ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

10 CFR 50.59(b)(2)
SUMMARIES

(SEE ATTACHED)

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT

JUNE 1, 1999 - APRIL 30, 2001
SUMMARY OF
SAFETY EVALUATIONS

JUNE 1, 1999 - APRIL 30, 2001

SUMMARY OF SAFETY EVALUATIONS FOR CORE COMPONENT DESIGN CHANGE REQUESTS AND CORE OPERATING LIMITS REPORTS

TVA-COLR-BF3C10, Revision 0

The safety evaluation for TVA-COLR-BF3C10 addresses the BFN Unit 3 Cycle 10 reload core design, incore fuel shuffle, Core Operating Limits Report (COLR), and associated Safety Analysis Report (SAR) revisions. The reload core design and licensing analyses for Cycle 10 were performed by GNF with results documented in the Supplemental Reload Licensing Report. The reload analyses were performed in accordance with NRC approved methodology as described in General Electric (GE) licensing topical report GESTAR II (NEDE-24011-P-A-13). Operating limits for the cycle (i.e., Linear Heat Generation Rate, Minimum Critical Power Ratio, and Average Planar Linear Heat Generation Rate) are incorporated into the Unit 3 COLR.

The Cycle 10 core is a scatter loaded core design which includes 288 fresh GE 13 fuel assemblies. Operating flexibility options analyzed include increased Core Flow, Final Feedwater Temperature Reduction, Feedwater Heaters Out-of-Service, and EOC Recirculation Pump Trip Out-of-Service, Turbine Bypass Valves Out of Service, and Recirculation Single Loop Operation. A mid-cycle exposure point was included for additional Minimum Critical Power Ratio (MCPR) margin throughout the majority of the cycle. Consistent with NRC commitments, GE also performed a cycle specific Safety Limit MCPR analysis.

The Unit 3 Reload 9 refueling will be accomplished by an incore fuel shuffle using a TVA developed shuffle sequence. Shutdown margin calculations have been performed for each unique core configuration in the planned shuffle sequence.

There is no increase in the probability or consequences of an accident or malfunction of equipment from that previously evaluated in the SAR. The proposed activity does not create the possibility of a different type of accident or malfunction from that previously evaluated in the SAR. The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, this change does not involve an unreviewed safety question.

TVA-COLR-BF2C12

The safety evaluation for TVA-COLR-BF2C12 addresses the BFN Unit 2 Cycle 12 reload core design incore fuel shuffle, Core Operating Limits Report (COLR), and associated Final Safety Analysis Report/Technical Requirement Manual revisions. The reload core design and licensing analyses for Cycle 12 were performed by Global Nuclear Fuel (GNF - formerly General Electric [GE] Nuclear Energy) with results documented in the Supplemental Reload Licensing Report. The GNF analyses were performed in accordance with NRC approved methodology as described in GNF licensing topical report GESTAR II (NEDE-24011-P-A-13). Operating limits for the cycle (i.e., Linear Heat Generation Rate, Minimum Critical Power Ratio, and Average Planar Linear Heat Generation Rate) are incorporated into the Unit 2 COLR.

The cycle 12 core is a Low Leakage Scattered Loaded Core design with includes 256 fresh GE13 fuel assemblies equipped with debris filter-lower tie plates. Cycle 12 is designed to be a 2-year cycle licensed to operate at 105% power-uprate conditions. Operating flexibility options analyzed include Increased Core Flow, Final Feedwater Temperature Reduction, Feedwater Heaters Out-of-Service, EOC Recirculation Pump Trip Out Of Service (RPTOOS), Turbine Bypass Valves Of Out Of Service (TBVOOS), and Recirculation Single Loop Operation. As a first time application, Cycle 12 is also analyzed for both RPTOOS and TBVOOS in combination. Consistent with NRC commitments, GNF also performed a cycle specific Safety Limit MCPR analysis.

The Unit 2 Reload 11 refueling will be accomplished by an incore fuel shuffle using a TVA developed shuffle sequence. Shutdown margin calculations were performed for each unique core configuration in the planned shuffle sequence.

There is no increase in the probability or consequences of an accident or malfunction of equipment from that previously evaluated in the Safety Analysis Report (SAR). The proposed activity does not create the possibility of a different type of accident or malfunction from that previously evaluated in the SAR. The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, this change does not involve an unreviewed safety question.

JUNE 1, 1999 - APRIL 30, 2001

SUMMARY OF SAFETY EVALUATIONS FOR FIRE PROTECTION REPORT REVISIONS

Fire Protection Report-R15

This safety evaluation for the Fire Protection Report (Volume 1) addresses changes to the Appendix R Safe Shutdown Program. The resulting changes enhance the Appendix R Safe Shutdown Program by ensuring that the proper equipment functions are listed for the applicable fire zones/areas thus ensuring that the necessary compensatory measures can be taken when necessary. The changes associated with the 24V DC diesel fire pump batteries bring these batteries in line with the other battery systems in use at BFN and will ensure the maximum service life for these batteries. There will not be any adverse impact on the function or operation of any safety related equipment or systems. Therefore, there will not be any adverse impact on the safe shutdown capability of the plant and thus no adverse impact on nuclear safety. These changes have been reviewed against the criteria of 10CFR50.59 and do not constitute an unreviewed safety question.

JUNE 1, 1999 - APRIL 30, 2001

SUMMARY OF
SAFETY EVALUATIONS
FOR
PLANT MODIFICATIONS

Engineering Change Notice P0320A

The safety evaluation for ECN P0320A is prepared to allow the use of the Low Level Radwaste, On-Site Storage Facility (LLRW OSF) module 2 for storage of up to 42,000 curies total in High-Integrity Container (HIC), minus any inventory in waste handling operations of up to 325 curies. The safety evaluation justifies the current physical configuration of the LLRW OSF. The total yearly generated activity stored in the facility shall not exceed 9,988 curies. This limitation is based on limiting the yearly cumulative exposure received by LLRW handling personnel. The curie limit will be included in the implementing procedures. This safety evaluation also supports required revisions to the Final Safety Analysis Report (FSAR). Neither the original design features of the mobile gantry crane or security features such as a security gatehouse, closed circuit TV, alarmed fence, back-up power, etc. are included in the current design. These features are not required to safely store LLRW on-site, based on the guidance provided by Generic Letter 81-38. Existing Critical Structures, Systems, and Components (CSSC) equipment is unaffected, and existing accident scenarios are not impacted by this change. Accident scenarios associated with operation of the LLRW OSF modules fall well within other qualified releases evaluated in the SAR. LLRW OSF module handling equipment and operations are not considered to be important to safety. However, the OSF modules themselves are considered safety-related, and are designed and constructed not to lose their integrity. The modules are designed for tornado, earthquake, flood, wind, precipitation, and handling events, making the possibility of accidents or malfunctions and consequences thereof not previously evaluated as implausible. The LLRW OSF facility modules are not addressed in the Technical Specifications, nor does it affect plant process systems or their operations. Therefore, the storage of radioactive material/waste in HICs, with the HICs stored in the LLRW OSF module 2 has been determined not to be an unreviewed safety question on the basis that it represents normal operation, and a total of 42,000 curies can be stored within the LLRW OSF module 2.

Design Change Notice S17557D

The safety evaluation for DCN S17557D addresses the Secondary Containment system which is a physical barrier designed to prevent the uncontrolled release of radioactive materials into the environment surrounding the plant. When Primary Containment is intact, the Secondary Containment serves as a secondary barrier which, in the event of release of radioactivity into the Reactor Building atmosphere, contains the necessary reliable redundant components and subsystems to isolate, contain and assure a filtered and controlled elevated release of the Reactor Building atmosphere. When the Primary Containment is not intact, such as during refueling and maintenance operations, the Secondary Containment membrane serves as the Primary Containment barrier. The reactor building exterior walls, floors, penetrations and the built up roof, stepped fascia panels, and metal siding of the structure covering the Refuel Floor forms the Secondary Containment membrane. This membrane is to limit the release of radioactivity to the environs after an accident so that the resulting exposures are kept to a practical minimum and are within the guideline values given in 10CFR20 and 10CFR100.

The Secondary Containment membrane is divided into four ventilation zones. The refuel floor zone is common to all three units and runs the entire length of the reactor building. The individual reactor units below the refuel floor form the other three zones. This four zone ventilation control configuration allows increased capability for localizing the consequences of an accident such that the effects in one or more isolated ventilation zones will not affect operation in an unaffected unit/zone.

This safety evaluation addresses a change which allows the four ventilation zones to be interconnected allowing free flow of each reactor zone atmosphere with the common refuel flow atmosphere. The combined zone Secondary Containment System is obtained by removing the blowout panels at the three equipment hatches, located along the south wall of the reactor building.

A reactor zone ventilation system is isolated whenever a group 6 isolation signal is initiated by high radiation levels in the ventilation exhaust duct leaving that zone, or by manual alignment. The refuel floor zone is isolated when any reactor zone isolates, by a high radiation signal on the refuel floor, or by manual action. When a ventilation zone is isolated, the ventilation system serving that zone shuts down and Standby Gas Treatment System (SGTS) is started and begins exhausting from the isolated zone(s). The SGTS is located on the west side (Unit 1 end) of the reactor building. SGTS has two parallel suction paths one from the southwest corner of the refuel floor and one from the three units. The unit suction points are connected in series, Unit 1, 2, and 3 with approximately 170 feet between the unit suction points. The SGTS creates a negative pressure within Secondary Containment to cause air to leak into the Secondary Containment rather than leak out to the surrounding environment.

When the reactor zones are physically isolated from the refuel zone, it is possible for a reactor zone to be unable to reach the required negative pressure. This is dependent on the amount of in-leakage in an isolated reactor zone and the distance from the SGTS fans. The refuel floor zone SGTS suction is parallel to the unit suctions and would not be affected unless the total in-leakage was more than the capacity of the SGTS.

The combined zone Secondary Containment System allows the total volume within the Secondary Containment membrane to be used to verify the integrity of Secondary Containment. Surveillance Procedure (SR) 0-SR-3.6.4.1.3, Combined Zone Secondary Containment Draw-down and Integrity Test, verifies that Secondary Containment in-leakage is less than 12,000 standard cubic feet per minute when Secondary Containment pressure is equal to or less than 0.25 inch of H_20 vacuum in accordance with the requirements of Technical Specifications SR 3.6.4.1.3 and SR 3.6.4.1.4. This SR requires verification that all reactor zones are interconnected to the refueling floor with an opening of at least 60 square feet.

Design calculations for evaluation of High Energy Line Breaks (HELB), Equipment Qualification (EQ), Tornado Depressurization, Pressure Transients within Containment, System Requirements, and Radiation doses were reviewed to determine impact by the combined zone Secondary Containment system. Based on review of the calculations, the combined zone Secondary Containment System is acceptable. However, dose calculations to the operators in the Control Room considered only Unit 2 and Unit 3 operation. The combined zone Secondary Containment

System will require re-evaluation should Unit 1 operation be resumed. This limitation will be reflected on design drawings and in the text of the FSAR.

From review of the Design Basis Accidents (DBA) in Chapter 14 of the Final Safety Analysis Report, the integrity of the Secondary Containment System was considered for the mitigation of the consequences resulting from a DBA. However, credit was taken for the entire volume within the Secondary Containment membrane which is not affected by this change and is verified to meet Technical Specifications as described above. No credit for zonal isolation within the Secondary Containment Boundary was taken.

The Abnormal Operational Transients considered include the Control Rod Removal Error during refueling and Fuel Assembly Insertion Error during refueling. These transients were considered because Primary Containment would be open leaving only Secondary Containment to contain radioactive releases. From review of Chapter 14, these transients are precluded by refueling interlocks to prevent inadvertent criticality. The Secondary Containment Boundary will be intact and unaffected with the reactor building ventilation zones interconnected.

The special event considered is tornado. In the event of a tornado, excessive internal pressures within the reactor building are relieved through blowout panels. These blowout panels are designed to relieve at 50 psf and vent the refueling room to the atmosphere. In the combined zone Secondary Containment configuration the reactor zones will also be vented and pressures within the Secondary Containment membrane will be equalized, decreasing pressure differences between ventilation zones. The Secondary Containment membrane will be breached which requires the operating reactors to be shutdown. However, Primary Containment will be intact to prevent excessive dose to the environment. Thus, this change does not change the evaluations performed for tornado depressurization effects.

There are no credible failure modes associated with this change. The Secondary Containment Boundary will remain intact and unaffected.

Therefore, this change does not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR), does not create a possibility for an accident of a different type than any evaluated previously in the SAR or reduce a margin of safety for any Technical Specifications. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T33433A

The safety evaluation for DCN T33433A addresses the disconnect of Turbidity Loop 0-TU-43-25G and its abandonment in place. The affected loop is designated a chemistry loop; however, the loop is located in a Radwaste line and was initially setup to control Radwaste flow control valves. The function of the control loop was to divert turbid liquid waste from going to the Cleanup Phase Separators (CPS) and instead direct it to the Waste Collector Tank (WCT). However, a return line from the CPS to the Waste Package Drain Tank (WPDT) allows the flow of bottoms from the CPS to the WPDT. Therefore, the turbidity control loop provides a redundant function and has been determined to be unreliable. Consequently, the loop no longer performs a required function and is being removed from service. The function of the loop is being replaced by grab samples, laboratory analysis, and manual valve control. Operation of the required function will be controlled via Radwaste System and Chemistry Procedures. Since the loop is no longer required to perform a function, removing it from service does not affect nuclear safety. No unreviewed safety question is involved.

Design Change Notice T37002A

The safety evaluation for DCN T37002A addresses the removal of various pieces of equipment in the Intake Pumping Station (IPS) associated with System 28, (Water Treatment). The system has not been used for many years, is not currently in use, and is partially abandoned in place. There is a TACF (0-92-1-28) which reflects the abandoned equipment and which can be closed by reflecting this configuration on the drawings. Makeup water is supplied by other means and there are no plans to ever use the original equipment. Some components not able to be removed shall be abandoned in place in accordance with SSP 9.3. They shall be placed behind walls constructed to maintain IPS functions and ingress/egress paths. Some of the equipment will remain in service and be changed from System 28 to System 40 (Station Drainage). This equipment is associated with a waste sump. Three shutoff valves and three check valves which provided water to the Water Treatment plant from Condenser Circulating Water are changed from System 28 to System 27. System 32 components which supplied Control Air to the Water Treatment System are disconnected since System 28 will not be operable. All Sampling System 43 components which tie into System 28 are disconnected. System 25 (Raw Service Water) provides seal water to the Waste Transfer Pumps. This connection is also disconnected and the System 25 valves spared.

