10 CFR 50.59(d)(2)

CP&L
A Progress Energy Company

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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 & Light (CP&L) Company submits the attached report in accordance with 10 CFR 50.59(d)(2), "Changes, Tests, and Experiments," for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The report provides a brief description of changes, tests, and experiments that were implemented pursuant to 10 CFR 50.59 between April 1, 2000, and April 1, 2002. A summary of the safety evaluation for each item is also included in the attached report. The report is being submitted by April 11, 2002, as required.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom.

Sincerely,

B. L. Fletcher III
Manager - Regulatory Affairs

CAC/cac

Attachment

c: Mr. L. A. Reyes, NRC, Region II
NRC Resident Inspector, HBRSEP
Mr. R. Subbaratnam, NRC, NRR

IE47

Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

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 prior to the 2001 rule change:

Evaluation No. 98-0099, Rev. 00

Description:

This activity was a permanent plant modification to mount the air conditioner in the Auxiliary Operator office, inside the Auxiliary Building.

Summary of Evaluation:

This permanent modification did not affect any system, structure, or component (SSC) directly related to the safe operation of the HBRSEP, Unit No. 2. The probability and consequences of accidents remain unaffected. The air conditioning unit is not near or connected to safety equipment. Therefore, it was determined that this modification did not pose an unreviewed safety question.

Evaluation No. 99-0011, Rev. 00

Description:

This modification installed different size movable racks that are used to store and move the reactor vessel closure bolts when they are removed from the reactor vessel.

Summary of Evaluation:

The original stud racks were a 3-by-3 design. The total weight was 7,300 lbs. This was reported to the NRC in response to NUREG-0612. The new design stud racks are a 2-by-3 design. The total weight will be 5,200 lbs. The new stud rack design will maintain the same required safety margins as were imposed on the original stud rack design. Additionally, the new stud racks will be controlled as a heavy load in the same manner as the original stud racks. As a result, the original analysis bounds the new design. The new design will maintain the same margin to safety as the original design, does not create any new failures, and does not change the probabilities of any accidents or malfunctions. Therefore, it was determined that this modification did not pose an unreviewed safety question.

Evaluation No. 99-0046, Rev. 00

Description:

This modification replaced the Instrument Air (IA) dryers "A" and "B." The dryers had utilized a chemical refrigerant that is no longer used in the United States (R-12). The new dryers use R-22

refrigerant. The piping was modified for the new dryers and the prefilters were deleted. A new breaker was installed for the "B" dryer to match the "A" dryer.

Summary of Evaluation:

After the modification, the IA system performed similarly to pre-modification and no new failures were introduced. The "A" and "B" IA dryers produce air quality consistent with descriptions in the Updated Final Safety Analysis Report (UFSAR). Therefore, this change does not pose an unreviewed safety question.

Evaluation No. 99-0071, Rev. 00

Description:

UFSAR pages 9.2.2-2 and 3.8.4-4c were revised to correct inconsistencies with original licensing basis information. The facility was not changed. Only facility descriptions were changed.

Summary of Evaluation:

The changes to the UFSAR clarified descriptions of seismic qualification of certain portions of the Service Water System. There were no plant alterations required to implement these UFSAR changes. Assumed malfunctions of the Service Water System remained unchanged. The Service Water System capability to operate for accident mitigation remained unchanged. Therefore, this change does not pose an unreviewed safety question.

Evaluation No. 99-0362, Rev. 00

Description:

This activity was a permanent plant modification to remove the Non-Contaminated Waste Oil System and the concrete wall that surrounded it.

Summary of Evaluation:

The Non-Contaminated Waste Oil System was not connected to systems or equipment involved in any accident analysis nor was it part of the mitigation strategy for any accident. Therefore, this modification has no impact on any accident analyses, nor on the creation of a new accident. There is no unreviewed safety question posed by this change.

Evaluation No. 99-1081, Rev. 01

Description:

This activity was an evaluation of a change to leave the power supply breakers for the Main Steam Isolation Valve (MSIV) bypass valves in the open position during plant operation. These

are motor-operated valves (MOVs) that are normally in the closed position.

Summary of Evaluation:

The required safety functions of the Main Steam System are not affected as a result of placing the MSIV bypass valve MOV breakers in the open position when the valves are not in use. In summary, this activity does not result in an unreviewed safety question as it is within the bounds of Technical Specifications and does not increase the probability or consequences of previously evaluated design basis accidents or equipment malfunctions.

Evaluation No. 99-1344, Rev. 00

Description:

Recent changes in titles of the Executive Officers within Carolina Power and Light (CP&L) Company are required to be reflected in the HBRSEP, Unit No. 2, UFSAR. The HBRSEP, Unit No. 2, UFSAR was changed as follows: The title of Executive Vice President and Chief Nuclear Officer-Energy Supply was replaced with Senior Vice President and Chief Nuclear Officer - Nuclear Generation Group. The President/Chief Executive Officer was replaced with Executive Vice President - Energy Supply.

Summary of Evaluation:

Based on the review of the UFSAR, Independent Spent Fuel Storage Installation (ISFSI) Safety Analysis Report (SAR), Technical Specifications, and the administrative nature of the specified changes, an unreviewed safety question does not exist. Changes that are administrative in nature do not increase the probability/consequences of an accident, affect the operation of equipment, or reduce the margin of safety.

Evaluation No. 99-1414, Rev. 00

Description:

The Technical Specification Bases, Section B3.4.10, were changed to correct the Pressurizer Safety Relief Valve capacity from 288,000 lbm/hr to 293,330 lbm/hr.

Summary of Evaluation:

The original design value for the Pressurizer Safety Relief Valve capacity was 288,000 lbm/hr. As listed in Section 4.8.2.4 of WCAP-12735, the actual relief capacity for the valves as procured is 293,330 lbm/hr. Chapter 15 of the UFSAR, Table 15.0.8-1, lists the capacity of the safety relief valves as 293,330 lbm/hr. This change to the Technical Specification Bases corrects the relief capacity to agree with that stated in UFSAR, Chapter 15. Therefore, the correction does not increase the probabilities or consequences of any analyzed accident or equipment malfunction, nor does it introduce any new accident or equipment malfunction. The margin of

the Pressurizer Safety Relief Valves remains unchanged. Therefore, this change does not pose an unreviewed safety question.

Evaluation No. 99-1421, Rev. 00

Description:

This activity was a permanent plant modification to add a sprinkler system to the new Maintenance Fabrication Building.

Summary of Evaluation:

The new system was designed in accordance with NFPA requirements and a calculation was performed to ensure the fire pumps can still provide 70 psig and 1000 gpm in addition to the demands of the new sprinkler system. Isolation valves are provided on the new system to ensure isolation of the building and underground piping. Based on the above design considerations and actual testing, the Fire Water System will operate the same as before. The Fire Water System will still be able to perform its required function. This modification does not increase the probabilities or consequences of any analyzed accident or equipment malfunction, nor does it introduce any new accident or equipment malfunction. Therefore, this change does not pose an unreviewed safety question.

Evaluation No. 99-1432, Rev. 00

Description:

This activity was a revision to the Physical Security and Safeguards Contingency Plan, which is a 10 CFR 50.54(p) change to the HBRSEP, Unit No. 2, Physical Security and Safeguards Contingency Plan. The proposed changes involve the elimination of activities described in the plan that have no basis in regulation, administrative changes to update titles, corporation name changes, and correction of typographical errors. Revised wording was also added regarding audit frequency consistent with final rulemaking (64 FR 14814).

Summary of Evaluation:

These changes to the Physical Security and Safeguards Contingency Plan did not affect any SSC used for the mitigation of analyzed accidents. Additionally, these changes did not introduce any new accident sequences or precursors. The requirements for the Physical Security and Safeguards Contingency Plan are not included in the determination of any margin of safety for HBRSEP, Unit No. 2. Therefore, these changes did not pose an unreviewed safety question.

Evaluation No. 99-1619, Rev. 00

Description:

This activity was a change to the UFSAR Liquid Radwaste Processing Sections. UFSAR Sections 1.2.2.4, 9.3.4.2.4, Table 9.3.4-2, and Sections 11.2, 11.2.2.1, 11.2.2.2, and 11.4.2.2 were updated to include items inadvertently omitted during previous UFSAR changes.

Summary of Evaluation:

The changes covered by this evaluation are made to allow the UFSAR to correctly reflect radioactive waste operating practices that are discussed for information purposes in the text. Two identified items in the UFSAR were not consistent with current operating practices for the processing of liquid radioactive waste. The activities involved had 10 CFR 50.59 evaluations performed before implementation; however, the UFSAR sections were not changed. If administrative controls would fail, it would likely result in a release of fewer curies of radioactivity, as the current wastewater management produces water with fewer curies in the waste Holdup Tanks. Additionally, the assumptions in UFSAR, Section 15.7.3, for the maximum instantaneous release of curies to remain below the 10 CFR 20 limits are not affected. Therefore, these changes did not pose an unreviewed safety question.

Evaluation No. 99-1665, Rev. 00

Description:

CP&L Nuclear Generation Group Procedure, REG-NGGC-0004, Revision 2, "Assessment Process," was changed to accurately reflect commitments and divisions of responsibility.

