TABLE OF CONTENTS

(continued)

(continued)

TABLE OF CONTENTS

BASES

ACTIONS

B.2, B.3, B.4, B.5, and B.6 (continued)

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits (i.e., SR 3.2.1.1 and SR 3.2.2.1) ensures that current operation at ≤ 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Accident for the duration of operation under these conditions. As a minimum, the following accident analyses shall be re-evaluated:

a. Rod insertion characteristics;
b. Rod misalignment;
c. Small break loss of coolant accidents (LOCAs);
d. Rod withdrawal at full power;
e. Large break LOCAs;
f. Main steamline break; and
g. Rod ejection.

A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

When Required Actions of Condition B cannot be completed within their Completion Time, the plant must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to at least MODE 2 with K_{eff} < 1.0 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 2 with K_{eff} < 1.0 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

U.1 and U.2

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms (i.e., diverse trip features) in MODES 1 and 2. Condition U applies on a RTB basis. This allows one diverse trip feature to be inoperable on each RTB. However, with two diverse trip features inoperable (i.e., one on each of two different RTBs), at least one diverse trip feature must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this time interval.

With one trip mechanism for one RTB inoperable, it must be restored to an OPERABLE status within 48 hours. The affected RTB shall not be bypassed while one of the diverse trip features is inoperable except for the time required to perform maintenance to one of the diverse trip features. The allowable time for performing maintenance of the diverse trip features is 6 hours for the reasons stated under Condition T. The Completion Time of 48 hours for Required Action U.2 is reasonable considering that in this Condition there is one remaining diverse trip feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

V.1

If the Required Action and Associated Completion Time of Condition R, S, T, or U is not met, the plant must be placed in a MODE where the Functions are no longer required to be OPERABLE. To achieve this status, the plant must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner without challenging plant systems.

It should be noted that for inoperable channels of Functions 16a, 16b, 16c, and 16d, the MODE of Applicability will be exited before Required Action V.1 is completed. Therefore, the plant shutdown may be stopped upon exiting the MODE of Applicability per LCO 3.0.2.

(continued)

BASES

ACTIONS (continued)

W.1 and W.2

Condition W applies to the following reactor trip Functions in MODE 3, 4, or 5 with the CRD System capable of rod withdrawal or all rods not fully inserted:

- RTBs;
- RTB Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

With two trip mechanisms inoperable, at least one trip mechanism must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this time interval.

With one trip mechanism or train inoperable, the inoperable trip mechanism or train must be restored to OPERABLE status within 48 hours. For the trip mechanisms, Condition W applies on a RTB basis. This allows one diverse trip feature to be inoperable on each RTB. However, with two diverse trip features inoperable (i.e., one on each of two different RTBs), at least one diverse trip feature must be restored to OPERABLE status within 1 hour.

The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

X.1 and X.2

If the Required Action and Associated Completion Time of Condition W is not met, the plant must be placed in a MODE where the Functions are no longer required. To achieve this status, action be must initiated immediately to fully insert all rods and the CRD System must be incapable of rod withdrawal within 1 hour. These Completion Times are reasonable, based on operating experience to exit the MODE of Applicability in an orderly manner.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

e. Auxiliary Feedwater - Undervoltage - Bus 11A and 11B

The Undervoltage - Bus 11A and 11B Function must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or RHR will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

A loss of power to 4160 V Bus 11A and 11B will be accompanied by a loss of power to both MFW pumps and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each bus. Loss of power to both buses will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip. Each bus is considered a separate Function for the purpose of this LCO.

f. Auxiliary Feedwater - Trip Of Both Main Feedwater Pumps

A Trip of both MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal. The MFW pumps are equipped with a breaker position sensing device. An open supply breaker indicates that the pump is not running. Two OPERABLE channels per MFW pump satisfy redundancy requirements with two-out-of-two logic. Each MFW pump is considered a Separate Function for the purpose of this LCO. A trip of both MFW pumps starts both motor driven AFW (MDAFW) pumps to ensure that at least one SG is available with water to act as the heat sink for the reactor. However, this actuation of the MDAFW pumps is not credited in the mitigation of any accident.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

f. Auxiliary Feedwater - Trip Of Both Main Feedwater Pumps (continued)

During MODES 1 and 2, the AFW pumps may be providing for removal of decay heat with the MFW pumps removed from service. To prevent an unnecessary actuation of both MDAFW pumps under these conditions, a MFW pump breaker may be placed in the test position provided it is capable of being tripped on undervoltage and overcurrent conditions on the associated 4160 V bus.

(continued)

BASES

ACTIONS (continued)

M.1

If the Required Actions and Completion Times of Condition L are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < 2000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

N.1

Condition N applies if a AFW Manual Initiation channel is inoperable. If a manual initiation switch is inoperable, the associated AFW or SAFW pump must be declared inoperable and the applicable Conditions of LCO 3.7.5, "Auxiliary Feedwater (AFW) System" must be entered immediately. Each AFW manual initiation switch controls one AFW or SAFW pump. Declaring the associated pump inoperable ensures that appropriate action is taken in LCO 3.7.5 based on the number and type of pumps involved.

SURVEILLANCE REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1. Each channel of process protection supplies both trains of the ESFAS. When testing Channel 1, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel 2, Channel 3, and Channel 4 (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

(continued)

BASES

LCO 19, 20. AFW Flow (continued)

The AFW System provides decay heat removal via the SGs and is comprised of the preferred AFW System and the Standby AFW (SAFW) System. The use of the preferred AFW or SAFW System to provide this decay heat removal function is dependent upon the type of accident. AFW flow indication is required from the three pump trains which comprise the preferred AFW System since these pumps automatically start on various actuation signals. The failure of the preferred AFW System (e.g., due to a high energy line break (HELB) in the Intermediate Building) is detected by AFW flow indication. At this point, the SAFW System is manually aligned to provide the decay heat removal function.

SAFW flow can also be used to verify that AFW flow is being delivered to the SGs. However, the primary indication of this is provided by SG water level. Therefore, flow indication from the SAFW pumps is not required.

Each of the three preferred AFW pump trains has two redundant transmitters; however, only the flow transmitter supplied power from the same electrical train as the AFW pump is required for this LCO. Therefore, flow transmitters FT-2001 (MCB indicator FI-2021A) and FT-2006 (MCB indicator FI-2023A) comprise the two required channels for SG A and FT-2002 (MCB indicator FI-2022A) and FT-2007 (MCB indicator FI-2024A) comprise the two required channels for SG B.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The LOP DG start instrumentation is required for the ESF Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS). Undervoltage conditions which occur independent of any accident conditions result in the start and bus connection of the associated DG, but no automatic loading occurs.

Accident analyses credit the loading of the DG based on the loss of offsite power during a Design Basis Accident (DBA). The most limiting DBA of concern is the large break loss of coolant accident (LOCA) which requires ESF Systems in order to maintain containment integrity and protect fuel contained within the reactor vessel (Ref. 2). The detection and processing of an undervoltage condition, and subsequent DG loading, has been included in the delay time assumed for each ESF component requiring DG supplied power following a DBA and loss of offsite power.

The loss of offsite power has been assumed to occur either coincident with the DBA or at a later period (40 to 90 seconds following the reactor trip) due to a grid disturbance caused by the turbine generator trip. If the loss of offsite power occurs at the same time as the safety injection (SI) signal parameters are reached, the accident analyses assumes the SI signal will actuate the DG within 2 seconds and that the DG will connect to the affected safeguards bus within an additional 10 seconds (12 seconds total time). If the loss of offsite power occurs before the SI signal parameters are reached, the accident analyses assumes the LOP DG start instrumentation will actuate the DG within 2.75 seconds and that the DG will connect to the affected safeguards bus within an additional 10 seconds (12.75 seconds total time). If the loss of offsite power occurs after the SI signal parameters are reached (grid disturbance), the accident analyses assumes the DG will connect to the bus within 1.5 seconds after the feeder breaker to the bus is opened (DG was actuated by SI signal). The grid disturbance has been evaluated based on a 140°F peak clad temperature penalty during a LOCA and demonstrated to result in acceptable consequences.

