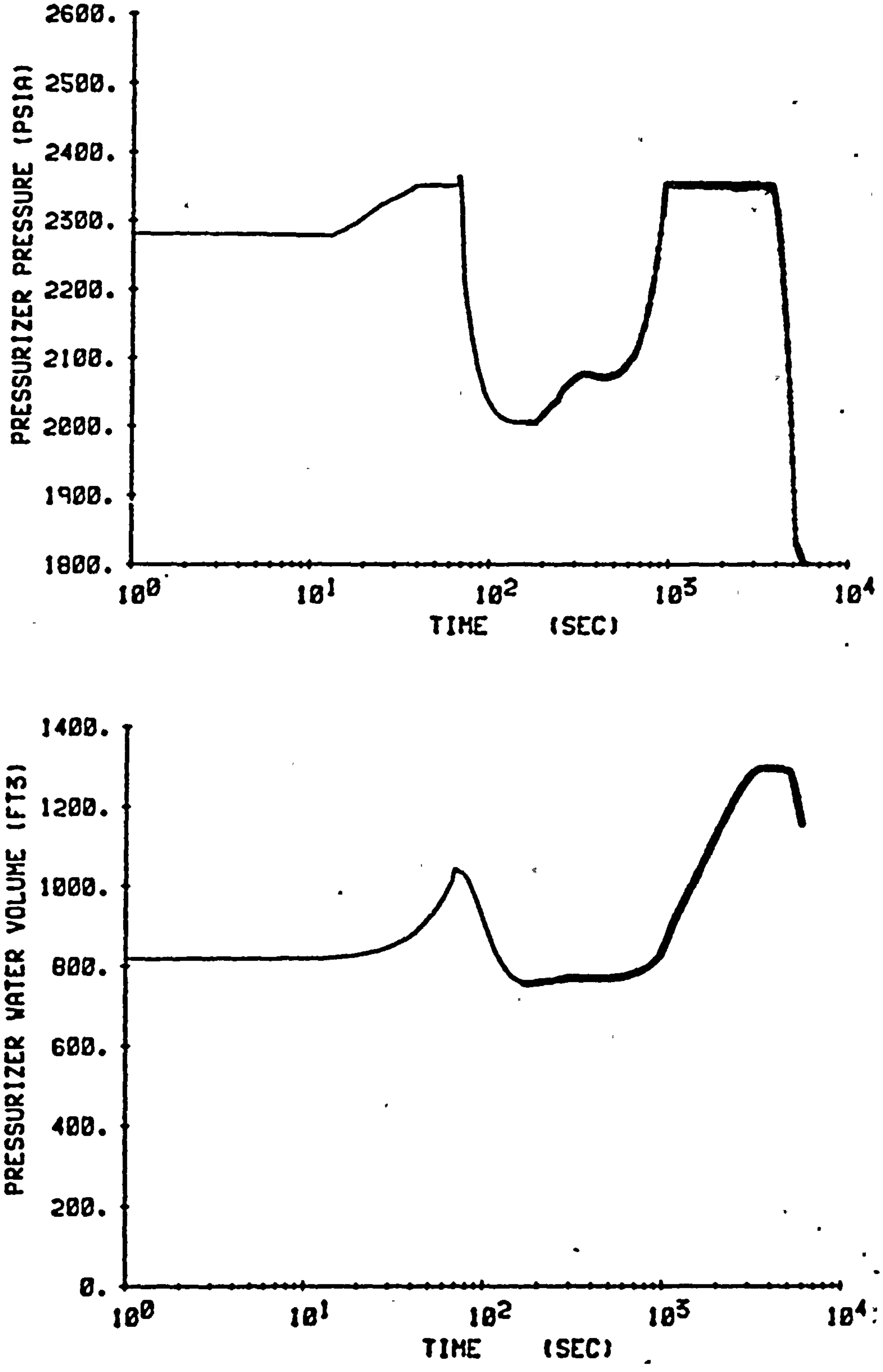

Figure 14.1.12-1. Pressurizer Pressure and Water Volume Transients for Loss of Offsite Power

