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

APR 0 6 2001

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)(2). Enclosure 1 provides the evaluations for the Updated Final Safety Analysis Report Amendment 2 and includes other evaluations performed during the period from October 1, 1999 to March 29, 2001. Also included in Enclosure 2, are five evaluations which were inadvertently omitted from the previous Summary Report submitted on October 18, 1999.

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
and Industry Affairs

Enclosures
cc: See page 2

IE47

APR 0 6 2001

cc (Enclosures):

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ENCLOSURE 1

10 CFR 50.59 SUMMARY REPORT

ABBREVIATIONS

ABGTS	Auxiliary Building Gas Treatment System
ABI	Auxiliary Building Isolation
ABSCE	Auxiliary Building Secondary Containment Enclosure
ACR	Auxiliary Control Room
ACU	Air Conditioning Unit
A/D	Anchor Darling
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
AHU	Air Handling Unit
ALARA	As Low As Reasonably Achievable
AMSAC	Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC)
AOI	Abnormal Operating Procedure
ARC	Automatic Recirculation Control
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BE	Best Estimate
BMI	Bottom Mounted Instrumentation
BP	Buisness Practice
CAM	Continuous Air Monitor
CBP	Condensate Booster Pumps
CCD	Configuration Control Drawing
CCP	Centrifugal Charging Pumps
CCS	Component Cooling System
CCTV	Close Circuit Television
CID	Component Identification
CILRT	Containment Integrated Leak Rate Test
CISI	Containment Inservice Inspection
CLA	Cold Leg Accumulator
COLR	Core Operating Limits Report
COT	Channel Operational Test
CPDS	Condensate Polishing Demineralizer System
CRDM	Control Rod Drive Mechanism
CRI	Control Room Isolation
CREVS	Control Room Emergency Ventilation System
CSST	Common Station Service Transformer
CTB	Cooling Tower Blowdown
CVCS	Chemical Volume and Control System
CVI	Containment Vent Isolation
DAW	Dry Active Radioactive Waste
DBA	Design Basis Accident
DBE	Design Basis Event
DCN	Design Change Notice
DD	Drawing Deviation
DNB	Departure from Nucleate Boiling

ABBREVIATIONS

ECCS	Emergency Core Cooling System
EGTS	Emergency Gas Treatment System
EDC	Engineering Document Change
EDS	Environmental Data Station
EHC	Electrohydraulic Control
EOC	End of Cycle
EOL	End of Life
EMI	Electromagnetic Interference
ENS	Emergency Notification System
EPG	Energy Products Group
EQ	Equipment Qualification
ERCW	Essential Raw Cooling Water
FAC	Flow Accelerated Corrosion
FHA	Fuel Handling Area
FHSS	Fuel Handling and Storage System
FPR	Fire Protection Report
FSSD	Fire Safe Shutdown Analysis
FTC	Fuel Transfer Canal
FVBR	Fifth Vital Battery Room
GDC	General Design Criteria
GL	Generic Letter
GO	General Operating Procedure
HCT	High Crud Tank
HDT	Heater Drain Tank
HIC	High Integrity Container
HUT	Holdup Tanks
HVAC	Heating, Ventilation, and Air Conditioning
ICS	Integrated Computer System
IFBA	Integral Fuel Burnable Absorber
I/O	Input/Output
IST	Inservice Testing
LBLOCA	Large Break Loss of Coolant Accident
LCC	Lower Compartment Coolers
LCO	Limiting Condition for Operation
LCV	Level Control Valve
LEFM	Leading Edge Flow Meter
LLD	Low Level Detection
LLRT	Local Leak Rate Test
LOCA	Loss of Coolant Accident
LONF	Loss of Normal Feedwater
LOOP	Loss of Offsite Power
LRPS	Liquid Radwaste Processing System
MCR	Main Control Room
MDAFW	Motor Driven Auxiliary Feedwater
MEL	Master Equipment List
MFP	Main Feedwater Pump

ABBREVIATIONS

MFW	Main Feedwater
MI	Maintenance Instruction
MOV	Motor Operated Valve
MSR	Mositure Seperator Reheater
M&TE	Maintenance and Test Equipment
MTC	Moderator Temperature Coefficient
MTOT	Main Turbine Oil Tank
NEI	Nuclear Energy Institute
NFE	New Fuel Elevator
NIS	Nuclear Instrumentation System
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
OM	Operational and Maintenance
OTO	One Time Only
PAI	Plant Administrative Instruction
PD	Positive Displacement
PORV	Power Operated Relief Valve
PCT	Peak Clad Temperature
PCV	Pressure Control Valve
PER	Problem Evaluation Report
PLC	Programmable Logic Controller
PM	Preventative Maintenance
PRT	Pressurizer Relief Tank
PT	Pressure Transmitter
PTLR	Pressure Temperature Limits Report
PWHT	Post Weld Heat Treatment
QA	Quality Assurance
RCCA	Rod Cluster Control Assembly
RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RCW	Raw Cooling Water
REMP	Radiological Environmental Monitoring Program
RIS	Regulatory Issue Summary
RHR	Residual Heat Removal
RRTN	Replacement Reconstitutable Top Nozzles
RTP	Rated Thermal Power
RWST	Refueling Water Storage Tank
SCCW	Supplemental Condenser Circulating Water
SDD	System Description Document
SER	Safety Evaluation Report
SPP	Site Procedures and Processes
SFPCCS	Spent Fuel Pool Cooling and Cleanup System
SGBD	Steam Generator Blowdown
SGTR	Steam Generator Tube Rupture
SI	Safety Injection

ABBREVIATIONS

SOI	System Operating Instruction
SR	Surveillance Requirement
SRDS	Solid Radwaste Disposal System
SRPS	Spent Resin Packaging System
SRST	Spent Resin Storage Tank
SSC	Structure, System, or Component
SSC	Scaling and Setpoint Document
SSE	Safe Shutdown Earthquake
TACF	Temporary Alteration Control Form
TGCPS	Turbine Generator Control and Protection System
TI	Technical Instruction
TIC	Temperature Indicating Controller
TDCT	Tritiated Drain Collector Tank
TRM	Technical Requirements Manual
TS	Technical Specification
TSC	Technical Support Center
TSTF	Technical Specification Task Force
UFSAR	Updated Final Safety Analysis Report
UPS	Uninterruptible Power Supply
USST	Unit Station Service Tranformers
VCT	Volume Control Tank
V+/P+	Vantage+/Performance+
VT	Visual Test
WABA	Wet Annular Burnable Absorbers
WAW	Wet Active Wastes
WDS	Waste Disposal System
WGDT	Waste Gas Decay Tank
WLRS	Wet Layup Recirculation System
WO	Work Order

SA-SE Number: BASES CHANGE 2000-001

Implementation Date: 4/13/2000

Document Type:	***Affected Documents:***	***Title:***
TS Bases	TS Bases B.3.1.3, Core Reactivity TS Bases B.3.1.4, Moderator Temperature Coefficient	Moderate Temperature Coefficient

Description and Safety Assessment:

The Technical Specification Bases for B 3.1.3, Core Reactivity and B 3.1.4, Moderator Temperature Coefficient (MTC) are being changed to reflect the behavior of the MTC as a result of the use of Integral Fuel Burnable Absorbers (IFBA).

The proposed Technical Specification Bases change does not introduce an unreviewed safety question because the Technical Specification upper and lower limits are not being changed, as a result of these Bases changes, nor is the surveillance testing methodology being changed. The surveillance testing continues to be conducted within the requirements of the Technical Specification (TS) Surveillance Requirements (SR). The addition of the details describing the effects on IFBA on the Moderator Temperature Coefficient does not affect Nuclear Safety and does not pose an unreviewed safety question.

SA-SE Number: BASES CHANGE 2000-002

Implementation Date: 09/28/2000

Document Type:	***Affected Documents:***	***Title:***
Technical Specification Bases Change	TS Bases B 3.1.10, Physics Tests Exceptions Mode 2, Bases B3.1.9	Physics Testing Program

Description and Safety Assessment:

The Technical Specification Bases for B 3.1.9, PHYSICS TEST Exceptions - MODE 1, and B 3.1.10, PHYSICS TESTS Exceptions - MODE 2 are being changed to remove the details of the ANSI/ANS - 19.6.1 Physics Test Program and to generically address the physics testing program.

