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see reports

SUBJECT: "Oconee Nuclear Station,Units 1,2 & 3,1997 Annual 10CFR50.59 Rept," containing descriptions of changes,tests & experiments,including summaries of SEs.W/980630 ltr.

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June 30, 1998

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Oconee Nuclear Site
Docket Nos. 50-269, 50-270, 50-287
10 CFR 50.59 Annual Report

Attached are descriptions of Oconee facility changes, tests, and experiments which were completed subject to the provisions of 10 CFR 50.59 between January 1, 1997, and December 31, 1997. This report is submitted pursuant to the requirement of 10 CFR 50.59 (b)(2).

If there are any questions, please contact Edwin Price at (864) 885-4388.

Very truly yours,

W. R. McCollum

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Attachment
Oconee Facility Changes - 1997

I. NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Main Steam, FDW

NSM ON-32873 added safety related detection circuitry that created signals to trip the main feedwater (FDW) pumps, close FDW valves to stop FDW flow, and inhibit autostart or initiate autostop of the turbine driven emergency feedwater (EFW) pump when instrumentation indicates a main steam line break (MSLB). The circuitry is designed with two trains of 2 out of 3 logic (one for each main steam header) for FDW pump trip, turbine driven EFW pump autostart inhibit/autostop initiate, and FDW valve isolation actuation. One new safety related pressure transmitter per header was added. The control circuitry has a timer circuit in order to prevent a spurious signal from unnecessarily isolating FDW. The pumps need to be tripped or autostart inhibited/autostop initiated and the valves isolated to help prevent the containment pressure from exceeding the design pressure. Solenoid valves were added to the control air signal for both flow paths of FDW main and startup control valves. The 80% interlock to open the FDW main block valves was increased to a 90% interlock in order to minimize the chance of transients. Also, the EFW interlocks that close the startup block valve when EFW is initiated was deleted. New statalarm and computer alarms were added to indicate a MSLB has occurred. Note: This change has been installed on all three units. UFSAR updates not being made in a timely manner was addressed in PIP 97-1801.

SAFETY EVALUATION SUMMARY

The new circuitry is designed so that a single failure will not cause a loss of FDW to the steam generator unless a MSLB (and possibly a FDW line break) is detected. Operators are currently instructed to isolate FDW on indication of MSLB. The new circuitry will automatically stop FDW to eliminate the need for this Operator action. Thus the probability of the stopping (loss) of FDW is not increased. The NRC has also stated that the stopping of FDW to mitigate a MSLB is an acceptable response to address the concerns of I&E Bulletin 80-04. The new FDW and EFW circuitry will assist in reducing the potential for the pressurization of containment. The new circuitry, as it relates to the EFW System, is designed so that it will not prevent the turbine driven EFW pump from starting or running as needed. This new circuitry creates no new credible single failures that could prevent the turbine driven EFW pump from autostarting (except for the MSLB and FDW line break). The motor driven EFW pumps and EFW flow control valves are not adversely affected by this modification and will provide EFW flow for scenarios other than Station Blackout and ATWS. Both FDW and EFW will still provide their design functions of supplying feedwater to the steam generators. The modification does not adversely affect containment integrity and radiological release pathways. The power source non-safety related/safety related connections are isolated with safety related devices. There are no seismic concerns. The effects and mitigation of EFW pump runout is within the acceptance criteria of the NRC. This modification does not involve any USQs or safety concerns. No Technical Specification changes are required. UFSAR Sections 7.4.3.1, 10.4.6, 10.4.7, and Figure 8-5 were revised accordingly. Note: A letter has been sent to the NRC describing the modification and exceptions to the IE Bulletin 80-04 acceptance criteria.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Main Steam

NSMs 1, 2, and 3 2903 changed out the 18" manual turbine bypass isolation valves (1, 2 ,3MS-20, 23, 29 and 32) on Units 1, 2, and 3, respectively, with 12" isolation valves of a higher pressure class rating. The turbine bypass control valves (1, 2, 3MS-19, 22, 28, 31) were also replaced with a more reliable, easier to control, reduced leakage design. The turbine bypass system is designed to route steam to the condenser after a turbine trip (or shutdown) and after large load reduction transients The old turbine bypass valves (TBVs) tended to leak, and required extensive maintenance. The old manual isolations did not meet code requirements. These modifications resolve both issues.

SAFETY EVALUATION SUMMARY

The new smaller diameter TBVs are capable of passing the required bypass flow of 25% rated steam flow. They provide a tortuous path for steam flow that allows a size reduction while maintaining flow volume. The old valves and piping were Class G non-QA components, as are the replacements. The new isolation valves were uprated to meet higher upstream temperature and pressure design criteria. All replacement valves perform the required design functions.

The TBVs are required to change position to mitigate a S/G tube rupture event. The new valves meet this requirement, but have a longer stroke time leading to slightly longer main steam relief valve open time. Dose analyses show releases remain well within the 10CFR Part 100 guidelines. The steam mass release assumed in UFSAR 15.8 is the upper boundary limit. The increased flow has negligible effect on vessel fracture mechanics. The new valves provide better control capabilities to the operators.

No USQs were created by or involved with this modification. No Technical Specification changes are required. The TBVs are mentioned in the UFSAR, but no detailed description is given, and this change is not significant enough to warrant changing the existing description. UFSAR section 15.9 was revised to reflect a more accurate value for the amount of primary coolant released through the Main Steam Relief Valves (MSRVs) of 13,000 lbm.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Main Steam (MS) and Heater Drain (HD)

NSM ON-32941 made numerous changes to the Main Steam and Heater Drain Systems on Unit 3. Major changes are as follows: Replace the internals/trim in valves 3MS-112 and 3MS-173. Replace the existing control systems for these valves with new digital valve controllers. Install check valves in the piping associated with the second stage reheater (SSRH) drain tanks "3A" and 3B", upstream of valves 3HD-92 and 3HD-95. Also install check valves in the piping associated with the First Stage Reheater (FSRH) drain tanks "3A" and "3B", upstream of valves 3HD-66 and 3HD-81. Additionally, a test connection with double isolation valves was added upstream and downstream of the new check valves. Replace pressure switches 3HDPS0377 and 3HDPS0378. Replace the MSRH (Moisture Separater Reheater), FSRH, SSRH drain tank normal level transmitters associated with the drain tank feedforward valves. Relocate and replace the MSRH, FSRH, SSRH drain tank HI level transmitters. Install a drain line from the heater drain piping loop seal and route to the "3A2" Feedwater (FDW) heater. Replace some existing heater drain sections of piping and fittings. A small connection, with an isolation valve, was added to the piping to allow for a temporary connection of a dynamic pressure transmitter. A low point drain was added to the SSRH header and routed to the west condenser dump header. The drain line contains an air operated valve and a manual valve. Replace the control systems for the MSRH feedforward and recirculation valves with a digital valve controller. Install a drain line in the SSRH System piping low point and route to the east condenser dump header. Each leg of the drain line was provided with a remote isolation valve and a manual isolation valve, with an associated control system that allows automatic operation. Hangers were added/modified as required.

SAFETY EVALUATION SUMMARY

The purpose of the Main Steam System change is to modify the Main Steam admission control system for the MSRH SSRH's to provide a more accurate automatic control of the startup/operating relationship of SSRH Tube Supply Pressure versus Turbine Reheat Pressure and also control within all SSRH Temperature and Heatup Rate Limits. The purpose of the Heater Drain System change is to modify the SSRH Drain Tank to '1A' FDW Heaters pipe and FSRH Drain Tank to "1B" FDW heaters, by the addition of check valves, to prevent the possibility of heater drain back-flow and to add a drain line and an automatic prewarming line on the SSRH piping. These actions prevent exceeding startup/operating limits on the SSRH's, thus prolonging their life span, minimize water-hammers (and the resultant damage to SSRH drain pipe, tanks, supports, and insulation), minimize water hammers in the MSRH drain piping, and alleviate operator burden of manual operation of the SSRH startup function. These activities do not affect the design basis of the units. The modification does not change the operation of the FSRHs, SSRHs, MSRHs, drain system, and affected piping as currently described in the SAR. This modification does not create any conditions or events which lead to accidents previously evaluated in the SAR. The affected piping and components are non-radioactive and do not mitigate any accidents. There is no adverse affect on containment integrity and no new release paths are created. There are no effects on the Appendix R fire analysis. The piping, valves, and other added/replaced components are non-QA, except for the QA-4 mounting of the components on the control boards. The modification does not add any new safety/non-safety interfaces. Environmental Qualification of the new components is not required. A control board seismic review revealed no adverse effects. The control circuitry is not safety related and is not required to be designed to the single failure criterion. This modification involves no USQs or safety concerns. No technical specification changes are required. UFSAR figure 10-4 was revised in the 1996 update to show the new check valvess for all three units.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Operator Aid Computer

NSMs ON-1, 2, and 3 2962 replaced the Honeywell Operator Aid Computer (OAC) on Units 1, 2, and 3, respectively. The purpose of the OAC is to supplement control indications and provide clear concise plant information that may otherwise not be directly available. The only control function formerly associated with the OAC was to inhibit operation of selected pressurizer bank heaters if the pressurizer level is less than a setpoint value. This function is performed by other equipment and was deleted from the OAC by the modification. The original OACs were experiencing increased failures and replacement parts were difficult to obtain. The new computer is an easily expandable, open architecture, data acquisition system that can utilize commercially available components.

SAFETY EVALUATION SUMMARY

The new OACs and indicators perform the same function as the existing equipment, but through the use of more advanced, modernized, and readily available equipment. The functionalities of the Low Temperature Overpressure Protection (LTOP), Inadequate Core Cooling Monitor (ICCM), RCS subcooling margin monitor, Safety Parameter Display System (SPDS), Emergency Response Data System (ERDS), Radiation Information Alarms (RIAs), and incore instrumentation are maintained. The new OAC components are seismically qualified where required.

The OAC is not safety related, not QA-1, and not required to be single failure proof. The OAC does not perform a function required to mitigate an accident, does not trip the reactor or actuate a safety-related system, and is not significant to any Emergency Operating Systems. There are no design basis requirements for the OAC. The OAC is not required to be operable for the plant to operate. Per existing operating procedures and with appropriate compensatory actions, a unit may be operated indefinitely with the OAC out of service. Technical Specifications (TSs) do not directly address the OAC as being required, but rather mention it peripherally in Sections 3.5.4, 4.7.2.1, Table 4.1-1(34) and Table 6.1-1. TS 3.5.1 and Table 3.5.1-1 address the nuclear instrumentation (NI) system requirements, which continue to be met.

These modifications involve no safety concerns or USQs, and no Technical Specification changes are required. Numerous UFSAR sections, tables, and figures (including Sections 2.3.3.2, 3.1.32, 3.1.43, 3.4.1.1.2, 4.3.3, 4.3.5, 4.3.7.1.2, 4.3.7.2.2, 4.3.7.3.2, 5.4.6.2, 6.4.2.1, 7.3, 7.4, 7.5, 7.6, 7.7, 7.8, 8.3.2, 9.3, 9.5, 10.4.7.2, 13.1.2, 13.5, 14.5, 15.0, and Figure 7-15) and the bases of Technical Specification 4.7.2 were reviewed and revised as necessary to address installation of these NSMs.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Fuel Handling

NSM ON-12970 replaced the four existing 1 ton steam generator cavity jib hoists with new 3 ton capacity hoists and fortified the existing support structures to accommodate.

SAFETY EVALUATION SUMMARY

The criteria for control of heavy loads is met. There are no seismic or Appendix R concerns. The old hoists were simply replaced with new heavy duty types. The new hoist provide the same functions. This modification does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. This modification involves no USQs or safety concerns. No Technical Specification or UFSAR changes are required.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: High Pressure Injection

NSM ON-12975 removed stop-check valves 1HP-126, 1HP-127, 1HP-152, and 1HP-153 on Unit 1, and replaced each of these valves with an angle check valve and a globe valve. The angle check valve performs the desired checking function while the globe valve performs the isolation capability. The new globe valves were given the existing check valves' numbers. The existing check valves are 2 1/2 inches. The new valves are also 2 1/2 inches. Some class B piping was replaced with class A so the system is class A from the inlet of the first valve (angle check valve) to the Reactor Coolant System (RCS). Also, the Class A pipe from the globe valves to the High Pressure Injection (HPI) safe ends and a portion of the warming lines near valves 1HP-126 and 1HP-127 was replaced. The angle check valve includes a drain connection on the downstream side of the seating surface to allow future leak testing. The drain connection has double isolation valves.

SAFETY EVALUATION SUMMARY

The HPI System continues to perform its normal and emergency functions. The valves are designed to appropriate system requirements and conditions. Missile protection is not degraded. There are no additional seismic interaction concerns of non-safety related components with this modification's safety related components. The information concerning the safe ends, the HPI piping, and the connection between the piping and the safe ends still meet the same design requirements specified in the SAR after the modification is installed. The mitigation of a 10 CFR 50 Appendix R fire is not adversely affected. Increased hydrogen generation due to additional amounts of zinc or aluminum is not a concern. The test connection has the two valves closed to provide a double QA Condition 1, Class A pressure boundary for the RCS. Pipe stress analysis and support designs are complete. The RCS pressure boundary is not degraded as evaluated in the SAR. This modification involves no USQs or safety concerns. No technical specification changes are required. UFSAR Section 5.0, and Figures 6-1 and 9-17 were reviewed/ revised as necessary to show the new check valve identification numbers for Unit 1.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: High Pressure Injection

NSMs ON-1,2, 3976 changed out the letdown control valves (HP-5). The existing air operated 2.5" Rockwell valves were replaced by a 2" anchor darling valve with a spring actuated closure function. The new valves retain the existing numbers.

SAFETY EVALUATION SUMMARY

Letdown valves 1,2,3HP-5 are required to close on an ES Channel 2 signal to isolate penetration number 6. This valve changeout alleviates the problem of keeping the valve shut in the event of continuous air leakage. The HPI System continues to perform its normal and emergency functions. This change does not create any conditions or events which lead to accidents previously evaluated in the SAR. The valves are designed to appropriate conditions. Missile protection is not degraded. There are no additional seismic interaction concerns of non-safety related components with this modification's safety related components. The mitigation of a 10 CFR 50 Appendix R fire is not adversely affected. Increased hydrogen generation due to additional amounts of zinc or aluminum is not a concern. The RCS pressure boundary is not degraded. These modifications and associated UFSAR changes involve no USQs or safety concerns. No technical specification changes are required. UFSAR Section 6.3.2.6.3 AND Table 6-16 were appropriately revised.

DESCRIPTION

SYSTEM: Low Pressure Injection

NSMs 1 and 3 2977 replaces the Low Pressure Injection (LPI) cooler shell outlet valves (1,3LPSW-4 and 1,3LPSW-5), the Reactor Coolant Pump (RCP) inlet isolation valve (1,3LPSW-6), and the RCP outlet isolation valve (1,3LPSW-15) on Units 1 and 3, respectively. Check valves 1,3LPSW-75 and 1,3LPSW-76 were removed. Vent valves were added to facilitate routing system testing. A line stop (Marbo plug) was used during installation, using existing permanent pipe "line stop" components. The new valves are stainless steel so they are better suited for service water applications than the current carbon steel valves.

SAFETY EVALUATION SUMMARY

This change does not create any conditions or events which lead to accidents previously evaluated in the SAR. The LPSW System still functions as designed and still provides the cooling function for the loads it serves. The SAR evaluates a Turbine Building flood that is caused by the large CCW expansion joint failure. The valves that are being removed are check valves that are downstream of 1LPSW-4 and 1LPSW-5 and do not serve a design basis or operational purpose. The removal of the checks valves does not adversely affect the flooding in the Turbine Building. The LPI System still provides its emergency function of providing cooling to the LPI coolers and other emergency loads. The replacement valves fail in the same position as the old. Valves 1LPSW-6 and 1LPSW-15 still provide the containment isolation function. No adverse effects on the Appendix R fire analysis was found. Appropriate design conditions were used. The replacement valves, piping, piping components, and existing Marbo components are all Class F (QA Condition 1) valves. The modification does not adversely affect the LPSW flow to the LPI coolers, RBCUs, RCP coolers, or any other LPSW supplied loads. Replacement valves LPSW-4 and LPSW-5 have throttle capability. The power supply, cabling, breakers for the valves, and the valves' control signals are still adequate after replacement. The replacement components are all environmentally qualified or in a mild environment. There are no new safety/non-safety electrical interfaces. There are no design temperature and pressure changes. The pipe stress analysis is not adversely affected. There is no adverse effects on separation or missile protection requirements. There are no seismic interaction concerns with non-seismic structures, systems, or components with the new or replacement QA-1 equipment. These modifications involve no USQs or safety concerns. No UFSAR or technical specification changes are required.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Reactor Coolant

NSMs ON-1,2 and 3 2978 replaced reactor coolant system (RCS) vent valves 1,2,3RC-155, 156, 157, 158, 159, and 160 on Units 1, 2 and 3, respectively. All six of these valves are used to vent steam or non-condensable gases from the RCS high points during operational events or accidents. The vents are used to ensure that gas buildup does not interfere with natural circulation flow during events which involve a loss of reactor coolant pumps (RCPs). The new valves use more reliable position indication and require less work to maintain. This change reduces outage time and worker radiation dose.

SAFETY EVALUATION SUMMARY

The RCS continues to perform its normal and emergency functions. The new valves meet the design condition requirements. Missile protection is not degraded. There are no additional seismic interaction concerns of non-safety related components with this modification's safety related components. The mitigation of a 10 CFR 50 Appendix R fire is not adversely affected. There is no increase in potential hydrogen generation since no additional amounts of zinc or aluminum are introduced to containment. The RCS pressure boundary, as evaluated in the SAR, is not degraded. This modification involves no USQs or safety concerns. No UFSAR or technical specification changes are required.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Main Steam

NSMs 1 and 3 2979 replaced valves 1,3MS-126 and 1,3MS-127, and replaced the pneumatic control loop for valves 1,3MS-126 and 1,3MS-129 on Units 1 and 3, respectively. The existing control loops are located in the Turbine Building. The controls for these valves were replaced with electronic controls relocated to the appropriate control room. Also, instrumentation that provides input to an annunciator window and the computer were also replaced.

SAFETY EVALUATION SUMMARY

The new valves do not adversely impact the heat removal capability of the MS or EFW Systems. Requirements for the elimination of AC independence for the EFW System are maintained. These modifications do not impact containment integrity or open any new radiological release pathways. The original and new valves are both non-QA Condition. The new valves are designed to the same design temperature and pressure as the adjoining pipe. The existing air supply is adequate to operate valves MS-126 and MS-129. The modification does not adversely affect flowrates. Power supply, breakers, and cabling are adequate for the new electronic circuitry. An electrical 10 CFR 50 Appendix R fire review was completed for the design phase. No changes to the damage control measures for an Appendix R fire are needed. The pipe stress analysis is not adversely affected. Valve MS-126 is also sized so that the Auxiliary Steam header pressure is not overpressurized if it and valve MS-129 fail open. The existing instrument loops are non-QA Condition. The new instrument loop is non-QA Condition and non-safety related, except for the QA-4 mounting on the control boards. A control board seismic review was performed and no adverse effects were found to exist. This modification involves no USQs or safety concerns. No UFSAR or Technical Specification changes are required.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Integrated Control

NSMs ON-1, 2, and 3 2989 replaced the Integrated Control System (ICS) and upgraded the Non-Nuclear Instrumentation (NNI) and Auxiliary Control System (ACS) on Units 1,2, and 3, respectively. The new ICS implements the major concepts of the B&W Owner's Group (BWOG) designed PCS algorithm. The new ICS utilizes Framatome Technologies' Incorporated (FTI) design and manufactured Control STAR™ modules. Major changes as part of this modification are as follows:

- The separation of HAND/AUTO (H/A) power supplies and redistribution of major loads,
- The installation of a new control algorithm by implementing the major concepts of the Plant Control System (PCS) algorithm which was designed by the Babcock and Wilcox Owner's Group (BWOG),
- Expansion of some ICS concepts (bumpless transfer, tracking, operating limits and integrated control) by the new algorithm to improve system response,
- Modifications to the load control panel (operator interface on control boards) to accommodate the new algorithm,
- Signal buffering or isolation for signals leaving the ICS cabinet,
- Employing the Steam Generator/Reactor H/A station as strictly a Feedwater Master demand station,
- Changing the emphasis/control of the ICS from megawatt generation to core thermal power (CTP)
- Allowing automatic operation above 2% power (Nuclear Instrument-- NI-- flux),
- Calculation of its own thermal power best for comparison with the OAC power calculation, and
- Providing automatic signal selection through the use of a median signal selection scheme and selective use of the Smart Automatic Signal Selector (SASS) for many plant parameters.

SAFETY EVALUATION SUMMARY

The new digital ICS continues to be a non-safety related system. ICS provides no protective functions. The new ICS fulfills the functions of the current analog ICS and maintains separation between it and safety related plant protective systems (such as the Emergency Feedwater System, and the Reactor Protective System, etc.). It does not hinder the operators from taking any necessary manual action that could be currently performed, nor does it alter the characteristics of the components under ICS control (such as control rods, control rod drive speeds, feedwater regulation valves, etc.). The functionality of the HAND/AUTO stations has not changed.

SAR required functions performed by the ICS were retained in the new design, which also complies with the UFSAR Chapter 15 accident analyses. Additionally, compliance with responses to IE Bulletin 79-27, NUREG-0667, Generic Letter 89-19, and NUREG-0737 was not compromised by this modification. Post accident indication (Reg. Guide 1.97) functions were not affected by this modification.

This modification involves no USQs or safety concerns. No Technical Specification changes are required. The following UFSAR sections were identified as needing review and revision, as necessary, to reflect installation of the new ICS: 3.1.12, 4.3.3.1.4, 5.1.2.2.2, 7.1.2.5, 7.3.2.3, 7.4.1, 7.4.1.2, 7.4.1.3.3, 7.4.2.2.1, 7.4.2.2.2, 7.4.2.2.3, 7.4.2.3, 7.4.2.3.1.1, 7.4.2.3.1.2.3, 7.5.2.4, 7.5.2.5, 7.5.2.39, 7.6.1.1.8, 7.6.1.2, 8.3.2.1.6, 15.7.2, 15.7.3, 15.7.4, 15.8.2, 15.13.3.2, Figures 7-6, 7-6a, 7-14 through 7-17a, 7-19, 7-19a, and 8-8, and Tables 7-5, 7-6 and 7-6a.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Essential Siphon Vacuum (ESV), Fire Protection

NSM ON-43000, Part BL2 installed and tested the Non-QA Condition redundantly configured power supplies for the ESV building and related equipment. The number and locations of new fire detectors installed required a change to UFSAR Table 9-12 and SLC Table 16.9-6.

SAFETY EVALUATION SUMMARY

In addition to the power supplies and related components, six new fire detectors were installed per this modification part. This change does not initiate, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. This modification involves no USQs or safety concerns, and Technical Specification changes are required. UFSAR Table 9-12 and SLC Table 16.9-6 were revised to address the new fire detectors.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Essential Siphon Vacuum (ESV), Fire Protection

NSM ON-43000, Part BM1 added fire hose stations to the new ESV building.

SAFETY EVALUATION SUMMARY

The new fire protection hose stations were designed to appropriate codes. The High Pressure Service Water (HPSW) system provides the water suppy, and is not adversely affected. This modification does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. This modification involves no USQs or safety concerns. No technical specification changes are required. UFSAR Section 9.5.1.6 and SLC 16.9.4 were revised to address the new fire hose strations.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Essential Siphon Vacuum (ESV)

NSM ON-43000, Part B1 added a seismic pad/foundation, with grounding mat, for the vacuum pumps, vacuum tanks, and other ESV equipment. An exterior light pole was also removed. This part also relocated the facility trash compactor to make space for the pad. The pad/foundation is designed to contain ESV systems and equipment. The modification also installs portions of some mechanical systems that lie in and/or below the pad. The mechanical systems include the seal water discharge line (non-QA Condition) and the High Pressure Service Water (HPSW) fire protection line (QA-3).

SAFETY EVALUATION SUMMARY

The new, relocated, and deleted equipment has no adverse effects on the Site Security Plan. The pad/foundation is designed QA-1 (excluding analysis for tornado and equipment missiles) and is a Class 2 structure with respect to MHE loads. This design matches the current ECCW structure design. The catch basin is non-QA Condition. The pad/foundation and other new equipment does not adversely impact any SSCs. There are no adverse seismic interactions. The pad/foundation is not subject to direct wind forces. The 9th edition of the AISC Steel Construction Code was used for the ESV Building only. Currently, UFSAR Section 3.8.5.2 lists the 6th edition of the code for Class 2 structures. UFSAR Section 3.2.2.2 states that due to the numerous code references located throughout the UFSAR, no attempt is made to revise these references as codes are amended, superseded, or substituted. This section also states that the intent of Duke Power is to comply with the latest version of existing codes unless material and/or design commitments have progressed to a state of completion such that it is not practical to make a change. This section is specifically addressing piping classifications, but this intent is used for this UFSAR variation also.

This modification does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. This modification involves no USQs or safety concerns. No technical specification changes are required. FSAR Sections 3.2.1.1.2 and 3.8.5 were revised to address the ESV pad/foundation as a new Class 2 structure.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Essential Siphon Vacuum (ESV)

NSM ON-43000, Part BS2 added a building that is used to enclose some of the ESV System's equipment, such as the ESV pumps and motors. The building was erected on the pad/foundation provided by Part B1 of this modification. The ESV building is part of the overall Oconee Service Water project.

SAFETY EVALUATION SUMMARY

The addition of the new ESV building has no adverse effects on the Site Security Plan. The building was erected on a QA-1, Class 2 structure concrete pad and the building itself is a Duke Class 2 structure qualified for the MHE. The building is designed QA Condition 4 to prevent seismic interaction concerns with the contained equipment. This design matches the current Emergency Condenser Circulating Water (ECCW) structure design. The new ESV building itself does not support any important to safety SSCs. The building is adequately separated from non-fire rated buildings. Class 2 loads are designed for 95 mph wind loads. The building is not designed for tornado loads or tornado missiles. The existing ECCW System is not designed to withstand a tornado or resulting tornado missiles.

This modification does not initiate, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. This modification involves no USQs or safety concerns, and no technical specification changes are required. UFSAR Sections 3.2.1.1.2 and 3.8.5 were revised (in conjunction with the changes for Part BS1 of this NSM) to address the ESV building as a new Class 2 structure.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Low Pressure Service Water (LPSW)

NSM ON-13001 Part AK1 and AM1 installed a portion of the Units 1&2 LPSW minimum flow lines. The lines were replaced to ensure minimum flow after the ES signals were removed from valves 1LPSW-4 & 5. See part AM-2 for more details

SAFETY EVALUATION SUMMARY

The piping, valves, and breakdown orifices are QA-1 and Class F and seismically qualified. The discharge from each pump will not be returned to its own suction supply. This modification does not initiate, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. This modification involves no USQs or safety concerns, and no technical specification changes are required. No UFSAR changes are required.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Low Pressure Service Water

NSM ON-13001 Part AM2 replaces each Low Pressure Service Water (LPSW) pump's discharge elbow with an elbolet and then connects the elbolet on each of the new discharge elbows to the minimum flow lines installed by NSM ON-13001 Parts AM1 and AK1. The elbolet is used to allow the minimum flow lines to branch at an angle and thus allow for full flow through the minimum flow lines. The minimum flow lines were broken into different parts to separate the lines into an innage and an outage portion. Part AM2 is the outage portion. Parts AM1 and AK1 are the innage portion, which is to install portions of the minimum flow lines from the Condenser Circulating Water (CCW) wet tap valves, LPSW-957 and LPSW-958, to just beyond each pump's multi-stage breakdown orifice plate and bypass, at valves LPSW-952, LPSW-953, and LPSW-954. The lines are needed to assure minimum flow after Engineered Safeguards (ES) signals are removed from valves 1LPSW-4 and 1LPSW-5.

SAFETY EVALUATION SUMMARY

The LPSW System still has adequate flow for normal and emergency loads with flow diverted through the minimum flow lines, even with a single failure. The piping, valves, and breakdown orifices are QA-1 and Class F and seismically qualified to the MHE. The minimum flow instrumentation is non-QA-1 in function, but QA-1 for pressure boundary. Containment integrity is not degraded and no new radiological release pathways are created. The minimum flow lines do not have to be designed for tornado loads since the existing ECCW and LPSW Systems are not required to be designed for tornado loads. The minimum flow lines are not adversely affected by a turbine missile since they are not susceptible to a Low or High Trajectory Missile. There is no adverse impact on Appendix R damage control measures or pipe stress analyses. The design of the minimum flow lines prevents the concerns identified in NRC Bulletin 88-04. This modification involves no USQs or safety concerns. No UFSAR or Technical Specification changes are required.

DESCRIPTION

SYSTEM: Low Pressure Service Water

Part C (CM1, CL1) of NSM ON-13001 adds a new valve, 2LPSW-139, downstream of valve LPSW-941. This new valve receives the power source that is currently on valve LPSW-139 (Unit 2 Class 1E power supply) and it operates from the switch that currently controls LPSW-139 (outside the Unit 1 control room). The control switch for valve LPSW-139 was relocated to inside the Unit 1 control room. The power source for LPSW-139 comes from Unit 1's Class 1E power supply. Valve LPSW-139 was renumbered to be unit 1 specific (1LPSW-139). Valves LPSW-940 and LPSW-941 were renumbered to be unit specific valves (1LPSW-940 and 2LPSW-941). The elbow between valves 2LPSW-941 and 2LPSW-139 was replaced with a Duke Class F fitting.

SAFETY EVALUATION SUMMARY

The LPSW System continues to have adequate flow for both normal and emergency loads with the new valve and pipe components. There are no single active failures introduced that would prevent the LPSW system from performing its safety related function. The new piping elbow that was installed downstream of valve 1LPSW-139 and upstream of the new valve 2LPSW-139, as well as the new valve 2LPSW-139 are Class F and QA-1 and seismically qualified to the MHE. The new controls are QA-1 and seismically qualified, except for one cable that provides valve indication to the non-safety related Operator Aid Computer (OAC). The power sources for valves 1LPSW-139 and 2LPSW-139 are 1E (safety related). The new valve 2LPSW-139 meets the description of seismic/non-seismic boundaries as defined in the UFSAR, has been selected and specified to the requirements of Generic Letter 89-10. The new design satisifies the electrical specifications identified in UFSAR. The QA-1 equipment is qualified for the mild environment. There are no adverse effects on control board seismic qualifications and no 10 CFR 50 Appendix R fire concerns. The new components' design temperature and pressure are the same as the adjoining pipe. There is no adverse affect on piping stresses.

This modification involves no USQs or safety concerns. No Technical Specification changes are required. UFSAR Section 9.2.2.2.3 and Figure 9-11 were revised appropriately to reflect the new design and configuration.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Low Pressure Service Water

NSMs ON-1,2, and 33001, Part D, removed the Engineered Safeguards (ES) signal from the "A" Low Pressure Injection (LPI) cooler outlet isolation valve, 1,2,3,LPSW-4, and the "B" LPI cooler outlet isolation valve, 1,2,3LPSW-5. The RZ modules associated with these valves located on the control boards were disabled and the labeling removed. The cables indicating valve position from these valves to the ES cabinets were deleted. Cables from the ES cabinet providing valve ES position to the computer for the valves were deleted. The ES cabinet internal wiring was not disconnected. New cables for each valve were added from the valve limit switches to the computer to provide valve position indication.

SAFETY EVALUATION SUMMARY

The removal of the ES signal from valves LPSW-4 and 5 was reviewed and approved by the NRC. Operator action is now credited during a LOCA/LOOP scenario. The modification does not eliminate valve position indication or valve control from the control room. The affected valves are QA-1. The modification is QA-1. The existing ES cables are all being removed or spared (left in place with no function). These cables currently provide ES control, valve position indicating lights on the RZ modules, and valve position input to the computer. New cables are added such that position indication is provided to the computers. The existing valve position indication and valve control in the control room is still available. The RZ modules were disabled (including the valve position indication) and labeling removed are associated with the ES function of the valves. The new cables are QA-1. The electrical components are adequate for their function. An electrical 10 CFR 50 Appendix R fire review was completed for the design phase. No seismic interaction concerns were found to exist with the new QA-1 cables and existing non-seismic structures, systems, and equipment. The new cables are designed for protection against tornadoes and missiles from hypothesized plant equipment failures (including turbine missiles). There are no new safety/non-safety electrical interfaces. The new cables are environmentally qualified for their environment. The cables meet the electrical specifications identified in the UFSAR.

This modification involves no USQs or safety concerns. No technical specification changes are required. UFSAR sections 6.3.2.2.2, Table 7-3, and Figures 6-1 and 9-12 were revised accordingly.

DESCRIPTION

SYSTEM: CCW

NSM 13003, Part A, upgraded a portion of the Condenser Circulating Water (CCW) pump discharge valve (CCW-10, CCW-11, CCW-12, and CCW-13) control circuitry from non-QA condition to QA-1. This change assists in assuring that the discharge valve remains in its existing position (opened or closed) following power restoration after a LOOP event.

SAFETY EVALUATION SUMMARY

Upgrading the CCW pump discharge valve control circuitry from non-QA condition to QA-1 does not create any conditions or events which initiate, or adversely impact the mitigation of, any accidents evaluated in the SAR. Neither the method of operation, nor the function of the CCW pump discharge valve controls is changed. All non-safety/safety electrical interfaces are protected with safety related (QA-1) isolation devices. No single failure exists which could result in the discharge valve closing during a LOOP or LOCA/LOOP. In addition, the valve circuitry ensures that the discharge valve for the last pump running remains open to make siphon flow available. All power sources are QA-1. The design meets the electrical specifications identified in UFSAR Sections 8.3.1.5, 9.5.1.4.3, and the applicable portions of Section 8.3.1.4. No new cabling was added. There is no effect on the control board seismic qualifications. The QA-1 components are located in mild environments. The capability to close the CCW pump discharge valve with a control room push-button is still available to reduce the potential of Turbine Building flooding due to siphoning of CCW. The mitigation of an Appendix R fire is not adversely affected. No new radiological release pathways are created. There are no seismic interaction concerns due to existing non-safety structures, systems, and components impacting the newly classified/designed QA-1 equipment. The valve controls do not have to be designed for protection against tornado loads. This modification part meets the separation/protection criteria for turbine missiles. No hypothetical missile could cause the modified discharge valve to change position following power restoration after a LOOP. This modification involves no USQs or safety concerns. No UFSAR or technical specification changes are required.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Main Feedwater

NSM ON-13007 removed the main feedwater pump (FDWP) discharge pressure input to the Reactor Protection System (RPS) Anticipatory Reactor Trip System (ARTS) and the Emergency Feedwater (EFW) System on Unit 1. This modification reconfigures the ARTS to initiate an anticipatory reactor trip solely in response to low FDWP control oil pressure. The EFW circuitry is reconfigured to automatically initiate on low control oil pressure and low-low steam generator level. The Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) initiation of EFW on low FDWP discharge pressure will remain unchanged. In addition, the control circuitry for the Unit 1 motor-driven Emergency Feedwater pumps (MDEFWPs) was rewired to match the Units 2 and 3 circuitry. Note: Until like permanent modifications can be installed on Units 2 and 3, the pressure trip function has been disabled via temporary modifications.

SAFETY EVALUATION SUMMARY

A submittal was made to the NRC for changes to Technical Specifications 3.4.2 and 4.1 to permit the removal of ARTS and EFW inputs from these switches. Approval for the proposed changes to Technical Specifications 3.4.2 and 4.1 was received (TS Amendments 216/216/213; April 1996).

Other diverse means are available to ensure that the design requirements for the ARTS and EFW actuation are still met. There is a potential reduction in the possibility of a reactor trip caused by secondary side instrumentation. UFSAR Chapters 10 and 15 analyses remain conservative and bounding for the plant configuration following removal of the FDWP discharge pressure switches. Thus, the associated dose analyses are bounding for the new plant configuration. No new electrical or mechanical failure mechanisms are postulated. This modification involves no safety concerns or USQs. UFSAR sections 7.4.3, 10.4.7, and the bases of Selected Licensee Commitments Section 16.7.3 were updated to reflect this modification.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Low Pressure Service Water

NSM 13022 replaced and relocated control valves 1LPSW-251 and 1LPSW-252 on Unit 1. The associated solenoid valves were also relocated. Isolation valves 1LPSW-254 and 1LPSW-256 were replaced. Vent and drain lines were modified/added. Some carbon steel piping around these valves was replaced with stainless steel pipe. Marbo plugs were installed through existing fittings. The existing control valve E/P converters were deleted. The existing pressure regulators were reused. A permanent differential pressure gauge across each control valve was added. A core drill through the wall structure was performed to provide access for instrument tubing, power, and control wiring.

SAFETY EVALUATION SUMMARY

The modification does not adversely affect the LPSW flow used for normal or emergency operation. Travel stops are still on the valves to prevent excessive flow to the LPI Coolers. No new LPSW System operational function occurs as a result of this change. Relocation of the valves and solenoids does not degrade their qualifications. All other electrical components are in the control room and are adequate for the environment. The Appendix R fire scenario is not impacted. There are no new safety/non-safety electrical interfaces. The control board seismic qualification is not impacted. The power supply, cabling, and breakers are adequate for the new/replaced loads. The replacement valves and components are all rated for the existing system design parameters of temperature and pressure. The new valves, components, and the component relocations do not create any undesireable seismic interactions. The pipe stresses and support/ restraints were analyzed as adequate. There is no adverse effects to separation or missile protection requirements. The new differential pressure gauge across each of the control valves is used monitor for blockages. The gauges are not required to function during an accident and are non-QA. Although, the new differential gauges are not QA-1, they are qualified to maintain the pressure boundary. The relocation of the control valves neccesitated a core drill through the nearby wall. The affected wall is a pressure boundary, but is not a fire barrier, and its integrity is not reduced due to the core drill.

This modification involves no USQs or safety concerns and no Technical Specification changes are required. UFSAR figure 8-5 currently shows the detailed DC and AC Vital Power Systems One-Line Diagram for Unit 1, which includes loads coming off the power panelboard breakers of which two shown involve the control valves. (Note Electrical Figure 8-5 was completely revised in this update per change package 97-111 to clarify, and remove unnecessary detail, and address all 3 Units)

DESCRIPTION

SYSTEM: Siphon Seal Water (SSW)

NSM ON-52932 part AM2 installed the buried portion of the new Siphon Seal Water (SSW) Headers to facilitate supplying Essential Siphon Vacuum (ESV) pump sealing water and Condenser Circulating Water (CCW) pump sealing and oil cooler water from Low Pressure Service Water (LPSW). The buried piping is part of the new SSW System, which was not functional in 1997. The purpose of the SSW system is to assist in ensuring a reliable siphon to the LPSW pump suction following a LOCA/LOOP event. Core drill holes were made to the existing Radwaste Facility (RWF) trench for the new piping. The pipe ends were capped. Note: This modification is simply one small part of the overall Oconee Service Water project.

SAFETY EVALUATION SUMMARY

The installation of the SSW buried headers does not create any conditions or events which initiate, or impact the mitigation of, any accidents evaluated in the SAR. The installation has no adverse effects on the Security Plan. This modification creates no new radiological release pathways. The buried SSW piping is QA-1 (Class F) and seismically qualified to the MHE. No equipment, other than the SSW buried piping, is installed under this part of the modification. No electrical components or electrical changes are being made. There are also no adverse effects on the structural integrity of any piping in the area, such as tornado protection of existing pipe or supports of the pipe. Heat tracing/freeze protection is not required since the piping is buried below the frost line. Snow, ice, and wind effects are not a factor since the piping is buried. The piping has been designed to provide adequate flow when connected to the other SSW piping. The piping stress analysis has shown the new arrangement is satisfactory to perform the design function under the required conditions. There are no seismic interaction concerns. The core drills and access openings do not adversely affect the structural integrity of the RWF trench. The RWF trench is currently non-QA Condition. Since the buried piping is non-functional for this part of the modification, the trench can be non-QA. The RWF trench will be seismically upgraded before piping installed by other parts of the Service Water Project can make the SSW System operational. Buried piping does not have to be designed for protection against tornado loads or turbine missiles. This modification involves no USQs or safety concerns and no technical specification changes are required. Once the SSW system is declared functional and its description is added to the UFSAR, UFSAR Sections 3.2.2 and 3.7.3.8 should also be revised, to list that the SSW System can withstand an MHE and that the SSW piping is a seismically designed safety related buried line, respectively.

NUCLEAR STATION MODIFICATIONS

DESCRIPTION

SYSTEM: Essential Siphon System (ESV)

NSM 53000, Part A. added a trench that runs across the intake dike. The trench is designed to contain new systems and equipment. It contains the vacuum lines, Siphon Sealing Water (SSW) lines to the intake, and necessary instrument cabling used in the Essential Siphon System (ESV) for the Service Water Project. Note: This modification is simply one small part of the overall Oconee Service Water (OSW) project.

