Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

SEP 2 0 2002

10 CFR 50.59(d)(2)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

In the Matter of	)	Docket No.50-390
Tennessee Valley Authority	)	

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - 10 CFR 50.59, CHANGES, TESTS AND EXPERIMENTS SUMMARY REPORT

Pursuant to 10 CFR 50.59(d)(2), this letter provides the Summary Report of the implemented changes, test, and experiments in which evaluations were performed in accordance with 10 CFR 50.59(c). The enclosure provides a summary of the evaluations for the Updated Final Safety Analysis Report Amendment 3 and includes other evaluations performed during the period from March 29, 2001 to August 9, 2002.

No regulatory commitments are included in this report. If you have any questions about this report, please contact me at (423) 365-1824.

Sincerely,

P. L. Pace
Manager, Site Licensing

IE47

SEP 2 0 2002

cc (Enclosure):

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ABBREVATIONS

%	Percent
ABGTS	Auxiliary Building Gas Treatment System
ABI	Auxiliary Building Isolation
ABSCE	Auxiliary Building Secondary Containment Enclosure
ACU	Air Cleanup Units
AFW	Auxiliary Feedwater
ANSI	American Nuclear Society Institute
CCW	Component Cooling Water
CFA	Charger Failure Alarm
CIV	Containment Isolation Valves
CPDS	Condensate Polishing Demineralizer System
CPU	Central Processing Unit
CRDM	Control Rod Drive Mechanism
CVCS	Chemical Volume Control System
DBA	Design Basis Accidents
DBE	Design Basis Event
DCN	Design Change Notice
ECCS	Emergency Core Cooling system
ECI	External Communication Interface
EPRI	Electrical Power Research Institute
EQ	Equipment Qualification
ERCW	Essential Raw Cooling Water
ERFDS	Emergency Response Facilities Data System
ESFAS	Engineered Safety Features Actuation System
FHA	Fuel Handling Area
FHE	Fuel Handling Exhaust
FMEA	Failure Modes & Effects Analysis
FTC	Fuel Transfer Canal
FTS	Fuel Transfer System
HDT	Heater Drain Tank
HVAC	Heating, Ventilation, and Air Conditioning
ICCM	Inadequate Core cooling Monitor
ICS	Integrated Computer System
I/O	Input/Output
LCP	Loop Control Processor
LOCA	Loss-of-Coolant Accident
LONF	Loss of Normal Feedwater
LOOP	Loss of Offsite Power
LPS	Loop Processor Subsystem
MCCB	Molded Case Circuit Breakers
MCR	Main Control Room
MFW	Main Feedwater
MSIV	Main Steam Isolation Valve
NEI	Nuclear Energy Institute
PDP	Power Distribution Panel
PRT	Pressurizer Relief tank
PWR	Pressurized Water Reactor
RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump

ABBREVATIONS

RCS	Reactor Coolant System
RCW	Raw Cooling Water
RHR	Residual Heat Removal
RVLIS	Reactor Vessel Level Instrument System
RWSt	Refueling Water Storage Tank
SEC	Serial to Ethernet Controller
SFP	Spent Fuel Pool
SFPCCS	Spent Fuel Pool Cooling and Cleaning System
SSC	Structure, System, or Component
SSPS	Solid State Protection System
TACF	Temporary Alteration Control Form
TDR	Time Delay Relay
TID	Total Integrated Dose
TPBARS	Tritium Producing Burnable Absorber Rods
TSP	Test Sequence Processor
UFM	Ultrasonic Flow Meter
UFSAR	Updated Final Safety Analysis Report
USST	Unit Station Service Transformers
VCT	Volume Control Tank
WGA	Waste Gas Analyzer
WGDT	Waste Gas Decay Tank
WO	Work Order

SA-SE Number: WBPLEE-99-108-0

Implementation Date: 03/17/2002

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50301-A	Integrated Computer System Phases 4 and 5 Upgrades

Description and Safety Assessments:

This change implements Phases 4 and 5 of the Integrated Computer System (ICS) upgrade plan in 34 stages. Phases 1-3 were previously implemented by DCN M-39911-A. Additional points are added to ICS including remaining Emergency Response Facilities Data System (ERFDS) data not included in previous Design Change Notice (DCN) and upgrading the inadequate core cooling monitor (ICCM)/reactor vessel level instrumentation system (RVLIS) points data link. Eagle 21 External Communication Interface (ECI) is added which includes a Serial to Ethernet Controller (SEC) board located in each of the Eagle 21 multi-bus chassis The SEC is installed in the half of the chassis devoted to the Test Sequence Processor (TSP). The SEC uses the multi-bus for obtaining power only; it will not be able to communicate on the multi-bus. The SEC 'eavesdrops' on the Loop Control Processor (LCP) to TSP data link message by receiving the message in parallel with the TSP, but with no return communications. ICS software is revised to calculate points such as U1118 (Reactor Total Thermal), using digital data whenever available, however analog data will continue to be available as a default.

The Main Control Room (MCR) annunciator printer is replaced with a central processing unit (CPU) and monitor.

The ICS is a non-safety-related system and is isolated from safety-related equipment. Consequently, there are no safety questions associated with it. The additional points and output/display hardware changes have no safety-related functions. The Updated Final Safety Analysis Report (UFSAR) changes are minor text changes with related drawing changes. This modification introduces no increased probability of an accident or malfunction of a different type than any evaluated previously in the UFSAR. This modification introduces no increased radiological consequences of an accident or malfunction of equipment important to safety preciously evaluated in the UFSAR. Quality Related process calculations previously completed by the P2500/ERFDS are now completed by ICS.

This modification will not degrade the ability of any equipment important to safety to perform its intended safety function, thereby reducing any margin of safety in the technical specifications. Therefore there is no reduction in the margin of safety as defined in the basis for any technical specification. No safety questions related to the ICS exists and therefore this activity does not constitute a license amendment.

SA-SE Number: WBPLEE-00-036-0

Implementation Date: 01-23-2002

Document Type:	*Affected Documents:*	*Title:*
Design Change	DCN D-50365-A	Deletion of the Charger Failure Alarms

Description and Safety Assessments:

This design change deletes the Charger Failure Alarm (CFA) on 125V Vital Battery Chargers, and lowers the ammeter relay load settings and increases time delay relay settings associated with the 125V Vital Battery Boards III and IV battery discharge alarm circuits. The reason for deleting the CFA is due to the electronic alarm cards being problematic, unreliable, and obsolete.

Each CFA unit is connected across a blocking diode to indicate when the battery charger output current decreases to a preset value (near zero) or actually reverses. This condition is indicative of a charger failure because the charger is not producing current and the load is being fed by the battery. When this condition occurs, the CFA provides annunciation in the MCR. Also paralleled with this annunciator contact is a battery discharge alarm contact which alarms when the battery discharges for a predetermined length of time. This contact provides essentially the same alarm information as the CFA. Also paralleled with these two contacts are, DC power undervoltage, and charger annunciator contacts for AC power failure, DC breaker trip, and charger overvoltage. This combination provides charger/battery "ABNORMAL" annunciation on the same MCR annunciator assembly window.

The safety function of the chargers is to maintain the vital batteries at the proper charge level. The sole purpose of the CFAs is to provide annunciation on the annunciator system when the battery charger output current decreases to a preset value (near zero) or actually reverses. Each vital battery board has a battery discharge annunciator input that also alarms in the MCR indicating a problem in the battery charging system as a result of a battery charger output current malfunction The annunciator system does not perform any nuclear safety-related function. This design change does not affect the operation or safety function of the chargers. The overall system design is in full conformance with IEEE Standard 308-1971 criteria for Class 1E Systems as stated in UFSAR Section 8.3.2.2.

The proposed change has been evaluated against (1) Coincident Loss of Onsite and External (Offsite) AC Power to the Station - LOOP to the Station, and (2) Loss of External Electrical Load and/or Turbine Trip design basis accident (DBA) that have been previously evaluated in Chapter 15 of the UFSAR. The change was determined to have no adverse effect on the analyzed accidents. This is a non-safety-related system and is not used in the mitigation of any accident. Since this change revised annunciator system alarm circuits only, there will be no design bases accidents involved.

This change will not inhibit the operation or adversely affect the functional requirements of the vital battery chargers. There will be no new credible failure modes introduced by this DCN. Deleting the CFAs does not change the vital battery charger failure modes. The sole purpose of the CFA is to provide MCR annunciation on the non-safety-related annunciator system when the battery charger output current decreases to a preset value (near zero) or actually reverses.

The proposed design change does not increase the probability of an accident or occurrence of a malfunction of equipment important to safety since the design basis requirements of the systems have not been adversely changed by this DCN. The consequences of an accident or a malfunction of equipment will not be increased. No new accidents or malfunctions of a different type than evaluated in the UFSAR are created since the vital battery chargers will continue to function as specified by the design basis documents. The proposed design change does not affect any technical specifications; therefore, no margins of safety are reduced. Based on the above, the proposed design change does not involve a license amendment request.

SA-SE Number: WBPLMN-00-042-2

Implementation Date: 06/15/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50424-A	Auxiliary Building General Supply Fan Rooms Floor and Equipment Drains

Description and Safety Assessments:

This change redesigns the floor and equipment drains for the Unit 1 and 2 Auxiliary Building general supply fan rooms to allow for proper drainage and ensure adequate Auxiliary Building Secondary Containment Enclosure (ABSCE) protection. The existing carbon steel drain piping associated with the Auxiliary Building general ventilation system air intake heating and cooling coils will be replaced with stainless steel tubing. The drain pans for this equipment will also be replaced and fabricated from stainless steel. Floor drains in the Unit 2 fan room area with Unit 1/2 interface cover plates will have the cover plates removed to allow for proper drainage. The condensate drain piping for the Control Building Electrical Boarding Room Air Handling Units will also change to stainless steel tubing.

In addition two floor drain traps are being added in the Unit 1 Pipe Chase. These floor drains are linked to the floor drains in the Unit 1 South Valve Vault which have traps installed. The installation of the new traps in the floor drains of the Unit 1 pipe gallery will ensure the ABSCE safety-related interface points.

There is no effect on the operation or response of any systems or components described in the UFSAR. The proposed changes will not increase the likelihood of any design basis event (DBE) occurring. This will not prevent the plant from achieving safe shutdown in the event of an accident. Since the ability of heating, cooling, and ventilating is unchanged and controlling the building environment is enhanced, radiological releases resulting from a this modification are unchanged. The modification does not create the possibility of a different type of accident than what has been previously evaluated, nor does the change introduce any new initiator or failure. The modification will not cause the system to be operated in a manner different than originally designed The change does not involve or impact technical specification components, therefore, the margin of safety is unaffected. The modification described in the design change for changing out of carbon steel piping to stainless steel tubing, making some of the existing piping larger, and replacing some of the water traps for the removal of condensation from the air handling components, does not constitute a license amendment request.

