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SUBJECT: Application for proposed amend 120 to license DPR-43, incorporating changes to TS 3.4, "Steam & Power Conversion Sys" to modify & clarify operability requirements for MSSVs, AFW sys & condensate storage tank sys.

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May 20, 1994

10 CFR 50.90

U.S. Nuclear Regulatory Commission
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Washington, D.C. 20555

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Proposed Amendment 120 to the Kewaunee
Nuclear Power Plant Technical Specifications

References: 1) Letter from L. Robert Greger (NRC) to C.A. Schrock (WPSC) dated August 20, 1993 (Inspection Report 93008)

2) Letter from L. Robert Greger (NRC) to C.A. Schrock (WPSC) dated November 19, 1993 (Inspection Report 93019)

This proposed amendment (PA) to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) is being submitted to incorporate changes to TS 3.4, "Steam and Power Conversion System." The proposed changes modify and clarify the operability requirements for the Main Steam Safety Valves, the Auxiliary Feedwater (AFW) System, and the Condensate Storage Tank System. Also, the Basis Section is being modified and minor administrative changes incorporated.

A001

These changes are being proposed to eliminate inconsistencies within TS section 3.4 and provide the basis for acceptable operation of the Auxiliary Feedwater System below 15% reactor power. WPSC's evaluation of reference 1 revealed that inconsistencies exist within the Technical Specifications regarding the allowed outage time for the Auxiliary Feedwater System. Reference 2 details concerns regarding the low-power and non-power operation and alignment of the Auxiliary Feedwater System. This proposed change is intended to address these items.

Attachment 1 to this letter contains a description, a safety evaluation, a significant hazards determination and an environmental considerations for the proposed changes. Attachment 2 contains the affected TS pages. Note that previously submitted PA 125 affects TS 3.4.

In accordance with the requirements of 10 CFR 50.30(b), this submittal has been signed and notarized. A complete copy of this submittal has been transmitted to the State of Wisconsin as required by 10 CFR 50.91(b)(1).

Sincerely,

Clark Steinhardt

Clark R. Steinhardt
Senior Vice President - Nuclear Power

DJK/cjt

Attach.

cc - US NRC - Region III
US NRC Senior Resident Inspector
Mr. Robert Cullen, PSCW

Subscribed and Sworn to
Before Me This 20th Day
of May 1994

Jeanne M. Ferris
Notary Public, State of Wisconsin
My Commission Expires:

June 18, 1995

lic\nrc\pa120.wp

ATTACHMENT 1

To

Letter from C.R. Steinhardt (WPSC)

to

Document Control Desk (NRC)

Dated

May 20, 1994

Proposed Amendment 120

Description of Proposed Changes

Safety Evaluation

Significant Hazards Determination

Environmental Considerations

Introduction

TS section 3.4 is being restructured to clarify and modify the operability conditions for the systems required for steam generator operability. The entire section is being restructured to improve the clarity of the KNPP Technical Specifications and improve consistency with "Westinghouse Standard Technical Specifications," NUREG-1431.

TS 3.4.a and TS 3.4.b currently specify requirements for the Auxiliary Feedwater (AFW) System, the Main Steam Safety Valves (MSSVs), the Condensate Storage Tank and Secondary Iodine-131 activity. The proposed change splits TS 3.4.a and TS 3.4.b into four distinct new specifieations, one for each of the components affecting steam generator operability:

1. TS 3.4.a "Main Steam Safety Valves"
2. TS 3.4.b "Auxiliary Feedwater System"
3. TS 3.4.c "Condensate Storage Tank"
4. TS 3.4.d "Secondary Activity Limits"

In addition, an administrative change is being proposed to renumber TS 3.4.c, "Turbine Overspeed Protection System," to TS 3.4.e. These changes are being proposed to eliminate inconsistencies within the Technical Specifications and provide the basis for the acceptable operation of the Auxiliary Feedwater System below 15% reactor power. A Safety Evaluation and a Significant Hazards Determination for each new proposed Technical Specifications section follows.

Description of Proposed Changes to Technical Specification (TS) 3.4, "Steam and Power Conversion System"

TS section 3.4., "Steam and Power Conversion System," is being split into five distinct new specifications, one for each of the components affecting Steam Generator operability, specifically:

- Existing TS 3.4.a.1.A.2 to proposed TS 3.4.a "Main Steam Safety Valves"
- Existing TS 3.4.a.1.A.1 and TS 3.4.b to proposed TS 3.4.b "Auxiliary Feedwater System"
- Existing TS 3.4.a.1.B to proposed TS 3.4.c "Condensate Storage Tank"
- Existing TS 3.4.a.1.C to proposed TS 3.4.d "Secondary Activity Limits"
- Existing TS 3.4.c to proposed TS 3.4.e "Turbine Overspeed Protection System."