SAFETY EVALUATION SUMMARY

The construction of a new trench does not cause, or affect the mitigation of, any prevously analyzed SAR accidents. The trench has no adverse effect on the Site Security Plan. Trench components that support QA-1 components are designed QA-1. Some components are designed QA-4 to prevent seismic interaction, and other non-seismic interacting components are designed non-QA. There are no adverse effects on the structural integrity of the dike or the existing piping in the area. Like the existing ECCW system, the new trench is not designed for tornado loads or turbine missiles. Redundant drains were added to the trench to allow rain water to drain out and not adversely affect the equipment contained within. The concrete around the holes in the trenches is QA-1 and will provide the draining function even if the drain piping is damaged. The new trench causes the dike to have a lower water access level than 815 feet MSL. The difference in anticipated water levels are compared to the dam and dike's lowest points. The wave height at the location of the dike where the trench is to be located will be lower than the wave height at the dam. Using the expected lower wave height at the trench, the margin between the minimum specified height of the dam and the maximum wave runup water level is less than the trench's minimum height and its expected wave runup water level. The potential for site flooding is not a concern since the amount of overwash into the trench and onto the site, if any, would be minimal. While earthen dams and dikes can have erosion and degradation of the structures due to overwash, the new trench is made of concrete. The anticipated maximum water level and any overwash would not cause degradation of the trench or dike. This modification involves no USQs or safety concerns and no technical specification changes are required. UFSAR Sections 3.2.1.1.2, and 3.8.5 were revised to address the trench as a new Class 2 structure. UFSAR Sections 2.4.2.2 and 3.4 were revised to address the trench level being lower than the dike height of 815 feet.

DESCRIPTION

SYSTEM: Essential Siphon System (ESV)

Part CS1 of NSM 53000 added a trench that runs from the Radwaste Facility Trench to the ESV building. This trench is used to contain ESV electrical cabling. This part of the NSM will also added an extension of the dike trench (dike trench was added as part of another modification) to provide a transition zone for the piping and conduit bank (and associated cabling) to come together. Steel covers are installed on most of the new cable trench, new dike trench extension, and the existing dike trench that was installed under a previous modification. Sections of the trench that are designed for vehicular traffic are provided with removable reinforcing concrete covers designed to support highway loads. Some steel stairs are also to be installed to provide a safe access to the dike trench and intake bridge walking surfaces. Concrete anchor blocks are installed in the trenches to transfer the loads from the piping that is installed as part of other Service Water Project NSMs to the trenches.

SAFETY EVALUATION SUMMARY

The trench and addition are designed to contain systems and equipment. The trench and addition do not cause any accidents or adversely affect the Security Plan. The trench and addition do not adversely affect the mitigation of any accident. The trench and trench addition are QA-1 and classified as a Class 2 structure designed for MHE loads (excludes analysis for tornado and turbine equipment missiles). This design matches the current ECCW structure design. The concrete anchor blocks are QA-1 since they are for supporting QA-1 components. The trench covers and stairs are designed QA-4 to stay in position during a seismic event without falling into the trenches. In addition, the design loads are appropriate for the vehicular loads on the trench. There are no adverse effects to the structural integrity of the dike due to the trench addition. The trench and trench addition also do not cause adverse effects to the structural integrity of the piping in the area, such as tornado protection of the existing pipe or support of the pipe. The trench addition becomes an integral part of the ESV System Dike trench which was added to the appropriate sections of the UFSAR by a previous modification that added the Dike trench. This modification involves no USQs or safety concerns. No technical specification changes are required. UFSAR Sections 3.2.1.1.2 and 3.8.5 were revised to address the ESV cable trench.

II. MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Emergency Feedwater and Condenser Circulating Water

Minor modifications OE-7314 and OE-7302 replaced valves 1FDW-347 and 1CCW-269, which were gate valves, with globe valves. Accordingly, minor modifications OE-7371 and OE-7370 replaced the operators on these valves with more appropriate ones for the new type of valves. This replacement was done to alleviate industry concerns such as pressure locking and thermal binding.

SAFETY EVALUATION SUMMARY

The new valves are motor operated, and operate as described in the UFSAR. The globe valve has better flow control than the gate valve it replaced. Although the stroke time is slower, there is no stroke time requirement for these valves. Changing these valves and operators does will not affect the operability of the associated system in any way.

This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components will continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or the UFSAR are required.

MINOR MODIFICATION (ONOEs)

DESCRIPTION

SYSTEM: Reactor Protection System (RPS)

This 10CFR50.59 evaluation and safety review are for installation of ONOE-8790 which replaced the existing RPS flux/imbalance/flow Bailey 880 analog signal processing and trip initiation modules with modern BWNT STAR digital microprocessor-based and conventional analog-based signal processing, isolation, and trip modules on Unit 1.

SAFETY EVALUATION SUMMARY

The replacement of outdated, obsolete RPS electrical components with newer more reliable devices that provide all required SAR described functions does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. No new accidents are created. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. The RPS functions have not changed. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Sections 7.2.2, 3, 7.4.2, and Table 7-5 of the Oconee UFSAR were revised accordingly.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Low Pressure Injection System (LPI)

Minor modifications ONOE-8860, 9205, and VN-9463A changed the LPI test header to the BWST from single to double isolation. These mods resized LP-42 and rearranged and relocated the surrounding piping.

SAFETY EVALUATION SUMMARY

The affected valves and piping were modified to ensure a double isolation exists during LPI pump flow tests. The mod only reduces the potential for leakage from the LPI system to the BWST when aligned post-accident. The LPI system continues to perform its intended safety function. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications were required. UFSAR Figure 9-19 was revised to reflect the new arrangement.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Reactor Coolant System (RCS)

Minor modification OE-8888 installed a new globe valve, 3RC-202, downstream of valve 3RC-147 to provide sample line isolation. Likewise, minor modification OE-8889 installed a new globe valve, 3RC-203, downstream of valve 3RC-148 to provide sample line isolation. The purpose of these modifications was to alleviate concerns with isolating flow using valves 3RC-147 and 148, which are needle valves. These valves are not efficient at isolating flow for sampling.

SAFETY EVALUATION SUMMARY

The valves were installed to enhance normal operation of the RCS sampling system. Installation of these valves did not affect the ability of the RCS to perform its intended safety function. The new valves are of pedigree consistent with the existing system. Adding these valves did not affect the operability of the associated systems in any way.

This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or the UFSAR are required

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Various Electrical

Minor modification OE-9049 revised the generic calculation, OSC-5599, for Generic Letter 89-10 motor operated gate and globe valves. The calculation was revised to include the latest voltage assumptions, and to include more conservative assumptions regarding efficiency.

SAFETY EVALUATION SUMMARY

The revisions to the generic 89-10 calculation do not affect the configuration or design of the plant. Any changes which resulted from new calculational results were evaluated in separate 10CFR50.59 evaluations. Therefore, the revisions to this calculation are administrative in nature. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components will continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or the UFSAR are required

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Reactor Coolant System

Minor modification OE-9343 removed tubes in the 1A steam generator from service by plugging and stabilizing as necessary, based on eddy current testing results or for preventative reasons.

SAFETY EVALUATION SUMMARY

The steam generator tubes are only an accident initiator in the steam generator tube rupture event. Plugging or removing tubes from service does not increase the probability of a tube rupture. Removing tubes from service by plugging and stabilizing is an approved NRC method. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or the UFSAR are required.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Reactor Coolant System

Minor modification OE-9344 removed tubes in the 1B steam generator from service by plugging and stabilizing as necessary, based on eddy current testing results or for preventative reasons.

SAFETY EVALUATION SUMMARY

The steam generator tubes are only an accident initiator in the steam generator tube rupture. Plugging or removing tubes from service will not increase the probability of a tube rupture. Removing tubes from service by plugging and stabilizing is an approved NRC method.

This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components will continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or the UFSAR are required.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Control Battery Room Duct System

Minor modification OE-9540 revised a system drawing that formerly required the ducts to the battery room to be blanked off from the Auxiliary Building. With blanks installed, the battery room pressure is too low to pass the performance test

SAFETY EVALUATION SUMMARY

The Auxiliary Building ventilation system is not safety related, not required to mitigate an accident, and continues to function as before the modification. The battery room temperature and pressure are now maintained within acceptable limits. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components continue to perform their design functions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification and no Technical Specifications are required. UFSAR Figures 9-27 and 28 were updated accordingly.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: High Pressure Injection System

Minor modifications OE-9582 and 9591 installed two valves, a 1" swing check valve (3HP-393) and a 2" globe valve (3HP-285) that replaced an existing stop check valve. The purpose of this modification was to replace the existing stop check valve, since it failed its leak rate tests. SOER 88-03 commitments required replacement of this valve.

SAFETY EVALUATION SUMMARY

The valves were installed to enhance the performance of the High Pressure Injection System by better performing the flow control and leakage prevention functions of the stop check valve. Installation of these valves does not affect the ability of the RCS to perform its intended safety function. The new valves are of pedigree consistent with the existing system. Adding these valves did not affect the operability of the associated systems in any way. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications were required. UFSAR Figure 9-17 was revised per this modification.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Standby Shutdown Facility

Minor modification ONOE-9819 made the following changes:

- Added additional restrictions for SSF ASW flow rate to all three units.
- Lowered minimum required initial SSF ASW flow rate to a unit (with and without instrument uncertainty).
- Added basis for the 110 degrees F max limit on SSF Service Water System inlet temperature
- Revised a design document to change minimum required SSF ASW pump capacity to a higher value

SAFETY EVALUATION SUMMARY

The implementation of these additional limits provides assurance that the SSW Service Water pumps have adequate net positive suction head during an SSF event. Changes to the SSW ASW flow rate limits resulted from a detailed review of calculations and licensing documents which indicated that more restrictive test acceptance criteria were necessary. Therefore, the changes implemented in this minor modification are more restrictive and ensure larger operating margins in the SSF Systems. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These affected components continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Keowee Hydro Station Electrical System

Minor modifications ONOE-9997 and 10161 resolved an identified problem with the K1ELKRL0060 and K2ELKRL0060 relays, respectively. The problem was that no control power was wired to these two relays as required. The unavailability of control power prevented the relays from performing the intended functions of automatically placing the voltage regulator in manual if the Keowee generator potential transformer secondary voltage is lost. The modification consisted of wiring the appropriate control power to these two relays.

SAFETY EVALUATION SUMMARY

The installation of appropriate control power to these two relays allows them to perform their intended functions. This modification allows functioning of the Keowee generator voltage regulator as intended. Therefore, this change is appropriate and conservative. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: LPSW

Minor Modification ONOE-10177 provides two gate valves, 1LPSW-968 and 2LPSW-966, and two drains, 1LPSW-969 and 2LPSW-967, one each per connection. The associated implementation procedure (TN/0/A/10177/MM/01M) performs the installation of two 12 x 14" taps into the existing LPSW non-essential header to provide tie-in connections for an LPSW bypass line to be used as part of the overall Oconee Service Water upgrade project. The minor modification and the implementation procedure were evaluated under a single safety evaluation.

SAFETY EVALUATION SUMMARY

Neither Minor modification ONOE-10177, nor its implementation procedure adversely affects any important to safety plant SSCs. These activities do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. The LPSW system continues to function as designed during normal and accident conditions, when required. The installation and implementation of this minor modification involves no USQs or safety concerns, and no Tech Spec changes are required. All installation activities performed under ONOE-10177 conform to existing SAR descriptions and requirements, therefore no UFSAR changes are required.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: High Pressure Injection (HPI), RCS

ONOE-10361 removed, inspected and restored the affected HPI/RCS piping and components on Unit 2 to the original design criteria. The HPI System developed a leak at a weld joint between valve 2HP-127 and the HPI nozzles going into the Reactor Coolant System (RCS). A section of piping between this valve and the RCS was removed for inspections of the weld and the adjacent piping. Also removed was part of the HPI nozzle warming line, which connects into this section of piping.

SAFETY EVALUATION SUMMARY

For this evolution, the RCS was treated as only having pressure boundary integrity to the HPI nozzles. The appropriate SLCs and plant conditions were maintained as if the HPI nozzle level is the plant's condition. The higher water level is desired in the event of a loss of decay heat removal. Any leakage is considered a loss of RCS inventory that is not being taken credit for and would be a cleanup concern, not an accident issue. The potential for loss of decay heat removal is not increased since the requirements for DHR systems for the water level at the HPI nozzles are met. Per Technical Specifications, the HPI System is not required to be operable if the RCS temperature < 350 F. The constraints specified in the minor modification prohibit taking the RCS above 350 F. One HPI pump may be used to meet the requirements of SLC 16.5.3 which specifies providing two of three available options for adding inventory to the RCS. The requirements for the SLCs that address RCS level at the HPI nozzle are met. The appropriate SLCs and plant conditions will be maintained as if the HPI nozzle level is the plant's condition. Any leakage would be to containment, which is required to have integrity. This modification involves no USQs or safety concerns. No technical specification or UFSAR changes are required.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: SSF Diesel Generator (D/G)

Minor modification ONOE-10409 changed the minimum required number of SSF (Standby Shutdown Facility) D/G starts from ten to six. There are four compressed air starting tanks, so each set of two can provide air for three succesive starts.

SAFETY EVALUATION SUMMARY

Experience has shown that the SSF D/G starts during the first attempt as long as there is no equipment failure. If equipment failure prevents the SSF D/G from starting, the D/G could not be repaired in time for the SSF to be used for accident mitigation. Therefore, there is no advantage to having the ability to successively start the SSF D/G 10 times. During normal standby operation of the SSF D/G Air Start System, pressure switches on the air receiver tanks start the air compressors when the indicated pressure in the air receiver tanks drops to 160 psig. Once the air compressors are activated, they continue to charge the tanks until the indicated pressure reaches 190 psig. During an accident which requires the SSF D/Gs to be Emergency Started, the compressors are assumed to be unavailable for recharging the receiver tanks. Based on manufacturers performance data, an initial air receiver tank pressure of 190 psig is capable of successively starting the SSF D/Gs twelve times without recharging the air receiver tanks. An initial air receiver tank pressure of 160 psig is capable of successively starting the SSF D/Gs eight times without recharging the air receiver tanks. Therefore, the minimum pressure maintained in the Air Receiver Tanks during the standby mode is less than the minimum pressure required to ensure ten successive emergency starts of the Diesel Generators.

This change does not cause, or affect the mitigation of, any prevously analyzed SAR accidents. No new radiological release pathways or failure modes are created. This modification involves no USQs or safety concerns and no technical specification changes are required. UFSAR section 9.6.3.4.2 was updated accordingly.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: High Pressure Injection System (HPI)

Minor modification OE-10417 installed a new non-QA redundant pressure transmitter on the Unit 1 Letdown storage tank (LDST). UFSAR Figure 8-5 was identified as needing to be updated.

SAFETY EVALUATION SUMMARY

The installion of the redundant indication is a conservative measure. The power supply and cabling are adequate. The tubing is QA-1 class B seismically qualified. Installation of this transmitter does not affect the ability of the HPI system to perform its nomal and safety functions. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components will continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications were required. UFSAR Figure 8-5 was identified as needing a revision. The entire electrical figure was revamped per NSM-13022 and UFSAR change package 97-111.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Low Pressure Service Water (LPSW)

Minor Modification ONOE-10447 installed 2LPSW-986, a 1" drain off the 14" LPSW non-essential header, and LPSW-987, a 2" drain off the 42" LPSW "A" header. The associated implementation procedure (TN/1/A/10447/00/AM1) performed a 14" wet tap into the existing LPSW non-essential header and a 36" wet tap into the existing LPSW "A" header to enable line stop isolations to be performed for NSM-13001 work. This work supports the overall Service Water upgrade project. The minor modification and the implementation procedure were evaluated under a single safety evaluation.

SAFETY EVALUATION SUMMARY

Neither Minor modification ONOE-10447, nor its implementation procedure adversely affects any important to safety plant SSCs. These activities do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. The LPSW system continues to function as designed during normal and accident conditions, when required. During the implementation phase, LPSW Pump "C" was off and the non-essential bypass header was supplying the Units 1 and 2 Main Turbine Oil Tank coolers. A 72-hour LCO was entered while LPSW Pump "C" was off. The installation and implementation of this minor modification involves no USQs or safety concerns, and no Tech Spec changes are required. All installation activities performed under ONOE-10447 conform to existing SAR descriptions and requirements, therefore no UFSAR changes are necessary.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Reactor Coolant System (RCS)

Minor Modification ONOE-10577 installed NUKON insulation on the Unit 3 RCS piping. The NUKON insulation supplements the existing mirror insulation.

SAFETY EVALUATION SUMMARY

NUKON insulation was approved by the NRC for containment applications. This minor modification does not adversely affect any important to safety plant SSCs. These activities do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. The RCS continues to function as designed during normal and accident conditions. The installation of this minor modification and corresponding UFSAR change involves no USQs or safety concerns, and no Tech Spec changes are required. UFSAR Section 5.4.5 was changed accordingly.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Standby Shutdown Facility (SSF)

Minor modifications OE-10648, 10649, 10650, and 10651 installed new globe valves 1HP-507, -508, -509, and -510, respectively. These valves were installed to provide the capability to throttle the Standby Shutdown Facility Reactor Coolant Makeup (SSFRCMU) System to Reactor Coolant Pump seal injection lines. This throttling capability is necessary to support system testing.

SAFETY EVALUATION SUMMARY

The installation of these valves does not affect the capability of the SSFRCMU System to deliver flow to the reactor coolant pump seals. The Installation of these valves does not affect the ability of the RCS to perform its intended safety function. The new valves are of pedigree consistent with the existing system. Adding these valves did not affect the operability of the associated systems in any way. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components will continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications were required. UFSAR Figure 9-35 was revised accordinly.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Low Pressure Service Water (LPSW)

Minor modification OEC-11029 performed a 24 inch wet tap and line stop into the Units 1 and 2 LPSW System "A" train to allow removal of another 36 inch line stop which was installed to perform modifications to the LPSW piping under NSM-13001 Part AM2, which is a portion of the Service Water System upgrade at Oconee.

SAFETY EVALUATION SUMMARY

The installation of this 24 inch wet tap and line stop was seismically evaluated. The modification did not adversely affect the capability of the LPSW System flow to the Low Pressure Injection coolers, Reactor Building Cooling Units, Reactor Coolant Pump coolers, or any other LPSW supplied loads. Contingencies were developed to address all potential concerns with the wet tap, and were appropriately reviewed prior to wet tap operations. The vendor which installed the wet tap was appropriately qualified for this activity. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components will continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Reactor Building Cooling (RBC)

ONOE-11094 revises the Reactor Building Cooling (RBC) System DBD and the UFSAR to remove the statement "RBCU heat transfer capacity has been evaluated for LPSW entering temperatures up to 90°F and shows acceptable results."

SAFETY EVALUATION SUMMARY

The RBC system is one of two independent engineered safeguards (ES) systems provided to remove heat from the RB following an accident. The cooling medium for the RBCU coils is low pressure service water (LPSW). RBCU heat removal capability is tested peridically per Tech Spec requirements to ensure adequate post-LOCA capacity is available. Acceptance criteria for RBCU heat removal is determined by the performance of PT/0/A/0160/06 (RBCU Perfromance Test) and documented in calculations OSC-5665, 5666 & 5667 for Oconee Units 1, 2 & 3 respectively. The performance test assumes LPSW temperature entering the coils is high enough so as not to be exceeded prior to performance of the next test. Historical data shows that the maximum LPSW temperature experienced has been 82 °F. Requiring that the RBC System have sufficient heat removal capacity assuming an LPSW temperature of 90°F is not necessary for the system to perform its safety function. The Tech Specs do not require that heat removal rate be based upon any assumed LPSW temperature.

This change does not cause, or adversely affect the mitigation of, any prevously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The required periodic testing will ensure the post-LOCA containment atmosphere conditions are maintained below the environmental qualification (EQ) curve. This minor modification involves no USQs or safety concerns and no technical specification changes are required. UFSAR sections 6.2.2.2.4 and 6.2.2.3 were revised accordingly.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Fuel Handling

Minor mod ONOE-11124 disables the intended design function of the Unit 2 Reactor Building Auxiliary Bridge Fuel Handling Crane. Specifically, the modification removed the inner mast and lower tower structure on the unused Auxiliary Bridge Fuel Handling Crane so that they may be used to increase the efficiency and enhance the modification schedule on NSM 32914, which upgrades the Unit 3 Fuel Handling Crane. After implementation of this modification, the Unit 2 Auxiliary Fuel Bridge is no longer capable of handling fuel. The connecting tube and 'A' frame was disposed of during the implementation of this mod. The hoist motor was removed for evaluation.

SAFETY EVALUATION SUMMARY

The scope of this 50.59 evaluation only includes partial disassembly of the unused Auxiliary Bridge on Unit 2. The refurbishment of these parts and their installation in Unit 3 Main Bridge is addressed in the 50.59 evaluation for NSM ON-32914. This modification was implemented with the Unit at cold shutdown with no fuel handling activities in progress (with the vessel head on or an equivalent foreign material exclusion cover installed). All the loads lifted ware under 1500 pounds and safe load paths are available. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. This modification involves no unreviewed safety questions or safety concerns. No technical specification changes are required. UFSAR section 9.1.4.2.2 was revised per NSD-220 pkg 97-94 to update the information related to the Unit 2 Auxiliary Bridge Fuel Handling Crane.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: HPI/SSF RC Makeup

Minor Modification ONOE-11222 was written to change the safety function of valves 1/2/3HP-417, to remain closed with their breakers open during an accident that requires operation of the SSF. RCS inventory control will be accomplished using the SSF RC letdown line.

SAFETY EVALUATION SUMMARY

The SSF RC makeup system provides seal injection flow to the RC pump seals to prevent seal degradation/ failure and to insure that adequate inventory is injected into the RCS to maintain RCS natural circulation flow. Based on the original design, if the flow rate added by the SSF RC makeup system was greater than the flow rate required to make up for RCS leakage, the SSF RC makeup bypass line could have been used to reduce the flow rate delivered by the SSF RC makeup pump to the RCS and the SSF RC letdown line could have been used to letdown flow from the RCS. It was later determined that reducing the total flow rate delivered by the SSF RC makeup system to the RCS by using the bypass line was undesirable because it reduced the amount of cooling water provided to the RC pump seals. Modifications were performed to increase the capacity of the SSF RC letdown line so that full SSF RC makeup system flow could be provided to the RC pump seals. Operating procedures were changed to require the SSF RC letdown line to be fully open before the bypass line could be used to divert flow away from the seals. Since the capacity of the letdown orifices is greater than or equal to the capacity of the SSF RC makeup pump, the SSF RC makeup bypass line is no longer needed for RCS inventory control. Since the SSF is not required to withstand a single failure, maintaining the ability to use the SSF RC Makeup bypass line provides no additional margin of safety. Therefore, this modification requires 1/2/3HP-417 to remain closed with its breakers open. This change does not cause, or affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. This modification involves no USQs or safety concerns and no technical specification changes are required. UFSAR section 9.6 was updated accordingly.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Auxiliary Steam

ONOE-11376 allows two operating units (Units 2 and 3) to be aligned to supply the Auxiliary Steam system only during times of high load exceeding the recommended limit of 150,000 lbm/hr per Unit. This monor mod created a new operating configuration not previously addressed in the UFSAR.

SAFETY EVALUATION SUMMARY

The auxiliary steam system is a shared system between the three units. The normal alignment of more than one unit's MS system to supply the AS system does not result in an increase in probability of a steam line break for each unit. The probability of malfunctions of equipment remains unchanged. The consequences of a steam line break in the AS system affect the MS lines for any unit aligned to the AS system. If two or more units are aligned to supply the AS system, actions simply must be taken on each unit to mitigate the event. Off-site dose consequences from a steam line break in the AS system is bounded by the double ended MS line break. No new failure modes are being introduced on the AS supply from any unit. MS-24 and MS-33 still provide the seismic/non-seismic boundary isolation between the MS system and the AS system. Alignment of one main steam line from multiple operating units to supply the auxiliary steam system has been reviewed and determined that no USQs or safety concerns exist. No technical Specifications are being changed by this new operating configuration. UFSAR sections 3.1.4 and 10.3.2 were revised to reflect the acceptability of the new alignment.

MINOR MODIFICATIONS (ONOEs)

DESCRIPTION

SYSTEM: Once Through Steam Generators (OTSG)

Minor modification OE-11381 documents tube repairs in the 1A OTSG. These repairs included the removal of any existing plugs which might contain defects, and installation of stabilizers (as necessary) and plugs as required by the results of visual inspections (bubble or drip tests) and eddy current testing, and the tube stabilization criteria document.

SAFETY EVALUATION SUMMARY

All the repair parts are QA condition 1 and are no more likely to fail than the existing parts. Tube stabilization and plugging are accepted industry practices for removing heat exchanger tubes from service. Once the steam generator manways are closed up and secured the RCS pressure boundary of the steam generator is intact.

Presently there are 234 in-service sleeves installed in the 1A OTSG and 451 tubes plugged. Based on the information in TAC ONTC-0-100A-0001-001 Rev. 1 there must be greater than 13,978 tubes available in each steam generator to meet core thermal-hydraulic design criteria. Following the completion of the repair activities (tube plugging/sleeving) performed under this modification, the TAC will be re-evaluated using the revised plugging and sleeving numbers.

This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The OTSGs continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or the UFSAR are required.

III. TEMPORARY MODIFICATIONS (TSMs)

DESCRIPTION

SYSTEM: Low Pressure Injection (LPI)

This 10 CFR 50.59 evaluation was performed for TSM-1376 and the associated procedure TN/2/A/1376/TSM/00M to repair a bonnet leak on Valve 2LP-1 on Unit 2.

SAFETY EVALUATION SUMMARY

Valve LP-1 is the first isolation off the reactor coolant system. The valve is required to open for circulation of coolant through the LPI coolers and to align the secondary boron dilution pathway during emergency operations. The unit conditions for this repair were 250F/350 psig with steam generators used to remove decay heat. RCS leakage must be controlled below Tech Spec limits. This injection leak repair was evaluated as acceptable. The repair activities do not affect the design basis functions of the LPI system. The activities performed do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No plant SSCs are adversely affected. There is no adverse affect on containment integrity, no new release pathways are created, and no new failure modes are created. This temporary mod and procedure involve no USQ's or safety concerns. No Technical Specification or UFSAR changes are required.

IV. PROCEDURES

DESCRIPTION

SYSTEM: Reactor Coolant System (RCS)

Two separate methodologies for performing steam generator primary head decontamination (PCHD) concurrently with RCS drain-down activities were evaluated. Appropriate procedure changes are made to comply with the prerequisite system conditions and administration controls to ensure that: 1) The minimum required core shutdown margin (SDM) levels are met, and 2) Rapid deboration/ dilution events are precluded. These conditions and controls are described in detail in a supporting engineering shutdown risk evaluation.

SAFETY EVALUATION SUMMARY

There are no adverse effects for concurrent PCHD/RCS drain-down activities for end-of-cycle (EOC) refueling outages, provided that the stipulated requirements and administrative controls within the controlling procedure are utilized. The UFSAR design basis accidents are unaffected. The engineering shutdown risk evaluation for the decontamination activities demonstrates that the potential consequences of inadvertent RCS deboration/dilution from both decontamination spray water and demineralizers will not lead to core reactivity shutdown margins below the levels required in the Oconee Technical Specifications. The Technical Specification 3.1.2.9 requirements for low temperature overpressurization protection (LTOP), and the associated Selected Licensee Commitments (SLCs) are unaffected. The SLCs for prevention and mitigation of loss of LPI decay heat removal events were not adversely impacted. The decontamination process and any subsequent dilution of RCS core-region water is not postulated to cause any kind of equipment malfunctions; the performance of the LPI pumps, for example, is essentially independent of the boron concentration of the water being pumped. The RCS and LPI System pressures, temperatures, and boron concentrations remain within the equipment and piping design parameters. No other equipment is adversely impacted. The concurrent performance of the PCHD with RCS drain-down activities does not involve any USQs or safety concerns. No Technical Specification or UFSAR changes are required.

PROCEDURES

DESCRIPTION

SYSTEM: Main Feedwater

Procedures CP/0/B/2002/16, CP/1/B/3002/21, and Secondary Chemistry Guidelines were changed to allow introduction of titanium into the plant feedwater system to reduce corrosion. These procedure changes necessitated revisions to UFSAR 10.3.5.1 to address titanium addition.

SAFETY EVALUATION SUMMARY

Titanium in the proper concentrations may be injected into main FDW to fight intergranular attack and stress corrosion cracking of the Alloy 600 steam generator tubing. EPRI research has shown Titanium to be an effective agent with no detrimental effects on secondary side components. Neither the activity, nor the corresponding UFSAR changes in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No USQs are involved with either the modification or the corresponding UFSAR change, and no Technical Specification changes are required. UFSAR Section 10.3.5.1 was revised to address titanium addition.

PROCEDURES

DESCRIPTION

SYSTEM: Reactor Coolant System (RCS)

Operations procedure OP/1/A/1102/01, Controlling Procedure for Unit Startup, Revision 213, Change E was originated to change the procedure to allow Unit 1 to start up in a three reactor coolant pump configuration instead of the normal four reactor coolant pump configuration. The Integrated Control System was placed in an abnormal mode to support this configuration during startup.

SAFETY EVALUATION SUMMARY

The review of this configuration showed that this mode of operation was evaluated in the UFSAR, and is bounded by accident analysis in the UFSAR. Oconee Technical Specifications allow for operation at power with three reactor coolant pumps. The Integrated Control System is a non safety related system which is not credited for accident mitigation. A detailed review of the impacts of this activity on reactivity management, potential for accident initiation, and affect on plant parameters and setpoints was conducted with none of these items adversely impacted. It was determined this activity does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The components will continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

PROCEDURES

DESCRIPTION

SYSTEM: Auxiliary Steam (AS)

OP/2,3/A/1106/22 (Auxiliary Steam) procedures were changed to allow two operating units (Units 2 and 3) to be aligned to supply the Auxiliary Steam system only during times of high load exceeding the recommended limit of 150,000 lbm/hr per Unit. These procedure changes created a new operating configuration not previously addressed in the UFSAR.

SAFETY EVALUATION SUMMARY

The auxiliary steam system is a shared system between the three units. The normal alignment of more than one unit's MS system to supply the AS system will not result in an increase in probability of a steam line break for each unit. The probability of malfunctions of equipment remains unchanged. The consequences of a steam line break in the AS system affect the MS lines for any unit aligned to the AS system. If two or more units are aligned to supply the AS system, actions simply must be taken on each unit to mitigate the event. Off-site dose consequences from a steam line break in the AS system is bounded by the double ended MS line break. No new failure modes were introduced on the AS supply from any unit. MS-24 and MS-33 still provide the seismic/non-seismic boundary isolation between the MS system and the AS system. Alignment of one main steam line from multiple operating units to supply the auxiliary steam system has been reviewed and determined that no USQs or safety concerns exist. No Technical Specification changes were required by this new operating configuration. UFSAR sections 3.1.4 and 10.3.2 were revised to reflect the new alignment per ONOE-11376

PROCEDURES

DESCRIPTION

SYSTEM: Cranes

Maintenance Procedure MP/0/A/1400/034 was issued to identify and remedy potential corrosion inducing conditions around the containment tendons. It also provides guidance for inspecting the as found condition of tendon components. The reason this activity was not screened from the 50.59 regulation is that the corrosion inhibiting filler grease, named in the UFSAR as Visconorust 2090P, and the new designation of the equivalent grease, 2090P-4, have slightly different chemical and physical properties. Therefore, the facility was changed as described in the UFSAR Section 3.8.1.6.2.6. The UFSAR will be revised accordingly to document the use of Visconorust 2090P-4 as tendon sheath filler grease instead of Visconorust 2090P. Rev 2 of this procedure provided enhanced guidance for use of the mobile Manitowoc Crane during tendon surveillances.

SAFETY EVALUATION SUMMARY

No physical changes to the structure other than the tendon grease substitution resulted from implementation of this procedure. No components important to safety or nuclear safety related were modified. The Design Basis Accidents which produce high internal pressure inside the Reactor Building are the Loss of Coolant Accident (LOCA) and Steam Line Break accident (SLB). The containment tendon system provides the counter-acting force to the high internal pressure. The activities performed by this maintenance procedure will in no way prevent or hinder the tendons from performing their design basis function. The procedure accounts for heavy loads being lifted over safety related equipment. The procedure prohibits the crane boom and any lifted load from passing over the BWST, main steam lines, and valve LP-28 when such lifts would be detrimental to Nuclear Safety. The procedure also prohibits these inspections during an Integrated Leak Rate Test. Therefore, the procedure does not cause, or affect the mitigation of, any accidents or malfunctions of equipment previously evaluated in the SAR. No new accidents or malfunctions are introduced as a result of implementation of this procedure, and there is no reduction in the Margin of Safety as defined in the bases to any Technical Specifications. UFSAR Section 3.8.1.6.2.6 was revised accordingly.

PROCEDURES

DESCRIPTION

SYSTEM: Low Pressure Service Water (LPSW)

The purpose of procedure TN/1/A/3001/00/AM2 is to isolate the LPSW system to tie-in minimum flow piping and valves around each Low Pressure Service Water (LPSW) Pump to existing LPSW piping. The minimum flow lines are needed to assure minimum flow after Engineered Safeguards (ES) signals are removed from valves 1LPSW-4 and 1LPSW-5.

SAFETY EVALUATION SUMMARY

NSM-13001 AM2 along with other LPSW System ensure adequate Net Positive Suction Head is available at the LPSW pumps during all design basis conditions. NSM 13001, Part AM2 was implemented during defueled maintenance for Unit 1. This implementation procedure does not adversely affect any important to safety plant SSCs. The activities performed do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. The LPSW system continues to function as designed during normal and accident conditions, when required. Implementation of this modification involves no USQ's or safety concerns. No Tech Spec changes are required. All installation activities performed under TN/1/A/3001/00/AM2 conform to SAR requirements and descriptions, therefore no UFSAR changes are required.

PROCEDURES

<u>DESCRIPTION</u>

SYSTEM: Low Pressure Service Water (LPSW)

The purpose of TN/1/A/3001/00/CM1 is to implement NSM-13001, Part CM1. This modification separates Unit 1 and 2 LPSW non-essential headers. It installs valve 2LPSW-139 and renumbers valve LPSW-941 to 2LPSW-941, LPSW-139 to 1LPSW-139 and LPSW-940 to 1LPSW-940. In order to install valve 2LPSW-139, a 14" mechanical line stop performed to isolate existing LPSW piping. NSM-13001, Part CM1 is part of the Oconee Service Water upgrade project.

SAFETY EVALUATION SUMMARY

This implementation procedure does not adversely affect any important to safety plant SSCs. The activities performed do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. The LPSW system continues to function as designed during normal and accident conditions, when required. The implementation of this modification involves no USQ's or safety concerns. No Tech Spec changes are required. All installation activities performed under TN/1/A/3001/00/CM1 conform to SAR requirements and descriptions, therefore no UFSAR changes are required for this implementation procedure.

PROCEDURES

DESCRIPTION

SYSTEM: Integrated Control System (ICS)

The ICS Transient Test Procedures (MC#1) TT/1,2, 3/B/0326/001 were performed to test the new ICS response to turbine trips, loss of load, FDWP trips, runback, and RCP trip scenarios. Administrative controls were included in the test procedures to maintain the plant operating conditions within acceptable values.

SAFETY EVALUATION SUMMARY

The new ICS was verified in response to turbine trips, loss of load, FDWP trips, runback, and RCP trip scenarios. This procedure ensures that no on-site or off-site power system transients are induced by keeping the unit's electrical loads powered from the startup transformers as normal. The procedure cautions against tripping the RCP that provides continuous operation of the pressurizer spray system. The impact of the RCP trip test on the thermal transients experienced on the HPI nozzles is below the threshold of system transients described in the SAR. Plant experience has repeatedly shown that RCPs can be tripped at this power level without experiencing a reactor trip or causing other system anomalies. The test imposes very little affect on operator burden after commencement of the transients. The test has no adverse effects on protective or safety related system actuation setpoints or time responses. There are no safety systems being bypassed or defeated other than those normally permitted for normal startup activities. The plant conditions encountered in performing this test have been analyzed in the UFSAR. The turbine trip test is a normal startup test performed below the RPS/ARTS reactor trip setpoint as described in the SAR. The load rejection test is performed below the RPS/ARTS reactor trip setpoint and does not effect the bus alignment of the electrical system as described in SAR. The FDW trip test is performed at a power level well within the limits of single FDW pump operation. Therefore the test does not apply to the loss of feedwater or ATWS transients described in the SAR. The RCP trip test is performed at a power level below the RPS reactor trip setpoints for 3 pump operation. Therefore the test does not apply to the loss of RC flow transient as described in the SAR. There are no USQs or safety concerns involved with performing this test procedure. No Tech Spec or UFSAR changes are required.

PROCEDURES

DESCRIPTION

SYSTEM: Integrated Control System (ICS)

The ICS Loss of Power Procedures TT/1,2,3/B/0326/002 tested the ICS response to loss of ICS Hand Power and to loss of ICS Auto Power. The test procedure is included in the post-modification test plan for NSMs-1,2, 32989.

SAFETY EVALUATION SUMMARY

Administrative controls have been included in the test procedure to maintain the plant operating conditions within acceptable values. The test does not affect any design, material, and construction criteria as required by the SAR. The acceptance criteria were developed from ICS Design Basis Document requirements and from engineering evaluations. The test has little affect on operator burden after commencement of the loss of ICS power tests. The Operations staff is free to operate the plant in accordance with the SAR. The test has no adverse effects on protective or safety related system actuation setpoints or time responses. There are no safety systems being bypassed or defeated. All of the plant conditions encountered in performing this test have been analyzed in the UFSAR. Administrative controls have been written in the test to maintain the operating conditions at acceptable values to prevent unnecessary challenges to safety systems. The loss of ICS power test was written to be performed at 25% reactor power to maximize the steam generator inventory to reduce the likelihood of emergency feedwater actuation while in HAND operation. The 25% power level removes the risk of a reactor trip at the Anticipatory Reactor Trip setpoint of 30% due to a turbine trip. The affected Unit's electrical loads will be powered from the switchyard via CT1,2, or 3 during the test. Therefore, no plant power system transfers would occur in the unlikely event of a turbine or reactor trip. The test has no adverse effects on safety system actuations, setpoints or time responses. There are no unreviewed safety questions involved with performing this procedure. No UFSAR or Technical Specification revisions are required to perform this activity.

PROCEDURES

DESCRIPTION

SYSTEM: SSF

TT/0/A/0600/17, Rev 1 performs a test which operates the SSF ASW pump, the SSF HVAC service water pumps, and the SSF DSW pump so that SSF Service water system performance data can be taken.

SAFETY EVALUATION SUMMARY

The SSF remains operable while this test is performed. If an accident which requires operation of the SSF occurs while performing TT/0/A/0600/17, the procedure provides adequate guidance to return SSF systems to the configuration needed for the SSF to perform its mission. While the SSF ASW system is operating, 1/2/3CCW-268 and 1/2/3CCW-287 remain closed to prevent SSF ASW flow from being directed to an operating unit's S/Gs. In addition, 1/2/3CCW-288 opened and drainage from these valves monitored to verify that there is no significant leakage past 1/2/3CCW-268. The SSF ASW system is operated similarly to the method used to test this system during quarterly IWP testing of the SSF ASW pump. The SSF is not operating outside of its normal system design parameters during this test. Therefore, there no SSF equipment important to safety should be adversely affected during the test. Locations of the test gauges during this test have been evaluated to insure that the SSF Service water system remain seismically qualified. Pump capacities are not be reduced or degraded due to leakage past test gauges during an accident which requires operation of the SSF. This test involves no USQs or safety concerns, and no Technical Specification changes are required. The UFSAR descriptions do not require any changes as a result of this test activity.

PROCEDURES

DESCRIPTION

SYSTEM: Standby Shutdown Facility (SSF)

Temporary test procedure TT/1/A/0400/025 was originated to conduct a test to verify that the Unit 1 SSF Reactor Coolant letdown flow path to the Units 1 & 2 Spent Fuel Pool is unobstructed. During cold shutdown conditions, the Unit 1 SSF Reactor Coolant letdown line is aligned to the Units 1 & 2 Spent Fuel Pool. A pressurizer level change of 3" was timed.

SAFETY EVALUATION SUMMARY

A detailed review of the impacts of this activity on reactivity management, potential for accident initiation, shutdown risk, decay heat removal, and affect on plant parameters and setpoints was conducted with none of these items being adversely impacted. As a result, it was determined that this activity does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The components continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

PROCEDURES

DESCRIPTION

SYSTEM: SSF

TT/2/A/0400/28, "SSF RC Makeup Pump Flow Distribution Test", is performed in response to commitment made by Duke Power to the NRC in a letter dated October 31, 1996. The test was required to be performed on Unit 2 during EOC16. The purpose of the test is to ensure an appropriate flow balance between each of the Reactor Coolant Pump (RCP) seal injection lines. The test aligns the SSF RC Makeup Pump to the RCP seals. The SSF RC Makeup pump is then operated in the injection mode, with flows recorded using an ultrasonic flow device.