SA-SE Number: WBPLMN-99-066-0

Implementation Date: 06/07/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50449-A	Condensate Polishing Demineralizer System Manual Isolation Valves

Description and Safety Assessments:

This design change adds six manual isolation valves to the condensate polishing demineralizer system (CPDS) to facilitate maintenance of downstream piping. The CPDS for each power generation unit consists of six mixed-bed demineralizer service vessels; the number of polishers in service varies with system conditions. The system also includes an external regeneration facility, shared between the demineralizer service vessels of the two generating units. The CPDS is not a safety-related system and is not required for the orderly shutdown of the reactor. The Turbine Building housing the CPDS equipment is a nonseismic structure. Piping hangers, and equipment in the CPDS are non-seismic. The system piping is in accordance with American National Standard Institute (ANSI) B31.1.

Adding six new manual isolation valves to the CPDS will not change the function of the condensate demineralizer system as it is described in the UFSAR. There are no design basis accidents or operational transients associated with the proposed modifications in Chapter 15 of the UFSAR. There are no Appendix R components or equipment, or any nuclear safety-related systems or portions of systems affected by the proposed modifications. The condensate demineralizer system does not perform any reactor safety-related function, nor will it compromise the ability of safety-related systems to perform their intended function. Therefore, this modification will not affect any design basis accidents or anticipated operational transients.

This design change will not change the credible failure modes analysis of the CPDS. The credible failure modes associated with the implementation of this DCN are:

- Failure of any of the new valves to change positions when being manually operated, or
- Catastrophic valve failure, such as stem or diaphragm breakage which would allow the diaphragm to become entrained in the system flow.

The possible occurrence of these failure modes is reduced by the selection of the valve materials compatible with existing valves in the system. Some of the selected valves are the same models as valves currently in the system. In addition, valves were selected to accommodate the pressures found in their respective portions of the CPDS. Therefore, there are no new credible failure modes associated with the implementation of this DCN.

This CPDS is required to remove dissolved and suspended impurities from the secondary system This modification adds 6 new manual isolation valves to the system to facilitate maintenance of downstream piping. This modification does not affect the basic design or operation of the CPDS. System design and functional requirements will remain the same for the CPDS. Failures of any of the new components does not contribute to or initiate any of the accident scenarios in the UFSAR; therefore, this modification will not increase the probability of occurrence or the consequences of an accident or a malfunction of a different type from those previously evaluated in the UFSAR. This modification to the CPDS does not reduce the margin of safety of any basis for any technical specification. For these reasons, this activity does not constitute a license amendment request.

SA-SE Number: WBPLEE-00-052-0

Implementation Date: 04/18/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50482-A FSAR Package 1639	Deletion of Radiation Monitors

Description and Safety Assessments:

This change deletes the following radiation monitors from the plant:

1-RE-90-106C, Containment Lower Compartment Monitor Iodine Channel, performs real-time detection of radioiodine in the Containment Building lower compartment. The gas and particulate channels of this monitor are used to provide compliance to Regulatory Guide 1.45, however, the iodine channel does not serve a primary safety function nor any regulatory compliance function.

1-RE-90-129, Condenser Vacuum Pump Exhaust Particulate and Iodine Sample Monitor, monitors, by means of a sampler, the condenser vacuum pump exhaust. It is used to quantify the amount of particulate and iodine effluent after a primary to secondary leak. The Condenser Vacuum Pump Exhaust noble gas monitors, 1-RE-90-119 and 1-RE-90-404, provide real-time detection of the Condenser Vacuum Pump Exhaust. 1-RE-90-119 provides early indication of a primary to secondary leak, as required by Regulatory Guide 1.97. Upon increased noble gas activity, a portable sampler could be put in place to provide means to quantify any particulate or iodine effluent, eliminating the need to maintain a permanent sampler. If there is a steam generator tube rupture, alternate means, such as recent primary and secondary system lab analysis, can be used to conservatively estimate particulate and iodine releases through the Condenser Vacuum Exhaust. These measures eliminate the need to maintain a permanent sampler. It should be noted that Regulatory Guide 1.97 requires monitoring of noble gases only. It does not require the continuous monitoring of particulates and iodine.

0-RE-90-132C, Service Building Vent Monitor Iodine Channel, performs real-time detection of radioiodine in the Service Building Vent effluent discharge. There is not a requirement for real time detection of iodine in the Service Building Vent Monitor.

1-RE-90-275 and 1-RE-90-276, Reactor Coolant Drain Tank (RCDT) Monitors, continuously provide real-time measurement of the gross gamma radioactivity of the water in the RCDT pump discharge. A high radiation signal from either of these monitors automatically generates a signal to initiate the isolation of the RCDT pump discharge. This isolation signal is not a primary safety function. The isolation is provided by a Containment Isolation Valve (CIV). This valve would have already closed from a Phase A Containment Isolation signal. The signal from these monitors serve no purpose in any credible scenario. The most likely result of a signal from the monitor would be a spurious alarm which would result in the isolation of the CIV.

1-RE-90-280, area radiation monitor, monitors ambient radiation in the Post Accident Sampling Room. Any post accident mission to the Post Accident Sampling Facility will have site RADCON support.

1-RE-90-290, 1-RE-90-291, 1-RE-90-292 and 1-RE-90-293, Residual Heat Removal (RHR) Post-Accident Monitors, provide real-time measurement of the gross gamma radioactivity of the post-accident recirculated sump water of the RHR line. The activity detected by these monitors would be dependent on the source of the supply to the pumps (refueling water storage tank (RWST), containment sump, reactor coolant system (RCS)) and the nature of the DBE that has occurred. Information provided is not used in diagnosing or mitigating an accident. By the time these monitors see the elevated activity in the water from the sump, there would have already been an accident signal. Any subsequent indication, by the RHR monitors, of high radiation from the containment sump would be repetitive information that would add little value to post-accident recovery.

These monitors are not required to mitigate any DBA. Although their use in an accident could be helpful to the operators, their failure as a result of an accident is acceptable. Therefore, their deletion is acceptable and does not constitute a license amendment request.

SA-SE Number: WBPLEE-00-025-0

Implementation Date: 03/22/2002

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50499-A	Replace Pressure Regulator with a Pneumatic Loading Station.

Description and Safety Assessments:

This change replaces pressure regulator, 1-PREG-6-106BA, with a pneumatic loading station. This pressure regulator supplies a constant pressure signal to the No. 3 Heater Drain Tank (HDT) level control valve. The constant pressure signal is connected to the positioner of the level control valve (by operation of solenoid valve, 1-LSV-6-106B) when the following conditions are present: 1) Turbine Power is ≥ 85 percent (%), and 2) a failure (trip) of any one of the three operating No. 3 HDT pumps Upon application of the constant pressure signal, the level control valve stem moves to a predetermined position in order to provide sufficient flow resistance to prevent run-out condition of the two remaining pumps. The use of a manual loading station provides two improvements: 1) improves accuracy and repeatability of the constant pressure signal, and 2) allows manual control of the level control valve following a tripped pump event. Additionally, the constant pressure signal value will be determined and shown on design output documents. Currently, the signal value corresponds to a 30% open valve stem position This value is too small.

UFSAR Chapter 15 accident analysis identifies two Condition II events associated with the feedwater system 1) Loss of Normal Feedwater (LONF), and 2) Excessive Heat Removal Due to Feedwater System Malfunction. The loss of normal feedwater event results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. The protective feature for this event is a reactor trip on low-low steam generator water level. The accident analysis assumes a complete loss of feedwater and is due to a loss of offsite AC power (bounding condition). The excessive heat removal due to feedwater system malfunction event results in excessive heat removal from the primary coolant system and accompanied by an increase in reactor core power (positive reactivity). The protective feature for this event is a feedwater isolation on high-high steam generator water level. This accident analysis assumes the full opening of one (or more) feedwater regulating valves due to equipment malfunction or operator error.

This change does not create any additional equipment failure modes that affect condensate/feedwater supply. The Heater, Drains and Vents System is not credited with mitigating any UFSAR Chapter 15 events. The change in the type of pneumatic component that is used to control the No.3 HDT level control valve does not create any additional failure mechanisms. The only failure mechanism related to this change is a loss of control air supply. This failure mechanism is common to the existing configuration. Therefore, this change does not add any additional failure modes.

This change does not affect any UFSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. The technical specification is not affected. This change is in compliance with safety requirements as specified in design basis documents. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no license amendment is required.

SA-SE Number: WBPLMN-00-060-0

Implementation Date: 04/03/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50551-A UFSAR Package 1648	Replacement of Alum Sludge Pond Sump Pumps

Description and Safety Assessments:

This design change removes the Alum Sludge Settlıng Pond sump pumps and replaces the sump pumps with a 4 inch stainless steel drain line with an isolation valve from the Alum Sludge Settling Pond Sump to the Low Volume Waste Holdıng Pond These sump pumps are not functioning as a result of age and usage. In addıtion, these pumps are obsolete and parts are not available. Hence, temporary pumps are currently being used as a replacement but are not economical. This change will remove these pumps and install a stainless steel gravity flow drain line from the sump to the low volume waste holding pond. A manual isolation valve will be installed to prevent back flow from the waste holding pond to the sludge settling pond. The sump would be isolated from the waste holding pond if the pond level was high enough to flow back to the sump. Due to the anticipated frequency of drainage and the fact that this line will be installed underground below the freeze line, no freeze protection or pipe supports will be required. The temporary pumps and associated equipment are no longer required after the gravity flow line is installed. Prior to exiting the sump enclosure, the two existing discharge lines from the sump pumps will have blind flanges installed. Electrical power for sump pumps/motors will be removed by several methods such as removing cables/conduits, terminating wires, removing fuses, sparing breakers, etc.

The sump pumps and associated equipment are not safety-related. UFSAR Chapter 15 accidents were reviewed, and no DBA or anticipated operational transients were determined to potentially be affected by this design change. Therefore, the proposed modıfications will not directly or indirectly impact the nuclear safety margins for operation or shutdown of the plant. This will not prevent the plant from achieving safe shutdown in the event of an accident. The modification does not create the possibility of a different type of accident than what has been previously evaluated, nor does the change introduce any new initiator or failure. The change will not increase the off-site does rates to the public, as analyzed in the UFSAR Chapter 15 nor will it increase the radiological consequences of accidents analyzed previously.

There are no credıble failure modes associated wıth the proposed modifications. As a result of removing the sump pumps and installing a gravity flow drain lıne, UFSAR Section 9.3.3 will be revised. These changes are acceptable since the equipment and components being changed are not required for the operation or safe shutdown of the plant.

