WOLF CREEK
NUCLEAR OPERATING CORPORATION

Clay C. Warren
Vice President & Chief Operating Officer

OCT 23 1998

WO 98-0082

U. S. Nuclear Regulatory Commission
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Reference: 1) Letter ET 98-0055, dated July 17, 1998, from R. A. Muench, WCNOC, to USNRC
2) Letter dated June 16, 1998, from K. M. Thomas, USNRC, to O. L. Maynard, WCNOC
3) Letter WO 98-0078, dated August 5, 1998, from C. C. Warren, WCNOC, to USNRC
4) Letter ET 97-0050, dated May 15, 1997, from R. A. Muench, WCNOC, to USNRC

Subject: Docket No. 50-482: Revision to Technical Specification 3.5.1, "Emergency Core Cooling Systems - Accumulators"

Gentlemen:

This letter transmits an application for amendment to Facility Operating License No. NPF-42 for Wolf Creek Generating Station (WCGS). This license amendment request proposes revising Technical Specification 3.5.1, "Emergency Core Cooling Systems - Accumulators." Specifically, it is being proposed that the allowed outage time in Action b. (to restore an inoperable accumulator, if inoperable for reasons other than not meeting its boron concentration requirements), be increased to 24 hours in lieu of the current allowed outage time of one hour.

In Reference 1 WCNOC stated our intention to resubmit an application for amendment, for the change described above, as the Westinghouse Owners Group lead plant for the review of WCAP-15049, "Risk Informed Evaluation of an Extension to Accumulator Completion Times." This letter provides that application. The approach used in WCAP-15049 utilizes the NRC Staff guidance presented in Draft Regulatory Guides DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-Specific Changes to the Current Licensing Basis," and DG-1065, "An Approach for Plant-Specific, Risk Informed Decision Making: Technical Specifications."

Reference 4 submitted an application for conversion of the current technical specifications to the improved standard technical specifications, and incorporated the proposed 24 hour allowed outage time. Reference 2 provided a Request for Additional Information associated with Section 3.5 of the improved technical specifications. Reference 3 responded to Comment 3.5.1-2 in which WCNOC revised the improved technical specifications to a 1 hour allowed outage time based on Reference 1. Upon approval of this license amendment request, a supplement to Reference 4 will be provided.

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Attachment I provides a description of the proposed change along with a Safety Evaluation. Attachment II provides a No Significant Hazards Consideration Determination. Attachment III provides the Environmental Impact Determination. The specific change to the current technical specifications proposed by this request is provided as Attachment IV. Attachment V provides a list of commitments made in this submittal.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Kansas State official. This proposed revision to the WCGS Technical Specifications will be fully implemented within 30 days of formal Nuclear Regulatory Commission approval.

If you have any questions concerning this response, please contact me at (316) 364-4048, or Mr. Michael J. Angus at (316) 364-4077.

Very truly yours,

Clay C. Warren

Clay C. Warren

CCW/dlc

Attachments:
- I - Safety Evaluation
- II - No Significant Hazards Consideration Determination
- III - Environmental Impact Determination
- IV - Proposed Technical Specification Change
- V - List of Commitments

cc: V. L. Cooper (KDHE), w/a
W. D. Johnson (NRC), w/a
E. W. Merschoff (NRC), w/a
B. A. Smalldridge (NRC), w/a
K. M. Thomas (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Clay C. Warren, of lawful age, being first duly sworn upon oath says that he is Chief Operating Officer 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.

By Clay C. Warren

Clay C. Warren
Chief Operating Officer

SUBSCRIBED and sworn to before me this 23RD day of October, 1998.

Mary. E. Gifford.

Notary Public

MARY E. GIFFORD
Notary Public - State of Kansas
My Appt. Expires 12/09/1999

Expiration Date 12/09/1999

ATTACHMENT I

SAFETY EVALUATION

PROPOSED CHANGE

This license amendment request proposes revising Technical Specification 3.5.1, "Emergency Core Cooling Systems - Accumulators." Technical Specification 3.5.1 Action b. currently specifies an allowed outage time (AOT) of one hour to restore a reactor coolant system accumulator to operable status when declared inoperable due to any reason except not being within the required boron concentration range. The AOT for Action b. would be increased to 24 hours in lieu of the current AOT of one hour.

