WOLF CREEK
NUCLEAR OPERATING CORPORATION

Robert C. Hagan
Vice President Nuclear Assurance

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NA 94-0089

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

Subject: Docket No. 50-482: Revisions to Technical Specifications Based on NRC Final Policy Statement on Technical Specifications Improvements

Gentlemen:

Attached is an application for amendment to Facility Operating License No. NPF-42 for Wolf Creek Generating Station (WCGS). This amendment would modify the WCGS Technical Specifications to incorporate improvements in scope and content endorsed by the Nuclear Regulatory Commission (NRC) in its Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, 58 FR 39132, July 22, 1993.

Attachment I provides a Safety Evaluation including a description of the proposed changes. Attachments II and III provide the No Significant Hazards Consideration Determination and an Environmental Impact Determination supporting the requested changes. Attachment IV consists of the revised Technical Specification pages, and Attachment V provides the results of application of the Policy Statement criteria to the WCGS Technical Specifications. Attachment VI provides draft mark-up pages for the proposed revised specifications that will be relocated to the Updated Safety Analysis Report (USAR) Chapter 16.

This license amendment application was developed in a joint undertaking with Union Electric Company, and the requested changes are, with only a few exceptions, identical to those changes in a similar request being submitted by Union Electric Company for the Callaway Plant. The attached Safety Evaluation, No Significant Hazards Consideration Determination, and the Environmental Impact Determination are basically identical for both plants.

P.O. Box 411 / Burlington, KS 66839 / Phone: (316) 364-8831

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In the process of applying the Policy Statement criteria, WCNOC decided to add Neutron Flux monitors and the Reactor Vessel Level Indicating System to the Post-Accident Monitoring (PAM) Specification 3.3.3.6, adopt the action statements given in the new Standard Technical Specifications (NUREG-1431) for PAM instrument functions, and delete Specification 3.6.4.1, since it is redundant and obsolete per NUREG-1431. Further, the main feedwater isolation valves were added to the Wolf Creek Technical Specifications in new Specification 3.7.1.7, with action statements also adopted from NUREG-1431. The relocated specifications will be added, in their entirety, to USAR Chapter 16. This shall provide for future changes to the relocated specifications via the 10CFR50.59 review process.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Kansas State Official. This revision to the Technical Specifications will be fully implemented within 120 days of formal NRC approval.

If you have any questions concerning this matter, please contact me at (316)364-8831 extension 4553 or Mr. Kevin J. Moles at extension 4565.

Very truly yours,

Robert C. Hagan

Robert C. Hagan

RCH/jra

Attachments:
- I. - Safety Evaluation
- II. - No Significant Hazards Consideration Determination
- III. - Environmental Impact Determination
- IV. - Proposed Technical Specification Changes
- V. - Results of Application of the NRC Final Policy Statement of Technical Specification Improvements
- VI. - Proposed Updated Safety Analysis Report Revisions

cc: G. W. Allen (KDHE), w/a
L. J. Callan (NRC), w/a
G. A. Pick (NRC), w/a
W. D. Reckley (NRC), w/a
T. Reis (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Robert C. Hagan, of lawful age, being first duly sworn upon oath says that he is Vice President Nuclear Assurance of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the content thereof; that he has executed that same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

NOTARY PUBLIC

By Robert C. Hagan
Robert C. Hagan
Vice President
Nuclear Assurance

SUBSCRIBED and sworn to before me this 24th day of May, 1994.

Sandra L. Elliott
Notary Public

Expiration Date 5/14/95

ATTACHMENT I

SAFETY EVALUATION

Safety Evaluation

Proposed Changes

This license amendment request proposes to revise the Technical Specifications to implement the improvements endorsed in the Nuclear Regulatory Commission's Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, 58 FR 39132, July 22, 1993 (the Policy Statement). These improvements involve focusing the Technical Specifications on those requirements that are of controlling importance to operational safety by screening each Technical Specification in Sections 3/4.1 through 3/4.11 using the criteria provided in the Policy Statement. Those criteria are intended to identify requirements derived from the analyses and evaluations included in the Updated Safety Analysis Report (USAR) that are of immediate concern to the health and safety of the public. Technical Specifications that meet one or more of the criteria must be retained. Those that meet none of the criteria may be removed from the Technical Specifications. The purpose of this amendment request is to remove the specifications that do not meet any of the four Policy Statement criteria.