The proposed Technical Specification Bases does not introduce an unreviewed safety question because the WBN Physics Testing Program is not changing. The change will eliminate the details described in ANSI/ANS-19.6.1 Physics Testing Program from the Background Sections of WBN's Bases Technical Specifications. This information does not affect Nuclear Safety and does not pose an unreviewed safety question because the tests either confirm design predictions/design methods and are conducted in accordance with the vendor recommendations or the tests are conducted within the requirements of the Technical Specification Surveillance Requirements.

SA-SE Number: BASES CHANGE 2001-01

Implementation Date: 03/07/2001

Document Type:	***Affected Documents:***	***Title:***
Other	TS Bases Change 01-01	Clarification of the Technical Specification Operability requirements for the Pressurizer Power Operated Relief Valves (PORVs)

Description and Safety Assessment:

The proposed revision to Technical Specification Bases Section 3.4.11 provides clarification of the OPERABILITY requirements for the PORVs 1-PCV-68-334 and 340A and associated block valves in Modes 1, 2 and 3. The PORVs are designed for both manual and automatic control. Automatic operation of the PORVs is not credited for accident mitigation in the design basis event (DBE) analyses in Modes 1, 2, and 3. The automatic function of PORVs is desirable for reducing challenges to the code safety valves for overpressure events. Manual actuation is assumed in the steam generator tube rupture (SGTR) analysis. For the purpose of defining Technical Specification OPERABILITY of PORVs, only the manual actuation function is required. The proposed revision provides this clarification.

The PORVs limit system pressure for a large power mismatch and thus prevent actuation of the high-pressure reactor trip. They are designed to limit reactor coolant system (RCS) pressure to a value below the high-pressure trip setpoint for all design transients up to and including a 50% step load decrease with steam dump actuation. The PORVs also provide a means for venting non-condensible gasses or steam from the pressurizer which may impair stabilization of the RCS following a DBE.

Additionally, the PORVs provide a means to depressurize the RCS following a SGTR event to reduce primary to secondary break flow as well as increase safety injection flow to refill the pressurizer, and to mitigate potential RCS cold overpressure transients. Each PORV can be manually operated from the control room. Also, each PORV can be individually isolated by a remotely operated, normally open, upstream block valve providing redundant isolation in the event of leakage. The block valves also permit performance of surveillances during plant operation.

This change is based on Industry Standard Technical Specification Task Force (TSTF) Change, TSTF-151 Revision 1, which has been reviewed and approved by the NRC. The change to the discussion of Actions C.1 and C.2 differs slightly from the Traveler to further clarify the basis for the Actions.

The analyses of overpressure events conservatively assume PORV operation where this produces more severe accident results. However, no credit is taken for the relief capability of the PORVs. The code safety valves are sized to protect against overpressure conditions resulting from the most limiting events. The applicable DBEs and UFSAR Chapter 15 section for this condition include Loss-of-Load/Turbine Trip, Loss of Normal Feedwater (LONF), Inadvertent Operation of emergency core cooling system (ECCS), and Main Feedwater Line Break. The SGTR event analysis assumes that the PORVs will be manually operated to depressurize the RCS if normal pressurizer spray is not available.

This change does not affect plant equipment or operation; equipment failure modes are not impacted.

The proposed change is based on Industry TSTF Standard Technical Specification Change Traveler TSTF-151 Revision 1, which has been reviewed and approved by the NRC. The change is a clarification to the Technical Specification bases language to distinguish that only the manual mode is needed for PORV OPERABILITY. It provides an improvement to the Technical Specification bases language and eliminates the potential for misinterpretation regarding the automatic PORV function with respect to PORV OPERABILITY. The change does not alter plant equipment or operation, setpoints, or assumptions in the accident analyses. No new failures modes will be introduced. Based on evaluation of the effects of the change, the proposed revision will not impact nuclear safety or result in a unreviewed safety question.

SA-SE Number: DCN D-50423-A

Implementation Date: 3/16/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50423-A	Gate Valve replaced with a ball valve.

Description and Safety Assessment:

Design Change D-50423-A replaces a leaking gate valve with a ball valve which performs an isolation function in the potable water system. The manufacturer of the backflow preventer no longer utilizes gate valves in this isolation capacity due to continued problems with the integrity of the gate valves. The replacement ball valve will offer more positive shutoff performance, reduced maintenance, and longer life due to its inherently robust mechanical design. The integral ball of the proposed replacement valve offers greater mechanical strength due to the geometric stability of the ball and offers greater sealing surface area. These features result in reduced probability that the valve will leak. The isolation function of the valve is enhanced due to the reduced probability of leaking and the reduced maintenance requirements of the replacement valve.

The replacement valve has essentially the same operational characteristics as the original. The UFSAR contains no detail about the operation of this valve, thus there is no change to the system operational characteristics as described in the UFSAR. No processes or procedures outlined, summarized, or described in the UFSAR are impacted by this change, either directly or indirectly.

The modification does not increase the probability of occurrence of a malfunction of equipment important to safety. The system design, function, or method of performing that function are not impacted by this change and the valve being revised is not discussed in the UFSAR. No automatic or manual features are added, deleted, or revised by this change. No system interactions are added deleted, or revised by this change. No classification, seismic qualification, or environmental qualification is affected by this modification. The failure position, size, design pressure and temperature, and operator of this replacement valve are equivalent to the existing valve.

The potable water system is completely independent of any system conveying radiological material or fluids and is not implicated in any accident scenario, either directly or indirectly. The modification is not associated with any equipment previously evaluated in the UFSAR for malfunction scenarios. The potable water system has no technical specification requirements and the modified valve can not impact any technical specification equipment or systems. The modification introduces no new failure pathways. Therefore, the replacement valve is not considered an unreviewed safety question.

SA-SE Number: DCN D-50440-A

Implementation Date: 08/09/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50440-A TS Bases Pkg 98-012	Power Distribution System Boards Required for Unit 1 Operation

Description and Safety Assessment:

Technical Specification Bases, Table B 3.8.9-1, AC and DC Electrical Power Distribution System, list power boards that are required for Unit 1 Technical Specification operability. However, a few of these boards are not necessarily functionally required to meet Technical Specification Section 3.8.9.

For the boards listed in Technical Specification Bases, Table B 3.8.9-1, each connected load was reviewed for Unit 1 operational and shutdown needs. Loads, which were Unit 1 required safety loads, automatically made the power boards required. Loads, which were not intended as the means for Unit 1 to operate, shutdown, or remain safely shutdown, were considered an operational convenience. Convenience loads and boards are fully tested and maintained; and if aligned, they may be intentionally substituted for a required load, making them required.

Also, while performing this review, it was noted that several minor errors associated with Calculation WBN-OSG4-129, Unit 2 Equipment within the Unit 1 Boundary needed to be corrected. This DCN was used as a vehicle for change; however, these corrections are not a direct result of the Technical Specification Change Package or subsequent documentation for that package. For example, on TVA single line drawing 1-45W755-4, an inadvertently omitted delta symbol associated with drawing Note 5 was added to the Computer Room Air Handling Unit No. 1 (0-MTR31-497) load. This revision corrects the safety classification of the associated load as shown on this drawing. Additionally, Problem Evaluation Report (PER) 99-011114-000 identified a problem with the Ul/U2 boundary for 2-LCV-2-132-A which is fed from Reactor motor-operated-valve (MOV) Board 2A1-A. Presently, the load breaker for 2-LCV-62-132-A is temporarily tagged out of service (de-energized), and the mechanical load is documented as not used for Unit 1 operation. This DCN corrects the boundary omission. Drawings 1-45W751-1, 1-45W760-62-6, and 45B2766-8B are revised to show the U1/U2 boundary.