SAFETY EVALUATION SUMMARY

Several contingencies for testing windows were evaluated: If testing is not performed during refueling, the SSF RC makeup system can take suction from the SFP via a fuel transfer tube. The required unit status at the start of the test is an RCS temperature ≤ 160 °F. The temperature of the SFP water used for RC pump seal injection is less than 150 °F. Due to the relatively small injection flow rate and the relatively small temperature difference between the water in the SFP and the water in the RCS, the 50 °F/hr RCS cooldown limit is not exceeded due to performing this test. If this test is performed after refueling is complete, the RCS is filled so that pressurizer level is greater than or equal to 140 inches, the 2A and 2B OTSG handhole covers or manways are removed, and the pressurizer is vented. Since the pressurizer is vented and OTSG handhole covers or manways are removed, no LTOP concerns are created by running this procedure. Operations drains water from the RCS per OP/2/A/1102/11, "Controlling Procedure for Cold Shutdown", as needed to avoid overfilling the RCS. If testing is not performed during refueling, some Unit 1 & 2 SFP level decrease occurs if makeup flow is not provided to the SFP. Therefore, makeup to the SFP is provided per OP/1&2/A/1104/06 as needed to maintain the proper SFP level when this test is performed.

The SSF RC Makeup pump discharge pressure is far below the pressure experienced during its design basis accidents. No adverse effects occur to the SSF RCMU Pump, but even so, they do not have an effect on the plant at the specified test conditions. The activities performed do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No unreviewed safety questions exist. No changes are required to the plant technical specifications. No changes are required to the UFSAR.

PROCEDURES

DESCRIPTION

SYSTEM: Nuclear Fuel, Fuel Handling

The evaluation provides a comprehensive review of the evolution for ultrasonic testing (UT) of fuel assemblies and the Fuel Assembly Inspection by UT Inspection that can be performed under PT/0/A/0124/006, “Fuel Assembly Post Irradiation Examinations”.

SAFETY EVALUATION SUMMARY

This evaluation covers the use of all of the tools necessary to accomplish fuel assembly inspection by UT. The methods used to move a fuel assembly for examinations are no different than those discussed in the SAR and therefore do not increase the probability of the fuel handling accident. The equipment used to examine the fuel assemblies is unobtrusive to the fuel pin cladding, therefore, damage to fuel pins is not credible. No criticality margins are reduced and no margins of safety as related to the spent fuel or pool as defined in the bases of the Technical Specifications are affected. No new radiological release pathways or failure modes are created. Based on the subject evaluation the process of performing UT inspections as controlled by PT/0/A/0124/006 involves no safety concerns or USQs. No Technical Specification changes are required as a result of this procedure change. A largely editorial clarification was made to UFSAR Section 4.2.3.1 to denote that based on radiochemistry data indicating potential failed rods, a UT campaign may be scheduled to test suspect fuel assemblies.

PROCEDURES

DESCRIPTION

SYSTEM: Fuel Handling

The purpose of this evaluation is to provide a comprehensive review of the entire fuel assembly Post Irradiation Examination (PIE) process that can be performed under PT/0/A/0750/005 Rev 8, Fuel Assembly Post Irradiation Examinations. This includes the equipment used, and the processes in place to control:

1) water channel measurements
2) fuel rod oxide measurements
3) fuel rod diameter measurements
4) crud sampling
5) fuel assembly length measurements
6) fuel assembly bow measurements
7) spacer grid position measurements
8) shoulder gap measurements
9) guide tube oxide measurements
10) guide tube plug gages
11) spacer grid oxide/width measurements

SAFETY EVALUATION SUMMARY

This evaluation covers the use of all of the tools necessary to accomplish the PIE for inspecting the lead test assemblies, and the interactions that the PIE tools may have with SSC's in the spent fuel pool and the fuel assemblies being examined. The methods used to move a fuel assembly for PIE examinations are no different than those discussed in the SAR and therefore do not increase the probability of a fuel handling accident. The PIE process of a fuel assembly does not involve the use of any equipment in the spent fuel pool in any manner different than those already discussed in the SAR. The equipment used to examine the fuel assemblies is unobtrusive to the fuel pin cladding and the examination heads involve only a few fuel pins at any time. Therefore, damage to fuel pins is not credible. No criticality margins are reduced in the PIE process of a fuel assembly. No margins of safety as related to the spent fuel pool or spent fuel as defined in the bases of the Technical Specifications are affected. Based on the subject evaluation the process of performing specific measurements on the LTAs as controlled by PT/0/A/0750/005, Rev 8 Fuel Assembly Post Irradiation Inspections involves no safety concerns or USQs. No Technical Specification or UFSAR changes are required as a result of this procedure change.

PROCEDURES

DESCRIPTION

SYSTEM: Low Pressure Injection (LPI), Core Flood (CF)

This 10 CFR 50.59 evaluation was performed for Operating Procedures OP/1/A/1104/04, OP/1/A/1103/11, and Temporary Test procedure TT/1/A/150/048 which were utilized to operate the Unit 1 LPI system at drain down conditions while a seat leak was repaired on valve 1CF-14. Valve 1CF-14 is a core flood tank isolation valve. The operating procedure was revised to allow a different LPI System configuration during the valve repair. A temporary test procedure was originated to test run the LPI pumps and take data to assess whether NPSH was adequate under the conditions established by the operating procedures for repair of 1CF-14.

SAFETY EVALUATION SUMMARY

All SLC 16.5.5 requirements were met because both LPI coolers and the associated Low Pressure Service Water supplies remain available. The repair activities do not affect the design basis functions of the involved systems. The activities performed do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No plant SSCs are adversely affected. There is no adverse affect on containment integrity, no new release pathways are created, and no new failure modes are created. These procedures involve no USQ's or safety concerns. No Technical Specification or UFSAR changes are required.

PROCEDURES

DESCRIPTION

SYSTEM: MS and HD

Operating Procedures OP/1,2,3/A/1106/14 provide procedural guidance for operating the Units 1,2, and 3 Moisture Separator Reheaters (MSRH's). The change was necessitated following the implementation of NSMs ON-1,2,32941. This procedure change affects the Main Steam (MS) and Heater Drain (HD) Systems. Procedure steps have been added to address operation of the new control systems for SSRH Steam Admission control via 1,2,3MS-112,173 Moore Controllers; SSRH Drain Feedforward control via 1,2,3HD-92,95 Moore Controllers; and MSRH Drain Feedforward and Recirc control via 1,2,3HD-37,52 Moore Controllers. Further, the operation of warming bypass valves around 1,2,3HD-37,52 is addressed along with the operation of a SSRH Feedforward Header Low Point Drain Valve. Also, the SSRH Non-Condensible Vent throttle valves have been conservatively specified full. A new SSRH Steam Admission Control Curve has been added to verify proper operation of 1,2,3MS-112,173 when in Auto and to support Manual operation of these valves. The MSRH, FSRH and SSRH Drain Tanks are operated on level control, by way of positioning the Shell Warm switch to "Shell Warm", for all operational modes. Provisions for removing both the steam supply and drains of FSRH and SSRH while on-line have been retained. Provisions for removing just the drain portion of FSRH and SSRH and for restoration of FSRH and SSRH following isolation while on-line have been removed from this procedure. The MS System change is to modify the operation of MS admission to the MSRH SSRHs to provide a more accurate automatic control of the startup/operating relationship of SSRH Tube Supply Pressure versus Turbine Reheat Pressure and also control within all SSRH Temperature and Heatup Rate Limits. The HD System change is to modify the operation of the feedforward control of MSRH, FSRH and SSRH drains to their respective Heaters and Drain Tanks in a manner that prevents HD back-flow and allows for HD pre-warming.

SAFETY EVALUATION SUMMARY

These actions prevent exceeding startup/operating limits on the SSRH's, thus prolonging their life span, minimize water-hammers in the MSRH, FSRH and SSRH drain piping, and alleviate operator burden of manual operation of the SSRH startup function. These activities do not affect the design basis of the units since the changes of significance only affect system components downstream of 1,2,3MS-76,79 and does not affect 1,2,3MS-76,79 except for their time-sequence of operation. The procedure change does not create any adverse effects concerning the operation of the modified systems as evaluated in the SAR. The procedure change does not change the operation of the FSRHs, SSRHs, MSRHs, drain system, and affected piping as currently described in the SAR. This change does not create any conditions or events which lead to accidents previously evaluated in the SAR. The affected equipment is non-radioactive and does not mitigate any accidents. There is no adverse affect on containment integrity and no new release paths are created. There are no adverse effects on the Appendix R fire analysis. The affected MS and HD equipment is non-QA, with the exception of 1,2,3MS-76,79. The change does not add any new safety/non-safety interfaces. The potential for water hammer can still occur if an input signal to the 1,2,3HD-92,95 Controller fails or if improper Manual Operator action of the Moore Controller occurs. But the control circuitry is not safety related and is not required to be designed to the single failure criterion. In addition, if a water hammer occurred and ruptured the piping, no QA-1 or equipment important to safety would be adversely affected. These procedures involve no USQ's or safety concerns. No Technical Specification or UFSAR changes are required.

PROCEDURES

DESCRIPTION

SYSTEM: Main Turbine (MT)

Procedure TN/1/B/8572/MM/01E Rev 0, 1 is the implementation procedure for minor modification OE-8572 to delete the thrust bearing wear detector. This change adds notes to section 4.2 that gives the technicians flexibility in performing the procedure when material and resources are not available to complete steps within section 4.2. It adds a note that allows voltage to be removed if it is present prior to performing step 4.2.25 as a matter of personnel safety. It deletes location designators from steps 4.4.7 through 4.4.38 and rewires two relay termination points to facilitate having less than three termination points on a relay terminal per VN-8572A. Steps were added to label new isolation valves and pressure switches in section 4.5. This ensures the instrumentation is labelled. Step 4.6.10 was deleted because OAC modification ON-12962 revised the computer. Step 4.6.1 Events recorder procedure was revised from IP/1,2/B/0125/006B to IP/0/B/0125/006A due to the procedure rearrangement. Rev 1 simply allows the technicians some flexibility in performing the procedure.

SAFETY EVALUATION SUMMARY

With the exception of minor wiring changes required by VN-8572A, these changes are administrative in nature and are for guidance purposes. This work is done while the unit is shutdown and is performed one time only. This safety evaluation addresses procedure implementation of the minor modification with the minor changes described above. The activities performed do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways, or failure modes are created. No plant SSCs are adversely affected. No safety concerns or unreviewed safety questions exist. No changes required to the plant technical specifications. No changes are required to the UFSAR.

PROCEDURES

DESCRIPTION

SYSTEM: High Pressure Injection (HPI)

Temporary test procedure TT/3/A/0251/065 was originated to perform testing on the replaced motor for the 3B HPI pump. Normally, the testing is done with the unit at cold shutdown conditions. However, the opportunity to replace the motor and test the pump occurred during an unexpected forced outage. During this forced outage, the unit remained above cold shutdown. Therefore, the evaluation was to determine if any unreviewed safety questions existed with a change to this test method to allow the HPI pump flow delta-P test to be done above cold shutdown conditions.

SAFETY EVALUATION SUMMARY

The implementation of this procedure provided data which validated proper operation of new equipment installed in the High Pressure Injection System. A detailed review of the impact of this testing on the probability of LOCAs, and impacts on reactivity management was conducted and it was concluded that these items would not be adversely impacted by the test. Compensatory actions, such as increased monitoring of the source range nuclear instruments, borated water storage tank level, and pressurizer level, were implemented. This activity does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The components continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

PROCEDURES

DESCRIPTION

SYSTEM: Heater Drain (HD)

Procedures TT/2, 3/B/0271/012 provide for testing and tuning for verification of the acceptable control behavior of Moore Controllers 2, 3MS SS0010, 2,3MS SS0011, 2,3HD SS0020, 2,3HD SS0021, 2,3HD SS0022 and 2,3HD SS0023 on Units 2 and 3. Testing and tuning will be accomplished by the procedural direction in TT/2,3/B/0271/012 that provides for additional control manipulations and monitoring of Main Steam and HD System controllers during certain critical times of MSRH System startup and operation by way of visual observation and note-taking. Implementation of NSMs-ON-2,32941, along with appropriate changes to Operating Procedures decrease the likelihood of MSRH heater drain system water hammer and thermal stress to the SSRHs and Low Pressure Turbines.

SAFETY EVALUATION SUMMARY

The activities addressed by these procedures involve small changes to plant equipment that is well within normal operational boundaries and only affect system components downstream of main steam to MSR isolation valves 2,3MS-76,79. The procedures do not create any adverse effects concerning the operation of the modified systems as evaluated in the SAR. The procedures do not change the operation of the FSRHs, SSRHs, MSRHs, drain system, and affected piping as currently described in the SAR. The procedures do not create any conditions or events which lead to accidents previously evaluated in the SAR. The affected equipment is non-radioactive and does not mitigate any accidents. There is no adverse affect on containment integrity and no new radiological realease pathways are created. There are no adverse effects on the Appendix R fire analysis. The affected equipment is non-QA condition. This procedure does not add any new safety/non-safety interfaces. The potential for water hammer could still occur if an input signal to the HD-92,95 Controller fails or if improper Manual Operator action of the Moore Controller occurs. But, the control circuitry is not safety related and is not required to be designed to the single failure criterion. In addition, if a water hammer occurred and ruptured the piping, no QA-1 or important to safety equipment would be adversely affected. This procedure involves no USQ's or safety concerns. No Technical Specification or UFSAR changes are required as a result of these procedures.

PROCEDURES

DESCRIPTION

SYSTEM: Heater Drain System

This temporary test procedure (TT/2/B/0271/013) was originated to perform three major activities, following installation of modifications to prevent waterhammers in the heater drain system:

- Provides procedural guidance and Engineering oversight with documentation of the results for operation of two heater drain valves with the new controllers during steady state operation.
- Document stem movement of these two valves during an increase in demand on the controllers for these valves.
- Document demand on these two valves during during closure of their condenser dump valves.

SAFETY EVALUATION SUMMARY

The implementation of this procedure provides data which validates proper operation of equipment designed and installed in the heater drain system to prevent waterhammers. The heater drain system is not safety related, and performs no accident mitigation functions. The heater drain system are made more resistant to waterhammers and resulting steam line breaks by virtue of this testing. This testing does not increase the risk of waterhammers occurring in the heater drain lines. As a result, this activity does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The components will continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

PROCEDURES

DESCRIPTION

SYSTEM: LPSW

This evaluation is for procedure TT/3/A/0251/63 "LPSW Pump Loop Simulation." The purpose of this procedure is to gather valuable input on system response, and the potential for water hammer, associated with a low pressure service water (LPSW) pump being shutoff and then restarted.

SAFETY EVALUATION SUMMARY

This test was performed with Unit 3 in a refueling shutdown (with fuel in the core). Under these conditions, there is no risk of unit trip or challenge to the Reactor Protective System (RPS), and most UFSAR analyzed accident scenarios do not apply. Strictly per the Tech Specs; Emergency Core Cooling Systems (ECCS) are not required with RCS pressure/temperature < 350 psi and 250°F, respectively. However, the decay heat removal systems remain operable during the test. The detailed procedure requirements for this test, including proper initial conditions and appropriate compensatory actions in place, provide assurance there are no safety concerns involved. The test activities performed do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways are created. There is no adverse impact on the function of any safety system, no increase in the probability or consequences of any previously analyzed accidents, and no new credible failure modes associated with this test. This test involves no USQs. No UFSAR or Technical Specification changes are necessary.

PROCEDURES

DESCRIPTION

SYSTEM: Standby Shutdown Facility (SSF)

Maintenance procedure MP/0/A/1300/059 Revision 8 was originated to change this procedure as follows:

- Change the procedure from a non safety related procedure to a safety related procedure
- Provide additional cautions and warnings during operation of CCW pumps, SSF submersible pump, and the SSF control room.

SAFETY EVALUATION SUMMARY

All of the revisions to this procedure are conservative in nature, in that they require additional precautionary measures, additional quality assurance measures, and additional information. As a result, it was determined this activity does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The components continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

PROCEDURES

DESCRIPTION

SYSTEM: High Pressure Service Water (HPSW)

Maintenance procedure MP/0/A/1800/021 was originated to document performance of a civil inspection of the HPSW System elevated water storage tank and to isolate and dewater the tank for the civil inspection and for repair, if needed.

SAFETY EVALUATION SUMMARY

The HPSW System supports operation of the Emergency Condenser Circulating Water (ECCW) System during a loss of offsite power event. To assure that this procedure, and the resultant draining of the elevated water storage tank did not impact the operation of the ECCW System, the procedure was performed in the spring during a condition of high lake level. The lake level was high enough to support gravity flow through the ECCW System so that the HPSW System did not affect operability of the ECCW System. As a result, it was determined that this activity does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The components will continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

PROCEDURES

DESCRIPTION

SYSTEM: Control Rod Drives

Restricted change 19R was made to the CRD Shim Drive - Leadscrew Uncoupling Procedure (MP/1&2/A/11405/001) to allow the checking of Unit 1 control rod group 5 rod 7, at core location O-7, for freedom of movement. The plant conditions under which this evolution are performed are: RCS at cold shutdown with a reduction of RCS inventory. Inventory is at a level below the CRDM flange to facilitate CRDM removal. This procedure provides guidelines for the uncoupling and parking CRDM type 'A' leadscrews. It is typically used during a shutdown to uncouple the leadscrews from the control rods prior to head removal. The procedure allows two methods of uncoupling and removing the leadscrew: preferred and optional. Under the preferred method, the leadscrew is uncoupled and then it is raised to the parking position with a lifting tool attached to a chain hoist with a spring scale. The scale provides an indication of the force necessary to lift the lead screw to alert the chain hoist operator of any abnormal conditions, as well as for the fine control of lead screw movement. Under this restricted change, the same process is used, except that the control rod is not uncoupled and the procedurally allowed force is increased to account for the expected weight increase from the presence of the control rod. This restricted procedure change allows lifting of the coupled leadscrew and control rod to a prescribed height, with a chain hoist and a prescribed force limit. The differences between this evolution and the normal leadscrew removal result from the attachment of the control rod to the leadscrew. The process used is basically the same, with the same types of controls. Several additional controls and actions are being added to the procedure for this evolution.

SAFETY EVALUATION SUMMARY

A Senior Reactor Operator (SRO), in communication with the control room was stationed in the Reactor Building to oversee the evolution. Reactor Coolant System conditions were such that a 1% shutdown condition was maintained with the worst case control rod and the control rod manipulated being fully withdrawn. With only one control rod affected by this evolution, the core will maintain a 1% shutdown condition even if the rod were postulated to be fully withdrawn. Thus, there are no criticality concerns. The spring scale limit was revised to only allow 35 pounds of force to be placed on the control rod. This force is sufficient to lift and move the control rod, but well below the point of damaging the control rod if it is stuck. A fall from one foot with no water in the CRDM would pose no additional consequences for the fuel assembly or control rod than any normal at power trip. During the evolution controlled by this procedure change, only one rod will be moved. The rod ejection accident involves a near instantaneous ejection of a control rod from the core. This rate of rod movement is not possible with the manually driven chain hoist. The only new pieces of equipment being introduced by this evolution, with respect to control rod movement are the handling tool and the chain hoist. These are standard leadscrew movement equipment. Any failure of these pieces would cause either the control rod to be stuck in place or dropped to the bottom of the core. The possibility of the manually driven chain hoist, independently of human interaction, providing an uncontrolled rod withdrawal is not considered credible. In summary, issues involving criticality, fuel assembly damage, component damage, control rod misalignment, and uncontrolled control rod withdrawal have been examined and found to be not credible scenarios. Since no fuel damage occurs as a result of this evolution, there will be no releases and there is no effect on the consequences of any SAR analyzed accident. There are no USQs. No Tech Spec or UFSAR changes are required.

PROCEDURES

DESCRIPTION

SYSTEM: Nuclear Fuel and Fuel Handling

The purpose of this evaluation is to provide a comprehensive review of the entire fuel work evolution that is performed under PT/0/A/0750/016 Rev 15, "Fuel Assembly Reconstitution and Recaging". When breached fuel pin(s) have been identified, an assessment of the value of the fuel assembly is made to determine if the fuel assembly should be repaired or permanently discharged. If the failure mechanism of the fuel assembly is not understood then the fuel assembly can be reconstituted to allow detailed examination of the failed fuel pins. Once examinations are complete the fuel pins may be reinserted in the fuel assembly, or placed in the failed rod storage container for permanent storage in the spent fuel pool. If the fuel assembly is deemed valuable enough to repair then the fuel assembly can be reconstituted or recaged. Reconstitution is the removal of the failed fuel pin(s) and their replacement with Natural U pin(s), or Stainless Steel pin(s). For cases where it is determined that the fuel pin is to badly degraded to be removed, the fuel pin pulls apart while being extracted, or the structure of the fuel assembly has been damaged to the point that it can no longer be used, the fuel assembly must be recaged. An empty fuel assembly cage is placed adjacent to the damaged fuel assembly, and <u>all</u> sound fuel pins are transferred one at a time from the damaged cage to the new cage. Natural Uranium (U) pin(s) are inserted into the damaged fuel pin(s) location(s).

SAFETY EVALUATION SUMMARY

This evaluation covers the use of all of the tools necessary to accomplish any of the mentioned tasks, and the interactions that those tools might have with SSC's in the spent fuel pool and the fuel assemblies being repaired. This procedure does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created, and no SSCs are degraded. The process of reconstituting or recaging a fuel assembly as described in this evaluation, and as controlled by PT/0/A/0750/016 Fuel Assembly Reconstitution and Recaging Revision 15 involves no safety concerns or USQs. No Technical Specification changes are required to perform this evolution. No UFSAR sections require revision due to this evolution.

PROCEDURES

DESCRIPTION

SYSTEM: Feedwater

This evaluation is for procedure TT/3/A/0251/062, "Unit 3 Final Feedwater Check Valve Testing". The final Feedwater (FFW) check valves are containment isolation valves. Therefore, in accordance with ASME Section XI, the valves need to be tested.

SAFETY EVALUATION SUMMARY

To meet containment requirements, the valve must close. A leak rate test is not required. This TT procedure will direct flow through the FFW check valves and then isolate the flow. When flow is isolated, each check valve was acoustically monitored. Acoustical monitoring verifies each valve's disc contacts its seat. LPI is required for core cooling while performing this TT. Flow will be directed through the check valves while unit 3 is operating in FDW Cleanup/Heatup mode per the Operating procedure. If necessary to control S/G level, the S/G Recirculation system may be used per Operating procedures. This procedure performs SAR required testing and does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created, and no SSCs are degraded. There are no USQs or safety concerns with this activity. No Tech Spec or UFSAR changes are required.

PROCEDURES

DESCRIPTION

SYSTEM: Reactor Building Cooling Units (RBCU)

Procedures TT/1,2,3/A/0160/10,11,17,19 start a third RBCU fan in high speed while the other two RBCU fans are also operating in high speed. In the past, operating procedures have prevented this alignment because of concerns with overpressurizing RBCU duct and history of the third fan failing to start due to high speed thermal overloads tripping on excessive startup current. Once the third RBCU fan is started successfully then LPSW will be valved into the cooler and all three trains will remain in operation to provide RB cooling. This lineup is necessary because we are currently operating with no LPSW being supplied to the RB auxiliary coolers due to waterhammer concerns.

SAFETY EVALUATION SUMMARY

The test activities performed do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways are created. There is no adverse impact on the function of any safety system, no increase in the probability or consequences of any previously analyzed accidents, and no new credible failure modes associated with this test. It has been determined that operating all three RBCU fans in HIGH does not involve any safety concerns or USQs. No Technical Specification changes are necessary. Even though the situation which necessitated this alignment is only temporary (see related Operability Safety Evaluation for PIPs 97-240,310, and 311), UFSAR changes to Sections 6.2.2.2, 9.2.2.2.3, and 9.4.6.2 are necessary to show three RBCUs operating in high speed as an acceptable alignment.

PROCEDURES

DESCRIPTION

SYSTEM: External Grid Protective System (EGTPS).

Procedure TT/0/A/0610/028 will operate the relays that trip the switchyard breakers in both frequency detection channels of External Grid Protective System (EGTPS). The output of these relays is blocked by opening terminal block sliding links. No switchyard breakers are operated by the proformance this procedure. Voltage is verified across the open links which demonstrates proper circuit configuration. The frequency detection channels are not operated simultaneously. The frequency channels for the EGTPS are initiated using the test push buttons to simulate an under frequency condition on the Duke Grid. The output for the tripping relays is blocked so that no breaker operation occurrs.

SAFETY EVALUATION SUMMARY

As stated in the UFSAR section 8.3.1.1.1, the EGTPS consists of two reduntant channels for both voltage and frequency. The voltage channels are not affected by this procedure. Both voltage channels remain operable throughout the performance of this procedure. The degraded grid system initiates isolation of the Keowee overhead power path through the voltage channels of the EGTPS. Therefore, both channels of the degraded grid system remain operable during performance of the procedure. Should a Design Basis Event (DBE) occur during the performance of this procedure, the overhead power path would be separated from the Duke grid and a Keowee Hydro Unit would be emergency started by the degraded grid system as discribed in the UFSAR Section 8.2.1.3.1. This procedure simulates an under frequency condition on the Duke grid which initiates a frequency channel of the EGTPS. The frequency channel of the EGTPS, initiated by a simulated input, operates as described in the Design Basis Document (DBD). The output of the frequency channel is blocked to prevent the separation of the overhead power path. The procedure doesl not affect the ability of the Emergency Power System to function as described in the UFSAR. During a DBE the overhead power path separates from the Duke grid and power is provided to the Startup Transformers by the Keowee Unit aligned to the overhead power path. Should a DBE occur during the performance of this procedure, the overhead power path would be separated and the Keowee Units started through the redundent voltage channels of the EGTPS by the Degraded Grid System. This procedure does not affect the ability of the switchyard breakers, the voltage channels of the EGTPS, or Keowee Emergency Start System to function as described in the UFSAR. This procedure will not affect the operating parameters or set point of any SSC. Therefore, no possibility or consquence of an accident or equipment failure is increased nor is the margin of safety decreased. No USQs or safety concerns result from this procedure. No Tech Spec or UFSAR changes are required.

PROCEDURES

DESCRIPTION

SYSTEM: Low Pressure Service Water (LPSW)

Implementation procedure TN/1/A/2977/AM1 was originated to provide guidance and documentation in the implementation of a modification on Unit 1. This modification replaced the Low Pressure Injection cooler outlet valves, LPSW-4 and -5, Reactor Coolant Pump inlet isolation valve LPSW-6, and Reactor Coolant Pump outlet isolation valve LPSW-15 with new valves of a different material. In addition, check valves LPSW-75 & 76 were also removed. Vent and drain lines were also added to the LPSW piping to facilitate testing and draining of the system.

SAFETY EVALUATION SUMMARY

The changes associated with the modification, NSM-12977, have been evaluated under a separate 10CFR50.59 evaluation. The implementation procedure is a conservative process used to document completion of the modification. Therefore, this activity is a conservative documentation measure only. As a result, it was determined that this activity does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The components will continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

PROCEDURES

DESCRIPTION

SYSTEM: LPSW

This procedure TN/2/A/3001/00/DL1 provides instructions and documentation for the Electrical portion of the removal of Limitorque Control Valves 2LPSW-4 and 2LPSW-5 (LPI Cooler Shell Outlet Valves) from the Engineered Safeguards (ES) System. These valves are EQ (Environmentally Qualified) QA1 valves. NSM ON-23001/00, Part DL1 installs new cables and removes the Engineered Safeguards signal from valves 2LPSW-4 and 2LPSW-5 (Low Pressure Injection Cooler Shell Outlet Valves). Cables and conductors will be disconnected and spared for future use in the Engineered Safeguards Cabinets affected by this modification. Affected RZ Modules have cables disconnected, leaving connectors attached to cables for future use. Labeling on the affected RZ Modules is removed. The OAC is affected by having computer points for valves 2LPSW-4 and 2LPSW-5 removed and revised. All functional verifications, cycling of valves to test indicating lights and switches, verifying operability of computer points, and verification of ES signal removal from valves 2LPSW-4 and 2LPSW-5 are performed per this procedure.

SAFETY EVALUATION

Valves 2LPSW-4 and 2LPSW-5 are active valves which throttle LPSW flow through the LPI & Decay Heat Coolers when valves 2LPSW-251 and 2LPSW-252 are not available. They are normally closed (unless the LPI Coolers are in service) and open upon actuation of ES Channel 3 & 4. Per this procedure, ES actuation is removed for valves 2LPSW-4 and 2LPSW-5 and the valves now open/operate per Control Switches S800 and S801. These valves are tested and functionally verified for indicating lights and computer points per this procedure. The modification was installed during Unit 2EOC16 Refueling Outage. Coordination of Electrical and MOV groups is essential for installation of this modification. When the Reactor Coolant System WITH FUEL IN CORE is in a condition with pressure equal to or greater than 350 PSIG or temperature of equal to or greater than 250 degrees F, TWO independent LPI trains SHALL BE operable. SLC 16.5.3, 4, & 5 require that TWO LPI trains be operable during various modes of Shutdown. Therefore, work on both LPI and Decay Heat Removal Train "A" (2LPSW-4) AND LPI and Decay Heat Removal Train "B" (2LPSW-5) SHALL ONLY OCCUR DURING Defueled Maintenance. Per SLC 16.6.6, during Fuel Loading and Refueling operations, at least ONE LPI Pump Cooler SHALL BE operable. Therefore, work on LPI and Decay Heat Removal Train "A" (2LPSW-4) OR LPI and Decay Heat Removal Train "B" (2LPSW-5) MAY BE performed during defueling or refueling. Since no operating plant systems are impacted during the performance of this procedure, this work does not affect the plant in any manner as stated in the UFSAR. No USQs or safety concerns result from this procedure. No Tech Spec or UFSAR changes are required. The performance of this procedure does not impede any plant operating system's ability to respond to an accident scenario.

PROCEDURES

DESCRIPTION

SYSTEM: ESV

The implementation procedure TN/5/A/3000/CS1-01 installs a reinforced concrete trench that is used for routing the future Essential Siphon Vacuum system's electrical and instrument cables. The new concrete cable trench runs from the existing Rad-waste trench to the new ESV building.

SAFETY EVALUATION SUMMARY

This new trench is installed around the south end of the station's yard in and adjacent to other operating systems. Adequate instructions, information, cautions, and warnings about the adjacent systems are provided to insure their integrity during the installation. This implementation procedure creates no adverse affects on these adjacent systems. The implementation activities performed do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways are created. There is no adverse impact on the function of any safety system, no increase in the probability or consequences of any previously analyzed accidents, and no new credible failure modes associated with this test. This modification involves no USQ's or safety concerns. No Technical Specification changes are required. The installation activities do not require a change to the UFSAR.

PROCEDURES

DESCRIPTION

SYSTEM: Essential Siphon Vacuum (ESV)

The implementation procedure TN/5/A/3000/CS1-02 constructs the northern closure section of reinforced concrete trench is used for routing future Essential Siphon Vacuum piping and electrical conductors. The piping starts in the plant yard continues up to the crest of the intake dike and parallels the dike's crest to each units CCW pipe manways.

SAFETY EVALUATION SUMMARY

This new trench is being installed in the plant yard and adjacent to other operating systems. Adequate instructions, information, cautions, and warnings about the adjacent systems, are provided to insure their integrity during the installation. The modification creates no adverse effects on adjacent systems. The implementation activities performed do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways are created. There is no adverse impact on the function of any safety system, no increase in the probability or consequences of any previously analyzed accidents, and no new credible failure modes associated with this test. This modification involves no USQ's or safety concerns. No Technical Specification changes are required. The installation activities do not dictate a change to the UFSAR.

PROCEDURES

DESCRIPTION

SYSTEM: HPI, RCS

TN/2/A/10361/001 is the implementation procedure for Minor Modification ONOE-10361. The Unit 2 High Pressure Injection (HPI) System developed a leak at a weld joint between valve 2HP-127 and the HPI nozzles going into the Reactor Coolant System (RCS). See Minor Modification ONOE-10361 description for more details.

SAFETY EVALUATION SUMMARY

The procedure addressed by this safety evaluation consists of largely routine maintenance activities (piping replacment) performed under special precautions. The implementation activities performed do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways are created. There is no adverse impact on the function of any safety system, no increase in the probability or consequences of any previously analyzed accidents, and no new credible failure modes associated with this procedure. This modification involves no USQ's or safety concerns. No Technical Specification changes are required. These implementation activities do not require a UFSAR change.

V. OPERABILITY EVALUATIONS

DESCRIPTION

SYSTEM: Reactor Building Auxiliary Cooling Units (RBCU)

The activity evaluated is an operable, but degraded, condition on the Unit's 1 , 2, and 3 Reactor Building (RB) Cooling System including its low pressure service water (LPSW) supply. It was determined that the potential for water hammer may exist in the RB Auxiliary Cooling Units (ACU) cooling water piping under certain accident conditions. PIPs 97-240, 310, and 311 describe the problem in detail. The short term resolution is to secure low pressure service water (LPSW) flow to the ACUs and drain the associated piping, thereby eliminating the water hammer concern.

SAFETY EVALUATION SUMMARY

While the ACUs are not safety related, their loss results in higher RB temperatures, i.e. a degraded condition. However, the RBCUs and the RB Spray System continues to perform their normal operating and accident mitigative functions. The operability evaluation contained within the respective PIPs set maximum limits of operation of 170°F for building average temperature based on a maximum allowable dome temperature of 180 °F. These limits cover both accident analysis assumptions, localized environmental (EQ) concerns, and RB structural concerns. As long as these limits are satisfied, all accident analyses assumptions will remain valid, there are no equipment environmental (EQ) concerns, and no degradation of concrete. This change does not increase the possibility or consequences of any SAR evaluated accidents or create any new accidents or failure modes. No plant safety limits are impacted. No new radiological release pathways or failure modes are created. There are no USQs or safety concerns. No Technical Specification changes are required. Even though this special plant configuration condition is only temporary, UFSAR Table 6-22 was updated to reflect the new initial ambient containment temperature assumption of 170 vs 125°F and Figure 6-43 was revised to reflect the higher peak temperature after a MSLB accident.

OPERABILITY EVALUATIONS

DESCRIPTION

SYSTEM: Reactor Vessel

During inspections of the lower reactor vessel internals, a "loose" component was found (PIP 1-O97-3976). This component was identified to be a reactor vessel guide block, which is an irregular solid stainless steel component with overall dimensions of about 3" x 5" x 6.5". The block was found to be trapped between the lower rib section assembly and the incore guide support plate about 2 feet below the bottom of the reactor core, and could not be retrieved. The loose guide block was evaluated when it was discovered missing in 1981. The new operability evaluation determined that the system remains operable with the guide block in place, with no compensatory actions.

SAFETY EVALUATION SUMMARY

The guide block does not pose a credible safety risk, does not involve any licensing commitments or requirements, and has no adverse impact on any equipment important to safety. The guide block does not pose a credible risk to the fuel, fuel cladding, RCS, or containment integrity. None of the accidents analyzed in the SAR could be caused by the guide block.. The block does not cause a credible increase in the probability of a LOCA. The guide block does not pose a credible risk to the reactor fuel, control rods, steam generators, reactor coolant pumps, reactor vessel internals, or RCS pressure boundary. No equipment important to safety could be adversely affected by the guide block. The guide block has no effect on offsite dose following an accident. The guide block does not pose a credible risk to any component used in mitigation of an accident. No credible mechanism for changing the consequences of an accident was identified. No new accident types or failure modes were identified which are different than evaluated in the SAR. The guide block does not adversely affect any plant safety limits, set points, or design parameters. There are no USQs or safety concerns. No Technical Specification changes are required. There are several figures in the UFSAR which show the reactor vessel internals, but none of them include the guide blocks. However, none of the discussion in the UFSAR is detailed enough to warrant inclusion of guide block descriptions. Therefore, no UFSAR changes are necessary due to the guide block.

OPERABILITY EVALUATIONS

DESCRIPTION

SYSTEM: Fire Protection System

The Fire Protection System for Unit 3 is considered degraded because a wall was installed in the Unit 3 computer room which resulted in two separate rooms. One room now has plant fire detection and the other room has a battery operated smoke detector. The battery operated smoke detector was installed as a compensatory measure until plant fire protection can be extended into this room.

SAFETY EVALUATION SUMMARY

Installation of a battery operated smoke detector is an adequate compensatory measure, in lieu of not having plant fire protection, to assure that a fire is detected and responded to during its incipient stage. Early mitigation of a fire in this area prevents an evacuation of the control room or loss of the operating aid computer by fire. The evaluation concluded that operators can adequately respond to the battery operated smoke detector. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components continue to perform its design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

OPERABILITY EVALUATIONS

DESCRIPTION

SYSTEM: High Pressure Injection (HPI)

The Unit 1 HPI pump recirculation orifices were discovered to be internally disfigured in the spacer areas based on a review of radiographic examinations. This problem resulted in potential concerns with pressure boundary integrity, loss of flow due to deformations blocking flow, and passage of excessive flow. Compensatory actions were taken. This 50.59 evaluated those compensatory actions. The compensatory actions were:

- Confinement of Unit 1 to a very limited operating period
- Frequent periodic ultrasonic testing and review of test results for flow through orifices
- Frequent periodic radiographic testing and review of test results for orifice geometry and condition
- Minimize running HPI pump with bad orifice
- Engineering involvement on adverse radiographic results

SAFETY EVALUATION SUMMARY

The implementation of these compensatory actions ensured that all concerns with pressure boundary integrity degradation, loss of flow due to deformations blocking flow, and passage of excessive flow, were adequately addressed. As a result, this change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. These components continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

V. SELECTED LICENSEE COMMITMENTS

DESCRIPTION

SYSTEM: Reactor Building Tendons

SLCs 16.6.2 was revised to incorporate the minimum lift-off and prescribed lower limits for the vertical and dome tendons.

SAFETY EVALUATION SUMMARY

The tendons do not initiate any SAR described accidents. This change does not increase the possibility or consequences of any SAR evaluated accidents or create any new accidents or failure modes. No plant safety limits are impacted. No new radiological release pathways or failure modes are created. There are no USQs or safety concerns. No Technical Specification or UFSAR changes are required. SLC 16.6.2 was revised accordingly.

DESCRIPTION

SYSTEM: Keowee Hydro

SLC 16.8.4 addresses Keowee operational restrictions. This SLC contains surveillances for verifying operability of the emergency functions during periods of commercial operation. The surveillances were removed because they are now located in Tech Specs 3.7.1k, 4.6.13, and 14 with descriptions in the bases.

SAFETY EVALUATION SUMMARY

The removal of redundant surveillances from the SLCs does not initiate any SAR described accidents. This change does not increase the possibility or consequences of any SAR evaluated accidents or create any new accidents or failure modes. No plant safety limits are impacted. No new radiological release pathways are created. There are no USQs or safety concerns. No Technical Specification or UFSAR changes are required. SLC 16.8.4 was revised accordingly.

SELECTED LICENSEE COMMITMENTS

DESCRIPTION

SYSTEM: High Pressure Service Water

The primary purpose of this revision to SLCs 16.9.7 and 16.9.8 is to incorporate changes that result from the Oconee Service Water (OSW) System Upgrade modifications. The equipment installed by OSW project modifications was placed in service for Unit 2 late in the End-of-Cycle 16 refueling outage. A Technical Specification amendment was reviewed and approved by the NRC staff and was implemented prior to placing the upgraded Service Water System equipment in service for Unit 2. These revised SLCs were implemented simultaneously with the approved Technical Specification amendment.

SLC 16.9.7 was revised to reflect the new licensing basis for the Unit 2 Service Water System, while retaining the existing licensing basis for the Units 1 and 3 Service Water Systems. In addition, for human performance improvements, SLC 16.9.7 was converted into the NUREG-1430 Improved Standardized Technical Specification format, and the Bases were clarified and better organized. SLC 16.9.8 was also revised to exclude consideration of Unit 2, since the High Pressure Service Water (HPSW) System was no longer credited for the safe shutdown function support for the upgraded Unit 2 Service Water System.

SAFETY EVALUATION SUMMARY

The revisions to SLCs 16.9.7 and 16.9.8 were; 1) editorial changes, 2) clarifications to the Bases which enhance an understanding of the existing licensing basis, 3) format conversion in accordance with the guidance of NUREG-1430 which does not change the technical intent, or 4) technical changes which have already been evaluated by the NRC. All changes in design and construction (via modifications), and operation (via procedural changes) were evaluated under separate 50.59 evaluations, or were evaluated by the NRC. Therefore, the revisions to these SLCs constituted a documentation activity which did not affect operation, testing, or design of the plant beyond that which was already evaluated under other 50.59 evaluations or by the NRC staff's review. This documentation activity did not change existing system design, construction, or operation.