These changes have been reviewed in accordance the NUREG-0498, Supplement No. 1, April 1995, "Final Environmental Statement," Section 3.4 "Chemical, Sanitary, and Other Waste Treatment" and SPP-5.05, RO, "Radiological and Chemistry Control." These changes do not impact the Environmental Impact Statement since the supernatants are still being processed to the same location.

These changes do not reduce the margin of safety identified in the applıcable technical specifications. These changes do not prevent any component from performing its function as described in the technical specifications. Therefore this change does not require a license amendment request.

SA-SE Number: WBPLMN-01-017-0

Implementation Date: *7/27/2001*

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50702-A	Waste Gas Analyzer Modification

Description and Safety Assessments:

The Waste Gas Analyzer (WGA) sequences through various samples for Units 1 and 2. The analyzer system consists of an oxygen (O_2) analyzer and a multipoint sampling header. The O_2 analyzer (0-102N-043-0232) is located in the hot sample room used to determine the volume percent of O_2 in various tanks to prevent an explosive gas mixture. This O_2 analyzer is a technical specification requirement and is required for all modes of operation. The following sample points are taken by the sequential O_2 analyzer: RCDT, Volume Control Tanks (VCT), Spent Resin Storage Tank, Pressurizer Relief Tanks (PRT), Waste Gas Decay Tanks (WGDT), and chemical volume control system (CVCS) Holdup Tanks.

A radioactive gas leak was found in sequential WGA Room, and the source was determined to be the second compressor. Since this compressor has failed several times, a trouble shooting work order was initiated to determine the cause of the failure and to obtain additional data using only one compressor (flow rates and pressure). Revise the applicable design documents to have only one compressor, and provide over pressure protection and adequate moisture drainage.

This design change addresses the issues with the Waste Gas Compressors 0-PMP-43-450A and 0-PMP-43-450B operation in conjunction with the oxygen analyzer. The following changes will insure proper operation of the compressors. This design change replaces the compressors and the flow controller meter, and adds a coalesce filter, a moisture trap, a pressure relief valve, a pressure indicator on the inlet and outlet of the compressor, and a pressure alarm on the inlet of the compressor.

The potential adverse failure mode of the new coalesce filter has been evaluated, and does not affect the design function of the sequential WGA. These changes do not create any equipment failure modes that would cause the Waste Gas Disposal System to be unable to perform its function of processing radioactive gases. The changes described above are still within the design requirements of the system. The existing equipment should operate as originally designed. These changes do not affect failure modes of equipment that are important to safety, therefore, no equipment failure modes are created that are not already covered by the UFSAR failure analysis. If a radioactive release from the sequential WGA equipment were to occur, radiation level would be detected by the Auxiliary Building Vent Radiation Monitor (0-RE-90-101). The likelihood of this failure is considered negligible. Any releases would be bounded by the WGDT safety analysis. This DCN does not add any additional or different types of failure modes that have not been addressed in the UFSAR. Therefore this change does not require a license amendment request.

SA-SE Number: WBPLMN-01-005-0

Implementation Date: 04/03/2002

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50821-A	Add Flush Connections to the ERCW System and Telltale Drain Clarification.

Description and Safety Assessments:

This evaluation addresses two system design changes:

(1) Adds essential raw cooling water (ERCW) flush connections to the supply piping to the Safety Injection Pump, Containment Spray Pump, and RHR Pump Room Coolers for Trains A and B. The ERCW flush connections will be used to facilitate clearing the supply lines to the subject coolers if blockages in the piping occur.

(2) Provides guidance for removal of the caps on the telltale drains that are located between the ERCW isolation valves in the suction to the Auxiliary Feedwater (AFW) pumps. The telltale drains and their associated caps are already installed in the subject piping. This design change is providing clarification concerning the operation of the telltale drain lines for maintenance and testing purpose.

The changes have no influence on the probability of accidents previously evaluated in the UFSAR. The guidance provided in the design output released by this design change assures that the equipment involved in the changes will be capable of performing their design basis functions when called upon to operate, as controlled by the technical specifications and plant procedures. This added guidance assures that the probability of occurrence of a malfunction of the ERCW system, systems supported by the ERCW system, or the AFW System previously evaluated in the UFSAR is not increased. Since guidance is included in the DCN to assure the ERCW and AFW systems will be capable of performing their design basis functions within the confines of the technical specifications requirements and plant procedures, the radiological consequences of currently evaluated accidents and equipment malfunctions will not be increased. The changes implemented by this DCN are not of the type that could create a different accident type. The possibility of a different malfunction is not created during the use of the ERCW flush connections added by the DCN or during the removal of the telltale drain caps during maintenance or testing. The existing flooding and water spray design basis bounds the use of these features. No design or licensing basis safety margins are impacted by the changes of this DCN. Therefore, it is concluded that these changes do not constitute a license amendment request.

SA-SE Number: WBPLEE-01-015-0

Implementation Date: 06/14/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN No. 50898-A FSAR Package 1674	Code Call All-Clear Alarm

Description and Safety Assessments:

WBN is responding to a effort to standardize the method that personnel located at Watts Bar, Sequoyah and Browns Ferry will be informed of the "all clear" condition following any evacuation, fire and medical emergencies. Presently, Operations in the Main and Auxiliary control rooms can either initiate interface with the Code Call, Alarms, and Paging System. This DCN deletes the "all clear" feature from the Code Call, Alarms, and Paging System Operations will use the paging system to make announcements that the "all clear" condition exists.

This change removes the "all clear" alarm tones which are described in the WBN UFSAR. Additionally, this DCN will revise System Description, "Code Call, Alarms, and Paging System" to reflect the removal of the "all clear" alarm.

The WBN Code Call, Alarms, and Paging System performance is unaltered by this change. It will maintain its intended functional capabilities. The components, structures and systems involved in this modification are not safety-related and are not required to support the operation of any safety or quality related components. Therefore, should a failure occur, there will be no impact on the safety of the plant, and from a Nuclear Safety standpoint this modification is acceptable.

SA-SE Number: WBPLMN-01-031-0

Implementation Date: 04/03/2002

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50948-A	Residual Heat Removal Valves Re-geared.

Description and Safety Assessments:

The following is a description of the changes that were evaluated:

Re-gear valve operators for valves 1-FCV-74-12 and -24. Re-gearing these valves will increase thrust margin (operator capacity that exceeds margin), thereby allowing longer intervals between required tests. Re-gearing will increase stroke time for these valves by approximately 3 seconds each.

Disconnect torque switches on valve operators for 1-FCV-74-12 and 1-FCV-74-24. These torque switches are currently installed in parallel with the limit switches (i.e., both switches are required to stop motor). Current settings for the torque switches are at the lower portion of their range. Thrust accuracy of the torque switches is ± 20 percent in this configuration, which has caused thrust to exceed design allowables (valves were not damaged). With torque switches disabled within the circuit, the limit switch will be relied upon to terminate the valve closing stroke, decreasing the likelihood that the valve will exceed the total allowable thrust. The torque switches in valve operators for 1-FCV-74-12 and 1-FCV-74-24 are inaccurate (±20%) at the lower portion of their range (where this valve operates), and therefore cannot be relied on to control the valve motor such that sufficient thrust is produced to close the valve, nor can the torque switch ensure that thrust in excess of the design thrust will not result because of these inaccuracies. Removal of the torque switch from the circuit decreases the likelihood that the valves will exceed the allowable thrust by removing the relatively inaccurate torque switch. Reliance on the limit switch to terminate closing stroke is acceptable because the limit switch itself is a positive displacement mechanical device and highly reliable. The post modification testing (as well as required periodic inspection testing) performed on the motor-operated valves ensures that the limit switch is set appropriately to achieve the required valve stroke time and proper torque between the valve plug and seat. In addition, the evaluation established that the proposed removal of the torque switches is acceptable because increased drift of the limit switch setpoint is not likely.

Valves 1-FCV-74-12 and 1-FCV-74-24 are sub-components of the RHR portion of the Emergency Core Cooling System (ECCS). Numerous accident evaluations within the UFSAR take credit for the ECCS. Re-gearing these valves will add additional margin (operator capacity that exceeds requirements) to the available thrust for these valves, but will increase stroke time from 12 to 15 seconds. For a long-term loss-of-coolant analysis (LOCA) Mass and Energy Release and containment peak pressure accident evaluation, this will mean a reduction in flow from the RHR pumps for 30 seconds instead of 24 (includes closed to full open and back to closed). This is acceptable because reduction in flow occurs during the accumulator injection period (approximately 56 seconds). During this time (56 seconds) the accumulators are capable of condensing available steam in the RCS loops, so a small reduction in RHR flow prior to then is insignificant. Short term LOCA Mass and Energy Release and Sub-Compartment Pressurization, Peak Cladding Temperature, Main Steamline Break Inside Containment, Best Estimate LOCA analysis, SBLOCA Analysis, non-LOCA Analysis, LOCA Forces, and Steam Generator Tube Rupture analysis are not affected by this design change. Therefore, no license amendment request is required.

SA-SE Number: WBPLEE-01-068-0

Implementation Date: 03/01/2002

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-51073-A FSAR Package 1698	Replace Controls for RCW Strainer Bypass Flow Control Valve

Description and Safety Assessments:

This change, replaces the control loop associated with the Raw Cooling Water (RCW) Strainer Bypass flow control valve, 1-FCV-24-191. The existing flow control allows the valve to provide automatic control action to maintain a constant bypass flow or allows the operator to manually position the valve to a desired flow value. An electronic controller (automatic mode) is used to modulate the bypass valve to maintain the desired bypass flow rate (i.e., setpoint control). RCW bypass flow is monitored by flow transmitter, 1-FT-24-191, which provides an input signal to the electronic controller. This change replaces this electronic controller with a manual loading station pneumatic controller. The manual loading station will allow Operations personnel to position the bypass valve to achieve a desired flow rate. This change will also remove the bypass flow loop, 1-LPF-24-191, from service.

The RCW system serves as a heat sink for cooling non-safety-related loads. The RCW system performs no safety-related function and shall therefore, not be required to operate during or after any DBEs. The design functions of the RCW Strainer Bypass flow control valve, 1-FCV 24-191, are to provide additional makeup to the CCW and to regulate RCW header pressure. This is not a safety-related functional requirement.

UFSAR Section 10.4.5.2 discusses the CCW system. Specifically, the following statement is given, "Normal water level is maintained by automatic operation of the RCW bypass strainer, and low level is alarmed in the main control room " As discussed above, this change removes the automatic control feature from the RCW Strainer Bypass Flow Control Valve. Thus, UFSAR Change Package deletes the term "automatic" from this statement. No further description of the CCW or RCW system related to this change is needed.