System 28 is an auxiliary system which has been replaced by a vendor (Ecolochem) supplied, vendor operated makeup water system. It has never had the ability to affect accident or transient conditions in the plant. The Condensate Storage Tanks maintain a qualified supply of demineralized water for plant emergency needs. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T39426A

The safety evaluation for DCN T39426A addresses the revision of plant drawings and equipment tagging to clearly define Unit 3 Electric Board Room Air Conditioning Unit (ACUs) as Shutdown Board Room ACUs. Also, Drawing 3-47E865-4 will be revised to rename the Shutdown Board Room Exhaust Fan to Electric Board Room Exhaust Fan to be consistent with the other drawing changes. The only physical work involved for this DCN is tagging associated components to reflect the correct equipment description. This change involves equipment labeling changes only and does not change the operation of any system or component. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T39462A

This safety evaluation for T39462A addresses replacement of the electrical heat strips on Air Particulate Radiation Monitor 3-RM-90-256 with electrical heat trace (heat tape). This change will prevent "Hot Spots" created by the heat strips. Existing Temperature Switches 3-TS-90-256B, 3-TS-90-256C, and 3-TS-90-256D will also be removed. In addition, the local alarm indicating lights and hand switch associated with the radiation monitor heating system will be removed. The existing heat trace on the 1 inch sample line tubing to the radiation monitor will be replaced with new heat tape. The Air Particulate Radiation Monitor 3-RM-90-256 only provides a monitoring function and is not an accident initiator. The electrical heat tape on the radiation monitor permits the monitor to accurately determine the radiation level of samples from the drywell by keeping the sample temperature above its dewpoint temperature. The only consequence of an accident credible to this proposed design change is an unmonitored drywell atmosphere since Emergency Core Cooling System (ECCS) and other safety functions are not affected by the radiation monitor. However, the radiation monitor will provide the control room operator with an annunication upon detecting an abnormal condition. Upon receiving the alarm, the control room operator is required to request Chemistry to provide periodic sampling of the drywell atmosphere. Replacing the electrical heat strips with the electrical heat tape does not increase the consequences of an unmonitored drywell atmosphere because the replacement heating system performs the same function and will fail in the same manner as the existing heating system. The proposed design change will not change any setpoint value associated with the radiation monitor or its sample flow. This change does not involve an unreviewed safety question.

Design Change Notice W39881A

The safety evaluation for DCN W39881A makes relatively minor wiring changes in Unit 3 Main Control Room Panel 9-14. These changes cause the Oscillation Power Range Monitor (OPRM) scram trip and control rod block functions to be enabled and to cause appropriate control room annunciations. The oscillation detection algorithms and the associated scram trip comprise the stability long term solution required to Generic Letter 94-02 for Unit 3. The change also revises NESSDs for the Unit 3 Neutron Monitoring System to include the effects of 24 month cycle operation. This change also includes change out of the Erasable Programmable Read Only Memory (EPROMs) in the Average Power Range Monitor/ Local Power Range Monitor (APRM/LPRM) modules, the APRM Operator Display Assembly (ODA) modules, and the Rod Block Monitor (RBM) ODA modules; these EPROM changes correct identified firmware defects, add test capability and improve human factors of the operator displays. This activity has been reviewed against the criteria of 10 CFR 50.59 and does not constitute an unreviewed safety question.

Design Change Notice T39933A

The safety evaluation for DCN T39933A documents the addition of Control Air (CA) System 032 Compressor G. This compressor will become the primary source for the CA system. The new compressor is capable of supplying 1445 SCFM at 120 psig and is set at 105 psig which provides all the control air needed for normal operation of the plant (based on operating experience) and also pressurizes the accumulator tanks for the Main Steam Isolation Valves (MSIV), Automatic Depressurization System Relief Valves, and the Reactor Building Equipment Access Lock inflated seals. The safety related portion of the CA system starts at the check valves upstream of the accumulator tanks and runs through to the isolation valves and inflatable seals. The four existing CA compressors will remain in service and will start/load sequentially as needed to supplement the new compressor. The new compressor is a configurable two stage compressor with the intercooler, aftercooler and lube oil cooler mounted on the same skid. The compressors (including the new compressor), receiver tanks and piping up to the accumulator tank check valves are in the non-safety related portion of the CA system. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T40195A

The safety evaluation for DCN T40195A addresses installation of new Unit 2 and Unit 3 Reactor Feedwater Pump (RFP) start-up bypass control valves, FCV-3-53 due to obsolescence. The new valve will provide the same function as the existing valve. The new valve will also fail to the closed position on loss of electrical power or signal since it has a direct acting position and the actuator is spring to close. This is a more desirable failure mode for the valve since it will prevent excess water from flowing to the reactor vessel and is bounded by Loss of Feedwater Flow evaluation at rated conditions. This change has been evaluated to ensure that the changed failure modes will not create an unreviewed safety question.

Design Change Notice T40241A

The safety evaluation for DCN T40241 addresses the design for the addition of a bypass line between the Residual Heat Removal (RHR) pumps and RHR pump seal heat exchangers. The design and required materials meet or exceed the existing system design criteria requirements. Equipment operational and safety parameters are not changed. The change was reviewed for impact against the Technical Specifications (TS) and the Final Safety Analysis Report (FSAR). The review found two FSAR Figures requiring revision. The change does not alter any equipment functions or features important to safety, nor does the change create any new accidents or malfunctions not previously evaluated in the SAR. The evaluation determined that the TS are not affected and therefore the margin of safety is not reduced. This change does not constitute an unreviewed safety question.

Design Change Notice W40314A

The safety evaluation for DCN W40314A addresses the installation of the Hydrogen Water Chemistry (HWC) Control Panels, Hydrogen Flow Control Module, Hydrogen Shutoff Valve Panel, Hydrogen Area Monitor Sensors, Offgas Oxygen Flow Control Module, and Offgas Oxygen Analyzer. The HWC Control panel provides total hydrogen and oxygen injection control by monitoring plant power level and monitoring initial start-up to determine the required flow rate. The HWC hydrogen injection flow control module directs flow from the hydrogen supply arriving from the tank farm to the hydrogen injection. The HWC oxygen injection flow module directs flow from the oxygen supply arriving from the tank farm to the injection line leading to the common Steam Jet Air Ejector suction line. Hydrogen Area Monitors Sensors are mounted above the hydrogen injection flow control module to detect hydrogen leakage to the atmosphere. These sensors will alarm at local panel if they detect a one (1) percent hydrogen in air mixture, and will initiate a system shutdown if a two (2) percent hydrogen in air mixture is detected. The Offgas Monitor Panel monitors the Offgas steam downstream of the recombiners for hydrogen and oxygen concentrations and provides HWC System shutdown signals on either high or low oxygen content. Applicable portions of the Safety Analysis Report (SAR) figures are being revised to show the Browns Ferry Nuclear Plant Unit 3 configuration with the HWC system installation. New procedures will be required to control the operation and testing of the HWC system. These new procedures will not differ with system operation characteristics nor conflict with or affect a process or procedure currently outlined, summarized, or described in the SAR. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T40592A

The safety evaluation for DCN T40592A addresses the removal of the check valves which allow flow through the Service Air piping system out of the Reactor Building. There are no anticipated functional modes which require the system to operate in this manner. These valves can be removed without affecting the operation of the Service Air system. The safety-related function of the valve to provide pressure boundary retention for Secondary Containment can be met by closing the piping system with standard piping fittings (welded cap) designed to meet the existing requirements (material, pressure and temperature) of the system.

The Secondary Containment shall be designed to act as a radioactive material barrier under the same conditions that require the Primary Containment to act as a radioactive material barrier. The Secondary Containment shall be designed to act as a radioactive material barrier, if required, whenever the Primary Containment is open for expected operational purposes. The Primary and Secondary Containments, in conjunction with other engineered safeguards, shall act to prevent the radiological effects of accidents resulting in the release of radioactive material to the containment volumes from exceeding the guideline values of applicable regulations.

During normal operation and when isolated, the Secondary Containment is maintained at a negative pressure relative to the building exterior. It is designed for an in-leakage less than 100 percent of building volume per day. The safety objective of the Secondary Containment System is to limit the release of radioactivity to the environs after an accident so that the resulting exposures are kept to a practical minimum and are within the guideline values given in published regulations (10CFR20 and 10CFR100 as applicable).

The check valves must be removed when the Service Air System is not operating. Isolating the piping system to remove the valve would require isolation of the entire Reactor Building served by the valve. Work on the valve cannot cause an accident or malfunction of a component required for safe shutdown of the plant. The Secondary Containment boundary through the Service Air piping system must be maintained during the removal of the valve. This can be accomplished by closing the shutoff valve in the Turbine Building which supplies that portion of the system. The zonal isolation between units must be isolated or the work shall be performed on one valve at a time. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T40618A

The safety evaluation for DCN T40618A addresses a modification to provide an alternative temporary means of bypassing an Residual Heat Removal (RHR) valve interlock, which prevents the opening of the suppression pool isolation valve when the shutdown cooling suction valves are open. This will allow flushing of the RHR shutdown cooling subsystem. To bypass the interlock and allow the system to be flushed, 2/3-OI-74 Section 8.7.1.5 places a jumper across the limit switches of the suction valves FCV-74-2/-13 (FCV-74-25/-36) allowing valves FCV-74-57 (FCV-74-71) to be open. DCN T40618 will install a locally mounted key-lock type hand switch to replace the jumpers and bypass both fully closed limit switches on valves FCV-74-2 and -13 (-25 and -36) during the Shutdown Cooling System (SCS) flush procedure. The hand switch will require a key to bypass the interlock. The keys will allow the hand switches to be placed in the bypass position but must be returned to the normal position to remove the keys. Also, an amber light, located with each switch will illuminate when the switch is in the bypass position. Putting the switches in the bypass position will replace the steps of adding jumpers to the limit switches and reduce the amount of time required to perform the flushing activity. The devices added by this modification will not cause the isolation valves to malfunction in such a way as fail to perform their safety function and the potential to drain down the reactor vessel is not increased since it would take an unlikely failure of the switch coupled with a series of deliberate operator actions to create the necessary valve alignment. Therefore the margin of safety for shutdown cooling is not affected. This change does not involve an unreviewed safety question.

Design Change Notice T40641A

The safety evaluation for DCN T40641 addresses the installation of flow limiting orifices in the Unit 1 Fuel Pool Cooling System (FPCS) heat exchangers' inlet lines, heat exchanger common bypass line and the demineralizer bypass line to limit demineralizer bypass flow and prevent pump run-out. Similar orifices are installed on Unit 2 under DCN T40643A and Unit 3 under DCN T40644A. Design drawings are revised to show this modification. These drawings will be put into the Final Safety Analysis Report (FSAR) as figures in Section 10.5. Therefore, this safety evaluation is provided to document the expected change to the FSAR figures created by this DCN.

The FPC and Demineralizing system is provided to remove decay heat from fuel assemblies and maintain fuel pool water within specified temperature limits and water quality. During accidents or operational transients, the FPCS piping that penetrates secondary containment is designed to confine radioactive material to the reactor building in order to prevent a radioactive release to the environment.

The FPCS has no active role in mitigating any accidents or transients. The non-safety related system is designed and built to Seismic Class II requirements so that equipment in the FPCS will not interfere with or prevent any safety related equipment located in the same area from performing its intended safety function during and following an earthquake. Installation of the orifices reduces the potential for equipment failure from pump run-out due to low surge tank level and maintains system cooling and demineralizer flow within original design parameters. Thus, neither the decay heat removal capability nor the demineralizer cleanup capability of the FPCS is altered by this modification. No new failure modes are created by this change and reliability of the cooling function may be improved by reducing the probability of pump run-out induced failures. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T40643A

The safety evaluation for DCN T40643A addresses the installation of flow limiting orifices in the Unit 2 Fuel Pool Cooling System (FPCS) heat exchangers' inlet lines, heat exchanger common bypass line and the demineralizer bypass line to limit demineralizer bypass flow and prevent pump run-out. Design drawings are revised to show this modification. The design drawings 2-47E855-1 & 2-47E610-78-1 appear as figures 10.5-1a and 10.5-1b, sheet 1 in the Final Safety Analysis Report (FSAR), therefore a SAR change is being made under this DCN. Similar orifices are installed on Unit 1 under DCN T40641A and Unit 3 under DCN T40644A.

The FPC and Demineralizing system is provided to remove decay heat from fuel assemblies and maintain fuel pool water within specified temperature limits and water quality. During accidents or operational transients, the FPCS piping that penetrates secondary containment is designed to confine radioactive material to the reactor building in order to prevent a radioactive release to the environment.

The FPCS has no active role in mitigating any accidents or transients. The non-safety related system is designed and built to Seismic Class II requirements so that equipment in the FPCS will not interfere with or prevent any safety related equipment located in the same area from performing its intended safety function during and following an earthquake. Installation of the orifices reduces the potential for equipment failure from pump run-out due to low surge tank level and maintains system cooling and demineralizer flow within original design parameters. Thus, neither the decay heat removal capability nor the demineralizer cleanup capability of the FPCS is altered by this modification. No new failure modes are created by this change and reliability of the cooling function may be improved by reducing the probability of pump run-out induced failures. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T40644A

The safety evaluation for DCN T40644A addresses the installation of flow limiting orifices in the Unit 3 Fuel Pool Cooling System (FPCS) heat exchangers' inlet lines, heat exchanger common bypass line and the demineralizer bypass line to limit demineralizer bypass flow and prevent pump run-out. Similar orifices are installed on Unit 1 under DCN T40614A and Unit 2 under DCN T40643A. Design drawings are revised to show this modification. Drawings 2-47E855-1 & 2-47E610-78-1 do not presently appear in the Final Safety Analysis Report (FSAR). However, these drawings will be put into the FSAR as figures in Section 10.5.

The FPC and Demineralizing system is provided to remove decay heat from fuel assemblies and maintain fuel pool water within specified temperature limits and water quality. During accidents or operational transients, the FPCS piping that penetrates secondary containment is designed to confine radioactive material to the reactor building in order to prevent a radioactive release to the environment.

The FPCS has no active role in mitigating any accidents or transients. The non-safety related system is designed and built to Seismic Class II requirements so that equipment in the FPCS will not interfere with or prevent any safety related equipment located in the same area from performing its intended safety function during and following an earthquake. Installation of the orifices reduces the potential for equipment failure from pump run-out due to low surge tank level and maintains system cooling and demineralizer flow within original design parameters. Thus, neither the decay heat removal capability nor the demineralizer cleanup capability of the FPCS is altered by this modification. No new failure modes are created by this chance and reliability of the cooling function may be improved by reducing the probability of pump run-out induced failures. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T40670A

The safety evaluation for DCN T40670A was initiated to reduce the unidentified inleakage to radwaste by providing a design for replacing the existing flange clamp seal with a more reliable seal which completely encapsulates the channel/shell flange joint of the 3A Reactor Water Cleanup (RWCU) Regenerative Heat Exchanger. The change was reviewed for impact against the Technical Specifications and the Final Safety Analysis Report (FSAR). This safety evaluation is required due to the change affecting FSAR Figure 4.9-5. The heat exchanger was supplied to withstand higher temperature and pressure values than are required by the system. The FSAR figure change is to document the temperature and pressure at the heat exchanger due to the new material not being qualified to the higher valves of the heat exchanger. The design and required materials meet the design criteria system requirements. The change does not alter any equipment functions or features important to safety, nor does the change create any new accidents or malfunctions not previously evaluated in the FSAR. The RWCU text is not altered due to the change nor is any other system description altered. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T40682A

The safety evaluation for DCN T40682A addresses the removal of Temporary Alteration Control Form (TACF) 0-97-01-77. This TACF currently allows installation of a portable sump pump which discharges to a floor drain through a base. The Radwaste Off-Gas Condensate Drain Sump pump discharge will be rerouted from the (clean) waste collector system to the (dirty) floor drain collector system. This action is required because the Off-Gas Condensate Drain Sump can become contaminated with ethylene glycol or raw water and then the waste sample tanks must be discharged to the river, rather than recycled to the plant. The water treated by the floor drain collector system is always considered to be dirty and is treated accordingly. This change will prevent waste sample tanks from having to be discharged to the river due to unacceptable quality water from the Off-Gas Condensate Drain Sump.

