ATTACHMENT B-1

PROPOSED CHANGES TO APPENDIX A,
TECHNICAL SPECIFICATIONS OF FACILITY
OPERATING LICENSES NPF-37 AND NPF-66,
BYRON STATION UNITS 1 & 2

Revision to:		X
		3/4 7-1
		3/4 7-2
		3/4 7-3
	B	3/4 7-1
	B	3/4 7-2

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With four reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5.

INSERT A

a. With up to 3 inoperable main steam line Code safety valves on any one steam generator, within 4 hours, either restore the inoperable valves to OPERABLE status, or reduce the Power Range Neutron Flux High Trip Setpoints per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With more than 3 inoperable main steam line Code safety valves on any one steam generator, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	87
2	65
3	43

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

±3%

VALVE NUMBER	LIFT SETTING (±1%)*#	ORIFICE SIZE
MS013(A-D)	1235 psig	16 in^2
MS014(A-D)	1220 psig	16 in^2
MS015(A-D)	1205 psig	16 in^2
MS016(A-D)	1190 psig	16 in^2
MS017(A-D)	1175 psig	16 in^2

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

#Main Steam line Code safety valve lift settings may have a ±3% tolerance until May 9, 1994, by which time the lift settings will be reset to ±1%.

... to ±1% tolerance.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary Coolant System pressure will be limited to within 110% (1320 psia) of its design pressure of 1200 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 102% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam dumps to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 17.958×10^6 lbs/h which is 119% of the total secondary steam flow of 15.135×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For four loop operation:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109).$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

INSERT B

The requirement that the main steam line Code safety valves be set to within ±1% of the appropriate setpoint is consistent with Section III of the ASME Boiler and Pressure Vessel Code. The allowed operating tolerance of ±3% is supported by the Commonwealth Edison Company, Byron/Braidwood Unit 1& 2 Overpressure Protection Report.

INSERT C

$$High\Phi = \frac{100}{Q}\left(\frac{w_s h_{fg} N}{K}\right)$$

Where:

$High\Phi$	=	Safety Analysis power range high neutron flux setpoint, in percent.
Q	=	Nominal NSSS power rating of the plant (including reactor coolant pump heat), in Mwt (= 3427.6 MWt).
K	=	Conversion factor = 947.82 (BTU/sec.)/MWt.
w_s	=	minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lbm/sec.
h_{fg}	=	Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in BTU/lbm.
N	=	Number of loops in the plant (= 4).

SAFETY VALVES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

The motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. The diesel-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the RHR System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water level of 40% ensures that sufficient water (200,000 gallons) is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line break. This dose also includes the effects of a coincident 1 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

ATTACHMENT B-2

PROPOSED CHANGES TO APPENDIX A,
TECHNICAL SPECIFICATIONS OF FACILITY
OPERATING LICENSES NPF-72 AND NPF-77,
BRAIDWOOD STATION UNITS 1 & 2

Revision to:	X
	3/4 7-1
	3/4 7-2
	3/4 7-3
	B 3/4 7-1
	B 3/4 7-2

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With four reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for Braidwood, Unit 1, Cycle 5, until the initial entry into MODE 2. The provisions of Specification 4.0.4 are not applicable for Braidwood, Unit 2, until the initial entry into Mode 2 following forced outage A2F27.

INSERT A

a. With up to 3 inoperable main steam line Code safety valves on any one steam generator, within 4 hours, either restore the inoperable valves to OPERABLE status, or reduce the Power Range Neutron Flux High Trip Setpoints per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With more than 3 inoperable main steam line Code safety valves on any one steam generator, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	~~87~~ 60
2	~~65~~ 43
3	~~43~~ 25

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (±1%)*#	ORIFICE SIZE
MS013(A-D)	1235 psig	16 in^2
MS014(A-D)	1220 psig	16 in^2
MS015(A-D)	1205 psig	16 in^2
MS016(A-D)	1190 psig	16 in^2
MS017(A-D)	1175 psig	16 in^2

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

#Main steam line Code safety valve lift settings may have a ±3% tolerance until May 9, 1994, by which time the lift settings will be reset to ±1%.