The revision to these SLCs did not result in any plant modifications, procedure changes, or other activities which could have resulted in an unreviewed safety question. Technical Specification changes were required, and the NRC staff review of these changes was completed prior to implementation of this SLC. SAR changes are required per the proposed ECCW Technical Specification amendment submittal dated August 28, 1997. The SLC is part of the SAR (specifically Chapter 16 of the UFSAR) and was revised accordingly.

DESCRIPTION

SYSTEM: CRVS, WC

A section to the Selected Licensing Commitments (SLC) was added to control maintenance and repair activities on the Control Room Ventilation System (CRVS) and the Chilled Water System (WC) which serves CRVS. Section 3.11.4 of the UFSAR states that redundant air conditioning and ventilation equipment is provided to assure that no single failure of an active component within these systems (CRVS and WC) prevents proper control area environmental control.

SAFETY EVALUATION SUMMARY

Presently no clear guidance is provided to Operations and Work Control when taking components/equipment out of service on these systems for maintenance/repair activities. This SLC requires that an LCO be entered whenever activities place CRVS or WC in a position that either system is not able to withstand a single active failure and still provide adequate temperature control to the control area (control room, cable room, electrical equipment room). This SLC adds an LCO time which does not presently exist for activities on these systems. This change is in the conservative direction and does not cause, or adversely affect the mitigation of, any prevously analyzed SAR accidents. No new radiological release pathways or failure modes are created, and no SSCs are degraded. This SLC change involves no USQs or safety concerns and no technical specification changes are required. The SLC is part of the SAR (specifically Chapter 16 of the UFSAR) and was revised accordingly.

SELECTED LICENSEE COMMITMENTS

DESCRIPTION

SYSTEM: CCW

This change to Selected Licensee Commitment (SLC) 16.9.11 provides additional flexibility to perform maintenance or modifications on certain CCW system equipment without compromising the original intent of the SLC. The lack of flexibility to perform maintenance is discussed in PIP 1-O97-2904. In addition, several editorial changes are proposed to clarify the intent of several provisions of the SLC, to clarify the bases, and to update the references. The Oconee UFSAR Section 3.4.1.1.1 describes the flood protection measures for the Turbine Building and Auxiliary Building. These measures are the basis for the requirements in SLC 16.9.11. The flood protection measures were implemented to reduce the overall core damage frequency as determined by the Oconee Probablistic Risk Assessment (PRA) study. UFSAR Section 3.4.1.1.1 states that the design basis flood is a failure of expansion joints in the CCW System near the condenser waterbox inlet or outlet nozzles. This change provides alternatives for mitigating the consequences of a Turbine Building flood.

SAFETY EVALUATION SUMMARY

This activity that involves additional methods for isolation during maintenance or modification activities that cannot cause failure of an expansion joint. These alternatives do not increase the system pressure above its design values. No other SAR described accidents are affected by these changes. These changes to SLC 16.9.11 are at least as effective as the existing SLC for limiting the Turbine Building flood level that could lead to other equipment failures. There are no adverse effects on containment integrity, radiological release pathways, fuel design, filtration systems, MSRV relief setpoints, or Radwaste systems. No new types of accidents or failure mechanisms are postulated. A Turbine Building flood is already evaluated in the SAR. No new malfunctions are postulated. The changes involves no physical modifications to the plant or changes in operating characteristics or procedures. The change involves no relaxation of seismic, environmental, or QA requirements. There are no reactivity management concerns. These changes do not affect any margins of safety defined in the basis for any technical specification. The proposed change does not affect any safety limits or limiting safety system settings. No plant safety limits, setpoints, or design parameters are adversely affected. There is no impact to the nuclear fuel, cladding, Reactor Coolant System (RCS), or containment integrity.

The Oconee Technical Specifications contain no requirements associated with Turbine Building flood. There are no Technical Specifications changes required. The change does not involve an unreviewed safety question. This revised SLC is consistent with other sections of the UFSAR, and no changes to the UFSAR are required, other than SLC 16.9.11 itself.

DESCRIPTION

SYSTEM: Radiation Information Alarms (RIAs)

SLC 16.7.1 requires an annual calibration be performed on the unit vent stack high range gas monitors (1,2,3RIA-56). The detectors for these monitors are located on the unit vent stack near the main steam relief valves, so in order to improve personnel safety, the calibration work orders for these monitors were changed in September of 1993 to be performed on a refueling basis instead of an annual basis. This change eliminated the need for personnel to access the detector with the unit on line. The problem is the requirement in SLC 16.7.1 was not changed. This activity changes the calibration requirement for RIA-56 in SLC 16.7.1 from an annual frequency to a refueling (18 month) frequency. This change makes the documentation in SLC 16.7.1 consistent with the calibration work orders.

SAFETY EVALUATION SUMMARY

No modifications to these monitors or any other SSCs are involved; therefore, the operating characteristics of these monitors are not affected. This activity does not affect the ability of the Radiation Monitoring System to monitor activity in the unit vent, nor does it affect any of the design bases requirements stated in Section 20 of OSS-0254.00-00-2022 (Design Basis Specification for the Process Radiation Monitoring System). No new or different failure modes are introduced as a result of this activity. The unit vent stack high range gas monitors do not have a safety-related function. The performance and reliability of these monitors are not degraded by this activity. These SLC changes do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created, and no SSCs are degraded. There are no physical changes to the plant or operating procedures. These changes create no USQs or safety concerns and no Technical Specification changes are required. No UFSAR changes, other than the affected SLC, are necessary.

VII. UFSAR CHANGES (Pkg 97-03)

DESCRIPTION

SYSTEM: Control Rod Drives (CRD)

This UFSAR change is based on ONOEs 2656, 2726, and 2727 which replaced the existing obsolete CRD programmers with solid state type available from B&W.

SAFETY EVALUATION SUMMARY

The replacement of obsolete equipment with modern updated types capable of performing the same design function does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Electrical diagram 7-4 of the Oconee UFSAR was updated accordingly to reflect the new programmers.

UFSAR CHANGES (Pkg 97-04)

DESCRIPTION

SYSTEM: Fuel Handling

This revision to UFSAR section 9.1.4 per PIP 97-1199 is largely editorial in nature. Revisions are made to fuel handling decriptions and processes to:(1) correct or clarify inaccurate or conflicting statements, (2) provide additional information, and (3) delete information that is too specific for the context of the activity being described.

SAFETY EVALUATION SUMMARY

The correction of unclear descriptions, sequences, and activities for the fuel handling equipment and processes with updated information reflecting present day operating practices does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Section 9.1.4 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-07)

DESCRIPTION

SYSTEM: Nuclear Instrumentation (NIs)

This revision to UFSAR Figures 7-6, 7, 9, 10 per PIP 97-0583 is simply completion of a change that was not made in a timely manner. These changes resulted from NSMs X2596 and X2909 which previously replaced the source and intermediate range NIs on all three units. In addition, Figure 7-1, which had been removed from previous editions, was restored.

SAFETY EVALUATION SUMMARY

The affected UFSAR text was already changed and the modifications were previously reported, thus this change is largely editorial in nature. Revisions are made to the appropriate figures to reflect installation of the new Reg Guide 1.97 full range detectors. The correction of the figures with updated information reflecting the present equipment does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Figures7-6, 7, 9, and 10 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-08)

DESCRIPTION

SYSTEM: Fuel Handling and Spent Fuel Pool (SFP)

This revision to UFSAR section 9.1.4 per PIP 96-2213 is largely clarification that should have been included in the previous update. Revisions are made to clarify that the normal refueling practice is to offload the entire core into the SFP versus performing an incore shuffle

SAFETY EVALUATION SUMMARY

The correction of unclear descriptions, sequences, and activities for the refueling processes with updated information reflecting present day operating practices does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. The SFP heat loads have been previously evaluated as adequate to handle full core offloads. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Section 9.1.4 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-09)

DESCRIPTION

SYSTEM: Main Steam, Feedwater, Core Flood

This editorial revision to UFSAR section 3.7.4.1 per OE-4170 VN is to clarifiy actual locations of the six 2g peak recordjng accelorometers.

SAFETY EVALUATION SUMMARY

This editorial correction to provide accurate description for the seismic measuring recorder locations does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Section 3.7.4.1 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-11)

DESCRIPTION

SYSTEM: Main Feedwater, Turbine Generator (TG)

This revision to UFSAR sections 10.2.1, 2, 4, and 10.4.6.3 per PIP 96-2632 is to correct the text to reflect the as-built plant configuration and present operating practices. This change is largely editorial in nature, and includes denoting that (1) the Mwe of the turbine is not an exact number, (2) secondary side systems can become contaminated, and (3) both supply lines from the upper surge tanks to emergency feedwater are either normally closed or capable of auto closure.

SAFETY EVALUATION SUMMARY

These revisions to provide more accurate descriptions of the Main FDW and TG systems does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Sections 10.2.1, 2, 4, and 10.4.6.3 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkg 97-12)

DESCRIPTION

SYSTEM: Radwaste

This USQ evaluation is for changes to UFSAR Sections 11.4 and 11.6 to reflect current operating practices. Discrepancies between this section and waste processing operations were found during the Preliminary Accuracy Review (Phase I) of this section and documented in PIP 97-2165. The UFSAR describes the Radwaste Facility as having NRC approved systems capable of processing liquid and solid radioactive waste by means of demineralization, evaporation incineration and concentration of liquid and solid wastes generated at the Oconee Nuclear Station. Disposal of the by-products generated by this process (i.e. bead resin, powdex resin, etc.) is also discussed. This change:

1.Deletes references to evaporation of liquid waste as this is no longer a process option.

2.Deletes references to recycle of reactor coolant since ONS does not recycle bleed water.

3.Updates references to the use of the Interim Radwaste Building (IRB) to reflect current operating practices.

4.Designates that the Radwaste Facility is the primary liquid waste processing facility.

SAFETY EVALUATION SUMMARY

At ONS, liquid wastes are accumulated in storage tanks according to the waste source and expected process train. The auxiliary Building coolant treatment header has been designed to facilitate the processing of liquid wastes from the high activity waste tanks, low activity waste tanks, and the miscellaneous waste holdup tanks in the Radwaste Facility. Waste is processed by filtration and/or demineralization, collected, sampled, analyzed and then disposed of under continuous radiation monitoring and control. The waste processing systems in the Radwaste Facility are not safety related and are not an accident initiator. This change is technical/editorial in nature but does not limit the ability of any plant Structure/System/Component (SSC) to perform its accident mitigation function. The Radwaste systems as described in the SAR are designed to process liquid waste through various options. This change is to describe current operating/processing practice. No changes are made to current practice described in the SAR. References to evaporation and recycle were deleted since these technologies are no longer being used, and references to the IRB were corrected to reflect that this is not the primary process option for liquid wasted. The 'bathtub' design of the Radwaste Facility contains any leaks as described in the SAR, and no new accident scenarios are created by this change. No plant SSCs are adversely affected by this change nor is the ability of any safety related SSC to perform it's intended function. All previous waste processing methodologies have been evaluated in the past and do not impact the safe operation of the plant. This change is to simply clarify which methodologies are no longer used and describe current operating practice. This change to the UFSAR has no effect on plant safety limits, setpoints, or design parameters. Based on the preceding, no USQs or safety concerns are created by this change. No Tech Spec changes are required. UFSAR Sections 11.4 and 11.6 were revised accordingly

UFSAR CHANGES (Pkg 97-13)

DESCRIPTION

SYSTEM: Radiation Information Alarms (RIAs)

UFSAR Tables 11-7 and 12-3 contain descriptions of RIAs. This revision to the tables per PIPs 96-2539 and 2637 corrects RIA locations and sensitivities.

SAFETY EVALUATION SUMMARY

No modifications to these monitors or any other SSCs are involved; therefore, the operating characteristics of the monitors are not affected. This activity does not affect the ability of the Radiation Monitoring System to monitor activity, nor does it affect any of the design bases for the Process Radiation Monitoring System. There are no physical changes to the plant or operating procedures. No new or different failure modes are introduced as a result of this activity. The performance and reliability of the RIAs are not degraded by this activity. These UFSAR changes do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created, and no SSCs are degraded. These changes create no USQs or safety concerns and no Technical Specification changes are required. UFSAR Tables 11-7 and 12-3 were updated as necessary.

UFSAR CHANGES (Pkg 97-14)

DESCRIPTION

SYSTEM: LPI

A 50.59 evaluation has been performed for the UFSAR changes to Section 6.3.2.3.8 per PIP 97-907. This UFSAR change is simply to correct an oversight from the 1993 revision. The actual system design temperature for Units 1 and 2 'B' coolers is 300F, not 250F. This change is editorial in nature.

SAFETY EVALUATION SUMMARY

Correcting the UFSAR described temperature of the 'B' LPI heat exchanger to the actual plant design number does not: (1) increase the consequences or, probability of a SAR described accident (2) affect the possibility of a different type of accident occurring, (3) cause any malfunctions of any type, or (4) reduce any margins of safety. There is no physical change to the plant or procedures. This largely editorial correction does not require any Technical Specification changes. There are no unreviewed safety questions. Section 6.3.2.3.8 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-15)

DESCRIPTION

SYSTEM: None

UFSAR Sections 12.4.5.2 and 12.4.7 were revised, respectively, to (1) clarify that the sampling of airborne radioactivity may be performed by using appropriate cartridges rather than restricting to use only of charcoal or silver zeolite type, and (2) reflect the use of both the self reading pocket dosimeters and the new electronic dosimeters types.

SAFETY EVALUATION SUMMARY

These changes simply reflect use of more modern means of personnel and airborne radiation monitoring with equivalent or better devices as approved by the plant Radiation Protection Staff. No modifications are made to any plant SSCs. This activity does not affect the ability of the Radiation Monitoring System to monitor activity, nor does it affect any of the design bases for the Process Radiation Monitoring System. There are no physical changes to the plant or SAR described operating procedures. No new or different failure modes are introduced as a result of this activity. The performance and reliability of no plant SSCs are degraded by this activity. These UFSAR changes do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created, and no SSCs are degraded. These changes create no USQs or safety concerns, and no Technical Specification changes are required. UFSAR Sections 12.4.5.2 and 12.4.7 were updated as necessary.

UFSAR CHANGES (Pkg 97-16)

DESCRIPTION

SYSTEM: None

UFSAR Section 12.1 "ALARA Program" was revised per PIP 97-1353 to clean up grammatical, format, and wording errors. Also to delete irrelevant information about ALARA concerns for design and construction of a new plant site. This change is largely editorial in nature

SAFETY EVALUATION SUMMARY

These UFSAR changes simply reflect largely editorial updating of the ALARA program information as approved by the plant Radiation Protection Staff. No changes or modifications made to any plant SSCs, including the Radiation Monitoring Systems. There are no physical changes to the plant or SAR described operating procedures. No new or different failure modes are introduced as a result of this activity. These UFSAR changes do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created, and no SSCs are degraded. These changes create no USQs or safety concerns, and no Technical Specification changes are required. UFSAR Section 12.1 was updated accordingly.

UFSAR CHANGES (Pkgs 97-17)

DESCRIPTION

SYSTEM: Condenser Circulating Water (CCW)

Minor modification ONOE-10592 and PIP 97-1801 changed OFD-133A-3.1 to show valve 3CCW-466 (CCW Booster Pump Alt. Suction Isolation) normally open. This 10 CFR 50.59 evaluation was revised per PIP 0-O97-3469. The original evaluation was screened from the 50.59 regulation, however, a change to the UFSAR was pending to show valve 3CCW-466 on UFSAR Figure 9-9 as a normally closed valve. The 12/31/96 update of the UFSAR did not show the valve and was therefore not affected by ONOE-10592. However, the pending change was affected since it would have shown the valve on UFSAR Figure 9-9 as normally closed. Therefore, the original evaluation should have been performed as a USQ evaluation due to the need for a UFSAR change that was different from the pending change. Valve 3CCW-466 is a 12-inch diameter, Class F, manual valve located in the piping from the CCW crossover to the suction of the CCW Booster Pumps. This piping provides an alternate suction supply for the CCW Booster Pumps whenever maintenance is performed that requires isolating the normal suction supply at 3CCW-341.

SAFETY EVALUATION SUMMARY

The CCW Booster Pumps are non-QA condition (non-safety related). This modification does not involve any physical change to the facility. This change does not affect any procedures or testing. Leaving the valve in the normally open position will have no adverse effect on the operation of the CCW Booster Pumps. It will provide another normally-open suction flow path to the pumps, but this will have a negligible effect on the pump operation. Either flow path can provide adequate net positive suction head. The piping downstream of 3CCW-466 is Duke Class G (non-seismic). The valve serves as a seismic/non-seismic boundary. Leaving this valve in the normally open position meets the criteria in UFSAR 3.7.3.9 for seismic/non-seismic boundaries. A postulated break at 3CCW-466 causes no loss of safety function, including effects of resultant flooding. This modification will not introduce any unwanted system interactions. The valve is located in a mild environment. There is no significant effect on the steady-state or transient characteristics of the system. The modification will not increase the system pressure above its design values. The modification will not affect the frequency of operation or alter any testing requirements. The modification will not prevent any actions described in the SAR. There are no adverse effects on containment integrity, radiological release pathways, fuel design, filtration systems, MSRV relief setpoints, or Radwaste systems. No new types of accidents, malfunctions or failure mechanisms are postulated. Seismic/LOOP and LOCA/LOOP accidents are already evaluated in the SAR. This change involves no physical modifications to the plant or changes in operating characteristics or procedures. The change involves no relaxation of seismic, environmental, or QA requirements. There are no concerns associated with reactivity management. Minor modification ONOE-10592 does not require a change to the Technical Specifications. The proposed changes do not involve a USQ. UFSAR Figure 9-9 was changed to show 3CCW-466 to be normally open.

UFSAR CHANGES (Pkg 97-18)

DESCRIPTION

SYSTEM: Electrical

NSMs 43000, 52932, and 53000 are part of the Oconee Service Water Project. These changes to electrical power cable and tray criteria in UFSAR sections 8.3.1.5.1,2 are made in support of the OSW project. The new trenches added for the ESV building contain electrical cabling.

SAFETY EVALUATION SUMMARY

The trench and additions are designed to contain systems and electrical equipment. Now power cables will be derated, and cable trays spaced, based on the current cable organization responsible for IEEE S-135, ICEA publication # P-46-426. Use of current industry guidance and electrical codes does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created, and no SSCs are degraded. These changes create no USQs or safety concerns, and no Technical Specification changes are required. UFSAR Sections 8.3.1.5.1, and 2 were updated accordingly.

UFSAR CHANGES (Pkg 97-19)

DESCRIPTION

SYSTEM: Electrical

NSMs X3000 and X2932 are part of the Oconee Service Water Project. These changes are applicable to electrical cable splices, routings, and trench criteria in UFSAR section 9.5.1.4.3, and are made in support of the OSW project. The new trenches added for the ESV building contain electrical cabling.

SAFETY EVALUATION SUMMARY

The ESV trench and additions are designed to contain systems and electrical equipment. Section 9.5.1.4.3 information on electrical cable splices, routings, and dedicated support systems (trenches) was updated based on the current cable organization responsible for IEEE S-135, ICEA publication # P-46-426. Use of current industry guidance and electrical codes does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. There are no Appendix R fire concerns. No new radiological release pathways or failure modes are created, and no SSCs are degraded. These changes create no USQs or safety concerns, and no Technical Specification changes are required. UFSAR section 9.5.1.4.3 was updated accordingly.

UFSAR CHANGES (Pkg 97-21)

DESCRIPTION

SYSTEM: LPI

Minor Modifications ONOE-5896, 97, 6006, 7 replaced Low Pressure Injections Valves (LPI) 3LP-17 and 3LP-18 and their valve operators. However, the UFSAR was not updated in a timely manner for these changes which were made per PIP 97-1110. These changes were made to fulfill Generic Letter 89-10 commitments.

SAFETY EVALUATION SUMMARY

The affected LPI valves are active Engineered Safeguards components that are required to be fully open within 15 seconds of ES actuation in order to supply borated water to the reactor core. The new valve operators change the stroke times of the valves, but will remain well within the design basis requirements. The new valves and operators will meet the same qualifications as the existing, i.e. seismic, environmental, and QA-1. The safety related power supplies are unchanged. There is no increased potential for leakage. The LPI system will function as designed during normal and accident conditions. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity has no effect on any margins of safety as previously evaluated in the SAR. This change simply reflects the actual plant configuration, so that the UFSAR descriptions will be more accurate. There are no USQs involved with this modification, and no Technical Specification changes are required. UFSAR section 6.3.2.6.3 was revised to reflect installation of the new ROTORK operators.

UFSAR CHANGE (Pkg 97-22)

DESCRIPTION

SYSTEM: Penetration Room Ventilation System (PRVS), Building Spray (BS), RB Cooling Units (RBCU), and RB Isolation

UFSAR Section 3.8.1.7.5 was revised to remove outdated wording concerning ES systems testing frequency that is inconsistent with Tech Spec requirements. This change is largely editorial and eliminates the discrepancies.

SAFETY EVALUATION SUMMARY

These UFSAR changes do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The affected systems and components will continue to be tested in accordance with Tech Spec and/or IST requirements and will remain capable of performing their design functions during normal and accident conditions. There are no physical changes to plant SSCs or operating procedures. This change is largely editorial and eliminates UFSAR discrepancies. No safety concerns or USQs are created by these revisions to UFSAR section 3.8.1.7.5. No Technical Specifications changes are required.

UFSAR CHANGE (Pkg 97-23)

DESCRIPTION

SYSTEM: Nuclear Fuel

Calculation OSC-7023 contains a 10CFR50.59 analysis and safety review for continued use of fuel assemblies with damaged grids at Oconee. This simple UFSAR change adds information to clarify that tests were performed with specimens of Zircaloy 4 and Inconel spacer grids.

SAFETY EVALUATION SUMMARY

This UFSAR changes does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSCs are adversely affected. There are no new malfunctions. The nuclear fuel remains capable of performing its design functions during normal and accident conditions. There are no physical changes to plant SSCs or operating procedures. This change is largely editorial and clarifies the use of Inconel 718 spacer grids for testing in the UFSAR. No safety concerns or USQs are created by this revision to UFSAR section 15.14.5. No Technical Specifications changes are required.

UFSAR CHANGES (Pkg 97-24, 116)

DESCRIPTION

SYSTEM: Standby Shutdown Facility (SSF)

This revision to UFSAR sections 9.6.2, 3, 4 per PIP 97-0120 corrects the text and figures to reflect the as-built plant configuration, present operating practices, and functions. This change also addresses deletion of (1) SSF Reactor Coolant makeup requirement for volume shrinkage and (2) to connect Reactor Vessel head vent valves to SSF diesel generator within 8 hours.

SAFETY EVALUATION SUMMARY

These revisions provide more accurate and up to date descriptions of the SSF systems, and do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. Makeup to RCS for volume shrinkage is not required for natural circualtion flow. The SSF letdown orifice was previously modified to pass a flow rate that is greater than the SSF RC makeup pump, so the RV head vents are no longer needed to avoid overfilling the RCS. Providing a better description of the ASME code and pipe class design for SSF systems is largely editorial, as is correcting flow diagrams to agree with the as-built plant. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Sections 9.6.2, 3, and 4 and Figure 9-15 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkg 97-25)

DESCRIPTION

SYSTEM: Auxiliary Instrument Air (AIA)

This revision to UFSAR sections 9.5.2.2 per PIP 97-1892 clarifies the text portion describing the AIA to reflect its relationship to the Maintenance Rule (MR).

SAFETY EVALUATION SUMMARY

This revision provides more accurate and up to date descriptions of the AIA system with respect to the MR. The AIA system provides a backup source of instrument air to key plant components following a loss of the normal source. Although the system may be available, it is not required for performing or supporting any operation. AIA fails in a safe condition. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Section 9.5.2.2 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGE (Pkg 97-28)

DESCRIPTION

SYSTEM: Reactor Coolant System

UFSAR Section 5.2.3.3.1 describes the stress evaluation which was originally performed on the reactor vessel in accordance with ASME Section III. Section 5.2.3.3.1 also references Table 5-6 and 5-7 for the stress evaluation summaries.

A review of the values in UFSAR Table 5-6 and Table 5-7 against the current reactor vessel stress evaluation documented in OSC-1815 shows that the values in UFSAR Table 5-6 and Table 5-7 values are historical, based on the original stress evaluation. The values do not reflect the current stress analysis. Therefore, in order to prevent confusion the following UFSAR changes were performed for this activity:

- In UFSAR Section 5.2.3.3.1, paragraphs 2 and 3 were reworded for clarification.
- In UFSAR Tables 5-6 and 5-7, a note was added for clarity.
- In UFSAR Table 5-7, the figure number was revised to reference the correct figure number for the points of stress analysis.

SAFETY EVALUATION SUMMARY

This activity provided clarifications to UFSAR Section 5.2.3.3.1, Table 5-6 and Table 5-7 to indicate that the stress evaluation summarized in UFSAR Table 5-6 and 5-7 are historical and that the latest stress evaluation is contained in calculation OSC-1815. Also a minor correction to UFSAR Table 5-7 was made to correctly reference UFSAR Figure 5-10 for the points of stress analysis. The proposed changes described in this activity do not affect the integrity of the reactor vessel as analyzed in OSC-1815.

These changes have been previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the proposed addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-29)

DESCRIPTION

SYSTEM: Reactor Coolant System (RCS)

UFSAR Section 5.2.3.3 describes conformance to fracture toughness requirements and material surveillance requirements of 10CFR 50 Appendices G and H and describes the methods for guarding against brittle fracture of the reactor vessel. The methodology uses fracture mechanics concepts and the reference nil-ductility temperature (RT_{NDT}). UFSAR Section 5.2.3.3, paragraph 5, sentence 3 currently states:

"The increase in the Charpy V-notch 30 ft-lb temperature, or the increase in the 35 mils of lateral expansion temperature, which ever results in the larger temperature shift due to irradiation, is added to the original RT_{NDT} along with a margin value to adjust the RT_{NDT} for radiation embrittlement."

This activity revised the above statement as follows as is described in 10 CFR 50 Appendix G and BAW-2050 and BAW-2051:

"The increase in the Charpy V-notch 30 ft-lb temperature~~, or the increase in the 35 mils of lateral expansion temperature, which ever results in the larger temperature shift due to irradiation,~~ is added to the original RT_{NDT} along with a margin value to adjust the RT_{NDT} for radiation embrittlement."

SAFETY EVALUATION SUMMARY

This activity changes the description of the reactor vessel material surveillance program described in UFSAR Section 5.2.3.3. The proposed change was provided to correctly show that the increase in Charpy-V notch temperature is measured only at the 30 ft-lb level and eliminates the alternate requirement of 35 mils of lateral expansion. This requirement conforms with 10CFR50 Appendix G and is consistent with the current evaluation methodology as described in BAW-2050 and BAW-2051. As such this activity does not reduce the margins of safety against brittle fracture of the reactor vessel and therefore does not present an unreviewed safety question. This activity provides a correction to the description of the Oconee reactor vessel surveillance program to indicate that the increase in Charpy V-notch temperature is measured only at the 30 ft-lb level; the alternate requirement for 35 mils of lateral expansion was eliminated since it is not a requirement stated in the current rules and regulations described in 10CFR 50 Appendix G, nor is it consistent with the current B&W evaluation methodology described in BAW-2050 and BAW-2051. No Technical Specification changes are required. This activity does not reduce the margins of safety against brittle fracture of the reactor vessel and therefore does not present an unreviewed safety question or safety concern.

UFSAR CHANGE (Pkg 97-30)

DESCRIPTION

SYSTEM: Reactor Coolant System

UFSAR Section 5.3.3.4 states that the fabrication inspection requirements imposed on the reactor vessel are summarized in UFSAR Table 5-10. In review of UFSAR Table 5-10 against the historical information shown in the original FSAR Table 4-12, the following errors were noted:

Item 1.5.10 of the Original FSAR Table 4-12 for Reactor "Weldments" shows that ultrasonic and dye penetrant was used in the fabrication inspection for cladding, and sealing surfaces. Current UFSAR does not show the dye penetrant process.

Item 3.3.5.1 in the original FSAR Table 4-12 shows that ultrasonic and dye penetrant was used in the fabrication inspection for "Pressurizer" heater tubing. Current UFSAR shows that magnetic particle and eddy current inspection process was used.

Item 3.3.5.2 the original FSAR Table 4-12 shows that radiographic testing was used in the fabrication inspection for "Pressurizer" heater element positioning. Current UFSAR shows that ultrasonic inspection process.

The UFSAR change performed by this activity revised UFSAR Table 5-10 to be consistent with the historical information shown in Original FSAR Table 4-12.

SAFETY EVALUATION SUMMARY

This activity provids a correction to UFSAR Table 5-10 to be consistent with the historical information originally presented in FSAR Table 4-12. UFSAR Table 5-10 provides a description of the Oconee Fabrication Inspection methods for RCS component. These changes were previously evaluated and approved, and do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the UFSAR addition performed does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-31)

DESCRIPTION

SYSTEM: Reactor Coolant System

UFSAR Sections 5.3.2.2, 5.3.2.3, and 5.3.3.5 provide historical information concerning initial Reactor Vessel fabrication/construction and installation. These sections were updated to more accurately reflect the historical nature of the information.

UFSAR Section 5.4.1.2 provides, in the first paragraph, a cross reference to UFSAR Figure 5-20 for the reactor coolant pump(RCP) characteristics of the Unit 2 and 3 pumps. This section was updated to more accurately reflect the historical nature of the information.

UFSAR Section 5.4.4.7, in the second paragraph, provides a listing of documents included in the procurement QA folder for the originally purchased RCP motors, as discussed in original FSAR Section 4.2.2.6. This section was updated to more accurately reflect the historical nature of the information.

UFSAR Section 5.4.4.7, in the third paragraph, indicates that the above RCP/M QA folders are in the control of the field quality control engineer, and that a copy is kept in the Design Engineering Department File. This historical information, applicable to the originally purchased components as discussed in original FSAR Section 4.2.2.6, is no longer accurate. Individual RCP/M QA information is available in the applicable controlled Procurement package. This section was updated to more accurately reflect the historical nature of the information and to provide a reference for current location.

SAFETY EVALUATION SUMMARY

The reactor vessel and reactor coolant pumps, which are the components of interest for this activity, were designed fabricated, inspected and tested in accordance with ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels, except that pump casings were not code stamped (Ref. 1 Section 5.2.2.1 and 5.2.2.3). Historical discussions of fabrication/inspection/installation processes, initial procurement documentation, and estimated performance haracteristics do not directly impact the design, function or integrity of the particular components. These discussions demonstrate the quality controls in place during assembly and construction and, thereby, provide additional justification of safe operation capability.

Therefore, clarification of the applicable statements in UFSAR Sections 5.3.2.2, 5.3.2.3, 5.3.3.5, 5.4.1.2, and 5.4.4.7 to more accurately reflect the historical nature of the information and, in the case of 5.4.1.2, to provide consistency with other UFSAR references provides clarification and will not impact system/component integrity/reliability. The design, integrity, operation and function of related systems, structures and components are not affected by the clarifications in this activity. As such, the clarifications described in this activity do not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-32)

DESCRIPTION

SYSTEM: Reactor Coolant System

UFSAR Tables 5-12 and 5-13 provide the physical and chemical properties for the various material heat numbers which were used in the construction of the reactor vessel. Table 5-12 lists ultimate and yield strengths, elongation, impact values and test temperatures for the various reactor vessel heat numbers. Table 5-13 presents a breakdown of the composition by element (C, Mn, P, S, Si, Ni, Mo, Co, V, and Cr) for each of the reactor vessel material heat numbers identified from Table 5-12. These tables provide a snapshot of the material properties at the time of vessel construction are therefore historical. A review of the historical information against the original FSAR Tables 4-16 and 4-17 indicated the following two discrepancies:

- In UFSAR Table 5-12, the heat number for the first outlet nozzle line item incorrectly shows the heat number as 122S316VA1. Original values from FSAR Table 4-16 shows the first outlet nozzle line item as 122S316VA2. All other values in UFSAR Table 5-12 are exactly identical to those shown originally in FSAR Table 4-16.

- In UFSAR Table 5-13, the composition of the element Vanadium (V) is incorrectly shown as <.02 and .02 for heat numbers 122S347VA1 and 125S535VA1, respectively. Original values from FSAR Table 4-17 show the composition as .02 and <.02 for the respective heat numbers. All other values in UFSAR Table 5-13 are exactly identical to those shown originally in FSAR Table 4-17

This activity revised UFSAR Table 5-12 and Table 5-13 to correct the above discrepancies to match the historical information shown in FSAR Table 4-16 and 4-17.

SAFETY EVALUATION SUMMARY

No Technical Specification changes are required. The changes, as described in the Safety Analysis above, revise UFSAR Tables 5-12 and 5-13 to be consistent with the historical values provided in FSAR Table 4-16 and 4-17. Since Oconee is part of an integrated material surveillance program which provides direct data for determination of reduction in fracture toughness properties due to radiation embrittlement, and a stress evaluation of the reactor vessel in OSC-1815 has determined that the integrity of the reactor vessel is maintained, this activity does not reduce the reactor vessel's margin of safety.

These changes were previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the UFSAR addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-33)

DESCRIPTION

SYSTEM: Reactor Coolant System

UFSAR Section 5.2.3.11.2 provides historical information concerning Reactor Coolant System (RCS) component inspection and testing during the manufacturing, construction and installation phases. UFSAR Section 5.2.3.11.2 paragraph three was affected by this minor change. This change was to update and make consistent with the historical nature of this section.

Included in paragraph three, UFSAR Section 5.2.3.11.2, are the Quality Assurance procedures for components purchased and manufactured outside of B&W. It describes the shipment routings and auditing processes. It also states that an individual audit will be done by B&W's Nuclear Power Generation Department Quality Assurance Section. Paragraph three, UFSAR Section 5.2.3.11.2, discusses Quality Assurance procedures which has been replaced by the Duke Power Topical Report Quality Assurance program. Information in paragraph three was revised to agree with the historical nature of UFSAR Section 5.2.3.11.2.

In order to provide consistency and clarification with the Original FSAR, the described activity changed the first and third sentences of paragraph three, UFSAR Section 5.2.3.11.2, to be consistent with the historical information of this section.

SAFETY EVALUATION SUMMARY

The Reactor Coolant System (RCS) provides the means to transfer the heat generated from the reactor core to the steam generators. The RCS components are designated as Class I equipment and are designed to maintain their functional integrity during earthquake. The design, fabrication, inspection and testing of the reactor vessel and closure head, steam generator (both reactor coolant side and secondary side), pressurizer and attachment nozzles on the vessels is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels. The historical discussions of Shop Inspections reflect the manufacturing, construction, installation and inspection requirements of RCS components prior to and during the time of plant construction. The changes do not affect the as-built design or operation of the RCS components or related systems and do not present an unreviewed safety question. These changes were previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the proposed addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-34)

DESCRIPTION

SYSTEM: Reactor Coolant System, OTSG

This activity updated certain once-through steam generator (OTSG) design data values for consistency with as-built design data provided in the Component Specifications, Vendor Documents and Drawings.

UFSAR Table 5-20 provides, per generator, Steam Generator Design data. This information is identical to original FSAR and FSAR supplement 9 information with the exception of the number of Secondary Side Handholes. However, certain values, such as Unit 1, 2 Steam Flow, Unit 1, 2 Feedwater Temperature, Unit 1, 2 Full Load Feedwater Temperature, Secondary Side Hydrotest Pressure for all three units, Temperature Well diameter for all three units, number of Unit 1, 2 EmergencyAux. Feedwater Nozzles, and number of Unit 3 Handholes, are not consistent with the as-built data provided in current vendor documents/drawings and component specifications.

This activity revised UFSAR Table 5-20 in accordance with as-built information provided in current vendor information and component specifications.

SAFETY EVALUATION SUMMARY

Clarification of the applicable statements in UFSAR Table 5-20 values to more accurately reflect the per unit as-built design data of the Oconee steam generators as provided in current vendor documents, drawings and component specifications do not affect the design, integrity, operation and function of related systems, structures and components, as previously evaluated in the SAR. These changes were previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-35)

DESCRIPTION

SYSTEM: Reactor Coolant System (Reactor Coolant Pumps)

UFSAR Sections 5.4.4, in the third paragraph, and 5.4.4.5, in the second paragraph, provide brief discussions of motor monitoring devices. The discussions are inconsistent in that vibration devices are not addressed in section 5.4.4. This activity updated the UFSAR statements to consistently reflect a more accurate description of the vibration devices provided for motor protection.

UFSAR Section 5.4.4.5, in the third paragraph, provides a discussion of bearing failure considerations. This discussion indicates that bearing failure with the motor shut down would result in melting of the bearing babbit, and does not provide discussion of operator action in response. This activity clarified the UFSAR to indicate that bearing failure with <u>motor operating</u> will result in melting of the bearing, and included related preventive measures.

SAFETY EVALUATION SUMMARY

Updating the existing information presented in UFSAR Sections 5.4.4 and 5.4.4.5 for consistency with as-built design documents and for clarity does not affect the design, function, operation, or integrity of related systems/components. These changes were previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-37)

DESCRIPTION

SYSTEM: Reactor Coolant System

This activity provided minor clarifications to the sixth paragraph of UFSAR Section 5.4.3 to reflect that Reactor Coolant System (RCS) piping connections, larger than 2", are butt-welded except for the flanged connections on the pressurizer relief valves.

UFSAR Section 5.4.3, in the sixth item, provides indication that all RCS piping connections are butt-welded except for the flanged connections on the pressurizer relief valves. This statement is identical to a corresponding statement in original FSAR section 4.2.2.4 and provides a description of RCS characteristics. Existing information in UFSAR Sections 5.2.1.5.1, 5.2.2.2 and Tables 1-3, 5-4 indicate that piping is in accordance with USAS B31.7, dated February 1968, and as corrected for Errata under date of June, 1968. This Standard, and the current ONS Pipe Installation Specification (OS-0243.00-00-0001) allow socket welds on Class A pipe up to 2". This activity revised UFSAR Section 5.4.3 statement for consistency with existing UFSAR statements, Standard B31.7 requirements, and current Pipe Installation Specification

SAFETY EVALUATION SUMMARY

The RCS piping was designed, fabricated, inspected and tested is in accordance with USAS B31.7, Code for Pressure Piping, Nuclear Power Piping, dated February 1968, and as corrected for Errata under date of June, 1968, excluding the pressurizer surge line and the spray line. Socket welds are not prohibited by USAS B31.7, except for corrosive applications, but are limited to connections less than or equal to 2 inches in diameter. Also, current Pipe Installation Specification allows socket welding on Class A piping up to 2 inches. The RCS is not a highly corrosive environment such that socket welds should be prohibited. Additionally, no Safety Evaluation Report or other licensing correspondence was identified that might indicate that RCS piping butt-welds are a critical characteristic. Piping pressure boundary function is maintained and socket welds for system piping smaller than 2 inch are allowed by the applicable code, identified in existing UFSAR Sections 5.2.1.5.1, 5.2.2.2 and Tables 1-3 and 5-4.

Clarification of the applicable statement in UFSAR Section 5.4.3 to more accurately reflect RCS piping connection characteristics, in accordance with code and specification requirements does not affect the design, integrity, operation and function of related systems, structures and components, as previously evaluated in the SAR. These changes have been previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-38)

DESCRIPTION

SYSTEM: Reactor Coolant System

UFSAR Section 5.2.3.5 describes the Reactor Coolant System's reliance on interconnected systems. The discussion is focused on the Reactor Coolant System's reliance on interconnected systems to provide decay heat removal during the hypothetical scenario of a complete loss of station power.

A review of UFSAR Section 5.2.3.5 against related descriptions in UFSAR Sections 8.3.2.2, 9.2.2.2.1, 10.4.7.1.3 indicated that Section 5.2.3.5 was unclear in its discussion of plant systems relied on during complete loss of station power. This situation occurs because the system functions for coping with a complete loss of station power, credited by Duke to comply with the Station Blackout Rule (10 CFR 50.63) and approved by the NRC in their safety evaluations, are different from those plant functions normally available during a complete loss of station power. Therefore, to prevent confusion the following UFSAR clarification was for this activity:

> In UFSAR Section 5.2.3.5 move the sixth sentence from its current location and create a new paragraph by modifying this sentence to read:
>
> > "The analysis of the plant component functions credited for coping with the unlikely condition of total loss of station power is presented in Section, "Station Blackout Analysis."

SAFETY EVALUATION SUMMARY

The changes associated with this proposed activity clarified the UFSAR's discussion of RCS reliance on interconnected systems as discussed in UFSAR Section 5.2.3.5. This change also ensured consistency between various UFSAR sections regarding its discussion of Station Blackout. This activity does not change any technical information and was provided for clarification purposes only. It did not affect any analysis previously evaluated in the UFSAR and did not change any discussion of system configuration, design basis, or safety related functions. Therefore, no Reactor Coolant System components or safety related components associated with interconnected systems were affected by this change. These changes were previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the proposed addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

FSAR CHANGE (Pkg 97-39)

DESCRIPTION

SYSTEM: Reactor Coolant System

This activity provided minor clarifications to the ninth paragraph of UFSAR Section 5.4.6 to remove information that is a duplicate of the statement in the fifth paragraph of that section. This activity also provided minor clarifications to UFSAR Section 5.4.9 to remove the fourth reference as not applicable to UFSAR Section 5.4 discussions.

