



Serial: RNP-RA/06-0030
APR 07 2006

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Ladies and Gentlemen:

Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc. (PEC), submits the attached report in accordance with 10 CFR 50.59(d)(2), "Changes, Tests, and Experiments," for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The report provides a description of changes that were implemented pursuant to 10 CFR 50.59 between April 1, 2004, and April 1, 2006. A summary of the evaluation for each item is also included in the attached report.

If you have any questions concerning this matter, please contact me at (843) 857-1253.

Sincerely,

A handwritten signature in black ink, appearing to read "C. T. Baucom".

C. T. Baucom
Supervisor – Licensing/Regulatory Programs

CTB/cac

Attachment

c: Dr. William D. Travers, NRC, Region II
NRC Resident Inspector, HBRSEP
Mr. C. P. Patel, NRC, NRR

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

JE47

**SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS FOR THE
H. B. ROBINSON STEAM ELECTRIC PLANT (HBRSEP), UNIT NO. 2**

Evaluations performed in accordance with 10 CFR 50.59 for the time period of April 1, 2004, to April 1, 2006:

Evaluation No. 04-0021:

Description:

This evaluation was conducted for certain manual actions in the applicable control room dedicated shutdown procedures (DSP) that are used in the event of a fire scenario (i.e., 10 CFR 50, Appendix R).

Summary of Evaluation:

The activity only impacts procedures used after the identification of a 10 CFR 50, Appendix R, fire scenario. For Appendix R fire scenarios, supporting systems, structures, or components (SSCs) are operated within approved design parameters. The event itself is an Appendix R-related fire in the designated area for which the applicable procedure is designed to mitigate, such that no analyzed accident, as described in the Updated Final Safety Analysis Report (UFSAR), will occur.

The scope of the change pertinent to this evaluation is the requirement for local manual actions. Performance of prescribed local manual actions does not affect the frequency of occurrence of an accident, the likelihood of occurrence of a malfunction of an SSC important to safety, the consequences of a UFSAR evaluated accident, or the consequences of a malfunction of an SSC important to safety.

Additionally, the activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated.

The specific methodology, criteria, basis, and assumptions have been previously established for 10 CFR 50.48, Appendix R, Section III.G.3 postulated fire events by design calculation. The same methodology, criteria, basis, and assumptions were included in the evaluation of manual actions for 10 CFR 50.48, Appendix R, Section III.G.2 applications. Therefore, the activity does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

None of the actions proposed within the revised procedures represent a challenge to a safety-related SSC. The evaluation of the manual actions concludes that design basis limits for any fission product barriers as described in the UFSAR are not affected. Therefore, NRC review and

approval of this activity prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 04-0148:

Description:

This evaluation pertains to changes in the Cycle 23 reload core design, which involves the use of the new or revised analyses of record in the UFSAR relative to those utilized in the previous operating cycle.

Summary of Evaluation:

The activity implements a new core loading pattern and associated analyses in support of Cycle 23 operation, including the new reload batch neutronic design, new core loading pattern, safety analysis, and updates to the UFSAR and plant procedures. The core continues to meet the applicable design, material, and fabrication standards. The reload core does not cause any systems to be operated outside of the current design and testing limits and does not alter the interface of any systems. The acceptance criteria for accident events are still met. Therefore, the Cycle 23 reload design does not result in more than a minimal increase in the frequency of occurrence of an accident, in the frequency of occurrence of a malfunction of an SSC, in the consequences of an accident, or in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR.

Additionally, the activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated.

The proposed activity does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. The described activity requires no changes to any element within the reload evaluation methodologies. Therefore, the applicable methodologies are not changed for Cycle 23 and the proposed activity does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 04-0224:

Description:

This evaluation pertains to changes made to operating procedure GP-007, "Plant Cooldown from Hot Shutdown to Cold Shutdown." The changes allow the use of electrical jumpers to defeat the reactor coolant system (RCS) 543 degrees F average temperature (Tavg) interlock that turns off the steam dump and limits the number of steam dump valves that can be used to three if the bypass is used. After the jumper is installed, the operator can use up to five steam dumps. The procedure changes allow the use of this jumper only when RCS temperature is below 450 degrees F. The intent is to reduce the time required to cooldown the plant during shutdown. Later in the procedure, when the cooldown has been accomplished, the jumper is removed.