Summary of Evaluation:

These changes were intended to achieve organizational efficiencies. The Nuclear Assessment Section (NAS) and Performance Evaluation Section (PES) both perform oversight and assessment. Those responsibilities are being divided differently while still maintaining organizational independence, and both NAS and PES are still being evaluated by an independent organization. The result is that the same areas are being evaluated with the same level of independence as previously committed to the NRC. There are no changes to the plant or procedures described in the UFSAR, and no changes related to accidents or equipment used for accident mitigation. Therefore, these changes did not pose an unreviewed safety question.

Evaluation No. 99-1684, Rev. 00

Description:

HBRSEP, Unit No. 2, procedure TMM-004, "Inservice Testing Program," was revised to include information relative to trending, and analysis period, to the Inservice Testing (IST) Program for

pumps and valves which are tested to satisfy the ASME Section XI Code requirements as mandated by 10 CFR 50.55a, the UFSAR, and Technical Specifications.

Summary of Evaluation:

These changes were intended to improve the documentation for audits and turnover, and the trace-ability and accuracy of information related to the IST program, in fulfilling the requirements of Technical Specifications, the UFSAR, and 10 CFR 50.55a. Changes to implementing procedures or processes are not intended as a result of this change. Thus, this revision cannot increase the probability of occurrence or consequences of an analyzed accident, nor create an increase in the probability of occurrence of a malfunction of equipment. The overall margin of safety is not reduced as a result of this proposed change. Therefore, these changes did not pose an unreviewed safety question.

Evaluation No. 99-1692, Rev. 00

Description:

This activity included the permanent plant modification to upgrade the existing meteorological tower instrumentation and data collection systems.

Summary of Evaluation:

The Meteorological (Met) Tower System is designed based Regulatory Guide 1.23, Revision 0 (Safety Guide 23), and Regulatory Guide 1.97. This upgrade modification has been evaluated against these documents as well as ANSI/ANS-2.5-1984, which is unofficially endorsed by the NRC via the Second Proposed Revision 1 to Regulatory Guide 1.23. The basic function of the Met Tower System is to provide on-site meteorological data to aid the Control Room and the Emergency Response Organization in determining the path and dispersion rate of an airborne radioactive release. The information provided by the Met Tower System is used for dose projections and assessment of the effect of a release on the health and safety of the general public. While the on-site meteorological data is an aid to Emergency Preparedness in protecting the health and safety of the public, the system does not perform any safety-related functions. The basic function of the Met Tower System will not be adversely affected by this modification. The replacement equipment is state-of-the-art technology supported by the manufacturers with available spare parts and replacements, which improves the system maintainability over the existing system. The new equipment will improve reliability and allow necessary maintenance to be accomplished more efficiently. This system is located north of the plant, outside the Protected Area, and does not interface with any safety-related system or any system otherwise required for safe operation of the plant. This system is not a mitigation system for any analyzed accident or to any analyzed malfunction of safety-related equipment. The proposed change does not affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, the proposed change is not an unreviewed safety question.

Evaluation No. 00-0049, Rev. 00

Description:

This activity was a change to the Off-Site Dose Calculation Manual (ODCM) to correct the due date for the Annual Radiological Environmental Operating report, and to redefine fish sampling locations for control sampling.

Summary of Evaluation:

The proposed activity, ODCM, Revision 18, was revised to correct the date required for submittal of the Annual Radiological Operating report to the date in the Technical Specifications, and to improve the definition of the sampling locations for obtaining control fish samples in ODCM Table 4.5-1. The date in the ODCM was relocated from the old Technical Specifications when the Technical Specifications were converted by Amendment No. 176. The ODCM and Technical Specifications submittal date is now consistent. The change to sample point 47 allows control samples to be taken from lakes at other locations that are not influenced by plant discharges. These changes to not impact the probability or consequences of accidents or equipment malfunctions, or equipment failures, nor do they introduce new failure modes or effects, nor do they impact the safety margins of the plant. Therefore, these changes do not pose an unreviewed safety question.

Evaluation No. 00-0058, Rev. 00

Description:

The note in UFSAR Section 15.6.2 was changed to correct the UFSAR description for the Small Break Loss-of-Coolant Accident (SBLOCA) to make the description consistent with the licensing basis for the event. Specifically, the statement that the SBLOCA is a Condition IV event analyzed to Condition III criteria was determined to be incorrect. Based on a review of the criteria in the UFSAR and the licensing basis for the event, the SBLOCA is a Condition IV event, and the analysis performed to meet 10 CFR 50.46 criteria was correctly applied to Condition IV event criteria.

Summary of Evaluation:

These UFSAR changes corrected the UFSAR description for the SBLOCA to make the description consistent with the licensing basis for the event. The relevant licensing basis information is Amendment No. 115, dated March 7, 1988, to the Operating License, which limited THERMAL POWER to 60% Rated Thermal Power (RTP), and Amendment 119, dated June 20, 1988, to the Operating License, which restored the authorized power level to 100% RTP. The reason for these license amendments was to ensure that the analyses bounded operation after the Emergency Core Cooling System (ECCS) was modified to remove the swing Safety Injection (SI) pump. Both CP&L submittals to the NRC, and both NRC Safety Evaluations, used the criteria of 10 CFR 50.46 for acceptance of the analyses, including the SBLOCA. The UFSAR defines Condition III and IV events based on American Nuclear Society

criteria as follows: CONDITION III - Infrequent Faults - Events which are expected to occur once during the lifetime of the plant. CONDITION IV - Limiting Faults - Events which are not expected to occur but which are evaluated to demonstrate the adequacy of the design. The acceptance criteria presented in UFSAR Section 15.0.1 for Condition IV events is 10 CFR 50.46. The described UFSAR change does not pose an unreviewed safety question, because the criteria by which the SBLOCA accident is analyzed does not affect the operation of accident mitigation systems, or the sequence or initiation of any accident.

Evaluation No. 00-0094, Rev. 00

Description:

A note was added to UFSAR Table 3.7.3-2 stating that the loading information in the table is the result of only one of two 2-direction shocks. The table is in the UFSAR only to demonstrate that the original reactor coolant loop analysis did not significantly change as a result of the revision of the analysis required by IE Bulletin 79-07 (Seismic Analysis of the Reactor Coolant Loop).

Summary of Evaluation:

The addition of the note clarifying the content of the table ensures that the information is not misinterpreted. The table is used to demonstrate that the original reactor coolant loop piping analysis remains valid by comparing the loads to the revised analysis required by IE Bulletin 79-07. Adding the note does not change the conclusion the table. These changes do not impact the probability or consequences of accidents or equipment malfunctions, or equipment failures, nor do they introduce new failure modes or effects, nor do they impact the safety margins of the plant. Therefore, these changes do not pose an unreviewed safety question.

Evaluation No. 00-0141, Rev. 00

Description:

This change was a permanent plant modification to replace the Dedicated Shutdown (DS) System Uninterruptable Power Supply (UPS) battery.

Summary of Evaluation:

The Dedicated Shutdown UPS is part of the Dedicated Shutdown System. The DS System provides a reliable source of power for loads required to achieve and maintain safe shutdown during the following conditions: (1) In the event of a fire in accordance with 10 CFR 50, Appendix R (Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979), Item III.g requirements; and (2) In the event of a loss of all AC power (Station Blackout - SBO) in accordance with 10 CFR 50.63 requirements (loss of all alternating current power). The DS UPS (and the new battery) has been designed to provide a reliable source of 120V AC power for operation of DS instrument loads. The battery has been sized to conservatively accommodate the instrument loads, to accommodate the addition of Annunciator Panel Procedure, APP-25, to account for prolonged float charging, and to account for the effects of aging. The sizing of the

battery was performed in such a manner as to provide an abundant margin of battery capacity. The new battery will not only maintain the required minimum voltage for one hour following a SBO event, but will continue to provide the minimum required voltage for much longer than one hour. The increased battery capacity will also provide the additional advantage of maintaining the output voltage much higher than the minimum voltage at the end of the one hour SBO discharge, therefore, the DS UPS inverter will operate more efficiently as a result of the higher battery capacity. The replacement of the DS UPS will not change the basic function of the DS UPS. The new replacement battery has been designed to drastically exceed this minimum discharge time, therefore, the ability of the DS UPS to perform its basic design function has been improved without altering the design function. This change does not affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not pose an unreviewed safety question.

Evaluation No. 00-0145, Rev. 00

Description:

This activity was an engineering operability evaluation of possible deficiencies in the support of instrument air tubes that are on the outlet of the instrument air solenoid valves that operate containment isolation valves PS-956A through PS-956H.

Summary of Evaluation:

The engineering operability evaluation concluded that applicable instrument air tubing for containment isolation valves PS-956A through PS-956H meets the applicable operability limit criteria. This demonstrates that the tubing will perform its safety-related design function, which is to ensure kinking of the tubing does not prevent proper venting through the fail-open, safety-related solenoid valves. This ensures the function of the containment isolation valves is unchanged and that all design commitments regarding the valves are maintained. Since the function of the containment isolation valves will not be impacted, the engineering operability evaluation concludes that the existing tubing configuration is acceptable until a long-term fix can be implemented. This change does not affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question. This type of operability evaluation meets the guidelines of NRC Generic Letter 91-18.