Unit 1 was evaluated to operate, shutdown, and be maintained in a safe shutdown condition without the use of any convenience power board or load. The convenience loads (2-HTR-062-0239-A, 2MTR-062-0230-A, 2-HTR-062-0228/4, 2-MTR-062-0232-8, and 2-HTR-062-0245-8) can be used to perform the same function as their Unit 1 counterparts, but are not credited for performing these functions. Likewise, removing power from these boards will de-energized 0-FCV-26-8 which is identified for manual or electrical operation. Removing control fuses and thermal overload heaters for a mechanical load which is identified as not within the Unit 1 boundary is likewise acceptable and does not constitute an unreviewed safety question. This DCN will not change testing or previous test results or cause an experiment; it will not affect the design basis accident (DBA) and credible failure mode evaluation. The change is acceptable to implement.

SA-SE Number: DCN D-50451-A

Implementation Date: 02/09/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50451-A	Lead Edge Flow Meter (LEFM) Installation

Description and Safety Assessment:

DCN D-50451-A provides for the installation of a Caldon LEFM in the Main Feedwater (MFW) 32 inch header piping. Installation of the spoolpiece required cutting the 32 inch MFW line to insert the weld end spoolpiece.

The LEFM System is comprised of one measurement section (spoolpiece), an electronic unit, and instrument cables for each transducer in the system. Each measurement section holds eight ultrasonic transducer assemblies, secured in their own transducer housing which forms the pressure boundary. Each transducer may be removed at full power conditions without disturbing the pressure boundary. The LEFM electronic unit controls magnitude of the ultrasonic signal and sequences the operation of the transducers, makes time measurements, calculates volume flow, temperature, and mass flow. The electronic unit has a touch-screen user interface.

The goal of the LEFM modification is to take advantage of the high degree of accuracy associated with the LEFM and reduce the margin required by 10CFR50 Appendix K for uncertainties. For all accidents under the Appendix K criteria, this margin currently is the 2% difference between the analyzed plant design basis, of 102% of the rated thermal power and the licensing basis of 100% of rated thermal power.

The LEFM technology can provide WBN Unit 1 with a minimum 1% power uprate by providing improved feedwater flow measurement accuracy. A chordal LEFM system is more accurate than external LEFM applications and the currently installed venturi nozzles. Improved feedwater flow accuracy measurement permits reduction in the uncertainties required by Appendix K; therefore permitting operation at a minimum of 1% above current power level. The reactor core power will increase from 3411 MWT to a minimum of 3445 MWT.

This technology, utilizing ultrasonic transducers installed in a spoolpiece in the MFW 32 inch header piping, may be used to correct for venturi fouling and the corrected value used in the plant calorimetric calculation, using the FW flow venturi nozzles as inputs.

Neither the DBAs or the credible failure modes are applicable to this DCN D-50451-A because the LEFM will only be installed and functioning to provide indication only and will not be fully operational by utilization of its input for calorimetric calculations until the return to operation of a later DCN D-50494-A which will justify the minimum 1% power uprate.

LEFM cannot influence safety related equipment to be more likely to malfunction because LEFM will not be fully operational. LEFM provides a monitoring function, and by definition cannot increase the consequences of an accident, since those consequences are a function of what takes place after information from a compliance instrument such as LEFM has not been used successfully to prevent or mitigate an accident occurrence. LEFM cannot increase the consequences of a malfunction because it does not change any equipment in the flow regime which could influence those consequences except the LEFM spoolpiece which has only minor internal discontinuities in the flow regime. LEFM cannot create the possibililty of an accident of a different type because it does not change existing operating equipment, procedures, operation, testing, or experiment under DCN D-50451-A. LEFM cannot create the possiblity of a malfunction of a different type because there is no new equipment to malfunction until LEFM goes fully operational, and no reason to believe that the existing equipment will be more likely to malfunction. The initial installation of LEFM under DCN D-50451-A will not reduce the margin of safety as defined in the basis of any Technical Specification because those margins apply to existing equipment and power ratings as used currently before the minimum 1% power uprate and will not change until the uprate. Based on the above discussion, the installation of LEFM under DCN D-50451-A does not involve an unreviewed safety question.

SA-SE Number: DCN D-50462-A-1

Implementation Date: 12/11/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50462-A	Replace the No. 3 Heater Drain Tank Level Controllers with Electronic Controllers.

Description and Safety Assessment:

The heater drain system is designed to remove and dispose of secondary source drainage during all modes of unit operation by returning the condensed water back to the Condensate-Feedwater System. During normal operation the No. 1 heater cascades into the shell of the No. 2 heater. The No. 2 heater drains, the No. 3 heater drains, and the moisture separator drains flow into the No. 3 heater drain tank (HDT). Water from the No. 3 HDT is then pumped forward into the condensate cycle by the No. 3 heater drain pumps.

Generally, the operating mode of the heater drains system has no effect on the reactor coolant system and the ability of the condensate and feedwater systems to deliver feedwater to the steam generators in sufficient quantity to meet all system demands. However, some transient conditions associated with the No. 3 HDT and drain pumps require proper interfacing with other secondary cycle systems to prevent a reactor trip.

The modifications associated with this DCN do not change the system operation of the No. 3 HDT controls but provides state of the art upgrades to the existing control instruments. The instruments being replaced by this change are pneumatic level controllers 1-LIC-6-105 and 1-LIC-6-106. The LIC-6-106 loop will be replaced by a 4-20 ma electronic control loop consisting of a new transmitter that will monitor 108 inches of the total 120 inches of the No. 3 HDT level, a new programmable logic controller (PLC) with an internal loop power supply, and a new current to pneumatic converter that will operate the level control valve

The No. 3 heater drain tank level standpipe arrangement will be modified to remove all unnecessarily piping and abandoned instrumentation. In addition, the associated condensate pot will also be reinstalled at the same elevation as it was previously. The 1 inch standpipe drain valve will also be relocated to the new section of standpipe.

UFSAR Chapter 7 and Section 10.4 were reviewed for potential impact from this change. The Heater Drains and Vents System and specifically the No. 3 heater drain tank and its associated level indication/controls are not described in the UFSAR text. However, the UFSAR does contain three figures that will be affected by this change. Figure 10.4-30, Figure 10.4-32, and Figure 10.4-34 will be revised by this DCN.

The No. 3 HDT level controls/indication do not perform a UFSAR Chapter 6 or Chapter 15 accident mitigation function. The equipment being replaced by this change is not safety-related, is not quality-related, is not technical specification or compliance related, and does not perform a UFSAR Chapter 6 or Chapter 15 accident mitigation function. The change only affects three figures in the UFSAR Section 10.4 which are the control diagram and the logic diagram. The change will be installed when the system is de-pressurized and removed from service. The modifications associated with this DCN does not change the system operation of the No. 3 HDT controls but provides state of the art upgrades to the existing control instruments. The replacement instrumentation does not change the system failure modes and does not increase the likelihood of a plant transient to occur. The HDT controls and indication are not used to perform any safety-related, safe shutdown, or accident mitigation functions, nor are they used to perform functions essential to the health and safety of the public. This change introduces no increased probability of an accident or malfunction of equipment important to safety, or create the possibility for an accident or malfunction of a type different than any evaluated previously in the UFSAR. This change introduces no increased radiological consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR. Therefore, the change does not result in an unreviewed safety question.

SA-SE Number: DCN D-50475-A

Implementation Date: 10/01/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50475-A	AMSAC Circuitry Replacement

Description and Safety Assessment:

DCN D-50475-A replaces the existing Anticipated Transients Without Scram (ATWS) Mitigation System Action Circuitry (AMSAC) microprocessor based logic circuitry with relay-based logic circuitry. Modifications are within the existing AMSAC cabinet and only internal components that perform the AMSAC operational logic functions are within the scope of this DCN. AMSAC inputs (turbine first stage impulse pressure input signals from AMSAC dedicated transmitters and existing narrow range steam generator level transmitters) are not changed by this DCN. Similarly, AMSAC output interfaces, including safety-related isolation devices, with the auxiliary feedwater (AFW) system and turbine trip circuitry remain unchanged.

AMSAC has no primary safety functions. AMSAC performs secondary safety functions in that it is of sufficient quality and reliability to perform its intended function without contributing to transients which may challenge safety systems. AMSAC does not degrade the existing reactor protection system.