Summary of Evaluation:

During the cooldown of the plant, the Steam Dump System is operated in the Steam Pressure mode with the operator controlling the rate of the cooldown using controls available in the control room on the reactor-turbine gauge board. UFSAR Section 10.4.4.3 states that a malfunction of the Steam Dump System, which causes the valves to fail open, results in a load increase equivalent to a small steam line break. A bounding steam line break is analyzed in UFSAR Section 15.1.5. A malfunction of the Steam Dump System that causes the valves to fail open is also similar to USFAR Section 15.1.4, Inadvertent Opening of a Steam Generator Relief or Power Operated Relief Valve. This procedure change does not alter the design or operation of the control circuitry such that the steam dump valves are more likely to fail open. This change does increase the amount of steam dump capacity available (from three valves to five valves) when the Tavg is below 450 degrees F. This change does not increase the likelihood of a malfunction of the Steam Dump System. The changes do not affect the design or function of the circuitry which generates the 543 degrees F Tavg interlock and does not alter the manner in which the Tavg interlock functions when the jumpers are not installed. The procedure will administratively limit the installation of the jumpers to RCS temperatures below 450 degrees F. Therefore, this change does not increase the frequency of occurrence of an accident, the frequency of occurrence of a malfunction of an SSC, the consequences of an accident, or the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Additionally, the activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. This change does not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered, because this change remains bounded by existing accident analyses. This change does not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses. The configuration created by this change remains fully bounded by existing analyses using current evaluation/safety analysis methodologies.

Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 04-0563:

Description:

This evaluation pertains to the to the engineering change to the facility to remove the emergency diesel generator (EDG) back-up field flash battery for the "A" and "B" EDGs.

Summary of Evaluation:

The function of the EDG back-up field flash batteries was to provide back-up supply of 125 Volt DC to the EDG rotor windings, which enables the EDGs to produce an output voltage if the primary supply is lost. The subject batteries were originally installed to mitigate the consequences of a main turbine failure that was postulated to result in a turbine blade being ejected, penetrating the battery room ceiling, and damaging both station batteries. This scenario would result in both EDGs becoming inoperable.

The replacement of the turbine rotor with a fully integral rotor, along with the new analysis provided in a license amendment to delete the turbine redundant overspeed trip system, which shows that the probability of turbine missile ejection is $7.5E-8$, provides the basis that the probability for a destructive overspeed event is below the threshold for consideration. The NRC acceptance criteria for this event is $1.0E-5$, therefore the probability of occurrence of a turbine missile ejection is well below the NRC acceptance criteria. In addition to the extremely low probability of turbine missile ejection, the probability that a turbine ejected missile would penetrate the battery room through the 10-inch concrete ceiling, damage both station batteries, damage the four battery chargers while simultaneously creating a loss of offsite power and disabling the dedicated shutdown system, is not credible. The EDG back-up field flash batteries are not credited for the prevention of any of the events analyzed in Chapter 15 of the UFSAR, nor are they used to reduce the frequency of occurrence of a Chapter 15 accident. Therefore, the removal of the EDG back-up field flash batteries will not increase the frequency of occurrence of an accident previously evaluated, increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR.

Additionally, the activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. This change does not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered, because the consequences of this change remain bounded by existing accident analyses. This change does not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses. The configuration created by this change remains fully bounded by existing analyses using current evaluation and safety analysis methodologies.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 04-0636:

Description:

This evaluation pertains to an engineering change to the facility to replace the existing Nuclear Measurements Corporation stack radiation monitor, R-14, with a state-of-the-art, skid-mounted monitor manufactured and supplied by General Atomic Electronic Systems, Inc. (previously called Sorrento Electronics).