The changes to proposed TS 3.4.c, TS 3.4.d and TS 3.4.e are formatting changes only. Technical changes are being proposed for TS 3.4.a and TS 3.4.b as follows:

1. Proposed TS 3.4.a is being modified to allow reactor coolant system heatup > 350°F with a minimum of two Main Steam Safety Valves per steam generator operable, and specifies that the reactor shall not be made critical unless five MSSVs per steam generator are operable.

2. Current TS 3.4.a.1.A.1 and TS 3.4.b are being combined and renamed "Auxiliary Feedwater System." This change more clearly defines operability of the Auxiliary Feedwater (AFW) System as being based on an entire AFW train, and not just an AFW pump. Additionally, the following changes are being proposed.

   - An action statement to allow one steam supply to the turbine driven AFW pump to be inoperable for seven (7) days.
   - The requirement to maintain a Service Water supply to the AFW pumps is being relocated from current TS 3.4.a.1.B, to proposed TS 3.4.b.1.
   - The TS 3.4.a.1.A.1 limiting condition for operation (LCO) action statement which allows all system piping and valves directly associated with providing auxiliary feedwater flow to the steam generators to be inoperable for 48 hours is deleted, and
   - A condition is being added to permit the AFW pump control switches located in the control room to be placed in the "pull out" position and valves AFW-2A and AFW-2B to be in a throttled position without declaring the corresponding AFW train inoperable when less than 15% reactor power.

3. The Table of Contents and Basis sections are revised accordingly and are being submitted for your information.

Safety Evaluation for Proposed Change to Technical Specifications (TS) 3.4.a "Main Steam Safety Valves"

Currently, TS 3.4.a.1.A.2 requires five MSSV's per steam generator to be operable prior to heating the reactor > 350 °F. The proposed change will require a minimum of two MSSVs per steam generator to be operable prior to heating the reactor coolant system > 350 °F, and five MSSVs per steam generator to be operable prior to reactor criticality. If these conditions cannot be met within 48 hours, within 1 hour action shall be initiated to achieve hot standby within 6 hours, achieve hot shutdown within the following 6 hours, and achieve and maintain the reactor coolant system temperature < 350°F within an additional 12 hours.

The MSSVs are relied upon to function in each of the following Updated Safety Analysis Report (USAR) analyzed accidents: Reactor Coolant Pump Locked Rotor, Loss of External Electrical Load, Loss of Normal Feedwater, Uncontrolled Rod Cluster Control Assembly Withdrawal at Power, Uncontrolled Rod Cluster Control Assembly Withdrawal From a Subcritical Condition, Steam Generator Tube Rupture, and Anticipated Transients without Scram.

The initial conditions in the analysis of a Loss of External Electrical Load Accident, Loss of Normal Feedwater Accident, Uncontrolled Rod Withdrawal at Power Accident, Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical Condition Accident and an Anticipated Transients without Scram Accident assume critical conditions. Because this proposed TS requires all MSSVs to be operable prior to reactor criticality, there will be no adverse effect on the health and safety of the public.

Two operable MSSVs are capable of relieving the maximum steam generated during a Reactor Coolant Pump Locked Rotor Accident or a Steam Generator Tube Rupture from a subcritical condition. The maximum required steam relief rate resulting from a Steam Generator Tube Rupture is 270,000 lbm/hr, well below the relief capacity of 1,532,076 lbm/hr of two MSSVs on a single steam generator. Two MSSVs on one steam generator are capable of relieving the steam generated corresponding to approximately 20% reactor power. This relief capacity is well above the maximum steam generation rate resulting from a Reactor Coolant Pump Locked Rotor Accident from a subcritical condition.

In all cases, the relieving capacity of the MSSVs exceeds the maximum steam generation rate, and reactor criticality is not permitted unless all MSSVs are operable. Therefore, this change will have no adverse effect on the health and safety of the public.