Rerouting the discharge from the Radwaste Off-Gas Condensate Drain Sump from the waste collector tank to the floor drain collector tank does not create any different failure modes of the radwaste system. The floor drain collector tank has a slightly lower volume (38,000 gallons vs. 32,000 gallons). Increasing the flow to the floor drain collector tank will only increase the frequency at which system is required to operate. The quality of the processed fluids will not be affected. The piping will be designed and installed in accordance with the required code and quality requirements.

Release doses from the radwaste system will not be affected by changing the path used for the treatment of the waste. Dose rates must still be within the limits of 10CFR20 and 10CFR100. The rerouted piping is installed in the overhead of the same room as the existing piping. A new shielding analysis is not required. The system will be operated within these parameters. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T40713A

This safety evaluation addresses DCN T40713A which provides the design to replace Reactor Core Isolation Cooling (RCIC) steam supply inboard and outboard isolation gate valves FCV-71-2 and FCV-71-3 in Unit 2 and 3. Also, DCN T40713A replaces approximately 12 feet of 4" piping downstream of the Unit 3, FCV-71-3 with 3" piping. The 3" valves and associated piping are being replaced to improve component performance and are being installed in accordance with approved plant procedures and specifications. The RCIC system piping and equipment, including support structures, shall be designed to withstand the effects of an earthquake without a failure which could lead to a release of radioactivity in excess of the guideline values given in 10CFR100 as the systems design basis. This change does not introduce any new credible failure modes. It was found that this modification does not affect any accidents evaluated in the Final Safety Analysis Report, since the changes does not alter how the RCIC system performs safety functions nor are any other system's required safety functions impacted. Therefore, this modification does not involve an unreviewed safety question.

Design Change Notice T40719A

The safety evaluation for DCN T40719A addresses the replacement of the 500kV Potential Transformers (PTs), the addition of surge arresters to the 500kV line termination, and the addition/deletion/replacement of various breaker trip auxiliary relays which do not diminish the capability of the 500kV switchyard. The DCN controls the work process such that the required offsite power sources will not be challenged. The addition of these devices will improve the 500kV switchyard system by limiting incoming surges to levels below the capabilities of the associated circuit breaker insulation. This safety evaluation shows that:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is <u>not</u> increased and

2. The possibility of an accident or malfunction of a different type than previously evaluated is <u>not</u> created and

3. There is <u>no</u> reduction in a margin of safety as defined in the bases for any Technical Specification.

Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T40720A

The safety evaluation for DCN T40720A addresses the addition of vacuum interrupters and surge arresters to the 161kV line termination which does not diminish the capability of the 161kV switchyard. The DCN controls the work process such that the required offsite power sources will not be challenged. The addition of these devices will improve the 161kV switchyard system by limiting incoming surges to levels below the capabilities of the associated circuit breaker insulation. The replacement relays for the type AR relays in the 161kV breaker trip circuits and in the cooling tower relay boards do not perform a safety related function. These relays will reduce the probability of nuisance operations due to voltage transients. The tripping diodes in the cooling tower transformer sudden pressure relay auxiliary circuit do not have a safety function. Removing these diodes will increase the life of the sudden pressure auxiliary relay contacts and will not increase the probability of an accident evaluated in the Final Safety Analysis Report (FSAR) and Living FSAR. This safety evaluation shows that:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is <u>not</u> increased and

2. The possibility of an accident or malfunction of a different type than previously evaluated is <u>not</u> created and

3. There is <u>no</u> reduction in a margin of safety as defined in the bases for any Technical Specification.

Therefore, this change does not involve an unreviewed safety evaluation.

Design Change Notice T40746A

The safety evaluation for DCN T40746A addresses changing the Off-Gas System Dilution Fan manual outlet dampers 2-DMP-066-1101 and 3-DMP-066-1102 from a normally open position to a throttled position. Since the total dilution flow through the stack is not significantly changed, the new configuration does not involve an unreviewed safety question.

Design Change Notice T40749A

The safety evaluation for DCN T40749A addresses the replacement of the Reactor Water Cleanup (RWCU) blowdown valves with new valves that have similar operating characteristics, but provide better shutoff and are designed to withstand the severe service conditions of high differential pressure. The valves will receive the same opening and closing signals as the existing valves, are controlled by air or electric motor and will fail to the same position (closed or as-is) on loss of air or power.

There are no design basis accidents for which the valves are required to operate. During normal operation, the valves are closed and must stay closed to eliminate lost inventory. During various modes of startup, operation or shutdown, the valves are open to allow control of reactor water level (blowdown). Replacement of the valves must occur by isolating this portion of the system from the RWCU system and from the condenser. Neither replacement of the valves nor operation with the new valves can cause an accident or malfunction of equipment important to safety. Operation of the valve is not required to mitigate the consequences of any design basis accident or the malfunction of equipment important to safety. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T40757A

The safety evaluation for DCN T40757A was initiated to modify the Unit 2 and Unit 3 High Pressure Coolant Injection System (HPIC), System 073 and the Reactor Core Isolation Cooling System (RCIC), System 071, turbine steam supply drain piping. The drain piping for both systems in Units 2 and 3 are very similar, perform the same function, and have the same design parameters. To prevent the HPCI and RCIC steam supply piping from filling with water, a condensate drain pot, steam line drain, and appropriate valves are provided in a drain pipeline arrangement just upstream to the turbine supply valve. During normal operation, water collected in the condensate drain pot is routed to the main condenser through a steam trap. Currently, valves are provided upstream and downstream of the steam trap for maintenance on the trap. A bypass, around the steam trap is provided which contains a level control air operated solenoid valve. This valve is controlled by a level switch in the condensate pot which will automatically open the valve due to high water level in the condensate drain pot. An alarm is also sounded in the control room for high water level in the condensate drain pot on panel 9-3. Downstream of the steam trap and the bypass piping, both systems have two flow control valves. These valves are normally open and automatically close with turbine initiation. When the turbine is operating, condensate in the supply piping is discharged from drain lines from the turbine stop valve seat, steam chest areas, and the turbine casing exhaust drains.

This modification will provide double isolation valves upstream and downstream of the subject trap and provide flange fittings on each side of the steam trap. This modification will allow maintenance to be performed on the steam traps without closing the HPCI containment outboard isolation valve, 73-3, or the RCIC containment outboard isolation valve, 71-3. The change is requested because existing level control valves are obsolete and like for like replacement is costly.

These modifications do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR). Thus, the possibility of an accident or malfunction of a different type than evaluated previously in the SAR is not created and no margins of safety are reduced. Based on this discussion, these modifications do not involve an unreviewed safety question.

Design Change Notice T40760A

This safety evaluation addresses DCN T40757A which replaces the site's main sewage lift station. The operation of the lift station is changed from pneumatically lifting the collected sewage to a holding tank and then pumping to the holding ponds; to lifting and pumping the collected sewage directly to the holding ponds through electric motor operated pumps. The new system has capacity equivalent to the existing and will reduce the required maintenance on the existing obsolete components. This change does not affect the accident analysis for the plant and does not affect the operation of any equipment required for the safe shutdown of the plant. This change has been reviewed against the criteria of 10CFR50.59 and does not constitute an unreviewed safety
question.

Design Change Notice W40964A

The safety evaluation for DCN W40964A addresses the installation of a connection for hydrogen injection into condensate booster pumps to reduce free oxygen concentration to mitigate the possibility of intergranular stress corrosion cracking in sensitized stainless steel.

The hydrogen injection is a part of Hydrogen Water Chemistry system (system #004) and is a non-safety related system. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T41019A

The safety evaluation for T41019A addresses rerouted 2" Main Steam piping, new isolation valves and a new actuator for valve 3-PCV-001-0147. This change will provide reasonable assurance that leakage past the closed Main Steam Isolation Valves (MSIVs) will be directed to the condenser through the preferred pathway. The seismically rugged pathway will help mitigate the effects of a Loss of Coolant Accident (LOCA) by providing a reliable pathway for any leakage past the closed MSIVs to the condenser.

These design changes do not alter the Technical Specification limits imposed on MSIV leakage, nor do they alter the design or operating characteristics of the system. This design change does change the failure position for the 3-PCV-001-0147 valve, but this change increases the reliability of the alternate leakage treatment pathway.

This activity has been reviewed against the criteria of 10CFR50.59 and does not constitute an unreviewed safety question.

Design Change Notice T41101A

The safety evaluation for DCN T41101A addresses a proposed modification for Units 1, 2, and 3. A process computer is provided for Units 2 and 3, which supplements procedural requirements for the control of rod worth during control rod manipulations during reactor startup and shutdown. Unit 1 is in "lay-up" with no fuel in the reactor. However, data from some equipment in Unit 1 is provided to the process computers in Units 2 & 3.

This change will delete the Radiation Release Rate Monitor System (RRRMS) computer after relocating the remaining RRRMS computer Inputs to the Process Computer System (PCS), also referred to as the Integrated Computer System (ICS). This DCN will also connect the Units 1, 2 and 3 Continuous Air Monitors (CAM) and Area Radiation Monitors (ARM) to the Integrated Computer System to facilitate the dissemination of "real-time" radiological data to emergency control centers and will connect temperature data from the Residual Heat Removal system, Fuel Pool Cooling system, and Recirculation Pumps to enhance monitoring and data trending of these plant parameters. The proposed modification affects equipment associated with monitoring only. This change does not involve an unreviewed safety question.

Design Change Notice T41118A

The safety evaluation for DCN T41118A addresses the revision of flow control diagram 0-47E845-2 to correct discrepancies between the actual plant configuration and the design basis drawings. Moisture drain traps for Units 1 and 3 Service Air Headers in the Reactor Building are shown on the flow control diagram but are not installed in the plant. The configuration for each of these lines is actually a 1" hose service connection with a normally closed shutoff valve similar to the Unit 2 configuration. (Unit 2 is depicted correctly.) The flow control diagram is revised to delete the moisture traps for Units 1 and 3 and show valves 1,3-SHV-33-1062 as normally closed service connections. This drawing appears in the Final Safety Analysis Report (FSAR) as Figure 10.14-2b, so this safety evaluation change supports the FSAR change.

The non-safety related Service Air system has no role in the initiation or mitigation of any Design Basis Accident. This change only affects plant documentation and component labeling for identification. No physical change is made to any plant system. No system or component is made to function, to be operated, maintained or tested in any different manner. Thus, no new failure modes are created. This is an insignificant documentation and labeling change on non-safety related equipment. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T41146A

The safety evaluation for DCN T41146A addresses the removal of the Gland Seal Water (GSW) supply shutoff valves to the Feed Water (FW) startup recirculation (long cycle) valves on Unit 2 and 3. The GSW supply lines will be sealed off with threaded caps or plugs. The long cycle valves' leak-off lines will be sealed with welded caps or plugs. The affected GSW supply lines and valves provide no safety related function. The lines are outside the boundary which provides integrity to the primary containment or reactor coolant pressure boundary. Removing the GSW supply to the FW system's long cycle valve stem does not affect the function or operation of the FW system or create the potential for a new leakage path. The plant drawings and Final Safety Analysis Report figures are revised to depict the appropriate configuration for the unused GSW supply lines and FW system valves' stem packing leak-off lines. These actions do not create any new failure modes. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T41188A

The safety evaluation supports the Final Safety Analysis Report change for DCN T41188A which installs tie-in piping and shutoff valves in the Unit 2 and 3 Drywell Control Air (DWCA) and Containment Inerting (CI) systems to allow future installation of a source of nitrogen from the CI system to the DWCA system. Installation of tie-in piping to the DWCA and CI systems does not alter their function or operating characteristics. No new failure modes are created by this change. The new piping and components are procured and installed to specifications which satisfy the material and quality requirements appropriate for these TVA Class M systems. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice W41191

The safety evaluation for DCN W41191 addresses replacement of the existing single channel instrument with the dual channel instrument from the same manufacturer on condensate feedwater panel 3-LPNL-25-103. The modification installs a new dissolved hydrogen sensor in the panel hood area. The new hydrogen sensor and the existing oxygen sensor is connected to the new dual channel Orbisphere analyzer. An existing nitrogen generator provides a continuous purge to the hydrogen sensor located in the condensate feedwater panel hood. The dissolved oxygen and hydrogen instruments are non safety and non-essential instruments. The power to these instrument is supplied from the spare instrument power sockets located in the panels. The modification does not require change to the Technical Specification or Technical Requirement Manual. However, Safety Analysis Report Figures 10.17-1D and 10.17-1E must be revised to reflect the plant configuration per the DCN. This DCN does not create an unreviewed safety question.

Design Change Notice T41259A

The safety evaluation for DCN T41259A addresses the modification of the Unit 3 Main Generator protection circuitry to eliminate an identified single point failure that could cause an unnecessary Unit 3 generator and turbine trip. The circuit for Generator 3 overcurrent relay 351G will be modified by DCN T41259A to block a trip signal which could be generated improperly by the 351G relay on opening of a supplying potential transformer fuse. The proposed change does not result in any physical or procedural changes to the plant which could alter the function or mode of operation of any plant system, nor does the proposed change adversely affect the safe shutdown capability of the plant. This modification has been reviewed against the criteria of 10CFR50.59. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T41261A

The safety evaluation for DCN T41261A addresses replacement of the existing generator breaker conductivity sensor, which is an obsolete component with readily available equivalent devices. The loop performs a non-quality, non-safety function providing Main Control Room indication and alarm. The measuring range and high conductivity setpoints are not changed and operator response remains the same. System operability and design basis are not affected by the replacement components. Therefore, this change does not introduce any new credible failure modes and does not introduce an unreviewed safety question.

Design Change Notice T41266A

The safety evaluation evaluates the implementation of DCN T41266A which makes permanent Temporary Alteration Change Form (TACF) 2-98-003-003. TACF 2-98-003-003 connects the output of level transmitters 2-LT-003-0208A, 2-LT-003-0208B, 2-LT-003-0208C, and 2-LT-003-0208D to the Intergated Computer System (ICS). In addition, this DCN will also connect Unit 3 level transmitters 3-LT-003-0208A, 3-LT-003-0208B, 3-LT-003-0208C, and 3-LT-003-0208C, and 3-LT-003-0208D to the Integrated Computer System (ICS). The non-safety related portion of the instrument loops 2-LT-003-0208A, 2-LT-003-0208B, 2-LT-003-0208C, 2-LT-003-0208D, 3-LT-003-0208A, 3-LT-003-0208B, 3-LT-003-0208C, and 3-LT-003-0208D are not associated with the initiation or mitigation of an accident. The proposed modification, which adds integrated computer system connections to the electrically-isolated non-safety related portion of the subject instrument loops, will not introduce any failure modes not bounded by the existing loop failure loops. The proposed modification affects the electrically-isolated non-safety related indication portions of the subject instrumentation loops, which are not described in the Technical Specifications. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice T41301A

The safety evaluation for DCN T41301 addresses the Reactor Core Isolation Cooling (RCIC) System inboard injection valves, 2/3-FCV-71-39, which are being modified. This modification consists of drilling a ¼" hole in the disc face on the feedwater side into the cavity between the disc faces for each valve. This hole will hydraulically connect the process piping on the feedwater side with the valve bonnet. This results in RCIC injection valve being more reliable to meet Technical Specifications and Final Safety Analysis Report requirements by ensuring that these valves are not susceptible to pressure locking. This activity is not an unreviewed safety question because it does not affect the Technical Specification requirements for the RCIC system nor any functional capabilities of any RCIC system components. In addition, this change does not affect any current accident analysis nor any design basis documents.