Evaluation No. 00-0154, Rev. 00

Description:

This activity was the evaluation of a change in the material used for the reactor vessel head adapter plugs from A182-Type 304 to type A276-Type 304 stainless steel.

Summary of Evaluation:

During manufacture and installation of the HBRSEP, Unit No. 2, reactor vessel, ASTM A276-Type 304 stainless steel material was used, in lieu of A182-Type 304, to produce the reactor vessel head adapter plugs. Eleven of these plugs were installed which cap the unused head penetration adapters and provide a Class 1 pressure boundary. The use of ASTM A276 material is not authorized for use as a primary pressure boundary component per the ASME Code. Therefore, acceptance of the as-found condition constitutes a change to the plant as described in the FSAR. The evaluation completed for the Westinghouse Nuclear Safety Advisory Letter (NSAL), NSAL-98-008, shows there is essentially no difference in the chemical and mechanical properties, and the material supplied is manufactured to have a quality as good as forgings of A182-F304 at that time. The substitution was previously evaluated under the Westinghouse QA program at the time of manufacture, and again for the NSAL, as functionally equivalent and not representing a substantial safety hazard. A 10 CFR Part 21 notification was not considered necessary and the recommendation was the licensee should disposition the material pursuant to NRC Generic Letter 91-18, as a "non-conforming material, accepted as-is." The material evaluation confirmed the material acceptability. Based on the evaluation of the material acceptability, this change does not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, the use of ASTM A276-Type 304 material for the HBRSEP, Unit No. 2, reactor vessel head adapter plugs does not result in an unreviewed safety question.

Evaluation No. 00-0161, Rev. 00

Description:

This activity was a change to the inservice test frequency for the letdown orifice isolation valves, CVC-200A, CVC-200B, and CVC-200C, from quarterly to cold shutdown.

Summary of Evaluation:

The test frequency change was technically justified and determined to be in compliance with ASME Code requirements. This change in frequency did not increase the probability or consequences of an accident, or the malfunction of equipment important to safety. This change did not create a new or different accident, or a malfunction of equipment important to safety. This change did not affect any margin of safety. Therefore, the change did not pose an unreviewed safety question.

Evaluation No. 00-0163, Rev. 00

Description:

This activity was a change to the inservice test frequency for the Containment Spray Pump discharge isolation valves, SI-891A and SI-891B, from quarterly to cold shutdown.

Summary of Evaluation:

This activity involves the revision to the Inservice Testing Program for pumps and valves outlined in procedure, TMM-004, "Inservice Testing Program," which are tested to satisfy the ASME Section XI Code requirements as mandated by 10CFR 50.55a, the UFSAR, and Technical Specifications. This change will ensure continued ASME Code compliance, while assuring that the safe operation of the facility is maintained. The revision does not alter ASME Code requirements, nor require relief or prior approval from the NRC to implement. Since these changes are technically justified and in compliance with ASME Code requirements, they will not increase the probability of an accident, increase the probability of equipment important to safety or otherwise to malfunction, increase the consequences of an accident due to equipment failure, nor reduce the margin of safety as defined in Technical Specifications, and as intended by the UFSAR, and 10 CFR 50.55a. Therefore, this change does not pose an unreviewed safety question.

Evaluation No. 00-0184, Rev. 00

Description:

This activity revised the ISFSI fence as described in Sections 3.3.5.1 and 7.1.3 of the ISFSI SAR to agree with Figure 1.1-2 of the ISFSI SAR, Figure 1.2.2-1 of the UFSAR, and the "as-built" condition. The ISFSI was installed in two (2) phases. Phase I, Modification 885, installed the first three (3) storage modules. Phase II, Modification 939, installed the next five (5) storage modules. A fence was included in the installation. The fence was to provide normal radiological access control. ISFSI SAR Sections 3.3.5.1 and 7.1.3 describe the fence as "around" the ISFSI. The fence for the ISFSI is not described in the UFSAR other than the layout plan, which depicts the "as-built" configuration.

Summary of Evaluation:

The current configuration for the ISFSI fence adequately provides radiological access control as was discussed in the ISFSI SAR section discussed above. There is no need to modify the fence. The above-listed ISFSI SAR sections do require minor clarification. The UFSAR is unaffected by this change. As stated above, the UFSAR currently shows the "as-built" configuration. There are no equipment or accidents as described in the UFSAR that are affected by this change. Clarifying the ISFSI SAR does not impact any accidents described in the UFSAR and does not create any new accidents not described in the UFSAR. As a result, no unreviewed safety question was created by the fence configuration clarification in the ISFSI SAR.

Evaluation No. 00-0187, Rev. 00

Description:

This UFSAR revision reorganized the Plant Nuclear Safety Committee (PNSC) to be composed of "at least six (6) members." The minimum number of six corresponds to the number of disciplines in the UFSAR.

Summary of Evaluation:

This HBRSEP, Unit No. 2, UFSAR revision reorganizes the PNSC to be composed of "at least six (6) members." The minimum number of six corresponds to the number of disciplines in the UFSAR. An underlying assumption to this change is that the PNSC members present in any meeting include the requisite disciplines to knowledgeably review the subject matter of the meeting. PNSC requirements were relocated from the Technical Specifications to UFSAR Chapter 17 (Quality Assurance Program Description) during the Technical Specification conversion to Improved Standard Technical Specifications. Throughout the history of the licensing basis for PNSC membership, management functions on the PNSC have included the necessary disciplines to ensure that required reviews would be performed by PNSC members with requisite knowledge of the subject matter. Recent changes to 10 CFR 50.54(a)(3) (64FR359034) have defined changes to the Quality Assurance Program Description (QAPD) which are not considered to be a "reduction in commitment" and which may be made without prior NRC approval. Re-organizing the PNSC to consist of "at least six (6) members" versus "seven to nine" members has no impact on the required activities of the PNSC. Since the number of members of the PNSC have no bearing on the authority, organizational freedom, and independence of the PNSC, the actual number of PNSC members in the QAPD may be changed without prior NRC approval. Since this change does not involve a change in the design, construction, installation, operation, testing or operations of SSCs described in the SAR, this change has no effect of the probability of accidents, the consequences of accidents, failure modes, probabilities, and effects of equipment, or safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0189, Rev. 00

Description:

This activity was the permanent plant modification to remove the rest of the N-05 incore flux thimble, which is currently cut and capped inside of its guide tube. The modification also installed thimble assemblies into core locations N-05, F-13, and D-12. The pressure boundary was re-established at the seal table through a previously designed high-pressure fitting. The replacement of thimbles occurred during Refueling Outage 20.

Summary of Evaluation:

The incore instrumentation is designed to yield information on the neutron flux distribution and fuel assembly outlet temperatures at selected core locations. The system provides means for acquiring data and performs no operational plant control. The information provided by the incore thermocouple instrumentation system is available through the system's indication system, which consists of two plasma display panels (one per instrumentation channel) installed in the Control Room. Both instrumentation channels comply with Regulatory Guide 1.97 requirements. Chromel-alumel, bottom-mounted thermocouples are inserted into the neutron flux thimble tubes that enter the reactor vessel through the seal table, and terminate at the end of the thimbles. Thermocouple outputs are recorded in the Control Room. This modification added redundancy

to the movable incore and Inadequate Core Cooling Monitor (ICCM) systems. The configuration was restored to original, less L-05 core location, which is dedicated to the Reactor Vessel Level Indicating System (RVLIS). There are no safety concerns as a result of this evaluation. This change did not affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this modification did not pose an unreviewed safety question.

Evaluation No. 00-0196, Rev. 00

Description:

This change reviewed the use of Safeguards Test Switches to block Containment Spray. The Safeguards Test Switches are part of the Safeguards System and are normally used for periodic testing of the individual Safeguards functions.

Summary of Evaluation:

The Safeguards Test Switches are part of the Safeguards System and are normally used for periodic testing of the individual Safeguards functions. During the testing, the actual initiation signal is blocked by placing the test switch for the specific input relay into the "test" position. A set of contacts in the initiation circuit is opened when the switch is placed in this position, thus preventing the energizing of the actuation relay. The Safeguards Test Switches are installed in the Safeguards Cabinets that are located in the E1/E2 area directly below the Control Room and are easily accessible to the Operator. No special tools or wiring changes are required for placing the switches into the "test" position. There will be sufficient time after the accident initiation to allow the Operator to place the switches in the required position; therefore, the likelihood of the mispositioning of a switch is not increased. The use of the Safeguards Test Switches cannot cause the inadvertent actuation of any ECCS equipment. The use of these switches will be procedurally limited by the plant procedures to situations where automatic Engineered Safety Feature (ESF) actions have already occurred and the accident is being controlled by the Operator in accordance with Emergency Operating Procedure (EOP) guidance. Removing the containment high-high pressure signal input to the Safeguards System following actuation will not cause any Containment Spray, Containment Phase B Isolation, or Steam Line Isolation component to change position or state. The Containment Spray, Containment Phase B Isolation, or Steam Line Isolation component will remain in their emergency configuration until changed by Operator action. The use of the Safeguards Test Switches as described does not exceed the design of the switches and does not create any new requirements for the electrical distribution system. Therefore, this activity is not an unreviewed safety question.