<u>Significant Hazards Determination for Proposed Changes to Technical Specification (TS) 3.4.a "Main Steam Safety Valves"</u>

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed changes will not:

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

   Currently, TS 3.4.a.1.A.2 requires five MSSV's to be operable prior to heating the reactor > 350 °F. The proposed change requires a minimum of two MSSVs per steam generator to be operable prior to heating the reactor coolant system > 350 °F, and five MSSVs per steam generator to be operable prior to reactor criticality. If these conditions cannot be met within 48 hours, within 1 hour action shall be initiated to achieve hot standby within 6 hours, achieve hot shutdown within the following 6 hours, and achieve and maintain the reactor coolant system temperature < 350°F within an additional 12 hours.

   The MSSVs are relied upon to function in each of the following USAR analyzed accidents: Reactor Coolant Pump Locked Rotor, Loss of External Electrical Load, Loss of Normal Feedwater, Uncontrolled Rod Cluster Control Assembly Withdrawal at Power, Uncontrolled Rod Cluster Control Assembly Withdrawal From a Subcritical Condition, Steam Generator Tube Rupture, and Anticipated Transients without Scram.

   The initial conditions in the analysis of a Loss of External Electrical Load Accident, Loss of Normal Feedwater Accident, Uncontrolled Rod Withdrawal at Power Accident, Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical Condition Accident and an Anticipated Transients without Scram Accident assume critical conditions. Because this proposed TS requires all MSSVs to be operable prior to reactor criticality, there will be no adverse effect on the health and safety of the public.

Two operable MSSVs are eapable of relieving the maximum steam generated during a Reactor Coolant Pump Locked Rotor Accident or a Steam Generator Tube Rupture from a subcritical condition. The maximum required steam relief rate resulting from a Steam Generator Tube Rupture is 270,000 lbm/hr, well below the relief capacity of 1,532,076 lbm/hr of two MSSVs on a single steam generator. Also, this condition results in no increase in the radiological dose to the public. Two MSSVs on one steam generator are capable of relieving the steam generated corresponding to approximately 20% reactor power. This relief capacity is well above the maximum steam generation rate resulting from a Reactor Coolant Pump Locked Rotor Accident from a subcritical condition.

In all cases, the relieving capacity of the MSSVs exceeds the maximum steam generation rate, and reactor criticality is not permitted unless all MSSVs are operable. Therefore, there is no adverse effect on the health and safety of the public and no significant increase consequences of an accident previously evaluated.

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

   The proposed change does not alter the plant configuration, operating setpoints, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

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

   The USAR safety analysis assumes five MSSVs per steam generator are operable. However, as shown above, this change results in no steam generator overpressure event or increase in the radiological dose. Therefore, this change will not involve a reduction in the margin of safety.

<u>Safety Evaluation for Proposed Change to Technical Specifications (TS) 3.4.b "Auxiliary Feedwater System"</u>

Current TS 3.4.a.1.A.1 and TS 3.4.b governing auxiliary feedwater flow to the steam generators are being combined and titled, "Auxiliary Feedwater System." This change is consistent with the format of "Westinghouse Standard Technical Specifications," NUREG-1431. In addition to the formatting changes, a number of technical changes are being proposed. These are:

- the correction of an inconsistency between current TS 3.4.a.1.A.1 and current TS 3.4.6.2.A.
- the addition of a seven (7) day Limiting Condition for Operation (LCO) for one steam supply to the turban driven auxiliary feedwater pump.
- an addition to allow the AFW control switches located in the control room to be placed in the "pull out" position and valves AFW-2A and AFW-2B to be in a throttled position when below 15% reactor power without declaring the corresponding AFW train inoperable.

An inconsistency exists between current TS 3.4.a.1.A.1 and current TS 3.4.b.2.A. TS 3.4.a.1.A.1 requires the system piping and valves directly associated with providing auxiliary feedwater flow to the steam generators to be operable, with a corresponding 48 hour limiting condition for operation (LCO) action statement if this requirement is not met. TS 3.4.b.2.A allows one auxiliary feedwater pump to be inoperable for 72 hours. This arrangement can cause a conflict regarding which TS is applicable depending on which component in the auxiliary feedwater flowpath to the steam generators is inoperable. By moving all TS action statements to TS 3.4.b, the inconsistency between TS 3.4.a.1.A.1 and TS 3.4.b.2.A will be eliminated. The requirement to maintain the operability of the system piping and valves directly associated with providing auxiliary feedwater flow to the steam generators remains, but is being modified to prevent removing both AFW supply headers from service for 48 hours.