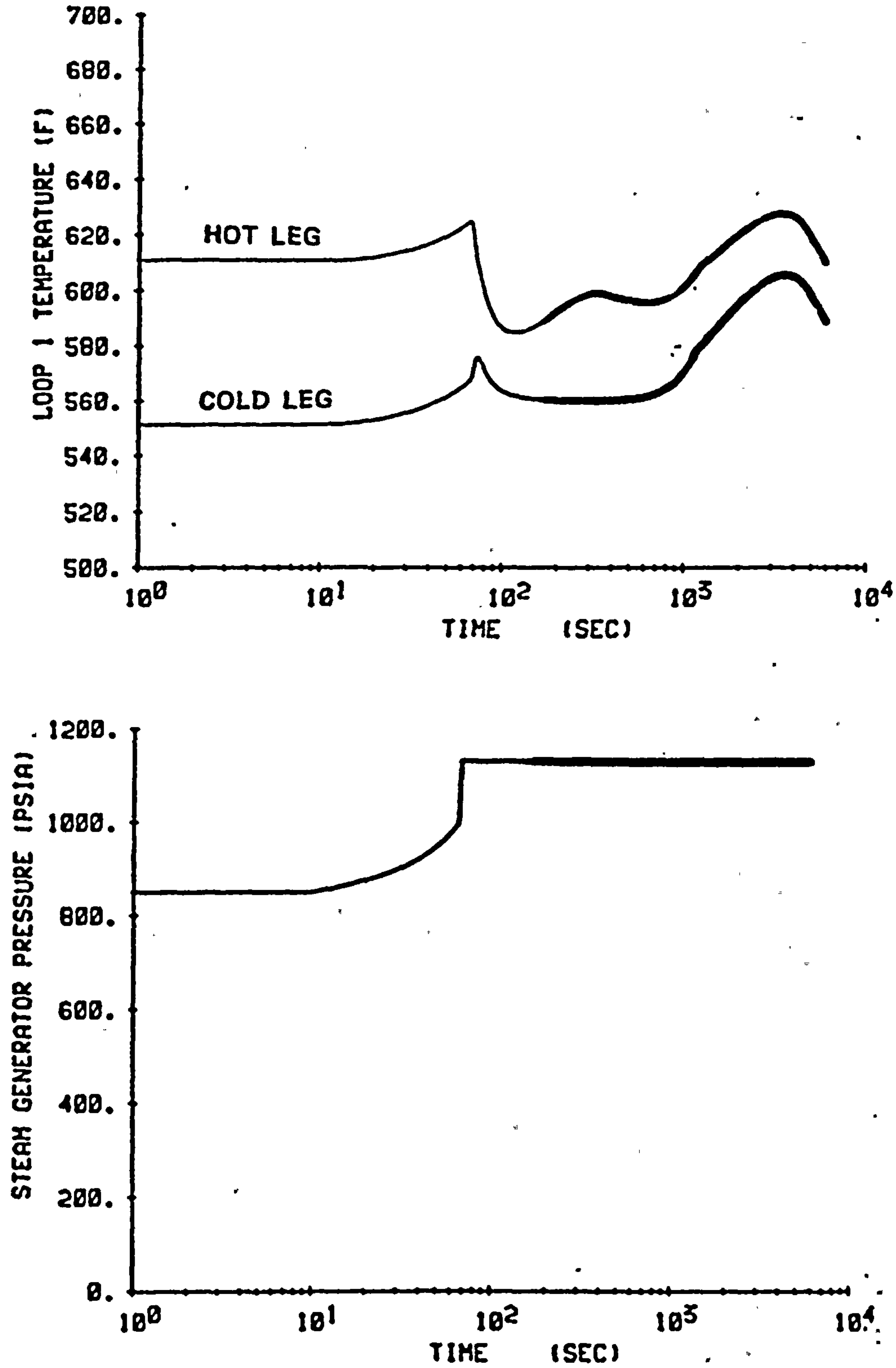

Figure 14.1.12-2. Loop Temperatures and Steam Generator Pressure for Loss of Offsite Power