Evaluation No. 00-0215, Rev. 00

Description:

This activity was the permanent facility change to install a new Contaminated Material Storage Building.

Summary of Evaluation:

The new Contaminated Material Storage Building has been installed to store contaminated materials used during outages, which were previously stored in Building 430. This change provides easier access to the material during outages and eliminates the need to move the materials outside of the main radiologically controlled area. The new Contaminated Material Storage Building is not connected to systems or equipment involved in accident mitigation or initiation. This change does not affect the design, construction, installation, maintenance, operation, or testing of SSCs associated with any accident. This change does not create any new or different types of accidents or malfunctions of equipment important to safety. This change to the facility does not affect any margin of safety. Therefore, this activity is not an unreviewed safety question.

Evaluation No. 00-0230, Rev. 00

Description:

This activity was a change to UFSAR Table 15.0.9-1 to correctly characterize the Auxiliary Feedwater (AFW) System. The table incorrectly lists AFW as ESF System. This proposed revision will change this to "Other Equipment" in the table.

Summary of Evaluation:

The change corrects the characterization of the system in the UFSAR and does not change the equipment functioning. The AFW System is already correctly characterized as non-ESF in UFSAR Sections 6.0, 7.3, and 10.4. No changes are made to the way the equipment is operated, tested, or maintained. The AFW System will continue to operate and to perform its accident mitigating function as described in the UFSAR. No physical changes to the facility were made. The system continues to perform its intended function(s) and no changes are made that can create new or different accidents or malfunctions, or increase the probabilities or consequences of accidents or malfunctions, and there will not be a reduction in a margin of safety. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0285, Rev. 00

Description:

This change to the plant was to replace the damaged Service Water System piping downstream of the Component Cooling Water heat exchanger with a higher-grade pipe that is not as susceptible to corrosion.

Summary of Evaluation:

This modification replaced the damaged Service Water System piping downstream of the Component Cooling Water heat exchangers. The piping was changed to Inconel type instead of

carbon steel lined. In addition, valves SW-739 and SW-740 were replaced with a stainless steel material. The conclusion is that the new piping did not result in a reduction in safety and the Service Water System operates the same as before. The new piping is stronger and less susceptible to erosion. The new piping did not change the overall flows or pressures in the Service Water System. The new pipe will have greater structural integrity and will maintain its ability as a support system to mitigate analyzed accidents (i.e., heat removal). Therefore, this change does not pose an unreviewed safety question.

Evaluation No. 00-0301, Rev. 00

Description:

Input discrepancies in the original HBRSEP, Unit No. 2, Large Break Loss-of-Coolant Accident (LBLOCA) and Main Steamline Break (MSLB) containment calculations were found in 1998. A new containment analysis was performed. Revised containment heatsink and containment volume were used as inputs into a re-analysis of the containment response. The results of the re-analysis are documented in WCAP-15304, "Carolina Power & Light Company H. B. Robinson Steam Electric Plant Unit No. 2, LOCA Containment Integrity Analysis," and WCAP-15305, "Carolina Power & Light Company H. B. Robinson Steam Electric Plant Unit No. 2, Steamline Break Containment Integrity Analysis." In addition, a sensitivity analysis was performed by Westinghouse for the containment peak pressure and peak steam/air temperature to changes in the containment initial temperature over a range of 120 °F to 130 °F for LBLOCA and MSLB scenarios, and a sensitivity analysis was performed of the containment peak pressure and peak steam/air temperature to changes in the service water temperature over a range of 95 °F to 100 °F for LBLOCA and MSLB scenarios.

Summary of Evaluation:

As a result of the described containment re-analysis, the peak containment pressure for the LBLOCA event increased from 40.0 to 40.5 psig, and the peak containment pressure for the limiting MSLB event increased from 40.5 to 41.85 psig. The increase in containment pressure for these events was determined to be an unreviewed safety question. A license amendment was submitted and subsequently approved by the NRC in License Amendment No. 187, by letter dated April 18, 2000.

Evaluation No. 00-0330, Rev. 00

Description:

This activity was the permanent plant modification to eliminate turbine runback due to a dropped rod signal.

Summary of Evaluation:

This change is allowed due to the reactor fuel safety analysis, which documents that the HBRSEP, Unit No. 2, will not exceed design or safety margins associated with Technical

Specifications upon receipt of a dropped control rod. The analysis is based on the assumption that a control rod would drop and a turbine runback would not occur to mitigate the event. UFSAR Chapter 15.4 (Dropped Control Rod), as revised by the new reactor fuel safety analysis, supports the removal of this feature. This modification does not create a new accident, or increase the probability of an accident or its consequences previously evaluated in the SAR. USFAR Chapter 15.4 (as revised by the new fuel analysis) documents that a dropped control rod does not require a turbine runback to mitigate its effects. Therefore, the consequences of this accident have not changed. Since there is no change to the Rod Control System, there is no increase of the probability of a dropped control rod. This modification does not affect the function of any equipment important to safety, nor can it increase the chance of a malfunction of equipment important to safety, nor introduce a new malfunction of equipment important to safety. The turbine runback circuitry has never been classified as safety-related equipment. The Siemens Fuel Analysis documents that this equipment is not required to operate to mitigate a dropped rod accident. This modification eliminates initiation of the runback circuitry from two systems, Rod Position Indication and Nuclear Instrumentation. No other function of either of these systems is being changed. Since the runback circuitry is not required to operate for this accident condition, the malfunction is of this equipment is no longer an issue. Therefore, this change does not pose an unreviewed safety question.

Evaluation No. 00-0337, Rev. 00

Description:

This activity was the setpoint change to lower the Containment Spray System actuation setpoint from 20 psig to 10 psig.

Summary of Evaluation:

As a result of changes to the LBLOCA and MSLB containment pressure analyses (see Evaluation No. 00-0301 above), the setpoint for the six channels monitoring containment pressure were modified. The setpoint for six comparators to containment pressure monitoring (PC-950, PC-951A, PC-952, PC-953A, PC-954, and PC-955A) were revised to 10 psig from 20 psig. The revised setpoint lowers the initiation setpoint for Containment Spray, Containment Isolation Phase B, and Steam Line Isolation. This change was associated with an unreviewed safety question and Technical Specifications changes associated with the LBLOCA and MSLB containment pressure analyses described in Evaluation No. 00-0301. A license amendment was submitted and subsequently approved by the NRC in License Amendment No. 187, by letter dated April 18, 2000.

Evaluation No. 00-0342, Rev. 00

Description:

This change installed a new channel cover for the "B" Component Cooling Water (CCW) heat exchanger.

Summary of Evaluation:

The conclusion is that the new cover and pipe penetrations will not result in a reduction in safety. The new cover will not increase the probability of a flood in the CCW pump room. The loss of one anode on the heat exchanger will not degrade the heat exchanger. The new pipe penetrations will not increase the consequences of a fire. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0414, Rev. 00

Description:

This activity was a temporary modification to provide alternate cooling for "B" CCW heat exchanger.

Summary of Evaluation:

This activity was a temporary modification to supply temporary cooling to the "B" CCW heat exchanger during work on the normal supply line. As a support system the Service Water System is relied upon for the mitigation of analyzed accidents. The temporary modification maintained the CCW heat exchanger function during the refueling outage. The probability of an accident previously evaluated in the SAR was not increased. The margin of safety as defined in the basis of Technical Specifications was not reduced during this temporary condition. Therefore, this temporary modification to the facility did not pose an unreviewed safety question.

Evaluation No. 00-0422, Rev. 00

Description:

This activity involved the permanent plant modification to remove the shell side pressure gauges from the Nos. 3, 4, 5, and 6 Feedwater Heaters.

Summary of Evaluation:

The shell side pressure gauges on the feedwater heaters were removed and the lines were capped. This change did not make any changes to the Technical Specifications or Operating License. The feedwater heater drains and vents system are not safety-related. The pressure gauges have no accident mitigating function. The margin of safety as originally defined in the basis of the Technical Specifications will not be reduced by removing these pressure gauges. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0450, Rev. 00

Description:

This activity was an evaluation to allow the use of reactor vessel nozzle plugs during full core off-load conditions to facilitate the inspection of the reactor vessel and steam generators simultaneously during the outage.

Summary of Evaluation:

The temporary installation of the reactor vessel nozzle plugs during defueled conditions cannot induce an accident as described in the UFSAR. No existing procedures or systems, or their functions described in the UFSAR, are affected. Failure of the plugs has been analyzed and is bounded by existing analysis. The margin to safety as defined in the basis for any Technical Specifications are not affected. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0475, Rev. 00

Description:

This activity involved changes to the Fire Protection Program (FP-012) including general clarifying statements and editorial updates.

Summary of Evaluation:

Changes in Revision 7 of FP-012 do not involve a change to the facility, however, some of the changes do represent a change to the Fire Protection Program. There is no specific reference to the content of procedure FP-012 in the UFSAR. This procedure specifies the minimum fire protection systems required and compensatory actions to be taken when the minimum equipment becomes inoperable. Statements in the UFSAR concerning this aspect of the Fire Protection Program remain unchanged. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0482, Rev. 00

Description:

The change to the Spent Fuel Cask head operation was an improved method for obtaining levelness of the cask, the acceptability of plywood material in the Spent Fuel Pool (SFP), and the acceptability of using shims to level the cask.