Proposed TS 3.4.b.2.A is being added to allow one steam supply to the turbine driven auxiliary feedwater pump to be inoperable for seven days. This addition is consistent with "Westinghouse Standard Technical Specifications," NUREG-1431. The seven day completion time is reasonable based on the redundant steam supplies to the pump, the availability of the redundant motor-driven AFW pumps, and the low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump. For these reasons, this change will have no adverse affect on the health and safety of the public.

Proposed TS 3.4.b.5 permits the AFW Pump control switches located in the control room to be placed in the "pull out" position and valves AFW-2A and AFW-2B to be in a throttled position when below 15% reactor power without declaring the corresponding AFW train inoperable. This change is proposed to resolve concerns regarding the cycling of the AFW pumps and the throttling of valves AFW-2A and AFW-2B during plant startups and shutdowns. Analysis shows that control room operators have a minimum of ten minutes to initiate auxiliary feedwater flow after a design basis accident with no steam generator dryout or core damage.

All accidents which rely on AFW flow for mitigation were reanalyzed to support this change. These analyses were completed assuming an initial reactor power level of 100%. However, a 15% reactor power restriction has been imposed on placing the AFW pump control switches located in the control room in the "pull out" position and throttling valves AFW-2A and AFW-2B. This restriction in effect limits use of TS 3.4.b.5 to plant start-ups, shutdowns and other low power operating conditions.

This change alters the assumptions of the safety analysis for the Small-Break Loss of Coolant Accident, the Steam Generator Tube Rupture and the Loss of Normal Feedwater due to their dependence on the AFW system to start and supply AFW for heat removal. To support this change, the Westinghouse Electric Corporation performed an analysis of the Small-Break Loss-of-Coolant Accident using the NOTRUMP code assuming a ten minute delay in AFW system operation for operator action to initiate auxiliary feedwater. This analysis resulted in a Peak Cladding Temperature (PCT) of 1053°F from an initial power level of 100%. All other acceptance criteria of 10 CFR 50.46 were met. This large margin to the 2200°F PCT limit supports ten minutes for operator action to initiate auxiliary feedwater.

Furthermore, WPSC has analyzed the Loss of Normal Feedwater and the Steam Generator Tube Rupture Accident assuming delays in the initiation of auxiliary feedwater. The Loss of Normal Feedwater Accident with a ten minute delay in the initiation of Auxiliary Feedwater does not result in any adverse condition in the core. It does not result in water relief from the pressurizer safety valves, nor does it result in uncovering the tube sheets of the steam generators. Also, at all times the Departure from Nucleate Boiling Ratio (DNBR) remained greater than 1.30. The Steam Generator Tube Rupture Accident with no auxiliary feedwater flow was also analyzed. The results of this analysis indicate that neither steam generator empties of liquid and at least 20°F of reactor coolant system subcooling is maintained throughout the transient. Also, there is no increase in the radiological dose to the public.

Ten minutes is an acceptable time for operator action because four independent alarms in the control room would initiate operator action to place the AFW pump control switches to the "auto" position and initiate AFW flow to the steam generators when necessary. These include two steam generator lo level alarms (one per steam generator), and two steam generator lo-lo level alarms (one per steam generator). Provisions also exist to add additional low level alarms on the plant process computer. In addition to these alarms, control room operators have twelve other indications of insufficient, or no, AFW flow to the steam generators. These indications include three auxiliary feedwater pump low discharge pressure alarms (one per AFW pump), two auxiliary feedwater flow meters (one per steam generator), two AFW pump motor amp meters (one per motor-driven AFW pump), two "ESF in Pullout" alarms (one per Engineered Safety Features train) and three pump running lights (one per AFW pump). The ten minutes for operator action was discussed in a telephone conversation between WPSC and Mr. R. Laufer (NRR). Ten minutes for operator action is further supported by Branch Technical Position EISCB 18. Scenarios have been completed on the KNPP simulator to support ten minutes for operator initiation of AFW flow. In all cases, operators manually initiated AFW flow within the allowed ten minutes. For these reasons, these changes will not adversely effect the health and safety of the public.

