BASES

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of 17 of the below listed 18 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed with OPERABLE position indication.

a. 4 safety/relief valves @ 1205 psig +1%, -3%
b. 4 safety/relief valves @ 1195 psig +1%, -3%
c. 4 safety/relief valves @ 1185 psig +1%, -3%
d. 4 safety/relief valves @ 1175 psig +1%, -3%
e. 2 safety/relief valves @ 1150 psig +1%, -3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 110°F, close the stuck-open relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

b ~~c.~~ With one or more of the above required safety/relief valve stem position indicators inoperable, restore the inoperable stem position indicators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.4.2.1 The safety/relief valve stem position indicators of each safety/relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 31 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.**

4.4.2.2 The low-low set function shall be demonstrated not to interfere with the OPERABILITY of the safety relief valves or the ADS by performance of a CHANNEL CALIBRATION at least once per 18 months.

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*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

#Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling outage.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

INFO ONLY - NO CHANGES

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 131,900 ft^3 and 128,800 ft^3, equivalent to a level between +3 inches** and -4 1/2 inches**, and a

2. Maximum average temperature of 105°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

b) 120°F with the main steam line isolation valves closed following a scram.

b. Drywell-to-suppression charmer bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 0.03 ft^2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than or equal to 105°F, stop all testing which adds heat to the suppression pool, and restore the average temperature to less than or equal to 105°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

2. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

#See Specification 3.5.3 for ECCS requirements.

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

INFO ONLY - NO CHANGES

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. Analysis has shown that with the safety function of one of the eighteen safety/relief valves inoperable the reactor pressure is limited to within ASME III allowable values for the worst case upset transient. Therefore, operation with any 17 SRV's capable of opening is allowable, although all installed SRV's must be closed and have position indication to ensure that integrity of the primary coolant boundary is known to exist at all times.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 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.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the higher limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so high concentrations of chlorides are not considered harmful during these periods.

INFO ONLY-NO CHANGES

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 45 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1020 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 45 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. (See Figure B 3/4.6.2-1)

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 39.6 psig which is below the design pressure of 45 psig. Maximum water volume of 131,900 ft^3 results in a downcomer submergence of 12.4 ft and the minimum volume of 128,800 ft^3 results in a submergence approximately 8 inches less. The majority of the Bogeda tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 135°F immediately following blowdown which is below the 200°F used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak bulk temperature of the suppression pool is maintained below 200°F during any period of relief valve operation with sonic conditions at the discharge exit for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

when suppression pool average water temperature is 110 °F or greater

DEPRESSURIZATION SYSTEMS (Continued)

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event of safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, ~~and~~ (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

Primary Containment Isolation Valves (PCIVs) form a part of the primary containment boundary. The PCIV safety function is related to control of primary containment leakage rates during accidents or other conditions to limit the untreated release of radioactive materials from the containment in excess of the design limits.

The automatic isolation valves are required to have isolation times within limits and actuate on an automatic isolation signal. The valves covered by this specification are listed with their associated stroke times, and other design information for lines penetrating the Primary Containment, in UFSAR Section 6.2.

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact.

Main steam lines through the isolation valves and hydrostatically tested valves must meet alternative leakage rate requirements. Other PCIV leakage rates are addressed by specification 3/4.6.1.1, "PRIMARY CONTAINMENT INTEGRITY". UFSAR Section 6.2 also describes special leakage test requirements and exemptions.

This specification provides assurance that the PCIVs will perform their designed safety functions to control leakage from the primary containment during accidents.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication

BASES

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of 17 of the below listed 18 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed with OPERABLE position indication.

a. 4 safety/relief valves @1205 psig ±3%
b. 4 safety/relief valves @1195 psig ±3%
c. 4 safety/relief valves @1185 psig ±3%
d. 4 safety/relief valves @1175 psig ±3%
e. 2 safety/relief valves @1150 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 110°F, close the stuck open relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

b ~~c.~~ With one or more of the above required safety/relief valve stem position indicators inoperable, restore the inoperable stem position indicators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The safety/relief valve stem position indicators of each safety/relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 31 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.**

4.4.2.2 The low low set function shall be demonstrated not to interfere with the OPERABILITY of the safety/relief valves or the ADS by performance of a CHANNEL CALIBRATION at least once per 18 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. Following testing, lift settings shall be within ±1%.

#Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling outage.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

INFO ONLY - NO CHANGES

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 131,900 ft^3 and 128,800 ft^3, equivalent to a level between +3 inches** and -4 1/2 inches**, and a

2. Maximum average temperature of 105°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

b) 120°F with the main steam line isolation valves closed following a scram.