Summary of Evaluation:

The engineering change involves the removal of the continuous monitoring for iodine and particulate in the plant vent exhaust in the normal range. The sampling of the plant vent for both iodine and particulate was not removed. A removal of continuous monitoring while retaining the sampling of the plant vent exhaust and subsequent counting of the filter assemblies using laboratory equipment does not alter the frequency of any analyzed event, nor is an initiator of or used to mitigate any analyzed event. An increase in iodine or particulate activity in the plant vent exhaust is not an accident, but may be the result of an accident. An increase in these sample activities may occur as the result of other conditions or accidents. Regulatory Guide 1.97 states that continuous monitoring for iodine and particulate is not required and determination of activity levels using laboratory methods is acceptable. This change does not increase the frequency of occurrence of an accident evaluated in the UFSAR, because the manner in which the iodine and particulate activity in the plant vent is determined does not initiate any analyzed event or transient described in the UFSAR. It has also been determined that this modification did not increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR.

Additionally, the activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. This change does not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered, because continuous monitoring of the iodine and particulate activity is not an element of any fission product barrier design basis limit. The sampling function is not an evaluation methodology. Therefore, this engineering change does not result in a departure or change in any evaluation methodology described in the UFSAR.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 05-0058:

Description:

This evaluation pertains to changes made to the maximum critical load value and the restricted load path settings that were implemented for the use of the Spent Fuel Cask Handling (SFCH) Crane to move the new dry fuel storage containers at HBRSEP, Unit No. 2.

Summary of Evaluation:

The information submitted to the NRC for Phase II of NUREG-0612 stated the maximum critical load (MCL) for the SFCH Crane as 70 tons plus the weight of the redundant yoke for the IF-300 cask system. This was the MCL handled by the SFCH Crane at the time of the Phase II review. The MCL is stated in maintenance procedure MMM-009. The revised MCL is being re-established at 110 tons, which will envelope the maximum loaded dry fuel cask weight of 108.7 tons. The change in MCL is consistent with the original design basis for the SFCH Crane established in docketed correspondence and accepted by the NRC via issuance of a Safety Evaluation Report dated March 22, 1977.

The modifications to the restricted load path are consistent with the original design requirements and do not change the function. The function of the restricted load path is to ensure the cask is transported along the "safe load path" while in proximity of the Spent Fuel Pool, which prevents the cask from being moved over or into irradiated fuel or safe shutdown equipment, and to prevent the crane from operating in an area where "load hang-up" could occur while handling the spent fuel cask.

Magnetic proximity limit switches are adjusted to provide these limitations. The vertical height adjustment provides more clearance between the pool operating floor and the bottom of the cask. The functionality and accuracy of the revised restricted load path settings shall continue to be verified as required by Technical Requirements Manual (TRM) Section 3.15.

This activity does not increase the frequency of occurrence of an accident, increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR, because these changes maintain the function of the SFCH Crane and associated restricted load path such that the possibility of a load drop is not feasible.

Additionally, the activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to

safety with a different result than any previously evaluated. This change does not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered, because none of the fission product barriers are being affected. These changes do not change any of the elements of the method described in the UFSAR or change from a method described in the UFSAR to another method used in establishing the design bases or in the safety analyses, because the method of evaluation of the fuel handling building structure is not identified or described in the UFSAR. Therefore, these changes are not considered to be a departure from the method of evaluation described in the UFSAR.

Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 05-0196:

Description:

This evaluation pertains to changes made to revise UFSAR Sections 6.3.2.2.3, 6.3.2.2.11, and 6.3.2.2.16 to be consistent with USFAR Figure 6.3.2-1 and the HBRSEP, Unit No. 2, design for safety injection (SI) relief valve SI-857A. Specifically, Section 6.3.2.2.3 is revised from, "Collection of discharges from pressure relieving devices into closed systems..." to state, "Pressure relieving devices discharge into closed systems or areas monitored for radwaste leakage by instrumentation with reliable power sources." Also, Sections 6.3.2.2.11 and 6.3.2.2.16 are revised to remove the statement, "Relief valves are totally enclosed."

Relief valve SI-857A is located in the Boron Injection Tank (BIT) room. The discharge for this valve is to an open drain in the BIT room.

Summary of Evaluation:

The operation of this relief valve is not an initiating event to any accident in the UFSAR. The failure of this valve is excluded for consideration for operability of the SI system based on the existing design and licensing basis for these types of passive components. The offsite dose analysis assumes conservative integrated leakage for ECCS recirculation loop leakage. Leakage through SI-857A is monitored when system pressure is supplied by the high head SI pumps. A hypothetical actuation of SI-857A during the use of the SI system for accident mitigation due to a pressure spike would not be considered a failure of the valve, because the relief valve would be performing its design function.