<u>Significant Hazards Determination for Proposed Changes to Technical Specification (TS) 3.4.b "Auxiliary Feedwater System"</u>

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed changes will not:

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

   Current TS 3.4.a.1.A.1 and TS 3.4.b governing auxiliary feedwater flow to the steam generators are being combined and titled, "Auxiliary Feedwater System." This change is consistent with the format of "Westinghouse Standard Technical Specifications," NUREG-1431. In addition to the formatting changes, a number of technical changes are being proposed. These are:

- the correction of an inconsistency between current TS 3.4.a.1.A.1 and current TS 3.4.6.2.A.
- The addition of a seven (7) day Limiting Condition for Operation (LCO) for one stem supply to the turban driven auxiliary feedwater pump.
- An addiction to allow the AFW control switches located in the control room to be placed in the "pull out" position and valves AFW-2A and AFW-2B to be in a throttled position when below 15% reactor power without declaring the corresponding AFW train inoperable.

An inconsistency currently exists between current TS 3.4.a.1.A.1 and current TS 3.4.b.2.A. TS 3.4.a.1.A.1 requires the system piping and valves directly associated with providing auxiliary feedwater flow to the steam generators to be operable, with a corresponding 48 hour limiting condition for operation (LCO) action statement if this requirement is not met. TS 3.4.b.2.A allows one auxiliary feedwater pump to be inoperable for 72 hours. This arrangement can cause a conflict regarding which TS is applicable depending on which component in the auxiliary feedwater flowpath to the steam generators is inoperable. By moving all TS action statements to TS 3.4.b, the inconsistency between TS 3.4.a.1.A.1 and TS 3.4.b.2.A will be eliminated. The requirement to maintain the operability of the system piping and valves directly associated with providing auxiliary feedwater flow to the steam generators remains, but is being modified to prevent removing both AFW supply headers from service for 48 hours.

Proposed TS 3.4.b.2.A is being added to allow one steam supply to the turbine driven auxiliary feedwater pump to be inoperable for seven days. This addition is consistent with "Westinghouse Standard Technical Specifications," NUREG-1431. The seven day completion time is reasonable based on the redundant steam supplies to the pump, the availability of the redundant motor-driven AFW pumps, and the low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump. For these reasons, this change will have no adverse affect on the health and safety of the public.

Proposed TS 3.4.b.5 permits the AFW Pump control switches located in the control room to be placed in the "pull out" position and valves AFW-2A and AFW-2B to be in a throttled position when below 15% reactor power without declaring the corresponding AFW train inoperable. This changes ids proposed to resolve concersn regarding the cycling of the AFW pumps and the throttling of valves AFW-2A and AFW-2B during plant startups and shutdowns. Analysis shows that control room operators have a minimum of ten minutes to initiate auxiliary feedwater flow after a design basis accident with no steam generator dryout or core damage.

All accidents which rely on AFW flow for mitigation were reanalyzed to support this change. These analyses were completed assuming an initial power of 100%. However, a 15% reactor power restriction has been implemented on placing the AFW pump control switches located in the control room in the "pull out" position and throttling valves AFW-2A and AFW-2B. This restriction in effect limits use of TS 3.4.b.5 to plant start-ups, shutdowns and other low power operating conditions.

This change alters the assumptions of the safety analysis for the Small-Break Loss of Coolant Accident, the Steam Generator Tube Rupture and the Loss of Normal Feedwater due to their dependence on the AFW system to start and supply AFW for heat removal. To support this change, the Westinghouse Electric Corporation performed an analysis of the Small-Break Loss-of-Coolant Accident using the NOTRUMP code assuming ten minutes for operator action to initiate auxiliary feedwater. This analysis resulted in a Peak Cladding Temperature (PCT) of 1053°F from an initial power level of 100%. All other acceptance criteria of 10 CFR 50.46 were met. This large margin to the 2200°F PCT limit supports ten minutes for operator action to initiate auxiliary feedwater.

Furthermore, WPSC has analyzed the Loss of Normal Feedwater and the Steam Generator Tube Rupture Accident assuming delays in the initiation of auxiliary feedwater. The Loss of Normal Feedwater Accident with a ten minute delay in the initiation of Auxiliary Feedwater does not result in any adverse condition in the core. It does not result in water relief from the pressurizer safety valves, nor does it result in uncovering the tube sheets of the steam generators. Also, at all times the Departure from Nucleate Boiling Ratio (DNBR) remained greater than 1.30. The Steam Generator Tube Rupture Accident with no auxiliary feedwater flow was also analyzed. The results of this analysis indicate that neither steam generator empties of liquid and at least 20°F of reactor coolant system subcooling is maintained throughout the transient. Also, there is no increase in the radiological dose to the public.

