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DEC 2 7 2005

PSEG
Nuclear LLC

LR-N05-0469
TS Bases Changes SCN 05-042
SCN 05-048
SCN 05-049
SCN 05-052

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

TECHNICAL SPECIFICATION BASES CHANGES SCN 05-042, SCN 05-048, SCN 05-049 AND SCN 05-052
SALEM GENERATING STATION UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

PSEG Nuclear, LLC (PSEG) has revised the Technical Specification (TS) Bases as described below. The changes were reviewed in accordance with the requirements of the Technical Specification Bases Control Program and 10 CFR 50.59.

Unit 1 and Unit 2 TS Bases 3/4.1.2, Boration Systems, were changed to maintain consistency with changes to the Technical Specifications approved in Amendment Nos. 264 and 246.

Unit 1 TS Bases, 3/4.4.5, Steam Generator Tube Integrity, and 3/4.4.6.2, Operational Leakage, were changed to maintain consistency with changes to the Technical Specifications approved in Unit 1 Amendment No. 268.

Unit 1 TS Bases 3/4.6.1.7, Containment Ventilation System, was added to describe the basis for the Limiting Condition for Operation and methods for meeting the required Action.

Unit 1 and Unit 2 TS Bases 3/4.6.2.1, Containment Spray System, and 3/4.6.2.3, Containment Cooling System, were changed to maintain consistency with changes to the Technical Specifications approved in Amendment Nos. 266 and 248.

A001

Attachment 1 contains the revised pages for the Salem Unit 1 Technical Specification Bases. Attachment 2 contains the revised pages for the Salem Unit 2 Technical Specification Bases. In accordance with the TS Bases Control Program, PSEG has incorporated these changes into the Bases.

Should you have any questions regarding this transmittal, please contact Mr. Paul Duke at (856) 339-1466.

Sincerely,

Darin Benyak

Darin Benyak
Director - Regulatory Assurance

Attachments (2)

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Attachment 1

LR-N05-0469
TS Bases Changes SCN 05-042
SCN 05-048
SCN 05-049
SCN 05-052

SALEM GENERATING STATION UNIT NO. 1
FACILITY OPERATING LICENSE NO. DPR-70
DOCKET NO. 50-272
REVISIONS TO THE TECHNICAL SPECIFICATIONS BASES

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3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, and 5) offsite power or an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature ≥ 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability (minimum boration volume) requirement is established to conservatively bound expected operating conditions throughout core operating life. The analysis assumes that the most reactive control rod is not inserted into the core. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires borated water from a boric acid tank in accordance with TS Figure 3.1-2, and additional makeup from either: (1) the second boric acid tank and/or batching, or (2) a maximum of 41,800 gallons of 2,300 ppm borated water from the refueling water storage tank. With the refueling water storage tank as the only borated water source, a maximum of 73,800 gallons of 2,300 ppm borated water is required. However, to be consistent with the ECCS requirements, the RWST is required to have a minimum contained volume of 364,500 gallons during operations in MODES 1, 2, 3 and 4.

The boric acid tanks, pumps, valves, and piping contain a boric acid solution concentration of between 3.75% and 4.0% by weight. To ensure that the boric acid remains in solution, the tank fluid temperature and the process pipe wall temperatures are monitored to ensure a temperature of 63°F, or above is maintained. The tank fluid and pipe wall temperatures are monitored in the main control room. A 5°F margin is provided to ensure the boron will not precipitate out.

Should ambient temperature decrease below 63°F, the boric acid tank heaters, in conjunction with boric acid pump recirculation, are capable of maintaining the boric acid in the tank and in the pump at or above 63°F. A small amount of boric acid in the flow path between the boric acid recirculation line and the suction line to the charging pump will precipitate out, but it will not cause flow blockage even with temperatures below 50°F.

With the RCS temperature below 350°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to assure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE assures that the plant will be able to establish natural circulation.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.i, "Steam Generator (SG) Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that significantly affect burst or collapse. In that context, the term "significant" is defined as, "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." The determination of whether thermal loads are primary or secondary loads is based on the ASME definition

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (Continued)

in which secondary loads are self-limiting and will not cause failure under single load application. For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB and draft Reg. Guide 1.121.

The accident induced leakage performance criterion ensures that the primary-to-secondary leakage caused by a design basis accident, other than a steam generator tube rupture (SGTR), is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm per SG. The accident induced leakage rate includes any primary-to-secondary leakage existing prior to the accident in addition to primary-to-secondary leakage induced during the accident.

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2, "Operational Leakage," and limits primary-to-secondary leakage through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

The ACTION requirements are modified by a Note clarifying that the Actions may be entered independently for each SG tube. This is acceptable because the ACTION requirements provide appropriate compensatory actions for each affected SG tube. Complying with the ACTION requirements may allow for continued operation, and subsequent affected SG tubes are governed by subsequent ACTION requirements.

If it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program, an evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. An action time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (Continued)

SG tube that may not have tube integrity. If the evaluation determines that the affected tube(s) have tube integrity, plant operation is allowed to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering HOT SHUTDOWN following the next refueling outage or SG inspection. This action time is acceptable since operation until the next inspection is supported by the operational assessment.

If SG tube integrity is not being maintained or the ACTION requirements are not met, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. The action times are reasonable based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

During shutdown periods the SGs are inspected as required by surveillance requirements and the Steam Generator Program. NEI 97-06, "Steam Generator Program Guidelines," and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period. The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find existing and potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities and inspection locations. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines. The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.4.i contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.8.4.i are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in size measurement and future growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). NEI 97-06 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria. The Frequency of prior to entering HOT SHUTDOWN following a SG inspection

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (Continued)

ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

<u>Primary-to-Secondary Leakage Through Any One SG</u>

The primary-to-secondary leakage rate limit applies to leakage through any one Steam Generator. The limit of 150 gallons per day per steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines. The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator program is an effective measure for minimizing the frequency of steam generator tube ruptures.