The removed Technical Specifications will be relocated to USAR Chapter 16, "Technical Specifications." In general, the Technical Specifications that are proposed for relocation would be incorporated into the USAR with the same format and content they possessed as part of the Operating License.

In some cases, the Technical Specification Limiting Condition for Operation (LCO) did not meet any of the criteria for retention, but an associated Surveillance Requirement (SR) was required to support an LCO that was being retained in Technical Specifications. In those cases, the SR was retained and added to the LCO it supports.

And, finally, some additions of new requirements are proposed where they are necessary to effect the implementation of the overall improvements encouraged by the Policy Statement.

The specific changes that are proposed are identified in the marked-up Technical Specification pages in Attachment IV.

Evaluation

The Statement of Considerations for the final rule issuing 10 CFR 50.36, Technical Specifications, discusses the scope of Technical Specifications as including the following:

> In the revised system, emphasis is placed on two general classes of technical matters (1) those related to prevention of accidents, and (2) those related to mitigation of the consequences of accidents. By systematic analysis and evaluation of a particular facility, each applicant is required to identify at the construction permit stage, those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity. Such items are expected to be the subject of Technical Specifications in the Operating License.

The Policy Statement also cites the subjective statement of the purpose of Technical Specifications expressed in ASLAB-531, 9 NRC 263 (1979): Technical Specifications are reserved for those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

Over the years, various requirements have been incorporated into Technical Specifications even though they do not satisfy the criteria for inclusion in Technical Specifications stated in the above documents. To remedy this situation, the Policy Statement encourages licensees to implement a program to upgrade Technical Specifications by screening existing requirements using four criteria intended to refocus the Technical Specifications consistent with the Atomic Energy Act, 10 CFR 50.36, and previous interpretations of the regulations governing Technical Specifications. The Policy Statement says that LCOs that do not meet any of the four criteria may be proposed for removal from the Technical Specifications and relocation to licensee-controlled documents.

The Policy Statement further endorses the premise that removal of specifications that do not meet one or more of the retention criteria would constitute an enhancement to safe plant operation by focusing greater attention on the significant operational conditions that would remain in the Technical Specifications.

A screening of the Technical Specifications has been performed using the four criteria specified in the Policy Statement. The details of the screening are provided in Attachment V which is considered part of this Safety Evaluation. Based on the screening, all or parts of 38 Technical Specifications were identified as not meeting any of the criteria and, therefore, as candidates for removal. Since they do not satisfy any of the criteria, the Technical Specifications that are proposed for relocation do not constitute performance requirements necessary to ensure safe operation of the plant. Therefore, relocating these specifications would not have a detrimental effect on safe operation of the plant.

Technical Specification provisions that would be relocated to USAR Chapter 16 would be subject to the controlled process that governs USAR revisions. USAR changes are evaluated in accordance with 10 CFR 50.59 requirements. The results of these safety evaluations are documented and reported to the NRC per 10CFR50.59(b)(2), and the USAR is periodically revised per 10CFR50.71(e). The effect of relocating specifications to a licensee-controlled document is to assure that requirements that are not important for accident prevention or mitigation, but may be important to the licensing design basis of the plant, are maintained and controlled. For these specifications, relocation to the USAR is appropriate because that document contains the licensing basis description of the plant.

The Policy Statement also encourages licensees to use the new Standard Technical Specifications as the basis for proposed plant-specific Technical Specification changes. To this end, the new STS were reviewed to identify appropriate additional requirements to be incorporated.