The RCW and CCW systems are non-safety-related. These systems are not used to detect or mitigate any DBEs. This change does not affect accident mitigation, radiological consequences, fission product barriers, or evaluation methodologies. Also, this change does not increase the likelihood of malfunction or the possibility of a new or different malfunction. Therefore, this change does not constitute a license amendment request.

SA-SE Number: WBPLMN-01-084-0

Implementation Date: 03/12/2002

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-51102-A FSAR Package 1702	Eliminate Pressurizer Backup Heaters of High Level Signal

Description and Safety Assessments:

This evaluation evaluates the effect of revising the control system logic to eliminate the actuation of the pressurizer backup heaters on a high level deviation signal. At present, a bistable is actuated on a high level deviation; the output of this bistable is used to both actuate an alarm on the control board and to actuate the backup heaters. The functional change is to eliminate the backup heater actuation on the bistable actuation, but to maintain the control board annunciator actuation.

The pressurizer level control system is used to control the RCS water inventory whenever a steam bubble is present in the pressurizer. This system also provides indications, alarms and manual controls for operator monitoring and control The pressurizer level control system utilizes three level channels to generate signals for pressurizer level control, anticipatory alarms, indication, and recording. The level channel inputs are isolated outputs from the reactor protection system. The three level channels provide individual indication on the MCR board.

During power operation, the pressurizer level setpoint is varied as a linear function of T_{avg} as previously described The difference between the level signal selected for control and the reference level is compensated to eliminate any steady-state error between indicated and its setpoint The output signal is the automatic charging flow demand signal. The uncompensated level error signal feeds two bistables. One bistable is used to provide a low level deviation alarm. This alarm is set below the programmed level by a preset amount to warn the operator that the charging system is not supplying enough charging flow to maintain pressurizer level. The second bistable actuates a high level deviation alarm if the measured level increases above the programmed level by a preset amount This bistable also energizes the pressurizer backup heaters.

The LONF/LOOP events are currently analyzed for the purpose of showing the AFW has sufficient long-term heat removal capacity. The single failure assumed for the LONF/LOOP event is failure of the turbine driven AFW pump which is the most limiting failure. Since the pressurizer backup heaters and their control circuits do not perform a primary safety function, single failure criteria does not apply. The definition of a single failure is a failure which results in the loss of a safety-related component to perform its intended safety function. Failures in systems not required to mitigate an accident are assumed when the component is in the zone of influence of the accident and is not designed and specified to remain functional in the accident environment. Since there is no zone of influence for this transient, an unintended spurious operation of the backup heaters is not required to be assumed.

UFSAR Section 7.7 documents that the pressurizer water level control system is not required for safety. Analysis has shown that the plant design transients are not impacted by the change, and the safety analyses are not adversely impacted In addition, implementation of the logic change does not make any changes to the protection grade isolated inputs into the pressurizer water level control system. Therefore, based on the above, the logic modification to eliminate the automatic backup heater actuation on a high pressurizer water level deviation, and maintain manual heater actuation capability, will not adversely affect the UFSAR described design functions of any structure, system, or component (SSC). Therefore, the control system logic modification to the automatic backup heater actuation on high pressurizer water level does not require a license amendment.

SA-SE Number: WBPLEE-97-121-0

Implementation Date: 03/21/2002

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39399-A FSAR Package 1489	Elimination of Condensate Booster Pump Suction Isolation Valve

Description and Safety Assessments:

This design change eliminates the automatic opening of the condensate booster pumps' suction isolation valve when the pressure in the main feedwater (MFW) pump suction header decreases to less than 130 psi above the Number 2 heater shell side pressure. Elimination of the suction isolation valve automatic opening feature also eliminates an automatic start of the condensate booster pumps because the pumps receive a start signal when their associated suction valve becomes fully open. The purpose of the design change is to remove the automatic start feature of the condensate booster pumps as described above in order to eliminate unnecessary starting/stopping of the pumps caused by expected process transients. Start up of a MFW pump or initiation of feedwater isolation can cause an expected drop in MFW suction header pressure, which can cause an unneeded start of the condensate booster pumps. An existing MCR annunciator alarm to alert the operator of decreasing differential pressure between the MFW pump suction header pressure and the Number 2 heater shell side pressure will remain intact.

The normal functions of the condensate system are to collect the condensed exhaust from the low pressure turbines in the hotwell section of the condenser, and pump it through feedwater heaters to the suction side of the feedwater pumps. The condensate booster pumps raise the condensate pressure to predetermined values required by the feedwater system. The condensate system does not serve any safety-related function. It is not required to operate for safe shutdown of the plant following any initiating event.

The condensate booster pumps provide flow and suction pressure for the MFW pumps. Failure of the booster pumps could result in the inability of the MFW pumps to supply sufficient water to the steam generators, and in the worst case, booster pump failure could cause a LONF. This modification removes the automatic start of the condensate booster pumps when the MFW pumps' suction header pressure decreases to a minimum value above the Number 2 heater shell side pressure. The modification removes electrical devices from the condensate booster pumps' suction isolation valve control circuit, and deletes a relay control circuit in its entirety. No failure modes are created as a result of this modification. The modification does not alter the system function or operation, and there is no interface with safety-related systems. There are no new failure pathways to create an unanalyzed accident. Deletion of the automatic pump start feature allows for manual operation of the pumps without adding any additional responsibilities to the operators. This change does not affect the frequency classification of the LONF event (i.e., Condition II: Incidents of Moderate Frequency). The changes have no interface with safety-related power systems, control systems or circuits, and the condensate booster pumps have no interface with a safety-related system. The modification does not increase the probability of occurrence of a malfunction of equipment important to safety. In the event of a LONF, the AFW system and reactor protection system are relied upon to operate properly in order to limit the consequences of the event. Since this modification has no interface with either of these systems, the consequences of a malfunction of equipment important to safety is not increased. The consequences of an accident would be the same regardless of whether the failure was due to operator failure to start a pump or failure of the automatic pump start feature that is being deleted. The net result is that the MFW pump would not be able to supply sufficient water to the steam generators. Any radiological consequences would be the same for either failure condition. These changes do not reduce any margin of safety identified in the technical specifications. These changes do not prevent any safety-related component from performing its function as described in the technical specifications. Therefore, based on the above, a license amendment is not required.

SA-SE Number: WBPLEE-99-063-0

Implementation Date: 05/17/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39851-A FSAR Package 1600	Deletion of Silica Analyzer and Associated Equipment

Description and Safety Assessments:

This design change deletes the condensate polisher outlet silica analyzer instrument loop and the condensate polishers common inlet silica analyzer instrument loop hardware, which include solenoid valves, pressure gauges, alarms, silica analyzers, silica recorder, and silica programmer. These instrument loops are located in the Turbine Building. This requires a change to UFSAR Section 10.4.6 to remove the text discussion for monitoring of dissolved silica content. This change will also remove control air distribution piping and isolation valves and feeding panels.

Silica analyzer instrument loops have a history of unsatisfactory performance and have become unserviceable due to unavailability of replacement parts. The instrumentation was originally installed to provide chemistry with information regarding the silica content of the inlet and outlet to the condense polishers. The silica content is an indicator of when the anion resin is depleted and needs to be replaced. Chemistry no longer needs this function because boric acid is used to prevent corrosion in the steam generators. This acid is a load for the anion resin and is accepted by the anion resin more readily than silica This means that in the process of borating a new resin charge, silica will be removed from the resin early in its life and therefore, is no longer a good indicator of the need to regenerate the anion resin. Silica would start to be removed from the resin as the bed was borated. Furthermore, TVA has committed to incorporate the Electrical Power Research Institute (EPRI) pressurized water reactor (PWR) secondary water chemistry guidelines into the WBN secondary water chemistry program. The EPRI guidelines do not establish any diagnostic requirements or limits for the condensate/feedwater train with respect to silica content. Since there is no requirement for their use, these instrument loops will be deleted.

The condensate polishing demineralizer system is non-safety-related and is not used in the mitigation of any accident. None of the affected instrumentation is used for post accident monitoring or sampling. This system does not interface with any safety-related equipment and its removal will therefore not degrade the performance of any safety-related equipment. There will be no DBA introduced by this change.

Removal of this equipment will improve plant operational efficiency because potential equipment failures are eliminated. The equipment performed no safety function and did not interface with any safety-related equipment in any way in the plant. Therefore, there will be no credible failure modes introduced by this change.

The equipment being removed performs no safety function, and does not interface with any equipment that does. The proposed design change does not increase the probability of an accident or occurrence of a malfunction of equipment important to safety. The consequences of an accident or a malfunction of equipment will not be increased. No new accidents or malfunctions of a different type than evaluated in the UFSAR are created. The proposed design change does not affect any technical specifications; therefore, no margins of safety are reduced. Based on the above amendment, the proposed design change does not involve a need for a license amendment request.

SA-SE Number: WBPLMN-98-078 R0

Implementation Date: 10/11/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN W-39951-A	Replacement of Cast Iron Butterfly Valves in the Raw Cooling Water System.

Description and Safety Assessments:

This change modifies the RCW system by replacing miscellaneous RCW valves that were identified in the single point failure study as being recommended for upgrade with stainless steel valves.

Replacing the existing RCW carbon steel valves with more corrosion resistant, stainless steel valves will not change the function of the RCW system as it is described in the UFSAR. There are no DBA or operational transients associated with the proposed modifications in Chapter 15 of the UFSAR. There are no Appendix R components or equipment, or any nuclear safety-related systems or portions of systems affected by the proposed modifications. The RCW system does not perform any reactor safety-related function, nor will the RCW compromise the ability of safety-related systems to perform their intended functions. Therefore, this modification will not affect any DBA or anticipated operational transients.

The credible failure modes associated with the implementation of this DCN are:

1. Failure of either of the valves to change positions when being manually operated, or
2. Catastrophic valve failure, such as stem or disc breakage which would allow the disc to become entrained in the system flow.

The possible occurrence of these failure modes is reduced by the selection of stainless steel valve materials for use in the raw water service.

Failures of these components do not contribute to or initiate any of the accident scenarios in the UFSAR; therefore, this modification will not increase the probability of occurrence or the consequences of an accident or malfunction previously evaluated in the UFSAR or create the possibility of an accident or a malfunction of a different type from those previously evaluated in the UFSAR. This modification to the RCW system does not reduce the margin of safety for any basis for any technical specification. It increases the RCW system margin of safety by using a more corrosion resistant material for the valves. Therefore, this change does not constitute a license amendment request.