(continued)

BASES

ACTIONS (continued)

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.5-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment ventilation isolation radiation monitor channel. Since the two containment radiation monitors measure different parameters, failure of a single channel may result in loss of the radiation monitoring Function for certain events. Consequently, the failed channel must be restored to OPERABLE status. The 4 hour allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

B.1

Condition B applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the system and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the system and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place each valve in its closed position or the applicable Conditions of LCO 3.9.3, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.5-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.5.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred and the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The CHANNEL CHECK agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1 (continued)

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.5.2

A COT is performed every 92 days on each required channel to ensure the entire channel will perform the intended Function. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). This test verifies the capability of the instrumentation to provide the containment ventilation system isolation. The setpoint shall be left consistent with the current plant specific calibration procedure tolerance.

SR 3.3.5.3

This SR is the performance of an ACTUATION LOGIC TEST. All possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay is tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 24 months. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.4

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11.

2. NUREG-1366.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNB design criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the plant that could impact these parameters must be assessed for their impact on the DNB design criterion. The transients analyzed include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The limit for pressurizer pressure is based on a ± 30 psig instrument uncertainty. The accident analyses assume that nominal pressure is maintained at 2235 psig. By Reference 2, minor fluctuations are acceptable provided that the time averaged pressure is 2235 psig.

The RCS coolant average temperature limit is based on a ± 4°F instrument uncertainty which includes a ± 1.5°F deadband. It is assumed that nominal T_{avg} is maintained within ± 1.5°F of the nominal T_{avg} specified in the COLR . By Reference 2, minor fluctuations are acceptable provided that the time averaged temperature is within 1.5°F of nominal.

The limit for RCS flow rate is based on the nominal T_{avg} and SG plugging criteria limit. Additional margin of approximately 3% is then added for conservatism.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature, and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

(continued)

BASES (continued)

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The RCS loops are considered filled until the isolation valves are opened to facilitate draining of the RCS. The loops are also considered filled following the completion of filling and venting the RCS. However, in both cases, loops filled is based on the ability to use a SG as a backup. To be able to take credit for the use of one SG the ability to pressurize to 50 psig and control pressure in the RCS must be available. This is to prevent flashing and void formation at the top of the SG tubes which may degrade or interrupt the natural circulation flow path (Ref. 2). One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least one SG is required to be ≥ 16%.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODE 1 > 8.5% RTP";
LCO 3.4.5, "RCS Loops – MODES 1 ≤ 8.5% RTP, 2, AND 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – Water Level ≥ 23 Ft" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Water Level < 23 Ft" (MODE 6).

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and both SGs have secondary side water levels < 16%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore at least one SG secondary side water level. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. The action to restore must continue until an RHR loop is restored to OPERABLE status or SG secondary side water level is restored.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.4.7.3 Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the standby RHR pump. If secondary side water level is ≥ 16% in at least one SG, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.
REFERENCES	1. UFSAR, Section 14.6.1.2.6 2. NRC Information Notice 95-35.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 59.8 psig and the peak containment temperature is 374°F (both experienced during an SLB). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5," Containment Temperature," for a detailed discussion.) The analyses and evaluations assume a plant specific power level of 102%, one CS train and one containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 7).

The effect of an inadvertent CS actuation is not considered since there is no single failure, including the loss of offsite power, which results in a spurious CS actuation.

The modeled CS System actuation for the containment analysis is based on a response time associated with exceeding the containment Hi-Hi pressure setpoint to achieving full flow through the CS nozzles. To increase the response of the CS System, the injection lines to the spray headers are maintained filled with water. The CS System total response time is 28.5 seconds for one pump to the upper spray header and 26.5 seconds for two pumps (average time between upper and lower spray headers). These total response times (assuming the containment Hi-Hi pressure is reached at time zero) includes opening of the motor operated isolation valves, containment spray pump startup, and spray line filling (Ref. 8).

(continued)

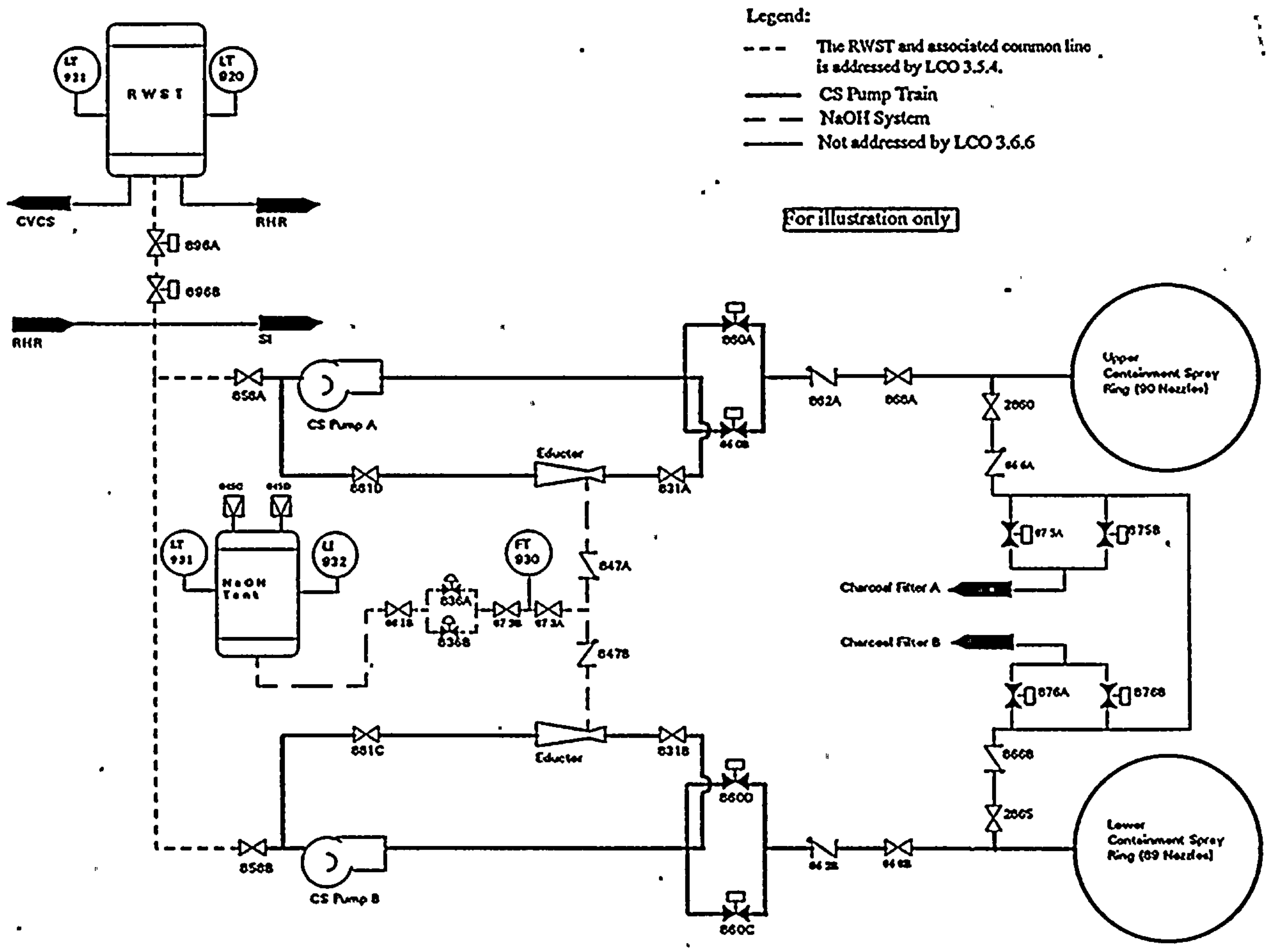
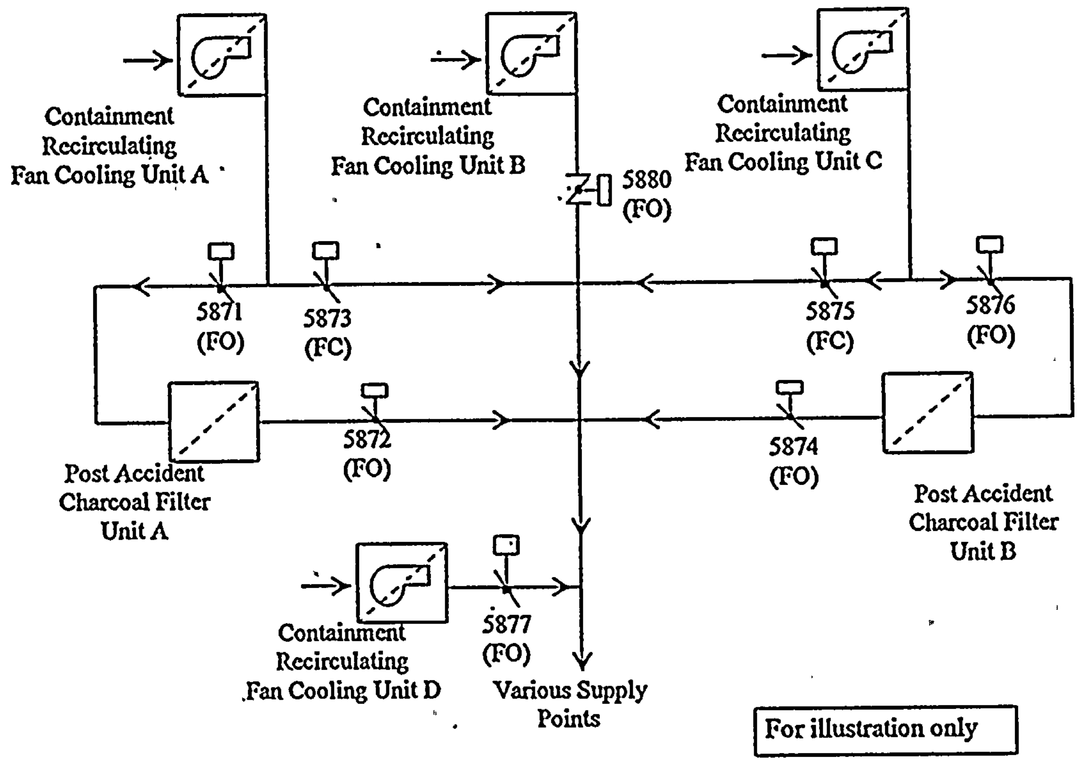