SAFETY EVALUATION SUMMARY

UFSAR Section 5.4.9 lists reference B&W - 1543, Rev. 2 which is applicable to Reactor Vessel Material Surveillance. Existing UFSAR Section 5.2.3.13 provides a detailed discussion of the reactor vessel material surveillance program and existing UFSAR Section 5.2.4 lists reference B&W - 1543, Rev. 3. The discussions in UFSAR (Ref. 1) Section 5.4 address the major RCS components but do not address the reactor vessel other than brief statements regarding the vessel support. As such, removal of reference that is correctly listed in UFSAR Section 5.2.4 has no impact on technical content of the UFSAR. Clarification of the applicable statements in UFSAR Section 5.4.6 to remove duplicate information, and UFSAR Section 5.4.9 to remove a reference which is not applicable to the subject matter of the section does not affect the design, integrity, operation and function of related systems, structures and components, as previously evaluated in the SAR.

These changes have been previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the proposed addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-40)

DESCRIPTION

SYSTEM: Reactor Coolant System

UFSAR Section 5.2.2 describes the component codes and classifications applicable to the Reactor Coolant System. These codes and classifications are summarized in Table 5-4. Similarly, UFSAR Section 5.2.3.4 provides a historical discussion of the research and development and fabrication of the steam generators. The locations of the steam generator fabrication welds are shown on Figure 5-11.

Reviews of UFSAR Table 5-4 and Figure 5-11 against UFSAR Section 5.2.2 discussion and original FSAR Figure 4-5A indicate that the UFSAR Table and Figure are lacking clarity. UFSAR Table 5-4 contains Reactor Coolant System component codes and classification information and is supported by UFSAR Table 1-3 which contains an abbreviated listing of these codes for important Reactor Coolant System components. The code information in UFSAR Table 5-4 represents original design/construction component code information and is unchanged from the information presented in original FSAR Table 4-2. In a similar fashion, UFSAR Figure 5-11 reflects information contained on original FSAR Figure 4-5A with the exception that two notes that were on the original FSAR figure do not appear on UFSAR Figure 51 1 (the figure shows where the notes apply but there is no legend explaining the notes). Therefore, in order to ensure the proper use of UFSAR Table 5-4 and Figure 5-11 the following UFSAR changes were clarified as to their applicability and historical significance.

SAFETY EVALUATION SUMMARY

The changes associated with this activity provide clarifying detail to existing UFSAR information and do not change any technical information currently presented in the UFSAR. Additionally, these changes will not invalidate any other UFSAR sections and will not change any of the evaluations or conclusions discussed in the Safety Analysis. The technical information on Table 5-4 represents original design/construction code information while Figure 5-11 represents historical information. In both cases, the information presented is unchanged from the original FSAR as reviewed and approved by the NRC in their original safety evaluations. Addition of clarifying information as part of this activity does not change any system component design or prevent the ability of any safety related components from performing their safety function. Therefore, the integrity of the Reactor Coolant System components is maintained and the proposed changes described in this activity do not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-42)

DESCRIPTION

SYSTEM: Reactor Coolant System

UFSAR Section 5.2.3.10.5 describes the seal leakoff path for reactor coolant pumps for all three units. The description in the UFSAR is inconsistent with existing Oconee design drawings and flow diagrams which have been verified to be correct. Therfore, this change corrects the erroneous description by making revisions to this UFSAR section to accurately describe the seal leakoff path and destination of seal leakoff water.

SAFETY EVALUATION SUMMARY

The proposed change provides a clarifying description of the as-built design of the reactor coolant pump seal leakoff piping as described in Oconee flow diagrams and design information. This change clarifies the available flowpaths and does not restrict normal operational flowpaths. These changes have been previously evaluated and approved, and do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGES (Pkg 97-44)

DESCRIPTION

SYSTEM: Alarms and Communications

This revision to UFSAR sections 7.7.3, 7.7.4.1, and 7.7.4.2 per PIPs 97-3447 and 3583 is to clarify the text portions for Security Alarms and emergency internal/external communications to reflect current operating practices.

SAFETY EVALUATION SUMMARY

This revision provides more accurate and up to date descriptions of the subject items. This change is non-technical in nature, and clarifies that Security, not Operations, receives alarms in the event of unauthorized entry into vital areas. This change also addresses the fact emergency communciations are performed via a fiber optic system, not a microwave. These changes do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Sections 7.7.3, 7.7.4.1, 2 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGE (Pkg 97-45)

DESCRIPTION

SYSTEM: Reactor Coolant System

This activity provides minor editorial clarification/correction changes to the "Auxiliary Feedwater" designation in UFSAR Sections 5.2.2.2, 5.4.2, and 5.4.7.4 to be consistent with the existing "Emergency Feedwater" system/component designations in UFSAR Sections 5.1.2.4, 5.2.3.5, 5.2.3.10.3, 7.5.2.40, throughout Chapter 10, 15.13.3.2, and 15.14.3.2.

SAFETY EVALUATION SUMMARY

This activity provided minor clarifications/corrections to UFSAR Sections 5.2.2.2, 5.4.2, and 5.4.7.4 for consistency with the existing information in Sections 5.1.2.4, 5.2.3.5, 5.2.3.10.3, 7.5.2.40, Chapter 10, 15.13.3.2, and 15.14.3.2. These minor clarifications for consistent system/component designation do not affect the design or function of the system/component, as previously evaluated in the SAR. These changes were previously evaluated and approved, and do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-46)

DESCRIPTION

SYSTEM: Reactor Coolant System

This activity provided minor clarification to UFSAR Section 5.4.2.4 to provide an accurate cross reference consistent with similar feedwater and reactor coolant quality references in existing UFSAR Sections 5.2.3.2 and 9.3.1.2.

SAFETY EVALUATION SUMMARY

This revision to update the cross reference in UFSAR Section 5.4.2.4 is consistent with other portions of the UFSAR which use the same cross reference. The other portions of the UFSAR were verified accurate. Therefore, these changes were previously evaluated and approved, and do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-47)

DESCRIPTION

SYSTEM: Reactor Coolant System (Reactor Coolant Pumps)

This activity updated Unit 2, 3 Reactor Coolant Pump (RCP) injection water design values for consistency with the more conservative as-built parameters provided in the vendor documents. It also updated RCP pump dry weight for consistency with as-built vendor supplied values utilized in the applicable stress analysis.

UFSAR Table 5-17 provides Reactor Coolant Pump Design Data for Oconee 2, 3. Certain values, such as Injection Water Temperature and Pump without motor Dry weight are not consistent with the as-built design data provided in current vendor documents and drawings. These UFSAR values are however, identical to the corresponding values in original FSAR Table 4-7.

This activity updated UFSAR Table 5-17 in accordance with as-built data provided in current vendor information, and for consistency with values specified in current operating procedures, as follows:

Injection Water Temperature, °F 120 °F ± 10 °F

Dry Weight Without Motor, lb 100,000

SAFETY EVALUATION SUMMARY

Clarification of the applicable parameters in UFSAR Table 5-17 to accurately reflect the as-built design values of the Oconee 2, 3 RCPs as provided in current vendor documents/drawings does not affect the design, operation and function of related systems, structures and components, as previously evaluated in the SAR. Component Integrity as evaluated in OSC-3613 stress analysis is maintained. These changes were previously evaluated and approved, and do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question. No Technical Specification changes are required.

UFSAR CHANGES (Pkg 97-49)

DESCRIPTION

SYSTEM: Steam Generator Pressure

This revision to UFSAR sections 7.5.2.5, 7.5.2.55, and 56 per PIP 98-181 added pertinent information pertaining to S/G pressure instrumentation (see NSMs X2873) and clarifies the Reg Guide 1.97 wind direction and speed instrumentation variables.

SAFETY EVALUATION SUMMARY

This revision provides more accurate and up to date descriptions of the subject items. This change is largely editorial in nature, and adds missing information on S/G pressure instrumentation related to the Main Steam Line Break mods and clarifies that the wind direction and speed instrumentation are adequate to fulfill the intended purpose. These changes do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Sections sections 7.5.2.5, 7.5.2.55, and 56 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkg 97-50)

DESCRIPTION

SYSTEM: Low Pressure Injection (LPI), Reactor Coolant (RCS)

PIP 97-1801 documents this UFSAR change to Section 5.2.3.10.5, that was not included in a timely manner. Minor Modifications ONOE-8912, et al installed valves and modified piping to to vent the bonnets on LP-2 (second valve off RCS) to address pressure locking relief concerns of GL-95-07. This change eliminates the requirement for a two sets of packing with leakoff line for POVs containing reactor coolant in the RB.

SAFETY EVALUATION SUMMARY

These type modifications are an industry recognized method of alleviating pressure locking concerns (ref NUREG/CP-0146) and meeting GL 95-07 requirements. All Design Basis requirements are maintained and the LP-2 function is the same. While this change eliminates the requirement for a two sets of packing with leakoff line for POVs containing reactor coolant in the RB, there is NO increase in leakage. The leakage for both packing styles will be the same. The valves are located inside containment, thus any leakage would be to the RB sump. The activities performed do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR decribed accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new uncontrolled radiological release pathways, or failure modes are created. No USQ is involved with the modifications or corrsponding UFSAR change. No Technical Specification changes are required. UFSAR Section 5.2.3.10.5 was revised accordingly.

UFSAR CHANGES (Pkgs 97-51, 76, 99, & 115)

DESCRIPTION

SYSTEM: High Pressure Injection

PIPs 97-1801 and 98-0819 document UFSAR changes to Section 6.3.2.6.3 and Table 6-16, including some that were not included in a timely manner. Minor Modifications ONOE-8402, 9033, 9034, 9035, 9438, et al and NSM X2976 modified and replaced the valves/operators on HP-3, 4, 5, 26, 27. In each case safety evaluations were performed, but not all were reported in a timely manner. These modifications enhanced the capability of the valves/operators to have sufficient margin to correctly position the components in their safety positions during a Design Basis Accident or Event.

SAFETY EVALUATION SUMMARY

These valve/operator changeouts comply with GL 89-10 requirements All Design Basis requirements are maintained. The activities performed do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR decribed accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No USQ is involved with these modifications. No Technical Specification changes are required. UFSAR Chapter 6, Appendix 6, Table 6-16 was revised to reflect the new valve/motor characteristics.

UFSAR CHANGE (Pkg 97-53)

DESCRIPTION

SYSTEM: Penetration Room Ventilation System (PRVS)

1) UFSAR Section 3.8.1.7.5 was revised to remove the reference that operability of PRVS is verified by being placed into operation periodically to maintain satisfactory temperatures within the penetration rooms.

2) Section 9.4.5 was revised to remove all references to the recirculation operating mode of the Reactor Building Purge System.

3) Editorial changes were made to sections 9.4.1 & 9.4.3 concerning the AHUs descriptions.

SAFETY EVALUATION SUMMARY

Normal temperatures are maintained in the penetration rooms by the use of air handling units and chilled water system. PRVS is tested on a monthly basis to assure that the system is capable of performing its safety function. Although an NSM was installed several years ago to add recirculation capability to the system it was never used due to concerns with duct leakage. Elimination of this mode of operation has no impact on safety. These UFSAR changes do not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The PRVS and AHUs will continue to perform their design functions during normal and accident conditions. There is no physical changes to plant SSCs or operating procedures. No safety concerns or USQs are created by these revisions to UFSAR sections 3.8.1.7.5 and 9.4. No Technical Specifications changes are required.

UFSAR CHANGES (Pkg 97-55)

DESCRIPTION

SYSTEM: Nuclear Fuel

Calculation OSC-7035 contains a 10CFR50.59 evaluation and safety review for correcting UFSAR 4.2.2.1 and 4.2.4.3.3 text per PIP 97-3440. The change corrects the number of spacer grids per fuel assembly from 6 to 7 and clarifies that surveillance includes spacer grid "position" determination at Oconee.

SAFETY EVALUATION SUMMARY

The actual fuel assembly design including thermal and mechanical limits, and testing has not changed. These changes are clarifications to reflect the as-built as-tested fuel assembly. They do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Sections 4.2.2.1 and 4.2.4.3.3 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkgs 97-56)

DESCRIPTION

SYSTEM: Control Room Pressurization and Filtering System

This UFSAR change updates Section 2.2.3.1.3 to reflect current use and storage of hydrazine. Clarification was made to base the limits for on-site storage of this material from number of containers to number of equal pounds of product(hydrazine). The major SSC affected by this activity is Control Room Pressurization and Filtering System

SAFETY EVALUATION SUMMARY

The following accident and conditions were addressed in the safety review: hydrazine leaks from on site storage containers. There are no new or unanalyzed accidents or conditions created by this activity. This update reflects current use and storage of hydrazine and clarifies base the limits for on-site storage of this material from number of containers to number of equal pounds of product(hydrazine). It does not affect the ability of the any SSC to perform it's design function. The changes to reflect current use and storage of hydrazine and clarification to base the limits for on site-storage of this material from number of containers to an equal number of pounds of product does not change the ability to maintain a non-toxic environment in the control room. Typical numbers of tote bins of hydrazine stored on site are discussed and UFSAR section 2.2.3.1.3 states if leaks were to occur this should not result in dangerous concentrations in the control room. Changing the on-site storage criteria from number of tote bin/containers to a limit based on an equal amount of pounds of product will help ensure compliance with section 2.2.3.1.3. The total limit of hydrazine allowed on site has not been changed but an equivalent pounds of product(hydrazine) has been provided. The ability to isolate the control room from the outside environment and provide portable breathing air was not affected. The ability to isolate the control room from the outside environment and provide portable breathing air was not affected. The total limit of hydrazine allowed on site has not changed but an equivalent pounds of product(hydrazine) has been provided. There are no Technical Specifications changes required. There are no USQs or safety concerns. UFSAR section 2.2.3.1.3 was revised to reflect current operating practices.

UFSAR CHANGES (Pkg 97-58)

DESCRIPTION

SYSTEM: Switchyard

This evaluation provides clarification and enhancement to UFSAR Figure 8-1 Electrical Single Line Diagram. The following changes were made to reflect current configuration: (1) Add motor operated disconnect symbol for 525 KV switchyard, (2) add transformer size information to AT-1, and (3) show there is no Power Circuit Breaker (PCB-29) by having a solid piece of bus connect PCB-30 red bus diconnect to PCB-28 yellow bus disconnect. These changes make the drawing consistent with the information provided in Chapter 8.0.

SAFETY EVALUATION SUMMARY

These changes correct and clarify the SAR electrical diagrams and are largely editorial in nature. The activities performed do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR decribed accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There are no physical changes to the plant, operating procedures, or electrical lineups. There are no concerns which will affect safety related applications. This UFSAR change involves no safety concerns or USQs. No Technical Specification changes are required. UFSAR figure 8-1 was revised accordingly.

UFSAR CHANGES (Pkg 97-59)

DESCRIPTION

SYSTEM: Hydrogen Recombiners

As a result of calculations OSC-6926 and 6191 and PIP 96-2628, revisions were made to UFSAR section 15.16. The changes clarify the Hydrogen Recombiners are located in a mild environment, and therefore not within the scope of 10CFR50.49. Also, further changes were made to revise the calculations for hydrogen concentrations and times to reach certain concentrations in containment to reflect more realistic conditions.

SAFETY EVALUATION SUMMARY

This UFSAR update does not affect the design, operation or function of the components associated with the Reactor Building Hydrogen Recombiners or integrity of related systems, structures and components. The change performed does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR decribed accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. This change simply reflects the results of the new H2 generation/concentration calculations so that the UFSAR descriptions will be more accurate. This activity does not involve an Unreviewed Safety Question. No changes to the Technical Specifications are required. UFSAR Section 15.16 was revised accordingly.

UFSAR CHANGES (Pkg 97-60)

DESCRIPTION

SYSTEM: None

This 10CFR50.59 evaluation and safety review updated UFSAR sections 15.15.2 and 4 text per PIP 97-1801 to reflect the results of the latest offsite dose calculations.

SAFETY EVALUATION SUMMARY

The most recent offsite dose calculations were performed using the more modern dose conversion factors from ICRP-30 and updated flow uncertainties for the RB spray system. There is no physical change to the plant or procedures. These changes do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Sections 15.15.2 and 4 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkg 97-61)

DESCRIPTION

SYSTEM: High Pressure Injection, Reactor Coolant

This 10CFR50.59 evaluation and safety review updated UFSAR section 15.4.1 text per PIP 97-1801. The "CRD continuous boron dilute permissive" statalarm window on all 3 units was changed from "PERMISSIVE" to "INHIBIT" per ONOEs 8559 et.al. but the UFSAR was not updated in a timely manner to reflect such.

SAFETY EVALUATION SUMMARY

The "CRD continuous boron dilute permissive" is activated during normal power operations to indicate to the operator that boron dilution capabilites are available and HP-14 can be positioned to "bleed" at the same time HP-16 is open to permit faster turnover of dilution water into the RCS. The permissive is based upon position of control rod groups 1-6. Since the statalarm was a permissive, it actuated during normal operations. In order to provide the operators with a more logical alarm, the circuitry was changed to reflect when boron dilution is "inhibited". The inputs to the annunciator are non-QA. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Section 15.4.1 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-62)

DESCRIPTION

SYSTEM: Penetration Room Ventilation , Purge

This 10CFR50.59 evaluation and safety review for update UFSAR section 9.4.7.2, Table 6-9, and Figures 6-4, 6-25 per PIP 97-1801. Vacuum relief valves PR-112 and 113 were deleted from all 3 units per ONOEs 9470-9475, but the UFSAR was not updated in a timely manner to reflect the change.

SAFETY EVALUATION SUMMARY

The subject relief valves were designed to lift when vacuum pressure in the Pen Rooms exceeds 1.73 inches of water wrt the Purge Equipment Room. It was determined that due to system design and configuration, the purge system can only develop a vacuum pressure of 0.20 inches of water. The deletion of the valves does not affect the ability of the system to maintain a negative pressure in the Pen Rooms. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Section 9.4.7.2, Table 6-9, and Figures 6-4, 6-25 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkg 97-63)

DESCRIPTION

SYSTEM: Building Spray

This 10CFR50.59 evaluation and safety review update UFSAR Table 6-2, and Figures 6-2, 9-19 per PIP 97-1264. Valves BS-5 and 6 were deleted from all 3 units per ONOEs 7143,44, 7411,12 and 8086,87 and replaced with piping and flanges, but the UFSAR was not updated in a timely manner to reflect the change.

SAFETY EVALUATION SUMMARY

The subject check valves were designed to prevent reverse flow on the BS system when it is aligned to the HPI/LPI piggyback piping. The deletion of these valves improves the NPSH to the BS pumps and eliminate the optional BS lineup with HPI/LPI piggyback which is not required in any EOP. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. The BS, LPI, and HPI functions have not changed. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Table 6-2, and Figures 6-2, 9-19 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkg 97-65, 158)

DESCRIPTION

SYSTEM: Nuclear Fuel

Calculation OSC-6979 contains a 10CFR50.59 analysis and safety review for use of the Mk-B10L fuel assembly design (radial zoned fuel) at Oconee. Also see NRC SER of FCF Topical Report BAW-10186P "Extended Burnup Evaluation".

SAFETY EVALUATION SUMMARY

The new fuel assembly design is the same as the previous Mk-B10G except that a set pattern of rods have pellets with a lower enrichment in the active fuel region. The normal reload analyses verify the effects on power distribution and accident analyses are acceptable. The Mk-B10L will be enveloped by the current limits and SAR requirements. There are no external changes made to the assemblies that would interfere with control rod components or fuel handling equipment. This change does not increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. The fuel assembly functions have not changed. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. UFSAR text 3.1.6, 3.9.2.6, 4.2.1, 2, and 3, 4.5.1.2.3, Tables 3-24, 4-1, 2 and Figures 4-4, 18, and 37 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkg 97-66)

DESCRIPTION

SYSTEM: Nuclear Fuel

Calculation OSC-5864 performed a 10CFR50.59 analysis and safety review for use of the Mk-B10 Axial Blanket Fuel at Oconee. The associated UFSAR changes to Sections 4.3.2, 3, and Tables 4-2, 5 are per PIPs 97-0448 and 2511. Also see License Amendments 209/209/206 for Oconee Units 1, 2, and 3, respectively.

SAFETY EVALUATION SUMMARY

A new methodology for determining maximum fuel assembly burnups was approved by the NRC. The maximum enrichment limits were changed via Licensing amendment. This change updates the applicable UFSAR sections accordingly to be consistent with the Tech Specs. There are also several editorial type clarifications and corrections made to the UFSAR text. The normal fuel reload analyses still verify the effects on power distribution and accident analyses are acceptable. There are no external changes made to fuel assemblies that would interfere with control rod components or fuel handling equipment. This change does not increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. No new radiological release pathways, or failure modes are created. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No SSCs are degraded. The fuel assembly functions have not changed. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No new Technical Specification changes are required for this UFSAR update. There are no unreviewed safety questions or safety concerns. UFSAR text 4.3.2, 3, and Tables 4-2, 5 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkg 97-68)

DESCRIPTION

SYSTEM: Component Cooling (CC)

This 10CFR50.59 evaluation was performd per PIP 96-1151 to update UFSAR Figure 9-8 to reflect installation of ONOEs 9106-9113. Existing stop-check CC valves 25,29,33, and 37 are difficult to operate and were replaced by a two valve combination that perform the same functions. The new valves shown on the flow diagrams are designated CC-164,165,166,167.

SAFETY EVALUATION SUMMARY

The subject valve arrangement does not change the CC system functions or interactions. The seismic and environmental qualifications are maintained. The CC system still performs its design functions, with no adverse effects on the steady state or transient characteristics. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Figures 9-8 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-69)

DESCRIPTION

SYSTEM: Fire Protection

This 10CFR50.59 evaluation was performed to update UFSAR Table 9-12 and SLC Table 16.9-6 per PIP 97-1483 to reflect installation of a new fire detector in the Unit 3 Operator Aid Computer (OAC) room.

SAFETY EVALUATION SUMMARY

The subject changes simply update the affected tables to accurately reflect the quantity and location of fire detectors on Unit 3. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. UFSAR Table 9-12 and SLC Table 16.9-6 were updated accordingly.

UFSAR CHANGES (Pkg 97-70)

DESCRIPTION

SYSTEM: Coolant Sampling (CS)

This 10CFR50.59 evaluation and safety review were performed to update UFSAR Section 9.3 per PIP 98-0485. This revision corrected the list of auxiliary sample sinks to show the ones that are out-of-service and disconnected.

SAFETY EVALUATION SUMMARY

The subject items are permanently out-of-service because the boron recycle equipment is not used. at Oconee. This revision simply makes the UFSAR conform with the as-built plant configuration and current operating practices. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. The CS system functions have not changed. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Section 9.3 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-72)

DESCRIPTION

SYSTEM: None

This 10CFR50.59 evaluation and safety review was performed to update UFSAR Figure 2-5 "Radioactive Effluent Site Boundaries" per PIP 97-1801. This revision shows the existing site boundary fence in the area of Keowee Dam.

SAFETY EVALUATION SUMMARY

This revision simply corrects the affected UFSAR figure to agree with the actual site configuration. This change is non-technical in nature. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Figure 2-5 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-74)

DESCRIPTION

SYSTEM: Standby Shutdown Facility (SSF) HVAC

This revision to UFSAR sections 9.6.3.6.4 per PIP 97-0120, CA#3 is based on TT/0/A/0160/013 "SSF HVAC Fan Air Flow Test". This test verifes that SSF HVAC fan flows are adequate to maintain SSF areas within their acceptable temperature limits. The safety evaluation for the test was also utilized to add acceptable upper SSF temperature limits to the UFSAR.

SAFETY EVALUATION SUMMARY

This revision adds additional detail to the UFSAR in the form of acceptable upper SSF temperature limits to assure the HVAC testing is adequate. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Section 9.6.3.6.4 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-75)

DESCRIPTION

SYSTEM: Hydrogen Recombiners (CHRS)

This revision to UFSAR section 6.2, Tables 6-7 and Figures 6-9 and 15-110 are perfromed per PIP 97-1801 is based on NSMs X3008. These NSMs provided a permanent fix to restore the Units Containment H2 Recombiner Systems (CHRS) to a fully operable status.

SAFETY EVALUATION SUMMARY

This revision completes the UFSAR update for the subject NSMs thath was not performed in a timely manner. The temporary mods were removed and a QA-1 closed loop seismically qualified drainage system was installed on the CHRS piping. The new mods preclude the possibility of moisture buildup in the recombiner lines rendering the CHRS inoperable. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Section 6.2, Table 6-7 and Figures 6-9 and 15-110 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGE (Pkg 97-77, 114)

DESCRIPTION

SYSTEM: Fuel Tranfer Canal and Spent Fuel Pool (SFP)

This very minor change revised UFSAR Sections 9.1.4 and 15.11.1 by substituting a reference to the Core Operating Limits Report (COLR) in place of the existing specific boron concentration requirements for fuel transfer canal and fthe Spent Fuel Pool. The pool water boron concentration limits are cycle specific, and along with many other nuclear limits and setpoints, are contained in the controlled COLR.

SAFETY EVALUATION SUMMARY

Many nuclear parameters that are cycle dependent (startup physics information, tilt/imbalance limits, boron concentrations, etc.) are contained in the COLR. The COLR information, which is part of the SAR, is updated as necessary through out the core cycle. Since the detailed cycle information is available in the COLR, there is no value added by repeating the same information in the UFSAR. For the SFP and Fuel Tranfer Canal boron concentrations, a reference to the COLR is now provided. This UFSAR change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The SFP, RCS, and FTC will continue to perform their design functions during normal and accident conditions. There are no physical changes to the plant SSCs. No safety concerns or USQs are created by this largely editorial revision to UFSAR sections 9.1.4 and 15.11.1. No Technical Specification changes are required.

UFSAR CHANGES (Pkg 97-78)

DESCRIPTION

SYSTEM: Main Turbine

This UFSAR change is per PIP 97-1801 and reflects changes made via ONOEs 8571,72,73, some of which were not reported in a timely manner. These minor mods deleted the thrust bearing wear detector on the main turbines for Units 1,2,3.

SAFETY EVALUATION SUMMARY

Because of excessive unit trips, the secondary side was re-evaluated to determine where improvements could be made to reduce unnecessary reactor trips. These minor mods deleted the thrust bearing wear detector on the main turbines. The MT will still trip on low oil pressure. The MT trip is not required to safely shutdown the plant. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions. Section 10.4.6.5.1 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-79)

DESCRIPTION

SYSTEM: Main Turbine

This UFSAR change was performed per PIP 97-1801 and reflects changes made via ONOEs 8551, et al, some of which were not reported in a timely manner. These minor mods deleted the exhaust hood high temperature trips on the main turbines for Units 1,2,3.

SAFETY EVALUATION SUMMARY

Because of excessive unit trips, the secondary side was re-evaluated to determine where improvements could be made to reduce unnecessary reactor trips. These minor mods deleted the the exhaust hood high temperature trips on the main turbines because they were redundant to the vacuum trips. The MT trip is not required to safely shutdown the plant. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions. Section 10.4.6.5.1 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-82)

DESCRIPTION

SYSTEM: Low Pressure Service Water

This UFSAR change was performed per PIP 98-105 and is largely editorial. It updates and clarifies Sections 1.2.2.4, 3.1.1.1, 3.1.4, and Table 1-2 to be consistent with the more detailed descriptions given in 9.2.2.2.3.

SAFETY EVALUATION SUMMARY

This change simply clarifies that LPSW is an ES system, normally in operation, and is independent on each unit with the exception of Unit's 1 and 2 shared LPSW systems. LPSW also cools HPI pump motor coolers. This change is largely editorial in nature, reflects the as-built plant configuration and present operating practices, and helps make the UFSAR cosnistent throughout. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions. Sections 1.2.2.4, 3.1.1.1, 3.1.4 and Tale 1-2 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-85)

DESCRIPTION

SYSTEM: Reactor Vessel

This change to UFSAR section 4.4.3.3.5 is non-technical, editorial in nature, and simply clarifies that the vessel flow distribution described in the DNB analyses is based on "inlet" flow. It is reported for information only.

SAFETY EVALUATION SUMMARY

No 10CFR 50.59 evaluation was required for this largely editorial change that was covered by NRC SER on DPC-NE-2003, "Core Thermal - Hydraulic Methodology using VIPRE-01. UFSAR text 4.4.3.3.5 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-86)

DESCRIPTION

SYSTEM: Radwaste Facility (RWF)

This 10CFR50.59 evaluation and safety review updated UFSAR Section 11.3 per PIP 97-1189. This revision deleted the description of the RWF incinerator exhaust filtration process. The incinerator is not in use.

SAFETY EVALUATION SUMMARY

The subject incinerator has never been utilized at Oconee, and is out-of-service. Therefore filtration of the exhaust is not applicable. This revision will simply make the UFSAR conform with the actual plant configuration and current operating practices. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. Section 11.3 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-87)

DESCRIPTION

SYSTEM: None

The purpose of this evaluation was to determine if any unreviewed safety questions (USQs) were involved with a revision to UFSAR Sections 1.2, 1.3, and Table 1-3. UFSAR Section 1.2.2.3 was revised to include the reactor building's capability to withstand a spectrum of main steam line breaks. UFSAR Section 1.2.2.4 was revised to clarify the description of the Engineered Safeguards Systems. A editorial correction was made to Section 1.2.2.9 to correct a misspelled word. Section 1.3 and associated Table 1-3 were deleted because the information contained in this table and section are fully and more accurately addressed in other sections of the UFSAR.

SAFETY EVALUATION SUMMARY

The revision of this information in the UFSAR is to reflect licensing basis positions which already exist in other locations of the UFSAR. The revision to the UFSAR did not result in any plant modifications, procedure changes, or other activities which could result in an unreviewed safety question. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. No Selected Licensee Commitment changes were required. Sections 1.2, 1.3, and Table 1-3 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES. (Pkg 97-88)

DESCRIPTION

SYSTEM: Reactor Core

Several wording changes were made to Sections 3.1.29 and 3.1.30 per PIP 98-0058. These changes were made for clarification and had no effect on the meaning or intent of the discussion parts of these sections. The last sentence of the discussion in section 3.1.29 referees to a minimum shutdown margin for Oconee 1 of 5.5% $\Delta k/k$. It makes no reference to a particular cycle. Therefore, the last sentence was deleted and replaced with; "Table 4-6 illustrates a shutdown margin calculation for a sample Oconee fuel cycle." Table 4-6 shows how a shutdown margin calculation is performed but the numbers will be somewhat different for each cycle. The following underlined section was added to the discussion in section 3.1.30. The reactor meets this criterion with control rods for hot shutdown under normal operating conditions and for shutdown under the accident conditions set forth in Chapter 15, "Accident Analyses" <u>except for the Steam Line Break Analysis. For details of this analysis, refer to Section 15.13 "Steam Line Break Accident".</u>

SAFETY EVALUATION SUMMARY

Concerning Section 3.1.29, shutdown margin is cycle dependent and therefore one number cannot be accurate for each cycle. Shutdown margin is verified for each cycle by Nuclear Design by use of the methods referenced to in NFS-1001, Ononee Nuclear Station Reload Design Methodology. The NRC approved this methodology with a SER on July 29, 1981.

Section 3.1.30 was changed to reflect the analysis that is in Chapter 15. This analysis has always been in Chapter 15 and the NRC was aware of this analysis when the SERs for Oconee 1,2, and 3 were issued. The criteria for unit protection and the release of fission products to the environment are all met for the steam line break accident.

These changes do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. No Selected Licensee Commitment changes were required. Sections 3.1.29 and 3.1.30 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGE (Pkgs 97-90, 147, 148)

DESCRIPTION

SYSTEM: Reactor Core

A 10CFR50.59 safety review and USQ evaluation (OSC-7150) was performed for revisions to the Oconee UFSAR Sections 6.2, 6.3, and 15.14 and Technical Specification Bases 3.3 and 3.5.2. These revisions incorporated the methodology and results of the RELAP5/MOD2-B&W evaluation model that has been approved by the NRC.

SAFETY EVALUATION SUMMARY

The results of the large break loss of coolant (LBLOCA) analyses performed using the NRC-approved evaluation model show that all acceptance criteria continue to be met. No station procedures are affected and there is no physical change to the plant. This UFSAR change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The new analyses show that the required time for low pressure injection into the core, post LOCA/LOOP assuming a single failure, is now 53 versus 48 seconds. High pressure injection is still required in 48 seconds. However, there are no physical changes to the ES systems or their actuation setpoints, i.e. the ES systems capability has not changed. The cycle specific reactor cores continues to perform as designed during normal and accident conditions.

There are no unreviewed safety questions associated with the revisions to UFSAR Sections 6.2, 6.3, and 15.14 and Technical Specification Bases 3.3 and 3.5.2. No changes to the Technical Specifications are required.

UFSAR CHANGES (Pkg 97-95)

DESCRIPTION

SYSTEM: Reactor Vessel

This change to UFSAR sections 4.2.1.2.1 and 4.3.1 is made per PIP 98-0058 and simply clarifies that while codes and specific values provided in the Topical Report NFS-1001A "Reload Design Methodology" have been revised overtime, the methodology of designing to a DNBR limit has not. The SER for the above TR dated 7/29/81 is the basis for this change. It is reported for information only.

SAFETY EVALUATION SUMMARY

No 10CFR 50.59 evaluation was required for this change that was covered by the NRC SER on NFS-1001A. The SER describes how Duke designs cores to ensure a minimum DNBR, not absolute peak-to-average power distributions as was stated in the affected UFSAR text. Sections 4.2.1.2.1 and 4.3.1 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkg 97-98)

DESCRIPTION

SYSTEM: Incore Detectors

Section 7.6.2.3 of the UFSAR states that there is a calibration system that consists of detectors positioned by hand in selected assembly calibration tubes. This system was installed in Unit 1 to provide on line confirmation of the experimentally derived compensation calculation methods. This calibration system is no longer used and these statements are being deleted from the UFSAR per PIP 97-3263.

SAFETY EVALUATION SUMMARY

Calibration of detectors is not required because experimental programs have determined the magnitude of the calibration factors, and these have been incorporated into calculations to correct the output of the detectors. There are no commitments for this calibration system. The incore detectors are not safety related. This UFSAR update does not affect the design, operation or function of any components associated with the Incore Instrumentation System, or the integrity of any related plant systems, structures and components. The change performed does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. This change simply reflects the actual plant configuration, so that the UFSAR descriptions will be more accurate. This activity does not involve an Unreviewed Safety Question. No changes to the Technical Specifications are required. UFSAR Section 7.6.2.3 was revised accordingly.

UFSAR CHANGES (Pkg 97-100)

DESCRIPTION

SYSTEM: Spent Fuel

This UFSAR change is per PIP 97-1801. UFSAR sections 1.2.2.8, 9.1.4.2.3, 15.11.3 were revised to reflect dry storage construction. NSM-52959 added the third phase of horizontal storage modules (HSMs) to provide for continued dry storage of spent fuel discharged from the Oconee reactors. The scope of this NSM was receipt, placement, and alignment of twenty horizontal storage modules (HSMs) of the NRC approved VECTRA General License "Standardized" NUHOMS 24-P design. This evaluation addresses Parts BS-1 & BL-1 of NSM-52959, which provides for installation of the first eight modules complete with temperature monitoring equipment. The new HSMs were pre-fabricated at Bayshore Concrete Co., shipped to Oconee, assembled onsite and placed on a permanent storage pad location already constructed within the existing ISFSI boundaries.

SAFETY EVALUATION SUMMARY

The new generic license dry storage system is similar to the Oconee site specific system (License SNM-2503), and can utilize the existing fuel handling equipment, dry storage canisters (DSC) design, transport/loading equipment, and site location. For the Phase III GL design, Duke utilized its QA-4 designation for the HSMs. Although some of the license conditions may differ between the site specific and General License systems, there is no conflict since each system will be treated as a separate entity, both procedurally and in licensing space. These changes do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No USQs are involved with either the modification or corresponding UFSAR change, and no Technical Specification changes are required. The plant UFSAR sections 1.2.2.8, 9.1.4.2.3, 15.11.3 were revised, accordingly.

UFSAR CHANGES (Pkg 97-102)

DESCRIPTION

SYSTEM: Hydrogen (H2) Control

This UFSAR change is per PIP 97-1062. UFSAR figures 6-4 and 15-110 were revised to remove RB H2 Purge System. Conservative calculations have shown the dose rates in the vicinity of the H2 purge cart to be prohibitively high during a design basis event. Also see change 97-59.

SAFETY EVALUATION SUMMARY

Adequate containment H2 control still exists to keep the concentration below the flammability limits via the Containment H2 Recombiner System (CHRS). Thus, the removal of the purge cart is of no consequence. These UFSAR changes do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No USQs are involved with either the modification or the corresponding UFSAR change, and no Technical Specification changes are required. The plant UFSAR Figures 6-4 and 15-110 were revised, accordingly.

UFSAR CHANGES (Pkg 97-105)

DESCRIPTION

SYSTEM: Fire Protection

This UFSAR change is per PIP 97-38. UFSAR section 9.5.1.6.19 was revised to denote the Radwaste Facility (RWF) foam fire suppression system has been abandoned.

SAFETY EVALUATION SUMMARY

The RWF polymer fill station was never placed in service at Oconee. The machinery is de-energized, isolated, and abandoned in place. The respective foam fire suppression system has also been abandoned because there is no fire hazard present. These UFSAR changes do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No USQs are involved with either the abandonment or the corresponding UFSAR change, and no Technical Specification changes are required. UFSAR Section 9.5.1.6.19 was revised, accordingly.

UFSAR CHANGES (Pkg 97-108)

DESCRIPTION

SYSTEM: none

This change to the Oconee UFSAR Section 12.4.3 is to add descriptions of the Reactor Coolant Pump Motor Refurbishment Building (RCPMRB) and the Carbon Dioxide Blast Facility. The Reactor Coolant Pump Motor Refurbishment Building is available for maintenance activities that will not release uncontrolled airborne radioactivity to the environment. The Carbon Dioxide (CO2) Blast Facility is available for decontamination activities that will not release uncontrolled airborne radioactivity to the environment.

SAFETY EVALUATION SUMMARY

Controls are imposed by the radiological procedure governing the work to ensure that uncontrolled airborne radioactivity is not released to the environment from the RCPMRB. The radiological control procedure specifies conditions under which work can be performed in an enclosure with a HEPA-filtered exhaust. The HEPA-filtered exhaust will be monitored for the discharge of radioactivity during periods of HEPA system operation.

Controls are imposed by the radiological procedure governing the decontamination work to ensure that uncontrolled airborne radioactivity is not released to the environment from the CO2 Blast facility. The blast facility is housed within a building that does not exhaust to the environment. Additionally during periods of operation, the process is exhausted through a HEPA filtration unit, to the outer facility. The HEPA-filtered exhaust is constantly monitored for the discharge of radioactivity during periods of HEPA system operation.

Adding this information to the UFSAR does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. The subject addition has been evaluated and has been found to have no impact on either accident analyses or equipment important to safety. The subject additions to Oconee UFSAR 12.4.3 involve no USQs. No changes to Technical Specifications are required. UFSAR Section 12.4.3 was revised accordingly.

UFSAR CHANGES (Pkg 97-110)

DESCRIPTION

SYSTEM: None

This change to UFSAR Section 3.9.2.1 "Piping Vibration, Thermal Expansion, and Dynamic Effects." is performed per PIP 96-1906. Section 3.9.2.1 of the Oconee UFSAR describes specific testing and analysis done for the initial startup of the plant. This change intends to clarify that the actions described were for the initial startup only. The text in section 3.9.2.1 of the current UFSAR was first added to revision 24 of Oconee's FSAR. In FSAR Supplement 12 (dated 7/26/72), Duke responded to several questions asked by the Atomic Energy Commission. Question 8 in FSAR Supplement 12 states in part "Paragraph 1701.5.4 of the ANSI B31.7 Nuclear Power Piping Code requires that piping shall be supported to prevent excessive vibration under startup and initial operating conditions." Duke's response to this question was "See FSAR Section 1C.3.7". This section in the 24th FSAR revision is now UFSAR Section 3.9.2.1. It is evident from this correspondence that the information in UFSAR Section 3.9.2.1 applied to the specific evolution of initial startup. A thorough review of Oconee licensing commitments was performed in order to ensure that no other licensing commitment was affected by this change.

SAFETY EVALUATION SUMMARY

This UFSAR change is technical and editorial in nature. The change performed does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR decribed accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This UFSAR change involves no USQ's. No changes to the Technical Specifications are required. UFSAR Section 3.9.2.1 was revised accordingly.