Therefore, these UFSAR changes will not increase the frequency of occurrence of an accident previously evaluated, increase the likelihood of occurrence of a malfunction of an SSC important to safety, result in more than a minimal increase in the consequences of an accident previously evaluated, or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety, previously evaluated in the UFSAR.

Additionally, the activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. This change does not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered, because the consequences of this change remains bounded by existing accident analyses.

This change does not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses. The plant design and associated design description remain fully bounded by existing analyses using current evaluation and safety analysis methodologies.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 05-0212:

Description:

This evaluation pertains to changes made to the TRM Section 4.4.1 requirements to temporarily decrease the frequency for the turbine valve test during operating Cycle 23 from once per six months to once per 12 months. This change includes a note for TRM Section 4.4.1 Test Requirement (TR) that states, "This TR is not required to be performed during operating Cycle 23, from October 14, 2004 until the completion of Refueling Outage 23, unless the turbine is tripped or taken offline for other reasons."

Summary of Evaluation:

Turbine overspeed event is not an accident evaluated by Chapter 15 of the UFSAR. The UFSAR does discuss this event in Chapter 3.5. This change does not increase the likelihood of a turbine trip, only the probability of an overspeed after the trip. The probability of overspeed after the trip is dependent on the likelihood of the failure of the turbine overspeed protection. The evaluation provided with the TRM change demonstrates that while the probability of such an overspeed event is increased by this change, it is still within the applicable acceptance criteria. Additionally, the estimated increase in probability does not include the partial completion of the required TR 4.4.1 testing on March 5, 2005, that included complete testing of the turbine valves, except the right side stop and governor valves. Specifically, the right governor valves were stroke tested to approximately 17% open and the right stop valve was not tested. Although it was not quantifiable, the actual testing that was conducted results in a lessened effect of the change to the turbine valve test frequency. Thus, the proposed activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

There is no significant increase in the likelihood of turbine missile ejection that could cause malfunction of equipment important to safety, based on maintaining the likelihood of turbine overspeed acceptably low. Thus, the proposed activity will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

It has also been concluded that this TRM change will not result in more than a minimal increase in the consequences of an accident previously evaluated or result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Additionally, the activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. This change does not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered, because the likelihood of a turbine overspeed event remains acceptably low. This change does not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses. The plant design and associated design description remains consistent with existing analyses using current evaluation and safety analysis methodologies.

Therefore, NRC review and approval of this change prior to implementation was not required based on the conclusion that the proposed change did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 166687:

Description:

This evaluation pertains to changes in the Cycle 24 reload core design, which involves the use of the new or revised analyses of record in the UFSAR relative to those utilized in the previous operating cycle.

Summary of Evaluation:

The activity implements a new core loading pattern and associated analyses in support of Cycle 24 operation, including the new reload batch neutronic design, new core loading pattern, safety analysis, and updates to the UFSAR and plant procedures. The core continues to meet the applicable design, material, and fabrication standards. The reload core does not cause any systems to be operated outside of the current design and testing limits and does not alter the interface of any systems. The acceptance criteria for accident events are still met. Therefore, the Cycle 24 reload design does not result in more than a minimal increase in the frequency of occurrence of an accident, in the frequency of occurrence of a malfunction of an SSC, in the consequences of an accident, or in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Additionally, the activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated.

The proposed activity does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. The described activity requires no changes to any element within the reload evaluation methodologies. Therefore, the applicable methodologies are not changed for Cycle 24 and the proposed activity does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 184146:

Description:

This evaluation pertains to the creation the new procedure SP-1529, "RCS to BIT Header Leak Path Determination." The UFSAR discusses over-pressure protection of the SI system in Section 6.3.2, which states, "Two relief valves are associated with the post loss-of-coolant recirculation. One is located outside the containment at the BIT discharge to prevent over-pressure in the header and in the BIT. The high head SI piping leading to the hot legs is protected by a relief valve inside the containment in the test line." With valve SI-878B closed, the section of pipe between valves SI-878B and SI-869, which is normally protected from over-pressurization by relief valve SI-857A, will no longer have over-pressure protection. Procedure SP-1529 monitors the pressure in this line and has steps to reduce pressure, if necessary. Additionally, the testing performed in SP-1529 is not specifically described in the UFSAR. Closure of valve SI-878B is not typical for full power operation and not consistent with normal system operating alignment, as shown in UFSAR Figure 6.3.2-1.