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary Coolant System pressure will be limited to within 110% (1320 psia) of its design pressure of 1200 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 102% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam dumps to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 17.958×10^6 lbs/h which is 119% of the total secondary steam flow of 15.135×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For four loop operation:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109).$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line, SAFETY VALVES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

The motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. The diesel-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the RHR System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water level of 40% ensures that sufficient water (200,000 gallons) is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line break. This dose also includes the effects of a coincident 1 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

INSERT B

The requirement that the main steam line Code safety valves be set to within ±1% of the appropriate setpoint is consistent with Section III of the ASME Boiler and Pressure Vessel Code. The allowed operating tolerance of ±3% is supported by the Commonwealth Edison Company, Byron/Braidwood Unit 1& 2 Overpressure Protection Report.

INSERT C

$$High\Phi = \frac{100}{Q}\left(\frac{w_s h_{fg} N}{K}\right)$$

Where:

$High\Phi$	=	Safety Analysis power range high neutron flux setpoint, in percent.
Q	=	Nominal NSSS power rating of the plant (including reactor coolant pump heat), in Mwt (= 3427.6 MWt).
K	=	Conversion factor = 947.82 (BTU/sec.)/MWt.
w_s	=	minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lbm/sec.
h_{fg}	=	Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in BTU/lbm.
N	=	Number of loops in the plant (= 4).

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Commonwealth Edison has evaluated the proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92 (c), a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

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

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

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

Commonwealth Edison (ComEd) proposes to revise Technical Specification (TS) 3/4.7.1, Turbine Cycle Safety Valves, and the associated Bases. The proposed revisions include:

1. Revising Technical Specification (TS) 3.7.1.1, Action a., to require the unit to be in hot shutdown, rather than cold shutdown, for consistency with NUREG 1431, "Standard Technical Specifications for Westinghouse Plants," and adding a new Action b. to clarify the shutdown requirements when there are more than 3 inoperable main steam line Code safety valves on any one steam generator.

2. Revising Technical Specification Surveillance Requirement (TSSR) 4.7.1.1 to clarify that Specification 4.0.4 does not apply for entry into Mode 3 for Byron and Braidwood, and, for Braidwood only, deleting the one-time requirements for Unit 1, Cycle 5 and Unit 2 after outage A2F27.

3. Revising the maximum allowable power range neutron flux high trip setpoints in Table 3.7-1.

4. Revising Table 3.7-2 to increase the as-found main steam safety valve (MSSV) lift setpoint tolerance to ±3%, provide an as-left setpoint tolerance of ±1%, and change a table notation.

5. Deleting the orifice size column from Table 3.7-2.

6. Revising the Bases for TS 3.7.1.1 to be consistent with the proposed changes to TS 3.7.1.1.

A. **The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The text describing reactor coolant loops and steam generators is redundant. TS 3.4.1.1, "Reactor Coolant Loops and Coolant Circulation - Startup and Power Operation," and 3.4.1.2, "Reactor Coolant Loops and Coolant Circulation - Hot Standby," provide restrictions on the number of operating reactor coolant loops and steam generators. Therefore, deleting the text that requires having four reactor coolant loops and associated steam generators in operation from TS 3.7.1.1, Action a., has no impact on any analyzed accident.

The proposed change to TS 3.7.1.1, Action a., to require the final mode to be hot shutdown rather than cold shutdown is consistent with the Applicability section of the specification, which does not require the MSSVs to be operable in hot shutdown. There are no credible transients requiring the MSSVs in modes 4 and 5. The steam generators are not normally used for heat removal in modes 5 and 6, and thus cannot be overpressurized. The change also eliminates the unnecessary transient that had been imposed on the unit by forcing entry into cold shutdown.

The new Action b. for TS 3.7.1.1 and text changes to Action a. clarify the shutdown requirement times based on the number of inoperable valves. There are no changes to these times.

Changing TSSR 4.7.1.1 to delete the one-time requirements imposed by previous amendments and allow entry into Mode 3 prior to performing the requirements of TSSR 4.0.5 has no impact on any accident. The change permits testing the MSSVs in accordance with the applicable codes and allows a reasonable amount of time for completion of the surveillance. The conditions requiring the one-time requirements have been corrected, so the one-time requirements are no longer required.