SYSTEM DESIGN

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. During normal operation, each accumulator is isolated from the Reactor Coolant System (RCS) by two check valves in series. In addition, each accumulator has a motor-operated isolation valve which is normally open with power removed. Should the RCS pressure fall below the accumulator pressure, the check valves open and borated water is forced into the RCS. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. One accumulator is attached to each of the cold legs of the RCS. Mechanical operation of the swing-disk check valves is the only action required to open the injection path from the accumulators to the reactor core via the cold legs.

Connections are provided to remotely adjust the level and boron concentration of the borated water in each accumulator during power operation. The accumulator water level can be adjusted either by draining to the recycle holdup tank or by pumping borated water from the refueling water storage tank (RWST) to the accumulator. Samples of the solution in the accumulators are taken periodically to verify proper boron concentration.

The motive force for accumulator injection is provided by a nitrogen cover gas. A connection is provided to a normally isolated nitrogen gas supply to allow adjustment of the accumulator nitrogen cover gas pressure, as required, during power operation. The accumulators are also protected from pressures in excess of design pressure by gas relief valves. The accumulator gas pressure is monitored by indicators and alarms. Solenoid-operated vent valves are provided to depressurize the accumulators during emergencies or cold shutdown conditions.

DESIGN BASIS EMERGENCY CORE COOLING SYSTEM ANALYSIS

A Loss-of-Coolant Accident (LOCA) is defined as a rupture of the RCS piping or of any line connected to the system from which the break flow exceeds the flow capability of the normal makeup/charging system. Ruptures of small cross-sections will cause expulsion of the reactor coolant at a rate which can be accommodated by the centrifugal charging pumps maintaining an operational water level in the pressurizer, permitting the operator to execute an orderly shutdown.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the RCS through the postulated break against the centrifugal charging pump makeup flow at normal RCS pressure, i.e., 2,250 psia. A makeup flow rate from one centrifugal charging pump is adequate to maintain the pressurizer level (at 2,250 psia) for break sizes less than or equal to a 0.375 inch diameter hole. This break results in a loss of approximately 17.5 lb/sec (127 gpm at 130 degrees Fahrenheit and 2,250 psia) of reactor coolant.

For the analyses reported in Chapter 15 of the Updated Safety Analysis Report (USAR), a small break is defined as a rupture of the RCS piping with a cross-sectional area less than 1.0 square foot (ft^2), in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. A small break LOCA is classified as an ANS Condition III event (an infrequent fault), as defined in USAR Section 15.0. A major break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the life of a plant but is postulated as a conservative design basis.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 as follows:

A. The calculated peak fuel element clad temperature shall not exceed the requirement of 2,200 degrees Fahrenheit.

B. The amount of the fuel element cladding that reacts chemically with water or steam, shall not exceed 1 percent of the total amount of Zircaloy in the fuel cladding.

C. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

D. The core remains amenable to cooling during and after the break.

E. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

Should a pipe break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. A reactor trip occurs and the safety injection system is actuated when their respective pressurizer low pressure trip setpoints are reached. Reactor trip and safety injection system actuation may be provided by a high containment pressure signal, depending on the actual break size. These countermeasures limit the consequences of the accident in two ways:

A. The reactor trip and borated water injection provide additional negative reactivity insertion to supplement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. However, no credit is taken in the analysis of a LOCA blowdown for negative reactivity due to the boron content of the injection water.

B. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. Thereafter, the core heat transfer is based on local conditions with transition boiling, film boiling, and forced convection to steam as the major heat transfer mechanisms.