Summary of Evaluation:

Cask handling operations are not adversely affected by the performance of the described activities. This change established levelness criteria of one degree for fuel and cask head operations, and two degrees for seismic events. The cask head operations will allow the use of plywood to shim, in addition to the use of plywood as a shield for contamination control. Plywood has been analyzed for use in the SFP by CP&L's laboratory at the Harris Environmental and Energy Center (HE&EC). The total concentration of ions that are known to have an effect on stainless steel and other components in the SFP were found to be of sufficiently low quantities to not affect the SFP, fuel, and fuel racks. The organic materials that make up the plywood, cellulose and lignin, are not water-soluble. Additionally, they are not detrimental to the SFP, fuel, and fuel racks. The use of shims ensures the cask is vertical and will meet the calculation assumptions. There are no accidents in the cask SAR that are affected. The normal fuel handling procedures and precautions will be used. Additionally, levelness criteria are established, which ensures that fuel will not be damaged. This provides additional assura nce that the probability of occurrence of a fuel handling accident, as described in UFSAR Section 15.7.4, is not increased. Movement of the cask is not affected such that the Spent Fuel Cask Drop Accident described in UFSAR Section 15.7.5 remains not credible. There are no Technical Specification requirements or margins of safety in any basis of a Technical Specification that are affected by these changes. The Technical Specifications and associated bases discuss the equipment required for fuel movement and spent fuel cask movement. There is no affect on any of this equipment. Therefore, there is no reduction in the margin of safety as defined in the basis of any Technical Specification. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0497, Rev. 00

Description:

This activity was the change to implement the Siemens LBLOCA methodolgy for HBRSEP, Unit No. 2. This change resulted in a change to the Technical Specifications, Section 5.6.5.

Summary of Evaluation:

The change addressed by this evaluation is limited to incorporation of approved Siemens LBLOCA analysis methodology in the HBRSEP, Unit No. 2, Technical Specifications. The methodology is described in Siemens Report EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications." This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question. A license amendment was submitted and subsequently approved by the NRC in License Amendment No. 188, by letter dated August 3, 2000.

Evaluation No. 00-0498, Rev. 00

Description:

This activity was the change to implement the Siemens Steam Line Break Methodology for HBRSEP, Unit No. 2. This resulted in a change to Technical Specifications, Section 5.6.5.

Summary of Evaluation:

The change addressed by this evaluation is limited to incorporation of approved Siemens LBLOCA analysis methodology in the HBRSEP, Unit No. 2, Technical Specifications. The methodology is described in Siemens Report EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs." This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question. A license amendment was submitted and subsequently approved by the NRC in License Amendment No. 188, by letter dated August 3, 2000.

Evaluation No. 00-0539, Rev. 00

Description:

This activity involved the temporary modification to the facility to provide an alternate path for the discharge of service water from the "B" Emergency Diesel Generator (EDG).

Summary of Evaluation:

The temporary modification provided an alternate path for the discharge of service water from the "B" EDG. The EDG was evaluated to allow operation under this temporary condition. The EDG will perform as designed, therefore, there is no increase in the probability of any accident previously evaluated. The temporary modification was installed during no-load conditions; therefore, there was no increase in the consequences of an accident previously evaluated in the UFSAR. The failure of the temporary piping is bounded by previously analyzed conditions. The

EDG remained operable; therefore, no margin of safety as defined in the basis for any Technical Specification will be decreased. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0551, Rev. 00

Description:

This activity was a temporary leak repair of service water line CW-50A-3 to repair a small leak.

Summary of Evaluation:

A temporary repair of the service water line was performed. As a support system, service water is relied upon for the mitigation of analyzed accidents. No new accident or failure mode is created by this repair. The operation of the system with the repair could not credibly increase any radiological releases. The repair was evaluated for any new malfunctions and found not to create any. The repair of degraded system piping until the affected pipe can be replaced does not make any SSC inoperable or reduce the margin of safety of any Technical Specification. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0556, Rev. 00

Description:

This activity was a change to the ISFSI SAR for waste oil storage to appropriately account for the existing plant configuration.

Summary of Evaluation:

UFSAR Section 11.2.2.3 describes the Contaminated Waste Oil Storage System. This section states that the 10,000 gallon capacity is for "accumulation of contaminated waste oil for an estimated 10 years." ISFSI SAR Section 2.1.2.1 has a description of the Contaminated Waste Oil Storage System that specifies a "two year temporary storage facility." UFSAR Section 11.2.2.3 was revised as part of Modification M-840, which installed the Contaminated Waste Oil Storage System. ISFSI SAR Section 2.1.2.1 has a description of the Contaminated Waste Oil Storage System that specifies a "two year temporary storage facility." This description was incorporated in Amendment 1 to the ISFSI SAR prior to the completion of Modification M-840. The safety evaluation for M-840 discusses changing UFSAR Section 11.2.2.3, but does not discuss the change to ISFSI SAR Section 2.1.2.1. There are no systems that either rely on or are impacted by the Contaminated Waste Oil Storage System. The operation of the Contaminated Waste Oil Storage System is not altered by this change, hence it will continue to be operated in accordance

with UFSAR Section 11.2.2.3 and Modification M-840. This change does not impact the design, construction, operation, maintenance or testing of any SSCs in the UFSAR. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0564, Rev. 00

Description:

The Technical Specification Bases, Section 3.3.2, were clarified to consistently discuss the arrangement of two sets of three channels in the APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY discussions, to consistently discuss that one channel per set can be placed in trip status, and to eliminate the distinction between "failed" and out-of-service channels.

Summary of Evaluation:

The Bases are clarified to consistently discuss the arrangement of two sets of three channels in the APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY discussions, to consistently discuss that one channel per set can be placed in trip status, and to eliminate the distinction between "failed" and out-of-service channels. The change to the Bases does not affect accident initiators or impact accident does consequences. The change does not impact the design, installation, configuration, or failure modes and effects of the Containment Spray System. The margin of safety associated with the actuation of the Containment Spray System is not affected. Therefore, the change to the LCO 3.3.2 Bases does not constitute an unreviewed safety question.

Evaluation No. 00-0578, Rev. 00

Description:

This activity was the deletion of Technical Requirements Manual (TRM), Section 3.9, Control Rod Misalignment Monitors.

Summary of Evaluation:

TRM Section 3.9, "Control Rod Misalignment Monitors," describes conditions, required compensatory measure, and completion time in case the ERFIS rod position deviation channel or quadrant power tilt monitor, or a combination of both become inoperable. The entire TRM Section 3.9 has been deleted. The ERFIS rod position deviation channel continuously monitors the potential misalignment of control rods via the Rod Position Indicating (RPI) System. The quadrant power tilt monitor (upper or lower flux deviation comparator) continuously monitors for an upper or lower power tilt via nuclear instrumentation power range detectors. Based on specific setpoints, operators are alerted to a potential rod misalignment condition. The scope of this change includes evaluation of removal of TRM Section 3.9. Technical Specifications LCO

Sections 3.1.4, 3.1.7, and 3.2.1 adequately address the loss of rod position monitors. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0583, Rev. 00

Description:

The change was to address a revision of the UFSAR for clarification of the roles of the Environmental and Radiation Control Manager and personnel reporting to Radiation Control (RC) Superintendent.

Summary of Evaluation:

This UFSAR revision re-institutes the Superintendent - Radiation Control position back into the UFSAR. The position of Manager, Environmental and Radiation Control will remain in the UFSAR. In addition, the statement, "If either the position of Manager, Environmental and Radiation Control or Superintendent - Radiation Control are vacant then the responsibilities related to radiation protection described in these chapters will be performed by the occupied position has been inserted in both Chapter 12 and 13 and will eliminate the need to make a change of this type in the future." If both positions are occupied, the Superintendent - Radiation Control will have the formal support for the plant's ALARA program. This change also states that the Analyst - ALARA may report to an RC Supervisor that provides the management direction for the ALARA Program, if deemed necessary by the Superintendent - Radiation Control. The Analyst - ALARA duties will be unchanged. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0672, Rev. 00

Description:

This activity was the permanent plant modification to improve the instrument bus (IB) power sources to allow replacement of Reactor Protection System (RPS) Relays.

Summary of Evaluation:

This modification upgraded and improved the capacity of the IB power supplies and distribution system. These improvements increase the reliability of the IB power supply system. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs

failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0675, Rev. 00

Description:

This activity was the change to Fire Protection Training Program Procedure, TPP-219, to generally upgrade the procedure, including clarifications and enhancements; the addition of definitions; changes in the retraining frequency for classroom training and practical exercises; changes in the frequency of training with offsite Fire Department(s) from annually to once per calendar year; changes in how the fire protection auxiliary operator candidate is tested from a walkdown and written test to an oral board to be in-line with other watch stations; the addition of control measures for fire drills; the addition of drill acceptance criteria; addition of drill scheduling into the plan of the week; and, enhancements to the Fire Brigade training matrix to increase proficiency.

Summary of Evaluation:

This change to TPP-219 did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0688, Rev. 00

Description:

This activity was a change to the EOPs to improve the steps to perform unit off-site electrical power back-feed by adding the lifting of additional wires. This revision added a lead to be lifted in order to accomplish the back-feed portion of Emergency Operating Procedure EPP-25, "Energizing Supplemental Plant Equipment using the Dedicated Shutdown Diesel Generator."