Design Change Notice T41356A

The safety evaluation for DCN T41356A addresses the proposed modification that will remove the disc position indication for 2-FCV-75-54, 2-FCV-75-26, 3-FCV-75-54, and 3-FCV-75-26 Core Spray (CS) testable check valves by removing the indicating lights and repairing the holes on panels 2-9-3 and 3-9-3. The associated wires will be lifted and spared. This activity will not affect the ability of the CS testable check valves to open in response to system flow from the CS pumps, or to close as required to isolate the reactor pressure vessel from the low pressure sections of the CS piping. This activity will no longer require American Society Mechanical Engineers Section XI testing of the disc position indication. No other plant equipment will be affected. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice 50020A

The safety evaluation for DCN 50020A addresses the Radiochemical Laboratory and related areas being refurbished/restored. The electrical changes associated with the Radiochemical Laboratory refurbishment do not affect any safety related equipment. The changes affect equipment and electrical service within the Radiochemical Laboratory, make related changes to the 480V radwaste, and remove obsolete recorders from panel 25-128 in the Radiochemical Laboratory. The existing interface of the 480V Radwaste Boards to the 480V Diesel Auxiliary Boards during flood conditions is not affected.

With the exception of Reactor Water Cleanup Conductivity, the changes associated with removal of recorders on panel 25-128 in the Radiochemical Laboratory affect systems whose operation is not required with Unit 1 shutdown and de-fueled. The Reactor Water Cleanup Conductivity instrument loops have recorders and alarms in the Main Control Room that monitor the same variable as those monitored by the recorders removed from panel 25-128. Additionally, samples may be taken to fulfill water quality monitoring requirements. Therefore, this activity does not constitute an unreviewed safety question.

Design Change Notice 50051B

The safety evaluation for DCN 50051B addresses the required plant changes for demolition and setting up a temporary Chemical Laboratory in the current Clean/Dirty Laundry Room. The temporary Chemical Laboratory will allow required Chemistry Department functions which support the plant operation to be performed while modifications are in the progress in the existing Chemical Laboratory. Refurbishment of the Chemical Laboratory does not change the ability of the permanent Chemical Laboratory to meet its required functions. There are not safety related functions for any equipment in the permanent Chemical Laboratory. Therefore, no unreviewed safety question is associated with this change.

Design Change Notice 50052A

The safety evaluation for DCN 50052 addresses changes to the controls and control schemes for Reactor Building Closed Cooling Water (RBCCW) flow through the Reactor Water Clean Up (RWCU) heat exchangers.

The new control scheme will measure the RWCU flow temperature coming out of the RWCU heat exchange to control the RBCCW cooling water flow through the heat exchangers. This will place the emphasis on maintaining the RWCU temperature at an optimum setting for the filter demineralizers. Temperatures above 130°F degrades the resin in the filter demineralizers and temperatures above 140°F cause an insolation of the system.

The revised control scheme will improve the thermal performance of both the RWCU and the RBCCW systems. RWCU will be improved by cooling it low enough for safe and efficient use of the filter demineralizers, while not cooling it too much before sending it back to the reactor. Since the RWCU flow will not be overcooled, the performance of the RBCCW system will also be increased.

This change does not impact nuclear plant safety, nor does it affect any accidents/accident analysis equipment important to safety/margin of safety. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice 50063A

The safety evaluation for DCN 50063A addresses Service Air and Demineralizer Water (DW) line routed through Drywell penetrations X20 and X21 on Units 2 and 3 which will be cut, capped, plugged and seal welded. These lines which supply service connections inside the Drywell for Service Air and DW are rarely used. Cutting and plugging these lines at their Drywell penetrations will eliminate the need for periodic Local Leak Rate Test (LLRT) testing of these penetrations.

This change cuts and caps/plugs the Service Air and DW supply lines to service connections in the Primary Containment. Since the Primary Containment is considered isolated in any accident scenario, the new configuration can not involve an unreviewed safety question.

Design Change Notice 50083A

The safety evaluation for DCN 50083A provides for replacement of the sudden pressure relays on the Unit 3 Main Bank (Phase A, B and C) and Unit 3 Station Service Transformers 3A and 3B. In addition, a second sudden pressure relay will be added in series with the replacement sudden pressure relay and will be connected in a two-out-of-two logic trip scheme. The new (and replacement) relays will be mounted to a pedestal which will be installed in the ground near the transformer. The relays will be piped to the transformers with flexible high-pressure steel-reinforced hose. This arrangement will reduce the potential for spurious operation due to transformer vibration, as will also the two-out-of-two trip logic. Both relays must operate in order to produce sources and a generator trip. There will not be any adverse impact on the safe shutdown capability of the plant and thus no adverse impact on nuclear safety. This modification has been reviewed against the criteria of 10CFR50.59 and does not constitute an unreviewed safety question.

Engineering Equivalent Change 50093A

The safety evaluation for EEC 50093 addresses the modification which will replace obsolete displacer level switches 2-LS-71-5, 2-LS-73-5, and 2-LS-73-8B. Test tees will be added to facilitate calibration. Also, isolation valves and test and drain valves will be replaced. The vent valves for 2-LS-73-8B will be deleted. Thus these changes do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR). These modifications do not change the automatic functions of the high pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC) systems. The intent of these modifications is to replace obsolete displacer level switches. These modifications do not change the function of the HPCI or RCIC turbine steam supply and exhaust drain piping systems. Thus, the possibility of an accident or malfunction of a different type than evaluated previously in the

SAR is not created and no margins of safety are reduced. Therefore, this change does not involve an unreviewed safety question.

Engineering Equivalent Change 50095A

The safety evaluation for EEC 50095 addresses a modification that does not affect the function of the High Pressure Coolant Injection (HPCI) or Reactor Core Isolation Cooling (RCIC) steam supply and turbine exhaust drain piping. The changes are only made to replace obsolete displacer level switches with similar switches qualified for the process pressure and temperature.

This modification will replace obsolete displacer level switches and test tees will be added to facilitate calibration. Also, isolation valves and test drain valves will be replaced. Thus these changes do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR). These modifications do not change the automatic functions of the HPCI or RCIC systems. The intent of these modifications is to replace obsolete displacer level switches. These modifications do not change the function of the HPCI or RCIC turbine steam supply and exhaust drain piping systems. Thus, the possibility of an accident or malfunction of different types than evaluated previously in the SAR is not created an no margins of safety are reduced. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice 50097A

The safety evaluation for DCN 50097A addresses installation of a block valve and associated test connection on the Core Spray system on Units 2 and 3 to allow Appendix J testing across 2/3-FCV-75-57. A ¾" test connection will be installed upstream of 2/3-FCV-75-57 and consists of piping and valves. The block valve will be installed upstream of 2/3-FCV-75-57 valve which is located on elevation of 519', NW quad, of both Units 2 and 3 and is in the suction line to the Pressure Suppression Chamber (PSC) Head Tank Pumps. The area in the Core Spray systems, at which these test connections will be added, are within the primary containment pressure boundaries for these systems.

The addition of one normally open block valve and associated test connections to allow Appendix J testing of the 2/3-FCV-75-57 does not affect the described safety functions of the Core Spray system. This modification will have no affect on the ability of 2/3-FCV-75-57 valves to perform their Primary Containment Isolation function, which is to close on an Isolation signal of either Reactor Water Vessel Low Water Level or High Drywell Pressure. Therefore, these changes do not constitute an unreviewed safety question.

Design Change Notice 50133A

The safety evaluation for DCN 50133A addresses removal of the Reactor Water Cleanup System (RWCU) sample return pumps and associated piping. The RWCU Sample Return pump removal is a result of a cost/benefit analysis done by Chemistry. This analysis concluded that the operation of the RWCU Sample Return pumps cannot be economically justified at this time, but did not recommend the removal or abandonment of the associated equipment. However, since the pumps are obsolete, and future operations of this system would require modifications, it was decided to remove this equipment now, via this DCN, and install new equipment in the future, via another DCN, if it is later determined economically feasible to operate this system. Since none of the equipment affected by this change is safety-related or important to safety, it is not included in any accident scenario. Therefore, the new configuration does not involve an unreviewed safety question.

Engineering Document Change 50134A

The safety evaluation for EDC 50134A provides the design documentation to change the Containment Atmospheric Dilution (CAD) injection valves from a solenoid operated gate valve to a solenoid operated globe valve on affected plant drawings, Safety Analysis Report table, system design criteria, and plant procedures. This change will show the affected valves as originally designed, evaluated, constructed, and installed. The plant design and calculations are not altered nor affected. Therefore, no unreviewed safety question is involved.

Engineering Equivalent Change 50138A

The safety evaluation supports EEC 50138A which reroutes the Unit 2 Raw Cooling Water (RCW) 3" discharge piping from the Generator Breaker Cooling Water skid directly into one of the main 18" RCW discharge headers located overhead on EL 557" T-8, C-Line in the Turbine building. This configuration frequently forces operators to throttle the 2-TCV-24-40 manual bypass valve flow to provide adequate cooling to the Generator Breaker Cooling Water Skid while not overcooling the Generator Hydrogen Coolers. Since none of the equipment affected by this change is safety-related, it does not involve an unreviewed safety question.

Design Change Notice 50140A

The safety evaluation for DCN 50140A addresses a change which adds Differential Pressure Indicating (DPIs) tubing and connectors to the Reactor Feedwater Pump Turbine (RFPT) oil control subsystem. The DPIs will be redundant and diverse to the existing Differential Pressure Switch (DPS) and their annunciators. This modification provides an alternate method for the operators to determine when the oil filters need to be swapped.

This change does not impact nuclear plant safety, nor does it affect any accident /accident analysis/ equipment important to safety/ margin of safety. Therefore, an unreviewed safety question is not created by this change to the facility.

Engineering Document Change 50147

The safety evaluation for EDC 50147A provides the design documentation to change systems 76 and 84 valves from a solenoid operated gate valve to a solenoid operated globe valve on affected plant drawings, Safety Analysis Report table, system design criteria, and plant procedures. This change will show the affected valves as originally designed, evaluated, constructed, and installed. The plant design and calculations are not altered nor affected. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice 50154

This safety evaluation addresses DCN 50154 which provides the design to cut and cap the vents and drains of the regenerative and non-regenerative reactor water cleanup heat exchangers in Unit 2. Also, this change installs an improved seal design for the channel flange to shell joint on regenerative heat exchangers 2B and 2C. These changes will reduce the unidentified in-leakage to radwaste. The design and required materials meet the existing system design criteria requirements. Equipment operational and safety parameters are not changed. The change does not alter any equipment functions or features important to safety, nor does the change create any new accidents or malfunctions not previously evaluated in the Safety Analysis Report. The evaluation determined that the Technical Specifications are not affected and therefore the margin of safety is not reduced. Therefore, this change does not constitute an unreviewed safety question.

Engineering Document Change 50166A

This safety evaluation is written in support of EDC 50166A which clarifies existing limitations for Direct Current Motor-Operated Valve (DC MOV) testing and operation of FCV-73-44 when a 250V DC Reactor Motor-Operated Valve board is placed on alternate supply. They are supported by existing electrical calculations that verify the adequacy of the battery to perform under design basis conditions provided the limitations for MOV operation are observed. The limitations for operation of the MOVs will not prevent their operation as required to mitigate accident conditions, obtain safe shutdown, or comply with Technical Specifications. Therefore, the activity does not constitute an unreviewed safety question.

Design Change Notice 50180A

The safety evaluation for DCN 50180A addresses the installation of new inlet air filters on each of the Off Gas Dilution Fans. The purpose of this change is to prevent dust and debris from the outside air entering the dilution air duct work from fouling the safety-related dilution air duct work backdraft dampers. Since this change should improve the operation of the dilution air backdraft dampers and improve their ability to mitigate ground-level radioactive releases, the new configuration does not involve an unreviewed safety question.

Design Change Notice 50210A

The safety evaluation for DCN 50210A addresses the routed 2" Main Steam piping and new isolation valves in conjunction with the new actuator for valve 2-PCV-001-0147 (failed closed) and the procedure controls for valves 2-FCV-001-0058 & 0059 (open on Loss of Coolant Accident [LOCA]) and valve 2-PCV-001-0147 (closed following a LOCA). This change will provide reasonable assurance that leakage past the closed Main Steam Isolation Valves (MSIVs) will be directed to the condenser through the preferred pathway. The seismically rugged pathway will help mitigate the effects of an accident LOCA by providing a reliable pathway for any leakage past the closed MSIVs to the condenser.

These design changes do not alter the Technical Specification limits imposed on MSIVs leakage (TS 3.6.1.3), nor do they alter the design or operating characteristics of the system. Current offsite dose and control room dose calculations are based on MSIV assumed leakage rates of 100 scfh for each MSIV or 150 scfh total for all four MSIV lines. This design change does change the failure position for the 2-PCV-001-0147 valve, but this change increases the reliability of the alternate pathway.

This activity has been reviewed against the criteria of 10CFR50.59 and does not constitute an unreviewed safety question.

Engineering Document Change 50247A

The safety evaluation for EDC 50247A addresses changes to the drawing location of the condensate booster pump lube oil and seal water subsystems so that future changes to the piping subsystems will be easier to make. Changing the drawing location of the condensate booster pump lube oil and seal water does not change the ability of the equipment to meet its required functions. There are no safety related functions for any equipment in these subsystems. Therefore, there is no unreviewed safety question with this change.

Engineering Document Change 50282A

The safety evaluation for EDC 50282A addresses the revision to the design pressures for the torus spray line/torus spray headers and the drywell spray headers. A discrepancy between engineering design input (calculations) and design output (drawings) was resolved by revising the design pressure of the subject sections of piping shown on the flow prints to envelope the pressures concluded in the calculations. Testing was satisfactorily conducted to verify the adequacy of the engineering conclusions. No physical changes are being made to the plant and the affected sections of the subject piping will continue to perform their respective safety functions unabated. Therefore, there is no unreviewed safety question.

Design Change Notice 50286

The safety evaluation for DCN 50286 addresses the installation of a vacuum port to the annulus area of the Containment Atmosphere Dilution (CAD) tank B (BFN-0-TNK-084-0636). The change is governed and implemented by applicable design and construction procedures. The installed components are not required to operate in response to any accident scenario and only serve to reduce maintenance time for the CAD tanks by facilitating faster vacuum draw down on the annulus areas. The CAD system will continue to perform its respective safety functions unabated. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice 50310A

The safety evaluation for DCN 50310A addresses an upgrade to various obsolete instruments in the Unit 2 Off-Gas System to ensure proper, reliable, and continuous operations. The modifications implemented by this proposed design change package do not increase the frequency of occurrence of any existing failure modes which could increase the probability of an accident. Implementation of the proposed design change package will not introduce any new failure modes or affect the probability of occurrence of any existing failure modes which could prevent any systems or components from fulfilling their safety related functions. Implementation of the modifications proposed by this design change package does not affect the function or operation of the affected instrument loops but will improve the operability and maintainability of the instrument loops. The modifications implemented by this proposed design change package will not prevent any system or component from performing their intended design basis functions or result in greater off-site doses than previously analyzed. Implementation of the proposed design change package will not alter the ability of affected equipment to perform their intended design basis functions nor will it result in any new or different accident initiators or change the method of initiation of existing accident initiators. Implementation of the proposed design change package will not introduce any new failure modes or affect the probability of occurrence or the severity of any existing failure modes. The equipment affected by the proposed design change package provide no safety related functions other than to maintain pressure boundary integrity. Therefore, it does not involve an unreviewed safety question.