When the RCS depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2,300 psia) falls to a value approaching that of the containment atmosphere. Prior to, or at the end of the blowdown, the expulsion or entrainment mechanisms that are responsible for the bypassing of emergency core cooling water injected into the RCS are calculated to no longer be effective. At this time (called end-of-bypass), refill of the reactor vessel lower plenum begins. Refill is complete when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods (called bottom of core recovery time).

The reflood phase of the transient is defined as the time period lasting from the end-of-refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the later stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The centrifugal charging, safety injection, and residual heat removal pumps aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

The accumulators are assumed operable in both the large and small break LOCA analyses at full power. In performing the LOCA calculations, conservative assumptions are made concerning the availability of emergency core cooling system (ECCS) flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The limiting large break LOCA is a double-ended guillotine break at the discharge of the reactor coolant pump. During this event, the accumulators discharge to the RCS as soon as RCS pressure falls below the accumulator pressure. No credit is taken in the analysis for ECCS pump flow until an effective delay time has elapsed. This delay accounts for the emergency diesel generators starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 second increase to account for generation of a safety injection signal. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow (centrifugal charging pumps and safety injection pumps) then providing continued cooling. As break size decreases, the accumulators no longer play a role in terminating the rise in fuel clad temperature and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

EVALUATION

The proposed license amendment increases the accumulator allowed outage time (AOT, referred to as Completion Time in the Improved Technical Specifications) from 1 hour to 24 hours for one accumulator inoperable for conditions other than boron concentration not within specification. In support of this proposed license amendment, the Westinghouse Owners Group initiated a program to evaluate the impact of this change on plant risk on a generic basis using representative calculations. The approach used in this program is consistent with the Nuclear Regulatory Commission's approach for using probabilistic risk assessment in risk-informed decisions on plant-specific changes to the current licensing basis. This approach is presented in the Draft Regulatory Guides DG-1061 ("An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis") and DG-1065 ("An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications"). The approach addresses, as documented in WCAP-15049, the impact on defense-in-depth and the impact on safety margins, as well as an evaluation of the impact on risk. The risk evaluation uses the three-tiered approach as presented by the NRC in Draft Regulatory Guide DG-1065. Tier 1, PRA Capability and Insights, assessed the impact of the proposed AOT change on core damage frequency, incremental conditional core damage probability, large early release frequency, and incremental conditional large early release probability. Tier 2, Avoidance of Risk-Significant Plant Configurations, considered potential risk-significant plant operating configurations. Tier 3, Risk-Informed Plant Configuration Control and Management, was not addressed in WCAP-15049, but is addressed below on a plant-specific basis for WCGS.

Several sets of accumulator success criteria, ranging from that required for design basis analysis to best estimate success criteria used in a number of probabilistic safety analysis (PSA) models, were evaluated in WCAP-15049. The analysis considered 2-, 3-, and 4-loop plants. Sensitivity cases were also evaluated that considered increased initiating event frequencies for medium and small break LOCA events. The following was concluded from this analysis:

- The impact of the increase in the accumulator AOT on core damage frequency (CDF) for all the cases evaluated is within the acceptance limits set by the NRC. The acceptance limit is 1E-06/yr CDF increase providing the total plant CDF is less than 1E-03/yr. The specific values for 4-loop plants are provided in Table 8-7 of the WCAP and are reproduced in Table 1 of this safety evaluation.

- The calculated incremental conditional core damage probabilities (ICCDP) meet the criterion of 5E-07 set by the NRC for the increased AOT except for those that are based on design basis success criteria. Design basis accumulator success criteria is not considered necessary to mitigate large break LOCA events and is only included as a worst case data point. In addition, the NRC has indicated that an ICCDP greater than 5E-07 does not necessarily mean the change is unacceptable. The ICCDP values are provided in Table 8-8 of the WCAP and are reproduced in Table 2 of this safety evaluation.