The DBEs for the AMSAC are LONF/ATWS and the Loss of Load/ATWS. The scenarios of those two events are described below:

A complete loss of normal feedwater occurs as a result of a malfunction in the feedwater/condensate system or its control system from such causes as the simultaneous trip of all condensate pumps, the simultaneous trip of all main feedwater pumps or the simultaneous closure of all main feedwater control, pump discharge or block valves. Because if a postulated common mode failure in the RPS, the reactor is incapable of being automatically tripped when any of several plant process variables have reached their reactor trip setpoints.

The most severe plant conditions that could result from a loss of load occur following a turbine trip from full power when the turbine trip is caused by a loss of main condenser vacuum. Because of a common mode failure in the protection system, the reactor is incapable of being automatically tripped as a result of the turbine trip or as the result of any several other reactor trip signals that occur later in time when several plant process variables reach their reactor trip setpoints.

The components installed to perform the AMSAC logic functions have been utilized in other nuclear applications to demonstrate a history of quality and reliability. The relay-based logic circuitry has been designed that output relays shall be energized to actuate in order to prevent spurious trips and false status indication on loss of power or logic. Therefore, no new failure modes are introduced by this modification.

The purpose of the AMSAC system remains the same: To function diverse from the reactor trip system to automatically start the AFW pumps and to initiate a turbine trip under conditions indicative of an ATWS.

This modification does not change any system trip setpoints, and loop accuracy, as shown by calculation, is within the previous calculated loop accuracy, therefore, this modification will support the operational limits defined for AMSAC. For these reasons, this activity does not constitute a unreviewed safety question.

SA-SE Number: DCN D-50494-A-1

Implementation Date:01/24/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50494-A TRM Change Pkg WB-TS-00-007	Leading Edge Flow Measurements

Description and Safety Assessment:

DCN D-50494-A provides the design basis document and physical modifications required to implement the power increase that is obtained by installation and use of an approved LEFM system for the Feedwater supply to the steam generators.

The flow measurement system utilizes ultrasonic transducers placed in a section of the main feedwater pipe (a spoolpiece and electronics unit that was installed by DCN D-50451-A) and measures transient time of ultrasonic sound waves.

The LEFM measurement is very accurate and can substantially decrease the uncertainty associated with using existing venturi based flow measurement in the secondary side power calorimetrics to determine thermal output of the core. The existing measurement system consists of nozzle venturis placed in the feedwater lines to the individual steam generators. Originally, accuracies associated with reactor power measurement required a 2% uncertainty per 10CFR50 Appendix K. A revision to K rule by the NRC has been approved which will allow a power increase of 1.4% or 48 MWt, increasing allowable core power to 3459 MWt with an associated NSSS power of 3475 MWt. This increase provides an allowance of 0.6% instrument uncertainty associated with reactor power measurements.

The core power calculation, as determined by secondary side calorimetrics, will be made using the LEFM inputs of feedwater mass flow and temperature. Control of feedwater flow will be by the existing controls from the nozzle venturis. If the LEFM becomes unavailable for a duration that exceeds the conditions of the Technical Requirements Manual, TR 3.3.7 and Technical Specification SR 3.3.1.2, the secondary side calorimetrics, as performed with inputs from the nozzle venturis, will require a core power adjustment toward a lower core power based on the 2% uncertainty associated with the nozzle inaccuracies. This evaluation is applicable to the Technical Requirements Manual change TR 3.3.7.

1. When the LEFM is available, the plant should be operated in a manner consistent with the LEFM based calorimetric measurement and at 3459 MWt (100% Rated Thermal Power) (RTP).

2. If the LEFM is unavailable, the plant may be operated at 3459 MWt (100% RTP) using the nuclear instrumentation system (NIS) indication or normalized feedwater venturi-based calorimetric measurement until the next performance of SR 3.3.1.2 is due.

3. If the LEFM based calorimetric measurement is unavailable at the time SR 3.3.1.2 is due, the normalized feedwater venturi-based calorimetric measurement should be used for the performance of SR 3.3.1.2. However, to maintain consistency with the uncertainty analysis, the maximum allowable power should be reduced to 3411 MWt or 98.6% RTP. Either the NIS indication or the normalized feedwater venturi-based calorimetric power indication may be used to control the unit power.

Based on the review of the Technical Requirements Manual (TRM) change and the discussions above, it is concluded that the activity covered by this safety evaluation does not involve an unreviewed safety question.

SA-SE Number: DCN M-39771-A

Implementation Date: 2/05/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39771-A	Condensate Polishing Demineralizer System (CPDS) System Modification

Description and Safety Assessment:

DCN M-39771-A removes the conductivity elements associated with the CPDS high crud tanks (HCT) 1 and 2 and the neutralization tank. These conductivity elements are no longer required for system operation. The instrument loops associated with each conductivity element are also being removed. The root valve associated with the conductivity element on the neutralization tank is being changed to a normally closed position, and the root valves associated with the elements on the HCTs were removed on a previous modification that abandoned the elements in place.

The CPDS 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, sampling, or Technical Specification compliance. 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 are no design bases accidents affected by this DCN.

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 DCN.

The equipment being removed performs no safety function, and does not interface with any equipment important to safety. The proposed design change does not increase the probability of an accident or the occurrence of a malfunction of equipment important to safety. The consequences of an accident or a malfunction of equipment will not be increased. No accidents or malfunctions of a different type than previously evaluated in the UFSAR are created. The proposed design change does not affect any technical specification or margin of safety identified in the Technical Specification Bases. Therefore, this change does not involve any unreviewed safety question.

SA-SE Number: EDC E-50120-A

Implementation Date: 11/10/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50120-A UFSAR Change Pkg. # 1586	Integral Fuel Burnable Absorber Fuel Rods

Description and Safety Assessment:

WBN changed to the Vantage+/Performance+ (V+/P+) fuel assembly in Cycle 2 for Reload 1. This was a very comprehensive fuel change encompassing skeleton, clad material, and fuel rod/pellet changes. One feature of the V+/P+ product line which was not included for Reload 1 was the annular axial blanket pellets. Starting in Cycle 3, Reload 2, WBN cores use IFBA fuel rods with annular axial blanket pellets instead of solid axial blanket pellets. This change is the subject of EDC E-50120-A.

A significant fuel performance issue has developed for fuel rods manufactured by Westinghouse which incorporate the zirconium diboride IFBA. It has been determined by Westinghouse that under certain circumstances the IFBA fuel rods may exceed the fuel pellet to clad gap no reopening design basis criterion specified in the UFSAR. In addition, it is possible that these fuel rods may exceed the 10CFR50.46 criterion for 17% local oxidation following a design basis Loss of Coolant Accident (LOCA). These conditions are possible in IFBA rods due to high internal rod gas pressure. The internal gas pressure is much higher in IFBA rods than in non-IFBA rods because the zirconium diboride neutron poison generates helium gas during irradiation. Additional margin can be gained to both the gap no reopening and 17% local oxidation criteria by reducing the rod internal gas pressure in the IFBA rods. This pressure reduction can be achieved by incorporating a larger plenum volume within the fuel rod to accommodate helium gas generated from the zirconium diboride IFBA and other fission gases. The only practical way to increase this plenum volume is to reduce the UO_2 volume present within the fuel rods and this is accomplished by utilizing some fuel pellets which have an axial hole in the center of the pellets.

The use of annular pellets in the top and bottom six inch region in the IFBA rods provides additional plenum space to address the increased fuel rod internal pressure. The annular axial blanket pellets will not be used in the fuel rods without IFBA because the additional plenum space is not needed to meet the rod internal pressure criterion for the non-IFBA fuel rods. Axial blankets consist of reduced enrichment fuel pellets at each end of the fuel stack in each fuel rod. These blankets reduce neutron leakage from the core and improve uranium utilization.

Fuel assemblies with IFBA rods with annular axial blanket pellets can weigh approximately 15 to 20 pounds less than fuel assemblies with solid axial blanket pellets. This difference has inconsequential impact on fuel handling. (Note: a fuel assembly weighs approximately 1470 pounds). Additionally, fuel assemblies containing annular axial blanket pellets in the IFBA rods will be indistinguishable in operation from fuel assemblies containing solid pellets and will be transparent to the operators.

UFSAR Change Package 1586 has been prepared to revise the UFSAR to accommodate the use of annular axial pellets in IFBA rods. This UFSAR revision is to paragraph 4.2.1.2.1 and adds several sentences to describe the use and purpose of annular axial pellets.