SA-SE Number: WBPLMN-01-013-0

Implementation Date: 08//28/2001

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	ECN E-50789-A FSAR Package 1662 TRM/TS Bases 01-07	Containment Purge Isolation Response Time

Description and Safety Assessments:

Various WBN design and licensing documents are currently inconsistent with regard to the maximum specified response time for containment purge isolation. In particular, two safety analysis calculations that evaluate containment purge, one for offsite dose and one for containment subcompartment differential pressure, each utilized assumptions for purge isolation response time that are less than 6 seconds now established as bounding value. To address these differences in WBN documentation regarding containment vent isolation response time, a plant specific analysis has been issued which documents that the offsite dose and containment subcompartment differential pressure analyses are bounding and conservative with an assumed 6.0 second containment purge isolation response time. The involved UFSAR accident analyses have been evaluated and revised to include the effects on containment subcompartment pressure and offsite radiological consequences of a LOCA during a containment purge. The 6.0 second total purge air valve response time will not adversely impact the results of peak clad temperature calculations during the first seconds immediately following a LOCA because the smaller mass release will not result in a containment pressure reduction at the core. The ability of the purge isolation valves to close will also not be adversely impacted during post-accident conditions. Therefore, the proposed activity will not increase the consequences of an accident previously evaluated in the UFSAR. Implementation of the design and licensing basis document changes addressed by this safety evaluation does not change the event classification of previously analyzed accidents or transients as defined in the UFSAR. The containment isolation function of purge vent isolation is a safety function and is not impacted by the changes addressed in this safety evaluation. Therefore the probability or consequences of a malfunction of equipment important to safety is not impacted. This change does not introduce new or impact existing component or system failure modes. There is no technical specification impacted, nor any new specification or surveillance created by this change. No other events addressed by the UFSAR or other licensing basis documents are impacted by this change. Therefore, the changes do not constitute a license amendment request.

SA-SE Number: WBPLMN-00-044-0

Implementation Date: 08/01/2001

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50612-A DD-00-0031	Diesel Lube Oil System Design Documents Consistency With Plant Configuration

Description and Safety Assessments:

Changes have been performed to the diesel lube oil system flow diagrams and the power systems lube oil system schematic diagrams to show two (2) drain valves located in the lube oil filter/scavenging oil pump strainer housing for each diesel engine. These changes clarify WBN's design bases regarding the plant's physical configuration. The changes have been evaluated for plant operability during the review process and do not affect the physical plant configuration or change the operating parameters of the diesel lube oil system. This change ensures that the design documents are consistent with each other and properly reflect the as-constructed plant configuration. In summation, the changes:

- clarify WBN's design bases regarding the diesel lube oil system and are intended to maintain accuracy and consistency between the UFSAR and other affected design documents with respect to the as-built configuration.
- have been evaluated for plant operability during the review process and do not affect the physical plant configuration or change the operational parameters of the affected systems
- are not expected to adversely affect the NRC's understanding of the design configuration or operation.
- will not alter the frequency class of any accident or event evaluated in the UFSAR to a higher frequency class.
- will not adversely affect the ability of the diesel lube oil system from performing its intended safety function.
- do not increase any challenges to the safety-related diesel lube oil system assumed to function in the accident analysis such that the system performance is degraded below the design basis.
- will not cause any undesirable interactions with other systems important to safety.
- have been evaluated with respect to the accident analysis and will not adversely affect any components that could cause, intensify, or mitigate any DBA event as described in the UFSAR, nor will they introduce any new malfunction pathways.
- will not increase the likelihood of a radiological release or have any adverse radiological impact on the diesel lube oil system as a result of an accident or malfunction of equipment.
- will not impede access to the vital areas of the plant, hamper actions required to mitigate an accident or a malfunction of equipment, or cause an increase in onsite or offsite radiological doses as a result of an accident or a malfunction of equipment.
- have been evaluated against the applicable accidents identified in the UFSAR with respect to the diesel lube oil system and determined not to introduce any new accident scenarios or failure pathways.
- do not increase the probability of any analyzed accident described in the UFSAR Chapters 6 and 15.
- do not involve any new single failures as the Failure Modes & Effect Analysis (FMEA) is not impacted.
- have been reviewed to determine if any margins of safety specified in the bases section of the technical specifications might be reduced and none were identified.

Therefore based on the above evaluation, implementation of the changes will not create the possibility of a new type of accident or equipment malfunction not previously evaluated in the UFSAR. These changes do not introduce any new accident scenarios or failure pathways, do not increase the probability of any analyzed accident, and do not involve any new single failures. The changes will neither increase the probability nor the radiological consequences of an accident or equipment malfunction important to safety previously evaluated in the FSAR due to the revision to the diesel lube oil flow diagrams, and the power system lube oil system schematic diagrams, to ensure that the design documents and the as-built configuration of the plant agree. The post-accident operation of the diesel lube oil system is not impacted. The changes do not infringe on any margin of safety defined in the basis for any technical specifications. Based on the results of this safety evaluation, it is concluded that the subject changes do not require a license amendment and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-00-094-0

Implementation Date: 08/01/2001

Document Type:	*Affected Documents:*	*Title:*
Engineering Document Change	EDC E-50683-A	Auxiliary Building Ventilation System Modification

Description and Safety Assessments:

This design document change defines an alternate configuration to the Auxiliary Building heating, ventilation, and air conditioning (HVAC) system which is to be implemented during any repair or upgrade around fuel transfer canal (FTC) or spent fuel pool (SFP) requiring an isolation of ventilation in these areas to minimize the potential spread of airborne contamination. Also temporary ventilation features shall be provided with HEPA filters as necessary, which would minimize the spread of airborne contamination.

The fuel handling area (FHA) ventilation system, a subsystem of the Auxiliary Building ventilating system, serves the FHA, the penetration rooms and the fuel waste, and cask handling areas. The system is designed to: (1) maintain acceptable environmental conditions for personnel access, operation, inspection, maintenance , and testing, (2) protect mechanical and electrical equipment and controls, and (3) control airborne activity during normal operation. The environmental control system is designed to maintain building temperatures between 60°F minimum and 104°F maximum,

During accident conditions, the FHA ventilation system is shutdown and environmental control is handled by the Auxiliary Building Gas Treatment (ABGTS). ABGTS is a safety-related nuclear filtration system with two 100% redundant Air Cleanup Units (ACUs), designed to reduce radioactive releases from the ABSCE during an accident to levels sufficiently low to keep the site boundary dose rates below the requirements of 10 CFR 100.

The ABGTS fans performance will not be impacted by this alternate configuration since the maximum flow rate of 9900 to ABGTS filter units will not change due to the two open access doors in FHA exhaust duct as the FTC alternate configuration is considered to be the worst case. Therefore, covering the ventilation openings around the FTC and SFP, closing the fire damper in the main duct serving these openings, and opening the access doors in the fuel handling exhaust (FHE) duct and covering holes created by the open access doors with screen, will not impact the ABGTS's ability to draw down and maintain the required negative pressure in the ABSCE.

The DBA and anticipated operational transients have the potential to increase offsite dose beyond 10 CFR 100 limits for ABGTS. The FHE system is non-safety-related, which stops on an Auxiliary Building Isolation (ABI) signal. The ABGTS primarily operates to assure that releases to the environment do not exceed 10 CFR 100 limits as a result of a LOCA, or a fuel handling accident, by maintaining the ABSCE at a negative pressure and processing all exhaust air through the ACU. The changes made by this alternate configuration will not adversely impact the FHE system, or the ABGTS, from performing their design functions, during normal operation, and in the event of an accident, respectively. LOCA is the only DBA which requires ABGTS to be functional.

The changes identified will have no effect on the ability of the FHE system to perform its non safety-related normal functions, which are to maintain the FHA under negative pressure and provide temperature control in the refuel floor spaces, during normal operation. In addition, these changes will have no effect on the ability of ABGTS to perform its safety-related function of maintaining -0.25 inches water guage pressure in ABSCE. The credible failure modes for the ABGTS have been evaluated against the accidents and it is concluded that the ABGTS does not introduce a failure pathway different from those identified and evaluated in the UFSAR. The applicable accidents and the equipment served by the ABGTS have been reviewed against these changes and no new malfunction pathways are introduced which have not previously been evaluated and identified. These changes will not increase the off-site dose rates to the public as analyzed in UFSAR. No change will occur to the radiological consequences of accidents analyzed in the UFSAR. These changes do not reduce the margin of safety identified in the applicable technical specifications. These changes do not prevent any component from performing its function as described in the technical specifications. Based on the results of this safety evaluation, it is concluded that the subject changes do not involve a license amendment.

SA-SE Number: WBPLMN-00-082-0

Implementation Date: 07/30/2001

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50824-A FSAR Package 1686	Increase Total Integrated Dose Values

Description and Safety Assessments:

This change will increase the total 40-year Integrated Dose in the Unit 1 RHR/containment spray heat exchanger rooms. The Total Integrated Dose (TID) was originally based on the sources in the room, Sequoyah Nuclear Plant health physics surveys, and the intended function of the room during normal plant operations. Actual radiation hot spots were evaluated against the radiological conditions assumed in the basis calculations for the TID values shown on the Environmental Data Drawings. A hot spot in the RHR heat exchanger room resulted in a higher dose rate/TID than calculated, calculations and Environmental Data Drawings were revised to reflect the higher dose rate/TID. Equipment Qualification (EQ) Binders were revised to reflect TID. Demonstrated accuracy calculations were revised to reflect the revised dose. UFSAR figure was revised to show the change in the radiation zone for the RHR/containment spray heat exchanger rooms. The equipment in the RHR/containment spray heat exchanger rooms have been evaluated and found not to be adversely affected.

This change will also add a note to Environmental Data Drawing that states "Due to hot particles moving through the system piping, localized hot spots are being created. Notify Engineering prior to relocating or adding equipment on Elevation 692' General Floor Area so a location-specific analysis may be done if necessary." The hot particle path through the system piping has been identified and it has been determined that no EQ equipment is adversely affected. However, this note is being added to prevent future additions or relocating of EQ equipment to Room A1 without an engineering evaluation.

Document changes have been evaluated for plant operability during the review process and found not to affect the physical plant or change the operational configuration of the affected systems. These changes do not modify any existing equipment.

The changes proposed do not affect any system or equipment design parameters. Radiation Zone maps are used to determine the potential dose rates in an area or room. However, actual Health Physics survey data is used to determine the actual posting requirements and worker requirements. The equipment in the RHR/containment spray heat exchanger room has been evaluated and found not to be adversely impacted. This change does not affect the mission dose since there are no post accident missions to the affected areas. Therefore this change does not require NRC approval.

SA-SE Number: EDC E-50894-A

Implementation Date: 03/07/2002

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50894-A FSAR Package 1707	Spent Fuel Pool Cooling Methodology

Description and Safety Assessments:

The reactor core design is being modified to provide for production of tritium through the irradiation of Tritium Producing Burnable Absorber Rods (TPBARS).