- The impact on the large early release frequency (LERF) and incremental conditional large early release probability (ICLERP) is similar to the impact on the CDF and ICCDP. Since the success or failure of the containment systems are independent of the accumulators, the LERF will increase only in direct proportion to the increased frequency of the core damage sequences involving accumulator failures. Since the impact of the accumulator AOT increase on CDF is small and the ICCDP is acceptable, the impact of the accumulator AOT increase on LERF will also be small and the ICLERP will also be acceptable.

- The impact of the AOT increase has no impact on defense-in-depth. There is no impact on maintaining a reasonable balance between prevention of core damage, prevention of containment failure, and consequence mitigation. There is no over reliance on programmatic activities. System redundancy, independence, and diversity is maintained; independence of barriers is not degraded; and defenses against common cause failures and human errors are maintained.

- Although the safety margin with regard to accumulator response to design basis large break LOCA events (i.e., a large break LOCA with loss of offsite power) is impacted by extending the AOT, the CDF increase for all large break LOCA cases considered in WCAP-15049 is less than the 1.0E-06/yr acceptance limit (see Table 1 below).

Applicability of the WCAP Evaluation to Wolf Creek Generating Station

To determine the applicability of the WCAP-15049 evaluation to the Wolf Creek Generating Station (WCGS) requires a review and comparison of several relevant PSA modeling parameters and assumptions used in the Westinghouse Owners Group (WOG) study against those comparable parameters and assumptions used in the WCGS PSA model. These parameters and assumptions include:

- Initiating events that require accumulators for mitigation.
- Initiating event frequencies for those events that require accumulators for mitigation.
- Accumulator success criteria for each event for which they are required for mitigation.
- Accumulator maintenance and test intervals.
- Accumulator failure modes.

The initiating events and accumulator success criteria used in the WOG analysis and the corresponding parameter or assumption value used in the WCGS PSA are summarized in Table 4. This shows a favorable comparison. Accumulator injection is included in the WCGS PSA model to mitigate large and medium break LOCAs, but not for mitigation of small break LOCAs. Including the accumulators to mitigate small break LOCAs as part of an alternate success path, as was done in the WCAP analysis, is conservative since accumulator unavailability changes will not impact the small break LOCA core damage frequency contribution in the WCGS PSA model.

With regard to large break LOCAs, the specific success criteria used in the WCGS PSA model, 2 accumulators to 2 of 3 intact loops, was evaluated in the WCAP analysis as PSA Model Basis 1 Case. With regard to medium break LOCAs, the specific success criteria used in the WCGS PSA model, 2 accumulators to 2 of 3 intact loops, was not evaluated in the WCAP analysis. The WCAP analysis medium break LOCA success criteria, 3 accumulators to 3 of 3 intact legs, conservatively envelops the WCGS success criteria.

The large break LOCA initiating event frequency used in the WCGS PSA is 5.0E-04. This is larger than the value used in the WCAP analysis and is considered a conservative value. As discussed in WCAP-15049, the initiating event frequencies for Westinghouse NSSS plants range from approximately 5E-04/year (yr) to 1E-04/yr for large break LOCA break sizes greater than 6 inches diameter, for the most part. Several plants use initiating event frequencies significantly less than this, but their minimum large break LOCA size is much larger (> 12 inches). The mean large break LOCA initiating event frequency is 3.1E-04/yr and a typical value is 3.0E-04/yr. Based on this information, the large break LOCA initiating event frequency that was used in the WCAP analysis was 3E-04/yr. Based on recent work done on risk-informed inservice inspection programs, large break LOCA initiating event frequencies considerably lower than 3.0E-04/yr, by a factor of 2 or 3, can be justified. In addition, the draft report INEEL/EXT-98-00401, "Rates of Initiating Events at U.S. Commercial Nuclear Power Plants, 1987 through 1995," indicates a large break LOCA break frequency of 2.7E-06/critical year for large break LOCAs. To remain consistent with plant specific PSA models, a value of 3.0E-04/yr was used in the WCAP base analysis. The sensitivity analysis on initiating event frequencies also used the same value since increasing it above this frequency would have been overly conservative. Therefore, the WCAP large break LOCA initiating event frequency is consistent with the WCGS PSA.