UFSAR CHANGE (Pkg 97-113)

DESCRIPTION

SYSTEM: Radwaste

This change to USFAR chapter 11, sections 11.2.2.2, 11.2.2.3, Table 11-1 and Table 11-5 was made per PIP 97-709 to clarify current operating practices for liquid waste processing equipment. Information was added to Table 11-1 to clarify that this information identifies potential sources of liquid waste and that actual quantities of liquid waste are reported in the Oconee Annual Effluent Report. Table 11-5 is being revised to include data which was omitted during a previous revision.

SAFETY EVALUATION SUMMARY

The intention of this largely editorial change is to correct, enhance and expand the Radwaste Systems descriptions contained in the UFSAR. Changing this information in the UFSAR does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This revision does not in any way change the physical characteristics of the Station or its operations. No safety concerns or unreviewed safety questions are created. No Technical Specification changes are required. USFAR Chapter 11, sections 11.2.2.2, 11.2.2.3, Table 11-1 and Table 11-5 were updated accordingly.

UFSAR CHANGES (Pkg 97-117)

DESCRIPTION

SYSTEM: RCS

This activity performed per PIP 97-3724 provides a more detailed description of the existing information presented in UFSAR Section 5.2.1.2 to more accurately reflect the as-built design values provided in the Oconee Flow Diagrams. The change updates the design temperature conditions for the pressurizer and other RCS components to be consistent with the Tech Specs.

SAFETY EVALUATION SUMMARY

This update is largely clarification, and does not affect the design, operation or function of the components associated with the pressurizer or integrity of related RCS systems, structures and components. This change will not increase the consequences, probability, possibility of any SAR described, or different type of, accident occurring. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR, and requires no Technical Specification changes. There are no unreviewed safety questions associated with the updates to Section 5.2.1.2 of the Oconee UFSAR.

UFSAR CHANGES (Pkg 97-119

DESCRIPTION

SYSTEM: Electrical 230KV

This activity performed per PIP 97-1801 updates UFSAR section 8.2.1.3.1. and Table 3-68 for NSM-52950 changes that were not incorporated in a timely manner. The NSM added three Capacitor Coupled Voltage Transformers and relays in the 230 KV switchyard for the degraded grid protection system. provides a more detailed description of the existing

SAFETY EVALUATION SUMMARY

The new Capacitor Coupled Voltage Transformers and relays are QA Condition 1 seismically mounted components that will reduce inaccuracies in the voltage measurements. The relays will still provide the same functions of alarm indication and switchyard isolation. Part of the change is editorial in nature. This change will not increase the consequences, probability, possibility of any SAR described, or different type of, accident occurring. No new radiological release pathways, malfunctions, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR, and requires no Technical Specification changes. There are no unreviewed safety questions associated with the updates to Section 8.2.1.3.1 and Table 3-68 of the Oconee UFSAR.

UFSAR CHANGES (Pkg 97-120)

DESCRIPTION

SYSTEM: Auxiliary Electrical

This activity performed per PIP 96-1779 updates UFSAR section 8.3.1.1.3 for ONOEs-7442,7447,7478 changes that were not incorporated in a timely manner. These minor mods added an auxiliary relay to the 4KV startup incoming breakers which seals in the existing fast bus transfer permissive time delay relays. Additionally, the slow transfer time delay setting was increased from 1 to 1.3 seconds to assure exceesive voltages are not applied till loads and systems recover following a slow aux power transfer.

SAFETY EVALUATION SUMMARY

The new aux relays are QA Condition 1 components. The mods enhance existing and assure time delsy relays will remain sealed in for all PCB-21 trips. The operational logic of the system is not being changed. The functions of the onsite emergency power systems are not adversely affected. This change will not increase the consequences, probability, possibility of any SAR described, or different type of, accident occurring. No new radiological release pathways, malfunctions, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR, and requires no Technical Specification changes. There are no unreviewed safety questions associated with this update to Section 8.3.1.1.3 of the Oconee UFSAR.

UFSAR CHANGES (Pkg 97-121)

DESCRIPTION

SYSTEM: Reactor Building

UFSAR sections 3.8.1.4, 3.8.1.5.2, and 3.8.1.7 were revised to update and correct RB structural integrity and tendon surveillance descriptions.

SAFETY EVALUATION SUMMARY

Reanalysis was used to establish minimum required lift-off values (MRV's) and were used in the establishment of predicted lift-off values (called Precsribed Lower Limits or PLL's). These MRV's and PLL's were included in SLC 16.6.2. A 50.59 evaluation was previously done on the addition of these values into Chapter 16. An improved tendon sheath filler was addressed in a 50.59 associated with the creation of Maintenance Procedure MP/0/A/1400/34. Tendon inspection procedures and are in accordance with an SER related to Amendment No 225 to FOL DPR-38, Amendment No 225 to FOL DPR-47, and Amendment No 222 to FOL DPR-55, OconeeUnits 1,2, 3 Tech Specs. The liner plate inspection change is editorial in that Tech Spec 4.4.1 already requires liner plate inspection in accordance with 10CFR50, Appendix J. Updating this RB information in the UFSAR does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This revision does not modify the physical plant or operating procedures. There are no safety concerns or USQs ivolved with this change. No Technical Specification changes are required. Applicable portions of UFSAR Sections 3.8.1.4, 3.8.1.5.2, and 3.8.1.7 were revised accordingly.

UFSAR CHANGES (Pkg 97-122)

DESCRIPTION

SYSTEM: Fire Protection

This UFSAR change is per PIP 96-2076. UFSAR section 8.3.1.4.6.2 was revised to clarify the description of fiberglass reinforced polyester barriers used in cable trays.

SAFETY EVALUATION SUMMARY

This clarification addressed use of the fiberglass polyester as an insulator and protection mechanism. However, no credit is taken for it and removal would not affect Oconee compliance with 10 CFR 50 Appendix R. This UFSAR change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents, including fires. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No SSCs are degraded. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No USQs are involved, and no Technical Specification changes are required. UFSAR Section 8.3.1.4.6.2 was revised, accordingly.

UFSAR CHANGES (Pkg 97-126)

DESCRIPTION

SYSTEM: Reactor Core

FCF identified a calculation inconsistency between the Mark-B fuel assembly horizontal faulted condition analyses and Emergency Core Cooling System (ECCS) calculations specific to the requirements of 10CFR50.46 (PSC-21-96-5). FCF re-analyzed this condition in BAW-2292P-A, Rev. 0. FCF determined that substantial margin exists between the applied load and the grid elastic load limit (control rod insertion will not be hindered under any faulted condition); therefore, control rod insertability is ensured. This analysis applies to all B&W designed plants (with skirt-supported and nozzle supported reactor vessels) fueled with Framatome Mark-B type fuel assemblies. Leak-before-break (LBB) analyses were used to establish the design breaks and the resulting reactor internals loads and displacement time histories, and fuel assembly impact loads. The results are applicable to Oconee Nuclear Station for any Mark-B type fuel assembly. BAW-2292P-A is a generic report written for the B&W Owner's Group. The information from BAW-2292P-A was incorporated into the UFSAR to depict the current licensing basis. An evaluation (OSC-6684, Rev. 1, "50.59 Revision to the Oconee FSAR, Chapter 3 to Reference Proper Licensing Methodology for Dynamic Analysis of Fuel Assemblies", Mar. 1998) was performed to review the Oconee SAR documents in order to update the applicable portions.

SAFETY EVALUATION SUMMARY

Changing this information in the UFSAR does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This revision does not modify the physical plant or operating procedures. FCF demonstrated that substantial margin exists between the applied load and the grid elastic load limit (control rod insertion will not be hindered under any faulted condition). Therefore, control rod insertability is ensured and the requirements of 10CFR50.46 are met. No technical specification change is required for this methods revision. The limits specified in the COLR were not affected by this methods revision. No licensing commitments were affected. The Duke methodology topical reports were not affected. There are no safety concerns or USQs involved with this change. No Technical Specification changes are required. Applicable portions of the UFSAR were revised (Sections 3.9.2.4, 3.9.6, 4.5.1.2, 4.5.5, 5.2.1.4, and 5.2.4, and Table 3-24).

UFSAR CHANGES (Pkg 97-127)

DESCRIPTION

SYSTEM: Reactor Building Normal Sump (RBNS)

This activity performed per PIP 97-3724 provides accurate detail to existing information on RBNS volumes gal/in and time frames for control room alarm indications presented in UFSAR Section 5.2.3.10.3 to more accurately reflect data from control documents.

SAFETY EVALUATION SUMMARY

This UFSAR update does not affect the design, operation or function of the components associated with the Reactor Building sump and the letdown storage tank or integrity of related systems, structures and components. The change performed does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR decribed accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. This activity does not involve an Unreviewed Safety Question. No changes to the Technical Specifications are required. UFSAR Section 5.2.3.10.3 was revised accordingly.

UFSAR CHANGE (Pkg 97-128)

DESCRIPTION

SYSTEM: None

UFSAR Section 2.5.2.1 and Table 2-94 "Significant Earthquakes in the Southeast United States (Intensity V or Greater)" were updated per PIP 97-1277 to reflect recent earthquake activity and clarify the table information. The table was revised to provide additional seismic activity since 1979. Company geologists conducted the research and provided the updated information to the UFSAR subsection owner. This UFSAR change to Table 2-94 provides current available updated seismic activity. The text within Section 2.5.2.1 was revised to reflect the following clarifying information that Figure 2-48 is a plot of the more significant earthquake shocks which occurred prior to 1961, and that the information provided in Table 2-94, described above, is based on the data available at the time of this UFSAR update. The information to be added to Section 2.5.2.1 will enhance the UFSAR readers understanding of information provided in Figure 2-48 and Table 2-94.

SAFETY EVALUATION SUMMARY

Addition of the latest available seismic Southeastern US seismic activity data to the UFSAR does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. The new data does not reveal trends which would cause reevaluation of the seismic design basis of the site. These largely editorial UFSAR revisions do not involve any physical changes to the facility or operating procedures as described in the SAR, nor do they alter the design bases.

This UFSAR revision involves no USQs, no safety concerns, nor any changes to the Technical Specifications. The wording within UFSAR Section 2.5.2.1 was revised for clarification and Table 2-94 updated.

UFSAR CHANGES (Pkg 97-130)

DESCRIPTION

SYSTEM: none

Currently, the UFSAR does not address the labeling/posting, control and storage of radioactive material at Oconee Nuclear Station. See Problem Investigation Report # 4-0-98-1223. Oconee currently stores radioactive material within the Radiation Control Area (RCA), within radiation control zones located inside the Restricted Area and within RCZs located within the Owner Controlled Area. In all cases, the material is labeled/posted, controlled and stored according to approved procedures and directives that meet the requirements of 10CFR20.

SAFETY EVALUATION SUMMARY

This UFSAR update activity does not involve any change to the facility and has no potential for impact on any SSC that is important to the safe operation of the station. Radioactive material consists of tools, equipment, parts and radioactive waste products generated during station operation. They may be low or high radioactivity material; however, storage requirements are indicative of the hazards involved in each case (i.e., the higher the activity, the higher the integrity of the required storage container). There is no physical change to the facility; no potential for degradation of any SSCs. This change is not an accident initiator and does not introduce any new failure modes or mechanisms. This change is largely editorial in nature to add information, and does not involve installation of any new equipment or components; does not create any new radiological release pathways from station systems; does not increase the probability that a release will occur because storage practices do not change; has no effect on the reactor coolant system, containment, filtration, radwaste systems, or main steam relief valvessetpoints, and does not change the current and previous radioactive material storage practices at Oconee.

There are no safety concern or USQs involved with this change. No Technical Specification changes arerequired. UFSAR Section 12.4.3 was revised accordingly.

UFSAR CHANGES (Pkg 97-131)

DESCRIPTION

SYSTEM: Nuclear Fuel

UFSAR Sections 4.2.1.2.1 and 4.3.1 were revised to reflect current core design methodology. Provided for information only.

SAFETY EVALUATION SUMMARY

By SER dated 4/3/95 the NRC approved Dukes use of the TACO-3 Fuel Performance Code. These minor changes to the UFSAR are consistent with that approval, so no 10 CFR 50.59 evaluation was required. The applicable text in sections 4.2.1.2.1 and 4.3.1 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkg 97-132)

DESCRIPTION

SYSTEM: Nuclear Fuel

UFSAR changes to sections 4.1, 4.2.1, 2, 3, 5, 4.4.5, Table 4-14 and Figure 4-18 are based NRC SERs (Reload Design Methodology II and Core TH Methodology using VIPRE-01) and Duke Calculation OSC-6526 which contains a 10CFR50.59 analysis and safety review for use of APSRs with extended hubs. These revisions reflect changes in fuel assembly and component design and T-H calculation methods.

SAFETY EVALUATION SUMMARY

The affected UFSAR sections were updated to reflect current T-H methodologies and fuel assembly/component design parameters. There are also several editorial type clarifications and corrections made to the text. The normal fuel reload analyses still verify the effects on power distribution and accident analyses are acceptable. There are no external changes made to fuel assemblies that would interfere with control rod components or fuel handling equipment. This change does not increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. No new radiological release pathways, or failure modes are created. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No SSCs are degraded. The fuel assembly functions have not changed. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No new Technical Specification changes are required for this UFSAR update. There are no unreviewed safety questions or safety concerns. UFSAR sections 4.1, 4.2.1, 2, 3, 5, 4.4.5, Table 4-14 and Figure 4-18 of the Oconee UFSAR were updated accordingly.

UFSAR CHANGES (Pkg 97-133)

DESCRIPTION

SYSTEM: None

Non technical changes were made to UFSAR Section 13.1, and Figures 13-1, 2, 3, 4, & 7 were made to reflect the new Duke Energy Corporation Organization. Provided for information only.

SAFETY EVALUATION SUMMARY

The new organization was approved by NRC License Amendment 226/226/223. Therefore, no 10CFR 50.59 safety evaluation is required.

UFSAR CHANGES (Pkg 97-134)

DESCRIPTION

SYSTEM: RCS

This activity performed per PIP 97-3724 provides a more accurate description and clarification of existing information presented in UFSAR Tables 5-22. A note was added to explain why the reference frame for the operating pressure of the pressurizer (2166 psig) is different than the RCS operating pressure (2155 psig) given in Table 5-1.

SAFETY EVALUATION SUMMARY

This clarification is largely editorial, and does not affect the design, operation or function of the components associated with the pressurizer or integrity of related RCS systems, structures and components. The RCS pressure transmitters located near the top of the hot-legs are measuring the pressure including effects of elevation and frictional pressure losses. This change will not increase the consequences, probability, possibility of any SAR described, or different type of, accident occurring. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR, and requires no Technical Specification changes. There are no unreviewed safety questions associated with the update made to Table 5.-22 of the Oconee UFSAR.

UFSAR CHANGES (Pkg 97-135)

DESCRIPTION

SYSTEM: Keowee Fire Protection

This activity is a revision to the Updated Final Safety Analysis Report (UFSAR), Chapter 9, Section 9.5.1, "Fire Protection System". This section describes the Keowee Hydro Station fire protection features. The added descriptions discuss Keowee Hydro Station's remote location and thus how Oconee equipment is not directly affected by fires at Keowee. Station equipment such as fire detectors, the fire pump, lighting, the water supply system, generator CO_2 system, sprinklers, hose stations, and deluge systems are described. This information gives the UFSAR reader a better understanding of Keowee fire protection systems and what systems are required for UFSAR Chapter 16 (Selected Licensee Commitments) surveillance. The Selected Licensee Commitments (SLC), Section 16.9 address Keowee surveillance on fire protection equipment.

SAFETY EVALUATION SUMMARY

The UFSAR change performed does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR decribed accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. Due to the changes being descriptions of existing physical characteristics and attributes of Keowee Hydro Station with regards to fire protection equipment and fire mitigation aids, this change is largely editorial and descriptive, not physical. There are no adverse affects on any plant SSC. There is no change to the method of operation, nor changes to testing, nor addition or removal of equipment. There are no changes required in the Technical Specifications. No changes in UFSAR Chapter 9 affect the surveillance in Chapter 16. There are no USQs or safety concerns related to this change to UFSAR Section 9.5.1.

UFSAR CHANGES (Pkg 97-136)

DESCRIPTION

SYSTEM: LPI

A 50.59 evaluation has been performed for the UFSAR changes to Section 6.3.3.2.1, "Boron Precipitation Evaluation,." and Table 6-20 per PIP 98-1592. A 50.59 evaluation was performed to determine the presence or absence of any unreviewed safety questions for this update to the Oconee UFSAR. Several changes were editorial in nature. One change, increasing the assumed beginning of cycle RCS boron concentration limit from 2000 to 2100 ppm, was technical in nature, and thus was the focus of the 50.59 evaluation.

SAFETY EVALUATION SUMMARY

Increasing the boron limit was the result of recent cycle designs which have had a 0 EFPD critical boron concentration in excess of 2000 ppm (specifically O2C17). The salient accident which is associated with this change is the post-LOCA core boron precipitation analysis. This change does not result in any changes to the post-LOCA plant response for boron precipitation concerns. Specifically, the alignment of the hot leg drop line dump-to-sump flow at 9 hours precludes the core boron concentration from reaching the solubility limit. Any pH concerns associated with an increased RCS boron concentration are addressed by Oconee Chemistry on a cycle-specific basis. Changing the assumed beginning of cycle RCS boron concentration limit did not result in any Technical Specification changes or procedure revisions. This change also did not increase the consequences, probability, possibility of a different type of accident occurring, or reduce the margin of safety. Therefore, there were no unreviewed safety questions associated with the updates to these sections of the Oconee UFSAR.

UFSAR CHANGES (Pkg 97-137)

DESCRIPTION

SYSTEM: Electrical Inverters

This activity performed updates UFSAR section 8.3.2.1.5 for ONOE-8487. This minor mod deleted the auto/manual retransfer toggle switch from the KU backup transfer switch to the normal source. During the mod review process, a UFSAR omission was identified in that the Aux power and computer systems are not included with ICS as having an auto transfer switch in addition to a static switch.

SAFETY EVALUATION SUMMARY

The essential inverter systems are not safety related and are not required for safe shutdwon. The operation of the switch has not changed. This change is largely editorial clarification.This change does not increase the consequences, probability, possibility of any SAR described, or different type of, accident occurring. No new radiological release pathways, malfunctions, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR, and requires no Technical Specification changes. There are no unreviewed safety questions associated with this update to Section 8.3.2.1.5 of the Oconee UFSAR.

UFSAR CHANGE (Pkg 97-138)

DESCRIPTION

SYSTEM: None

UFSAR Section 2.1 contains population density information about the area surrounding the Oconee Nuclear Site. PIP#0-O96-1780 identified the need for up to date population information in section 2.1. In order to satisfy corrective action 3 of PIP#0-O96-1780, the UFSAR was revised to add information referencing the licensing document containing the latest population statistics.

SAFETY EVALUATION SUMMARY

Section 2.1 of the UFSAR provides population data for 1970 and projected population data information to 2010. Actual population data is subject to constant change. This revision to UFSAR Section 2.1 makes reference to the Oconee Nuclear Station Emergency Plan which provides a more up to date source of population data. The intention is to enhance and expand the usefulness of the UFSAR. Addition of this information to the UFSAR does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. This revision does not in any way change the physical characteristics of the station or its operations. No unreviewed safety questions are created as a result of this revision to Section 2.1. No Technical Specification changes are required.

UFSAR CHANGES (Pkg 97-139)

DESCRIPTION

SYSTEM: High Pressure Injection

This 10CFR50.59 evaluation was for updating UFSAR Section 9.3.2.2.4 and Table 6-4 to reflect installation of ONOEs 7308 - 7313 and 7284, 85, et al. The existing stop-check valves HP1-144, 145, 146, 147 are obsolete and were replaced by a two valve combination that performs the same functions.

SAFETY EVALUATION SUMMARY

The subject valve arrangement did not change the HPI system functions or interactions. The seismic and environmental qualifications are maintained. The HPI system continues to perform its design functions, with no adverse effects on the steady state or transient characteristics. This change does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity also has no effect on any margins of safety as previously evaluated in the SAR. No Technical Specification changes are required. There are no unreviewed safety questions or safety concerns. UFSAR Section 9.3.2.2.4 and Table 6-4 of the Oconee UFSAR was updated accordingly.

UFSAR CHANGES (Pkg 97-144)

DESCRIPTION

SYSTEM: LPI, RCS

PIP 98-819 documents this UFSAR change to Tables 5-21, 5-22, and Figure 9-20, that were not included in a timely manner. NSMs-X2927 modified the RCS to add a dedicated shutdown vent line to the quench tank (QT). A new vent line was also installed on the pressurizer into the new A Steam Generator Line.

SAFETY EVALUATION SUMMARY

These modifications were fully evaluated under 10CFR 50.59 prior to installation. However, the UFSAR changes identified by the evaluation were not incorporated at the time. The High Point Vent system safety related functions were unaffected by the mods. The activities performed do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR decribed accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new uncontrolled radiological release pathways, or failure modes are created. No USQ is involved with the modifications or corresponding UFSAR change. No Technical Specification changes were required. UFSAR Tables 5-21, 5-22, and Figure 9-20 were revised accordingly.

UFSAR CHANGES (Pkg 97-149)

DESCRIPTION

SYSTEM: ESPS

The following change is being made to the UFSAR per PIP 98-1631. Engineered Safeguards Protective System (ESPS) UFSAR section 7.3.3.4 stated that the set points of the pressure switches used for ESPS channels 7 and 8 may be checked by connecting a source of pressure and a pressure gauge to the pressure transmitter connections provided inside the Reactor Building and that this check may be made regardless of reactor power when access to the building is attained. All test connections used to calibrate ESPS channels 7 and 8 pressure switches and ESPS channels 5 and 6 building pressure transmitters are located in the penetration rooms. Therefore, the UFSAR statement was revised to reflect the removal of exact test tee locations for checking the set points and references to checks made when the reactor building is accessible was removed.

SAFETY EVALUATION SUMMARY

No physical changes to any SSCs were made as a result of this revision. ESPS continues to meet all design criteria and functions as before. This change is for documentational purposes, and is editorial in nature. This UFSAR change does not cause, or adversely affect the mitigation of, any prevously analyzed SAR accidents. No new radiological release pathways or failure modes are created, and no SSCs are degraded. There are no physical changes to the plant or operating procedures. This change create no USQs or safety concerns and no technical specification changes are required. UFSAR section 7.3.3.4 was revised accordingly.

UFSAR CHANGES (Pkg 97-152)

DESCRIPTION

SYSTEM: Vital Power, Control Batteries

PIP 98-819 documents a UFSAR change to Figure 8-4, 6, & 7, that were not included in a timely manner. NSMs-X2881 replaced vital and essential inverters and control battery AC feeder breakers.

SAFETY EVALUATION SUMMARY

These modifications were fully evaluated under 10CFR 50.59 prior to installation. However, the UFSAR changes identified by the evaluation were not incorporated at the time. The changes were essentially like for like components, with the obsolete components being changed out for modern components that perform the same functions. The batteries, inverters, and chargers qualifications and safety functions were all retained. The activities performed do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR decribed accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new uncontrolled radiological release pathways, or failure modes are created. No USQ is involved with the modifications or corrsponding UFSAR change. No Technical Specification changes are required. UFSAR Figures 8-4, 6, and 7 were revised accordingly.

UFSAR CHANGES (Pkg 97-153)

DESCRIPTION

SYSTEM: Turbine Bypass (TB)

PIP 98-531 documents this UFSAR change to Section 7.6.1.2.2.2. A statement that the TB system permits load drop of 40%, or turbine trip at 40% load without safety valve operation, is incorrect because plant modifications over the years have changed out and added new equipment, ie Anticiaptory Reactor Trip System installed , TB valves and ICS replaced.

SAFETY EVALUATION SUMMARY

The ARTS was installed to reduce the challenges to RCS overpressure caused by large secondary side transients. The new TBVs exhibit a slower response time. The new ICS is not designed for 40% load drop without steam relief. All these modifications were fully evaluated under 10CFR 50.59 prior to installation, and appropriately revised the level of detail in the UFSAR. The activities performed do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR decribed accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new uncontrolled radiological release pathways, or failure modes are created. No USQ is involved with the modifications or corrsponding UFSAR change. No Technical Specification changes are required. UFSAR Section 7.6.1.2.2.2 was revised accordingly.

UFSAR CHANGES (Pkg 97-154)

DESCRIPTION

SYSTEM: None

This USQ evaluation was performed per PIP 96-2627 for a revision to Table 5-2 of the UFSAR to reflect a reduced number of thermal cycles used to calculate the fatigue life of certain portions of the RCS. This reduction resulted from flaw tolerance evaluations performed for certain anomalies detected during In-Service Inspections that considered fewer additional cycles than necessary to achieve the original number of thermal cycles. The reduced number of cycles have been reflected in the Allowable Operating Transient Cycles (AOTC) program log books kept for Units 1 and 2 (there presently are no flaw tolerance evaluations for Unit 3 utilizing fewer cycles than contained in the RCS Functional Specification). In this manner, the number of transient cycles experienced by the plant are maintained within the numbers evaluated in the calculations that demonstrate code compliance.

SAFETY EVALUATION SUMMARY

The "limiting" flaw evaluation on Unit 1 considers a total of 207 heatups and cooldowns versus the RCS Functional Specification value of 360 (the actual number of logged heatups is 100 and cooldowns is 101 as of April 1998). The "limiting" flaw evaluation on Unit 2 considers a total of 330 heatups and cooldowns versus 360 in the RCS Functional Specification (the actual number of logged heatups is 113 and cooldowns is 114 as of April 1998). The change performed does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR decribed accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. The answer to the seven USQ Evaluation questions is NO. This activity does not involve an Unreviewed Safety Question. No changes to the Technical Specifications are required. UFSAR Table 5-2 was revised accordingly.

UFSAR CHANGE (Pkg 97-156)

DESCRIPTION

SYSTEM: Nuclear Fuel

Calculation OSC-7155 performed a 10CFR 50.59 evaluation of Fuel in Compression Ratio. Framatome performed a new series of experiments to re-evalaute the cladding ductility. As a result FCF removed the RCS cooldown criteria for fuel in compression. UFSAR section 4.2.3.1.1 was impacted.

SAFETY EVALUATION SUMMARY

The fuel rod in compression ratio was used to minimize the formation of radially oriented hydride platelets by limiting the clad tensile stress. This UFSAR change does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No potential loose parts are created. This activity does not modify the physical plant or operating procedures. This analysis has determined that there are no unreviewed safety questions involved, and no Technical Specification changes required for this UFSAR change. UFSAR section 4.2.3.1.1 was revised accordingly.

UFSAR CHANGE (Pkg 97-157)

DESCRIPTION

SYSTEM: Nuclear Fuel

The evaluation for the Unit 1 Cycle 10 Reload Report (that was submitted to the NRC in 1985), included a design change for the burnable poison rod (BPR)s. UFSAR Figure 4-1 was updated to show the new BPR .

SAFETY EVALUATION SUMMARY

The normal reload analyses verifies that the core design, including fuel and control components, will operate within the acceptance criteria limits. The reload design process is performed in accordance with NRC approved methodologies. The predicted physics parameters are bounded by the UFSAR analyses. This UFSAR change does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity does not modify operating procedures. This analysis has determined that there are no unreviewed safety questions involved, and no Technical Specification changes required for this UFSAR change which simply replaces an outdated BPR figure with the current design. UFSAR Figure 4-1 was revised accordingly.

UFSAR CHANGE (Pkg 97-160)

DESCRIPTION

SYSTEM: Spent Fuel Pool/Nuclear Fuel

This activity revised UFSAR section 9.1.1 to add a brief discussion of the restricted loading patterns that have been established to allow storage of fuel with an enrichment up to a nominal 5.00 weight percent U-235 in the fuel storage racks. The restricted loading patterns are not currently described in the UFSAR. The analysis which justified increasing the allowable enrichment is described in UFSAR section 9.1.2.3.2, Criticality Analysis.

UFSAR section 9.1.1 currently states: "New fuel will normally be stored in the spent fuel pool serving the respective unit."

This activity therefore revised the first paragraph of section 9.1.1 for completeness and consistency with UFSAR section 9.1.2.3.2 and existing licensing documentation as follows:

"New fuel will normally be stored in the spent fuel pool serving the respective unit. New or irradiated fuel assemblies with initial enrichments up to 5.00 weight percent U-235 which do not meet the requirements for unrestricted storage must be placed in a restricted loading pattern. Reactivity analyses for these assemblies, stored in every other row of the spent fuel pool, were performed using the methods discussed in section 9.1.2.3.2, Criticality Analysis. Acceptable fuel assemblies which qualify for storage in the alternating rows between adequately depleted assemblies are referred to as filler assemblies."

In addition, this activity added information to section 9.1.5 to document the basis for the above revision.

SAFETY EVALUATION SUMMARY

The Oconee fuel storage racks provide for storage of new and spent fuel assemblies in a flooded pool, while maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excess mechanical or thermal loadings. The fuel storage racks are composed of individual storage cells made of stainless steel interconnected by grid assemblies to form integral module structures. The racks utilize a neutron absorber, Boraflex, which is attached to each cell.

UFSAR section 9.1.1 states that new fuel is normally stored in the spent fuel pool serving the respective unit. This section does not include a discussion of the restricted loading patterns that were established to allow storage of more highly enriched fuel as documented by the Safety Evaluation Report (SER) for License Amendments 209/209/206. This activity adds a description of the restricted loading patterns to section 9.1.1 for completeness and consistency with existing UFSAR section 9.1.2.3.2 and licensing documentation. These restricted loading patterns have been previously analyzed and approved as described in UFSAR section 9.1.2.3.2, the licensing submittal , and the associated SER. Since these changes have been previously evaluated and approved, this activity does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-161)

DESCRIPTION

SYSTEM: Spent Fuel Pool/Nuclear Fuel

This activity revised UFSAR section 9.1.2.3.2.3 for consistency with the licensing submittal.

UFSAR section 9.1.2.3.2.3, paragraph 14, describes the computer codes used to analyze the boundary conditions between rows of restricted and unrestricted fuel assemblies stored in the fuel storage racks. Minor descriptive differences exist between the statement, the licensing submittal , and the associated calculation.

This activity therefore revised the fourteenth paragraph of section 9.1.2.3.2.3 for completeness and consistency with the licensing submittal as follows:

"This methodology utilizes three dimensional Monte Carlo theory. Specifically, this analysis method used the CSAS25 sequence contained in Criticality Analysis Sequence No. 4 (CSAS4). CSAS4 is a control module contained in the SCALE-4.2 system of codes. The CSAS25 sequence utilizes two cross section processing codes (NITAWL and BONAMI) and a 3-D Monte Carlo code (KENO Va) for calculating the effective multiplication factor for the system. The 27 Group NDF4 cross section library was used exclusively for this analysis."

SAFETY EVALUATION SUMMARY

The Oconee fuel storage racks provide for storage of new and spent fuel assemblies in a flooded pool, while maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excess mechanical or thermal loadings. New or irradiated fuel assemblies with initial enrichments up to 5.00 weight percent U-235 which do not meet the requirements for unrestricted storage must be placed in a restricted loading pattern. Since fuel will be stored in the spent fuel pools according to two different loading configurations to accommodate both restricted and unrestricted storage, the boundary conditions between these configurations were analyzed to determine the effects of neutronic coupling.

UFSAR section 9.1.2.3.2.3, briefly describes the computer codes used to analyze the boundary conditions between rows of restricted and unrestricted fuel assemblies stored in the fuel storage racks. Minor descriptive differences exist between the statement, the licensing submittal, and the associated calculation. Therefore, this activity revises section 9.1.2.3.2.3 to specify three-dimensional Monte Carlo theory and the SCALE-4.2 system of computer codes for completeness and consistency. Existing UFSAR section 9.1.2.3.2.3 discussion only identifies the computer codes as SCALE-4. This revision does not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the activity described does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-162)

DESCRIPTION

SYSTEM: Nuclear Fuel

This activity updated UFSAR sections 9.1.4.2.2 and 9.1.4.2.3 to reflect currently authorized core design and refueling practices to be consistent with existing section 4.5.2, Table 4-20, and Figure 4-1.

UFSAR sections 9.1.4.2.2 and 9.1.4.2.3 discuss shuffling of fuel assembly inserts during reactor refueling. The inserts currently included in the discussion are Control Rod Assemblies (CRA) and Orifice Rod Assemblies (ORA). This is not consistent with the currently authorized core design and refueling practices. Orifice rod assemblies are no longer used at ONS. Also, Burnable Poison Rod Assemblies (BPRA) are now included in the core design. Design and operation of the fuel handling system that is used to shuffle the inserts is not changed.

This activity therefore revised the ninth paragraph of Section 9.1.4.2.2 and the thirteenth paragraph of section 9.1.4.2.3 to remove ORAs from the statements and to add BPRAs to the statements:

> "The control rod handling mechanism is used to transfer the control rod or ~~orifice~~ burnable poison rod assembly to a new fuel assembly waiting in the second fuel transfer carriage basket. This new fuel assembly with control rod or ~~orifice~~ burnable poison rod assembly is carried to the reactor by the main bridge and located in the core while the spent fuel assembly is being transferred to the spent fuel pool."

> "The fuel handling bridges are limited to handling of fuel assemblies and control rod and ~~orifice~~ burnable poison rod assemblies only."

SAFETY EVALUATION SUMMARY

Fuel assemblies are built with a standardized design to allow interchangeability among core locations. Some core locations have CRAs, while most do not. The standard assembly design includes guide tubes which restrain lateral movement of the control rods. Orifice rod assemblies were used in early core designs to control the amount of reactor coolant flow that would bypass the fuel rods by flowing through empty control rod guide tubes in the fuel assemblies that did not have CRAs. The core design has since been re-analyzed to allow operation without use of ORAs. The core thermal-hydraulic analyses have been revised and approved by the NRC to permit operation without ORAs. Burnable poison rod assemblies are used to suppress excess core reactivity early in the operating cycle for cores using higher enrichments to extend the length of the cycle. As the cycle progresses, the poison burns out. This produces a positive reactivity effect that tends to counter the reactivity loss associated with fuel burnup. The core nuclear designs have been revised and approved by the NRC to permit operation with BPRAs.

This activity revises sections 9.1.4.2.2 and 9.1.4.2.3 to delete ORAs from, and add BPRAs to the discussion of refueling practices. This activity updates the UFSAR to match the design, function and operation of structures, systems and components as previously evaluated in the SAR. Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-163)

DESCRIPTION

SYSTEM: Spent Fuel Pool/Cranes

This activity revised UFSAR section 9.1.4.2.2, to correct the span listed for the spent fuel cask handling crane. The span is listed in this section as 11 foot 6 inch. Drawings O-0018-C, O-2308-D and OM 0109-0122 indicate that the span is 13 foot, 6 inch. Review of drawings indicates that the span listed in the UFSAR is actually the travel range of the hook, not the span of the rails.

UFSAR section 9.1.4.2.2 stated: "The spent fuel cask handling facility consists of a 100-ton capacity overhead bridge crane with an 11 foot 6 inch span."

This activity therefore revised the fifteenth paragraph of section 9.1.4.2.2 to conform to the as-built configuration as follows:

"The spent fuel cask handling facility consists of a 100-ton capacity overhead bridge crane with a 13 foot 6 inch span."

SAFETY EVALUATION SUMMARY

The spent fuel cask handling overhead bridge crane is used to move an empty shipping cask from a transport trailer to the spent fuel pool. Once the cask is loaded with spent fuel assemblies, the crane moves the cask from the spent fuel pool to a decontamination area. After decontamination the crane moves the cask to a transport trailer for shipment.

UFSAR section 9.1.4.2.2 incorrectly lists the span of the spent fuel cask handling overhead bridge crane as 11' 6". The actual as-built span is 13' 6". The span of the crane is not used as an input to the cask drop accidents analyzed in the UFSAR. This activity revises the span of the crane listed in section 9.1.4.2.2 to match the as-built span shown on applicable drawings. Revising the span listed for the crane does not affect the design, integrity, operation or function of structures, systems and components as previously evaluated in the SAR. Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-164)

DESCRIPTION

SYSTEM: Condenser Circulating Water (CCW) System

UFSAR Section 9.2.2.2.1 provides a system description of the Condenser Circulating Water (CCW) System and makes the following statement:

> "The Little River arm of Lake Keowee is the source of water for the CCW systems. Figure 9-9 shows the arrangement of the systems with respect to the two branches of Lake Keowee."

UFSAR Figure 9-9 is a summary flow diagram of the CCW system. This figure does not show the arrangement of the system with respect to Lake Keowee as described in the above statement. However, UFSAR Figure 2-4 does show the arrangement. Therefore, this activity revised the statement cross reference from Figure 9-9 to Figure 2-4.

SAFETY EVALUATION SUMMARY

UFSAR Section 9.2.2.2.1 states that "The Little River arm of Lake Keowee is the source of water for the CCW systems. Figure 9-9 shows the arrangement of the systems with respect to the two branches of Lake Keowee." UFSAR Figure 9-9 is a summary flow diagram for the CCW System; however, this figure does not provide any detail describing the arrangement of the system with respect to Lake Keowee. Existing UFSAR Figure 2-4 does provide the arrangement detail and would therefore be a more appropriate cross reference in the affected sentence. This activity therefore changed the cross reference from "Figure 9-9" to "Figure 2-4" and was more appropriate within the statement context.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-166)

DESCRIPTION

SYSTEM: Chemical Addition System

Existing discussions for individual Chemical Addition (CA) components in UFSAR section 9.3.1.2 address the unit segregation (i.e. Unit 1 & 2, Unit 3). However, the Boric Acid Mix Tank (BAMT) discussion in the fourth paragraph of UFSAR Section 9.3.1.2 does not address the distribution of tanks per unit.

Existing statements in UFSAR 9.3.1.2.6 indicate that the only consideration following a Loss of Coolant Accident (LOCA) is the operation of isolation valves. These statements are inconsistent with existing statements in UFSAR sections 9.3.1.1, 9.3.1.2 (fifth paragraph), and 9.3.1.2.1 (first paragraph), which indicate that CA caustic addition is utilized to minimize zinc-boric reactions in the Low Pressure Injection (LPI) system following a LOCA. Post-LOCA Caustic Addition is also addressed by Technical Specification 6.4.1.i and by docketed response to IEB 77-04, Post-LOCA pH Control.

Therefore, this activity clarified the first sentence of the fourth paragraph of UFSAR Section 9.3.1.2 and the first sentence of UFSAR Section 9.3.1.2.6 as follows for consistency with existing UFSAR discussions:

- Two Boric Acid Mix Tanks, one shared between units 1 and 2, and one for unit 3, are provided as a source of concentrated boric acid solution.

- Since the system serves no engineered safeguards function, the only consideration immediately following a loss-of-coolant accident is the operation of the isolation valves.

SAFETY EVALUATION SUMMARY

Existing discussions in UFSAR Section 3.1.4, Figures 9-15 and 9-16 address the sharing of CA system components between units 1 and 2. This sharing is also clearly identified in existing UFSAR Section 9.1.3.2 system and component discussions, with the exception of BAMT. Existing BAMT discussions address the total number of tanks but do not identify the actual sharing between units. Sodium Hydroxide (Caustic) is added to the LPI system following a LOCA to minimize zinc-boric reactions in recirculated coolant. This addition is a Technical Specification administrative control requirement and must be initiated within 30 minutes of switchover to the recirculation mode of core cooling to adjust the pH to a range of 7.0 to 8.0 within 24 hours (see Specification 6.4.1.i). In response to IEB 77-04, Duke committed to sufficient Caustic at the station to maintain the pH of the containment sump post LOCA solution within specification under all operating conditions.

The minor revisions associated with this activity provided clarification to UFSAR Sections 9.3.1.2 and 9.3.1.2.6 to be consistent with existing discussions in UFSAR Sections 3.1.4, 9.3.1.1, 9.3.1.2, and 9.3.1.2.1 regarding component sharing between units, and post-LOCA CA caustic addition considerations. Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGES (Pkg 97-168)

DESCRIPTION

SYSTEM: CRVS and WC

The changes which were made to the UFSAR per PIP 98-1165 can be divided into two categories: 1) Clarify definitions of control area and control room zone. 2) Clarify single active failure requirements of CRVS and WC.

SAFETY EVALUATION SUMMARY

The definition of control area includes the control, cable and electrical equipment rooms. This is the definition as stated in the original FSAR. During the 1982 revision this definition was left out. This change added the definition back in. This change was largely editorial in nature. The definition of control room zone was incorrectly changed during rev 4 of the FSAR. The control room zone as defined in section 6.4.2.1 is made up of the control room, offices, computer rooms, operator's break area, and operator's toilet. This is the same as the control room envelope as defined in that section. Correcting the definition of control room zone back to its previous one has no impact on safety.

Requirements for single active failure were added to section 9.4.1.1 to reflect statements in section 3.11.4. Redundant air conditioning and ventilation equipment was required to assure that no single failure of an active component within the control area prevents proper environmental control. This issue has been addressed in PIP 98-1165. The control rooms along with the Unit 3 cable and electrical equipment rooms have two-100% capacity AHUs. The units 1&2 cable and electrical equipment rooms have redundant AHUs which allow the system to maintain acceptable temperatures in the rooms assuming a failure of one AHU. This arrangement was shown to be acceptable in OSC-7141.