Design Change Notice 50340A

The safety evaluation for DCN 50340A addresses the modification of the drywell sump pump flow measuring instrument loops. This DCN is changing the internal electronics of the transmitters to convert them into "Smart Transmitters". The smart transmitters have many advantages over the existing transmitter electronics, they can filter process, they have built in square root converters, and they are more accurate. The square root converters on panel 9-19 are no longer needed, and they will be removed. The refurbished transmitters will have a different output and lower power supply voltage input. This will require minor changes (resistors and diodes) in the flow totalizer and the power supply. The instrumentation is improved in reliability and accuracy to reduce process noise. This modification will provide operations with more accurate determination of drywell leakage. The Primary Containment isolation equipment is not impacted, the isolation logic is not affected and neither is the piping, valves nor the instrument tubing. The radwaste equipment pathways, release rate and release pathways are not impacted by these modifications, the radwaste system is instead enhanced by eliminating the current malfunction of the equipment allowing for an accurate determination of the drywell sump flow. This change does not impact nuclear plant safety, nor does it affect any accidents/accident analysis/equipment important to safety/margin of safety. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice 50364

The safety evaluation for DCN 50364 addresses the modification of the automatic rod scram timing circuitry to make it more reliable. Since the rod scram timing was implemented as part of the Integrated Computer System (ICS) it has not functioned properly because of the failure to detect the drift signals that the splitter cards were installed to detect and provide the computer (ICS) for calculation of scram times. The existing design utilizes the same drift signals that the Control Rod Drive Selection Relay Panel (9-28) uses from the Rod Position Information System Panel (-27). The signals are split in 9-28 and one goes to the original circuits in panel 9-28 and the other goes to Change of State cards in panel 9-37 to provide time sequence information for the movement of the Control Rods to the ICS.

The new design will utilize the Rod Position Indication System lamp driver change of state for the "6" light to determine when the rod passes through the 46, 36, 26, 16 and 06 positions from full out. The signals generated for the Rod Position Indication in panel 9-27 consist of a "tens" signal from 0 to 4 and a "units" signal from 0 to 9 which allows the display of rod positions from 0 to 48. The "6" light signal will change state each time a rod passes through a position that has a units 6 component. The new design will use these change of states for the "6" light as input in the Change of State cards in panel 9-37. This will meet the requirement in the Technical Specifications that the timing from full out to positions 46, 36, 26 and 06 are less than the values in table 3.1.4-1 and less than 7 seconds from full out to position 06. It is concluded that this modification does not constitute an unreviewed safety question. The conclusion is based on the facts that the modification does not affect the ability to mitigate an accident, does not create the potential a new type of accident nor new type equipment malfunction, it doesn't affect the affected equipment in such a way as to increase the likelihood it will fail, nor does it reduce the Margin of Safety as determined from the Bases of the Technical Specifications.

Design Change Notice 50371A

The safety evaluation for DCN 50371 addresses the design change that allows the removal from service of selected turbine bearing lift pumps to provide rotational drag on the turbine shaft during turning gear operation. The method of providing rotational drag on the turbine generator during turning gear operation is approved by the turbine-generator vendor. The change was reviewed for impact against the Technical Specifications (TS) and the Safety Analysis Report (SAR). The review found several SAR wiring diagrams requiring revision, Figures 8.5-7b, 7f, 8b, and 8f 480-Volt Reactor motor-operated valve Boards 2A, 2B, 3A, and 3B Single Lines. This is due to the fact that the wiring diagram depicts local, remote, and auxiliary control switches. The change does not alter any equipment functions or features important to safety, nor does the change create any new accidents or malfunctions not previously evaluated in the SAR. The evaluation determined that the TS are not affected and the margin of safety is not reduced. Therefore, this change does not involve an unreviewed safety question.

Engineering Document Change 50383A

The safety evaluation for WO 99-011184-000 addresses the removal of the lower layer of reactor cavity shield plugs after the reactor has achieved shutdown and is placed in mode 3, hot shutdown. The reactor cavity shield plugs provide missile protection for primary containment and radiation shielding from the reactor vessel during power operation. Removal of the lower layer of reactor cavity shield plugs at hot shutdown is evaluated to allow early removal of the shield plugs as an outage schedule reduction method. The normal practice at Browns Ferry Nuclear Plant had been to wait until the reactor was in mode 4, cold shutdown, to remove the bottom layer. Waiting until mode 4 ensured missile protection for primary containment which is required to be operable in modes 1, 2, and 3. An evaluation was performed to determine the effects of impact from tornado generated missiles including load drop of a bottom layer reactor cavity shield plug on the drywell head. The analysis concludes that although damaged, the drywell head will not be breached and the containment will remain intact. The reactor building crane and load handling system meets the requirements of NUREG-0612, Phase I. Plant instructions are in place for load handling and personnel are trained and qualified to these procedures for reactor vessel assembly and disassembly and the equipment is designed for the loads and has been tested which makes the probability of a load drop extremely small. A lift height restriction is imposed of approximately 12" above the refuel floor, elevation 664'. This will be reflected in the procedures and on a drawing for vessel disassembly and for heavy load lifts in the critical lift zone.

The design basis accident considered is the Loss of Coolant Accident (LOCA). With the reactor unit shutdown and reactor pressure and temperature decreasing, this accident is not likely. However, this accident is considered because it represents the accident that would result in the greatest radiation dose with the shielding provided by the reactor cavity shield plugs removed. A calculation has been performed postulating a LOCA after the reactor is shutdown. The calculation demonstrates that no fuel damage will occur resulting from a LOCA after shutdown and the dose will not be increased above that normally experienced during refueling outages. This calculation specifies an initial cool down period to provide a margin between the post shutdown fuel peak clad temperature and the worst case fuel peak clad temperature determined for the design basis LOCA. Based on this limitation, the lower layer of reactor cavity shield plugs should not be removed until the unit has achieved shutdown and an initial period no less than 30 minutes.

The special events considered are tornado and tornado generated missiles. Because the reactor cavity shield plugs are lifted by the reactor building crane, removal of the shield plugs will follow the same rules as provided in the procedures for the reactor building crane in the event of a tornado. Even though the drywell head has been shown by evaluation to maintain primary containment, this caution will ensure conservative conditions that maintain primary containment integrity without damage. The paragraph of the UFSAR which describes the function of the reactor cavity shield plugs will be revised to state that missile protection is provided by one layer of shield plugs. From a review of design documents, both layers of reactor cavity shield plugs were designed for identical loading and are equivalent structurally and one layer is sufficient for missile protection.

This safety evaluation also addresses a change to Refueling Laydown Space Drawing 0-47E200-18. This drawing was revised by DCN W40116A for installation of the Auxiliary Decay Heat Removal

(ADHR) system. The ADHR piping was routed through the area used to place vessel components when the vessel was disassembled. The weight of the 65 ton Unit 2 drywell head was incorrectly restated as 45 tons. This change is based on review of previous revisions of the drawing which specifies the weight as 65 tons.

Therefore, this activity does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR), does not create a possibility for an accident of a different type than any evaluated previously in the SAR or reduce a margin of safety for any Technical Specifications. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice 50426A

The safety evaluation for DCN 50426 addresses the replacement of the Diesel Generator A, B and C Battery Exhaust Fans. The replacement fans are of similar design and function to the existing fans. Minor differences in motor horse power and weight have been evaluated, and it has been determined that these have no adverse affect on the operation of any system. The fans are not required to perform any safety related functions and there are no changes in failure modes associated these changes. Therefore, this change does not involve an unreviewed safety question.

Engineering Document Change 50492A

The safety evaluation for EDC 50492A addresses the revised drawing 0-47E866-3 to depict bypass valve 3-BYV-031-1954 as normally open, which is the normal flow path for the system to function in Units 1 and 2, deletes depicted flows to previously deleted loads, and deletes incorrect depicted composite heating water flows which are not required. Since this change only involves changes in the heating water portion of the Control Bay Heating, Ventilation and Air Conditioning and is not required for mitigation of any accidents or events, this change does not involve an unreviewed safety question.

Design Change Notice 50497A

The safety evaluation for DCN 50497 addresses the replacement of an existing standby lighting transformer with one that has the same voltage stabilizing characteristic and very similar component rating parameters. The change was reviewed for impact against the Technical Specifications (TS) and the Safety Analysis Report (SAR). The review found one SAR, Figure 8.5.7a, 480-Volt Reactor motor-operated valve Board 2A Single Line, wiring diagram requiring revision. This is due to the fact that the wiring diagram depicts detailed information on the transformer's input and output voltage ranges. The change does not alter any equipment functions or features important to safety, nor does the change create any new accidents or malfunctions not previously evaluated in the SAR. The evaluation determined that the TS are not affected and the margin of safety is not reduced. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice 50498A

The safety evaluation for DCN 50498 addresses the proposed design change that removes an alternate power source from the Unit 3 Shutdown Boards (SDBDs) and from the Unit 1 and Unit 2 Shutdown Buses by removing the supply to the Bus Tie Board. The proposed design change affects only one segment of the distribution system downstream of the 500- and 161-kV switchyards. Thus, the proposed design change will not affect the capability of the 500- and 161-kV switchyards to provide plant power. Even though an alternate power source is removed from the Unit 3 SDBDs and the Unit 1 and Unit 2 Shutdown Buses, there are other alternate power sources that meet the Design and Licensing Bases and that can be utilized to the SDBDs and their associated safety related electrical loads. The postulated Loss of Offsite Power (LOP) event encompasses any event that could occur associated with the Unit 3 (SDBDs) and the Unit 1 and Unit 2 Shutdown Buses. Therefore, this change does not involve an unreviewed safety question.

Engineering Document Change 50528A

The safety evaluation for 50528 addresses the discrepancy which mandated the actual plant configuration as depicted on drawings 3-47E812-1 and 3-47E610-73-1 which appear in the Safety Analysis Report as Figure 7.4-1B Sheet 2 and Figure 6.4-3, respectively. The revision made to these drawings adds the depiction of thermowell 3-TW-73-42 on the outlet side of the High Pressure Coolant Injection (HPCI) Lube Oil Cooler (cooling water side), as there is currently a thermowell installed in the location specified. The unique identification for this component was made inactive and its depiction on the applicable drawings was removed during Unit 3 restart efforts. After investigation it was determined that a thermowell is installed at this location on both Units 2 and 3 but that a pipe plug is screwed into the opening where the temperature element would normally be. Based on this information, the thermowell will be labeled as such on both Units 2 and 3, and the Unit 3 drawings will be revised to reflect the current configuration. The component is identified as safety related in EMPAC with the safety function being pressure boundary only. There is no active function performed or required by the plug and no changes to the physical plant are being made. Therefore, this change does not involve an unreviewed safety question.

Engineering Document Change 50530A

The safety evaluation for EDC 50530A addresses a fuel assembly design change to incorporate a debris filter-lower tie plate for the General Electric (GE) 13 fuel design manufactured by Global Nuclear Fuel (GNF), formerly GE Nuclear Energy. Foreign debris fretting is currently the leading cause of fuel failures for boiling water reactors. The debris filter grid plate contains a larger number of flow holes but smaller in size compared to the standard tie plate which reduces the possibility of debris getting into the fuel assembly. Final Safety Analysis Report Section 3.2, "Fuel Mechanical Design", is being revised to indicate that some fuel assembly designs may include a debris filter-lower tie plate and to provide a brief description of the debris filter.

The debris filter-lower tie plate is made from a single casting, with no separate parts than can break loose in operation. The outside envelope of the debris filter-lower tie plate is identical to the standard lower tie plate as is the grid structure of the debris filter plate in so far as fuel and water rod seating, rod spacing, and rod capture. The debris filter meets all structural requirements to ensure that it will not fail during operation. The pressure drop across the debris filter may add slightly to the pressure drop across the core, and thus could have a small effect on the thermal-hydraulic characteristics of the fuel at hot operating conditions. These effects, however, will be reflected in the cycle-specific reload licensing analyses performed to establish the cycle-specific operating limits. Operation within the established limits will ensure that actual fuel operation is maintained within the fuel rod nuclear and thermal-mechanical design and safety analysis bases. Potential for flow blockage was also evaluated with the conclusion that significant flow blockage due to buildup of crud or debris is not expected to occur.

There is no increase in the probability or consequences of an accident or malfunction of equipment from that previously evaluated in the Safety Analysis Report (SAR). The proposed activity does not create the possibility of a different type of accident or malfunction from that previously evaluated in the SAR. The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, this change does not involve an unreviewed safety question.

Design Change Notice 50600A

The safety evaluation for DCN 50600 addresses the revision of the NRC Emergency Notification System (ENS) phone system to use a TVA supplied commercial grade telephone line instead of NRC provided leased lines. The telephone system is not an accident or anticipated operational transient initiator. TVA's phone system has proven reliable in daily operation since 1992. This system is available for the same events as the current NRC leased lines. This change does not involve an unreviewed safety question.

Design Change Notice 50605A

The safety evaluation for DCN 50605A addresses the upgrade of the Unit 2 Turbine Building bridge crane main hoist from 180 tons to 210 tons according to industry standards. This upgrade will assure that all anticipated lifts performed by this crane meet industry standards. This change will require revision of section 12.2.3.3 (Cranes) of the Final Safety Analysis Report which indicates that the rating of the main hoists of all the turbine building bridge cranes is 180 tons.

This modification does not constitute an unreviewed safety question because the new configuration only affects the Unit 2 turbine building bridge crane and supporting structure and does not affect the function or operation of any quality or safety related plant systems. Therefore, this modification has been found not to affect the safety of the plant.

JUNE 1, 1999 - APRIL 30, 2001

SUMMARY OF SAFETY EVALUATIONS FOR PROCEDURE REVISIONS

Emergency Operation Instruction Program Manual, Revision 15 (including EOI flowcharts and EOI Appendices)

The safety evaluation for EIOPM addresses the changes to the EOIPM and Emergency Operating Instruction (EOI) flowcharts and Appendices being made in response to the Unit 3 Cycle 10 fuel dependent parameter changes, in response to the new Main Steam Isolation Valve leakage allowances in Technical Specifications, in response to Probabilistic Safety Assessment insights and insights from the Station Black Out (SBO) analysis, and in response to changes in the plant. The changes have been compared with the design bases from the living Safety Analysis Report and with the generic guidance for EOIs and shown to provide the correct technical guidance and do not violate the design bases assumptions. Therefore, this change does not involve an unreviewed safety question.