- The use of the annular axial blanket pellets for IFBA fuel rods In V+/P+ fuel remains within the current fuel design bases. The manufacturing, material, and design features of the annular axial blanket pellet do not modify the current fuel rod functional requirements. No new credible failure modes are created by using annular pellets starting in Cycle 3 with Reload 2.

- The change to annular axial blanket pellets does not have any effect on the probability or consequences of previously analyzed accidents nor on the probability or consequences of a previously analyzed malfunction of equipment important to safety. No new or different accident or malfunction has been created by the use of annular axial blanket pellets in IFBA fuel rods in V+/P+ fuel.

- The WBN safety analyses (large and small break LOCA and Non-LOCA), have been generically evaluated. The only impact identified is an assessment of a 10°F peak clad temperature (PCT) penalty on the small break LOCA NOTRUMP-EM analysis to account for the addition of the annular pellets. This assessment will remain in effect until NRC approval of Westinghouse annular pellet heat conduction model. The small break LOCA PCT is still below the regulatory and WBN licensing basis limit of 2200°F and the 10°F annular pellet allocation will be included in the required 1OCFR 50.46 PCT report.

- Implementation of annular axial blanket pellets within IFBA fuel rods will provide additional margin to the following fuel rod design basis criteria: 1) The internal gas pressure of the lead rod (maximum internal pressure) in the reactor will be limited to a value below that which could cause the diametral gap to increase due to outward cladding creep during steady state operation, and 2) Maximum fuel rod cladding corrosion is limited to a maximum of 17% local oxidation following a design basis LOCA.

- The margin to these criteria is Increased by the additional plenum volume provided by the annulus in the axial blanket pellets. Increase of the plenum volume will provide additional space for the accumulation of fission gases during irradiation and will, therefore, reduce the rod internal gas pressure at end-of-life. The annular pellets will not be used in the fuel rods without IFBA because the additional plenum volume is not needed in these rods to increase the rod design basis margins.

- Fuel assemblies containing annular axial blanket pellets in the IFBA rods will be indistinguishable In operation from fuel assemblies containing solid pellets and will be transparent to the operators. The WBN Cycle 3 Reload design will consider the annular pellets to correctly account for their presence on the power distributions and peaking factors. The required 10CFR 50.46 PCT report will include the 10°F PCT assessment allocated for the annular pellets in the small break LOCA.

This change is the subject of EDC E-50120-A and does not constitute an unreviewed safety question.

SA-SE Number: EDC E-50455-A

Implementation Date: 11/231999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50455-A	Drawing Discrepancies Concerning Fire Dampers

Description and Safety Assessment:

Drawing Discrepancy (DD) 99-0079 identified discrepancies in TVA drawings 1-47W866-10 Rev. 26, 1-47W866-8 Rev. 19, and 47W92O-39 Rev. 16 pertaining to the phrase "abandoned in place" directed towards fire dampers 0-ISD-031-2425, -2427. -2428, 1 -ISD-031-2987, and 2-ISD-031-2990. The phrase inappropriately implies two things: 1) the fire dampers are no longer required to support Unit 1 operation, and 2) the fire dampers are no longer in required compartmentation fire barriers and are no longer required for periodic inspection. However, per inspection of these drawings, the fire dampers are clearly shown to be within the boundary of Unit 1, therefore, making them necessary to support Unit 1 operation. As a result, deleting the phrase "abandoned in place" will only clarify the proper status of these fire dampers, and not affect their functions in any way. Pertaining to Fire Protection, DCNs F-34313-A, M-05736-D, and F-28970-A were issued to "abandoned in place" the above mentioned fire dampers for the purpose of eliminating unnecessarily exposure time to the Fire Protection Unit during periodic inspection of these fire dampers. The phrase was used to indicate that these fire dampers were no longer required to be periodically inspected. Table 14.8.2 of the Fire Protection Report (Part 11) lists all the fire dampers required for periodic inspection, and per investigation of this table, it was found that dampers 0-ISD-031-2427, I -ISD-031-2987, and 2-ISD-031-2990 are among the ones required for inspection. They are also listed in Table 9.4 of System Description N3-30AB4001. Therefore, regarding these three particular fire dampers, the drawings are in contradiction with the Fire Protection Report and the System Description. By deleting the phrase, the drawings will then agree with the Fire Protection Report and the System Description regarding these fire dampers. As to dampers 0-ISD-031-2425 and -2428, DCN F-34313-A also stated that they are no longer in compartmentation fire barriers and are currently used for ventilation purposes. Although they are no longer in compartmentation fire barriers and no longer required for periodic inspection, these dampers are within the boundary of Unit 1, and therefore, necessary to support Unit 1 operation.

As a result, EDC E-50455-A is issued to delete the phrase "abandoned in place" from TVA drawings 1-47W866-10 Rev. 26, 1-47W866-8 Rev. 19, and 47W920-39 Rev. J.

These changes will not affect the design of the plant, physically modify any equipment in the plant, or affect how the plant is operated. The proposed changes do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR. A possibility of an accident or malfunction of a different type than evaluated previously in the UFSAR is not created. These changes will not reduce any margin of safety as defined in the basis for any Technical Specifications. Therefore, the changes do not constitute an unreviewed safety question.

SA-SE Number: FIRE PROTECTION REPORT

Implementation Date: 11/01/1999

Document Type:	***Affected Documents:***	***Title:***
Fire Protection	Fire Protection Report Revision 12	Fire Protection Report (FPR), Revision 12

Description and Safety Assessment:

This Safety Evaluation is provided for revision 12 to the FPR. The changes include:

1. Adding the core empty period to the fire watch requirements of Mode 5 and 6.
2. Expanding the use of constantly manned positions to provide fire watch coverage,
3. Correcting the nitrogen pressure requirements for the Appendix R manual action stations in accordance with the associated system descriptions,
4. Providing consistent placement of fire hose during outages,
5. Providing an evaluation of the existing design for the fusible link configuration to release sliding fire doors,
6. Changing fire barrier requirements as evaluated in DCN D-50182, and
7. Other non-intent changes.

The changes deal with the actions to prevent and mitigate a fire. The changes do not provide a different accident or failure than those already evaluated. The changes only reflect a re-defining of the methods to satisfy the design basis.

The "core empty" period presents hazards similar to Modes 5 and 6 thus this change will clarify the acceptability to allow the same fire watch requirements for Modes 5 and 6 and "core empty".

Personnel at a constantly manned location are uniquely qualified to provide the equivalent observation capabilities of a fire watch for the specified location. The personnel will continue to be able to perform their assigned functions. The reporting of a fire has always been part of the responsibilities of any plant employee and involves a minimal amount of time.

The nitrogen pressure change reduces the requirement for the steam generator PORVs to the documented information specified by the associated system description and not the higher nitrogen pressure requirement of the AFW level control valves (LCVs). Thus the steam generator PORVs can still provide the intended valve operations for an Appendix R fire event but not have an excessive maintenance requirement of the higher pressure.

The placement of fire hose at the entrance to lower containment during refueling and non-refueling outages permits the consistent use of this hose during times of comparable hazards (i.e., outages).

The evaluation for fusible link configuration for releasing of sliding fire doors only documents the plant's design since fuel load and does not change anything in the plant.

The fire barrier changes are addressed by the safety evaluation for DCN E-50182-B.

The non-intent changes include such items as adding a table of contents for Part VI, correcting symbols, and revising/updating the Table of Contents and Effective Page Listing.

For these reasons the changes of the FPR, Rev. 12 do not constitute an unreviewed safety question, does not reduce the margin of safety, and is safe from a nuclear safety standpoint.

SA-SE Number: GO-6 R13 OTO-1

Implementation Date: 09/10/2000

Document Type:	***Affected Documents:***	***Title:***
Procedure Change	GO-6 R13 OTO-1	Additional Condenser Dump Valves to maintain Cooldown Rate

Description and Safety Assessment:

At present, the Watts Bar steam dump logic will block all of 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. Based on the Watts Bar Simulator, it appears that the three cooldown valves would have insufficient steam flow capacity to maintain the administrative cooldown limit of 60°F/hr below approximately 285°F. It is being proposed that sometime during the cooldown phase, the P-12 interlock would be disabled to allow the use of additional condenser dump valves to aid in maintaining a cooldown rate of 60°F/hr.