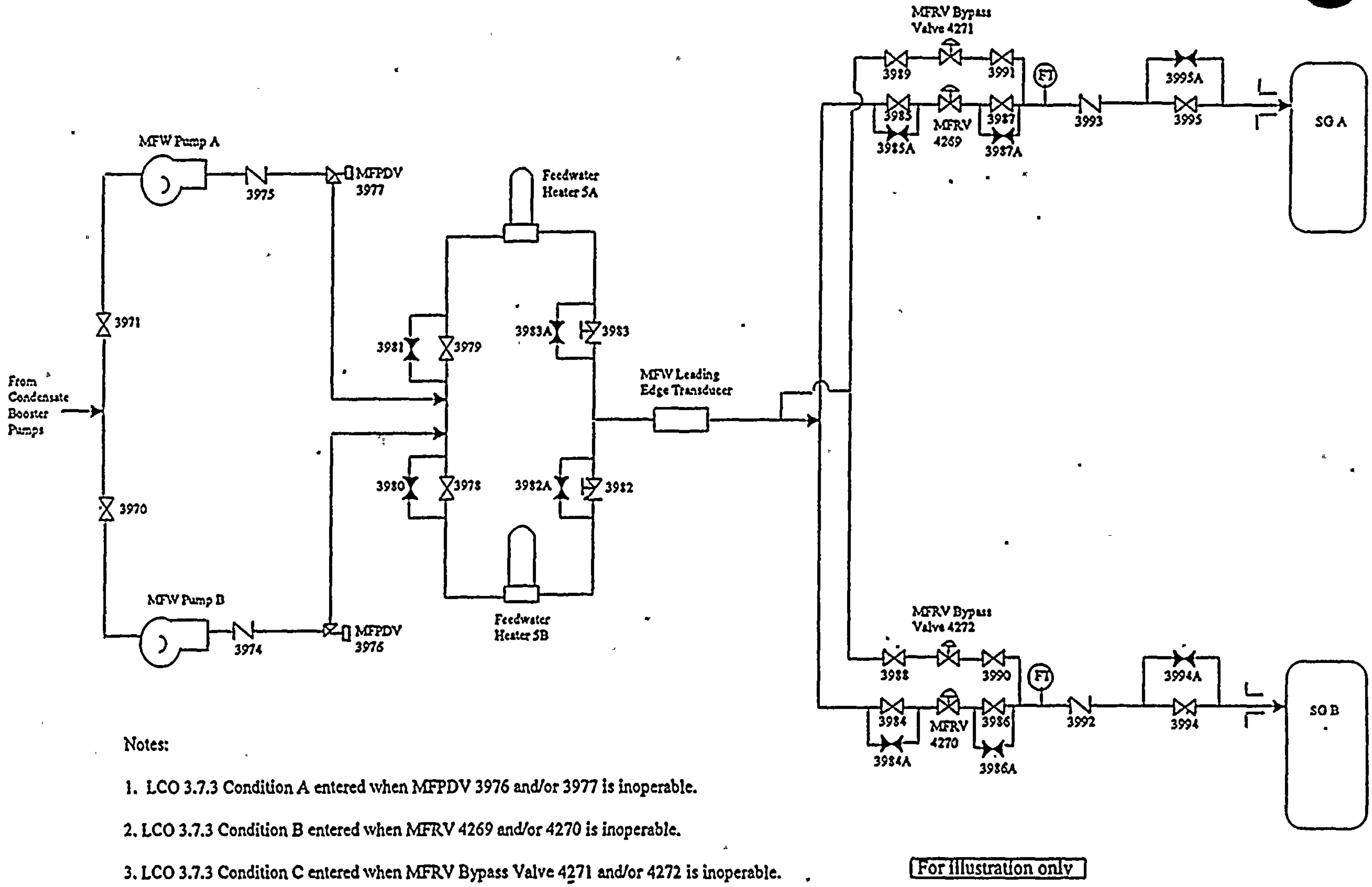
Figure B 3.6.6-1
Containment Spray and NaOH Systems



Notes:

1. Dampers 5871 and 5872 are associated with Post Accident Charcoal Filter Unit A
2. Dampers 5874 and 5876 are associated with Post Accident Charcoal Filter Unit B
3. Damper 5873 is assoicated with both CRFC Unit A and Post Accident Charcoal Filter Unit A
4. Damper 5875 is associated with both CRFC Unit C and Post Accident Charcoal Filter Unit B

Figure B 3.6.6-2
CRFC and Containment Post-Accident Charcoal Systems



Notes:

1. LCO 3.7.3 Condition A entered when MFPDV 3976 and/or 3977 is inoperable.
2. LCO 3.7.3 Condition B entered when MFRV 4269 and/or 4270 is inoperable.
3. LCO 3.7.3 Condition C entered when MFRV Bypass Valve 4271 and/or 4272 is inoperable.
4. LCO 3.7.3 Condition E entered when any combination of valve inoperabilities results in an unisolable flowpath from the condensate booster pumps to one or more SGs.

Figure B 3.7.3-1
MFRVs, Associated Bypass Valves and MFPDVs

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System supplies feedwater to the steam generators (SGs) to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The SGs function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the SGs via the main steam safety valves (MSSVs) or atmospheric relief valves (ARVs). If the main condenser is available, steam may be released via the steam dump valves. The AFW System is comprised of two separate systems, a preferred AFW System and a Standby AFW (SAFW) System (Ref. 1).