These UFSAR changes do not cause, or adversely affect the mitigation of, any prevously analyzed SAR accidents. No new radiological release pathways or failure modes are created, and no SSCs are degraded. There are no physical changes to the plant or operating procedures. These changes create no USQs or safety concerns and no technical specification changes are required. UFSAR sections 3.11.4, 6.4.2.3, and 9.4.1 were revised accordingly.

UFSAR CHANGE (Pkg 97-169)

DESCRIPTION

SYSTEM: High Pressure Injection (HPI) System

UFSAR Section 9.3.2.2.4 discussion of High Pressure Injection (HPI) System isolation is not consistent with information in existing UFSAR Figures 6-9 and 9-17 concerning Reactor Coolant Pump (RCP) seal injection line isolation valve configuration. The above figures are consistent with existing HPI flow diagrams and minor modifications in that stop check valves are provided outside containment rather than a remotely operated valve.

Also, the existing system isolation configuration in UFSAR section 9.3.2.2.4 is inconsistent in the identified number of lines, for a given line type, in relation to the number of penetrations and actual lines on existing UFSAR Figures 6-9, 9-17, and 9-18 and existing HPI flow diagrams.

This activity therefore revised UFSAR Section 9.3.2.2.4, in the third, fourth, and eighth sentences, as follows to reflect as installed HPI system isolation configuration that is consistent with information in existing UFSAR Figures 6-9 and 9-17, and HPI flow diagrams.

SAFETY EVALUATION SUMMARY

A detailed discussion of the Reactor Building Containment Isolation System, including the HPI system isolation valves associated with the above operations, is presented in existing UFSAR Section 6.2.3 and referencing Figure 6-9 and Table 6-7 for individual line/penetration information. A brief discussion of the configuration of the HPI system isolation valves is also presented in UFSAR Section 9.3.2.2.4 and detailed in Figure 6-17 and 6-18.

The existing UFSAR Section 9.3.2.2.4 discussion is not consistent with the information in existing Figures 6-9, 9-18 and Table 6-7 in that four (4) RCP seal injection lines and individual penetrations exist for each unit rather than one (1), and the isolation valves outside the reactor building for these lines are check valves rather than remotely operated valves. The configuration discussed in the existing figures and table are consistent with the applicable HPI flow diagrams for each unit and with various minor modifications associated with valve replacement for the outside containment check valves. Additionally, existing UFSAR Figure 9-17 and the applicable HPI system flow diagrams specify a single Auxiliary Pressurizer Spray line and a single Nozzle Warming line per unit rather than the multiple lines as identified in existing UFSAR Section 9.3.2.2.4.

Therefore, this activity revised the discussion of HPI system isolation valve configuration in UFSAR Section 9.3.2.2.4, for consistency with existing UFSAR Figures 6-9, 9-17, 9-18 and Table 6-7, applicable HPI system Flow Diagrams, and minor modifications for RCP seal injection outside containment isolation valve replacements. This clarification does not impact the design, function, or performance of systems, structures or components as evaluated in the SAR and no unreviewed safety question exists.

UFSAR CHANGE (Pkg 97-170)

DESCRIPTION

SYSTEM: Low Pressure Injection (LPI) System

Existing UFSAR discussions of the Low Pressure Injection (LPI) decay heat removal pumps, specifically in section 9.3.3.2, do not presently address the minimum flow recirculation orifices provided for thermal protection of each pump. These orifices were resized, based on updated manufacturer recommendations, to address concerns expressed in NRC Bulletin (IEB) 88-04.

This activity, therefore, revised the second paragraph of UFSAR Section 9.3.3.2 as follows to include a discussion of the minimum flow orifice configuration for completeness in relation to the IEB 88-04 licensing basis:

Three decay heat removal pumps are arranged in parallel with electric motor operated valves in the suction line to each pump. Each pump has a separate minimum flow recirculation line with an orifice between pump discharge and pump suction. The bore of each orifice was increased to address considerations detailed in IEB 88-04, Safety Related Pump Loss. The two outboard pumps are normally available for emergency operation, and the center pump is valved off on both the suction and discharge sides of the pump. During decay heat removal, any two of the three pumps are lined up to the decay heat removal coolers.

SAFETY EVALUATION SUMMARY

In response to NRC Bulletin (IEB) 88-04, Duke evaluated the effect of increasing the minimum flow orifice bore diameter to meet updated manufacturer's recommendations. This evaluation, including vendor recommendations for pump continuous, short duration and start/stop operation, is documented in OSC-3077. The evaluation, along with proposed procedure changes to address minimum flow concerns during LPI operations such as during draining of the FTC, were accepted by the NRC as documented in Inspection Report IR-90-23. Existing UFSAR discussions, specifically in section 9.3.3.2, do not presently address the LPI minimum flow orifices. A discussion should be included in UFSAR section 9.3.3.2 for completeness in relation to the licensing basis associated with IEB 88-04.

This activity updates UFSAR Section 9.3.3.2 to document the minimum flow orifices provided for the LPI pumps. The installation and size of these orifices has been previously evaluated in relation to IEB 88-04. Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-171)

DESCRIPTION

SYSTEM: Coolant Storage and Radwaste Systems

This activity provides a rewording of UFSAR Section 9.3.4.2 to more accurately state that reactor coolant is processed by the Radwaste Facility as opposed to the Coolant Treatment System which is no longer in use.

UFSAR Section 9.3.4.1 correctly states that coolant processing is performed by the Radwaste Facility. Likewise, UFSAR Section 9.3.5 correctly states that the boron recycling portion of the Coolant Treatment System never functioned properly and that only the coolant storage portion of the system is still in use at Oconee. However, UFSAR Section 9.3.4.2 still indicates that coolant processing is performed by the Coolant Treatment System.

This activity therefore revised the last sentence of the first paragraph of Section 9.3.4.2 to read as follows:

"Liquid from the coolant bleed holdup tanks can be pumped to the Radwaste Facility for processing."

SAFETY EVALUATION SUMMARY

Coolant processing is performed by the Radwaste Facility. The Coolant Treatment System was originally designed and installed to both store RC bleed and to treat RC bleed for recycling. Since the boron recycling portion of the original Coolant Treatment System never functioned properly, the coolant storage portion is the only part of the system still in use at Oconee. UFSAR Section 9.3.4.2 states that liquid pumped from the coolant bleed holdup tanks is pumped to the Coolant Treatment System for processing. Only the coolant storage portion of the Coolant Treatment System is still in use and all coolant processing is now performed by the Radwaste Facility. This is consistent with RC bleed transfer operations allowed by procedure and statements made in other UFSAR sections.

Therefore, the last sentence of the first paragraph of UFSAR Section 9.3.4.2 was rewritten to state that coolant processing is now performed by the Radwaste Facility and not by the Coolant Treatment System. Changing the UFSAR in this manner does not invalidate any conclusions reached in the SAR and makes UFSAR Section 9.3.4.2 more consistent with the discussions in other UFSAR sections. Performing the change associated with this activity does not change any design, operation, or analyses currently evaluated in the SAR nor does it change any information that could be used to prevent safety related components from performing their safety functions.

UFSAR CHANGE (Pkg 97-172)

DESCRIPTION

SYSTEM: Containment Hydrogen Monitoring System

This activity provides clarification to UFSAR Section 9.3.7.3 to make the text more accurately reflect the actual system accuracy for the Containment Hydrogen Monitoring System (CHMS) indicator loop. Currently the UFSAR states that the system accuracy for the CHMS indicator loop is 5.0% of full scale which is inconsistent with the accuracy values provided by Duke to the NRC and approved by the NRC in an SER.

This activity therefore modifies the fourth sentence of the first paragraph of UFSAR Section 9.3.7.3 to read as follows:

"The CHMS indicator loop has a system accuracy of 3.0% of full scale."

SAFETY EVALUATION SUMMARY

UFSAR Section 9.3.7.3 should be changed to state that the CHMS indicator loop system accuracy is 3.0%. This value represents the "reference accuracy" of the system using the SRSS method and is the appropriate value to use since it is the value documented in licensing correspondence from Duke and in an NRC SER. Making this change does not impact the design, function, or any analyses associated with the CHMS system or its interfaces. Existing Duke calculations and analyses are not impacted since the "reference accuracy" value is considered when calculating the total loop uncertainty for the CHMS remote indication string equipment. Therefore, performing the change associated with this activity does not impact any existing SAR evaluations and does not change any information that could be used to prevent safety related components from performing their safety functions.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-173)

DESCRIPTION

SYSTEM: Control Room Ventilation System

This activity revised UFSAR Section 9.4.1.1 to include a reference to Selected Licensee Commitment (SLC) 16.8.1 and to provide a cross-reference to Section 6.4 of the UFSAR. This revision is provided for completeness and consistency with similar discussions in existing UFSAR Sections 3.11.4, 6.3.2.2.1, 6.3.2.2.2, 6.4.2.2, 6.5.1.2, 9.3.2.2.6, 9.3.3.1, 9.4.6.1 and 9.4.7.1.

This activity inserted the folowing paragraph into Section 9.4.1.1 for consistency with existing UFSAR sections:

"Control Room Zone temperatures related to Station Blackout are addressed by Selected Licensee Commitment 16.8.1. The pressurization and filtration of the control room envelope is discussed further in Section 6.4, 'Habitability Systems'."

SAFETY EVALUATION SUMMARY

The clarifications associated with this activity update UFSAR Section 9.4.1.1 to the content level provided in existing UFSAR sections Sections 3.11.4, 6.3.2.2.1, 6.3.2.2.2, 6.4.2.2, 6.5.1.2, 9.3.2.2.6, 9.3.3.1, 9.4.6.1 and 9.4.7.1. These cross-references direct the UFSAR reader to other UFSAR sections for related system discussions and to Selected Licensee Commitments related to section discussions as applicable. Addition of the applicable cross reference discussed above to UFSAR section 9.4.1.1 increased section content for consistency with the content of existing UFSAR sections.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-174)

DESCRIPTION

SYSTEM: Fire Protection System

This activity revised the second paragraph of UFSAR Section 9.5.1.1 to add the following fire suppression methods for consistency and completeness with UFSAR Sections 9.5.1.5.3 through 9.5.1.5.5: fixed sprinklers, Halon and CO_2 fire suppression systems.

> "Total reliance is not placed on a single automatic fire suppression method. Fire hose stations, fixed sprinklers, Halon and CO_2 fire suppression systems, and portable extinguishers are provided."

SAFETY EVALUATION SUMMARY

UFSAR Section 9.5.1.1 paragraph one describes the design bases for the fire protection program. The program is based on the analysis potential for fire hazards in the Auxiliary Building, Turbine Building, Reactor Building and associated adjacent areas.

This activity revised the second paragraph of UFSAR Section 9.5.1.1 to add the following fire suppression methods for consistency and completeness with UFSAR Sections 9.5.1.5.3 through 9.5.1.5.5: fixed sprinklers, Halon and CO_2 fire suppression systems. The proposed change provides an update to UFSAR Section 9.5.1.1, for the addition of fire suppression systems which are described in the UFSAR Sections 9.5.1.5.3 through 9.5.1.5.5.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGES (Pkg 97-176)

DESCRIPTION

SYSTEM: None

The follow UFSAR discrepancies in Section 3.9.3.1.3 were addressed per PIP 97-2605:

- Routing of instrument and impulse tubing previously described in 3.9.3.1.3(1) is no longer established by Construction. Currently, the specific routing requirements of safety related and non safety related tubing lines are established by Specification OSS-0060.00-00-0001.
- Routing and erection of Class E, G & H piping is specified per Specification OSS-0027.00-00-0003.
- Seismic design requirements for field routed piping previously described in 3.9.3.1.3(3b) are no longer established by appropriate Design Engineering personnel after erection. Currently, seismic design requirements for all piping are established before erection.
- The engineer surveillance program conducted after erection to review all seismically designed structure interaction with non seismically designed structures previously described in 3.9.3.1.3(3c) has been eliminated. Currently, such reviews are made prior to erection by engineering and reviewed by QA/QC after installation.
- Design Engineering visits to the Oconee site previously described in 3.9.3.1.3(3g) have been eliminated since the Design Engineering organization has been combined with the Oconee site engineering staff.
- Inspection by Construction QC Group, Operating personnel, and Design Engineering of field run piping and impulse lines after installation previously described in 3.9.3.1.3(3h) have been eliminated. A more formalized process replaces the former, and relies on engineering review of modifications prior to installation, engineering specification of routing and erection requirements prior to installation, and post installation inspection by QA/QC.

SAFETY EVALUATION SUMMARY

Updating of the process of routing and erection of instrument and impulse tubing, and Class E, G, & H piping did not increase the probability of a malfunction of equipment important to safety since the process change is an improved, more formalized process than previously described in Section 3.9.3.1.3. This revision does not involve any physical changes to the facility as described in the SAR, nor do they alter the design bases. The activity does not change procedures, methods of operation, or alter a test or experiment, either described or not described in the SAR. As such, this UFSAR change does not adversely affect any SSC necessary to operate the plant in accordance with the SAR. This change is not an accident initiator and does not introduce any new failure modes or mechanisms. No new radiological release pathways are created. This largely editorial in nature change is to correct the old information and add new updated information.

This UFSAR revision involves no USQs, no safety concerns, and no changes to the Technical Specification. The wording within Section 3.9.3.1.3 was updated to reflect the improvements in the process of routing and erection of instrument and impulse tubing, and Class E, G, & H piping; and to reflect the changes in the organization responsibilities relative to the routing and erection of such tubing and piping.

UFSAR CHANGE (Pkg 97-177)

DESCRIPTION

SYSTEM: Fire Protection System

This activity revised a number of locations in the UFSAR where lower tier documents which control the fire proteciton program at Oconee are now incorrectly referred to. For example:

- The second paragraph of UFSAR Section 9.5.1.3 was revised to replace "Site Directive 3.2.7, "Control of Combustible Materials,"" with Nuclear System Directive 313, "Control of Combustible and Flammable Material".
- UFSAR Section 9.5.1.3 was revised to replace "Site Directive 3.2.10" with Nuclear System Directive 314.
- UFSAR Sections 9.5.1.3 paragraph 6 was revised to replace "Site Directive 3.2.7" with Nuclear System Directive 313.
- UFSAR Section 9.5.1.3 paragraph 9 was revised to replace "Site Directive 3.2.9" with Nuclear System Directive 316.
- UFSAR Section 9.5.1.6.1 paragraph 6 was revised to replace "Site Directives" with Nuclear System Directives.

SAFETY EVALUATION SUMMARY

The fire protection program is structured to detect and suppress fires with out the loss of safety functions, and protect the public from undue release of radioactive material.

The facility procedures which implement the controls to minimize the amount of combustibles that a safety-related area may be exposed to include provisions to: limit the use and storage of combustibles in safety-related areas; establish work controls and require additional fire protection where transient fire loads are introduced; assure the removal of waste, debris and scrap materials following work activities; and provide for periodic housekeeping inspections. Facility procedures for the control of combustibles are designed to minimize the quantity of flammable material in safety related areas. Controlling ignition sources during routine work projects minimizes the potential of exposure for existing combustible material. Control of location, storage and use of flammable materials is dictated by administrative procedures. Location control of combustible material extends to all structures, systems and components related to the safety of the plant. Administrative procedures define acceptable storage areas and associated quantities that can be stored there.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-178)

DESCRIPTION

SYSTEM: Fire Protection System

This activity revised the first sentence in paragraph ten of UFSAR Section 9.5.1.3 to replace Site Directive 3.2.8 with Nuclear System Directive 112, "Fire Brigade Organization, Training and Responsibilities". In the second sentence, the proposed activity changed the title of Fire Chief to Fire Brigade Leader and delete Assistant Fire Chief. Therefore, paragraph ten was revised to read as follows:

> The Oconee Fire Brigade organization is addressed by Nuclear System Directive 112 which describes the functions and duties of each position and identifies individuals by title to fill these positions. The organization provides for a Fire Brigade Leader and Shift Coverage.

SAFETY EVALUATION SUMMARY

UFSAR Section 9.5.1.3 paragraph 10 indicates that Site Directive 3.2.8 describes the Fire Brigade functions, duties, and hierarchy. This directive has been deleted. Current information for the Fire Brigade Organization is contained in Nuclear System Directive 112, "Fire Brigade Organization, Training and Responsibilities".

This proposed change provides an update to UFSAR Section 9.5.1.3 paragraph 10, to provide correct directive cross references. It also revises the title of Fire Chief to Fire Brigade Leader and deletes Assistant Fire Chief for consistency with existing organizational titles. This activity does not effect the design or operation of any system, structure, or component relied on by the Fire Protection Program.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the UFSAR addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-179)

DESCRIPTION

SYSTEM: Fire Protection System

This activity revised the first sentence in paragraph one of UFSAR Section 9.5.1.5.1 to reflect the correct year designation for the National Fire Code from 1975 to 1976. Therefore, sentence one was revised to read as follows:

> "Deviations from NFPA 72D are identified and justified by paragraph number per National Fire Code, 1976: "

SAFETY EVALUATION SUMMARY

The fire protection program is structured to detect and suppress fires with out the loss of safety functions, and protect the public from undue release of radioactive material. The plant has a protective signaling system which transmits alarms from fire detectors and water spray system actuation to the control room. Water flow on wet pipe sprinkler systems does not alarm in the control room. In general, the system complies with those provisions of NFPA 72D which are considered essential for the facility, including requirements for emergency power supply and circuit supervision. There is no distinct audible fire alarm signal provided in the control room . Selected Licensee Commitments provides the detector locations and procedural operability requirements and actions. Existing UFSAR Section 9.5.1.5.1 discussion identifies the applicable year of the National Fire Code as of 1975. As documented in the Response, the correct edition is 1976. This proposed change would update UFSAR section 9.5.1.5.1 to reflect the correct year designation for consistency with applicable licensing correspondence.

Since these changes were previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the UFSAR addition described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-180)

DESCRIPTION

SYSTEM: Instrument Air and Breathing Air Systems

UFSAR Section 9.5.2.1 describes the design basis for the Instrument and Breathing Air Systems. This section also states that "the Instrument and Breathing Air Systems are designed to provide clean, dry, oil free instrument air to all air operated instrumentation and valves, and breathing air at ANSI Z86.1 Grade D standards to minimize personnel exposure in areas of airborne contamination."

This activity revised the statement as follows to address the additional air quality standard requirements implemented on the Instrument Air System under Oconee's response to GL 88-14:

"The Instrument and Breathing Air Systems are designed to provide clean, dry, oil free instrument air to all air operated instrumentation and valves. Instrument air is supplied to ISA -S7.3-1975 standards, and breathing air is supplied at ANSI Z86.1 Grade D standards to minimize personnel exposure in areas of airborne contamination."

SAFETY EVALUATION SUMMARY

This activity changed UFSAR Section 9.5.2.1 to include the air quality standards for the Instrument Air system. In response to Generic Letter 88-14, Oconee committed to an instrument air quality testing program based on ISA-S7.3-1975. OSC-4462 states that the most restrictive air quality testing standards were those that reference ANSI/ISA-S7.3-1975(R1981). The 3 elements of air quality standards are dew point, particulate and oil content.

This activity involves only a UFSAR change to document the standards to which instrument air is supplied to. These acceptance criteria were provided in response to GL 88-14 to ensure a highly reliable source of instrument air to plant equipment. Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-181)

DESCRIPTION

SYSTEM: Standby Shutdown Facility

UFSAR Section 9.6 describes the design, function, and operation of the Standby Shutdown Facility (SSF). The existing UFSAR information in Section 9.6 is a compilation of design and licensing basis information regarding the requirements associated with the SSF. Two situations have been identified where UFSAR Section 9.6 does not identify the functions provided by the SSF for station blackout nor accurately describe the events which would result in SSF actuation. A summary level discussion of the SSF is also provided in UFSAR Section 1.2.2.10. This discussion also inadequately addresses the SSF's role in station blackout. Therefore, the changes associated with this activity provided additional detail to UFSAR Sections 1.2.2.10, 9.6.1, and 9.6.5 to account for the SSF's role during station blackout.

SAFETY EVALUATION SUMMARY

During the UFSAR Chapter 9 Verification Project, UFSAR Section 9.6 was reviewed for content adequacy based on a review of existing licensing basis information. From that review several instances were found where the UFSAR text contained an inadequate level of detail to address the licensing basis information for the SSF regarding it's role in station blackout. The SSF functions credited for station blackout are discussed in two NRC Safety Evaluation Reports (SERs) dated March 10 and December 3, 1992 (Ref. 2). These SERs indicate that the SSF Auxiliary Service Water (ASW) System is the credited source of decay heat removal and the SSF Diesel is the credited alternate AC (AAC) power source for the required station blackout coping duration.

The overall Oconee capability to cope with a station blackout is discussed in UFSAR Section 8.3.2.2.4. However, the existing discussion in UFSAR Section 9.6 inadequately addresses the licensing basis requirements of the SSF for coping with a station blackout. Likewise, the summary level discussion provided in UFSAR Section 1.2.2.10 for the SSF also requires additional detail. Based on this background it is reasonable to add additional detail to the UFSAR in an effort to better address existing licensing basis information associated with the SSF's role in coping with a station blackout.

Updating the UFSAR in this manner did not invalidate any other UFSAR sections and did not change any of the evaluations or conclusion discussed in the SAR. In fact, adding the additional text associated with this change ensures more consistency with other UFSAR sections addressing station blackout and better reflects current licensing basis requirements associated with the SSF. These revisions to the UFSAR did not change any component design information discussed in the SAR nor did it change any information such that safety related components could be prevented from performing their safety functions.

UFSAR CHANGE (Pkg 97-182)

DESCRIPTION

SYSTEM: Standby Shutdown Facility

UFSAR Section 9.6 describes the design, function, and operation of the standby shutdown facility (SSF). The existing UFSAR information in Section 9.6 is a compilation of design and licensing basis information regarding the requirements associated with the SSF. UFSAR Section 9.6.1 discusses the overall functionality and design of the SSF. In the second paragraph of this section, the UFSAR contains a misleading statement indicating that the SSF isolates all other sources of Reactor Coolant System (RCS) addition with the exception of the SSF Reactor Coolant Makeup (RCM) System. In actuality, the SSF provides no such isolation, but, by design, would only be utilized after all normal sources of RCS addition had become unavailable.

This activity therefore clarifirf the design function of the SSF RCM system as currently described in the second paragraph of UFSAR Section 9.6.1 by revising the fourth bullet contained in the paragraph to read as follows:

- "4.Maintain the reactor subcritical, after all normal sources of RCS makeup have become unavailable, by providing makeup via the Reactor Coolant Makeup Pump System which always supplies makeup of a sufficient boron concentration."

SAFETY EVALUATION SUMMARY

The SSF RCM system is designed to supply makeup to the RCS in the event that normal makeup systems are unavailable. The SSF RCM System supplies borated makeup to the RCS to provide Reactor Coolant Pump (RCP) seal cooling and RCS inventory. The RCM system is designed to ensure that sufficient borated water is available from the spent fuel pools to allow the SSF to maintain hot shutdown conditions to all three units for approximately 72 hours. The RCM pump is capable of delivering borated water from the Spent Fuel Pool to the RCP seal injection lines. A portion of this seal injection flow is used to makeup for RC pump seal leakage while the remainder flows into the RCS to makeup for other RCS leakage. The SSF is used during extreme emergency conditions to achieve and maintain hot shutdown conditions following postulated fire, sabotage, or flooding events. During these events, safe shutdown of the reactor is initially performed by the insertion of control rods from the control room. Reactor coolant inventory and reactor shutdown margin are maintained from the SSF control panel by the SSF RCM pump taking suction from the spent fuel pool.

Based on this discussion, it is appropriate to change UFSAR Section 9.6.1 to describe the actual design function of the SSF RCM System as that of providing RCS makeup after all other sources of makeup have become unavailable rather than that of isolating all sources of RCS addition. Clarifying the UFSAR in this manner did not invalidate any other UFSAR sections and did not change any of the evaluations or conclusion discussed in the SAR regarding the emergency functions of the SSF. Performing the change associated with this activity did not affect the design, operation, or analysis associated with any SSF system, structure or components (SSCs) nor did it impact any information in such a way that could prevent safety related components from performing their safety functions.

UFSAR CHANGE (Pkg 97-183)

DESCRIPTION

SYSTEM: Standby Shutdown Facility

The UFSAR Section 9.6 describes the design, function, and operation of the safe shutdown facility (SSF). The existing UFSAR information in Section 9.6 is a compilation of design and licensing basis information regarding the requirements associated with the SSF. UFSAR Section 9.6.3.1 discusses the design of the SSF structure and incorrectly states that the maximum expected site water level resulting from a turbine building flood is 795.5 feet. Licensing correspondence and Oconee calculations indicate this level should be 796.5 feet.

This activity provided the following changes to the UFSAR:

- The second sentence of the sixth paragraph of UFSAR Section 9.6.3.1 will be revised to more accurately reflect that the maximum expected water level within the site boundary occurring as a result of a turbine building flood is 796.5 feet.

SAFETY EVALUATION SUMMARY

The SSF has been designed to withstand flooding events resulting from 1) turbine building flood caused by a break in the non-seismic condenser circulating water (CCW) piping system and 2) infiltration of normal groundwater. In correspondence regarding the SSF, the NRC acknowledged in their SER for the SSF that the maximum expected water level within the site boundary is 796.5 feet. The NRC further concluded that since this maximum expected water level is below the elevation of the grade level entrance to the SSF, the structure is not flooded by such an incident This conclusion is consistent with the flooding protection measures already in existence for the Turbine Building and Auxiliary Buildings.

It was appropriate to change UFSAR Section 9.6.3.1 to reflect the actual flood level expected within the site boundary as a result of a Turbine Building flood to 796.5 feet which is the value that has been calculated and included in the SSF licensing correspondence. Changing the UFSAR in this manner did not invalidate any other UFSAR sections and did not change any of the evaluations or conclusions discussed in the SAR regarding the emergency functions of the SSF. Performing the change associated with this activity did not change any design, operation, or analysis associated with any SSF SSCs nor did it change any information that could be used to prevent safety related components from performing their safety functions.

UFSAR CHANGE (Pkg 97-184)

DESCRIPTION

SYSTEM: Standby Shutdown Facility

UFSAR Section 9.6.3.3 does not contain a detailed enough description of the Standby Shutdown Facility (SSF) portable pumping system. Currently, there is no discussion of how the SSF submersible pump is used and how it is powered. As a result, the last sentence should be reworded with additional detail to make it clear that the SSF submersible pump is portable, is installed in the intake canal, and is powered from the SSF.

This activity will therefore revise the second paragraph of UFSAR Section 9.6.3.3 by adding additional detail to the existing UFSAR discussion as follows:

> "A portable submersible pump that can be installed in the intake canal and powered from the SSF is available to replenish the water supply in the embedded CCW pipe if siphon flow through the CCW pipe is lost."

SAFETY EVALUATION SUMMARY

In the past it has been shown that under certain SSF scenarios, the CCW buried intake pipe could be depleted in less than 4 hours. Corrective action based on these scenarios included the use of a portable submersible pump and a flow path to replenish the SSF water supply by taking water from the intake canal and discharging it into the Unit 2 CCW pipe. As a result of this situation, Oconee proposed revisions to the SSF technical specifications to address operability requirements for the portable pumping system. The NRC subsequently approved these revisions in their safety evaluation for License Amendments 195/195/192.

Based on this background it is reasonable to add additional detail to the UFSAR in an effort to better address the use and functionality of the SSF portable pumping system.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the UFSAR change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-185)

DESCRIPTION

SYSTEM: Standby Shutdown Facility

UFSAR Section 9.6 describes the design, function, and operation of the standby shutdown facility (SSF). The existing UFSAR information in Section 9.6 is a compilation of design and licensing basis information regarding the requirements associated with the SSF. Several subsections within UFSAR Section 9.6 have been identified which contain slight wording errors resulting in inaccurate statements existing in the UFSAR. Therefore, the changes associated with this activity provide minor wording changes to UFSAR Sections 9.6.3.5.1, 9.6.3.6, and 9.6.3.6.3 to resolve the inaccurate UFSAR statements. These clarifications include:

- Pressurizer temperature is not monitored at SSF control panel.
- Potable water is provided to SSF.
- Diesel Service water pump operates during system testing.

SAFETY EVALUATION SUMMARY

It is reasonable to provide minor rewording to the UFSAR in an effort to increase the accuracy of the existing UFSAR discussions of the SSF. Updating the UFSAR in this manner corrects minor wording discrepancies and does not change any of the evaluations or conclusions discussed in the SAR regarding the SSF or its emergency support functions. Making these revisions ensures consistency with other UFSAR sections and better reflects existing design and operational requirements associated with the SSF.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the proposed change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-187)

DESCRIPTION

SYSTEM: Condenser Circulating Water System

This activity revises UFSAR Table 9-4 to correctly show the Condenser Circulating Water Pump total developed head as 12.4 psig to be consistent with the design information shown on the pump curves (Ref. OSC-5739). This activity also changed terminology from "design press" to "total developed head at rated flow", which is the more appropriate term to be used for describing the pump attribute.

SAFETY EVALUATION SUMMARY

The Condenser Circulating Water (CCW) System provides cooling water to the condensers during normal operation and serves as the ultimate heat sink during plant cooldown. The CCW System also serves as the suction source for the High Pressure Service Water, Low Pressure Service Water, Auxiliary Service Water, and SSF Auxiliary Service Water System and acts as a heat sink for the Recirculated Water System. Portions of the CCW System and the LPSW System are designed so that no single component failure impairs emergency safeguard operation. Each unit consists of 4 CCW pumps which supplies a common intake header via two 11 ft conduits. This activity revises UFSAR Table 9-4 to reflect the design characteristics of as-installed pumps in the CCW System.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-188)

DESCRIPTION

SYSTEM: Recirculated Cooling Water System

This activity revises the Recirculated Cooling Water (RCW) Heat Exchanger design information presented in UFSAR Table 9-4 to be consistent with the vendor design information. For the Unit 1&2 RCW heat exchangers the Condenser Circulated Water (CCW) inlet temp was changed from 75 F to 80 F to be consistent with the design information shown in OM-266-1. For the Unit 3 RCW heat exchanger shell material was changed to SA515-70 to be consistent with the design information shown in OM-266-0007-001.

SAFETY EVALUATION SUMMARY

The RCW System provides inhibited closed cycle cooling water to the following components outside the Reactor Building:

- RC pump seal return coolers
- spent fuel cooling
- sample coolers
- evaporator systems
- various pumps and coolers in the Turbine Building

During a review of UFSAR Table 9-4 data for the RCW heat exchangers, some inconsistencies were noted when compared with the vendor design information.

This activity changes UFSAR Table 9-4 to provide consistency with the "as-installed" RCW heat exchanger component design information:

- CCW inlet temperature for Unit 1 & 2 RCW heat exchangers was changed to 80 F

- shell material for the Unit 3 RCW heat exchangers was changed to SA515-70.

Since these changes were previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-189)

DESCRIPTION

SYSTEM: Low Pressure Injection System

UFSAR Table 9-8 provides Low Pressure Injection (LPI) System Performance Data including 1 and 2 hours to fill and drain the Fuel Transfer Canal (FTC) respectively. These times are unchanged from the original FSAR. However, these times are not consistent with historical B&W design information, OSC-3560 or the established procedures. Additional information is provided in the response to IEB 88-04 that the LPI pumps are operated at minimum flow for FTC draining operations.

This activity therefore updated UFSAR Table 9-8 as follows to reflect a LPI time to drain and fill the FTC consistent with LPI Hydraulic Model information, 88-04 response statements as to pump minimum flow modes of operation and current procedures OP/1,2,3/A/1102/15:

"Fuel Transfer Canal Fill Time, hr"Not Used
"Fuel Transfer Canal Drain Time, hr"8 (nominal)

SAFETY EVALUATION SUMMARY

One of the safety function of the Low Pressure Injection System (LPI) is to fill and/or drain the Fuel Transfer Canal (FTC), although it is not normally used for filling as the Spent Fuel Cooling (SFC) system is normally utilized. No requirements were identified through review of applicable calculations, Technical Specifications or Docketed Correspondence for times to fill and drain the FTC using the LPI. Original FSAR listings of these times are consistent with the current UFSAR values of 1 and 2 hours respectively and appear to represent maximum capabilities. Historical B&W design information identifies approximately 4 hours for filling and 2 hours for draining the FTC. Response to IEB 88-04 provides a commitment to revised procedures OP/1,2,3/A/1102/15 to address minimum flow concerns. These procedures provide instruction for filling the FTC using the SFC system, and for draining the FTC using either the SFC or LPI systems and provide further precautions to ensure that pumping rate does not exceed BWST venting rate during FTC draining operations. The time to drain or fill the canal is not a design input to the Chapter 15 Fuel Handling Accident analysis.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-192)

DESCRIPTION

SYSTEM: Auxiliary Service Water System

UFSAR Section 9.2.3.2 provides a system description of the Auxiliary Service Water (ASW) and states "The pump bypass is equipped with a globe valve." In review of the statement against flow diagram OFD-121D-1.2, no piping bypassing the ASW pump is indicated. However, the flow diagram does show a discharge path to the Condenser Circulating Water (CCW) cross connect discharge header.

This activity revised the statement to be consistent with the as-built configuration depicted in flow diagram OFD-121D-1.2 as follows:

> "The pump is equipped with a minimum flow path to the CCW discharge crossover line, which is isolated by a globe valve."

In addition, the following reference was added as reference 3 to UFSAR Section 9.2.5 in support of the above proposed change:

> **3.** "Letter from H. B. Tucker (Duke) to USNRC Document Control Desk, dated December 5, 1989, "NRC Bulletin No. 88-04 Potential Safety-Related Pump Loss Action 4 Report Status Update"

SAFETY EVALUATION SUMMARY

The ASW system is designed to provide decay heat removal for up to 37 days following a concurrent loss of Feedwater, Emergency Feedwater, and Decay Heat Removal Systems. The system consists of a single ASW pump which takes a suction from the Oconee 2 intake. The system utilizes the water inventory in the intake and discharge piping which are interconnected together. The discharge of the pump is fed into each steam generator via separate connections to the Emergency Feedwater piping. Water in the steam generators is vaporized to the atmosphere to provide for the removal of decay heat. All valves required for ASW operation are either check valves or manual valves. The pump suction is equipped with a manually operated butterfly valve and the discharge with a check valve and manually operated gate valve.

This activity involves only a documentation change to ensure consistency between the UFSAR description and the ASW as-built configuration.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-193)

DESCRIPTION

SYSTEM: Low Pressure Injection System

This activity updated UFSAR Tables 6-9 and 9-9 to reflect current Low Pressure Injection (LPI) pump qualification pressures, based on Ingersoll-Rand requalification, and update UFSAR Table 9-9 to reflect design temperatures for the Unit 1 & 2 “A” LPI Coolers consistent with existing UFSAR Table 6-9 values and nameplate ratings.

UFSAR Tables 6-9 and 9-9 specify design pressures of 470/505 psig for the LPI pumps. The pumps have been requalified per Ingersoll-Rand letter dated June 23, 1992 to reflect an increased design pressure of 560/580 psig. Additionally, UFSAR Table 9-9 specifies the Units 1 and 2 “A” LPI Cooler design temperature as 200°F which is inconsistent with the related temperature data in existing UFSAR Table 6-9 and site documentation.

This activity therefore updated UFSAR Tables 6-9, 9-9 to reflect a Low Pressure Injection Pump design pressure of 560/580 psig in accordance with the Ingersoll-Rand letter, as incorporated in the applicable OM vendor documents, and will update UFSAR Table 9-9 to reflect a design temperature, for the Units 1 and 2 “A” LPI Coolers, of 250°F to be consistent with UFSAR Table 6-9 and OM 201-3132.

SAFETY EVALUATION SUMMARY

The changes that are covered by this evaluation provide updates of the tabular information in Tables 6-9 and 9-9 to reflect accurate pump design pressures, per vendor requalification, and to provide consistent design temperatures for the Units 1 and 2 “A” LPI Coolers.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the UFSAR change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-194)

DESCRIPTION

SYSTEM: Low Pressure Injection System

This activity revised the Component Cooler (CC) design information provided in Table 9-13 for consistency with the vendor's heat exchanger specification sheet. For the Unit 1 CC Cooler the following changes were made:

Capacity, btu/h: 19×10^6
Component Cooling Water Inlet Temp, °F 150
Component Cooling Water outlet Temp, °F 100

For the Unit 2 and 3 CC Coolers the following changes were made:

Component Cooling Water Inlet Temp, °F 150
Component Cooling Water outlet Temp, °F 100

SAFETY EVALUATION SUMMARY

The Component Cooling System is designed to provide cooling water to the following components in the Reactor Building :

- letdown coolers,
- reactor coolant pump cooling jacket and seal coolers,
- quench tank cooler, and
- control rod drive cooling coils

The CC system is not an engineered safeguards system and therefore performs no emergency function. The CC coolers for all three units are identical. In comparing the heat exchanger design data against the CC cooler design data shown in UFSAR Table 9-13, several inconsistencies were noted. This activity revised some design parameters in UFSAR Table 9-13 to be consistent with the heat exchanger design specification sheet:

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the UFSAR change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-195)

DESCRIPTION

SYSTEM: Spent Fuel Pool/Nuclear Fuel

This activity revised UFSAR section 9.1.2.3.3 to accurately describe construction of the fuel storage racks. This activity made section 9.1.2.3.3 consistent with the as-built configuration and with section 9.1.2.1.2. UFSAR section 9.1.2.3.3 describes the construction of the fuel storage racks. This section currently states: "The entire fuel assembly storage rack is constructed of type 304 stainless steel." The installed racks also use Boraflex as a neutron poison.

This activity therefore revised the first paragraph of section 9.1.2.3.3 for completeness and consistency with existing UFSAR sections 9.1.2.1.2, 9.1.2.2.2, and 9.1.2.3.2.2, other licensing and design documentation :

"The entire fuel assembly storage rack is constructed of type 304 stainless steel, with Boraflex panels attached to each cell."

SAFETY EVALUATION SUMMARY

The Oconee fuel storage racks provide for storage of new and spent fuel assemblies in a flooded pool, while maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excess mechanical or thermal loadings. The fuel storage racks are composed of individual storage cells made of stainless steel interconnected by grid assemblies to form integral module structures. Each cell has a lead-in opening which is symmetrical and is blended smooth to facilitate fuel insertion. The cells are open at the top and bottom to provide a flow path for convective cooling of spent fuel assemblies through natural circulation. The fuel assembly storage cells are structurally connected to form modules which limit structural deformations and maintain a nominal center-to-center spacing between adjacent storage cavities during design conditions including the Safe Shutdown Earthquake. The racks utilize a neutron absorber, Boraflex, which is attached to each cell. UFSAR section 9.1.2.3.3, *Material, Construction, and Quality Control*, briefly describes the construction of the fuel storage racks. This section does not include a description of the Boraflex panels used as a neutron poison in the rack design. The Boraflex panels are described in sections 9.1.2.1.2, 9.1.2.2.2, and 9.1.2.3.2.2. This activity adds a description of the Boraflex panels to section 9.1.2.3.3, for completeness and consistency with existing UFSAR sections, other licensing documentation, and design documentation. Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-196A)

DESCRIPTION

SYSTEM: Spent Fuel Pool (SFP)

This change revised UFSAR Section 9.1.4 by substituting a reference to the Core Operating Limits Report (COLR) in place of the existing specific boron concentration requirements for the Fuel Transfer Canal and Spent Fuel Pool. The pool water boron concentration limits are cycle specific, and along with many other nuclear limits and setpoints, are contained in the controlled COLR.

SAFETY EVALUATION SUMMARY

Many nuclear parameters that are cycle dependent (startup physics information, tilt/imbalance limits, boron concentrations, etc.) are contained in the COLR. The COLR information, which is part of the SAR, is updated as necessary throughout the core cycle. Since the detailed cycle information is available in the COLR, there is no value added by repeating the same information in the UFSAR. For the SFP and FTC boron concentrations, a reference to the COLR is now provided.

This UFSAR change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The SFP, RCS, and FTC continue to perform their design functions during normal and accident conditions. There is no physical changes to plant SSCs or operating procedures. No safety concerns or USQs are created by this largely editorial revision to UFSAR section 9.1.4. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-196B)

DESCRIPTION

SYSTEM: Spent Fuel Pool (SFP)

This very minor change revised UFSAR Section 15.11.2.5.1 by substituting a reference to the Core Operating Limits Report (COLR) in place of the existing specific boron concentration (2210 ppmb) requirements for the SFP to support dry cask canister loading. The pool water boron concentration limits are cycle specific, and along with many other nuclear limits and setpoints, are contained in the controlled COLR.