The medium break LOCA initiating event frequency used in the WCGS PSA is 1.1E-03. This is larger than the medium break LOCA initiating event frequency value used in the WCAP analysis. As discussed in WCAP-15049, the initiating event frequencies for Westinghouse NSSS plants range from approximately 3.4E-05/yr to 2.3E-03/yr for medium break LOCA break sizes ranging from approximately 2 inches to 6 inches. The mean medium break LOCA initiating event frequency is 7.1E-04/yr and a typical value is 8.0E-04/yr. Based on this information, the medium break LOCA initiating event frequency that was used in the WCAP analysis was 8.0E-04/yr with an initiating event sensitivity analysis value of 1.0E-03. The WCAP typical medium break LOCA initiating event frequency is less than the WCGS PSA model medium break LOCA initiating event frequency of 1.1E-03. The WCAP sensitivity analysis medium break LOCA initiating event frequency value of 1.0E-03 is nearly identical to the WCGS PSA model medium break LOCA initiating event frequency value.

The accumulator maintenance and test intervals used in the WCAP analysis are conservative with respect to the WCGS PSA model. Consistent with the WCAP assumption, test activities that cause the accumulators to be inoperable are not done while the plant is at power. In addition, maintenance activities at power that cause an accumulator to be inoperable are restricted to repair or unplanned activities. The WCGS accumulator model assumes these are infrequent activities. That is, their frequency of occurrence is less than that used in the WCAP analysis, which used a frequency of 0.1/year/accumulator.

Accumulator failure modes between the WCAP and WCGS fault tree models are not completely consistent. However, those that differ have no impact on the increase in core damage frequency. They are constants in the analysis which are not impacted by the increased AOT and, therefore, have no impact on the change in CDF.

Finally, the CDF for WCGS is calculated to be 6.3E-05/yr for at power internal events including flooding. This value is significantly below the CDF guideline provided by the NRC in Draft Regulatory Guide DG-1065 for allowing small increases in risk providing the total plant CDF is less than 1.0E-03/yr. A small increase is indicated in Draft Regulatory Guide DG-1065 to be less than 1.0E-06/yr. Although this CDF value does not include external events and shutdown operation, when added to the internal event CDF values, the total CDF for WCGS would remain below 1E-03/yr.

Three Tiered Approach

As discussed previously, the WCAP-15049 risk evaluation uses the three-tiered approach consistent with that presented by the NRC in Regulatory Guide DG-1065. Tier 1, PRA Capability and Insights, which assesses the impact of the proposed AOT change on core damage frequency, incremental conditional core damage probability, large early release frequency, and incremental conditional large early release probability, has been presented and discussed in WCAP-15049 and summarized above, yields acceptable results for WCGS.

Tier 2, Avoidance of Risk-Significant Plant Configurations, is discussed in Section 8.4 of WCAP-10549. As noted in this section, restrictions or limitations on plant system unavailability while one accumulator is unavailable, beyond those currently contained in the Standard Technical Specifications (NUREG-1431, Revision 1) are not necessary. This conclusion is also applicable to WCGS since the supporting analysis in WCAP-15049 is applicable to WCGS.

For Tier 3, Risk-Informed Plant Configuration Control and Management, the risk impact associated with performance of maintenance and testing activities is evaluated in accordance with the Wolf Creek Operational Risk Assessment Program (AP 22C-003). An Operational Risk Assessment is performed for activities within a weekly schedule. Compensatory measures are addressed for activities deemed to be risk significant. The weekly scheduled activities and associated Operational Risk Assessment are reviewed by the WCGS PSA Group and approved by the Plant Manager or his designee. The Operational Risk Assessment Program also addresses the impact on the Operational Risk Assessment due to added or emergent activities and activities which have slipped from the scheduled completion time.