Ten minutes is an acceptable time for operator action because four independent alarms in the control room would initiate operator action to place the AFW pump control switches to the "auto" position and initiate AFW flow to the steam generators when necessary. These include two steam generator lo level alarms (one per steam generator), and two steam generator lo-lo level alarms (one per steam generator). Provisions also exist to add additional low level alarms on the plant process computer. In addition to these alarms, control room operators have twelve other indications of insufficient, or no, AFW flow to the steam generators. These indications include three auxiliary feedwater pump low discharge pressure alarms (one per AFW pump), two auxiliary feedwater flow meters (one per steam generator), two AFW pump motor amp meters (one per motor-driven AFW pump), two "ESF in Pullout" alarms (one per Engineered Safety Features train) and three pump running lights (one per AFW pump). The ten minutes for operator action was discussed in a telephone conversation between WPSC and Mr. R. Laufer (NRR). Ten minutes for operator action is further supported by Branch Technical Position EISCB 18. Scenarios have been completed on the KNPP simulator to support ten minutes for operator initiation of AFW flow. In all cases, operators manually initiated AFW flow within the allowed ten minutes.

For these reasons, this change will have no adverse affect on the health and safety of the public or significantly increase the probability or consequences of an accident previously evaluated in the USAR.

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

   The proposed change does not alter the plant configuration, operating setpoints, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

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

   This change alters the assumptions of the safety analysis for the Small-Break Loss of Coolant Accident, the Steam Generator Tube Rupture and the Loss of Normal Feedwater due to their dependence on the AFW system to start and supply AFW flow for heat removal. To support this change the Westinghouse Electric Corporation has performed an analysis of the Small-Break Loss-of-Coolant Accident using the NOTRUMP code assuming ten minutes for operator action to initiate auxiliary feedwater. This analysis resulted in a Peak Cladding Temperature (PCT) of 1053°F from an initial power level of 100%. All other acceptance criteria of 10 CFR 50.46 were met. This large margin to the 2200°F PCT limit supports ten minutes for operator action to initiate auxiliary feedwater.

   Furthermore, WPSC has analyzed the Loss of Normal Feedwater and the Steam Generator Tube Rupture Accident assuming delays in the initiation of auxiliary feedwater. The Loss of Normal Feedwater Accident with a ten minute delay in the initiation of Auxiliary Feedwater does not result in any adverse condition in the core. It does not result in water relief from the pressurizer safety valves, nor does it result in uncovering the tube sheets of the steam generators. Also, at all times the Departure from Nucleate Boiling Ratio (DNBR) remained greater than 1.30. The Steam Generator Tube Rupture Accident with no Auxiliary Feedwater flow was also analyzed. The results of this analysis indicate that neither steam generator empties of liquid and at least 20°F of reactor coolant system subcooling is maintained throughout the transient. Also, there is no increase in the radiologieal dose to the public.

   For these reasons, these changes will not adversely effect the health and safety of the public or involve a significant reduction in the margin of safety.

Safety Evaluation for Proposed Administrative Changes to Technical Specification (TS) 3.4 "Steam and Power Conversion System"

Administrative changes are being made to renumber the following TS:

1. Current TS 3.4.a.1.B to proposed TS 3.4.c, "Condensate Storage Tank"
2. Current TS 3.4.a.1.C to proposed TS 3.4.d, "Secondary Activity Limits"
3. Current TS 3.4.c, "Turbine Overspeed Protection System" to proposed TS 3.4.e.

Also, the requirement of TS 3.4.a.1.B to maintain the ability of the Service Water System to deliver water to the Auxiliary Feedwater Pumps has been moved to proposed TS 3.4.b.1.

These changes have been reviewed to ensure that they do not alter the intent or interpretation of the specification; therefore, there is no adverse effect on public health or safety.

Significant Hazards Determination for Proposed Administrative Changes to Section TS 3.4 "Steam and Power Conversion System"

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will 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 the margin of safety.

The proposed changes are administrative in nature and do not alter the intent or interpretation of the TS. Therefore, no significant hazards exist.

Additionally, the proposed change is similar to example C.2.e(i) in 51 FR 7751. Example C.2.e.(i) states that changes which are purely administrative in nature; i.e., to achieve consistency throughout the Technical Specifications, correct an error, or a change in nomenclature, are not likely to involve a significant hazard.

Environmental Considerations

This proposed amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or a change to a surveillance requirement. WPSC has determined that the proposed amendment involves no significant hazards considerations and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in the individual or cumulative occupational radiation exposure. Accordingly, this proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with this proposed amendment.