The proposed setpoints in Table 3.7-1 are more limiting than those currently allowed in Specification 3.7.1.1. Westinghouse determined that the current setpoints are non-conservative for some combinations of reduced MSSV availability and reactor power levels. By reducing the setpoints, the original design margins for safety will be met. Reduced reactor trip setpoints due to reduced availability of the MSSVs are not precursors to any accidents, but are used in the safety analysis to establish that plant response will be within required margins for accidents of concern.

Increasing the as-found valve setpoint tolerance from ±1% to ±3% does not have a significant impact on any accident. The peak primary and secondary pressures remain below 110% of design at all times. The departure from nucleate boiling ratio and peak cladding temperature values remain within the specified limits of the licensing basis. All of the applicable loss-of-coolant accident (LOCA) and non-LOCA design basis acceptance criteria remain valid.

The MSSVs are actuated after accident initiation to protect the secondary systems from overpressurization. Increasing the as-found setpoint tolerance will not result in any hardware modification to the MSSVs. Therefore, there is not an increase in the probability of the spurious opening of a MSSV. Sufficient margin exists between the normal steam system operating pressure and the valve setpoint with the increased tolerance to preclude an increase in the probability of actuating the valves. The MSSVs also remain capable of relieving any unlikely system overpressure during all applicable operating modes.

Although increasing the as-found valve setpoint tolerance may increase the steam release from the ruptured steam generator above the Updated Final Safety Analysis Review (UFSAR) value by approximately 2%, the steam generator tube rupture analysis indicates that the calculated break flow is still less than the value reported in the UFSAR. Therefore, the radiological analysis indicates that the slight increase in the steam release is offset by the decrease in the break flow such that the offsite radiation doses are less than those reported in the UFSAR. The evaluation also concluded that the existing mass releases used in the offsite dose calculation for the remaining transients (i.e., steam line break, rod ejection) are still applicable. Therefore, based on the above, there is no increase in the dose releases.

Neither the mass and energy release to the containment following a postulated LOCA, nor the analysis of containment response following the LOCA credit the MSSVs in mitigating the consequences of an accident. Therefore, changing the MSSV lift setpoint tolerances would have no impact on the containment integrity analysis. In addition, based on the conclusion of the transient analysis, the change to the MSSV tolerance will not affect the calculated steam line break mass and energy releases inside containment.

Deleting the orifice size column from Table 3.7.1-2 has no impact on previously evaluated accidents. There is no change to the orifice size, which is stated in the UFSAR and incorporated as needed in the accident analyses.

The proposed changes do not introduce any new equipment, equipment modifications, or any new or different modes of plant operation. The MSSVs are not precursors to any analyzed accident. The proposed changes will not affect the operational characteristics of any equipment or systems.

Therefore, these proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. **The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

Deleting the text describing reactor coolant loops and steam generators from TS 3.7.1.1 Action a. has no impact on plant operation since the specific restrictions on the number of operating reactor coolant loops and steam generators are provided in TS 3.4.1.1 and 3.4.1.2.

The proposed change to TS 3.7.1.1, Action a., to require the final mode to be hot shutdown rather than cold shutdown is consistent with the Applicability section of the specification, which does not require the MSSVs to be operable in hot shutdown. There are no credible transients requiring the MSSVs in Modes 4 and 5. The steam generators are not normally used for heat removal in Modes 5 and 6, and thus cannot be overpressurized. NUREG-1431 does not include requirements for the MSSVs to be operable in these modes. The change will also eliminate the unnecessary transient that had been imposed on the unit by forcing entry into cold shutdown.

The new Action b. for TS 3.7.1.1 and text changes to Action a. clarify the shutdown requirement times based on the number of inoperable valves. There are no changes to the times.

The proposed change to TSSR 4.7.1.1 to clarify that TSSR 4.0.4 does not apply for entry into Mode 3 will allow ComEd to continue to perform MSSV testing at normal operating pressure and temperature as required by the applicable codes. The change precludes having to enter an action statement to perform the testing and eliminates severe time restrictions on the valve testing and conflicts with other plant startup requirements.