Conclusions

Based on the above discussion, the WCAP-15049 analysis is applicable to the WCGS and consistent with the WCGS PSA model. Therefore, the results and conclusions in the WCAP are applicable to the WCGS and extending the allowed outage time to 24 hours for the conditions when one accumulator is inoperable for reasons other than boron concentration not within specification is acceptable.

Based on the above discussions and the considerations presented in Attachment II, the proposed change does not involve a significant increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report; or create a possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report; or involve a significant reduction in the margin of safety as defined in the basis for any technical specification. Therefore, the proposed change does not adversely affect or endanger the health or safety of the general public or involve a significant safety hazard.

Table 1 Summary of Impact of AOT Time Change on Core Damage Frequency			
Case	CDF w/current AOT	CDF w/extended AOT	CDF Increase
Base Case			
4-loop Plant, Design Basis Case	6.93E-07	9.24E-07	2.31E-07
4-loop Plant, PSA Model Basis 1 Case	6.23E-08	7.77E-08	1.54E-08
4-loop Plant, PSA Model Basis 2 Case	4.57E-08	6.09E-08	1.52E-08
Sensitivity Case Initiating Event Frequencies			
4-loop Plant, Design Basis Case	6.93E-07	9.24E-07	2.31E-07
4-loop Plant, PSA Model Basis 1 Case	1.25E-07	1.61E-07	3.60E-08
4-loop Plant, PSA Model Basis 2 Case	1.09E-07	1.45E-07	3.60E-08

Description of Cases:

Design Basis Case - This case requires accumulator injection only for mitigation of large break LOCA events and uses design basis accumulator success criteria (3 accumulators to 3 intact loops).

PSA Model Basis 1 Case - This case credits improved accumulator success criteria for large break LOCA events, as discussed below, and credits the use of accumulators in responding to medium and small break LOCA events following failure of high pressure injection. The success criteria used in this case are:

- large break LOCA - 2 accumulators injecting into 2 of 3 intact loops required for success, this is the same as the success criteria used in several PSA models and is conservative compared to those that do not require any accumulator injection.
- medium break LOCA - 3 accumulators injecting into 3 of 3 intact loops, this is conservative or the same as the success criteria used in PSA models that model this success sequence.
- small break LOCA - 3 accumulators injecting into 3 of 3 intact loops, this is conservative or the same as the success criteria used in PSA models that model this success sequence.

PSA Model Basis 2 Case - This case credits improved accumulator success criteria for large break LOCA events, as discussed below, and credits the use of accumulators in responding to medium and small break LOCA events following failure of high pressure injection. The success criteria used in this case are:

- large break LOCA - 0 accumulators injecting into 0 of 3 intact loops required for success, this is the same as the success criteria used in several PSA models.
- medium break LOCA - 3 accumulators injecting into 3 of 3 intact loops, this is conservative or the same as the success criteria used in PSA models that model this success sequence.
- small break LOCA - 3 accumulators injecting into 3 of 3 intact loops, this is conservative or the same as the success criteria used in PSA models that model this success sequence.

Initiating event frequencies: The values used in the Base Cases and the Sensitivity Cases are provided in Table 3.

Table 2 Summary of Incremental Conditional Core Damage Probability Results	
Case	ICCDP
Base Case	
4-loop Plant, Design Basis Case	8.20E-07
4-loop Plant, PSA Model Basis 1 Case	5.53E-08
4-loop Plant, PSA Model Basis 2 Case	5.41E-08
Sensitivity Case	
4-loop Plant, Design Basis Case	8.20E-07
4-loop Plant, PSA Model Basis 1 Case	1.30E-07
4-loop Plant, PSA Model Basis 2 Case	1.28E-07

Description of cases: see Table 1

Table 3 Initiating Event Frequencies		
Initiating Event	**Base Case**	**Sensitivity Case**
Large break LOCA	3.0E-04/yr	3.0E-04/yr
Medium break LOCA	8.0E-04/yr	1.0E-03/yr
Small break LOCA	7.1E-03/yr	2.0E-02/yr