EDC E-50894-A supports the documentation changes required to allow higher decay heat loads to be placed in the SFP. The higher decay heat loads are based on off-design values by taking credit for lower CCW system water temperatures and low SFP heat exchanger fouling factors. The existing analysis of record utilizes design basis values and regulatory specified fuel discharge scenarios for predicting maximum SFP temperatures based on limiting decay heat loads in the pool. The design basis maximum temperature for the SFP remains unchanged and is bounding for the higher decay heat loads.

EDC E-50894-A will also specify a Unit 1 Cycle 4 specific maximum allowable decay heat value of 38 MBtu/hr provided that CCW temperature is maintained at 83°F or less which is based on design fouling of the SFP heat exchanger. This value is based on the analysis and methodology being implemented by this change. EDC E-50894-A also provides clarification to existing Note 32 on TVA drawing 1-47W845-5. The clarification allows full opening of normally throttled and locked valve 2-FCV-67-143 during Modes 5 and 6, consistent with the system description.

The CCW and ERCW systems provide cooling for the SFP cooling and cleaning system (SFPCCS) As such, any increase in allowable SFP decay heat will be rejected to the CCS and ERCW. CCS and ERCW analyses have been revised as part of the proposed change. The results of these analyses conclude that the increased decay heat load in the SFP, when rejected to the attendant cooling systems, remains within their design basis heat removal capability.

A License Amendment Request for these changes was previously submitted to the NRC and approved by the NRC in Amendment 37 to the Operating License.

SA-SE Number: WBPLMN-01-025-0

Implementation Date: 10/31/2001

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50952-A	Alternate method for Use of Condenser Dump Valves

Description and Safety Assessments:

This document change provides for the disabling of the P-12 interlock which thereby provides an alternate method of using additional condenser dump valves for unit cooldown. At present, the steam dump logic will block the condenser dump valves when the plant T_{AVG} is reduced below the low-low T_{AVG} interlock (P-12); value of 550°F. A manual interlock bypass switch is provided to permit the use of the designated cooldown valves. These cooldown valves are three out of the total of twelve condenser dump valves. The P-12 interlock will be disabled by lifting wires for the K631 relay contacts associated with the two independent steam dump control circuits. The disablement will be performed procedurally with no permanent hardware modifications to the unit. The use of additional condenser dump valves will be optional for the Operator. The condenser dump valves are controlled using the steam pressure controller before and after the P-12 interlock is disabled. This procedurally controlled temporary alteration will allow the use of additional condenser dump valves below 350°F to aid in the cooldown of the RCS during unit cooldown.

Normal cooldown now uses all twelve condenser dump valves as needed above T_{AVG} temperature of 550°F and the three cooldown valves below this temperature value. The cooldown valves are capable of being operated below 550°F by using the MCR hand switches, which disable the automatic blocking performed by the P-12 interlock for these 3 condenser dump valves.

The effectiveness of the cooldown valves for unit cooldown decreases as RCS temperature and pressure decrease. The proposed method provides for disabling the P-12 interlock and permits the use of additional condenser dump valves during cooldown below 350°F which is the Mode 4 entry temperature.

There is not an increase in the frequency of occurrence of an accident or malfunction of a component important to safety previously evaluated in the UFSAR. Overcooling events involving increased heat removal by the secondary system are analyzed in Chapter 15 of the UFSAR. These include "Excessive Heat Removal Due to Feedwater System Malfunctions" associated with a rapid increase in steam flow, "Excessive Load Increase Incident" associated with a rapid increase in steam flow, "Accidental Depressurization of the Main Steam System" associated with an inadvertent opening of a single condenser dump, relief or safety valve, and "Major Rupture of a Main Steam Line." Since the reactor is shutdown, Mode 4 shutdown margin will be assured by increasing boron concentration (i.e., Mode 4 at 200.1°F concentration) prior to allowing additional condenser dump valves overcooling event involving return to criticality is not credible at this phase of shutdown operation. The RHR system is available as before to prevent uncontrolled heatup of the RCS. The steam generator power operated relief valves are also available as a means of primary cooling via the secondary main steam system. Thus, the probability of an uncontrolled heatup due to failure of a cooling system or component (e.g., steam dumps) is not increased.

The consequences of the accidents and malfunctions of the associated equipment remains unchanged whether the three cooldown valves or the additional steam dump valves are used. The potential to create a new type of event not previously evaluated in the UFSAR is not found with this change. No new accident are created by this change and no new control features have been incorporated. The disablement of the P-12 interlock will be performed procedurally with no permanent hardware modifications to the unit. The use of this cooldown method does not affect the design basis limit of any fission product barriers. The rate of cooldown is maintained within the acceptance limits for RCS cooldown, as specified in the technical specifications (which ensure compliance with 10 CFR 50 Appendix G). The plant cooldown is controlled in the same manner as previously considered which is manual operator action. The method of evaluation for the identified accidents is not modified nor is a new method created. This change does not have an impact on evaluation methodologies described in the UFSAR. Therefore, this change does not require a license amendment.

SA-SE Number: WBPLMN-02-005-0

Implementation Date: 02/14/2002

Document Type:	***Affected Documents:***	***Title:***
UFSAR	FSAR Package 1712	Service Water Process to Drains

Description and Safety Assessments:

This safety evaluation evaluates drainage of raw water into radwaste sumps inside the radiological control area. Currently the UFSAR does not allow drainage of service water in the Auxiliary or Reactor Buildings to a floor drain. This is being evaluated to expedite local leak rate testing.

The liquid radwaste processing system, its associated components, and piping do not perform any accident mitigation function except for CIVs. Processing service water to the containment drains, has no adverse affect on the CIVs. These components do not serve to limit the consequences of a malfunction of equipment important to safety, and this change will not increase the 10 CFR 100 post accident dose limits previously established for the facility or restrict access to vital areas or otherwise impede action to mitigate the consequences of reactor accidents. This change does not affect any equipment required for safe operation or shutdown. In the event of a DBE, safety-related equipment is expected to operate as designed to limit the consequences of the DBE. UFSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunction in liquid radwaste processing system such as pump or valve failures or evaporator failures. The addition of service water (that may contain high conductivity, silt, and debris) to the drains may have long term effects leading to increased maintenance or other corrective measures to prevent degradation of the drain lines and/or processing equipment. However, the likelihood of occurrence of this is not any different than a presumed malfunction of the existing liquid radwaste processing system equipment.

There are no DBAs that could be associated with this activity. This change is not associated with the protective features used to detect and mitigate the effects of any accident. The equipment involved in the UFSAR does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the UFSAR. This change does not change or affect the design basis for any system that is important to safety, and does not cause the frequency level of any accident to change. The equipment potentially affected by this UFSAR change is quality related and Seismic Category IL related and located in the Auxiliary and Reactor Buildings. This change will not cause this system or any system important to safety to fail to fulfill its functional requirements. This change does not change the logic or function of any system that is important to safety. Therefore, this change does not require a license amendment.

SA-SE Number: WBPLMN-01-049-0

Implementation Date: 07/192001

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-01-07-043	Isolation of RCS Manual Leaking Valve

Description and Safety Assessments:

Temporary Alteration Control Form (TACF) 1-01-007-043 will close isolation valve 1-SMV-68-548. This manual isolation valve is normally open while the plant is operating and is part of the line used to sample RCS from Hot Leg Loop No.1. A compression fitting downstream of the manual isolation valve is leaking. The leak could not be completely eliminated by tightening the associated compression fitting during the forced outage. Minor seepage was observed at the fitting when normal pressures and temperatures were attained in the RCS. Experience with compression fittings in RCS sample lines suggests that the leakage rate could increase during the remainder of Cycle 4. Since the fitting is located inside the Crane Wall, access for repair work is not possible in Modes 1 or 2.

The sampling and water quality is not safety-related except for valves associated with containment isolation. This system is not required to operate during or after a DBE and, therefore, does not need to comply with single failure requirements. Sampling of the RCS is used to detect failed fuel. RCS sampling is used to determine gross specific activity and dose equivalent I-131 analyses. The RCS is assumed to be a homogenous solution. Therefore, it is inconsequential whether the sample is pulled from either RCS Hot Leg Loops No. 1 or No. 3. The samples taken from this sample point are UFSAR and Technical Specification required samples, but where the samples are taken is not specified. The ability to sample from RCS Hot Leg Loop No. 3 is not impacted by this temporary alteration, as well as other alternate sample points which have been used by Chemistry in the past. While this temporary alteration does eliminate one sample location, there is no requirement that samples be taken at this location, and Chemistry is still left with the ability to not only sample from RCS Hot Leg Loop No. 3, but has other available alternate sample locations. It is concluded that the temporary elimination of this sample point will not impact the ability to meet the system design function, nor does it impact the ability to satisfy the technical specification surveillance requirements. The change is positive from the standpoint that the risk of continued leakage and possible increased leakage from the RCS is eliminated by closing the isolation valve until such time as a repair can be made. Therefore, this change does not require a license amendment request.

SA-SE Number: TACF 1-01-11-099-0

Implementation Date: 11/06/2001

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-01-11-099 ARI-109-115 WO 01-15469-000	Restoring Power to Loop Processor Subsystem

Description and Safety Assessments:

On October 20, 2001, WBN experienced a failure of the loop processor subsystem (LPS) in Eagle 21 process protection system rack. The symptoms for this failure were consistent with cycling of time delay relay (TDR) K1 in the power distribution panel (PDP) which powers the LCP/Primary Input/Output (I/O) power supply. This evaluation supports Rack 7 LPS and test sequence processor (TSP) subsystem/Secondary I/O power supply to full function

The LPS receives input and calibration digital data, performs 2-point gain and offset calculations, converts input data into electrical and engineering units, performs floating-point calculations, compares conditioned input data to plant protection setpoints, performs diagnostic routines, stores system parameters, provides analog output signals, initiates trip and alarm outputs, and communicates with the TSP.

The Eagle 21 process protection system monitors various plant parameters and provides inputs to the Reactor Trip and engineered safety features actuation system (ESFAS) logic for mitigation of various DBEs. These functions are performed by the LPS independently of the TSP. Altering the feeds to the Rack 7 LPS and TSP with the subsequent change in the Racks 7 and 8 load sequencing does not affect the accident mitigation functions performed by the LPS. The load sequence is not a safety function and does not affect the 120V ac vital board feeder to Racks 7 and 8. Installation of this temporary alteration restores the full functionality of the TSP including the control room annunciation of protection set trouble, protection set Channel II failure, and protection set Channel II bypass alarms. Since the TSP has no accident mitigation functions, failure of the TSP power feed will not impair the capability of the rack to perform required protective functions.

