ATTACHMENT A

UNIT 1 EXISTING SPECIFICATION

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	$\leq$ 4.0 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 12.0#/22.0##
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0
d. Containment Isolation-Phase "A"	≤ 17.0#/27.0##
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	≤ 14.0#/48.0##
h. Steam Line Isolation	≤ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0
d. Containment Fan Cooler	≤ 40.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	≤ 2.5
b. Feedwater Isolation	≤ 11.0
9. Steam Generator Water Level --Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60.0
b. Turbine-Driven Auxiliary Feedwater Pumps	≤ 60.0

ATTACHMENT B

UNIT 2 EXISTING SPECIFICATION

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	≦ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	≦ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	≦4 .0 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	≦ 2.0 seconds
10. Pressurizer Pressure--High	≦ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}/12.0^{(2)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0
d. Containment Isolation-Phase "A"	$\leq 18.0^{(2)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	$\leq 49.0^{(1)}/13.0^{(2)}$
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq 12.0^{(2)}/22.0^{(3)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0
d. Containment Isolation-Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	$\leq 13.0^{(2)}/48.0^{(3)}$
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	$\leq 14.0^{(2)}/24.0^{(3)}$
b. Reactor Trip (from SI)	≤ 4.0
c. Feedwater Isolation	≤ 9.0
d. Containment Isolation-Phase "A"	$\leq 19.0^{(2)}/29.0^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	$\leq 14.0^{(2)}/49.0^{(3)}$
h. Steam Line Isolation	≤ 9.0

ATTACHMENT C

UNIT 1 PROPOSED SPECIFICATION

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE ITEMS

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	$\leq$ 5.75 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE ITEMS

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Ventilation Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Service Water System	Not Applicable
Containment Fan Cooler	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Ventilation Isolation	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Ventilation Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0
d. Containment Isolation-Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	$\leq 13.0^{(2)}/48.0^{(3)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE ITEMS

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}/12.0^{(2)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0
d. Containment Isolation-Phase "A"	$\leq 18.0^{(2)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	$\leq 49.0^{(1)}/13.0^{(2)}$
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq 12.0^{(2)}/22.0^{(3)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0
d. Containment Isolation-Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	$\leq 13.0^{(2)}/48.0^{(3)}$
5. Steam Flow in two Steam Lines - High Coincident with T_{avg} --Low-Low	
a. Safety Injection (ECCS)	$\leq 15.75^{(2)}/25.75^{(3)}$
b. Reactor Trip (from SI)	≤ 5.75
c. Feedwater Isolation	≤ 10.75
d. Containment Isolation-Phase "A"	$\leq 20.75^{(2)}/30.75^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 61.75
g. Service Water System	$\leq 15.75^{(2)}/50.75^{(3)}$
h. Steam Line Isolation	≤ 10.75

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 12.0(2)/22.0(3)
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 7.0
d. Containment Isolation-Phase "A"	$\leq$ 17.0(2)/27.0(3)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Service Water System	$\leq$ 14.0(2)/48.0(3)
h. Steam Line Isolation	$\leq$ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 45.0
b. Containment Isolation-Phase "B"	Not applicable
c. Steam Line Isolation	$\leq$ 7.0
d. Containment Fan Cooler	$\leq$ 40.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps(4)	$\leq$ 60.0
b. Turbine-Driven Auxiliary Feedwater Pumps(5)	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Undervoltage RCP bus	
a. Turbine-Driven Auxiliary Feedwater Pumps	$\leq$ 60.0
11. Containment Radioactivity - High	
a. Purge and Exhaust Isolation	$\leq$ 5.0(6)
12. Trip of Feedwater Pumps	
a. Auxiliary Feedwater Pumps	Not Applicable
13. Undervoltage, Vital Bus	
a. Loss of Voltage	$\leq$ 4.0
14. Station Blackout	
a. Motor-Driven Auxiliary Feed Pumps	$\leq$ 60

TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2 Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(3) Diesel generator starting and sequence loading delays included. Response time limit opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(4) On 2/3 in any steam generator.

(5) On 2/3 in 2/4 steam generators.

(6) The response time is the time the isolation circuitry input reaches the isolation setpoint to the time the Containment Pressure-Vacuum Relief are fully shut.