<u>AFW System</u>

The preferred AFW System consists of two motor driven AFW (MDAFW) pumps and one turbine driven AFW (TDAFW) pump configured into three separate trains which are all located in the Intermediate Building (see Figure B 3.7.5-1). Each MDAFW train provides 100% of AFW flow capacity, and the TDAFW pump provides 200% of the required capacity to the SGs, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to the condensate storage tanks (CSTs). Each MDAFW train is powered from an independent Class 1E power supply and feeds one SG, although each pump has the capability to be realigned from the control room to feed the other SG via cross-tie lines containing normally closed motor operated valves (4000A and 4000B). The two MDAFW trains will actuate automatically on a low-low level signal in either SG, opening of the main feedwater (MFW) pump breakers, a safety injection (SI) signal, or the ATWS mitigation system actuation circuitry (AMSAC). The pumps can also be manually started from the control room.

(continued)

BASES

BACKGROUND (continued)

The SAFW Pump Building environment is controlled by room coolers which are supplied by the same SW header as the pump trains. These coolers are required when the outside air temperature is ≥ 80°F to ensure the SAFW Pump Building remains ≤ 120°F during accident conditions.

The AFW System is designed to supply sufficient water to the SG(s) to remove decay heat with SG pressure at the lowest MSSV set pressure plus 1%. Subsequently, the AFW System supplies sufficient water to cool the plant to RHR entry conditions, with steam released through the ARVs.

APPLICABLE SAFETY ANALYSES

The design basis of the AFW System is to supply water to the SG(s) to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the SGs at pressures corresponding to the lowest MSSV set pressure plus 1%.

The AFW System mitigates the consequences of any event with the loss of normal feedwater. The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows (Ref. 2):

a. Feedwater Line Break (FWLB);

b. Loss of MFW (with and without offsite power);

c. Steam Line Break (SLB);

d. Small break loss of coolant accident (LOCA);

e. Steam generator tube rupture (SGTR); and

f. External events (tornados and seismic events).

AFW is also used to mitigate the effects of an ATWS event which is a beyond design basis event not addressed by this LCO.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The AFW System design is such that any of the above DBAs can be mitigated using the preferred AFW System or SAFW System. For the FWLB, SLB, and external events DBAs (items a, c, and f), the worst case scenario is the loss of all three preferred AFW trains due to a HELB in the Intermediate or Turbine Building, or a failure of the Intermediate Building block walls. For these three events, the use of the SAFW System within 10 minutes is assumed by the accident analyses. Since a single failure must also be assumed in addition to the HELB or external event, the capability of the SAFW System to supply flow to an intact SG could be compromised if the SAFW cross-tie is not available. For HELBs within containment, use of either the SAFW System or the AFW System to the intact SG is assumed within 10 minutes.
	For the SGTR events (item e), the accident analyses assume that one AFW train is available upon a SI signal or low-low SG level signal. Additional inventory is being added to the ruptured SG as a result of the SGTR such that AFW flow is not a critical feature for this DBA.
	The loss of MFW (item b) is a Condition 2 event (Ref. 3) which places limits on the response of the RCS from the transient (e.g., no challenge to the pressurizer power operated relief valves is allowed). This analysis has been performed assuming no AFW flow is available until 10 minutes with acceptable results. The most limiting small break LOCA (item d) analysis has also been performed assuming no AFW flow with no adverse impact on peak cladding temperature.
	In summary, all limiting DBAs and transients have been analyzed assuming a 10 minute delay for actuation of flow.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition to its accident mitigation function, the energy and mass addition capability of the AFW System is also considered with respect to HELBs within containment. For SLBs and FWLBs within containment, maximum pump flow from all three AFW pumps is assumed for 10 minutes until operations can isolate the flow by tripping the AFW pumps or by closing the respective pump discharge flow path(s). Therefore, the motor operated discharge isolation valves for the motor MDAFW pump trains (4007 and 4008) are designed to limit flow to ≤ 230 gpm to limit the energy and mass addition so that containment remains within design limits for items a and c. The TDAFW train is assumed to be at runout conditions (i.e., 600 gpm).

The AFW System satisfies the requirements of Criterion 3 of the NRC Policy Statement.

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary or containment.

The AFW System is comprised of two systems which are configured into five trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the SGs are OPERABLE (see Figures B 3.7.5-1 and 3.7.5-2). This requires that the following be OPERABLE:

a. Two MDAFW trains taking suction from the CSTs as required by LCO 3.7.6 (and capable of taking suction from the SW system within 10 minutes), and capable of supplying their respective SG with ≥ 200 gpm within 10 minutes and ≤ 230 gpm total flow upon AFW actuation;

b. The TDAFW train taking suction from the CSTs as required by LCO 3.7.6 (and capable of taking suction from the SW system within 10 minutes), provided steam is available from both main steam lines upstream of the MSIVs, and capable of supplying both SGs with ≥ 200 gpm each within 10 minutes; and

(continued)

BASES

LCO (continued)

c. Two motor driven SAFW trains capable of being initiated either locally or from the control room within 10 minutes, taking suction from the SW System, and supplying their respective SG and the opposite SG through the SAFW cross-tie line with ≥ 200 gpm.

The piping, valves, instrumentation, and controls in the required flow paths are also required to be OPERABLE. The TDAFW train is comprised of a common pump and two flow paths. A TDAFW train flow path is defined as the steam supply line and the SG injection line from/to the same SG. The failure of the pump or both flow paths renders the TDAFW train inoperable.

The cross-tie line for the preferred MDAFW pumps is not required for this LCO. However, since the accident analyses have been performed assuming a 10 minute delay for AFW, and there are two separate systems, the use of this cross-tie line is allowed in MODES 1, 2, and 3. Also, provided that the AFW and SAFW discharge valves are set to provide the minimum required flow, the recirculation lines for the preferred AFW system and SAFW system pumps are not credited in the accident analysis. The recirculation lines are also not required to be OPERABLE for this LCO since the MSSVs maintain the SG pressure below the pump's shutoff head.

The SAFW Pump Building room coolers are required to be OPERABLE when the outside air temperature is ≥ 80°F. If one room cooler is inoperable, the associated SAFW train is inoperable.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW System is lost. In addition, the AFW System is required to supply enough makeup water to replace the lost SG secondary inventory as the plant cools to MODE 4 conditions.

In MODE 4, 5, or 6, the SGs are not normally used for heat removal, and the AFW System is not required.

(continued)