For the purpose of evaluation, the proposed Technical Specification changes have been categorized as follows:

A. Specifications relocated intact to USAR Chapter 16;

B. Specifications relocated with portions retained in the Technical Specifications;

C. Specifications relocated with programmatic requirements referenced in Section 6 of the Technical Specifications;

D. Modifications to retained specifications to accommodate relocation of other specifications; and

E. New specification requirements incorporated into the Technical Specifications.

Proposed changes in categories A, B, C, and D result directly from application of the four screening criteria in the Policy Statement and the guidance provided by previous NRC staff evaluations of Westinghouse Standard Technical Specifications (STS). Therefore, these categories of proposed changes involve Technical Specification provisions that are neither of controlling importance to operational safety of the plant nor derived from the safety analysis report or Probabilistic Safety Assessment (PSA) information. Those requirements that will be relocated to the USAR will be maintained in accordance with the administrative controls and 10 CFR 50.59 change process applied to the information and commitments contained in the USAR. Any changes to information in the USAR must undergo a review to assure that the changes do not involve an unreviewed safety question prior to implementation of the changes.

Based on the above discussion, the proposed changes in categories A through D involve a relocation of requirements without a reduction in scope or enforceability. The proposed changes would not result in changes to the operation of the plant prior to or after any postulated design basis events. Therefore, the changes would not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR or create a possibility for an accident or malfunction of a different type than any previously evaluated in the USAR. In addition, the proposed changes, since they do not involve changes in plant or equipment operation, would not cause a reduction in the margin of safety as defined in the Technical Specification Bases. With regard to this conclusion, the Bases for Boration Systems, which would be relocated under the proposed revisions, state that the boration systems ensure that negative reactivity control is available during each mode of facility operation. Based on the evaluations contained in WCAP-11618, "Methodically Engineered, Restructured, and Improved Technical Specifications, MERITS Program - Phase II," and USAR Section 15.4.6, the Boration Systems are not required to operate to mitigate the consequences of a design basis accident or transient. The required negative reactivity in Modes 1 and 2 is provided by the shutdown and control rods maintained at their insertion limits and, for certain accidents which add positive reactivity, the boration capability of the Emergency Core Cooling System (ECCS) is credited. During Modes 3, 4, and 5, mitigation of a postulated boron dilution event is accomplished by terminating the dilution. Relocation of these specifications would not cause a reduction in the safety margins stated in the Bases. Also, retention of the relocated specifications in Chapter 16 of the USAR would ensure the availability of these systems when required to initiate boration to regain the required shutdown margin. The screening forms in Attachment V provide additional information.

Proposed changes in category E consist of new Technical Specification LCOs that would be added to the plant Technical Specifications. These additions are necessary to (1) effect the retention of portions of relocated specifications as recommended in the NRC's evaluation of the Westinghouse STS, Revision 5, and (2) accommodate the Policy Statement recommendation to utilize the industry experience embodied in the new STS (NUREG-1431). The following additions to Technical Specifications were considered (the numbers in parentheses are from the new STS):

1. Core Reactivity (3.1.3). This specification has been added as LCO 3.1.1.5. Currently this requirement exists as a surveillance requirement under the shutdown margin specification.

2. Physics and shutdown margin test exceptions (3.1.9, 3.1.10, 3.1.11). Similar specifications are already contained in the Technical Specifications under 3/4.10, Special Test Exceptions. No new requirements were added.

3. Loss of Power Diesel Generator Start Instrumentation (3.3.5). These requirements already exist as Engineered Safety Features Actuation System (ESFAS) instrumentation. No new requirements were added.

4. Accident Monitoring Instrumentation for Neutron Flux and Reactor Vessel Level Indicating System (RVLIS). These instruments were added based on an evaluation of necessary operator actions following postulated accidents. Action Statements from the new STS were adopted for all of LCO 3.3.3.6 and LCO 3.6.4.1 was deleted, as discussed below.

5. Boron Dilution Protection System (3.3.9). Surveillance requirements already exist for this system. A future license amendment request will address the calculations discussed above, as well as the design basis and actuation setpoint for this system at Wolf Creek. At this time, no new requirements were added.

6. Reactor Coolant System (RCS) Loops Test Exceptions (3.4.19). A similar specification is already contained in the WCGS Technical Specifications under 3/4.10, Special Test Exceptions. No new requirements were added.