Providing power to the LPS from a different breaker in PDP will allow the LPS to perform its design and accident mitigation functions as specified in the design basis documents and the UFSAR. The alternate breaker is the same size with the only difference being a longer time delay for powering up the system. No protection system setpoints or design functions will be altered and no new failure modes will be created. No technical specifications requirements will be affected.

Eagle 21 is not an initiator of any accident event. The capability of the LPS to perform its safety functions will be restored by this temporary change and thus is an improvement from the present condition. The restoration of monitoring functions provided by the TSP restores Rack 7 to full functionality for monitoring equipment failures. This change does not affect accident mitigation, radiological consequences, fission product barriers, or evaluation methodologies. The changes does not increase the likelihood of malfunction or the possibility of a new and different malfunction of Rack 7. Therefore, this change does not require a license amendment.

SA-SE Number: WBOTSS-01-020-1

Implementation Date: 09/20/2001

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-01-01-030 R1	Control Rod Drive Mechanism (CRDM) Power Circuit

Description and Safety Assessments:

February 1, 2001, work order (WO) 01-001516-000, was initiated to troubleshoot the cause for CRDM Cooler 1B-B tripping off. CRDM Cooler 1B-B Motor 2's power circuit (1-MTR-030-0092/1-B) was identified shorted to ground in or near the motor. During plant operation in Mode 1 or 2, this motor is not accessible for maintenance.

This temporary alteration will allow CRDM Cooler 1B-B Motor 2 to be disabled while CRDM Cooler 1B-B Motor 1 will remain operational. CRDM Cooler 1B-B Motor 2 will be disabled by racking out and removing its control power fuses and to prevent CRDM Cooler 1B-B Fan 2 from running while allowing CRDM Cooler 1B-B Fan 1 to remain available for service, but only in BYPASS mode. This temporary alteration will maintain CRDM Cooler 1B-B in the BYPASS, i.e. supplemental cooling mode, by positioning its associated dampers in the following required alignment:

A. 1-TCO-30-94 OPEN, it's associated handswitches will be maintained in the OPEN position.
B. 1-TCO-30-93, CLOSED, it's associated handswitches will be maintained in the CLOSED position.

In this configuration, annunciator 1-XZ-44-4C, window 102A may alarm when CRDM Cooler 1B-B is operated due to less than design negative pressure at flow switch. CRDM Coolers 1A-A, 1C-A and 1D-B and their associated dampers are not impacted by this TACF.

There are no WBN design basis UFSAR Chapter 15 events for which the CRDM cooling system is required to operate. The CRDM coolers and associated duct/dampers are not safety-related and are not required to perform a primary nuclear safety function. However, the CRDM Coolers, combined with the lower compartment coolers, are required for safe shutdown per 10 CFR 50, Appendix R, (i.e., the Watts Bar Nuclear Plant Fire Protection Report) to keep containment temperatures from exceeding operability (EQ) limits on safe shutdown equipment inside containment. Therefore no license amendment is required.

SA-SE Number: WBPLMN-02-004-1

Implementation Date: 02/04/2002

Document Type:	Affected Documents:	Title:
Temporary Alteration	TACF 1-02-01-62 Rev. 1 SOI-68-02, ARI-95-101 AOI-24	Use of a Ultrasonic Flow Meter on Reactor Coolant Pump seal.

Description and Safety Assessments:

This TACF 1-02-1-62, will install an ultrasonic flow meter (UFM) on the No. 1 seal leakoff piping of reactor coolant pump (RCP) No. 2. The UFM will monitor flow above 6 gpm which is the current leakoff limit allowed by the Westinghouse vendor manual and the system description. The MCR indicators have a maximum reading, or flow range of up to 6 gpm. The UFM will monitor flow and allow temporary operation up to 7 gpm, as addressed in Westinghouse letter, "RCP No. 1 Seal Extended Operating Limits." The TACF will modify the Westinghouse allowance of time at the 7 gpm leakage to permit operation with up to 7 gpm until the Unit 1 Cycle 4 outage. The other RCP seal operating parameters will remain unchanged.

The UFM does not violate pressure integrity of the seal leakoff piping since the sensor is non-intrusive, strap-on type. The signal processor will be located near the sensors) inside primary containment raceway. The signal processor will convert the sensor acoustic signals into a corresponding milli-amp dc signal. This signal will be connected to indicating/recording equipment located in the MCR. This signal will be transmitted by use of existing response time testing circuits. The use of this ultrasonic flow monitor will allow operations to measure seal flow above the 6 gpm limits of the existing instrumentation. Operation personnel will be alerted to a high seal flow condition. The TACF will optionally disable the existing alarm provided that the recorder alarm is functional.

This temporary UFM installation does not interact with any Class 1E circuitry and was evaluated for adverse impacts to other programs such as Seismic Category I(L) and Appendix R. This temporary alteration does not adversely affect any safety-related SSCs and is safe from a nuclear standpoint.

SA-SE Number: TACF-1-02-005-246-0

Implementation Date: 03/09/2002

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-02-05-246 ARI-1-7 ARI-71-75	Unit Station Service Transformers Sudden Pressure Device Circuitry Trip Function Defeat

Description and Safety Assessments:

This TACF 1-02-005-246 affects two non-safety-related systems by defeating the Unit 1 Unit Station Service Transformers (USST) Sudden Pressure Device Circuitry associated with the generator/turbine trip This is accomplished by opening the Trip Cutout blocks to defeat the Unit trip function of the sudden pressure device on Unit 1 USSTs 1A and 1 B. The trip cutout blocks are installed in the circuit for this purpose and require no special tools or material. These blocks are located in the main relay boards on the Unit 2 side of the Control Building.

The USST 1A and 1 B overcurrent trip is described in the UFSAR. The sudden pressure device is one of the inputs to the USSTs 1A and 1B overcurrent trip. The USSTs 1A and 1B overcurrent trip will continue to function to trip the generator and turbine if an overcurrent condition is sensed. The USSTs 1A and 1B overcurrent trip function is one of seventeen non-safety, non-seismic automatic turbine trips due to electrical faults in the generator, transformers, 500KV bus or breakers. These trips are non-safety-related and are designed for equipment protection. If the plant load is greater than 50%, a reactor trip is also generated. This reactor trip function is not affected and the design accident analysis for loss of load and/or turbine trip is not affected.

TACF 1-02-005-246 defeats the trip function of the sudden pressure device on the USSTs 1A and 1B but not the operating characteristics of the transformer nor other protective relays designed to actuate on various transformer, generator, bus, and breaker faults. The TACF does not increase likelihood of equipment failure or malfunction. The risk of equipment damage is one of economics rather than nuclear safety. By minimizing the potential for Unit trips due to spurious sudden pressure device actuation, WBN is accepting a small risk of more severe damage to the Unit 1 USSTs in the event of a transformer failure. This TACF does not create a new type event or impact fission product barriers. No new or different accidents are introduced and no previously analyzed accidents are affected Implementation of this temporary alteration does not increase the likelihood of a challenge to the plant or reactor shutdown. The technical specification and safety margins are not affected.

SA-SE Number: WBPLMN-02-042-0

Implementation Date: 7/15/2002

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-02-11-043	Manual Isolation Valve Closed to Prevent Minor Leakage

Description and Safety Assessments:

Minor through valve leakage of valve 1-FCV-43-21 downstream of manual isolation valve 1-SMV-68-578 (RCS Loop No. 3 Hot Leg sample valve) was observed during recovery from the forced outage which began on July 13, 2002 when the RCS was heated up. Since the leaking valve is located inside the crane wall, access for repair work is not possible in Modes 1 or 2. Repair of the fitting leak will be accomplished during the next entry into Mode 5 or later, depending on the priority of work. Until then, TACF 1-02-011-043 will close 1-SMV-68-578 to isolate the valve leak. The design configuration for 1-SMV-68-578 is normally open to allow sampling of the RCS from Hot Leg Loop No. 3. The capability to sample RCS fluid is described in UFSAR Section 9.3.2.2.

The sampling and water quality system is not safety-related except for valves associated with containment isolation. This system is not required to operate during or after a DBE and, therefore, does not need to comply with single failure requirements. Sampling of the RCS is used to detect failed fuel. RCS sampling is used to determine gross specific activity and dose equivalent I-131 analyses. The RCS is assumed to be a homogenous solution. Therefore, it is inconsequential whether the sample point are UFSAR and technical specification required samples, but where the samples are taken is not specified. The ability to sample from RCS Hot Leg Loop No. 1 is not impacted by this temporary alteration, as well as other alternate sample points which have been used by Chemistry in the past (such as downstream of the CVCS letdown heat exchanger). While this temporary alteration does eliminate one sample location, there is not requirement that samples be taken at this location, and Chemistry is still left with the ability to not only sample from RCS Hot Leg Loop No. 1, but has other alternate locations available to sample from. It is concluded that the temporary elimination of this sample point will not impact the ability to meet the UFSAR described design function, nor does it impact the ability to satisfy the technical specification surveillance requirements. The change is positive from the standpoint that the risk of continued leakage and possible increased leakage from the RCS is eliminated by closing the isolation valve until such time as a repair can be made. Closure of 1-SMV-68-578 will also render pressure indicator PI-68-42 inoperable. This pressure gage gives local indication only in the raceway and is not required for plant operations.

Isolation of the RCS Hot Leg Loop No. 3 does not introduce the possibility of a change in the frequency of an accident or malfunction of equipment. Closure of this valve does not affect or initiate any accident analyses.

SA-SE Number: WBPLEE-00-061-0

Implementation Date: 05/15/2001

Document Type:	***Affected Documents:***	***Title:***
Technical Requirements Manual	TRM-00-010 and FSAR Package 1642	Molded Case Circuit Breakers Preventative Maintenance Frequency

Description and Safety Assessments:

This change extends preventative maintenance periodicity of molded case circuit breakers (MCCB) from 60 months to:

72 months for Class 1E MCCB, and

96 months for non-Class 1E MCCB.

This frequency will align MCCB preventative maintenance periodicity with industry guidance, e.g. EPRI NP-7410-V3, "Molded Case Circuit Breaker Application and Maintenance Guide," and, in the case of 1E MCCBs, coordinate with the present 18 month fuel cycles at WBN; thereby resulting in maintenance of 25% of the MCCB each fuel cycle.

MCCBs are used throughout the electrical power system in safety and non-safety applications. The breakers are at the bottom of the distribution system and, thus, are normally fed from larger switchgear type circuit breakers. Their function is to serve as both manual switches and automatic switches which actuate on overcurrent. These are sealed devices upon which internal maintenance is not possible. A credible failure mode of these devices is binding in the internal mechanism which could slow or stop the automatic opening of the circuit breaker, thereby causing an upstream circuit breaker to be called upon to isolate the cause of an overcurrent and consequently resulting in increased damage downstream of the impaired circuit breaker due to the time delay in interrupting the fault current. There have been a few, isolated instances of such binding of the internal mechanism reported in the industry. However, based on past operating experience at Watts Bar, binding has not been an identified problem.