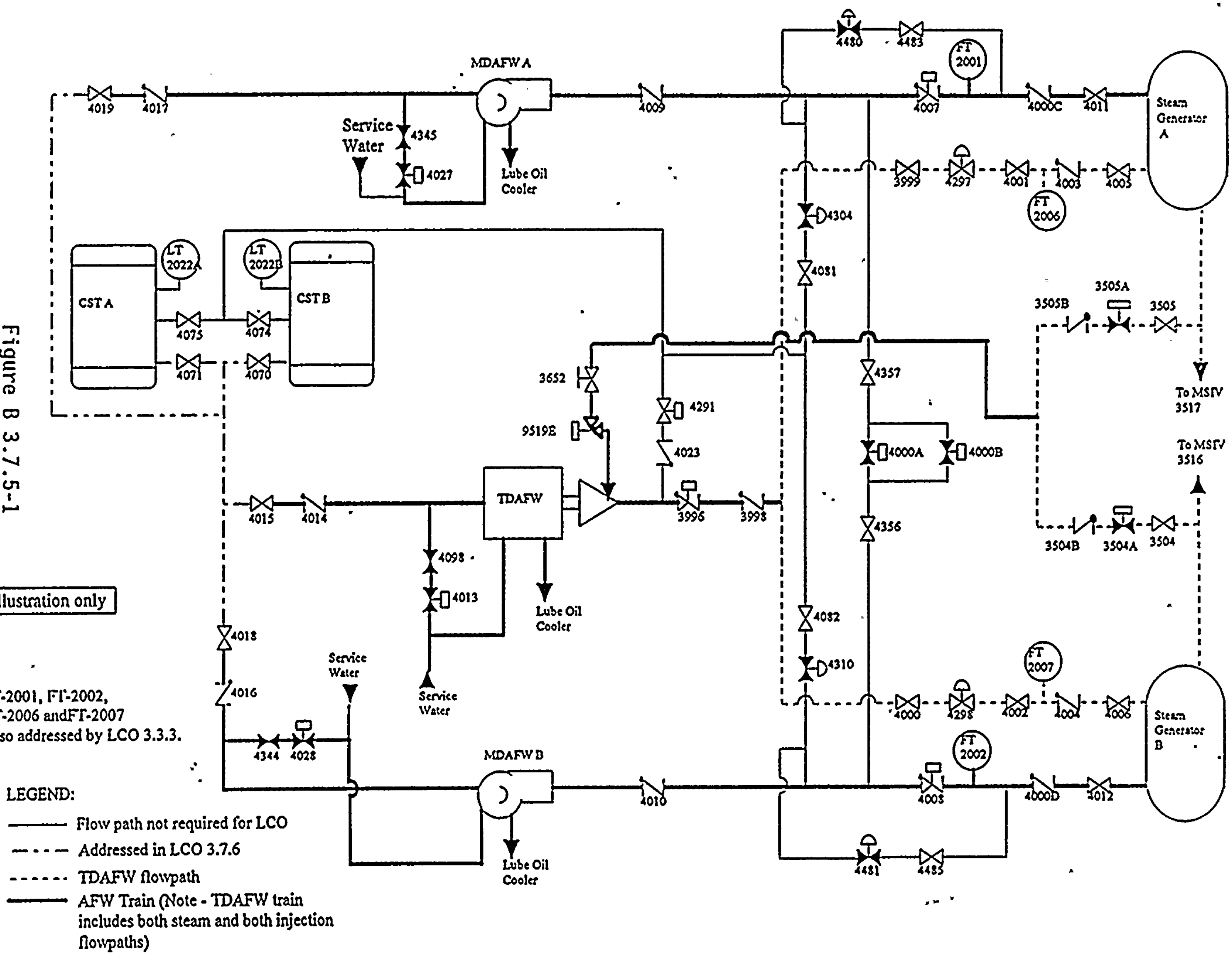
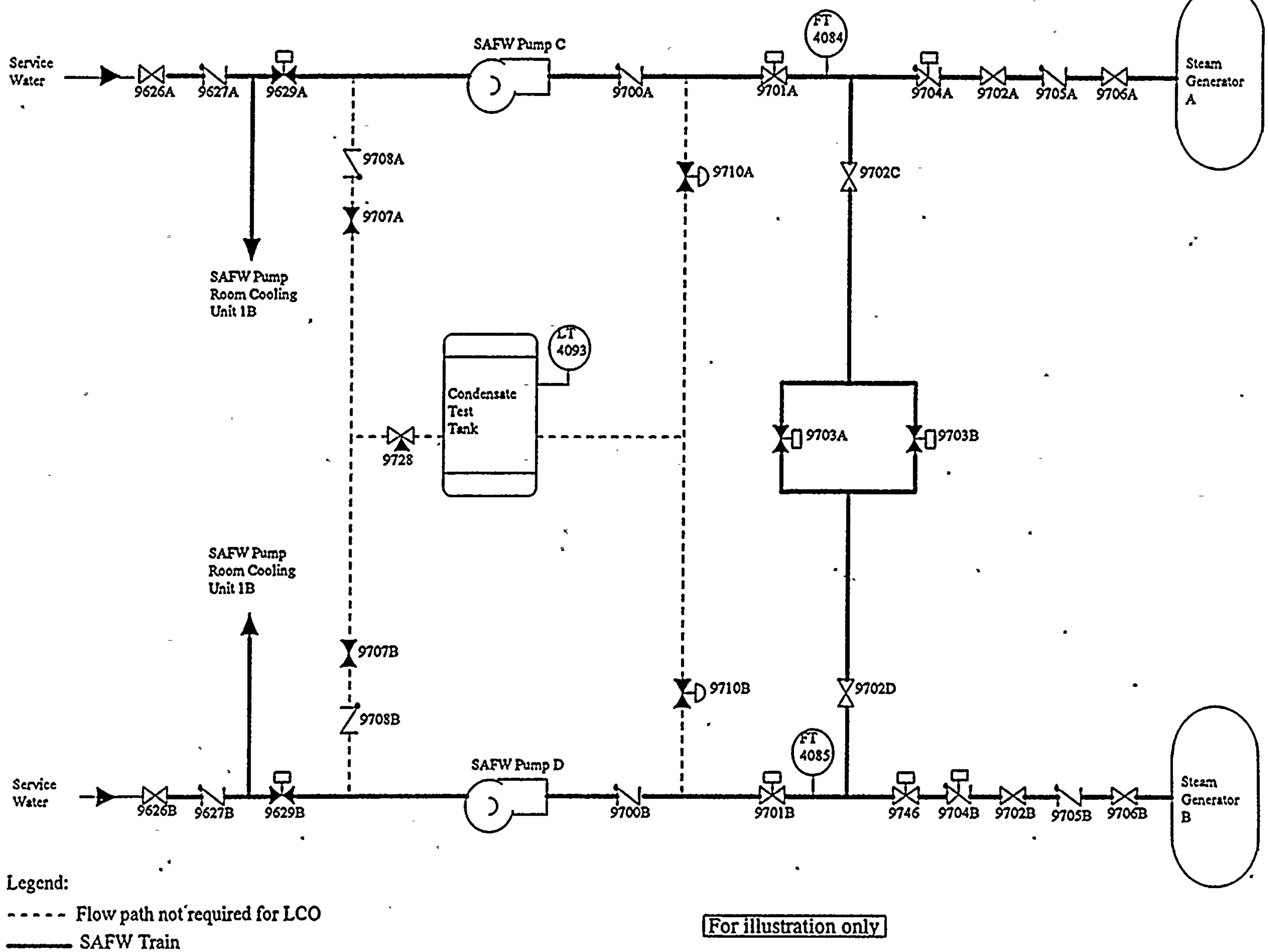


Figure B 3.7.5-1
Preferred AFW System

Figure B 3.7.5-2
Standby AFW System



BASES

BACKGROUND (continued)

The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. Since the removal of decay heat via the RHR System is only performed during the recirculation phase of an accident, the CCW pumps do not receive an automatic start signal. Following the generation of a safety injection signal, the normally operating CCW pump will remain in service unless an undervoltage signal is present on either Class 1E electrical Bus 14 or Bus 16 at which time the pump is stripped from its respective bus. A CCW pump can then be manually placed into service prior to switching to recirculation operations which would not be required until a minimum of 22.4 minutes following an accident.

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for one CCW train and one CCW heat exchanger to remove the loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase. The Emergency Core Cooling System (ECCS) and containment models for a LOCA each consider the minimum performance of the CCW System. The normal temperature of the CCW is ≤ 100 °F, and, during LOCA conditions, a maximum temperature of 120°F is assumed. This prevents the CCW System from exceeding its design temperature limit of 200°F, and provides for a gradual reduction in the temperature of containment sump fluid as it is recirculated to the Reactor Coolant System (RCS) by the ECCS pumps. The CCW System is designed to perform its function with a single failure of any active component, assuming a coincident loss of offsite power.

The CCW trains, heat exchangers, and loop headers are manually placed into service prior to the recirculation phase of an accident (i.e., 22.4 minutes following a large break LOCA).