Summary of Evaluation:

Repositioning of the SI valves and measuring system leakage has no effect on the frequency of occurrence of any accident previously evaluated in the UFSAR. Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.5.2, Condition B, limits the amount of time control power can be maintained on motor-operated valve (MOV) SI-857B to 24 hours while in MODES 1, 2, or 3. SP-1529 contains instruction to enter the LCO Action Statement associated with restoration of control power to SI-878B. The Bases for TS LCO 3.5.2, Condition B, discusses the accumulator discharge valves and the potential to inadvertently isolate an accumulator when control power is available. Having control power aligned to valve SI-878B while the valve is closed will allow for rapid restoration by operator action in the control room.

SP-1529 contains guidance to immediately open SI-878B should a valid safety injection signal occur during this test.

TS LCO 3.5.2, Condition A, allows for one train of ECCS to be inoperable for up to 72 hours while in MODES 1, 2, or 3. With the closure of SI-878B, SI Pump A is isolated from the hot leg injection path and SI Pump C is isolated from the cold leg injection path. The design functions performed by high head safety injection are cold leg injection, cold leg recirculation, and hot leg recirculation. Although both pumps are fully available, the closure of SI-878B essentially reduces the number of pumps available for each function to one. This condition is analogous to the condition entered during pump testing, where the pump is isolated from the injection flow path and operates in mini-flow recirculation.

Once SI-878B is closed, the section of pipe between SI-878B and SI-869 becomes isolated from the BIT header over-pressure protection relief valve (SI-857A). SP-1529 contains steps to depressurize this space as soon as SI-878B is closed. Additional steps monitor the pressure in this space and instruct Operations to drain fluid to decrease pressure as needed.

Following testing in SP-1529, Section 8.1, the only section which manipulates this valve, SI-878B is opened electrically. SP-1529 allows for the manual opening of SI-878B, but requires the final stroke be performed electrically via the motor operator.

With the procedural controls discussed above, there is no more than a minimal increase in the occurrence of a malfunction of any SSC due to activities covered by this evaluation. Furthermore, the likelihood of a system malfunction is consistent with TS allowed conditions and other routine activities that are conducted on the SI system (e.g., pump maintenance).

The ECCS acts to minimize the consequences of a design basis accident by delivering flow as needed to cool the core and also by acting as an extension to the containment boundary through which highly radioactive fluid flows. Since highly radioactive fluid flows through the ECCS, any leakage would negatively impact the dose consequences of a design basis accident. The current limit for ECCS leakage is 2 gallons per hour with the actual measured leakage being much less. If a LOCA were to occur and SI-878B failed closed, the ECCS total leakage could only be less or unchanged due to partial system isolation.

SP-1529 contains steps to immediately open SI-878B upon receipt of a valid safety injection signal. Assuming that the valve cannot be opened, either using the MOV or manually, the design functions of the ECCS will not be adversely affected. In this postulated condition, i.e., SI-878B failed closed coincident with a LOCA, sufficient capacity exists in one train of ECCS to provide the required ECCS flow and pressure. The reduction in redundancy under these postulated conditions will have no effect upon the ability of the ECCS to perform its safety-related design function.

Therefore, no increase in the consequences of any accidents will occur because the ability of the ECCS to retain and provide required cooling flow is maintained within limiting license basis requirements.

Furthermore, it has also been concluded that this test will not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Additionally, the activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated. This testing does not result in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered, because the testing activity remains bounded by existing accident analyses. This change does not involve a change to any evaluation methodology used to establish the design bases or to perform the safety analyses. The plant design and associated design description remains consistent with existing analyses using current evaluation and safety analysis methodologies.

Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).