Table 4 WCGS/WCAP-15049 Comparison Summary: Initiating Events				
Initiating Event	**Model**	**Initiating Event Frequency (/yr)**	**Accumulator Success Criteria**	**Comments**
Large break LOCA	WCAP-15049	3.0E-04, Base Case 3.0E-04, Sens. Case	0, 2, or 3 accumulators to 3 intact legs	Cases were run with accumulator requirements ranging from 3 accumulators to 3 intact legs to no accumulator required
	WCGS	5.0E-04	2 accumulators to 2 of 3 intact legs	Matches WCAP-15049 PSA Model Basis 1 Case success criteria for large break LOCA
Medium break LOCA	WCAP-15049	8.0E-04, Base Case 1.0E-03, Sens. Case	3 accumulators to 3 intact legs for alternate success path	WCAP-15049 PSA model requires depressurization, accumulator injection, and low pressure injection as alternate success path following failure of high pressure injection. Primary success path, high pressure injection, does not require accumulator injection.
	WCGS	1.1E-03	2 accumulators to 2 of 3 intact legs for alternate success path	WCAP-15049 medium break LOCA success criteria envelops the WCGS PSA model medium break LOCA success criteria
Small break LOCA	WCAP-15049	7.1E-03, Base Case 2.0E-02, Sens. Case	3 accumulators to 3 intact legs for alternate success path	WCAP-15049 PSA model requires depressurization, accumulator injection, and low pressure injection as alternate success path following failure of high pressure injection. Primary success path, high pressure injection, does not require accumulator injection.
	WCGS	2.5E-03	not required	WCGS PSA model does not credit depressurization, accumulator injection, and low pressure injection as alternate success path following failure of high pressure injection.

ATTACHMENT II

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

No Significant Hazards Consideration Determination

This license amendment request proposes revising Technical Specification 3.5.1, "Emergency Core Cooling Systems - Accumulators." Technical Specification 3.5.1 Action b. currently specifies an allowed outage time (AOT) of one hour to restore a reactor coolant system accumulator to operable status when declared inoperable due to any reason except not being within the required boron concentration range. The AOT for Action b. would be increased to 24 hours in lieu of the current AOT of one hour.

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

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The overall protection system performance will remain within the bounds of the accident analyses documented in Chapter 15 of the Updated Safety Analysis Report (USAR), WCAP-10961-P, and WCAP-11883, since no hardware changes are proposed. The impact of the increase in the accumulator AOT on core damage frequency for all the cases evaluated in WCAP-15049 is within the acceptance limit of 1.0E-06/yr for a total plant CDF less than 1.0E-03/yr. The incremental conditional core damage probabilities calculated in WCAP-15049 for the accumulator AOT increase meet the criterion of 5E-07 in Regulatory Guide DG-1065 for all cases except those that are based on design basis success criteria. As indicated in WCAP-15049, design basis accumulator success criteria are not considered necessary to mitigate large break LOCA events, and was only included in the WCAP-15049 evaluation as a worst case data point. In addition, WCAP-15049 states that the NRC has indicated that an ICCDP greater than 5E-07 does not necessarily mean the change is unacceptable.

The safety injection accumulators are credited in Section 15.6.5 of the Updated Safety Analysis Report for large and small break LOCA. 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 Section 15.6.5. Design basis accidents are not assumed to occur during allowed outage times covered by the Technical Specifications. As such, the ECCS Evaluation Model equipment availability assumptions made in Section 15.6.5 remain valid.

The safety injection accumulators 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.

The proposed technical specification change does not involve any hardware changes nor does it affect the probability of any event initiators. There will be no change to normal plant operating parameters, engineered safety feature (ESF) actuation setpoints, accident mitigation capabilities, accident analysis assumptions or inputs. Therefore, this change will not increase the probability of an accident or malfunction.