SAFETY EVALUATION SUMMARY

Many nuclear parameters that are cycle dependent (startup physics information, tilt/imbalance limits, boron concentrations, etc.) are contained in the COLR. The COLR information, which is part of the SAR, is updated as necessary through out the core cycle. Since the detailed cycle information is available in the COLR, there is no value added by repeating the same information in the UFSAR. For the SFP boron concentrations to support dry cask storage, a reference to the COLR is now provided. The COLR boron concentration values for the SFP are very conservative with respect to the dry cask storage criticality analysis requirements of 1810 ppmb. This UFSAR change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The SFP, RCS, and FTC will continue to perform their design functions during normal and accident conditions. There are no physical changes to the plant SSCs. No safety concerns or USQs are created by this largely editorial revision to UFSAR Section 15.11.2.5.1. No Technical Specification changes are required.

UFSAR CHANGE (Pkg 97-197)

DESCRIPTION

SYSTEM: Emergency Condenser Cooling Water (ECCW)

This UFSAR change, per PIP 98-978, deleted wording from Sections 9.2.2.2.1, 2 that states the ECCW second siphon is the preferred means of decay heat removal after a station black out (SBO) event. This change could have been made after the revision to Tech Spec 3.4.5 was issued in 1994.

SAFETY EVALUATION SUMMARY

The second siphon is not intended to be used for this purpose. Per commitments documented in the SBO SER, the inverters are off-loaded during this scenario per AP/1,2,3/A/1700/011. Loss of inverters causes the turbine bypass valves to fail closed removing steam flow to the condenser. Offloading the inverters is of primary importance to maintain control room habitability during an SBO. The UFSAR descriptions clearly reflect that the second siphon is not required. Therefore. This UFSAR change does not create any conditions or events which initiate, or adversely impact the mitigation of, any accidents evaluated in the SAR. No new radiological release pathways, failure modes, or malfunctions are created. There are no physical changes to the plant SSCs. This change involves no USQs or safety concerns. No Technical Specification changes are required. UFSAR Sections 9.2.2.2.1, 2 were revised accordingly.

UFSAR CHANGE (Pkg 97-198)

DESCRIPTION

SYSTEM: Reactor Building (RB)

This UFSAR change to Section 3.8.1.1 and Table 3-12 reflects the change of coating materials used in containment. The UFSAR described the use of inorganic zinc primer with Phenoline 305 topcoat. Since Phenoline is no longer available it was replaced by Carboline 890 which is a state of the art coating product.

SAFETY EVALUATION SUMMARY

Carboline 890 is a state of the art coating product that has been design basis accident (DBA) tested. Carboline provides the same level of corrosion protection, and is qualified for radiation exposure, pressure, temperature and water chemistry resulting from a DBA per ANSI N101.2. Coatings have no effect on RB sump clogging. This change does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. No SSC QA, seismic or environmental qualifications are degraded. The RB continues to function as designed during normal and accident conditions. This UFSAR change involves no USQs or safety concerns, and no Tech Spec changes are required. UFSAR Section 3.8.1.1 and Table 3-12 were changed accordingly.

UFSAR CHANGE (Pkg 97-199)

DESCRIPTION

SYSTEM: RCS

UFSAR Section 5.2.3.7 was revised to clarify the Oconee Low Temperature Overpressure Protection (LTOP) System licensing basis design requirements. These clarifications were made as a result of resolution of PIPs associated with licensing basis open items from the Oconee Safety Related Designation Clarification (OSRDC) Project. The revisions to UFSAR Section 5.2.3.7 clarified the licensing basis for LTOP System single failure, seismic, loss of air, and loss of power design requirements. UFSAR Section 5.2.4 was revised to include some key references.

SAFETY EVALUATION SUMMARY

Addition of this licensing position regarding the design requirements of the Oconee LTOP System, along with the appropriate references, to the UFSAR does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. The addition of this information to the UFSAR is simply to document a licensing basis position which already exists. This licensing basis position is based on a licensing interpretation of References (4) through (22) of the 50.59 as documented in the three licensing position white papers. This revision to the UFSAR Sections 5.2.3.7 and 5.2.4 does not result in any plant modifications, procedure changes, or other activities which could result in an unreviewed safety question or safety concers. No Technical Specification changes are required. No Selected Licensee Commitment changes are required.

UFSAR CHANGE (Pkg 97-201)

DESCRIPTION

SYSTEM: Standby Shutdown Facility

The UFSAR Section 9.6 describes the design, function, and operation of the safe shutdown facility (SSF). The existing UFSAR information in Section 9.6 is a compilation of design and licensing basis information regarding the requirements associated with the SSF. UFSAR Figure 9-38 provides a summary level diagram of the SSF Diesel Air Starting System and incorrectly indicates that the diesel generator "B" engine is 16 cylinders. All other design and operational references for the SSF diesel generator indicate that the "B" engine is 12 cylinders.

This activity revised Figure 9-38 to correctly indicate that the SSF diesel "B" engine is 12 cylinders rather than 16.

SAFETY EVALUATION SUMMARY

The SSF Power System is provided with standby power from a dedicated diesel generator. The power unit consists of two EMD diesel engines, a 12-645E4 and a 16-645E4, driving one Electric Products generator coupled with EMD tandem couplings, forming a diesel-generator assembly. It uses a compressed air starting system with four air storage tanks. The SSF diesel is the credited alternate AC power source relied for the required coping duration associated with station blackout conditions. UFSAR Figure 9-38 provides a summary diagram of the SSF Starting Air System. This diagram is consistent with the ONS flow diagrams with the exception that this figure indicates that the SSF diesel generator engine "B" is 16 cylinders. According to the SSF Diesel Generator vendor manual, the "A" engine is 16 cylinders while the "B" engine is only a 12 cylinder engine. Therefore, UFSAR Figure 9-38 was mistakenly labeled and was revised. Based on this discussion, it was appropriate to change UFSAR Figure 9-38 to indicate the correct number of cylinders for the SSF diesel generator engine "B." Changing the UFSAR in this manner did not invalidate any other UFSAR sections and did not change any of the evaluations or conclusion discussed in the SAR regarding the emergency functions of the SSF.

Since these changes have been previously evaluated and approved, they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-202)

DESCRIPTION

SYSTEM: Radiation Monitors

This activity updated UFSAR Figure 9-18 to delete Radiation Monitor RIA-36 and the associated piping and valves for consistency with existing UFSAR text, which no longer contains discussions of this radiation monitor (RIA).

UFSAR Figure 9-18 identifies process Radiation Monitor RIA-36 as monitoring, upstream of ES valve HP-5, the Reactor Coolant (RC) letdown to High Pressure Injection (HPI). This RIA for each unit, and the associated piping, valves, electrical connections, etc., were removed by Modifications ON-52832 and ON-32832 for units 1 & 2, and 3 respectively due to manual reactor coolant samples minimizing the need for these near-obsolete instruments. The applicable statements in UFSAR Section 9.3.2.2 were deleted during the 1993 UFSAR update and no additional UFSAR discussions could be identified.

This activity updated UFSAR Figure 9-18 to delete RIA-36 and associated tubing and valve for consistency with existing UFSAR text, which no longer includes discussion of this radiation monitor, and modifications ON-52832, ON-32832.

SAFETY EVALUATION SUMMARY

Radiation monitor RIA-36 (Reactor Coolant Letdown) was installed at Oconee for the early detection of failed fuel and crud bursts. To accomplish this, it monitored the Reactor Coolant letdown flow for fission product and activation product activity. As per modifications ON-52832, and ON-32832, this radiation monitor, and the associated piping and valves, had become inoperable and/or unreliable with high background counts due to internal contamination.

As per the above modifications, RIA-36 is of no value during or after a LOCA since the letdown system is quickly isolated. If the monitor were to be valved back in post-accident, it would quickly be contaminated. RIA-36 is not required to be operable by the Technical Specifications and was determined to not meet Reg. Guide 1.97 requirements.

These changes have been previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-203)

DESCRIPTION

SYSTEM: High Pressure Service Water System

UFSAR Figure 9-10 provides a summary flow diagram of the High Pressure Service Water (HPSW) System. Several minor errors were found on the figure when compared with the as-built configuration shown in OFD-124C series flow diagrams. These include:

- an extra branch line upstream of 3HPSW-269 and 3HPSW-270 which does not exist on OFD-124C-1.5.
- there are no cooling lines to the HPI pump motors from the First Floor Aux Bldg Header downstream of 1HPSW-17.
- The Primary and Breathing Air compressors are supplied from the M line header but not shown in UFSAR Figure 9-10 as indicated on OFD-124C-2.2.

This activity revised UFSAR Figure 9-10 to correct the above errors to be consistent with the as-built configuration shown on the OFD-124C series Flow Diagrams.

SAFETY EVALUATION SUMMARY

This High Pressure Service Water (HPSW) System is a non-QA, non-seismic system. This activity provided only a documentation change to the UFSAR for consistency with the as-built configuration of the HPSW System.

These changes have been previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-205)

DESCRIPTION

SYSTEM: Coolant Storage (CS)

This activity provided changes to UFSAR Figure 9-20 to make it more accurately reflect the actual Coolant Storage System configuration. This figure contained several inconsistencies with the Coolant Storage System configuration as shown on Oconee flow diagrams OFD-101A-1.1, 2.1, 3.1 and OFD-107A-1.1, 1.2, 2.1, 2.2, 3.1, 3.2.

This activity modified UFSAR Figure 9-20 as follows:

- Reverse the numbers shown on the figure for the number of SG shell side and tube side vents discharging to the Quench Tank.
- Add a Seal #1 header relief valve discharge line to the Quench Tank (Unit 1 only).
- Add a N2 FDW Vent line to the Quench Tank (Unit 2 and 3 only).
- Add a N2 blanketing line to the Quench Tank (all units).

SAFETY EVALUATION SUMMARY

The Coolant Storage System for each unit is designed to accommodate the accumulated coolant bleed over a core cycle, including startup expansion and coolant letdown to storage for boric acid reduction. A quench tank, located inside the reactor building, condenses and contains any effluent from the pressurizer safety valves and various vents. The quench tank and component drain pump portions of the Coolant Storage System is shown on UFSAR Figure 9-20. The as-built configuration of these portions of the Coolant Storage System are shown on Oconee flow diagrams. Several inconsistencies existed between Figure 9-20 and the flow diagrams. It is appropriate to modify UFSAR Figure 9-20 to make it more accurately reflect the actual as-built configuration of the Coolant Storage System. Changing the UFSAR in this manner does not invalidate any conclusions reached in the SAR and makes UFSAR Figure 9-20 more consistent with the actual plant configuration.

These changes have been previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-206)

DESCRIPTION

SYSTEM: Coolant Treatment System

This activity provided changes to UFSAR Figure 9-21 to make it more accurately reflect the actual Coolant Treatment System configuration. This figure contained several inconsistencies with the Coolant Treatment System configuration as shown on Oconee flow diagrams OFD-106A-1.1, 2.1, 3.1.

The RC Bleed Evaporator and its associated equipment and the Miscellaneous Waste Evaporator and its associated equipment are part of the Coolant Treatment System that is no longer used for coolant processing. These components were once considered for removal from the plant but ALARA and cost concerns prevented their removal from occurring. Although these components have been out of service for some time, the Oconee flow diagrams indicate this equipment is still connected to existing operating systems/components. UFSAR Figure 9-21 is not consistent with the flow diagrams in that the figure does not indicate that the RC Bleed Transfer Pump 2B discharges to the Miscellaneous Waste Evaporator and the RC Bleed Evaporator Demineralizer. Similarly, Figure 9-21 does not indicate that the RC Bleed Transfer Pump 3B discharges to the RC Bleed Evaporator Demineralizer. The Coolant Treatment System flow diagrams indicate these connections still exist.

This activity modified UFSAR Figure 9-21 as follows:

- Add lines to the Reactor Coolant (RC) Bleed Evaporator Demineralizer and the Miscellaneous Waste Evaporator from RC Bleed Transfer Pump 2B discharge.
- Add a line to the RC Bleed Evaporator Demineralizer from RC Bleed Transfer Pump 3B.

SAFETY EVALUATION SUMMARY

The Coolant Treatment System was originally designed and installed to both store RC bleed and to treat RC bleed for recycling. Since the boron recycling portion of the original Coolant Treatment System never functioned properly, the coolant storage portion is the only part of the system still in use at Oconee.

Based on this information, it was appropriate to modify UFSAR Figure 9-21 by adding connections to the RC Bleed Evaporator Demineralizer and the Miscellaneous Waste Evaporator from the 2B RC Bleed Transfer Pump discharge and by adding a connection to the RC Bleed Evaporator Demineralizer from the 3B RC Bleed Transfer Pump discharge. This change makes the UFSAR Figure more consistent with existing system configuration shown on the Oconee flow diagrams and does not invalidate any existing conclusions reached in the SAR. Performing the change associated with this activity does not change any design, operation, or analyses currently evaluated in the SAR nor does it change any information that could be used to prevent safety related components from performing their safety functions.

These changes have been previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the changes described in this activity do not present an unreviewed safety question.

UFSAR CHANGES (Pkg 97-208)

DESCRIPTION

SYSTEM: SFP Cooling, Reactor Building Spray

During the annual UFSAR update review, several discrepancies were identified in Chapter 9. The following changes are being made to the UFSAR per PIPs 97-1269 and 97-3353 as follows: (1) Correct minor errors pertaining to the description of the Spent Fuel Cooling System configuration and function, and (2) clarify the capacity requirements for the Reactor Building Spray System. This activity updated the UFSAR to provide a more accurate and detailed description of the Spent Fuel Cooling and Reactor Building Spray Systems.

SAFETY EVALUATION SUMMARY

The as-built design, configuration, function, performance, and integrity of the systems and components are not affected by these revisions. Changing this information in the UFSAR does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This revision does not in any way change the physical characteristics of the Station or its operations. No safety concerns or unreviewed safety questions are created. No Technical Specification changes are required. USFAR Chapter 9 sections 9.1.3 and 9.4.6 was revised accordingly.

UFSAR CHANGE (Pkg 97-209)

DESCRIPTION

SYSTEM: Low Pressure Injection and Core Flood Systems

The following minor inconsistencies were identified for UFSAR Figure 9-19, in relation to existing UFSAR Figure 6-1 and the as-installed configuration per the applicable Operational Flow Diagrams, OFD-102A-1.1, 1.3; 2.1, 2.3; 3.1, 3.3:

- Flow diagrams specify relief valve CF-15, for each unit, providing relief of Core Flood Tank (CFT) A and relief valve CF-17, for each unit, providing relief of CFT B. UFSAR Figure 9-19 specifies the reverse.
- Flow diagrams and existing UFSAR Figure 6-1 identify the Low Pressure Injection (LPI) pump B train as providing suction to the "B" Reactor Building Spray (RBS) pump. UFSAR Figure 9-19 specifies both the LPI A and B trains supplying the RBS B pump.

This activity updated UFSAR Figure 9-19 for consistency with the as-built configuration identified in the applicable flow diagrams and in existing UFSAR Figure 6-1.

SAFETY EVALUATION SUMMARY

This activity provided minor clarification of CFT relief valve and RBS suction header designations for UFSAR Figure 9-19. UFSAR Figure 9-19 identified both LPI headers providing suction to the "A" RBS pump and does not identify RBS "B" pump LPI suction. Additionally, the CFT Tank A and B relief valves identified on the current figure were the reverse of the as-built configuration detailed on the applicable flow diagrams. The changes were only for consistency with existing UFSAR information and the as-built configuration identified on the applicable flow diagrams, OFD-102A-1.1, 1.3; 2.1, 2.3; 3.1, 3.3.

These changes have been previously evaluated and approved, and they do not in any way increase the likelihood of initiation, or adversely affect the mitigation of, any SAR described accidents. There is no increase in the consequences of any SAR described accident. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There is no physical change to the plant or procedures. This activity also has no effect on any margins of safety as previously evaluated in the SAR. As such, the change described in this activity does not present an unreviewed safety question.

UFSAR CHANGE (Pkg 97-211)

DESCRIPTION

SYSTEM: None

This safety evaluation determines if any unreviewed safety questions (USQs) are involved for proposed revisions to UFSAR Sections 9.6.1 and 9.6.2. These clarifications were made as a result of resolution of PIPs associated with licensing basis open items from the Oconee Safety Related Designation Clarification (OSRDC) Project. The revisions to UFSAR Section 9.6.1 and 9.6.2 clarify the licensing basis for TBF with respect to the equipment which is credited for mitigation of this event.

SAFETY EVALUATION SUMMARY

This change to UFSAR Section 9.6.1 is acceptable because it is a clarification to more accurately reflect the licensing basis for use of electrical power supplies as described in the docketed correspondence between Duke and the NRC staff. The change to UFSAR Section 9.6.2 is acceptable because it is an addition of detail to the description of the mitigation strategy of a turbine building flood. A description was added to describe the mitigation strategy and licensing history of the TBF because no such description existed in the UFSAR. This description accurately reflects the Oconee licensing basis for a TBF as described in docketed correspondence between Duke and the NRC. These UFSAR revisions do not impact public health and safety. These revisions to the UFSAR did not result in any plant modifications, procedure changes, or other activities which could have resulted in an unreviewed safety question. No Technical Specification changes were required. No Selected Licensee Commitments were required. UFSAR Sections 9.6.1 and 9.6.2 were revised to clarify the Oconee turbine building flood (TBF) event licensing basis.

VIII. CALCULATIONS

DESCRIPTION

SYSTEM: None

This evaluation covers the utilization of vendor supplied firmware called MathCad (Versions 6.0 & Plus 6.0) in performing Engineering analysis and computations on Safety Related Systems, Structures, and Components (SSC's). Revision 3 to Calculation COM-208.00-00-0001, has been performed to certify the accuracy of MathCad versions 6.0 and Plus 6.0 and to provide documentation for allowing the checker of a calculation created in MathCad versions 6.0 & Plus 6.0 to utilize the self-checking features of MathCad when checking the calculation instead of independently duplicating all mathematical operations. MathCad (Versions 6.0 & Plus 6.0) will be used for the design and analysis of Safety Related SSCs.

SAFETY EVALUATION SUMMARY

Since software used for design or analysis of QA Condition 1, 2, 3, or 4 structures, systems, or components could potentially affect the accuracy of the calculations performed on and/or related to the SSC's, Revision 3 of Calculation COM-208.00-00-0001 documents the details of MathSoft Inc.'s Master Test Plan. This test was utilized to meet the requirements of IEEE Standard 829 to verify the accuracy of all calculations performed by MathCad versions 6.0 & Plus 6.0. In addition there are no "unseen" operations as no operations can be performed or data entered without direct permanent entry on the computer screen/printed page. As a result any errors in a MathCad version 6.0 or Plus 6.0 calculation are readily visible to the originator, to the checker, and to anyone looking at the page (on screen or paper printout) in the future.

Because of the aforementioned verifications, utilization of this software version does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. This activity does not modify the physical plant or operating procedures. The utilization of MathCad versions 6.0 & Plus 6.0 to perform Calculations for the design and analysis of QA Condition 1, 2, 3, & 4 Structures, Systems, and Components involves no USQ's or safety concerns and requires no UFSAR or technical specification changes.

CALCULATION

DESCRIPTION

SYSTEM: Nuclear Fuel

Calculation OSC-7078 contains a generic 10CFR50.59 analysis and safety review for fuel assembly reconstitution at Oconee. This analysis evaluated the substitution of a Mk-B11 natural uranium replacement rod for a standard Mk-B11 fuel rod.

SAFETY EVALUATION SUMMARY

The thermal and mechanical design limits of the standard Mk-B11 fuel rods envelope or bound the natural uranium replacement rods. The normal reload analyses verify the replacement rod operates within the criteria given in the UFSAR for internal pin pressure, creep collapse, stress, strain, strain fatigue, linear heat rate to melt, DNB, and LOCA initialization. The effects on the power distribution, of the fuel assembly reconstitution, are evaluated in the normal reload analyses process, using NRC approved methodologies, to ensure that the fuel does not exceed mechanical and thermal limits. The safety margin for the 1% failed fuel criterion (Technical Specification 3.1.4) is not impacted since the replacement rods are mechanically and functionally equivalent and the natural uranium replacement fuel rod fission product release is lower than the standard Mk-B11 fuel rod. Since the natural uranium replacement rods residence time, pin power levels, and fission product gas production are less than the standard Mk-B11 fuel rod design, the consequences of a fuel handling accident are bounded by the current analysis in Section 15.11.2 of the ONS UFSAR. The internal pin pressure analysis for the limiting fuel rod will bound the natural uranium replacement fuel rod. Considering a reactivity excursion due to control rod ejection. The replacement of a damaged rod by a natural uranium or stainless steel replacement rod results in a flux perturbation in that core location, but the pin power levels and reactivity are lower. The replacement rod and adjacent rods pin power are bounded by the peak pin. The replacement rods are functionally equivalent to the standard Mk-B11 fuel rods. The replacement rods could fail during Condition III and IV transients, but are contained in the fuel assembly. The use of natural uranium replacement rods does not adversely impact dynamic response of the fuel assembly subjected to seismic and LOCA loading or the capability to maintain a coolable geometry during a seismic and LOCA event. There are no USQs, and no changes to the UFSAR or Technical Specifications required.

CALCULATION

DESCRIPTION

SYSTEM: Nuclear Fuel

During the video examination of the fuel assemblies during the Oconee Unit 1 EOC 17 outage, Oconee personnel noted 22 fuel assemblies had damaged grids. The damage appeared to be similar in nature to the Oconee Unit 3 damage that previously caused three fuel assemblies to be recaged and one to be reconstituted. Calculation OSC-7023 justifies the continued usage of 14 of the damaged fuel assemblies in Oconee 1 Cycle 18 in their current condition and performs a 50.59 analysis of the impacts of doing so. The results of this calculation show that the fourteen fuel assemblies are acceptable for continued usage with damaged grid corners. The fourteen assemblies are listed below:

NJO763 NJO76M
NJO764 NJO76R
NJO768 NJO847
NJO773 NJO83W
NJO775 NJO84F
NJO75D NJO834
NJO75Y NJO84A

SAFETY EVALUATION SUMMARY

The continued use of the subject fuel assemblies is justified because (1) they have experienced one full cycle of irradiation without failure, (2) no fuel failures have ever been attributed to damaged grids, and (3) none of the rods are loose. The buckling of individual grid corners does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. No potential loose parts are created. This activity does not modify the physical plant or operating procedures. This analysis has determined that there are no unreviewed safety questions created by, or technical specification limits affected, by this activity. A largely editorial change was made to the Oconee UFSAR 9.1.4.2.3 per NSD-220 change package 97-57 for clarification of defective fuel examination.

CALCULATION

DESCRIPTION

SYSTEM: Nuclear Fuel

Calculation OSC-6907 includes a 10CFR50.59 analysis and safety review for the Oconee 1 Cycle 18 reload.

SAFETY EVALUATION SUMMARY

The normal reload analyses verify the subject core operates within the acceptance criteria limits. The reload design process was performed in accordance with NRC approved methodologies. The predicted physics parameters are bounded by the UFSAR analyses. This core is similar in operating characteristics to those of previous succesful designs. This reload design does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There are no USQs, and no changes to the UFSAR or Technical Specifications are required.

CALCULATION

DESCRIPTION

SYSTEM: Nuclear Fuel

Calculation OSC-6154 includes a 10CFR50.59 analysis and safety review for Oconee 1 Cycle 17 reload. Two revisions were performed; (1) account for a 10 ppmb change in beginning of cycle concentration due to an error in applying conservatisms and (2) to verify operation with three pumps for the remaining cycle length after 400 EFPDs was acceptable.

SAFETY EVALUATION SUMMARY

The normal reload analyses verify the subject core operates within the acceptance criteria limits. The reload design process and the subject revisions were performed in accordance with NRC approved methodologies. The predicted physics parameters are bounded by the UFSAR analyses. These changes/ reverifications of the O1C17 reload design does not in any way initiate, mitigate, or increase the consequences of any SAR described accidents. There is no adverse affect on any SSC, and no increase in the probability of a malfunction of equipment important to safety. No new radiological release pathways, or failure modes are created. There are no USQs, and no changes to the UFSAR or Technical Specifications are required.

XI. MISCELLANEOUS

DESCRIPTION

SYSTEM: Main Steam (MS) System

On the Unit 3 "B" Steam Generator, problems were being experienced with the turbine bypass valves (TBVs). It was necessary, due to circumstances associated with plant conditions, to take both TBVs out of service on this steam generator for maintenance to resolve the problems with the TBVs. This 50.59 evaluated whether or not there were any unreviewed safety questions associated with this activity.

SAFETY EVALUATION SUMMARY

Both of the TBVs on one Main Steam header can be isolated for maintenance purposes while the unit is on line. Isolation of both TBVs on a steam header has been analyzed with respect to accident consequences. Offsite dose following a loss of load or steam generator tube rupture are within the limits specified in the SAR. The TBVs are not required to be functional by Technical Specifications and the basis addresses the unavailability of all TBVs. As a result, it was determined this activity does not cause, or adversely affect the mitigation of, any previously analyzed SAR accidents. No new radiological release pathways or failure modes are created. The components continue to perform their design functions during normal and accident conditions. Based on the safety evaluation performed, no unreviewed safety questions are created by this minor modification. No changes to the Technical Specifications or UFSAR were required.

ACRONYMS AND ABBREVIATIONS

Absolute	Abs
Anticipated Transients Without Scram	ATWS
Accumulator	Accum
ATWS Mitigation System Actuation Circuitry	AMSAC
Acknowledge	Ack
Active	Act
Administration	Admin
Air circuit breaker	ACB
Air compressor	Air Comp
Air conditioner (ing)	A/C
Air handling unit	AHU
Alarm	Alm
Alternate	Alt
Alternating current	AC
Amperes	Amps
Approximate (ly)	≈ or ~ (Approx)
As Low As Reasonably Achievable	ALARA
Atmosphere	Atmos
Automatic	Auto
Auxiliary	Aux
*Auxiliary Instrument Air System	AIA
Auxiliary oil pump	AOP
Auxiliary service water	ASW
*Auxiliary steam system	AS
Auxiliary transformer	Aux Xformer
Average	Avg (av)

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Average temperature	T_{AVE} (Tave)
Backup	BKUP
Basement	BSMT
Battery	Batt
Battery charger	Batt chgr
Bearing	Brng
Bearing lift pump	BLP
Blanket	BLKT
Bleed	BLD
Bleed holdup tank	BHUT
Block	BLK
Block valve	Blk Vlv
Blower	BLWR
Borated water storage tank	BWST
Boric acid mix tank	BAMT
Boron 10	B_{10}
Breaker	BKR (Bkr)
*Breathing air system	BA
British thermal unit	BTU
Building	Bldg
*Building spray system	BS
Bypass	Byp
Cabinet	CAB
Carbon dioxide	CO_2
Carbon monoxide	CO
Center line	C_L

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Change	Chng (chg)
Channel	CH
Charger	chgr
Check valve	CHK VLV
Chemical	Chem
*Chemical addition system	CA
Chloride	Cl
Circuit	CKT
Circulating	Circ
Closed	CLSD
Column	COL
*Component cooling system	CC
Compressor	Comp
Computer	Comptr
Concentrate	Conc
Concentrated boric acid storage tank	CBAST
Condensate booster pump	CBP
Condensate monitor tank	CMT
Condensate steam air ejector	CSAE
Condensate storage tank	CST
*Condensate system	C
Condensate test tank	CTT
*Condenser circulating water system	CCW
Conductivity	Cond
Containment	CONT
Control	CTRL

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Control rod drive	CRD
Control rod drive mechanism	CRDM
Control Room	CR
Control valve	CV
Coolant	CLNT
*Coolant storage system	CS
*Coolant treatment system	CT
Cooldown Procedure	CP
Cooler	CLR
Cooling	CLNG
Core exit thermocouples	CETCs
*Core flood system	CF
Core flood tank	CFT
Correction	CORRT
Corridor	CORRD
Counts per minute	CPM
Counts per second	CPS
Crisis Management Center	CMC
Crossconnect	XCONN
Crossover	X-OVER
Cubic feet	ft^3
Cubic feet per minute	cfm
Current transformer	CT
Damper	Dmpr
Deborating (ate)	Debor

ENCLOSURE 5.4
ACRONYMS AND ABBREVIATIONS

Decades per minute	DPM
Decay heat removal	DHR
Decontamination (ate)	Decon
Degree	Deg
Degrees Centigrade	°C (Deg C)
Degrees Fahrenheit	°F (Deg F)
Dehumidifier	Dehum
Delta	Δ
*Demineralized water system	DW
Demineralizer	Demin
Desuperheater	Desuphtr
Detector	Det
Diameter	Dia
Diesel generator	DG (D/G)
Differential	DIFF
Differential pressure	ΔP (D/P)
Direct current	DC
Discharge	Disch
Diverse Scram System	DSS
Dose Equivalent Iodine	DEI
Double pole double throw	DPDT
Double pole single throw	DPST
Down	DWN
Downcomer	DNCMR
Drain valve	Drn Vlv
Drawing	DWNG (DWG)

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Dry Storage Canister	DSC
Effluent	EFF
Electrical	ELEC
Electro hydraulic control	EHC
*Electro hydraulic control system	HO
Elevated water storage tank	EWST
Elevation	ELEV
Emergency	EMER
Emergency bearing oil pump	EBOP
Emergency core cooling systems	ECCS
Emergency feedwater	EFDW
Emergency feedwater pump	EFDWP
Emergency feedwater pump turbine	EFDWPT
Emergency power switching logic	EPSL
Emergency seal oil pump	ESOP
Enclosure	Encl
Engineering safeguards	ES
Engineering safety feature actuation system	ES (ESFAS)
Equipment	Equip
Evacuation/ate	EVAC
Evaporator	EVAP
Exchanger	EXCHNGR
Exhaust	Exh
Exhauster	EXHTR
Expansion	EXPN
Expansion joint	EXPJT

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Extended SG SU Range	XSUR
External	EXT
Feeder	FDR
Feedwater pump	FDWP (FWP)
Feedwater pump turbine	FDWPT
*Feedwater system	FDW
Feet	ft (')
Feet per second	fps
Filter	FLTR
*Fire hydrant system	FH
First, second, third	1st, 2nd, 3rd, etc
First stage reheater	FSRH
First stage reheater drain tank	FSRHDT
Flow transmitter	FT
Forced draft fan	FD FAN
Forward	FWD
Frequency	FREQ
Fuel Assembly	FA
*Fuel oil system	FO
Full Power	FP
Gallon	gal
Gallons per hour	gph
Gallons per minute	gpm
*Gaseous waste disposal system	GWD
Gaseous waste disposal tank	GWD TK
Gaseous waste release	GWR

*Valve designator for that sytem

ACRONYMS AND ABBREVIATIONS

Generator	GEN (Gen)
Governor	GOV
Governor valve	GOV VLV
Ground	GND
Header	HDR (Hdr)
Heater	HTR
Heater drain pump	HDP
*Heater drains system	HD
*Heater vent system	HV
Heating, ventilation and air conditioning	HVAC
High	HI
High activity waste tank	HAWT
High efficiency particulate air	HEPA
*High pressure extraction system	HPE
High pressure injection pump	HPIP
High pressure injection system	HPI
*High pressure injection system	HP
*High pressure service water system	HPSW
High range	HR
Holdup	HU
Horizontal	Horiz
Horizontal Storage Module	HSM
Hotwell	HW
Hotwell pump	HWP
Hour	Hr
Hydraulic	HYDR

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Hydrazine	NH_4
Hydrogen	H_2
Hydrogen ion concentration	pH
*Hydrogen system	H
Inactive	IN/ACT
Inadequate Core Cooling	ICC
Inadequate Core Cooling Monitor	ICCM
Inboard	I/B
Inch	in.
Inches of water	in.H_2O
Inches of mercury	in Hg
Incore Thermocouples	CETC
Incorporated	INC
Independent Spent Fuel Storage Installation	ISFSI
Indication & Control	IC
Inhibit	INHIB
Injection	INJ
Inlet	INLT
Instrument	INST
*Instrument air system	IA
Instrument and Electrical Department	I&E
Instrument Root Valve	IRV
Insulation	INSUL
Integrated Control System	ICS
Integrated Leak Rate Test	ILRT
Interim rad waste	IRW

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Intermediate Range neutron detector	IR
Inverter	INVTR
Ion exchanger	IX
Irradiated Fuel Assembly	IFA
Isolation (ate) (ed)	Isol (ISOL)
Junction	JCT
Kilovolt	kV
Kilovolt-ampere	kVA
Kilovolt-ampere reactive	kVAR
Kilowatt	kW
Kilowatt-hour	kWH
Laundry and hot shower tank	LHST
Lead	Pb
*Leak rate test system	LRT
Letdown	L/D
Letdown storage tank	LDST
Level	LVL
Level transmitter	LT
Limiting Condition of Operation	LCO
Liquid	LIQ
*Liquid waste disposal	LWD
Liquid waste release	LWR
Lithium hydroxide	LiOH
Load center	LDCTR (LC)
Load frequency control	LFC
Locked closed	L.C.

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Locked open	L.O.
Loss of coolant accident	LOCA
Low activity waste tank	LAWT
*Low pressure extraction system	LPE
Low pressure injection	LPI
Low pressure injection pump	LPIP
*Low pressure injection system	LP
*Low pressure service water	LPSW
Low Range	LR
Lube oil purifier	LOP
Main Computer	MC
Main feeder bus	MFB
Main feeder bus monitoring panel	MFBMP
Main feedwater	MFDW
Main feedwater pump	MFDWP
Main seal oil pump	MSOP
*Main steam	MS
Main steam control valve	MSCV
Main steam intercept valve	MSIV
Main steam relief valve	MSRV
Main steam stop valve	MSSV
Main Turbine	MT
Main turbine oil tank	MTOT
Make up	M/U
Manual	MAN
Maximum	MAX

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Maximum Permissable Concentration	MPC
Mechanical	MECH
Megavolt ampere reactive	MVAR
Megawatt	MW
Megawatt electrical	MWe
Megawatt thermal	MWt
Mezzanine	MEZZ
Microcuries per milliliter	µCi/ml
Minimum	MIN
Minute	Min
Miscellaneous	Misc
Miscellaneous waste holdup tank	MWHUT
Moisture separator drain tank	MSDT
Moisture separator drain pump	MSDP
Moisture separator reheater	MSRH
Moisture separator reheater drain tank	MSRHDT
Monitor	MON
Motor	MTR
Motor control center	MCC
Motor driven emergency feedwater pump	MD EFDWP
Motor gear unit	MGU
Motor operated	MO
Motor operated disconnect	MOD
Motor operated valve	MOV
Motor speed changer	MSC
Mulsifyre	MLSFYR
Narrow range	NR

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Negative	Neg
Net positive suction head	NPSH
Neutral	NEUT
Nil ductility temperature	NDT
Nitrogen	N_2
*Nitrogen system	N
Non Licensed Operator	NLO
Non-Nuclear Instrumentation	NNI
Normally	Norm
Normally closed	N.C.
Normally open	N.O.
Nuclear instruments	NI
Nuclear Policy Manual	NPM
Oconee Nuclear Station	ONS
Oil circuit breaker	OCB
Oil lift pump	OLP
Operate	Oper
Operating Range	OR
Operations	Ops
Operations Management Procedure	OMP
Operations Support Center	OSC
Operator aid computer	OAC
Outboard	O/B
Outlet	OTLT
Overflow	OVF
Overhead	OVHD

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Overload	OVLD
Oxygen	O_2
Package	PKG
Panel	PNL
Panel board	PNLBD
Particulate, absolute, charcoal filter	PAC filter
Parts per billion	ppb
Parts per million	ppm
Parts per million boron	ppmb
Penetration	Pen(t)
Penetration room	Pen(t) Rm
Penetration room ventilation	PRV
*Penetration room ventilation system	PR
Phase	Ø
*Plant heating steam system	PH
Pneumatic	PNEU
Pneumatic circuit breaker	PCB
Polishing	POL
Polishing demineralizer system	POWDEX
Position	POSN
Positive	POS
Potential	Pot
Potential transformer	PT
Pounds mass per hour	lbm/hr
Pounds per hour	LB/HR (lb/hr)
Pounds per square inch	psi

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Pounds per square inch absolute	psia
Pounds per square inch differential	psid
Pounds per square inch gauge	psig
Power	PWR
Power factor	PF
Power operated relief valve	PORV
Power range	PR
Power supply	PS
Pressure	Press
Pressure & Temperature	P/T
Pressure gauge	PG
Pressure transmitter	PT
Pressurizer	PZR
Preventative maintenance	PM
Primary	PRI
Problem Investigation Report	PIR
Public address system	PA
Pump	pmp (P)
Purge	PRG
Purifier (cation)	Purif
Quality assurance	QA
Quantity	QTY
Quench tank	QT
Radial	RADL
Radiation monitor	RIA

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Radiation Protection	RP
Radioactive Waste Facility	RWF
Reactor	RX
Reactor building	RB
Reactor building cooling unit	RBCU
Reactor building normal sump	RBNS
Reactor building spray	RBS
Reactor building vent	RBV
Reactor coolant average temperature	T_{ave} (Tave)
Reactor coolant bleed holdup tank	RC BHUT
Reactor coolant cold leg temperature	T_c
Reactor coolant hot leg temperature	T_h
Reactor coolant inventory monitoring system	RCIMS
Reactor coolant makeup	RCMU
Reactor coolant pump	RCP
Reactor coolant system	RCS
*Reactor coolant system	RC
Reactor Operator	RO
Reactor protective system	RPS
Reactor vessel	RXV
Reactor vessel level instrumentation system	RVLIS
Recirculating (ate)	Recirc
*Recirculating cooling water system	RCW
Recirculating seal oil pump	RSOP
Recorder	RCDR
Rectifier	Rect

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Reference	Ref
Reflash	R/F
Refrigeration	Refrig
Regenerative	REGEN
Reheat stop valve	RSV
Reheater	RHTR
Relay	RLY
Relief valve	RV
Required	REQD
Resistance temperature detector	RTD
Return	RTN
Revision	REV
Revolutions per minute	RPM
Room	Rm
Sample	SMPL
Saturation pressure	P_{sat}
Saturation temperature	T_{sat}
Schematic	SCHEM
*Seal oil system	SO
Seal oil vacuum pump	SOVP
Second	Sec
Second stage reheater	SSRH
Second stage reheater drain tank	SSRHDT
Secondary	SEC
Section	SECT

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Senior Reactor Operator	SRO
Sequence	SEQ
*Service air system	SA
Shield wall	SH
Shielding	SHLD
Shut down	SD
Single pole double throw	SPDT
Single pole single throw	SPST
Small break loss of coolant accident	SBLOCA
Source range neutron detector	SR
Spare	SPR
*Spent fuel cooling system	SF
Spent fuel pool	SFP
Spent resin storage tank	SRST
Standard cubic centimeter per minute	SCCM
Standard cubic feet per minute	SCFM
Standard cubic feet per second	SCFS
Standby	Stby
Standby Shutdown Facility	SSF
Start up	SU
Startup range	SUR
Stator	STATR
*Stator coolant system	SC
Stator cooling water	SCW
Stator cooling water pump	SCWP

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Steam	Stm
*Steam drain system	SD
Steam generator	SG
Steam generator (restricted usage)	OTSG
Steam generator Operating Range level	O.R.
Steam generator tube rupture	SGTR
Steam packing exhausts	SPE
*Steam seal system	SSH
Stop valve	SV
Strainer	STRNR
Structure	STRUCT
Subcooling margin	SCM
Suction	SUCT
Superheater	Suphtr
Switch	SW
Switch board	SWBD
Switch gear	SWGR
Switch yard	SWYD
Synchronize	SYNC
System	SYS
Tank	Tk
Technical Specifications	T.S. (Tech Specs)
Technical Support Center	TSC
Temperature	Temp (T)
Temperature change	ΔT

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Temperature transmitter	TT
Thermal shock operating region	TSOR
Thermocouple	TC (T/C)
Thrust	THR
Transfer	Xfer
Transformer	Xformer
Transmitter	Xmitter
Tritium	H_3
Trouble	TRBL
Turbine	Turb
Turbine building	Turb Bldg (TB)
Turbine building sump	TBS
Turbine bypass valves	TBVs
Turbine driven EFDWP	TD EFDWP
Turbine generator	Turb Gen (T/G)
*Turbine lube oil system	TO
Turning gear	TG
Turning gear oil pump	TGOP
Unbalanced	UNBAL
Under voltage	UV
Uninterrupted Power Source	UPS
Upper surge tank	UST
Vacuum	Vac
Vacuum Drying System	VDS
*Vacuum system	V

*Valve designator for that system

ACRONYMS AND ABBREVIATIONS

Valve	Vlv (VLV)
Ventilation	Vent
Vibration	Vib
Volt	v
Volt ampere	VA
Volt ampere reactive	VAR
Voltage alternating current	VAc
Voltage direct current	VDc
Voltage regulator	VREG
Volume	Vol
Waste disposal	WD
Waste gas filter	WG filter
Waste monitor	WM
Water	WTR (H_2O)
Wide range	WR
Winding	WDNG
Withdrawal	WITHDRWL

*Valve designator for that system