(continued)

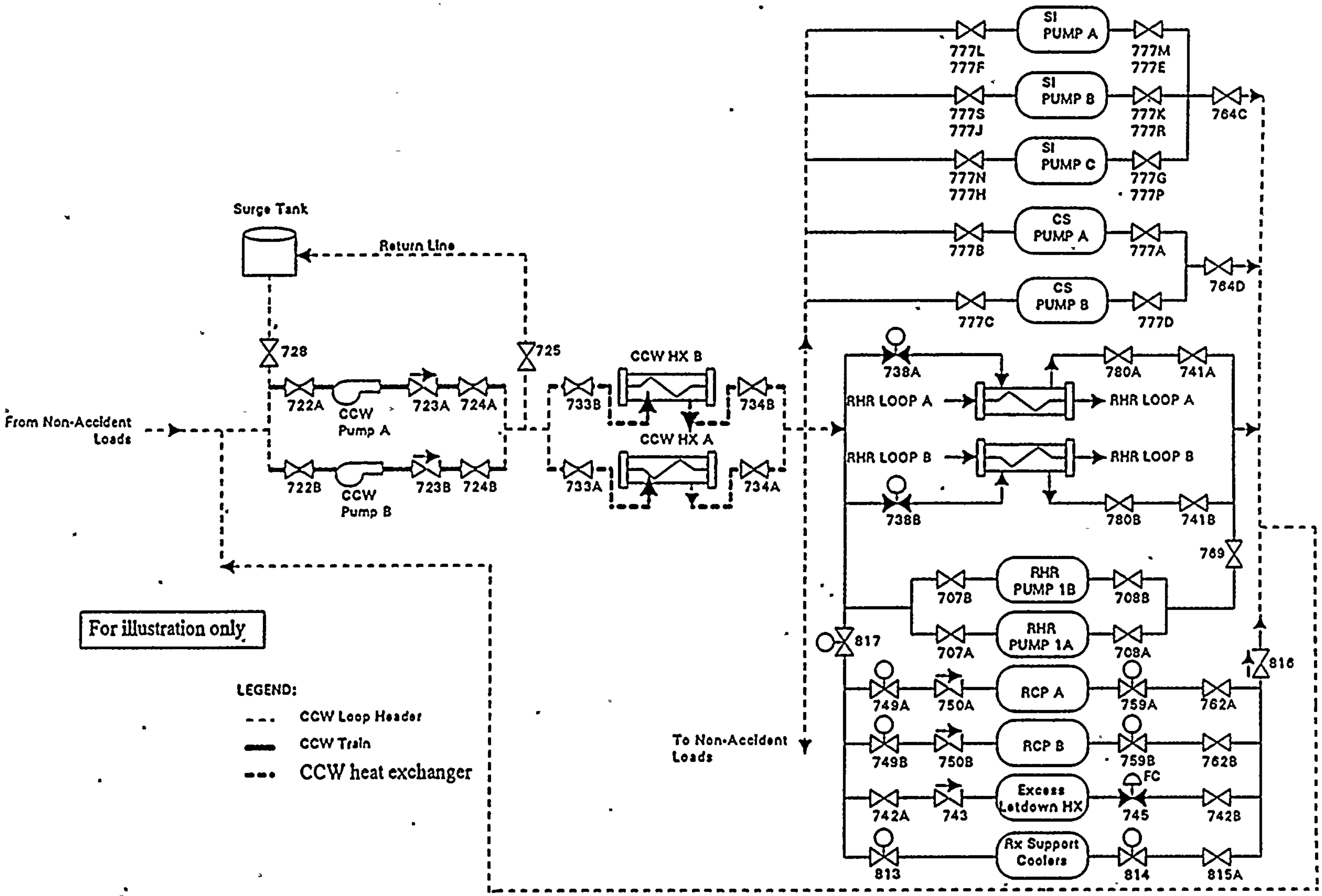


Figure B.3.7.7-1
CCW System

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SW trains and loop header are assumed to supply to following components following an accident:

a. The CRFCs, DGs and safety injection pump bearing housing coolers immediately following a safety injection signal (i.e., after the loop header becomes refilled);

b. The preferred AFW and SAFW pumps within 10 minutes following receipt of a low SG level signal; and

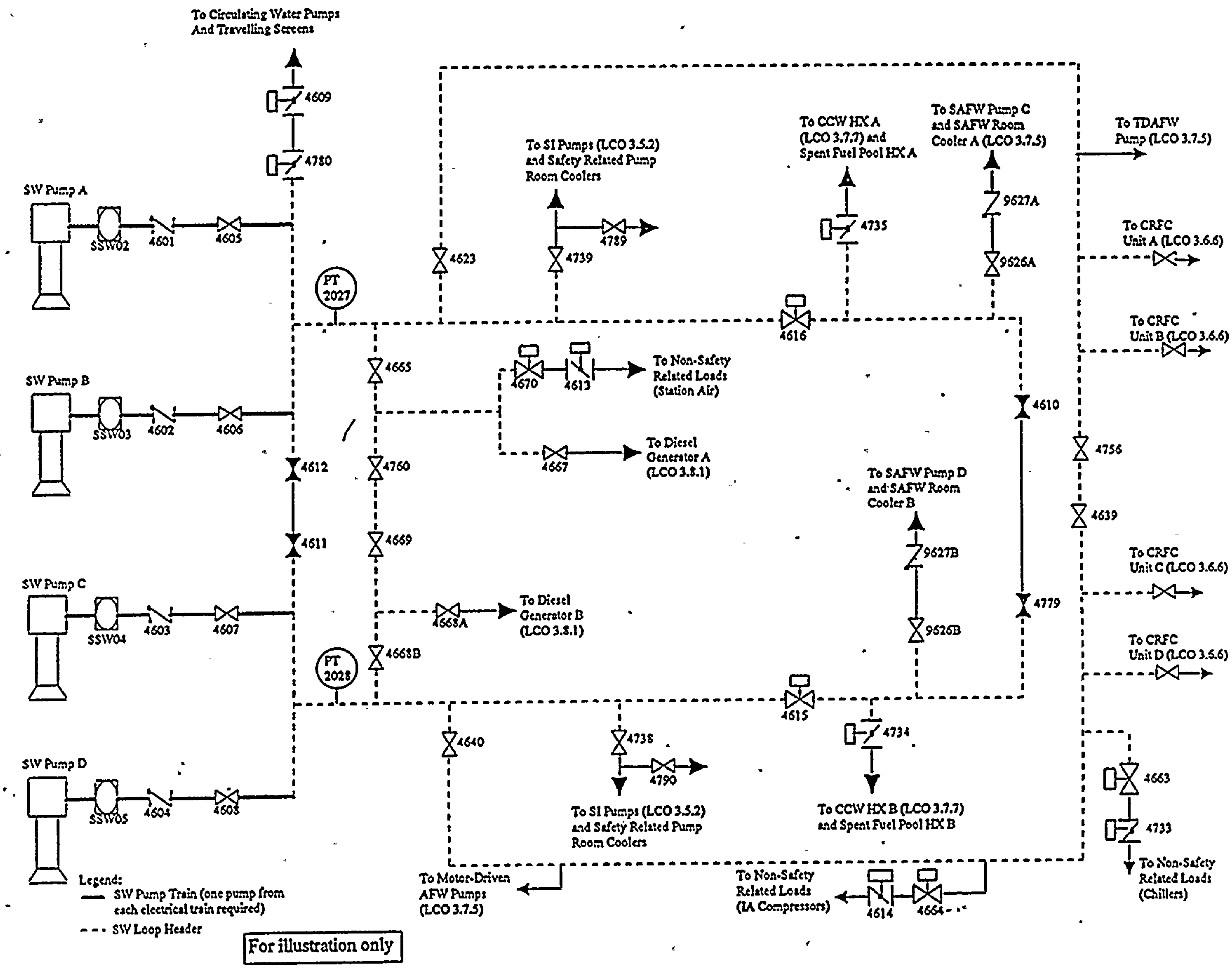
c. The CCW heat exchangers within 22.4 minutes following a safety injection signal.

The SW system, in conjunction with the CCW System, can also cool the plant from residual heat removal (RHR) entry conditions (T_{avg} < 350°F) to MODE 5 (T_{avg} < 200°F) during normal operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR System trains that are operating. Since SW is comprised of a large loop header, a passive failure can be postulated during this cooldown period which results in failing the SW System to potentially multiple safety related functions. The SW system has been evaluated to demonstrate the capability to meet cooling needs with an assumed 500 gal leak. The SW System is also vulnerable to external events such as tornados. The plant has been evaluated for the loss of SW under these conditions with the use of alternate cooling mechanisms (e.g., providing for natural circulation using the atmospheric relief valves and the AFW Systems) with acceptable results (Ref. 1).