General Operating Procedure (GO)-6 and Work Order 00-011299-000 provide for the disabling of the P-12 interlock which thereby provides an alternate method of using additional condenser dump valves for unit cooldown. 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 under specified conditions to aid in the cooldown of the RCS during unit cooldown following Unit 1 Cycle 3 operation.

There are no unreviewed safety questions associated with the use of additional condenser dump valves for unit cooldown when operated in accordance with the requirements and limits delineated within this safety evaluation. The technical specifications are not affected by this change. An evaluation was performed to assess the cooldown potential following failure of the steam dump controller after placing additional condenser dump valves in service with and without additional decay heat removal systems in operation. It was determined that the cooldown limit of 100°F/hr would not be exceeded due to a failure with the additional condenser dump valves opening at or below the permitted temperatures. No new equipment failure modes are malfunctions are created by this procedurally controlled temporary alteration.

SA-SE Number: ODCM Rev. 7

Implementation Date: 03/08/2001

Document Type:	***Affected Documents:***	***Title:***
Procedure Change	ODCM Rev. 7	Offsite Dose Calculation Manual - Revised Sampling Locations

Description and Safety Assessment:

The Offsite Dose Calculation Manual (ODCM), Revision 7 includes the following changes: Section 9.0, "Radiological Environmental Monitoring Program", changed sampling locations for milk, and an air sampling station. The changes in milk sampling locations are necessary due to the termination of milk production at two farms that were being sampled. These locations are Farm C at 16 miles SSW and Farm S at 19.5 miles SW. A new farm EH at 24 miles SSW has been added as, a second control. Replacement of the 2 farms with a single farm continues to meet regulatory sampling requirements.

The electrical substation at air sampling location station PM-5 located 6.2 miles S is being taken out of operation and a new substation has been constructed 8.0 miles S. The PM-5 air sampling station will be relocated to the new substation.

Revised the wording to be consistent with the guidance in NUREG-1301 to provide for the collection of vegetation from locations meeting the criteria for milk sampling when the collection of a milk sample is not possible from that location.

Eliminated Sr-89 and Sr-90 analyses on vegetation, drinking and ground water since NUREG-1301 does not require these analysis after 3 years of operation.

Since this change will not affect the operation of any plant equipment and that Radiological Environmental Monitoring Program (REMP) verifies the impact of routine radioactive effluent releases to the environment, there is not unreviewed safety question.

SA-SE Number: REMIC WBN2000-02

Implementation Date: 10/24/2000

Document Type:	***Affected Documents:***	***Title:***
Other	Meteorological Monitoring Equipment Change for Wind Direction/Speed Sensors	Installation of a New Ultrasonic Wind Sensor.

Description and Safety Assessment:

This change involves installing a new ultrasonic wind sensor (Vaisala model 425AH) at 91-, 46-, and 10-meters on the meteorological tower. The new sensor is a replacement for the current wind direction and wind speed sensors (Climet model 012-10 and Climet model 011 -1). The current sensors are no longer manufactured and spare parts are difficult to obtain, and there is a recent history of equipment problems. The change does not effect any safety related equipment, does not alter the data collected, does not reduce the margin of safety defined in any bases, and does not cause any requirements to be exceeded. Therefore, this equipment change does not constitute a unreviewed safety question.

SA-SE Number: TACF 1-00-11-244-0

Implementation Date: *11/15/2000*

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-00-11-244 Rev. 0, ARI-1-7, 1-PI-OPS-1-MCR, SOI-47.02	Reduce the Potential for a Unit trip caused by inadvertent actuation of Main Generator System Gound Fault Protection Relay

Description and Safety Assessment:

This temporary alteration (TACF) defeats the alarm and trip function associated with the 100% Generator System Ground Fault protection on the Unit 1 Main Generator. The purpose of this TACF is to reduce the potential for a spurious Unit trip caused by inadvertent actuation of the 100% Generator System Ground Fault protection relay. This relay was installed by DCN D-50341-A during Refueling Outage 03 and is unique to WBN and TVA. It is prudent to gain some operating experience and build confidence in the relay before activating the alarm and trip function and risk a spurious trip of the Unit. The control room indicator is not affected by this TACF and remains available for use by Operations.

This 100% Generator System Ground Fault senses stator ground faults via generator neutral overvoltages. The device utilizes a microprocessor based system to analyze voltages and provide an output signal. The device senses generator neutral over-voltages (60 hertz) and third harmonic (180 hertz) under-voltages. By monitoring both of these voltages, the protective device can provide detection of generator system ground faults for 100% of the stator winding. The 100% Generator System Ground Fault is designed to trip the main generator and turbine. If Unit power level is greater than 50%, the turbine trip will initiate a reactor trip.

A generator ground fault is the result of a degradation of stator winding (coil) insulation resistance. This degradation over time leads to a complete breakdown of the insulating quality and a fault to ground or to another phase conductor. This degradation can be the result of internal vibration induced rubbing that reduces insulation wall thickness, over-heating the insulation, or moisture intrusion. If the degradation is allowed to continue, the insulation will weaken to the point of arc-over with resulting damage due to high current. The Control Room indicator is not affected by this TACF and the indicator remains available for use by Operations

The WBN Unit 1 Main Generator underwent a major overhaul/inspection during Refueling Outage 03. The generator inspections did not reveal any discrepancies that would indicate abnormal insulation degradation.

All scenarios involving a generator failure (catastrophic or otherwise) have been encompassed by previously analyzed accidents or the design of equipment important to safety. There is not an increased likelihood of a failure of equipment important to safety to occur due to this TACF. The risk of equipment damage is one of economics rather than nuclear safety. By minimizing the potential for Unit trips due to spurious 100% Generator System Ground Fault device actuation WBN is accepting a small risk of more severe damage to the Unit 1 Main Generator. No new or different accidents are introduced and no previously analyzed accidents are affected. Implementation of this TACF does not increase the likelihood that safety related safe shutdown equipment will be challenged to shutdown the reactor. The technical specifications and safety margins are not affected. This TACF does not constitute an unreviewed safety question.

SA-SE Number: TACF 1-00-3-246-2

Implementation Date: 5/5/2000

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-00-3-246 Rev.2 Procedure ARI-1-2 Procedure ARI-71-75	Generator/Turbine Trip Function

Description and Safety Assessment:

This TACF defeats the generator/turbine trip function associated with the sudden pressure device on the Unit 1 Main Bank and Unit Station Service Transformers (USSTs). The purpose of this TACF is to reduce the potential for a spurious Unit trip caused by inadvertent actuation of the transformer sudden pressure device. There have been industry events associated with spurious actuation of sudden pressure devices. A design change is in process to install a two-out-of-three logic to the sudden pressure devices. This TACF will remain in place until the design change is implemented. The transformer sudden pressure device is designed to respond to the sudden increase in gas pressure in a power transformer which would be caused by an internal arc.

At WBN the sudden pressure device actuation initiates two distinct functions: 1) Trip the main generator and turbine, and 2) Actuate transformer fire protection. If Unit power level is greater than 50%, the turbine trip will initiate a reactor trip. The trip function is the only function affected by this TACF. This TACF does not affect the fire protection actuation circuitry.

The loss of external load and/or turbine trip is the direct result of a transformer fault. It is possible but not likely that collateral damage could occur to the reactor coolant pump (RCP) Boards resulting a partial or complete loss of forced reactor coolant flow. The RCP Boards are located approximately 100 feet away from the Unit 1 Main Bank and USSTs. It is also possible but not likely that collateral damage could cause damage to common station service transformer (CSST) C and D resulting in a loss of offsite power. These transformers are located approximately 300 feet from the Unit 1 Main Bank and USSTs. Fire suppression actuation functions are not affected by this TACF. Also, the Unit 2 transformers are directly adjacent to the Unit 1 transformers and are between the Unit 1 Main Bank/USSTs and the RCP Boards and CSST C and D. The Unit 2 transformers essentially provide shielding to prevent any direct damage to the RCP boards or CSST C and D. Other protective relays are unaffected by this TACF and will remain functional to clear a transformer fault with the sudden pressure relay trip function disabled. There are also generator trips that could be initiated if the transformer fault is felt at the generator. Protective trip relays in the 500KV switchyard and the transformer yard are designed to protect the equipment and these relays are unaffected by this TACF. In the event of a catastrophic transformer failure, there could be collateral damage to adjacent transformers due to explosion and fire. However, all possible scenarios of direct or collateral damage that could extend to safety related equipment have been previously analyzed. There are no different accident analyses required by this TACF.