UFSAR Chapter 15 accidents considered are:
Condition II - Loss of Offsite Power
Condition III - Small Break LOCA
Condition IV - Large Break LOCA, Fuel Handling Accident, Environmental Consequences of Accidents, Environmental Consequences of a Postulated LOCA.

This lengthening of the preventative maintenance periodicity of MCCBs which mechanically exercises the breaker is consistent with current industry (EPRI) guidance and better coordinates with the current 18 month fuel cycles. No new configurations, functional relationships, nor operational characteristics are introduced. Periodic testing for performance degradation remains as previously required and will detect any need for more frequent preventative maintenance; therefore, there is no increased probability of malfunction or accident than previously analyzed There is no increase in the consequences of an accident or equipment malfunction than previously analyzed. There is no possibility for different accidents nor malfunctions of a different type than evaluated in the UFSAR, nor are safety margins affected. Consequently, this lengthening of MCCB preventative periodicity does not constitute a need for a license amendment.

SA-SE Number: WBPLEE-02-016-0

Implementation Date: 03/06/2002

Document Type:	*Affected Documents:*	*Title:*
Procedure Change	Fuel Handling Instruction FHI-2, Fuel Handling Instruction FHI-3, Fuel Handling Instruction FHI-7	One Time Procedure Change

Description and Safety Assessments:

The Fuel Handing System is designed to handle new and spent fuel from the time it enters the site. This system includes three subsystems: Refueling Machine, Fuel Transfer System, and SFP Bridge Crane

The described change affects the FTS and SFP bridge crane subsystems. The change was a one-time change which was incorporated into the procedure for use and subsequently cancelled prior to implementation.

The FTS includes an electric gear motor driven transfer car that runs on a track extending into the SFP with a lifting arm (upender) located at both ends of the transfer tube. The normal fuel assembly transfer sequence is for the transfer car being driven to the SFP end of the track The upender is raised to a vertical position The SFP bridge crane, which has been positioned at least 20 feet from the upender, is now moved over the upender with the hoist in its full-up position. The SFP bridge movement is restricted unless the hoist is in the full-up position The fuel assembly, which is connected to the hoist using a special handling tool, is lowered into the upender and released. The SFP hoist is raised to its full up position, and the SFP bridge is then moved out of the 20 foot zone. The upender now lowers the fuel assembly to the horizontal position for transfer through the transfer tube and the upender on the other side raises the fuel cell to the vertical position. The SFP bridge and the fuel upender are interlocked so that the upender will halt operation if the SFP bridge has not been moved out of the transfer canal 20 foot buffer zone between the bridge and upender.

The one-time procedure change allows the physical separation between the FTS fuel upender and the SFP bridge crane to be maintained administratively. The function of the two non-safety-related travel limit switches will be administratively controlled with existing software capability such that the SFP bridge will be allowed to be inside the transfer canal 20 foot buffer zone, prior to or while the upender is attaining the required vertical position or to receive the fuel assembly, and while starting of the automatic sequence of lowering the upender to send the fuel assembly to the Reactor Building. The administrative control also allows the FTS upender to start moving once it has been verified that the SFP bridge crane hoist is unlatched and has been placed in the full-up position. The interlock will be restored in accordance with the design upon completion of the procedure. The following administrative controls will be put in place of the removed physical bridge interlock:

The change from a hardwired interlock to use of administrative controls to provide a barrier against a fuel handling accident is acceptable to be implemented. The failure mode introduced by this change, nonadherence to the procedure, does not increase the frequency of an evaluated accident or increase the likelihood of a malfunction. NEI-99-02 establishes guidelines that can be used to determine if a specific operator action is virtually certain to be successful. In order to be considered virtually certain to be successful the action must be contained in a written procedure, it must be uncomplicated (a single action or a few simple actions), it must not require diagnosis, and there must be a dedicated local operator for the duration of the activity. The operator action required by this procedure change fulfills these requirements Therefore, this activity does not increase the frequency of an accident or the likelihood of a malfunction. Any consequences resulting from this activity are bounded by the UFSAR Chapter 15 Fuel Handling Accident. This change does not affect the ability of the refueling system to perform its UFSAR described design function This change does not impact the fission product barriers and is not a departure from a method of evaluation described in the UFSAR. The technical specifications are not affected by this change. Therefore, a license amendment is not required.

SA-SE Number: WBPLMN-02-013-0

Implementation Date:02/25/02

Document Type:	***Affected Documents:***	***Title:***
Work Order	WO 01-001797-000 WO 01-005133-000 WO 01-005092-000 TS Bases TSB-02-02	Installation of Rubber Test Ball Plug for Containment Closure

Description and Safety Assessments:

This safety evaluation is for the portion of W. O. 01-001797-000, 01-005133-000, and/or 01-005092-000 that deals with installation and use of the Cherne Model 277-169 Urethane Rubber Test Ball plug for closure of containment when internals of the main steam isolation valves (MSIVs) are removed for maintenance. This screening review also covers the associated Tech Spec Bases change to replace the words "NRC approved" with "approved" in TS Bases 3.9.4.

During Unit 1 Cycle 4 outage in Modes 5 and/or 6, internals of the MSIVs will be removed to perform maintenance activities. If the steam generator manways or other penetrations are open, a flow path will exist from containment (inside the ABSCE) to the valve vault (outside the ABSCE). To attain containment closure during this time, a Urethane test plug will be installed in the 32-inch main steam piping at the MSIV location. The plug cannot be used for containment closure during reduced inventory/midloop operations.

Since the plug will not be used during reduced inventory/mid loop operations, the only accident to consider during the time the plug is used is a fuel handling accident. The plug is not procured to safety related requirements and is not redundant. However, based on guidelines provided by the NRC in Generic Letter 88-17 on containment closure for no power operations during reduced inventory/midloop conditions, a single non safety related containment closure method is acceptable if the closure method can reasonably assure the accident environment subsequent to an accident will be isolated and is similar in capability to those provided for containment isolation during power operations. Since a reduced inventory/midloop loss of RHR accident could result in core melt and subsequent elevated radiation, temperature, and pressure environment inside containment, a valve or flange is required for containment closure during reduced inventory/mid loop operations. The Urethane plug will be credited only for isolation of containment in the event of a fuel handling, accident. A fuel handling accident produces a radiation dose of 7.75 E+3 rads, but does not produce an increase in containment pressure or temperature. Urethane rubber is qualified to 4.0 E+7 rads, consequently, Urethane plug will reasonably assure containment of a fuel handling accident environment inside containment. The plug will be inspected periodically during use to ensure a failure of the plug (defined as loss of sealing capability) does not occur.

Technical Specification Bases Change No. TS-02-02 revises Technical Specification Bases 3.9.4 (Background, Page B 3.9-13) to clarify the approval authority for "equivalent isolation methods" for containment penetrations. The current LCO 3.9.4 allows, during refueling operations, that Containment Closure may be achieved by isolating valves or blind flanges, "or equivalent." The current Technical Specification Bases stipulates that the equivalent isolation methods must be approved by NRC. The subject change will clarify that "equivalent isolation methods" for achieving containment closure must only be "...approved.." (e.g., by station management), thereby removing the implication that NRC is the approval authority for "equivalent isolation methods."

The UFSAR currently requires that the ABSCE boundary is maintained during refueling operations. This safety evaluation concluded that the activities proposed in work orders and technical specification bases do not violate the ABSCE boundary. Isolation is required for mitigation of the fuel handling accident and therefore, these changes that implement the method of isolation do not a) impact the frequency of occurrence of an accident, b) create a possibility for an accident of a different type, c) result in a fission product barrier limit being exceeded or altered, or d) depart from a method of evaluation. Guidance prepared by the NRC in Generic Letter 88-17 and qualification of the urethane plug for the environment produced by a fuel handling accident provided justification for the adequacy of the urethane plug for containment closure during refueling operations (but excluding reduced inventory/m id loop operations) and thus no adverse effect on the remaining questions of this safety evaluation.

SA-SE Number: WO 01-008855-000 R0

Implementation Date: 06/21/2001

Document Type:	*Affected Documents:*	*Title:*
Work Order	WO-01-008855-000	Jumper Installed in the SSPS to Clear the General Warning Alarm

Description and Safety Assessments:

WO 01-008855-000 reconfigured solid state protection system (SSPS) TR-B general warning alarm to delete or mask the input from the failed primary 15V power supply failure. The primary 15V power supply failed and short term attempts to troubleshoot and correct have not been successful. To clear the alarm and unmask the remaining inputs to the alarm and the potential inadvertent reactor trip, a jumper is added to the warning circuit wiring to circumvent the loss of power supply input to the alarm function. The inputs to the alarm are:

- Failure of either of two 15 Volt power supplies.
- Failure of either of two 48 Volt power supplies.
- Test switch not in a normal position.
- Logic circuit card removed.
- Loss of AC power.
- Ground return fuse blown.
- Bypass Reactor Trip Breaker Closed.

The alarm can be characterized into two categories, degraded detection and off normal detection. The degraded detection is for the inputs that affect the reliability of the SSPS to perform the safety-related function i.e. loss of power supplies, logic card removed, loss of AC and ground fuse cleared. Off normal detection is the more important detection since it prevents rendering both trains of SSPS nonfunctional when in Bypass or test mode. The prevention is in the form of a reactor trip. The same relay is used for the degraded detection and off normal status, and thus a degraded condition in both trains results in a reactor trip. This change will only remain in effect till the problem with the power supply is corrected at which time the system will be restored to normal configuration.

The change involves blocking a single failed power supply input to the Train B SSPS general warning alarm monitoring circuit, therefore unmasking the general warning monitoring circuit to allow it to detect any other alarm condition. In addition this change removes the half reactor trip condition caused by the Train B SSPS general warning alarm, thus reducing the likelihood of a reactor trip from a Train A general warning concurrent with the unchanged Train B configuration which would challenge safety systems while not adding to overall plant safety. These are positive changes as a result of the temporary alteration of the alarm circuit. Conversely removing the half reactor trip condition from the Train B general warning alarm and isolating the power supply, thus reducing redundancy, are negative effects of the change. However, the SSPS is not an initiator of any accident event nor will the single failure design basis be changed. SSPS ability to perform it's safety functions are not compromised by this change. Thus the temporary alteration is an overall positive change for the present condition. The change does not affect accident initiation, radiological consequences, fission product barriers, or evaluation methodologies. The change involves SSPS which is important to safety. However the changes do not increase the likelihood of malfunction or the possibility of a new and different malfunction to SSPS. This change in configuration does not require NRC review.