7. Seal Injection Flow (3.5.5). This is adequately addressed in the Technical Specifications under 3/4.4.6.2. If flow characteristics are modified, Surveillance Requirement 4.5.2.h ensures that the effect of seal injection flow is considered in the flow balance. No new requirements were added.

8. Main Feedwater Isolation Valves (MFIVs) (3.7.3). The main feedwater isolation valves have been added to new WCGS Technical Specification 3/4.7.1.7. Adding this new specification requires a change to Note *** of TS Table 3.6-1, Containment Isolation Valves, to reference the new specification.

9. Unborated Water Source Isolation Valves (3.9.2). The requirement to close manual valves in potential boron dilution flow paths is currently included as a surveillance requirement for the boron concentration LCO under 3/4.9.1, Refueling Operations. No new requirements were added.

The addition of a core reactivity specification, accident monitoring instruments, and the MFIV specification (3/4.7.1.7) for WCGS involve enhancements to the operating safety of the plant by incorporating new requirements of importance to operational safety. These proposed changes would place additional emphasis on maintaining the core design parameters within design limits and on maintaining the operability of components (instrumentation and main feedwater isolation valves) that may be required to mitigate a design basis event. These changes would not involve an increase in the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR; nor would they create the possibility of an accident or malfunction of a different type than any previously evaluated or reduce the margin of safety as defined in the basis for any Technical Specification.

POST-ACCIDENT MONITORING (PAM) LCO CHANGES

Neutron flux indication is provided to verify reactor shutdown over the full range of flux that may occur post-accident. One channel of the Gamma-Metric neutron flux monitoring system provides source range (0.1 to 10^5 cps) and wide range (10^{-8} to 200% power) indication in the main control room (SE-NI-0060A, B). The second channel provides source and wide range indication at the auxiliary shutdown panel, as well as a two-pen indicating recorder (SE-NIR-0061) for both source and wide ranges in the main control room.

Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion. As such, it has been added to 3/4.3.3.6.

The Reactor Vessel Level Indicating System (RVLIS) is provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy. Indication is provided in the main control room (BB-LI-1311, 1312, 1321, 1322).

RVLIS utilizes two sets of two d/p cells. These cells measure the pressure differential between the bottom of the reactor vessel and the top of the vessel. This d/p measuring system utilizes cells of differing ranges to cover different flow behavior with and without pump operation as discussed below:

(a) Reactor Vessel - Narrow Range (ΔP_b)

This measurement provides an indication of reactor vessel level from the bottom of the reactor vessel to the top of the reactor during natural circulation conditions.

(b) Reactor Vessel - Wide Range (ΔP_c)

This instrument provides an indication of reactor core and internals pressure drop for any combination of operating RCPs. Comparison of the measured pressure drop with the normal, single-phase pressure drop provides an approximate indication of the relative void content or density of the circulating fluid. The indication of coolant density is significant only when the subcooling is near zero. This instrument monitors coolant conditions on a continuing basis during forced flow conditions.

To provide the required accuracy for level measurement, temperature measurements of the impulse lines are provided. These measurements, together with existing reactor coolant temperature measurements and wide-range RCS pressure, are employed to compensate the d/p transmitter outputs for differences in system density and reference leg density, particularly during the change in the environment inside the containment structure following an accident. The Wolf Creek design does not include a measurement of reactor vessel level above the hot legs. As such, RVLIS has been added to Technical Specification 3/4.3.3.6.

The PAM action statements have been revised per the new STS. New Action a applies when one or more functions have one required channel that is inoperable, requiring restoration of the inoperable channel to OPERABLE status within 30 days. The 30 day completion time is based on operating experience and takes into account the remaining OPERABLE channel for those functions with two or more channels, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval. If the channel(s) is not restored in time, a Special Report is submitted within 14 days per Specification 6.9.2. This report would discuss the results of the root cause evaluation of the inoperability and identify proposed restorative actions. This action is appropriate instead of a shutdown requirement because: 1) redundant channel(s) are available, and 2) the small likelihood that unit conditions during the period of inoperability would require information provided by this instrumentation.