Industry experience has indicated that the potential for such a spurious trip does exist. Installing this TACF will reduce the potential for challenges to the safe shutdown systems.

Implementation of this TACF does not increase the likelihood that safety related safe shutdown equipment will be challenged to shutdown the reactor. The technical specifications and safety margins are not affected. This TA does not constitute a unreviewed safety question.

SA-SE Number: TI-100.006-3

Implementation Date: 06/16/2000

Document Type:	***Affected Documents:***	***Title:***
Procedure Change	TI-100.006, Rev. 3	Incorporation of Approved Relief Requests PV-15 R1 and PV-16

Description and Safety Assessment:

Revision 3 to Technical Instructions (TI)-100.006, Inservice Testing (IST) Program, incorporates a revision to the existing request for relief, PV-15, and a new request for relief, PV-16. Both relief requests have been reviewed and accepted by the NRC.

PV-15, Revision 1, does not alter the testing methodology currently being used for the RCS head vent valves, but instead gathers together the various tests, calibrations, and preventative maintenance (PM) activities already being performed and proposes them as an "enhanced maintenance program", as discussed in NUREG-1482, Section 4.2.8. This "enhanced maintenance program" then supplies the basis for not periodically replacing the valves as committed to in Revision 0 of PV-15. The RCS head vent throttle valves are utilized in all accident scenarios that involve the generation of noncondensable gases inside the RCS.

PV-16 alters the periodic IST requirements for the cooling water inlet valves to the Auxiliary Air Compressors. These valves were originally included in PV-15, along with the RCS Head Vent Valves, and a program of periodic replacement was imposed in lieu of stroke timing. Revision 1 to PV-15 restricts PV-15 to the RCS Head Vent Valves only. PV16 was prepared to address the IST requirements for the cooling water inlet valves to the Auxiliary Air Compressors. PV-16 accepts the periodic testing of the Auxiliary Air Compressors as demonstration that the cooling water inlet valves are functioning adequately to be considered operationally ready. PV-16 does not utilize or rely upon periodic replacement of the valves as was originally planned in PV-15. The cooling water inlet valves to the Auxiliary Air Compressors are utilized in all accident scenarios that involve potential loss of the station service air compressors, such as loss of off-site power, station blackout, and seismic events.

Creditable failure modes for this change is for the proposed alternative testing to fail to maintain the valves in an operable condition.

These two relief requests were docketed with, reviewed by, and approved for use at Watts Bar for the remainder of the first Inservice Interval by the NRC via a safety evaluation transmitted to Watts Bar by an NRC Letter dated March 22, 2000, Richard P. Correia to Mr. J. A. Scalice which included the NRC safety evaluation as an attachment.

SA-SE Number: TI-100.006-4

Implementation Date: 09/01/2000

Document Type:	***Affected Documents:***	***Title:***
Procedure Change	TI-100.006 Rev. 4	Incorporation of Approved Relief Requests, PV-10, PV-13 and PV-17

Description and Safety Assessment:

Operating and Maintenance (OM)-10, which is invoked by American Society of Mechanical Engineers (ASME) Section XI for IST of valves, provides an alternative to the testing of check valves. This alternative [paragraph 4.3.2.4(c)] allows the check valve to be disassembled and inspected during refueling outages. Prior to Revision 4 of TI-100.006, the ASME Section XI IST required certain Essential Raw Cooling Water (ERCW) and Containment Spray system check valves to be disassembled and inspected during refueling outages in lieu of testing. Revision 4 implements revisions to Relief Requests PV-10 and PV-13, and new Relief Request PV-17, which alter the requirements of OM-10 to allow disassembly and inspection of these valves during unit operation, in conjunction with other maintenance scheduled on the affected components. No other intent changes are made by Revision 4. Revision 4 also makes an editorial correction in the UNID of the 1 B-B AFW pump.

For the containment spray check valves, the DBAs involved are any accident that releases energy into containment sufficient to cause a significant increase in containment pressure.

The ERCW check valves affected by this revision are in the cooling water supply line to the emergency diesel generators. Therefore the DBA involved is the loss of off sight power.

Since the only change is in the performance mode of the disassembly and inspection, and since the disassembly and inspection is to be accomplished within the existing time frames for the associated Technical Specification Limiting Conditions for Operation (LCO) and without causing increased system unavailability, no credible failure modes have been identified.

Since these changes are different than the provisions given in OM-10, invoked by ASME Section XI, for the inservice testing of check valves, they have been docketed with, and reviewed and approved by the NRC as provided in 10CFR50.55a. Since the NRC has reviewed and concurred with the specific changes, and has issued an safety evaluation documenting their review and concurrence, no unreviewed safety question is involved.

SA-SE Number: TI-100.006-5

Implementation Date: 09/27/2000

Document Type:	***Affected Documents:***	***Title:***
Procedure Change	TI-100.006 Rev. 5	Deletion of Two Abandoned Valves from the ASME Section XI Inservice for PD Pump

Description and Safety Assessment:

Revision 5 to TI-100.006 implements an impact from DCN D-50506-A. This DCN abandoned the reciprocating charging pump and its associated piping and valves in place. Two of the valves abandoned by this DCN were included in the ASME Section XI IST Program. This revision deletes the two abandoned valves from the ASME Section XI IST Program.

No DBA is involved in deleting the testing requirements for abandoned valves.

No credible failure mode is involved in deleting the testing requirements for abandoned valves.

The valves affected by this revision are relief valves that provided overpressure protection to the ASME Code Class 2 suction and discharge piping to the positive displacement (PD) charging pump. This pump itself did not provide an active, safety related function. However, when it was in service it was connected to safety related piping for which these relief valves provided overpressure protection. Since the pump and the associated piping have been isolated and abandoned in place, there is no longer a need to protect the piping. In fact, the DCN that abandons the positive displacement charging pump also isolates and abandons the piping containing these relief valves. Deletion of testing requirements for a relief valve that no longer performs a safety function does not constitute an unreviewed safety question.

SA-SE Number: TI-100.009-1

Implementation Date: 03/01/2000

Document Type:	***Affected Documents:***	***Title:***
Procedure Change	TI-100.009 Revision 1	Inservice Inspection Program - Pressure Test

Description and Safety Assessment:

Revision 1 to TI-100.009 updates and reformats Appendix B, Tentative Pressure Test Schedule. Revisions made to the table include: [1] reformatting to provide a column that identifies which specific outage within a given inspection period each pressure test will be performed; [2] revision of boundary descriptions to provide a clear relationship between the several implementing instructions and Appendix B; [3] replacing the use of Code Case N-522 with performance of a standard Inservice or Functional System Pressure Test for certain specific containment penetrations, [4] changing the specified test types from Functional to Inservice or from Inservice to Functional for certain portions of some systems, and [5] rescheduling the once per 10-year hydrostatic test alternatives.

There are no DBAs involved in the performance of ASME Section XI System Pressure Tests.

The only credible failure mode identified is the failure to comply with the requirements of the ASME Section X1 System Pressure Test Program. Since this revision has been reviewed for compliance with the requirements of ASME Section X1, any failure to comply would be the result of failure to comply with the requirements of TI-100.009 itself. Revision to TI-100.009 is not considered a reasonable basis for inducing failure to comply with its requirements.

The NRC has reviewed and approved the relief requests contained in Appendix A, and has reviewed and accepted the System Pressure Test Program description contained in the main body of TI-100.009. This review, approval, and acceptance is documented in the safety evaluation issued for the WBN System Pressure Testing Program. This revision to Appendix B of TI-100.009 is in compliance with the requirements of ASME Section XI and the NRC approved requests for relief contained in Appendix A of TI-100.009 for the performance and conduct of system pressure tests. Additionally it is in compliance with the System Pressure Test Program description contained the main body of TI-100.004. Since this revision is in compliance with the NRC reviewed, approved and accepted System Pressure Test Program, it does not involve an unreviewed safety question.