The corresponding increase in CDF due to the proposed change to increase the AOT of the accumulators from one hour to 24 hours is not significant. Pursuant to the guidance in Section 3.5 of NEI 96-07, Revision 0, "Guidelines for 10 CFR 50.59 Safety Evaluations," the proposed increase in AOT does not "degrade below the design basis the performance of a safety system assumed to function in the accident analysis," nor does it "increase challenges to safety systems assumed to function in the accident analysis such that safety system performance is degraded below the design basis without compensating effects."

Therefore, it is concluded that this change does not increase the probability of occurrence of a malfunction of equipment important to safety.

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

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. This change does not involve any change to the installed plant systems or the overall operating philosophy of WCGS.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. As described in Section 9.1 of the WCAP-15049 evaluation, the plant design will not be changed with this proposed Technical Specification AOT increase. All safety systems still function in the same manner and there is no additional reliance on additional systems or procedures. The proposed accumulator AOT increase has a very small impact on core damage frequency. The WCAP-15049 evaluation demonstrates that the small increase in risk due to increasing the accumulator AOT is within the acceptance criteria provided in Draft Regulatory Guide DG-1065. No new accident or transients can be introduced with the requested change and the likelihood of an accident or transient is not impacted.

The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore, the possibility of a new or different malfunction of safety related equipment is not created.

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

The proposed change does not involve a significant reduction in a margin of safety. There will be no change to the Departure from Nucleate Boiling Ratio (DNBR) Correlation Limit, the design DNBR limits, or the safety analysis DNBR limits discussed in Bases Section 2.1.1.

The basis for the accumulator LCO, as discussed in Bases Section 3/4.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of the WCAP-15049 evaluation, the proposed change will allow plant operation in a configuration outside the design basis for up to 24 hours, instead of 1 hour, before being required to begin shutdown. The impact of this on plant risk was evaluated and found to be very small. That is, increasing the time the accumulators will be unavailable to respond to a large LOCA event, assuming design basis accumulator success criteria is necessary to mitigate the event, has a very small impact on plant risk. Since the frequency of a design basis large LOCA (a large LOCA with loss of offsite power) would be significantly lower than the large LOCA frequency of the WCAP-15049 evaluation, the impact of increasing the accumulator AOT from 1 hour to 24 hours on plant risk due to a design basis large LOCA would be significantly less than the plant risk increase presented in the WCAP-15049 evaluation. It is therefore concluded that the proposed change does not involve a significant reduction in the margin of safety as described in Technical Specification Bases Section 3/4.5.1.

As discussed previously, the performance of the accumulators will remain within the assumptions used in the large and small break LOCA analyses, as presented in USAR Section 15.6.5. Also, 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.

Based on the above discussions it has been determined that the requested technical specification revision does not involve a significant increase in the probability or consequences of an accident or other adverse condition over previous evaluations; or create the possibility of a new or different kind of accident or condition over previous evaluation; or involve a significant reduction in a margin of safety. The requested license amendment does not involve a significant hazards consideration.

ATTACHMENT III

ENVIRONMENTAL IMPACT DETERMINATION

Environmental Impact Determination

10 CFR 51.22(b) specifies the criteria for categorical exclusions from the requirements for a specific environmental assessment per 10 CFR 51.21. This amendment request meets the criteria specified in 10 CFR 51.22(c)(9). The specific criteria contained in this section are discussed below.

(i) the amendment involves no significant hazards consideration

As demonstrated in the No Significant Hazards Consideration Determination in Attachment II, the requested license amendment does not involve any 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 requested license amendment involves no change to the facility and does not involve any change in the manner of operation of any plant systems involving the generation, collection or processing of radioactive materials or other types of effluents. Therefore, no increase in the amounts of effluents or new types of effluents would be created.

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

The requested license amendment involves no change to the facility and will not increase the radiation dose resulting from the operation of any plant system. Furthermore, implementation of this proposed change will not involve work activities which could contribute to occupational radiation exposure. Therefore, there will be no increase in individual or cumulative occupational radiation exposure associated with this proposed change.

Based on the above it is concluded that there will be no impact on the environment resulting from this change. The change meets 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.