ATTACHMENT B
PROPOSED AMENDMENTS TO THE
LICENSE/TECHNICAL SPECIFICATIONS

NPF-11

3/4 4-5
B 3/4 4-2*

*This page is provided for information only, no changes.

3/4.4.2 SAFETY/RELIEF VALVES

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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.

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.

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INSERT A

**Until Unit 1 enters Cold Shutdown at the end of Unit 1 cycle six or the next Cold Shutdown, whichever comes first, Limiting Condition for Operation 3.4.2 is modified as follows:

1. The number of SRVs required to be OPERABLE is changed from 17 to 18 of the 18 SRVs installed.

2. The provisions of specification 4.0.3 are not applicable to SRVs 1B21-F013B and 1B21-F013J with respect to the safety valve function lift setting setpoint test frequency specified in ASME Code Section XI, Table IWV-3510-1.

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.

ATTACHMENT C

SIGNIFICA*   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 consid-ration 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:

   a. There is no affect on accident initiators so there is no change in probability of an accident. The probability of a failed open Safety/Relief Valve (SRV) is not affected based on observed performance of setpoint drift.

   b. There is no effect or minimal affect on the consequences of analyzed accidents based on an evaluation that the highest reactor vessel pressure that will occur is still less than the Safety Limit of 1325 psig steam dome pressure, for the bounding vessel pressurization event. This evaluation assumed that both SRVs 1B21-F013B and 1B21-F013J fail to open.

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

   The SRVs are not being used in any other mode than original design. The only affect is from the safety mode setpoint drift. This issue does not involve any plant modifications or changes to operating procedures. Therefore, this issue does not create the possibility of a new or different kind of accident from any previously evaluated accident.

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

   The review of previous sensitivity analyses for peak accident pressure indicates that in the worst case postulated (both SRVs fail to open), the peak vessel pressure will not exceed approximately 1276 psig in the reactor bottom head,(1226 psig in the RPV steam dome). The 1276 psig value retains a margin of greater than 50 psig to the ASME limit of 1375 psig for Upset conditions, and will not result in exceeding the Safety Limit reactor pressure of 1325 psig steam dome pressure. Therefore, this issue does not involve a significant reduction in the margin of safety.

ATTACHMENT C

SIGNIFICANT HAZARDS CONSIDERATION

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 acceptable criteria with respect to the system or component specified in the applicable 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 action requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 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. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.