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 0.03 ft^2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than or equal to 105°F, stop all testing which adds heat to the suppression pool, and restore the average temperature to less than or equal to 105°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

2. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

#See Specification 3.5.3 for ECCS requirements.
**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

INFO ONLY - NO CHANGES

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. Analysis has shown that with the safety function of one of the eighteen safety/relief valves inoperable the reactor pressure is limited to within ASME III allowable values for the worst case upset transient. Therefore, operation with any 17 SRV's capable of opening is allowable, although all installed SRV's must be closed and have position indication to ensure that integrity of the primary coolant boundary is known to exist at all times.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

INFO ONLY - NO CHANGES

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 45 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1020 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 45 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. (See Figure B 3/4.6.2-1)

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 39.6 psig which is below the design pressure of 45 psig. Maximum water volume of 131,900 ft^3 results in a downcomer submergence of 12.4 ft and the minimum volume of 128,800 ft^3 results in a submergence approximately 8 inches less. The majority of the Bogeda tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 135°F immediately following blowdown which is below the 200°F used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak bulk temperature of the suppression pool is maintained below 200°F during any period of relief valve operation with sonic conditions at the discharge exit for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

when suppression pool average water temperature is 11Ø °F or greater

DEPRESSURIZATION SYSTEMS (Continued)

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event of safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

Primary Containment Isolation Valves (PCIVs) form a part of the primary containment boundary. The PCIV safety function is related to control of primary containment leakage rates during accidents or other conditions to limit the untreated release of radioactive materials from the containment in excess of the design limits.

The automatic isolation valves are required to have isolation times within limits and actuate on an automatic isolation signal. The valves covered by this specification are listed with their associated stroke times, and other design information for lines penetrating the Primary Containment, in UFSAR Section 6.2.

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact.

Main steam lines through the isolation valves and hydrostatically tested valves must meet alternative leakage rate requirements. Other PCIV leakage rates are addressed by specification 3/4.6.1.1, "PRIMARY CONTAINMENT INTEGRITY". UFSAR Section 6.2 also describes special leakage test requirements and exemptions.

ATTACHMENT C

SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated the proposed Technical Specification Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of LaSalle County Station Units 1 and 2 in accordance with the proposed amendment will not:

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

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the UFSAR. A stuck open SRV event is a mild transient which neither affects fuel limits nor radiological consequences. The two minute requirement to manually scram after a SRV becomes stuck open is not assumed or used in any transient or accident analysis in the FSAR. Removing the two minute requirement to manually scram after a SRV becomes stuck open does not change the probability of any accident evaluated in the FSAR. Removing the two minute requirement to manually scram after a SRV becomes stuck open also does not change the capability of the suppression pool during this event in case of any accident involving reactor blowdown, because the suppression pool average water temperature limit in Technical Specification 3.6.2.1 is still valid and enforced. The suppression pool average water temperature limit is the only requirement during operational conditions 1 and 2 that assures sufficient heat sink capacity in case of a LOCA in the containment. Therefore, removing the two minute requirement to manually scram after a SRV becomes stuck open would not increase the probability or consequences of any postulated accident analyzed in the FSAR.

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

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the UFSAR. This change does not effect any hardware. This is a procedural change to assure that the reactor will not be unnecessarily scrammed by the operator after a SRV is stuck open for two minutes. The reactor will still be scrammed if suppression pool average water temperature increases above 110 degrees F. Since the design basis of the suppression pool is protected by this average water temperature limit, this procedural change of removing the two minute requirement to manually scram after a SRV becomes stuck open introduces no new accident or malfunction.

3) Involve a significant reduction in the margin of safety because:

The proposed change does not reduce the margin as defined in the bases for any Technical Specification. On the contrary, if the two minute requirement to manually scram after a SRV becomes stuck open is not removed, the operator has to scram the reactor thus challenging the RPS, the reactor vessel, and other associated components, and reducing the related margin to safety. This scram would be unnecessary if the suppression pool average water temperature is below the 110 degree F limit allowed by the design basis of the suppression pool. Reactor safety or suppression pool design basis is not compromised because the suppression pool average water temperature limit alone guarantees that there would not be any reduction in margin of safety.

Guidance has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. These proposed amendments most closely fit the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the standard review plan.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR Part 51.21. It has been determined that the proposed changes meet the criteria for categorical exclusion as provided for under 10 CFR Part 51.22(c)(9). This conclusion has been determined because the changes requested do not pose significant hazards considerations or do not involve a significant increase in the amounts, and no significant changes in the types of any effluents that may be released off-site. Additionaily, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.