The temperature of the fluid supplied by the SW System is also a consideration in the accident analyses. If the cooling water supply to the containment recirculation fan coolers and CCW heat exchangers is too warm, the accident analyses with respect to containment pressure response following a SLB and the containment sump fluid temperature following a LOCA may no longer be bounding. If the cooling water supply is too cold, the containment heat removal systems may be more efficient than assumed in the accident analysis. This causes the backpressure in containment to be reduced which potentially results in increased peak clad temperatures.

(continued)

Figure B 3.7.8-1
SW System



BASES

APPLICABLE SAFETY ANALYSES (continued)

DG Load	DG A Time	DG B Time
480V safeguards buses and CS pumps	10	10
SI pump A and B	10	10
SI pump C	15	17
Residual heat removal pump	20	22
Selected service water pump	25	27
First containment recirculation fan cooler	30	32
Second containment recirculation fan cooler	35	37
Motor driven auxiliary feedwater pump	40	42

The pumps and fans are assumed to be running within 5 seconds following breaker closure.

Since the DGs must start and begin loading within 10 seconds, only one air start must be available in the air receivers as assumed in the accident analyses. The long term operation of the DGs (until offsite power is restored) is discussed in LCO 3.8.3, "Diesel Fuel Oil."

The AC sources satisfy Criterion 3 of NRC Policy Statement.

LCO

One qualified independent offsite power circuit connected between the offsite transmission network and the onsite 480 V safeguards buses and separate and independent DGs for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA.

An OPERABLE qualified independent offsite power circuit is one that is capable of maintaining rated voltage, and accepting required loads during an accident, while connected to the 480 V safeguards buses required by LCO 3.8.9, "Distribution Subsystems—MODES 1, 2, 3, and 4." Power from either offsite power circuit 751 or 767 satisfies this requirement.

(continued)

BASES

LCO (continued)

A DG is considered OPERABLE when:

a. The DG is capable of starting, accelerating to rated speed and voltage, and connecting to its respective 480 V safeguards buses on detection of bus undervoltage within 10 seconds;

(continued)

BASES

LCO (continued)

b. All loads on each 480 V safeguards bus except for the safety related motor control centers, CCW pump, and CS pump are capable of being tripped on an undervoltage signal (CCW pump must be capable of being tripped on coincident SI and undervoltage signal);

c. The DG is capable of accepting required loads both manually and within the assumed loading sequence intervals following a coincident SI and undervoltage signal, and continue to operate until offsite power can be restored to the safeguards bus (i.e., 40 hours);

d. The DG day tank is available to provide fuel oil for ≥ 1 hour at 110% design loads;

e. The fuel oil transfer pump from the fuel oil storage tank to the associated day tank is OPERABLE including all required piping, valves, and instrumentation (long-term fuel oil supplies are addressed by LCO 3.8.3, "Diesel Fuel Oil"); and

f. A ventilation train consisting of at least one of two fans and the associated ductwork and dampers is OPERABLE.

g. The service water (SW) Δp through the diesel generator heat exchangers is < 31 psid with two SW pumps operating and < 44 psid with three SW pumps operating.

The AC sources in one train must be separate and independent of the AC sources in the other train. For the DGs, separation and independence must be complete assuming a single active failure. For the independent offsite power source, separation and independence are to the extent practical (i.e., operation is preferred in the 50/50 mode, but may also exist in the 100/0 or 0/100 mode).

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

(continued)

BASES

APPLICABILITY (continued)

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

(continued)

BASES

LCO (continued)

A DG is considered OPERABLE when:

a. The DG is capable of starting, accelerating to rated speed and voltage, and connecting to its respective 480 V safeguards buses on detection of bus undervoltage within 10 seconds;

b. All loads on each 480 V safeguards bus except for the safety related motor control centers, component cooling water (CCW) pump, and containment spray (CS) pump are capable of being tripped on an undervoltage signal (CCW pump must be capable of being tripped on coincident safety injection (SI) and undervoltage signal);

c. The DG is capable of accepting required loads manually. Since most equipment which receives a SI signal are isolated in these MODES due to maintenance or low temperature overpressure protection concerns, and the DBA of concern (i.e., a fuel handling accident) would not generate a SI signal, manual loading of the DGs will most likely be required. These loads must be capable of being added to the OPERABLE DG within 10 minutes;

d. The DG day tank is available to provide fuel oil for ≥ 1 hour at 110% design loads;

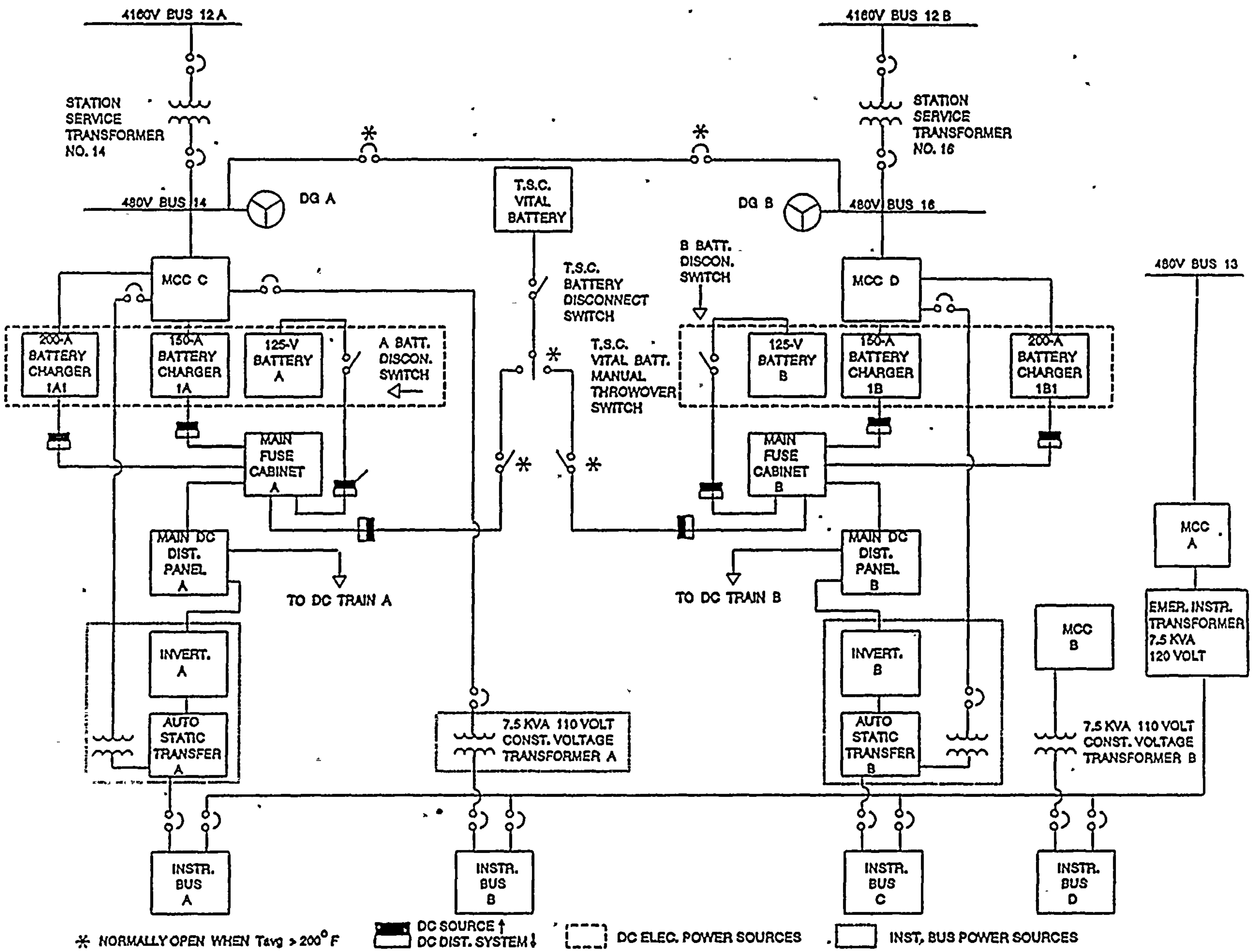
e. The fuel oil transfer pump from the fuel oil storage tank to the associated day tank is OPERABLE including all required piping, valves, and instrumentation (long-term fuel oil supplies are addressed by LCO 3.8.3, "Diesel Fuel Oil"); and

f. A ventilation train consisting of at least one of two fans and the associated ductwork and dampers is OPERABLE.

g. The service water (SW) Δp through the diesel generator heat exchanger is < 31 psid with two SW pumps operating and < 44 psid with three SW pumps operating.