Summary of Evaluation:

This change removed DC control power from the relays for the exhaust hood high temperature input to Generator Lockout 86P. This action is necessary to assure that an exhaust hood temperature signal does not cause a lockout and thus remove the pathway of current from the switchyard to the buses during back-feed. This protective feature is designed to protect the low-pressure turbine from overheat caused by low power operation or windage, by providing a signal that would result in a turbine trip. During the performance of back-feed operations, the turbine has already been tripped and this feature is not needed. The addition of this step does not adversely affect the times listed in the UFSAR. In addition, since the times used in the UFSAR are not based on analytical values used to show conformance to safety-related issues, no affect on the plant safety is made by the addition of the extra time. Defeating the high temperature input to the generator lockout after the turbine has already been tripped will not affect any safety function

associated with the turbine, since the required function has already been accomplished. Based on the above, no reduction in the margin to safety will occur with this revision, no conflict with NRC commitments or licensing documentation is present, and therefore, no unreviewed safety question is created by the revision.

Evaluation No. 00-0826, Rev. 00

Description:

This change was a revision to the HBRSEP, Unit No. 2, Emergency Plan related to administrative items, address and location changes for off site facilities, and referenced procedure changes.

Summary of Evaluation:

This revision was administrative in nature. This change included the replacement of drawings for legibility and the change of address for the Chesterfield County Emergency Operations Center. Also included are various formatting corrections, the addition of site evacuation assembly options, and the change for a new second tier medical response hospital. The future location of the alternate Operations Support Center was also added. This change to the Emergency Plan did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0867, Rev. 00

Description:

This activity was a permanent modification to the facility to discontinue the use of six chart recorders located in the Control Room.

Summary of Evaluation:

This modification abandoned in place six chart recorders (NR-41, NR-42, NR-43, NR-44, TR-409, YR-760, and NR-761). These recorders were no longer needed. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0970, Rev. 00

Description:

This activity was a change to the UFSAR to add the Generic Implementation Procedure (GIP) for New and Replacement Equipment (NARE).

Summary of Evaluation:

The evaluation shows that no unreviewed safety question is created by the UFSAR change and, as such, the use of the GIP method is acceptable. The use of the GIP will not affect the ability of safety-related equipment or equipment important to safety to perform required safety functions during or after a seismic event. The GIP methodology provides an equivalent or superior level of assurance that equipment will withstand various potential seismic failure modes. This change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0973, Rev. 00

Description:

This activity was a re-analysis of the Appendix R evaluations. This re-analysis included a review of the fire barriers in Fire Area A, Fire Area G, changes in the Control Room HVAC systems, Motor-Driven AFW Pump Room fans, removing valves from the safe-shutdown analysis that are not required for safe-shutdown, and clarification of the requirements for specific fire areas.

Summary of Evaluation:

The changes incorporated into the Appendix R evaluations by this activity did not alter the basic requirements and conclusions of the Appendix R evaluations. No design basis accidents are assumed to occur during any fire scenario or during the actions needed to place the plant in safe-shutdown. The changes that were effected by this re-analysis did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0975, Rev. 01

Description:

This change incorporates a generically approved change to the Bases to LCO 3.4.11, "Pressurizer PORV." This change clarifies in the LCO, APPLICABILITY, and ACTIONS discussions to

those already reflected in the APPLICABLE SAFETY ANALYSES discussion, and the clarifications are consistent with the UFSAR analyses.

Summary of Evaluation:

This change reflects the assumptions and required functions of the Pressurizer PORVs in MODES 1, 2, and 3 as delineated in the UFSAR accident analyses. The accident analyses in the UFSAR conservatively account for PORV operation by opening the valve by inadvertent operation when such operation results in more adverse consequences, or by assuming that the PORVs do not operate in situations where automatic operation would mitigate the event. Therefore, no required function exists in the accident analyses for automatic actuation of the PORVs. The discussion changes in the Bases provide consistency with the UFSAR and do not result in any change in the Operating Procedures or Emergency Operating Procedures, since those procedures already reflect the UFSAR accident analyses. Therefore, the change did not change the probability or consequences of an accident or equipment malfunction. The proposed changes do not result in any design, maintenance, operation, or testing of SSCs, and therefore, do not affect failure modes and effects, nor do they introduce any new accident or equipment malfunction. The safety margin associated with the PORVs as stated in the APPLICABLE SAFETY ANALYSES discussion in the Bases is not changed. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0984, Rev. 01

Description:

This activity is an evaluation of the acceptability of the installed motors for the EDG Fuel Oil (FO) transfer pumps, which differ from the original design. This evaluation was also performed to allow replacement with other similar pump and motor combinations.

Summary of Evaluation:

The installed EDG FO Transfer Pump motors are different than the original design in that there are some minor variations in full load amps and locked rotor amps. An evaluation was performed to document the operability determination and to evaluate changes to the thermal overloads for these pumps. Other similar pump and motor combinations were also evaluated for acceptability. The evaluation concluded that the motors evaluated are acceptable and will perform the required functions without negatively affecting the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0989, Rev. 00

Description:

This activity was a permanent plant modification to install a reactor cavity spiral staircase during refueling outages to allow easier access to the refueling cavity.

Summary of Evaluation:

The staircase is removable so that it can be removed during cavity flood-up. During other plant conditions, the staircase remains installed in cavity. The staircase is seismically mounted to avoid interaction concerns with reactor coolant system and other support systems. This modification did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-0990, Rev. 00

Description:

This activity was the permanent plant change to replace the existing control rod step counters located on the control board within the Control Room with a different model.

Summary of Evaluation:

This modification does not represent a change to the operation of the Rod Control System or the information available to the Control Room operators. The rod control step counters do not initiate nor mitigate any accident scenario and hence, have no effect on any accident or its consequences. This modification did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1029, Rev. 01

Description:

This activity was a permanent plant modification to the air intake for radiation monitor R-38 to improve water intrusion protection. R-38 is the radiation monitor for the Technical Support Center (TSC) and Emergency Operations Facility (EOF). This modification disconnected the existing line from the duct, installed a new blank plate over the duct opening, installed a new section of pipe that was turned down to prevent water intrusion, and covered the pipe with a bug screen.

Summary of Evaluation:

The basic operation of the R-38 radiation monitor was not changed. The monitor will still appropriately provide automatic actuation of the TSC and EOF HVAC system protective mode of operation. This modification did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1075, Rev. 00

Description:

This activity was the establishment of a hot weather procedure (PLP-118) for HBRSEP, Unit No. 2. The procedure contains instructions for a temporary connection to provide additional cooling to the Containment.

Summary of Evaluation:

The additional cooling described in the hot weather procedure (PLP-118) is achieved by using supplemental cooling water from the deep-well pump system. The water quality of the deep-well water is equivalent to (or better than) the water quality of the Service Water System. If the supply of the supplemental cooling water fails while supplemental cooling is being utilized, normal service water will remain adequate to provide the design water supply. PLP-118 also allows the use of a chiller system. The chiller unit would be maintained an appropriate distance from the control room, due to the possibility of a refrigerant leak. This procedure does not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1085, Rev. 00

Description:

This activity was the revision of UFSAR Figures 7.2.1-18, 8.3.1-1, 8.3.1-2 and 8.3.1-3. The revisions correct existing drawing errors and add typical one-line electrical detail that was not provided during configuration control modifications.

Summary of Evaluation:

There were no physical changes to the plant included with this activity. The UFSAR drawing changes were made to properly account for the actual condition of the associated equipment. These UFSAR drawing changes did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the

mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1143, Rev. 00

Description:

This activity was an engineering evaluation of plant operation with a potential loose part on Steam Generator (SG) "B" secondary side.

Summary of Evaluation:

On August 19, 2000, HBRSEP, Unit No. 2, began experiencing events above setpoint on Loose Parts Monitoring System (LPMS) Channel 757, SG "B" secondary accelerometer. Tapes were made of the suspect signal and provided to Westinghouse for analysis. The results of Westinghouse's analysis indicated the potential of a loose part on the secondary side in SG "B." Since performance of the LPMS acoustical analysis, the setpoint was adjusted to the Westinghouse recommended proper value and the LPMS signal has reduced below alarm setpoint. This activity was the engineering evaluation of plant operation with this potential loose part. The engineering evaluation concluded that the loose part would not cause significant tube damage during operation and that reactor coolant system pressure boundary integrity was not in jeopardy. Operation with a potential loose part in the "B" SG did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1155, Rev. 00

Description:

This activity was the change to the HBRSEP, Unit No. 2, organization described in UFSAR Chapter 13.

Summary of Evaluation:

This change revised the HBRSEP, Unit No. 2, site organization descriptions to reflect those reporting to the site Vice President as the Director – Site Operations, Manager – Site Support Services, Manager – Regulatory Affairs, and Manager – Nuclear Assessment; to the Director – Site Operations, the Plant General Manager, the Manager – Training, and the Manager – Engineering. In addition, minor title changes are made inside the site organization. The change does not affect the functional responsibilities of the plant staff sufficiently to cause operation of the facility in a manner different than that described in the UFSAR. This change does not impact the design, construction, operation, maintenance, or testing of SSCs assumed to fail when initiating an accident or assumed to mitigate an accident. This change does not affect SSC

design or plant analyses assumed in the Bases to any Technical Specification. Therefore, this change is not an unreviewed safety question.