JUNE 1, 1999 - APRIL 30, 2001

SUMMARY OF
SAFETY EVALUATIONS
FOR
TEMPORARY ALTERATIONS

TACF 0-99-001-205

This safety evaluation addresses TACF 0-99-001-205 which will provide power to office buildings and other support facilities while maintenance is being performed on the 4kv Cooling Tower Switchgear D bus. The alteration will be returned to normal after maintenance is complete on the bus for 4kV Cooling Tower Switchgear D. There is no unreviewed safety question because: (1) There are no Technical Specifications which are affected (2) the probability of occurrence and the consequences of an accident or malfunction of equipment important to safety are not increased, and (3) the possibility of an accident or malfunction of a different type is not created.

TACF 2/3-99-004-073

This safety evaluation for TACF 2-99-004-073 and 3-99-004-073 addresses installation of a pressure monitoring connection on the High Pressure Coolant Injection (HPCI) stop valve FCV-73-18 balance chamber for units 2 and 3. The alteration consist of ¾-inch threaded connection to two ¼ inch instrument isolation valves from the existing balance chamber test connection pressure which is normally plugged.

This TACF will provide pressure monitoring capability at the stop valve balance chamber. Pressure boundary integrity will be maintained at all times when system is operable and/or functionally available. The HPCI system will continue to function as described in the Safety Analysis Report. This will allow continued operability of the HPCI system, and will allow stop valve balance chamber pressure monitoring for troubleshooting as needed. This activity does not constitute an unreviewed safety question.

TACF 2-2000-10-069

The safety evaluation for TACF 2-2000-10-069 addresses the allowance for temporary alteration to the Unit 2 Reactor Water Cleanup (RWCU) System. The temporary alteration removes two manual Appendix J leak test valves (2-TV-069-0633 and 2-TV-069-0634) from the vendor supplied test connection to the valve body of 2-FCV-069-0001 and replaces them with a pipe cap. The TACF does not create any new failure modes or change the function of the RWCU system. Therefore, this change does not involve an unreviewed safety question.

TACF 2-2000-011-001

This safety evaluation addresses TACF 2-2000-011-001 which modifies the circuitry for the solenoid valves associated with the A Steam Line's Inboard Main Steam Isolation Valve (MSIV) 2-FCV-1-14. The circuitry is modified to provide an automatic transfer for its emergency power source (Unit Preferred) upon a loss of Reactor Protection System (RPS) A power or upon a Primary Containment Isolation System (PCIS) A Group 1 trip. This transfer will not be made if PCIS B Group 1 is tripped or if RPS B is not available. The purpose of this temporary alteration is to enhance the reliability of MSIV 2-FCV-1-14 remaining open upon an RPS A power failure or a spurious PCIS A Group 1 trip signal. This temporary change does not involve an unreviewed safety question.

TACF 2-2000-14-001

The safety evaluation for TACF 2-2000-14-001 addresses providing auto-transfer to emergency power for the AC solenoid for 2-FCV-1-26 upon loss of Reactor Protection System (RPS) A. The controls for Main Steam Isolation Valve (MSIV) 2-FCV-1-26 are similar after installation of the alteration. That is, the MSIV will automatically close when Primary Containment Isolation System (PCIS) logic is completed (1-out-of-2 taken twice) or will close when manually demanded in the Control Room. The difference introduced is: upon a PCIS A Group 1 trip signal or loss of RPS A power, an automatic transfer to emergency power is made for the AC solenoid valve for the B steam line's inboard MSIV. If this automatic transfer is made, the MSIV will be conservatively considered inoperable and appropriate Limiting Condition for Operation actions are to be taken. This automatic transfer is inhibited if a PCIS B Group 1 trip signal exists or RPS B power is unavailable. In addition, the manual operation capabilities of MSIV 2-FCV-1-26 are maintained. Therefore, this change does not involve an unreviewed safety question.

TACF 2-00-005-001

This safety evaluation addresses the proposed activity of placing the steam to preheater drain valve, 2-FCV-1-178A, in the closed position until repairs can be implemented. The drain valve is shown on 2-47E801-2 (Final Safety Analysis Report, Figure 11.1-1b) as being in the open position. Therefore, this safety evaluation supports the temporary realignment of the drain valve to the closed position.

Currently, the valve has a significant packing leak which is releasing radioactive gases, steam, and other non-condensable gases into the atmosphere.

The normal drain line flow will be diverted into the Offgas Preheater by this activity. The Offgas Preheater is a total condenser which means that the heat transfer is derived from the condensation of the steam. Therefore, the steam will exit the preheater in the form of the liquid condensate. The original General Electric performance specification for the Offgas Preheater indicates a normal steam consumption rate of 720 lbm. per hour. The additional moisture loading is expected to only be a very small fraction of the current flow rate.

The steam to preheater drain valve, 2-FCV-1-178A, is a non-safety related valve located in the Offgas Recombinder Room in the Turbine Building. The drain valve has no interactions with any safety-related equipment. There is no safety related equipment or actions which are dependent upon the position of this valve. There is no credible accidents or failures modes initiated or mitigated by this valve. The drain valve has no engineering safety functions. There is no protective or automatic logic associated with this drain valve.

Based upon this review, temporarily closing the 2-FCV-1-178A will not adversely affect the Offgas System function or operation. No new radioactive material release paths are created and no new
credible failure modes are created. Therefore, the proposed activity does not involve an unreviewed safety question.

TACF 2-00-004-043

This safety evaluation addresses TACF 2-00-004-043 which controls connecting sampling equipment to Post-Accident Sampling System liquid/gas return to torus line and involves no equipment or systems that have affected safety related functions and is included in no accident scenario. Therefore, this change does not involve an unreviewed safety question.

TACF 2-00-012-074 & 3-00-008-074

This safety evaluation for TACF 2-00-012-074 & 3-00-008-074 was written to ensure that the Residual Heat Removal (RHR) system will operate as designed. In an Appendix R event, the possibility exits that the operating RHR pumps could have their discharge path isolated by a spurious close signal, thereby dead-heading and damaging the RHR pumps. The various modes of operation for the RHR system were examined with the results being that the minimum flow valves for the RHR pumps should be placed in the open position and their respective breakers open to preclude this possibility. This activity can be implemented provided that the Technical Specification required flows for Low-Pressure Coolant Injection (LPCI) can be delivered. The LPCI flows envelope all other required flows for RHR. This verification of deliverable flow will be accomplished by the performance of 0-TI-409. In order to insure the RHR system is available for the Shutdown Cooling, Supplemental Fuel Pool Cooling, and RHR Cross-tie (adjacent unit) modes, the minimum flow valves must be able to close to provide system isolation and the breakers will be closed prior to entering these modes, per the TACF. Since the RHR pumps will perform their design functions for both design basis events and an Appendix R event, nuclear safety is not impacted. This activity does not create an unreviewed safety question.

TACF 2-99-003-085

This safety evaluation for TACF 2-99-003-085 increases the drive temperature trip set point from 250 F to 350 F for temperature recorders 2-TR-85-7A and 2-TR-85-7B. The sensor/trip point is from TE-85-7 (1 through 185) with trip point currently at 250 F. The alarm once annunciated will mask other high temperature alarms from the other Control Rod Drives (CRD) until the alarm is cleared or bypassed. The function of the CRD high temperature alarm is passive. It does not signal any safety related equipment or perform any initiating function. The alarm alerts the operator that a CRD is operating at an abnormally high temperature. Appropriate procedures are in place for responding to the alarm.

Increasing the drive temperature Trip Set Point from 250 F to 350 F for temperature recorders 2-TR-85-7A and 2-TR-85-7B does not involve an unreviewed safety question.

TACF 2-01-01-79

The safety evaluation for TACF 2-01-01-79 addresses disabling the Unit 2 refueling interlocks to allow bridge movement over the core and in-vessel local power range monitor replacement during rod position indication system maintenance. The design basis accidents associated with this activity include accidents which have been evaluated to ensure they could not potentially lead to criticality. These include a rod withdrawal error during refueling and a fuel assembly insertion error with a control rod withdrawn. As stated in the design basis document, refueling interlocks which inhibit fuel and rod movement will be in place under this TACF. The main hoist grapple will be disabled and the mode switch will be administratively controlled in the "Refuel Position". All rod block inputs will remain in place.

This TACF has been evaluated to ensure the design basis accidents and failure modes associated with this activity are not affected. Therefore, this change does not involve an unreviewed safety question.

TACF 1-00-001-032

The safety evaluation for TACF 1-00-001-032 addresses the replacement of existing Unit 1 pre-filter moisture trap with a manual valve (1-THV-32-2710) that can be throttled slightly open so that moisture can be continuously removed. Continuous removal of moisture will reduce Unit 1 control air dryer pre-filter damage and high control air dewpoints downstream of dryer. Moisture trap replacement is necessary due to it's unreliable operation. Also, no change is required to BFN Technical Specification. Therefore, this change does not involve an unreviewed safety question.

TACF 3-99-006-243

This safety evaluation addresses TACF 3-99-006-243 which will disable the trip functions of the Qualitrol Sudden Pressure relay device mounted on the Unit Station Service Transformer 3B (USST 3B). It will also disable the sprinkler actuation that occurs when the sudden pressure relay actuates. Implementation of this temporary alteration does not increase, nor does it introduce, any new unanalyzed failure mechanisms. This temporary alteration is being implemented to increase the reliability of USST 3B to provide offsite power to the 4kV shutdown boards. This activity does not constitute an unreviewed safety question.

Temporary Structure Control Form 3-00-003-RB

The safety evaluation addresses TSCF 3-00-003-RB addresses the implementation of a temporary structure for a Radcon Control Point use during the Unit 3 Cycle 9 outage. This structure is required to be climatically controlled because it will house the computers for Radcon control in the drywell during the outage. The installation of this temporary structure will not cause any type of accident, malfunction, or abnormal operational transients. There will be no increase in probability or consequences of an accident, no increase in the probability or consequences of equipment failure, no reduced margin of safety, and there will be no possibility of an accident of a different type. The proposed installation of this temporary enclosure, therefore, does not constitute an unreviewed safety question.

TACF 3-00-009-231

This safety evaluation for TACF 3-00-009-231 documents an existing temporary condition for breaker 3-BKR-231-0003A/2B. Amptector trip device is to be installed on 225A frame type AK-15 breaker. To enable installation of amptector trip device on this breaker, a spare 600A frame AK-25 breaker with amptector trip device and the same trip characteristics as the AK-15 breaker has been installed in its place. The AK-25 breaker has an additional contact per pole in order to carry the higher rated currents. To accommodate the additional contacts and higher current rating, the AK-25 has wider pole units, thicker upper and lower studs, wider flexible shunts, and wider insulating links on the contact assemblies. The larger frame size breaker does not present an operability concern since the breaker trip device is set to provide the same trip functions whether the breaker is a 600A or 225A frame breaker. This temporary condition will be removed during the Unit 3 Cycle 10 refueling outage. This temporary condition does not alter any equipment functions important to safety, nor does the condition create any new accidents or malfunctions not previously evaluated in the SAR. Therefore, this temporary condition does not constitute an unreviewed safety question.

JUNE 1, 1999 - APRIL 30, 2001

SUMMARY OF
SAFETY EVALUATIONS
FOR
UPDATED FINAL SAFETY
ANALYSIS REPORT
REVISIONS

BFN Final Safety Analysis Report 14.6 and Technical Specification Bases B 3.4.6

This safety evaluation is concerned with the methods used to evaluate the onsite and offsite radiological consequences of various Design Basis Accident (DBAs). The changing of initial assumptions and evaluation methodology has no affect on the actual operation of the plant during normal, abnormal, or accident conditions. This change does not result in any physical changes to plant equipment or any changes to equipment design requirements. The principle plant equipment relied upon to mitigate the radiological effects of a DBA consists of primary containment and isolation, secondary containment and isolation, Standby Gas Treatment, and Control Room Emergency Ventilation. There have been no physical changes to this equipment or the method in which it is operated. The change does not affect the design or function of any components required to operate in an accident scenario. The new control room, exclusion area boundary and low population zone doses are well within the dose guidelines of General Design Criteria 19 and 10CFR100 and conforms to current regulatory guidance. The changes are consistent with the information provided to the NRC regarding the implementation of the Power Uprate project.

The information in Technical Specification B 3.4.6 which is being eliminated is inconsistent with the general format for Standard Technical Specifications. The Bases paragraph above the paragraph to be deleted contains the conclusions regarding the radiological doses. The details of how these calculations are performed are contained in the Updated Final Safety Analysis Report.

This modification has been reviewed against the criteria of 10CFR.50.59 and does not constitute an unreviewed safety question.

BFN Final Safety Analysis Report Section 12.2.5.2, and 12.2.5.2.1

This safety evaluation for FSAR Section 12.2.5.2 and 12.2.5.2.1 addresses leaving Radwaste Building flood protection door 184 in the normally open position. This door provides access from the Service Building to the corridor leading to the Radiochemical Laboratory and the Radwaste Control Room in the Radwaste Building. Because this is a heavy-traffic area, it is desirable to leave this door in the open position.

The conclusion of this elevation is that flood protection is not degraded. Door 184 will remain functional and can be closed as required or as desired. Maintenance instructions are in place to maintain the door's functionality and to ensure door 184 will perform its design function. Abnormal Operation Instructions direct that door 184 be closed under external flooding conditions. From review of 0-AOI-100-3, door 184 will be closed and sealed well before flood levels require the operating units to be placed in cold shutdown.

There are no interfaces with other plant systems used to mitigate the consequences of any design basis accidents, abnormal operational transients, or special events.

The incorrect figure number in the FSAR paragraph 12.2.5.2 is a typographical error occurring when the figures in chapter 1 were renumbered. The figure only displays location of the flood protection doors to the Radwaste Building and does not impact any technical content of the SAR. This change is a non-significant SAR change and does not require a safety assessment or safety evaluation to change.

Therefore, this activity does not increase the probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the SAR. This change does not involve an unreviewed safety question.

BFN Final Safety Analysis Report Section 7.12 and 7.18

This safety evaluation is written in support of FSAR clarifications, enhancements and administrative changes. The changes are clarifications of the wording and descriptions of the systems and subsystems to match the plant. These changes include the following systems: Radiation Monitoring (Subsystems - Air Ejector Main stack, Process Liquid, and Main Steam monitoring) and Main Steam/Backup Control (Control Room abandonment).

Most of the evaluated changes are clarifications and corrections of ambiguous and vague text. Therefore, this change does not involve an unreviewed safety question.

BFN Final Safety Analysis Report Section 7.3 and 7.5

This safety evaluation supports the following changes to FSAR 7.3 and 7.5.

1) Standardizing the term isolation trip limits to Analytical Limit/Trip Setting to agree with the header title in Table 7.3-2.
2) Enter the Analytical Limit or Trip Setting (Setpoint) in place of the Allowable Values listed in Analytical Limit/Trip setting column of Table 7.3-2.
3) Remove reference to quartz fiber insulation cable for the Source Range Monitor and Intermediate Range Monitor detectors.
4) Remove reference to the use of a water seal on the end of the Local Power Range Monitor string which is to be removed from the reactor core.
5) The values given for the high (60 feet per minute) and low (7.5 feet per minute) speed of the drive motor for the traversing Incore Probe.

These changes are clarification of the text by removing details which are not necessary to understand the system operation and function or revising the values in the Analytical Limit/Trip Setting column. The values, which have been evaluated by Design Engineering Calculations, ensure that safe operation and shutdown can be maintained with the instrumentation. These change does not constitute an unreviewed safety question.