(continued)

Figure B 3.8.4-1



BASES (continued)

LCO

Various combinations of AC, DC, and AC instrument bus electrical power distribution subsystems, trains within these subsystems, and equipment and components within these trains are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components—all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

The LCOs which apply when the Reactor Coolant System is ≤ 200°F and which may require a source of electrical power are:

LCO 3.1.1	SHUTDOWN MARGIN (SDM)
LCO 3.3.1	Reactor Trip System (RTS) Instrumentation
LCO 3.3.4	Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation
LCO 3.3.5	Containment Ventilation Isolation Instrumentation
LCO 3.3.6	Control Room Emergency Air Treatment System (CREATS) Actuation
LCO 3.4.7	RCS Loops - MODE 5, Loops Filled
LCO 3.4.8	RCS Loops - MODE 5, Loops Not Filled
LCO 3.4.12	Low Temperature Overpressure Protection (LTOP) System
LCO 3.7.9	Control Room Emergency Air Treatment System (CREATS)
LCO 3.9.2	Nuclear Instrumentation
LCO 3.9.4	Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft
LCO 3.9.5	Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft

Maintaining the necessary trains of the AC, DC, and AC instrument bus electrical power distribution subsystems energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

(continued)

BASES

LCO (continued)

Bus-tie breakers required to be open during MODES 1, 2, 3, and 4 per SR 3.8.9.1 may be closed during MODES 5 and 6 provided that the distribution system alignment continues to support systems necessary to mitigate the postulated events assuming either a loss of all offsite power, loss of all onsite DG power, or a worst case single failure. The postulated events during MODES 5 and 6 include a boron dilution event and fuel handling accident. Examples of allowed configurations are as follows (note that other configurations are acceptable provided that they meet the above criteria):

a. Bus-Tie Breakers 16-15 and 14-13 (and their associated "dummy" breakers on non-safeguards Buses 13 and 15) provide the capability to cross-tie the safeguards and non-safeguards 480 V buses. Closure of these bus-ties is allowed provided that the OPERABLE DG per LCO 3.8.2 can accept all loads which would be automatically loaded from the safeguards and non-safeguards buses, and accept those loads which must be manually loaded to mitigate the accident.

b. Bus-Tie Breakers 14-16, 16-14, and 17-18 provide the capability to cross-tie the two safeguard electrical trains. Closure of these bus-ties is allowed provided that the OPERABLE DG per LCO 3.8.2 can accept all loads which would be automatically loaded, and accept those loads which must be manually loaded to mitigate the accident. In addition, the automatic trip logic of the bus-ties due to an undervoltage signal from either of the two cross-tied buses must be OPERABLE. This trip logic ensures that upon a fault of either 480 V safeguards bus as the single failure, the redundant bus is capable of mitigating the accident using either the DG or offsite power.

(continued)

BASES (continued)

APPLICABILITY

The AC, DC, and AC instrument bus electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the effects of a postulated event and maintain the plant in the cold shutdown or refueling condition are available.

The AC, DC, and AC instrument bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9, "Distribution Systems—MODES 1, 2, 3, and 4."

ACTIONS

A.1

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and operations involving positive reactivity additions. By allowing the option to declare required features associated with an inoperable distribution subsystem or train inoperable, appropriate restrictions are implemented in accordance with the LCO ACTIONS of the affected required features.

A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

With one or more required electrical power distribution subsystems or trains inoperable, the option exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Therefore, immediate suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions is an acceptable option to Required Action A.1. Performance of Required Actions A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control within established procedures.

(continued)

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5 (continued)

It is further required to immediately initiate action to restore the required AC, DC, and AC instrument bus electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

In addition to performance of the above conservative Required Actions, a required residual heat removal (RHR) loop may be inoperable. In this case, Required Actions A.2.1, A.2.2, A.2.3, and A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.5 requires declaring RHR inoperable, which results in taking the appropriate RHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the electrical power distribution trains are functioning properly, with all the required power source circuit breakers closed, required tie-breakers open, and the required buses energized from their allowable power sources. Required voltage for the AC power distribution electrical subsystem is ≥ 420 VAC, for the DC power distribution electrical subsystem ≥ 108.6 VDC, and for AC instrument bus power distribution electrical subsystem is between 113 VAC and 123 VAC. Required voltage for the twinco panels supplied by the 120 VAC instrument buses is between 115.6 VAC and 120.4 VAC. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The Frequency of 7 days takes into account the capability of the AC, DC, and AC instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

None.

BASES (continued)

LCO

This LCO requires two source range neutron flux monitors be OPERABLE to ensure redundant monitoring capability is available to detect changes in core reactivity.

To be OPERABLE, each monitor must provide visual indication and at least one of the two monitors must provide an audible count rate function in the control room.

With the discharge of fuel from core positions adjacent to source range detector locations, counts decreasing to zero is the expected response. Based on this indication alone, source range detection should not be considered inoperable. Following a full core discharge, source range response is verified with the initial fuel assemblies reloaded.

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity conditions in this MODE. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

<u>A.1 and A.2</u>

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Actions A.1 and A.2 shall not preclude completion of movement of a component to a safe position (i.e., other than normal cooldown of the coolant volume for the purpose of system temperature control within established procedures).

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

With no source range neutron flux monitor OPERABLE there are no direct means of detecting changes in core reactivity. Therefore, actions to restore a monitor to OPERABLE status shall be initiated immediately and continue until a source range neutron flux monitor is restored to OPERABLE status.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

This SR is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one monitor to a similar parameter on another monitor. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range monitors, but each monitor should be consistent with its local conditions.

The inoperability of one source range neutron flux channel prevents performance of a CHANNEL CHECK for the operable channel. However, the Required Actions for the inoperable channel requires suspension of CORE ALTERATIONS and positive reactivity addition such that the CHANNEL CHECK of the operable channel can consist of ensuring consistency with known core conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

SR 3.9.2.2

This SR is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to baseline data. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 7.7.3.2.

2. Atomic Industrial Forum (AIF) GDC 13 and 19, Issued for Comment July 10, 1967.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 5, there are no accidents of concern which require containment. In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be bolted in place. Good engineering practice dictates that a minimum of 4 bolts be used to hold the equipment hatch in place and that the bolts be approximately equally spaced. As an alternative, the equipment hatch opening can be isolated by a closure plate that restricts air flow from containment or by an installed roll up door and enclosure building.

(continued)

BASES

BACKGROUND (continued)

The containment equipment and personnel air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of plant shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed in the personnel and equipment hatch (unless the equipment hatch is isolated by a closure plate or the roll up door and associated enclosure building).

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System includes two subsystems. The Shutdown Purge System includes a 36 inch purge penetration and a 36 inch exhaust penetration. The second subsystem, a Mini-Purge System, includes a 6 inch purge penetration and a 6 inch exhaust penetration. During MODES 1, 2, 3, and 4, the shutdown purge and exhaust penetrations are isolated by a blind flange with two O-rings that provide the necessary boundary. The two air operated valves in each of the two mini-purge penetrations can be opened intermittently, but are closed automatically by the Containment Ventilation Isolation Instrumentation System. Neither of the subsystems is subject to a Specification in MODE 5.

(continued)

BASES (continued)

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that at least one valve in each of these penetrations is isolable by the Containment Ventilation Isolation System.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions, no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

If the containment equipment hatch (or its closure plate or roll up door and associated enclosure building), air lock doors, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Ventilation Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the plant must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

(continued)