Evaluation No. 00-1190, Rev. 00

Description:

This change was a permanent plant modification to install an Upgraded Fire Alarm Computer System with a new fire detection panel in the Condensate Polisher Building. The modification installed a new replacement fire detection panel in the Condensate Polisher Building that will monitor detection devices within the Condensate Polisher Building itself, and will monitor alarms from deluge systems located in the transformer yard, Hydrogen Seal Oil System, and the Turbine Lube Oil System.

Summary of Evaluation:

This modification to the Fire Alarm Computer System will not adversely affect the ability of the system to appropriately detect and alarm for fire safety of the affected facilities. The upgraded computer system has been designed to meet the intent of the applicable fire protection standard (NFPA-72, 1993 edition). The replacement detection panel has similar functionality to the one that was replaced and meets the intent of the applicable fire protection standard (NFPA-72D, 1986 edition). This modification did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1211, Rev. 00

Description:

This activity was an engineering evaluation to determine the post-accident differential pressure conditions for valves SI-861A and SI-891B (the Train A and Train B Containment Spray Pump discharge isolation valves) operation that are less than the current test pressure.

Summary of Evaluation:

The engineering evaluation determined the post-accident differential pressure conditions for SI-861A and SI-891B operation that are less than the current test pressure. Document changes were identified to allow testing of the valves at the reduced pressure. Testing at the reduced pressure alleviated an over-torque concern for the motor operator. The testing will continue to properly verify operability of these valves. This engineering evaluation and associated test changes did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1226, Rev. 00

Description:

This activity was the plant modification to various ECCS and Chemical and Volume Control (CVC) system valves that had packing leak-off lines piped directly to the waste disposal system. This modification removed some of the piping to allow measurement of leakage.

Summary of Evaluation:

Various ECCS valves have packing leak-off piping which is piped directly into the waste disposal system. The modification was performed to cut the leak-off piping such that packing leakage may be observed and measured. The leak-off fluid drains to the waste disposal system via the Auxiliary Building floor drains. This modification did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1270, Rev. 00

Description:

This activity was the revision to UFSAR Figures 9.2.1-1 and 11.3.2-1, based on valves SW-839, SW-845, SW-851, and SW-857 being previously modified to change valve type from diaphragm-type to ball-type obturator. Additionally, extraneous information was removed from Figure 11.3.2-1.

Summary of Evaluation:

There were no physical changes to the plant included with this activity. The UFSAR drawing changes were made to properly account for the actual condition of the associated equipment. These UFSAR drawing changes did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1304, Rev. 00

Description:

This activity was a change to the procedure (OMM-036) that describes operator real time training (RTT) to better define the operator real time training program with respect to administrative requirements. This change altered a commitment associated with a corrective action for an NRC Notice of Violation (NOV) by replacement of the corrective action with an equivalent or better action.

Summary of Evaluation:

The change that was made to OMM-036, as corrective action for the referenced NOV, was to require RTT on all changes that were not classified as "administrative." An equivalent or better corrective action has been determined to be the requirement to perform training before implementation for those items that are performed from memory. There were no physical changes to the plant included with this activity. This procedure change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1318, Rev. 00

Description:

This activity was a permanent plant modification to replace a portion of the steam generator blowdown system piping from carbon to stainless steel, and removal of temperature indicators TI-3100, TI-3102, and TI-3106 at the heat exchangers.

Summary of Evaluation:

This modification installed improved piping material and removed indicators that were determined to be unnecessary. The piping and instrumentation affected by this modification are in the non-safety-related portion of the steam generator blowdown system. This modification did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1322, Rev. 00

Description:

This activity was a change to plant procedure EST-001, "Source Range Statistical Reliability Test," to allow reduced testing prior to fuel handling and lowering the reactor vessel water level with fuel in the reactor vessel.

Summary of Evaluation:

The change to delete the requirement to perform EST-001 "prior to fuel handling in the core," and limiting the requirement to perform EST-001 "prior to lowering of the vessel water level with fuel in the reactor vessel" to the initial drain-down during an outage, will reduce the number of times this test is required to be performed while maintaining assurance of proper source range channel operation. The test will continue to be performed at least twice during a refueling

outage. Technical Specifications-required surveillances (described in Techncical Specifications 3.3.1 and 3.9.2) also continue to provide assurance of source range channel operability. This procedure change did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1356, Rev. 00

Description:

This activity was a permanent plant modification to eliminate the automatic monitoring of the particulate, iodine, and noble gas channels of R-23 for the Radwaste Building. Weekly manual grab samples will be utilized for each channel in order to maintain the requirements of effluent accountability.

Summary of Evaluation:

Radiation monitor R-23 for the Radwaste Building has no automatic function. The R-23 monitor location has been modified to consist of a continuous particulate- and iodine-sampling device, and grab sample connections for obtaining grab gas and tritium samples. Any measurable radioactive material will be accounted for in the effluent accountability program. This modification did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1358, Rev. 00

Description:

This activity was the reconstitution of fuel assembly S15-H and related activities.

Summary of Evaluation:

The fuel assembly S15-H was handled in the Spent Fuel Pool using existing approved plant procedures. The assembly was placed in the new fuel elevator in the Spent Fuel Pool for reconstitution. The fuel rods were removed from assembly S15-H using standard rod removal techniques. The removed fuel rods were loaded into a special shipping canister that will later be loaded into a spent fuel shipping cask for the offsite shipment. Required fuel assembly movements were performed using currently approved site fuel handling procedures. Only one fuel assembly (S15-H) was handled for this activity. All the necessary requirements for fuel handling in the Fuel Handling Building (FHB) were followed, i.e., FHB ventilation, Spent Fuel Pool boron concentration, fuel handling tool checkouts, etc. This fuel assembly reconstitution activity did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the

SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1408, Rev. 00

Description:

This activity was a revision to the Technical Requirements Manual (TRM), the reportability procedure (AP-030), and Radiological Effluent Technical Specifications procedure (EMP-024), to eliminate certain special reports and extend completion times.

Summary of Evaluation:

The TRM, AP-030 (NRC Reporting Requirements), and EMP-024 (RETS Surveillance) procedures were changed to eliminate the 14-day Special Report for inoperable Post Accident Monitoring Instrumentation and to extend the required Completion Time for one inoperable RVLIS channel from seven days to 30 days. In lieu of the 14-day Special Report, preplanned alternative methods must be in place within 14 days and the cause of the inoperability and the plans for restoring operability identified within 30 days, if the inoperable channel is not restored within seven days. The alternative requirements provide adequate assurance that system operability will be restored within appropriate time periods. These changes did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 00-1442, Rev. 01

Description:

This activity was the permanent plant modification to replace a condensate system flow-indicating switch with a non-indicating flow switch. This modification also revised the setpoints at which the valve operates to preclude valve cycling while still maintaining the minimum flow.

Summary of Evaluation:

This modification was performed to improve the control of condensate system flow during periods of low condensate system flow. This modification does not affect equipment important to safety. The modification increases the reliability of the condensate recirculation components and ensures the condensate pumps are not damaged during plant operation. Therefore, this modification reduces the likelihood for a plant transient due to condensate system failure during low flow conditions. This modification did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 01-0049, Rev. 00

Description:

This activity was the revision of UFSAR Figure 10.1.0-5 to incorporate previous changes made to Containment penetrations P-57 and P-58 and related valve alignments.

Summary of Evaluation:

There were no physical changes to the plant included with this activity. The UFSAR drawing changes were made to properly account for the actual condition of the associated equipment. These UFSAR drawing changes did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this change does not involve an unreviewed safety question.

Evaluation No. 01-0063, Rev. 00

Description:

This activity was the establishment of a special procedure, SP-1485, "S15-H Cask Loading," for the control of the shipping cask loading of fuel assembly S15-H high burnup fuel rods.

Summary of Evaluation:

The fuel rod canister movements were performed using a special fuel rod canister-handling tool. Only the fuel canister was handled during this activity, no other fuel movement was necessary. The requirements for fuel handling and cask movement in the Fuel Handling Building were in effect during this activity. Use of the special fuel rod canister-handling tool made the possibility of a fuel handling accident very remote. Had a fuel handling accident occurred, it was determined to be bounded by the accident as analyzed in the HBRSEP, Unit No. 2, UFSAR, Section 15.7.4. This activity did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this activity does not involve an unreviewed safety question.

Evaluation No. 01-0106, Rev. 00

Description:

This activity involved changes to procedure NUA-NGGC-1511, "Assessment and Independent Safety Review Personnel Training and Development, Qualification, and Certification."

Summary of Evaluation:

This procedure implements administrative controls contained in the UFSAR. The revision included administrative changes and the allowance for supervisory discretion to determine continuing training requirements for Independent Safety Review personnel. The evaluation of these changes concluded that the applicable UFSAR administrative control requirements were still satisfied. This activity did not affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, this activity does not involve an unreviewed safety question.

Evaluation No. 01-0148, Rev. 00

Description:

This activity was a revision to the Offsite Dose Calculation Manual (ODCM).