BFN Final Safety Analysis Report Section 8.8 and 10.18

This safety evaluation for FSAR Section 8.8 and 10.18 and FSAR Figure 10.18-4 are revised to modify the descriptions for the 48-V DC and ±24-V DC power systems and certain telephone switching node Uninterruptible Power Supply (UPS) systems batteries. The existing power generation design bases, evaluations and descriptions indicate specific time requirements and under-voltage alarm set-points for operation of the batteries, chargers, and communications UPSs. The batteries are not discharged under normal operation and in most cases the chargers are supplied with diesel generator backed power. The affected power systems are not required to mitigate any accident and are not required for safe shutdown. The sound powered telephone, which is required for shutdown from outside the main control room, and the power supplies for the in plant Very High Frequency radio systems, F1 and F4 which are allocated for manual shutdown actions during a fire, are not affected by the changes. Evaluations for the failure of the ±24-V DC, which supply power to the Source Range Monitor, Intermediate Range Monitor, and radiation monitors are not affected by the changes. Therefore, this change does not constitute an unreviewed safety question.

BFN Final Safety Analysis Report Section 9.2

This safety evaluation is written to revise FSAR Section 9.2, Liquid Radwaste Systems. Tables 9.2-1 and 9.2-2 provide details that are not useful and are not necessary in describing liquid radwaste system operation. Details related to radioactive liquid discharges such as the parameters listed in these tables are more accurately described in the Annual Radioactive Effluent Release Report. Therefore, Tables 9.2-1 and 9.2-2 are deleted in this revision, and text references to the tables have been revised as necessary. This evaluation does not constitute an unreviewed safety question.

BFN Final Safety Analysis Report Appendix N.2

This safety evaluation addresses a revision to the BFN FSAR Appendix N.2 which contains the Unit 2 Cycle 11 Supplemental Reload Licensing Report (SRLR). The revision will incorporate a General Electric (GE) letter containing supplemental information for the Loss of Coolant Accident (LOCA) section of the SRLR. This letter provides a corrected value for the GE9B Peak Clad Temperature (PCT) due to a generic LOCA error reported to TVA by GE. The corrected value remains well below the PCT licensing limit and the LOCA licensing basis remains valid.

There is no increase in the probability or consequences of an accident or malfunction of equipment. The proposed activity does not create the possibility of a different type of accident or malfunction from that previously evaluated in the SAR. The proposed activity does not reduce the margin of safety as defined in the basis of any Technical Specification. Therefore, it does not involve an unreviewed safety question.

Safety Analysis Report Change Request to Section 10.8.3

This safety evaluation for SAR Change Request to Section 10.8.3 addresses running the Raw Service Water (RSW) pumps in manual with the RSW Head Tank isolated. The configuration allowed for by this proposed SAR change will result in the RSW pump in manual not being controlled by the water level in the RSW tanks and the RSW pump in manual not being deenergized automatically if a fire pump is operated. In case of a fire pump start during this proposed SAR change the RSW Pump(s) will have to be returned to automatic by Operations. Review of SAR Section 10.8.2, indicates no change to the function of "...supplying all normal plant requirements for raw service water" is caused by performance of this proposed SAR change. SAR Section 10.11 references the Fire Protection Report Volume 1 and review of this report does not indicate any negative impact to the information presented for the High Pressure Fire Protection System (HPFP). The system configuration of this proposed SAR change does not negatively affect RSW/HPFP System components or prevent the HPFP system from performing its design function. Therefore, this change does not involve an unreviewed safety question.

JUNE 1, 1999 - APRIL 30, 2001

SUMMARY OF
OTHER
SAFETY EVALUATIONS

Work Order 00-004513-000

The safety evaluation for WO 00-04513-000 addresses corrective maintenance on the Unit 3 Main Generator Breaker PCB 234. This corrective work is necessary due to the air leak on the "B" Phase Pole. This air leak will eventually result in the breaker control logic locking out the breaker operating capability. This work order will remove the motive sources, air and 250vdc control power from the breaker. The partial loss or complete loss of auxiliary power to the unit boards and the loss of a single recirculation pump or both recirculation motor generator set drive motors has previously been evaluated.

During the performance of this work PCB 234 will remain in the closed configuration. Therefore, this change does not involve an unreviewed safety question.

Work Order 00-007891-000

The safety evaluation for Work Order 00-007891-000 addresses running one or more Raw Service Water (RSW) pumps in manual (control switches in hand) with the RSW Head Tank insolated. The configuration allowed for by this activity will result in the RSW pump in manual not being controlled by the water level in the RSW tanks and the RSW pump in manual not being deenergized automatically if a fire pump is operated. In case of a fire pump start during this activity the RSW Pump(s) will have to be returned to automatic by Operations. This action will be documented in Work Order 00-007891-001. Review of Safety Analysis Report (SAR) Section 10.8.2, indicates no change to the function of "…supplying all normal plant requirements for raw service water" is caused by performance of this activity. SAR Section 10.11 references the Fire Protection Report Volume 1 and review of this report does not indicate any negative impact to the information presented for the High Pressure Fire Protection System (HPFP). Cycling of a RSW head tank isolation valve and a HPFP main header isolation valve will also be performed to determine if piping flow restrictions exist. These valves will be cycled while monitoring RSW/HPFP pressure so that immediate action can be taken (either to reposition the valve or start a RSW Pump) in case a low pressure condition occurs. The temporary system configuration of this activity does not negatively affect RSW/HPFP System components or prevent the HPFP system from performing its design function. Therefore, this change does not involve an unreviewed safety question.

Work Order 00-005623-000

This safety evaluation for WO 00-05623-000 addresses the seal steam regulator valve, 3-PCV-1-147. The valve is indicating a throttled condition which is an indication that high pressure steam is being diverted to the Main Condenser through the seal steam header. Currently, Unit 3 power production is less than optimal. The intent of the proposed activity is to isolate the high pressure steam supply to the seal steam header in order to maintain cycle tightness. The proposed activity consists of closing the Main Steam Supply To Steam Seal valve, 3-FCV-1-146. The isolation of the high pressure steam supply will not adversely effect the Seal Steam System while in full power. The normal system supply is from the High Pressure Turbine seals and various high pressure steam valve packing leak-off lines.

The effected components are non-safety related equipment located in the Moisture Separator Room in the Unit 3 Turbine Building. The equipment impacted by the proposed activity does not initiate any credible Design Basis Accidents or Abnormal Operating Transients. However anytime equipment is operated, a remote potential exists for catastrophic equipment failure. Review of the proposed activity indicates that the worst case scenario would be loss of pressure boundary integrity of the main steam supply to steam seal valve, 3-FCV-1-146. However, this event is boundary by the analysis used to evaluate a Main Steam Line Break and Main Steam isolation Valve closure events.

There are no Technical Specifications or bases associated with the operation of the seal steam system. There is no equipment important safety which depends upon the operation of the seal steam supply valves.

Therefore, it is concluded that the proposed activity does not involve an unreviewed safety question.

2-XA-55 Panel Window 17, Rod 30-27

This safety evaluation addresses the annunicator for Control Rod Drive (CRD) 30-27 on Panel 9-5, 2-XA-55-5A, System 085, CRD Unit Temp High 2-TA-85-7, Window 17. The sensor/trip point is from TE-85-7 (1 through 185) with trip point at 250 F. Inputs from these CRD thermocouples are received by recorders 2-TR-85-7A, -7A/A and 2-TR-85-7B, -7B/A which causes the control room annunciator to alarm. The alarm once annunciated will mask other high temperature alarms from the other CRD's until the alarm is cleared.

Disabling the annunicator window for CRD individual high temperature does not involve an unreviewed safety question. The function of the CRD high temperature alarm is passive. It does not signal any safety related equipment or perform any initiating function. Its only purpose is to notify an operator that a control rod high temperature alarm has come in and to evaluate the event using appropriate alarm response procedures.

Unit 3 Noble Metal Chemical Application

The safety evaluation for Unit 3 Noble Metal Chemical Application (NMCA) addresses the changes associated with the following items:

1. 3-TI-394, "Noble Metal Injection" - This instruction controls the noble metal equipment setup, system connections, valve alignments and valve return-to-normal, system disconnections, and removal of equipment from the plant.

2. Safety Analysis Report (SAR) Section 10.23, "Hydrogen Water Chemistry System (HWC) Change" - This SAR change states that noble metal may be injected into reactor coolant and applied to reactor vessel internals for the purposes of lowering HWC hydrogen injection rates (which lower operating personnel doses) and improving corrosion protection of reactor vessel internals (lower electrochemical corrosion potentials).

3. Unit 1, 2, and 3 Technical Requirements Manual Section 3.4.1, "Coolant Chemistry", and Section 3.4.1, Bases Change - This change adds new reactor coolant conductivity and pH limits for the time period the NMCA and subsequent reactor coolant clean-up are performed.

4. General Electric NobleChem Injection and Analysis Procedures - These procedures control operation of the General Electric NobleChem injection equipment and chemical analysis equipment.

5. Effects of the Noble Metal Chemical Application During and Following the Application and Upon Return to Power Operation with and without HWC In-Service - This item covers the effects of NMCA seen by other plants and which are expected to occur at Browns Ferry.

NMCA helps prevent the initiation of new Inter-granular Stress Corrosion Cracking (IGSCC) and helps mitigate the growth of existing IGSCC, does not degrade or adversely affect the operation of the subjected components, and does not adversely affect post-injection plant operations. The probability and consequences of the abnormal events evaluated in the SAR and the safety-related equipment functional capabilities are not adversely affected by NMCA. The SAR safety analyses remain valid and bounding. NMCA does not create any new failure mode or abnormal sequence of events that can result in an accident or malfunction of a different type than previously evaluated in the SAR. Therefore, NMCA can be applied without affecting plant safety as established in the SAR or the health and safety of the public. Therefore, this change does not involve an unreviewed safety question.

RADIOLOGICAL ENVIRONMENTAL & METEOROLOGICAL INSTRUMENTATION COMMITTEE (REMIC) BFN 2000-02

The safety evaluation for REMIC BFN 2000-02 addresses the change that involves replacement of the current models of wind sensors (Climet 012-16 and Climet 011-4) with an ultrasonic wind sensor (Vaisala model 425AH). This replacement is necessary because the current sensors are no longer manufactured and spare parts are difficult to obtain.

- As part of the replacement, connections to data logging and recording equipment will be changed. Specifically, wind sensors will be directly connected to the Environmental Data Station (EDS) computer and will be disconnected from the wind translators, switch controller, and digital multimeter. Also, wind data will no longer be recorded on a strip-chart recorder in the EDS.

- Also during this replacement, the 91-meter sensor mount will be changed so it is consistent with other levels to facilitate maintenance, while maximizing data recovery from directions unobstructed by the tower structure.

- Finally, changes to the Final Safety Analysis Report will be initiated to incorporate the sensor change, simplify sensor and height descriptions, remove wind speed and direction data from strip-chart recorders, and update references.

The meteorological monitoring system does not include any equipment important to safety and does not change the conditions that are measured. Specifically, the instrumentation is used to access meteorological conditions but cannot influence those conditions. Further, the system is designed to meet the requirements of RG 1.23 and will still continue to meet those requirements after this change.

Therefore, the change does not involve an unreviewed safety question.

Interim Storage of Low Level Radwaste Resins

This safety evaluation is prepared to allow the interim storage of resins at the Low Level Radioactive Waste On-Site Storage Facility (LLRW OSF) in the casks or concrete vaults until the existing LLRW modules are repaired and brought under configuration control through issued design output. The total activity of stored radioactive waste and stored radioactive material shall not exceed the 325 curie limit as described in Final Safety Analysis Report Section 9.3.

Interim storage of resins stored in High-Integrity Containers (HICs), with the HIC stored in casks or vaults, have been determined not to be an unreviewed safety question on basis that it represents normal operation and that the amount of Radwaste exterior to the permanent plant buildings be limited to 325 curies.

Caution Order for 3-FCV-77-2B

This safety evaluation for Caution Order 3-FCV-77-2B evaluates closing 3-FCV-77-2B (Drywell Floor Sump Outboard Isolation Valve) between pump downs of the Drywell Floor Drain Sump. This activity will be performed as necessary to prevent loss of level due to the undesired siphon effect from the Drywell Floor Drain Sump. Since the required functionality of this component is to meet the isolation requirements of Primary Containment Isolation System (PCIS), maintaining the valve in the closed position will meet the PCIS requirements. Maintaining this configuration does not involve an unreviewed safety question.

Caution Order for Raw Service Water

This addresses the Caution Order for running one or more Raw Service Water (RSW) pumps in manual (control switches in hand) with the RSW Head Tank isolation valve(s) closed. The configuration allowed for by this activity will result in the RSW pump in manual not being controlled by the water level in the RSW tanks and the RSW pump in manual not being deenergized automatically if a fire pump is operated. In case of a fire pump start during this activity the RSW Pump(s) will have to be turned off manually by Operations. This action will be documented in the Caution Order. Review of Safety Analysis Report (SAR) Section 10.8.2 indicates no change to the function of "...supplying all normal plant requirements for raw service water" is caused by performance of this activity. SAR Section 10.11 references the Fire Protection Report Volume 1 and review of this report does not indicate any negative impact to the information presented for the High Pressure Fire Protection System (HPFP). The temporary system configuration of this activity does not negatively affect RSW/HPFP System components or prevent the HPFP system from performing its design function. Therefore, this change does not involve an unreviewed safety question.

Unit 3 Technical Requirements Manual Technical Requirement Specification 3.3.9.3 for Hydrogen Water Chemistry Modification

This safety evaluation for Unit 3 TRM change revises the calibration frequency for the Offgas Hydrogen Analyzer from once per operating cycle to once per 92 days. This change is consistent with design characteristics of the new Offgas Hydrogen Monitor equipment and is supported by issued setpoint and scaling calculations. The Offgas Hydrogen Analyzer is not safety-related equipment. It is designed to provide an operational warning of a buildup of Hydrogen in the offgas piping downstream of the recombiners, but it is not credited in mitigation of any accidents or transients, nor with controlling or monitoring the release of radioactive material.

This activity has been reviewed against the criteria of 10CFR50.59 and does not constitute an unreviewed safety question.

Technical Requirements Manual 3.4.3, Structural Integrity

The safety evaluation for TRM 3.4.3, Structural Integrity, addresses the revision made to better delineate Required Actions for the evaluation and disposition of conditions which place the structural integrity of Section XI components in questions according to Reactor Mode. The existing TRM provisions are unclear regarding a differentiation of appropriate actions to take or restrictions based on Reactor Mode. This change to the TRM does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, this revision does not affect or involve plant operation, transient or accident analyses, or malfunctions of equipment.

Regarding accident and malfunction analyses, Code conditions are likewise not assumed to exist as an initiator of accidents or malfunctions. Rather, the American Society Of Mechanical Engineers (ASME) Code and as reinforced by TRM 3.4.3 provides that conditions be evaluated and corrected in a timely manner. The overall application of the ASME section XI program is provided in SPP-9.1 and is unaffected by this TRM change. The revised TRM retains these objectives in a more precise manner according to Reactor Mode. Hence, it is concluded that the TRM revision satisfies the objectives of the ASME Code to evaluate and restore component integrity. The proposed revisions do not constitute an unreviewed safety question.