Two instrument functions are identified in Technical Specification Table 3.3-10 that have one indication per steam generator. The above discussion of new Action a is not applicable for these instrument functions that have no redundancy. However, even though channel redundancy is not available, diverse indications are available. Loss of the single channel would be addressed under new Action b for these two instrument functions.

There is one wide range water level indicator for each steam generator (AE-LI-0501 through -0504 in the main control room). Diverse indications are available from four narrow range level indicators for each steam generator (SG) when on scale, three steamline pressure indicators per loop, and one AFW flow indicator per SG as discussed in USAR Table 7A-3 Data Sheet 4.1. It is noted that wide range SG level is not a Type A variable at Wolf Creek and it is only being retained in Table 3.3-10 due to its relative significance from a human recovery action perspective as identified in the Wolf Creek IPE Summary Report.

There is one AFW flow rate indicator for each SG (AL-FI-0001A through -0004A in the main control room). Although not required for RG 1.97 Category 2 variables, diverse indications are available from one wide range level indicator and four narrow range level indicators per SG. As discussed in USAR Table 7A-3 Data Sheet 5.1, each of these four flow indicators is powered by a different separation group. Since only two of four SGs are required to establish a heat sink for the RCS, flow indication to at least two intact SGs is assured even if a single failure is assumed. Section 22 of the SER, NUREG-0881 (which refers to NUREG-0830), specifically accepted the response to NUREG-0737 Item II.E.1.2 Part 2 for AFW flow rate indication and also noted that wide range SG level is provided. Additional discussion is found in USAR Sections 10.4.9 and 18.2.8. It is noted that AFW flow rate indication is not a Type A variable at Wolf Creek, nor is it a RG 1.97 Category 1 variable, and it is only being retained in Table 3.3-10 due to its relative significance from a human recovery action perspective as identified in the Wolf Creek IPE Summary Report.

New Action b applies to the above single channel instrument functions as well as when one or more multiple channel functions have two required channels inoperable (i.e., two channels inoperable in the same function), requiring restoration of one channel in the functions(s) to OPERABLE status within 7 days. The completion time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation. Continuous operation in this status is not acceptable. Therefore, requiring restoration of one inoperable channel of the function limits the risk that the PAM function will be in a degraded condition should an accident occur. Action b excludes hydrogen monitor channels, as discussed below under new Action d. Action b also excludes the containment radiation monitors and RVLIS, based on the existence of preplanned alternate means of monitoring. If Action b is not met, the plant must be shutdown.

Alternate methods of monitoring containment area radiation and RVLIS have been identified. For containment radiation level (high range), diversity is provided by portable survey equipment with the capability to detect gamma radiation over the range 1E-03 to 1E04 R/hr, maintained in the site health physics instrument inventory as discussed in USAR Table 7A-3 Data Sheet 17.3 The post-accident sampling system (PASS) also provides diversity, as discussed in USAR Table 7A-3 Data Sheet 11.1. Although not designed with the same high range, further diversity is available from the containment atmosphere radiation monitors (GT-RE-0031 and -0032) which display at the digital radiation monitoring panel SP067. Section 22 of NUREG-0881 (which refers to NUREG-0830) specifically accepted the response to NUREG-0737 Item II.F.1 Attachment 3. Additional discussion is found in USAR Section 18.2.12. For RVLIS, diversity is provided by the 46 core exit thermocouples, pressurizer level indication (BB-LI-0459A, -0460A, and -0461A), and RCS subcooling monitor indication (BB-TI-1390A,B). Additional discussion is found in USAR Table 7A-3 Data Sheet 1.4. If these alternate methods are used, new Action c does not require a plant shutdown; rather, a Special Report is submitted within 14 days per Specification 6.9.2. The report provided to the NRC would discuss the preplanned alternate methods used, outline the cause of the inoperability, and provide a schedule for restoring the normal PAM channels.

New Action d applies when two hydrogen monitor channels are inoperable, requiring the restoration of one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour completion time is reasonable based on the backup capability of the Post Accident Sampling System to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA (which would cause core damage) would occur during this time. Consistent with the new Standard Technical Specifications, NUREG 1431, LCO 3.6.4.1 is deleted since LCO 3.3.3.6 contains the appropriate actions and surveillances.