ATTACHMENT D

UNIT 2 PROPOSED SPECIFICATION

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE ITEMS

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	$\leq$ 5.75 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE ITEMS

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Ventilation Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Service Water System	Not Applicable
Containment Fan Cooler	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Ventilation Isolation	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Ventilation Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 27.0 (1)
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 7.0
d. Containment Isolation-Phase "A"	$\leq$ 17.0(2)/27.0(3)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Service Water System	$\leq$ 13.0(2)/48.0(3)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE ITEMS

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 27.0(1)/12.0(2)
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 7.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0(2)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Service Water System	$\leq$ 49.0(1)/13.0(2)
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq$ 12.0(2)/22.0(3)
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 7.0
d. Containment Isolation-Phase "A"	$\leq$ 17.0(2)/27.0(3)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Service Water System	$\leq$ 13.0(2)/48.0(3)
5. Steam Flow in two Steam Lines - High Coincident with T_{avg} --Low-Low	
a. Safety Injection (ECCS)	$\leq$ 15.75(2)/25.75(3)
b. Reactor Trip (from SI)	$\leq$ 5.75
c. Feedwater Isolation	$\leq$ 10.75
d. Containment Isolation-Phase "A"	$\leq$ 20.75(2)/30.75(3)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 61.75
g. Service Water System	$\leq$ 15.75(2)/50.75(3)
h. Steam Line Isolation	$\leq$ 10.75

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12.0^{(2)}/22.0^{(3)}$
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 7.0
d. Containment Isolation-Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Service Water System	$\leq 14.0^{(2)}/48.0^{(3)}$
h. Steam Line Isolation	$\leq$ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 45.0
b. Containment Isolation-Phase "B"	Not applicable
c. Steam Line Isolation	$\leq$ 7.0
d. Containment Fan Cooler	$\leq$ 40.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps(4)	$\leq$ 60.0
b. Turbine-Driven Auxiliary Feedwater Pumps(5)	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Undervoltage RCP bus	
a. Turbine-Driven Auxiliary Feedwater Pumps	$\leq$ 60.0
11. Containment Radioactivity - High	
a. Purge and Exhaust Isolation	$\leq 5.0^{(6)}$
12. Trip of Feedwater Pumps	
a. Auxiliary Feedwater Pumps	Not Applicable
13. Undervoltage, Vital Bus	
a. Loss of Voltage	$\leq$ 4.0
14. Station Blackout	
a. Motor-Driven Auxiliary Feed Pumps	$\leq$ 60

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2 Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(3) Diesel generator starting and sequence loading delays included. Response time limit opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(4) On 2/3 in any steam generator.

(5) On 2/3 in 2/4 steam generators.

(6) The response time is the time the isolation circuitry input reaches the isolation setpoint to the time the Purge and Exhaust Valves Relief are fully shut.

BASIS FOR NO SIGNIFICANT HAZARDS DETERMINATION

The proposed change does not involve a significant hazards consideration because operation of Salem Generating Station Units 1 and 2 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The probability of previously analyzed accident is discussed first. The proposed change in measured response time will not increase the probability of such an accident because the numerical value of the LCO on OTΔT or steam flow in two steam lines - high coincident with T_{avg} - low-low ESF actuation response times is not a factor in the initiation of a previously evaluated accident. The removal and replacement of the existing RTDs and bypass line elimination will not significantly increase the probability of occurrence of an accident previously evaluated. The events of interest are those initiated by a failure of those components affected by the proposed change. There are four such events: (1) Uncontrolled Withdrawal of a Control Rod at Power, (2) Excessive Load Increase, (3) Accidental Depressurization of the Main Steam System, and (4) Small Break Loss of Coolant Accident (SBLOCA). The Uncontrolled Rod Withdrawal event is an ANS Condition II (moderate frequency) event potentially initiated by a failure of the reactor control system. The Excess Load and Accidental Depressurization of the Main Steam System events are also Condition II events. They are potentially initiated by a failure of the steam dump control system. The input to the reactor control system and steam dump control system from the replacement RTDs will be equivalent to those currently provided by the existing RTDs. The proposed modification will be done in a manner consistent with the plant design bases. As such, there will be no degradation in the performance of or increase of the number of challenges to safety systems assumed to function in the accident analysis. Furthermore, there will be no increase in the probability of failure of or degradation of the performance of the systems designed to reduce the number of challenges to safety systems. Hence, the first three events will remain Condition II events.