Summary of Evaluation:

This revision to the ODCM replaced the use of two computer programs (LADTAP and GASPAR) with the existing Effluent Management System Software. This methodology change continued to satisfy all applicable regulatory requirements for offsite dose determination. The Technical Specification limit for liquid effluent (TS 5.5.4.b) allows 10 times the concentration Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2401. This change to the ODCM administratively limited discharges to one times the applicable concentration limit. Radiation monitors R-22, E&RC Building Effluent Monitor, and R-23, Radwaste Building Monitor, were added to the table requiring compensatory measures in the event that the associated monitor is out of service and to the table that defines surveillances for these monitors. This change is consistent with the requirements of TS 5.5.4. The requirement for performing a land-use census was changed from once per year to once every two years. The areas surrounding HBRSEP, Unit No. 2, are rural and there are few significant changes noted on a year-to-year basis. Technicians perform weekly sampling, which provides the opportunity to observe and report changes in land use. The requirement to perform monthly summation of cumulative release was deleted. Cumulative dose projections are performed on each release permit, therefore, the monthly summation is not necessary. These changes still satisfy all applicable regulatory requirements. These ODCM changes did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, these ODCM changes do not involve an unreviewed safety question.

Evaluation No. 01-0187, Rev. 00

Description:

This activity was a revision to UFSAR Section 17.3A, Approval Authority.

Summary of Evaluation:

UFSAR Section 17.3A was changed to allow the Plant General Manager – Robinson Nuclear Plant (PGM) to perform approval, designation, and maintenance of the qualified reviewer list. The Vice President – Robinson Nuclear Plant (VP) will continue to retain this authority. The specific requirements of Section 17.3A are being changed, as follows: Subsection 1.1.5 states that procedures, tests, and experiments and permanent changes thereto (other than editorial or typographical) which have been determined neither to involve an unreviewed safety question, as defined in 10 CFR 50.59(a)(2), or a change to the Technical Specifications that affects nuclear safety, shall be approved prior to implementation by the manager of the functional area affected by the procedures, tests, and experiments and permanent changes thereto as previously designated by the Vice President – Robinson Nuclear Project. This requirement was changed to allow approval by the Plant General Manager (PGM) or Vice President, in addition to the manager of the functional area as previously designated by the Plant General Manager – Robinson Nuclear Plant or the Vice President – Robinson Nuclear Plant. The PGM was added to the requirement for "...functional area as previously designated by the Vice President – Robinson Nuclear Project." The PGM is also being added to the requirement for approving and maintaining a list of qualified persons. Subsection 1.1.5 is being further changed to correct an error that was introduced during the creation of this appendix. These changes still satisfy all applicable regulatory requirements. These UFSAR changes did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, these UFSAR changes do not involve an unreviewed safety question.

Evaluation No. 01-0195, Rev. 00

Description:

This activity was a revision to PLP-007, "Robinson Emergency Plan." This revision deleted the Forward Emergency Operations Center (FEOC) as a facility required for activation or use by the South Carolina Emergency Preparedness Division (SCEPD), the Chemistry Technician job category was added to potential first responder personnel, the fire detection description was improved, and the requirement to send a copy of the Emergency Plan to INPO was deleted.

Summary of Evaluation:

These changes to the Robinson Emergency Plan did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, these UFSAR changes do not involve an unreviewed safety question.

Evaluation No. 01-0231, Rev. 00

Description:

This activity was a revision to TPP-219, Fire Protection Training Program.

Summary of Evaluation:

This procedure controls the training aspects of the Fire Protection Program. This revision to the Fire Protection Training Procedure, TPP-219, was for general upgrade including clarifications and enhancements; the addition of Fire Areas A and B to the areas where timed objectives for fire drills apply; the incorporation of required table top drills to supplement (but not replace) graded fire drill evolutions; and, to revise Section 5.1 of this procedure that defines the types of reviews required for relocated requirements. The "reviews required" definition in Section 5.1 is consistent with guidance recently provided in 10 CFR 50.59(c)(4) and NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation." These changes to TPP-219 did not negatively affect the design, construction, installation, maintenance, operation, or testing of SSCs assumed in the initiation of an accident or in the mitigation of an accident, or the SSCs failure modes and effects, or the SSC design and safety margins. Therefore, these changes do not involve an unreviewed safety question.

Evaluation No. 01-0255, Rev. 01

Description:

Changes resulting from the Cycle 21 reload and plant changes as part of the Cycle 21 Plant Parameters Document review were evaluated against the analysis of record.

Summary of Evaluation:

This Cycle 21 reload safety analysis evaluates the impact of the Cycle 21 reload on the existing analysis of record as documented in the UFSAR and the Technical Specification and Bases. Changes resulting from the reload and plant changes as part of the Cycle 21 Plant Parameters Document review were evaluated against the analyses of record. As is demonstrated in this evaluation and the associated UFSAR mark-ups, the changes associated with the Cycle 21 reload are minor. Their impact has been evaluated and demonstrated to be acceptable and within the current licensing basis. The analyses have shown that the licensing basis, as defined by the Technical Specifications and Bases, and UFSAR, continued to be supported for Cycle 21.

The changes associated with this reload are limited to minor issues, which result in minor incremental impacts on the operation of the plant. There is no fundamental change to any equipment or operation of the plant that could introduce a new or different type of accident initiator or place the plant in a substantially different configuration from which an accident might be initiated. There are no plant changes that introduce equipment changes that could increase the probability of equipment failure. Since the new fuel design is functionally identical to that used in Cycle 20, the probability of failure due to fuel design itself is not increased. The operating

conditions of the fuel are also unchanged and each assembly has been demonstrated to be acceptable for the maximum burn-up projected throughout Cycle 21 operation. Neither the reload nor any plant changes addressed within this evaluation will increase the severity of the operating environment for any equipment, or impose additional loads or operating demands on equipment that would increase the probability of failure. As such, an unreviewed safety question does not exist.

Evaluation No. 01-0277, Rev. 00

Description:

This activity was a review of EMF-2286, "H. B. Robinson Unit 2 Extended Transfer to Cold Leg Recirculation Following a LBLOCA."

Summary of Evaluation:

A new analysis has been performed by Siemens Power Corporation (SPC) and is documented in EMF-2286(P) and EMF-2286(NP). This activity was an Owner's Review of EMF-2286(P), Revision 0, and EMF-2286(NP), Revision 0, "H. B. Robinson Unit 2 Extended Transfer to Cold Leg Recirculation Following a LBLOCA." The Owner's Review verified that the report provides the appropriate content and detail to; support revision of the LBLOCA switchover analysis of record as described in UFSAR Chapter 15.6.5; support 10 CFR 50.46 reporting requirements; support closeout of the NRC concern regarding the magnitude of the second fuel heat-up during the switchover to recirculation; and, demonstrate compliance with 10 CFR 50.46 and 10 CFR 50, Appendix K, documentation requirements for the methodology used. The switchover evolution presented in EMF-2286 is consistent with the descriptions in UFSAR Chapter 6 for the alignment and performance of the ECCS during recirculation. The analysis and Owner's review conclude that the applicable regulatory requirements are satisfied. Therefore, this change does not pose an unreviewed safety question.

Evaluation No. 01-0869, Rev. 00

Description:

This activity was a revision to the ODCM. Specifically, Table 2.6-1, Item 2.b, was revised to provide clarity on when to implement SG blowdown measurement device compensatory actions.

Summary of Evaluation:

This change to the ODCM was to improve the clarity of the guidance provided. The basic requirements were not changed. Therefore, this ODCM change did not pose an unreviewed safety question.

The following evaluations were performed in accordance with 10 CFR 50.59 after the 2001 rule change:

Evaluation No. 00-1213, Rev. 01

Description:

This activity changed the EDG overhaul frequency from 18 to 24 months.

Summary of Evaluation:

This change reduced the amount of time that the EDG will be out of service, thus increasing the availability of the EDG. Since the overhaul frequency will still be commensurate with the manufacturer's recommendations, the probability of the EDG failing prematurely is not increased. This activity does not involve a change to an SSC that adversely affects a UFSAR-described design function because the design function of the EDG is not changed by this activity. The EDG will still mitigate the effects of a loss of offsite power as described in the SAR. This activity does involve a change to a procedure that adversely affects how any UFSAR-described SSC design functions are performed or controlled because design functions are indirectly controlled by maintenance intervals and this change extends that interval. The evaluation methodology used in the design basis and safety analysis is not affected by this change since the change only extends the EDG maintenance interval and does not affect the design basis or safety analysis in any way. This activity is not a test or experiment. Following this change, the EDG will remain within the reference bounds of its design and will remain consistent with the analyses and descriptions within the UFSAR.

Evaluation No. 01-0412, Rev. 00

Description:

This activity was the establishment of a new procedure for removing the "A" or "B" 125V DC busses from service.

Summary of Evaluation:

This is a new procedure to provide the guidance to remove the safety-related DC busses from service for maintenance. This procedure required the reactor to be in a defueled condition. The Spent Fuel Pool level and temperature alarms are not functional in the Control Room for a period of time while the "A" DC bus is deenergized. The alarms are not required for accident mitigation. Therefore, a license amendment was not required for this change.