PORV and PORV block valve position indicators have been deleted from Technical Specification 3.3.3.6. Loss of position indication requires that the Actions associated with LCO 3.4.4 be entered; therefore, there is no need to also have these indicators under LCO 3.3.3.6. It is further noted that these indicators are not Type A variables at Wolf Creek, nor are they RG 1.97 Category 1. Monthly channel checks for these indicators have been added as SR 4.4.4.3 and SR 4.4.4.4.

DETERMINATION OF NO UNREVIEWED SAFETY QUESTION

The proposed changes to the Technical Specifications do not involve an unreviewed safety question because the operation of the WCGS in accordance with these proposed changes would not:

(1) Involve an increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Overall protection system performance will remain within the bounds of the accident analyses documented in USAR Chapter 15, WCAP-10961-P, and WCAP-11883, since no hardware changes are proposed.

    There will be no effect on these analyses, or any other accident analysis, since the analysis assumptions are unaffected and remain the same as discussed in the USAR.

    Safety-related equipment will continue to function in a manner consistent with the above analysis assumptions and the plant design basis. As such, there will be no degradation in the performance of nor an increase in the number of challenges to equipment assumed to function during an accident situation.

    These Technical Specification revisions do not involve any hardware changes nor do they affect the probability of any event initiators. There will be no change to normal plant operating parameters, ESFAS actuation setpoints, accident mitigation capabilities, accident analysis assumptions or inputs. These changes are administrative in nature, in that they relocate those requirements not important to accident prevention or mitigation from the Technical Specifications to USAR Chapter 16. Therefore, these changes will not increase the probability or consequences of an accident or malfunction.

(2) Create the possibility for an accident or malfunction of a different type than any previously evaluated in the USAR. As discussed above, there are no hardware changes associated with these Technical Specification revisions nor are there any changes in the method by which any safety-related plant system performs its safety function. The normal manner of plant operation is unaffected.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes. Therefore, the possibility of a new or different type of accident is not created.

There are no changes which would cause the malfunction of safety-related equipment, assumed to be operable in the accident analyses, as a result of the proposed Technical Specification changes. No new mode of failure has been created and no new equipment performance burdens are imposed. These changes are administrative in nature, in that they relocate those requirements not important to accident prevention or mitigation from the Technical Specifications to USAR Chapter 16. Therefore, the possibility of a new or different malfunction of safety-related equipment is not created.

(3) Involve a reduction in the margin of safety as defined in the basis for any Technical Specification. There will be no change to the DNBR limits, or the safety analysis DNBR limits discussed in Bases Section 2.1.1.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined, nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on DNBR limits, F_Q, $F\Delta H$, LOCA PCT, peak local power density, or any other margin of safety.

Conclusions

Based on the above evaluation, including the supporting information in Attachment V and the considerations presented in the No Significant Hazards Consideration Determination, the proposed changes to the Technical Specifications would not involve an increase in the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR; or create the possibility of an accident or malfunction of a different type than any previously evaluated; or reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, these proposed changes would not adversely affect or endanger the health or safety of the general public.

ATTACHMENT II

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

No Significant Hazards Consideration Determination

This license amendment request proposes to revise the Technical Specifications to implement the improvements endorsed in the Nuclear Regulatory Commission's Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, 58 FR 39132, July 22, 1993 (the Policy Statement). These improvements involve focusing the Technical Specifications on those requirements that are of controlling importance to operational safety by screening each Technical Specification in Sections 3/4.1 through 3/4.11 using the criteria provided in the Policy Statement. Those criteria are intended to identify requirements derived from the analyses and evaluations included in the Updated Safety Analysis Report (USAR) that are of immediate concern to the health and safety of the public. Technical Specifications that meet one or more of the criteria must be retained. Those that meet none of the criteria may be removed from the Technical Specifications. The purpose of this amendment request is to remove the specifications that do not meet any of the four Policy Statement criteria.

The removed Technical Specifications will be relocated to USAR Chapter 16, "Technical Specifications". In general, the Technical Specifications that are proposed for relocation would be incorporated into the USAR with the same format and content they possessed as part of the Operating License.