The SBLOCA is an ANS Condition III (infrequent) event. It could be initiated by the highly unlikely ejection of a thermowell or the failure of a cap covering one of the existing pump suction leg penetrations. The scoops, cross over leg buttweld caps RVLIS, and thermowells will be analyzed to the ASME Boiler and Pressure Vessel Code, Section III, Class 1 and installed in accordance with the requirements of Section XI of this Code. As such, the RCS pressure boundary will not be degraded. The SBLOCA will thus remain a Condition III event. Additionally, approximately 280 feet of small diameter pipe and the associated valves will be removed from the primary system pressure boundary, eliminating the possibility of a SBLOCA from these locations. Hence, there will be no significant increase in the probability of occurrence of an accident previously evaluated in the SAR.

There will be no increase in the consequences of a previously evaluated accident. In assessing the impact on the consequences of a previously evaluated accident, there are four events of interest: (1) Uncontrolled Boron Dilution During Full Power, (2) Loss of External Load, (3) Uncontrolled Withdrawal of a Control Rod at Power, and (4) Major Secondary Pipe Rupture. The first three events are of interest because the OTΔT trip is the primary trip credited in the safety analyses. The fourth event is considered because steam flow in two steam lines - high coincident with T_{avg}-low-low is one of the signals credited to initiate an Engineered Safety Features actuation. The OTΔT trip will continue to function in a manner consistent with the existing analysis assumptions for the first three events. The actual response time will be within the six seconds currently assumed. Similarly, the ESF response times will remain within those assumed in the safety analysis. Hence there will be no increase in the consequences of previously evaluated accident.

(2) create the possibility of a new or different kind of accident from any previously analyzed. The proposed change will be performed in a manner consistent with the applicable standards, preserve the existing design bases, and will not adversely impact the qualification of any plant systems. This will preclude adverse control/protection systems interactions. The design, installation, and inspection of the new

equipment will be done in accordance with ASME Boiler and Pressure Vessel Code criteria. By adherence to industry standards, the pressure boundary integrity will be preserved. As such, the possibility of a new or different kind of accident is not created.

(3) involve a significant reduction in a margin of safety. The applicable margins of safety are defined in Technical Specification Bases Sections 2.1.1 and 2.1.2. Bases Section 2.1.1 states that the minimum value of the Departure from Nucleate Boiling Ratio (DNBR) during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that Departure from Nucleate Boiling (DNB) will not occur. The restrictions of this fuel cladding integrity safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the coolant. The proposed change will not result in a decrease in the minimum DNBR reported in the UFSAR accident analyses.

Bases Section 2.1.2 states that the Safety Limit on maximum RCS pressure is 2735 psig. This Safety Limit protects the integrity of the RCS from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere. The proposed change will not result in an increase in the maximum RCS pressure reported in the UFSAR accident analyses.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing examples (51 FR 7751) of amendments that are considered not likely to involve significant hazards consideration. Example (ix) is:

A repair or replacement of a major component or system important to safety if the following conditions are met:

(1) The repair or replacement process involves practices which have been successfully implemented at least once on similar components or systems elsewhere in the nuclear industry or in other industries, and does not

involve a significant increase in the probability or consequences of an accident previously evaluated or create the possibility of a new or different kind of accident from any accident previously evaluated; and

(2) The repaired or replacement component or system does not result in a significant reduction in any safety limit (or limiting condition of operation) associated with the component or system.

The proposed changes to the Salem Units 1 and 2 Technical Specifications is similar to changes approved at Byron Station Units 1 and 2 (52 FR 2785). As discussed earlier, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated or create the possibility of a new or different kind of accident from any previously evaluated. It does not result in a significant change in a component or system safety function or a significant reduction in any associated safety limit or limiting condition of operation.

Therefore, based on the above considerations, it has been determined that the proposed change does not involve a significant hazards consideration.

The addition of response times to the Auxiliary Feed Pump Section of table 3.3.5 and the addition of Station Blackout requirements corresponds to example 2 of 48 FR 14870 as changes that impose an additional limitation not currently in the Technical Specifications. Changes to the footnotes were done to correct typographical errors and as such correspond to example 1 of 48 FR 14870. In either case the changes will not involve an increase in the probability or consequences of a previously analyzed accident, create a new or different kind of accident that previously analyzed, or reduce the margin of safety since the changes are being done to be consistent with previously reviewed and approved analyses.