<u>Actions</u>

Unidentified leakage or identified leakage in excess of the LCO limits must be reduced to within limits within 4 hours. This action time allows time to verify leakage rates and either identify unidentified leakage or reduce leakage to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the reactor coolant pressure boundary (RCPB). If any pressure boundary leakage exists, or primary-to-secondary leakage is not within limit, or if unidentified or identified leakage cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

its potential consequences. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. The reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary. The action 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. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Surveillances

Verifying RCS leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. Unidentified leakage and identified leakage are determined by performance of an RCS water inventory balance. The RCS water inventory must be met with the reactor at steady state conditions. The surveillance is modified by a Note that the surveillance is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The 72 hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If SR 4.4.6.2.c is not met, compliance with LCO 3.4.5, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines). If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one Steam Generator. The Surveillance is modified by a Note which states that the surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary-to-secondary leakage determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The Surveillance Frequency of 72 hours is a reasonable interval to trend primary-to-secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines).

3/4.4.7

THIS SECTION DELETED

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig and 2) the containment peak pressure does not exceed the design pressure of 47 psig during the limiting pipe break conditions. The pipe breaks considered are LOCA and steam line breaks.

The limit of 0.3 psig for initial positive containment pressure is consistent with the accident analyses initial conditions.

The maximum peak pressure expected to be obtained from a LOCA or steam line break event is ≤ 47 psig.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA or steam line break. In order to determine the containment average air temperature, an average is calculated using measurements taken at locations within containment selected to provide a representative sample of the overall containment atmosphere.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the design pressure. The visual inspections of the concrete and liner and the Type A leakage test both in accordance with the Containment Leakage Rate Testing Program are sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA. Maintaining these valves (or equivalent isolation device) closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system, when operated in conjunction with the Containment Cooling System, ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

Normal plant operation and maintenance practices are not expected to trigger surveillance requirement 4.6.2.1.d. Only an unanticipated circumstance would initiate this surveillance, such as inadvertent spray actuation, a major configuration change, or a loss of foreign material control when working within the affected boundary of the system. If an activity occurred that presents the potential of creating nozzle blockage, an evaluation would be performed by the engineering organization to determine if the amount of nozzle blockage would impact the required design capabilities of the containment spray system. If the evaluation determines that the containment spray system would continue to perform its design basis function, then performance of the air or smoke flow test would not be required. If the evaluation cannot conclusively determine the impact to the containment spray system, then the air or smoke flow test would be performed to determine if any nozzle blockage has occurred.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration, ensure that 1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and 2) corrosion effects on components within containment are minimized. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

The surveillance requirements for the service water accumulator vessels ensure each tank contains sufficient water and nitrogen to maintain water filled, subcooled fluid conditions in three containment fan coil unit (CFCU) cooling loops in response to a loss of offsite power, without injecting nitrogen covergas into the containment fan coil unit loops assuming the most limiting single failure. The surveillance requirement for the discharge

Attachment 2

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SCN 05-048
SCN 05-049
SCN 05-052

SALEM GENERATING STATION UNIT NO. 2
FACILITY OPERATING LICENSE NO. DPR-75
DOCKET NO. 50-311
REVISIONS TO THE TECHNICAL SPECIFICATIONS BASES

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, and 5) offsite power or an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature ≥ 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability (minimum boration volume) requirement is established to conservatively bound expected operating conditions throughout core operating life. The analysis assumes that the most reactive control rod is not inserted into the core. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires borated water from a boric acid tank in accordance with TS Figure 3.1-2, and additional makeup from either: (1) the second boric acid tank and/or batching, or (2) a maximum of 41,800 gallons of 2,300 ppm borated water from the refueling water storage tank. With the refueling water storage tank as the only borated water source, a maximum of 73,800 gallons of 2,300 ppm borated water is required. However, to be consistent with the ECCS requirements, the RWST is required to have a minimum contained volume of 350,000 gallons during operations in MODES 1, 2, 3 and 4.

The boric acid tanks, pumps, valves, and piping contain a boric acid solution concentration of between 3.75% and 4% by weight. To ensure that the boric acid remains in solution, the tank fluid temperature and the process pipe wall temperatures are monitored to ensure a temperature of 63°F, or above is maintained. The tank fluid and pipe wall temperatures are monitored in the main control room. A 5°F margin is provided to ensure the boron will not precipitate out.

Should ambient temperature decrease below 63°F, the boric acid tank heaters, in conjunction with boric acid pump recirculation, are capable of maintaining the boric acid in the tank and in the pump at or about 63°F. A small amount of boric acid in the flowpath between the boric acid recirculation line and the suction line to the charging pump will precipitate out, but it will not cause flow blockage even with temperatures below 50°F.

With the RCS temperature below 350°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE OPERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system, when operated in conjunction with the Containment Cooling System, ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

The containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

Normal plant operation and maintenance practices are not expected to trigger surveillance requirement 4.6.2.1.d. Only an unanticipated circumstance would initiate this surveillance, such as inadvertent spray actuation, a major configuration change, or a loss of foreign material control when working within the affected boundary of the system. If an activity occurred that presents the potential of creating nozzle blockage, an evaluation would be performed by the engineering organization to determine if the amount of nozzle blockage would impact the required design capabilities of the containment spray system. If the evaluation determines that the containment spray system would continue to perform its design basis function, then performance of the air or smoke flow test would not be required. If the evaluation cannot conclusively determine the impact to the containment spray system, then the air or smoke flow test would be performed to determine if any nozzle blockage has occurred.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration, ensure that 1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and 2) corrosion effects on components within containment are minimized. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.