Technical Requirements Manual 3.5.1.B (U3)

This safety evaluation supports a revision to the TRM for Unit 3 Section 3.5.1.B and TRM Bases 3.5.1.B to increase the completion time to restore the Residual Heal Removal (RHR) cross-connection capability from 10 days to 20 days. This change is being made because the current Allowed Outage Time (AOT) of 10 days does not always provide sufficient time to perform refueling outage maintenance on the RHR loop which provides the cross-connection capability. In these cases, maintenance work could be restricted or an unneeded shutdown be required on an operating unit. Therefore, an increase in completion time from 10 to 20 days is being proposed to allow time to perform required maintenance activities. In the event of the flooding of Elevation 519 in one unit due to rupture of the torus or other cause involving sufficient water to submerge the Emergency Core Cooling System (ECCS) pumps, the cross connection feature can provide core cooling and prevent further fuel damage. If the torus ruptured, the Standby Coolant System is used to flood the system in open loop mode until sufficient hydrostatic head is built up in the basement to provide adequate Net Positive Suction Head (NPSH) to the RHR pumps on the unit cross connection. The events for which the RHR cross connection is designed are outside of the design basis and therefore the feature is not part of the safety design given in Section 4.8.3 of the FSAR. This activity does not require a Technical Specification (TS) change because the RHR tie connection feature is not required by TS. A license amendment is not required because the methodology for determining AOTs is not described in the Final Safety Analysis Review (FSAR) and the increased AOT derived using NRC-approved methodology has a minimal impact on the mitigation of postulated events. This change does not involve an unreviewed safety question.

Technical Requirements Manual Bases Section TR 3.6.5

This safety evaluation supports a revision to TRM Bases Section TR 3.6.5 to clarify the basis for the 542 Standard Cubit Feet Per Hour (scfh) test acceptance criteria and to delete out dated information relative to the radiological doses following a design basis accident. The primary containment will continue to be tested in accordance with 10CRF50 Appendix J to the 2% per day leakage criteria which is the current BFN design basis leakage rate. The continuous makeup monitoring during reactor operation which provides an indication that primary containment integrity is maintained and will be conducted to the same 542 scfh acceptance criteria as currently exists. The activity does not change the physical characteristic of any component associated with the primary containment, secondary containment or the control room habitability zone. The radiological dose calculations and the Updated Final Safety Analysis Report are based on the design basis leakage limit of 2% of primary containment per day. This modification has been reviewed against the criteria of 10CFR50.59 and does not constitute an unreviewed safety question.

BFN Technical Requirement Manual 3.7.1

This safety evaluation was written for TRM 3.7.1 "Liquid Effluents" Bases Change for Unit 1, 2 and 3. The TR 3.7.1 Bases change is a result of a verification that the Condensate Storage Tanks (CSTs) are not considered liquid radwaste tanks as defined in NUREG-0473 (and, therefore, the requirements of TR 3.7.1 and Technical Specification 5.5.8 do not apply to the CSTs).

The site approved CSTs radioactivity sampling program ensures that should all five CSTs be filled to capacity and rupture at the same time, the resulting radioactivity concentration at the nearest portable water supply and the nearest surface water supply in the unrestricted area would be less than the limits of 10CRF20, Appendix B, Table 2, Column 2. The TRM Bases change does not add any new radioactive effluent (liquid or gaseous) release pathways or increase the frequency or consequences of any radioactive releases. Based on this information, this TRM Bases revision does not constitute an unreviewed safety question.

Technical Requirements Manual and Bases Section TR 3.7.6

The safety evaluation supports a new TRM Section 3.7.6 and associated Bases for Units 1, 2, and 3 Electric Board Room Air Conditioning systems for elevations 593' and 621' of the reactor building. The new TRM section enhances the controls and emphasis placed on the operability of the electric board room air conditioning systems.

The activity does not change the physical characteristic of any component inside the electric board rooms or any equipment associated with cooling the electric board rooms. The new TRM section will help ensure that both trains of the electric board room air conditioning systems for each unit are maintained operable and thus the electrical equipment in the rooms would be maintained within its as-designed 10CFR50.49 temperature limits. The air conditioning systems are designed to maintain the room temperature less than 104°F with either train of cooling. The new TRM section does not change the design or functional requirements of the systems.

As demonstrated by a Special Test (STI-00-01), the alternate method of cooling which would require the opening of the doors to the electric board rooms and closing dampers in the air conditioning system does not adversely affect the establishment of the control room habitability zone at a positive pressure. It has been demonstrated via calculation that the electric board rooms can be maintained cool with the alternate method of cooling and that the heat transferred to the control bay does not adversely affect the environment in these areas. Therefore, this change does not involve an unreviewed safety question.

Unit 1, 2, 3 Technical Requirements Manual 3.9.3.1

The safety evaluation for TRM 3.9.3.1 proposes a change for the sampling and analysis frequency of fuel pool water for conductivity and chloride content from “24 hours” to “7 days”. This change also deletes the requirement to sample and analyze fuel pool water for conductivity and chloride content every 8 hours when the fuel pool cleanup system is inoperable.

The purpose for the monitoring requirements for fuel pool water is to maintain the chemistry limits to ensure the integrity of the stainless steel components and the fuel racks for the design system lifetime.

The power generation design basis as listed in the Final Safety Analysis Report Section 10.5 is:

1. The Fuel Pool Cooling System shall minimize corrosion product buildup and control water clarity, so the fuel assemblies can be efficiently handled underwater.
2. The Fuel Pool Cooling and Cleanup system shall minimize fission product concentration in the water which could be released from the pool to the reactor building environment.
3. The Fuel Pool Cooling and Cleanup System shall monitor fuel pool water level and maintain a water level above the fuel sufficient to provide shielding for normal building occupancy.

Fuel Pool Coolant Chemistry is not a part of the safety design basis of the Fuel Pool Cooling and Cleanup system which is to “remove decay heat from the fuel assemblies and maintain fuel pool water within specified temperature limits”.

This change is a revision to the frequency for analysis and does not revise any limit for chemistry monitoring. Therefore, this change has no impact on equipment important to safety and no unreviewed safety question exists.

Technical Requirements Manual Units 1, 2 and 3 3.3.1A, 3.3.2.2A, 3.3.2.3A, and 3.3.3.5A

This safety evaluation addresses administrative changes to TRM 3.3.3.1.A, 3.3.2.2A, 3.3.2.3A, and 3.3.3.5A. The change adds a note to caution TRM users that specific Technical Specification Limiting Condition for Operation (TS LCOs) may be impacted when the TRM actions are entered. No physical changes to the plant are being proposed. There are no resultant changes in the accident analysis, equipment malfunctions, or radiological consequences. Consequently, there are no changes to an acceptance limit which may have been used by the NRC in evaluating the acceptability of the margin of safety provided in the TS, NRC Safety Evaluation Reports, or the Safety Analysis Report.

Technical Requirements Manual/Technical Requirements Bases 3.3.3.1

This safety evaluation for TRM/TRB 3.3.3.1 addresses the Emergency Core Cooling System (ECCS) Keep Fill instrumentation. This system provides the operator with a ready means of monitoring the pressure in the individual ECCS subsystems Residual Heat Removal (RHR) and Core Spray) to verify that they are fully charged in order to prevent water hammer in the event of an automatic initiation. TRM Limiting Condition for Operation (LCO) 3.3.3.1 provides actions in the event that one or more of these pressure indicators becomes inoperable during a mode in which the associated ECCS subsystem is required to be operable. The TRM and Bases Section 3.3.3.1 (ECCS Keep Fill Instrumentation) are being revised to change the completion time for LCO Action A from immediately to four (4) hours. These changes are needed to make the time allowed to complete actions in LCO 3.3.3.1 commensurate with similar actions already in LCO 3.5.4 (Maintenance of Filled Discharge Pipe). This change does not involve an unreviewed safety question.

Technical Specification B 3.1.4

This change to TS B3.1.4 addresses the effect on scram times for control rods whose Control Rod Drive (CRD) is operating at high temperatures. This change will ensure that conservative actions are taken to declare control rods slow when the CRD temperature is greater than 350°F. TS B3.1.4 contains the limiting condition for operation with regard to the number of slow rods allowed. The effects of this change have been evaluated and found not to increase the probability of occurrence or the consequences of an accident or equipment important to safety. Therefore, this change does not constitute an unreviewed safety question.

Technical Specification Bases - Example B3.0.6-1 and SR 3.1.4.4

The safety addresses a revision to improve TS Bases Example B3.0.6-1 for Loss of Safety Function and to add an omitted word to TS SR 3.1.4.4 Bases. The existing Example B3.0.6-1 is incomplete and is a subset of Example B3.0.6-3. Therefore, it is being revised to better demonstrate the example of Loss of Safety Function as discussed in the (a) text description. Also, a missing word is added to the SR 3.1.4.4 Bases for completeness. Therefore, this change does not involve an unreviewed safety question.

Technical Specification 3.8.1.B.4 Actions Bases

This change to TS 3.8.1.B.4 added text to provide a reference to the Configuration Risk Management Program (CRMP) in the Technical Requirement Manual (TRM) for operation with the Diesel Generator (DG) out of service. This is being done to augment a TS change which extends the Allowable Outage Time (AOT) for DG from 7 days to 14 days. As part of the TS approval BFN committed to apply additional risk informed measures for DG outages including a CRMP in the TRM. This Bases change is being made to provide direction to refer to the CRMP circumstances. This change does not involve an unreviewed safety question.

Unit 1, 2, 3 Technical Requirements Manual Section 3.3.9

The safety evaluation for TRM 3.3.9, Offgas Hydrogen Analyzer, addresses the deletion of requirements to "exert best efforts and return the instruments to operable status" for Offgas hydrogen analyzer instruments and associated NRC reporting requirements. The TRM 3.3.9 required actions to install a temporary monitor or take grab samples in the event the Offgas hydrogen analyzer equipment is declared inoperable is not being changed. Also, TRM 3.7.2 requirements for maintaining Offgas hydrogen below 4% remains unchanged. This activity has been reviewed against the criteria of 10CFR50.59 and does not constitute an unreviewed safety question.

Technical Requirement Manual Section 3.7.4 and Bases

The safety evaluation for TRM Section 3.7.4 and Bases addresses the revision proposal that allows for separate conditions to be entered for each system/train instead of each individual snubber as previously performed. This is a conservative decision since the TRM Section 3.7.4, does not distinguish between one or multiple snubbers being inoperable on a component or system. The proposed revision will ensure that required systems will remain operable to perform their safety function and ensure the plant stays within its design bases. Also, the proposed revision adds additional clarification to the TRM and Bases for better understanding by all personnel using them. It also does not affect or change any analysis of the Design Basis Accidents or Abnormal Operational Transients discussed in the Final Safety Analysis Report. Therefore, this change does not involve an unreviewed safety question.

Technical Requirements Manual Changes to Match Technical Specification 403

The safety evaluation addresses revisions made to TR Section 1.4, Frequency, Section 3.0, Limiting Condition for Operation (LCO) Operability, and 5.1.2, TRM Control Program. These revisions are being made to match the analogous changes made to the TS in TS-403. TS-403 was approved by NRC on November 21, 2000, and was recently implemented. These proposed

TRMs will result in the TS and TRM having analogous requirements in the subject common sections. Therefore, this change does not involve an unreviewed safety question.

Special Test Instruction STI-00-01

The safety evaluation for STI-00-01 addresses the change to the Control Room Habitability Zone (CBHZ) boundary to allow for data collection. The test will do this by opening Electric Board Room (EBR) doors. Should the need arise, the doors will be able to be closed to allow for proper maintenance of CBHZ pressure via the Control Room Emergency Ventilation (CREV's) system. This will assure that the dose analysis assumptions remain intact and unchanged, thus placing the CBHZ in its required configuration. No change is required to BFN Technical Specifications. Therefore, this change does not involve an unreviewed safety question.

Special Test Instruction 3-STI-00-02

The safety evaluation for 3-STI-00-02 is written to determine the safety significance of performing the High Pressure Coolant Injection (HPCI) flow test at rated pressure with the 3-FCV-73-36 redundant test return valve in a throttled position. This activity will not place HPCI in a test configuration which has not been previously analyzed in the Safety Analysis Report. The flow path will be the same as the one used to perform quarterly testing and for Condensate Storage Tank (CST) to CST operation per the existing operating instructions. The change which is introduced from this activity is the use of the 3-FCV-73-36 gate valve for throttling to develop initial testing differential pressure. This will be accomplished by removing the seal-in circuit from the 3-FCV-73-36 valve to allow the valve to be throttle in either the open or closed direction. This change does not disable the automatic realignment which would take place in the event of a HPCI initiation. Therefore, this change does not involve an unreviewed safety question.

Technical Instruction 2-TI-402

The safety evaluation for 2-TI-402 addresses the operation of the Post-Accident Sampling System (PASS) for the collection of reactor coolant samples to meet the requirements of Technical Requirement Manual (TRM) 3.4.1 and Technical Specification 3.4.6.1. Operation by this new procedure is not significantly different from the operation of the PASS for training/periodic testing or for post-accident sampling. This alternate sampling point will only be used under plant conditions which are infrequent and of short duration. The small increase in the frequency of PASS operation will not have a significant effect on the expected life of 2-FSV-43-70 or the primary containment isolation valves, 2-FSV-43-40 and -42, or the PASS. Performance of this procedure requires a temporary alteration be in place providing a connection downstream of 2-VTV-043-0173. This temporary alteration will be required for a short duration during periods when neither the normal reactor water cleanup nor the reactor recirculation sample points are available. It is not expected to be in place through the next periodic Final Safety Analysis Report (FSAR) update cycle, therefore, no FSAR change is required. Therefore, this change does not involve an unreviewed safety question.

Unit 2 Noble Metal Chemical Application

The safety evaluation for Unit 2 Noble Metal Chemical Application (NMCA) addresses the changes associated with the following items:

1. 2-TI-419, "Noble Metal Injection" - This instruction controls the noble metal equipment setup, system connections, valve alignments and valve return-to-normal, system disconnections, and removal of equipment from the plant.

2. Unit 1, 2, and 3 Technical Requirements Manual Section 3.4.1, "Coolant Chemistry", and Section 3.4.1, "Bases Change" - This change revises the reactor coolant conductivity and pH limits for the first several weeks of hydrogen injection following NMCA (due to increased conductivity and pH expected because of soluble iron (a non- Inter-granular Stress Corrosion Cracking [IGSCC] contributor).

3. General Electrical NobleChem Injection and Analysis Procedures - These procedures control operation of the General Electrical NobleChem injection equipment and chemical analysis equipment.

4. Effects of the Noble Metal Chemical Application During and Following the Application and Upon Return to Power Operation with and without HWC In-service - This item covers the effects of NMCA seen by other plants (including Browns Ferry Unit 3) and which are expected to occur on Browns Ferry Unit 2.

NMCA helps prevent the initiation of new IGSCC and helps mitigate the growth of existing IGSCC, does not degrade or adversely affect the operation of the subjected components, and does not adversely affect post-injection plant operations. The probability and consequences of the abnormal events evaluated in the Safety Analysis Report (SAR) and the safety-related equipment functional capabilities are not adversely affected by NMCA. The SAR safety analyses remain valid and bounding. NMCA does not create any new failure mode or abnormal sequence of events that can result in an accident or malfunction of a different type than previously evaluated in the SAR. Therefore, NMCA can be applied without affecting plant safety as established in the SAR or the health and safety of the public. Therefore, this change does not involve an unreviewed safety question.