In some cases, the Technical Specification Limiting Condition for Operation (LCO) did not meet any of the criteria for retention, but an associated Surveillance Requirement (SR) was required to support an LCO that was being retained in Technical Specifications. In those cases, the SR was retained and added to the LCO it supports.

And, finally, some additions of new requirements are proposed where they are necessary to effect the implementation of the overall improvements encouraged by the Policy Statement.

The specific changes that are proposed are identified in the marked-up Technical Specification pages in Attachment IV.

This proposed amendment has been reviewed per the standards provided in 10 CFR 50.92. Each standard is discussed separately below.

Standard I - Involves a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed Technical Specification changes involve relocating requirements that are not conditions or limitations on reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. The proposed changes were identified through the application of criteria designed to cull those requirements that are not important to operational safety from the Technical Specifications. In this process, selected provisions of the Technical Specifications identified for relocation were retained if necessary to support a Technical Specification that was to be retained. Thus, only specification requirements that have little or no operational safety significance are proposed for relocation. In addition, those requirements that would be relocated will be included in the Updated Safety Analysis Report (USAR) and, therefore, will be controlled and implemented as USAR commitments. In this manner, those requirements that have no operational safety significance but involve maintaining the plant in its as-designed state (for example, through surveillance programs) would be controlled.

In addition, the criteria for identifying requirements to be retained in the Technical Specifications specifically call out, for retention, those structures, systems, or components that are required to mitigate accidents previously evaluated.

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

Standard II - Create the Possibility of a New or Different Kind of Accident from Any Previously Evaluated

The proposed changes involve relocating Technical Specification requirements to another licensee-controlled document. No changes or physical alterations of the plant are involved. Also, no changes to the operation of the plant or equipment are involved. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Standard III - Involve a Significant Reduction in the Margin of Safety

The proposed changes involve relocating Technical Specification requirements to the USAR. The requirements to be relocated were identified by applying the criteria endorsed in the Commission's Policy Statement. Thus, those specifications that would be relocated do not impose constraints on design and operation of the plant that are derived from the plant safety analysis report or from probabilistic safety assessment (PSA) information and do not belong in the Technical Specifications in accordance with 10CFR50.36 and the purpose of the Technical Specifications stated in the Policy Statement. Therefore, relocation of these requirements does not involve a significant reduction in the margin of safety.

In addition, revisions to the USAR will be evaluated in accordance with the 10 CFR 50.59 process which considers the reduction in safety margin. Therefore, any future revisions to the provisions in the USAR will consider reductions in the margin of safety using the criteria for identifying an unreviewed safety question.

Based on the above, the requested Technical Specification changes do not involve a significant increase in the probability or consequences of a previously evaluated accident, create the possibility of a new or different kind of accident, or involve a significant reduction in the margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration in accordance with 10 CFR 50.92(c).

ATTACHMENT III

ENVIRONMENTAL IMPACT DETERMINATION

Environmental Impact Determination

10 CFR 51.22(b) specifies the criteria for categorical exclusions from the requirement for a specific environmental assessment per 10 CFR 51.21. This amendment request meets the criteria specified in 10 CFR 51.22(c)(9) as specified below:

(i) the amendment involves no significant hazards consideration

As demonstrated in Attachment II, the proposed changes do not involve a significant hazards consideration.

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

The proposed changes do not involve generation or release of effluents from the plant. The changes would result in the relocation of Technical Specifications related to radioactive effluents; however, these specifications do not affect generation or release of effluents from the plant. In addition, the other specifications to be relocated do not involve effluent generation or release. Therefore, the proposed changes will have no effect on normal plant effluents, and there will be no change in the types or amounts of any effluents released offsite.

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

The proposed changes will have no effect on general levels of radiation present in the plant; nor will additional quantities of radioactive materials be generated as a result of the proposed changes. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure associated with these proposed changes.

Based on the above, there will be no impact on the environment resulting from the proposed changes and the changes meet the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.21 relative to specific environmental assessment by the Commission.