The proposed recalculated setpoints of Table 3.7-1 are more limiting than those currently allowed in the Specification and ensure that the original design margins for safety are met. The secondary system pressure remains within design limits.

Increasing the as-found tolerance on the MSSV setpoint to ±3% will not increase the challenge to the MSSVs or result in increased actuation of the valves. The changes to the Bases document the method for calculating the reduced reactor trip setpoints based on reduced availability of MSSVs.

Deleting the orifice size column from Table 3.7-2 and the obsolete one-time requirements in TSSR 4.7.1.1 are administrative changes only.

Increasing the lift setpoint tolerance on the MSSVs does not introduce a new accident initiator mechanism. The proposed change does not introduce any new equipment, equipment modifications, or any new or different modes of plant operation. No new failure modes have been defined for any system or component important to safety nor has any new limiting single failure been identified. This change will not affect the operational characteristics of any equipment or systems. Thus, there is no change in the margin for safety.

Therefore, these proposed changes will not create the possibility of a new or different type of accident from any accident previously evaluated.

C. **The proposed change does not involve a significant reduction in a margin of safety.**

Deleting the text describing reactor coolant loops and steam generators has no impact on plant operation since the specific restrictions on the number of operating reactor coolant loops and steam generators are provided in TS 3.4.1.1 and 3.4.1.2.

The change requiring hot shutdown instead of cold shutdown entry is more appropriate than the existing specification since the action statement places the plant in a mode where operability of the MSSVs is not required. The Technical Specification is applicable in Modes 1, 2, and 3, therefore, entering Mode 4 places the plant in a condition where the MSSVs are not required to be operable. There are no credible transients requiring the MSSVs in Modes 4 and 5. The steam generators are not normally used for heat removal in Modes 5 and 6, and thus cannot be overpressurized. NUREG-1431 does not include requirements for the MSSVs to be operable in these modes.

Changing the mode in which the MSSVs are tested will not change the operational characteristics of the MSSVs. ComEd will continue to test the MSSVs at normal operating pressure and temperature as required by the applicable codes.

The proposed reactor trip setpoints in Table 3.7-1 are more limiting than the current setpoints in the Specification. Reactor trip settings were calculated using a revised methodology to account for the non-linear relationship of reactor trip setpoints and reduced MSSV availability. The revised setpoints ensure the original design margin of safety is maintained. The proposed changes to the Bases include the revised equation used to calculate the reduced reactor trip setpoints.

Increasing the as-found lift setpoint tolerance on the MSSVs will not adversely affect the operation of the reactor protection system, any of the protection setpoints, or any other device required for accident mitigation. The proposed increase in the setpoint tolerance does not invalidate the LOCA and non-LOCA conclusions presented in the UFSAR accident analyses. In letter CAE-91-209/CAE 91-219, Westinghouse concluded that the new loss of load/turbine trip analysis satisfied all applicable acceptance criteria and demonstrated that the conclusion presented in the UFSAR remains valid. For all the UFSAR non-LOCA transients, the departure from nucleate boiling design basis, primary and secondary pressure limits, and dose release limits continue to be met. Peak cladding temperatures remain well below the limits specified in the 10 CFR 50.46.

Deleting the orifice size column from Table 3.7-2 and the obsolete one-time requirements in TSSR 4.7.1.1 are administrative changes.

The proposed changes do not introduce any new equipment, equipment modifications, or any new or different modes of plant operation. These changes will not affect the operational characteristics of any equipment or systems. Therefore, no reduction in the margin of safety will occur as a result of changes.

Therefore, based upon the above evaluation, Commonwealth Edison has concluded that these changes involve no significant hazards considerations.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT

Commonwealth Edison Company (ComEd) has evaluated this proposed license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. ComEd has determined that this proposed license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) the amendment involves no significant hazards considerations

As demonstrated in Attachment C, this proposed amendment does not involve any significant hazards considerations.

(ii) there is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite

As documented in Attachment A, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.