SA-SE Number: TI-100.009-2

Implementation Date:09/27/2000

Document Type:	***Affected Documents:***	***Title:***
Procedure Change	TI-100.009, Revision 2	Incorporation of Approved Relief Request ISPT-08

Description and Safety Assessment:

Revision 2 to TI-100.009 incorporates the recently NRC approved Request for Relief, ISPT-08. This Request for Relief approves the deletion of the requirement to remove insulation from certain bolted connections when performing the Visual Test (VT)-2 visual examination, and is an alternative to the provisions of Code Case N-533 which allows removal of the insulation and inspection while the system is depressurized. No changes, other than the incorporation of this NRC approved Request for Relief are made by Revision 2.

The involved DBA is a LOCA, in that a LOCA could be initiated by a catastrophic failure of one of the bolted connections involved in this Request for Relief.

The credible failure mode considered for the activity is that the bolting involved in the bolted connections could suffer failures that would allow the system to successfully pass a VT-2 inspection with the insulation installed, but cause a subsequent failure of the joint.

The NRC has reviewed and approved Request for Relief ISPT-08, which is being implemented by Revision 2 to TI-100.009. This review, approval, and acceptance is documented in an NRC safety evaluation attached to a letter from Richard P. Correia to J. A. Scalice, dated September 7, 2000. Since this revision is in compliance with the NRC reviewed, approved and accepted Request for Relief ISPT-08, it does not involve an unreviewed safety question.

SA-SE Number: TI-100.012-0

Implementation Date:04/07/2000

Document Type:	***Affected Documents:***	***Title:***
UFSAR	UFSAR Pkg 1602 SI-88-4, R3 TI-100.012, R0	ASME Section XI IWE and IWL Requirements

Description and Safety Assessment:

UFSAR Change Package No 1602, new procedure TI-100.012, Revision 0, "ASME Section XI Containment Inservice Inspection (CISI) Program," and revised procedure 1-IS-88-4, "40 Month General Visual Inspection of Steel Containment Vessel" implement 10 CFR 50.55a requirements to perform containment inservice inspections in accordance with ASME Section XI. The NRC amended 10 CFR 50.55a requiring utilities to incorporate the inservice inspection requirements of ASME Section XI, Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light Water Cooled Power Plants, " and Subsection IWL, "Requirements for Class CC Concrete Components of Light Water Cooled Power Plants." The NRC amended its' regulations in the Federal Register, Volume 61, Number 154 dated Thursday, August 8, 1996. The amended regulation requires implementation of the containment inspection requirements by September 2001.

The purpose of the amended regulation is to incorporate the Subsection IWE and IWL requirements to assure that the critical areas of containments are routinely inspected to detect and take corrective action for defects that could compromise a containment's structural integrity. These amended requirements are in addition to the currently performed Appendix J requirements and does not replace any requirements currently performed. None of the existing requirements provide specific guidance on how to perform the necessary containment examinations. Incorporation of ASME Section XI will provide specific guidance for performance of containment inspections. The inspection requirements include data collection in the form of visual and volumetric non-destructive examinations, evaluation, and acceptance of Code Class MC pressure retaining components and their integral attachments and metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments.

The UFSAR Change Package documents the containment inservice inspection requirements mandated by the Code of Federal Regulations. Technical Instruction TI-100.012, Revision 0, is the site specific CISI program which implements the Section XI requirements. In addition to the standard site procedure requirements, this TI has been prepared in accordance with 10 CFR 50.55a(b)(2)(ix), 10 CFR 50.55a(g)(4), 10 CFR 50.55a(g)(6)(ii)(B), Standard Programs and Processes SPP-9.1, "ASME Section XI," Part E, "ASME Section XI Containment Inservice and Augmented Inspection Examinations," and the applicable portions of ASME Section XI. It is noted that the TI contains five Requests for Relief which provide alternatives to the ASME Section XI requirements but which have been approved for use at Watts Bar in the NRC's Safety Evaluation issued on November 24, 1999. The revision to I-SI-88-4 adds the 10 CFR 50.55a and specific ASME Section XI General Visual Examination requirements.

There are no DBAs involved in the performance of the ASME Section XI containment inspections.

This UFSAR change, new procedure and revised procedure does not constitute a unreviewed safety question as noted in the previous evaluation of effects. The incorporation of the ASME Section XI containment inservice inspection requirements is mandated by the amended Code of Federal Regulations. By nature of the regulatory approval process, these requirements have been reviewed by the NRC staff and determined not to precipitate unsafe conditions or practices and are required to be implemented. The inspection requirements include data collection in the form of visual and volumetric non-destructive examinations, evaluation and acceptance as defined by ASME Section XI. These inspection requirements are in addition to the existing Appendix J requirements and do not delete or replace any existing requirements. As noted, TI-100.012 contains Requests for Relief which provide alternatives to the ASME Section XI requirements but which have been approved for use at Watts Bar in the NRC's safety evaluation issued on November 24, 1999.

SA-SE Number: WBLMGR-99-002-0

Implementation Date: 03/13/2000

Document Type:	***Affected Documents:***	***Title:***
Technical Specification	TS Bases Revision 29 TRM Revision 21	Operability Limits Include Instrument Error

Description and Safety Assessment:

The revision to the Technical Specification and TRM Bases addressed by this evaluation is an administrative change. An editorial correction is being made along with a revision to add a note similar to "value does not account for instrument error" beside various level, pressure, temperature, or flow parameters identified in the Technical Specification and TRM Bases. This note will act as a reminder to plant personnel that other documents, such as setpoint and scaling documents, drawings, or procedures, should be reviewed. From these documents the involved personnel will establish the value that, when read from appropriate compliance instruments, will ensure compliance with the Technical Specification or TRM operability limit.

The Bases revision will correct and enhance the information available to the plant operators, and in no way will affect or degrade the operation of any plant structure, system, or component (SSC). This is evident since no changes to established plant parameters are being made but additional guidance, in the form of the drawings and the notes, are being provided to the plant staff. Therefore, this administrative enhancement to the Technical Specification and TRM Bases does not:

- increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR,
- increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR,
- increase the consequences of an accident previously evaluated in the UFSAR,
- increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR, create a possibility for an accident of a different type than any evaluated previously in the UFSAR,
- create a possibility for a malfunction of a different type than any evaluated previously in the UFSAR,
- increase the radiological consequences of plant operation, or
- reduce the margin of safety as defined in the basis for any technical specification.

Considering this, the proposed Bases revisions do not constitute a unreviewed safety question.

SA-SE Number: WBN Reload Cycle 3 Rev. 2

Implementation Date: 01/18/2000

Document Type:	***Affected Documents:***	***Title:***
Analysis	Watts Bar Unit 1 Cycle 3 Core Reload and Operation	Cycle 3 Reload Analysis Revision 2

Description and Safety Assessment:

The Reload evaluation considers reactor core reload and operation for cycle 3 operation in all modes to a maximum cycle core average burnup of 20,500 MWd/MTU, including a power coastdown.

Revision 2 justifies the measurement and evaluation of the transient heat flux hot channel factor, $F^W_Q(Z)$, in a portion of the core which is not currently required to be measured by the Technical Specification or TS bases. This revision is a corrective action to PER 99-000051-000 and will serve as a model for future reload evaluations if the issue has to again be addressed.

Surveillance Requirement 3.2.1.2 requires the periodic measurement and evaluation of the transient heat flux hot channel factor, $F^W_Q(Z)$. The associated TS bases states that the top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

The reload design for WBN Cycle 3 resulted in the steady state $F_Q(Z)$ predicted to occur within the lower excluded region for a short time early in cycle life. Cycle 3 is currently beyond this time in life. Because $F_Q(Z)$ was predicted to occur in the lower excluded region, it was decided to conservatively exclude only the top and bottom 10% instead of the normal 15%. To support this core surveillance, the W(Z) values were generated to exclude the top and bottom 10% of the core. The additional monitoring insured that the most limiting $F_Q(Z)$ would be measured and action taken, if necessary, to prevent violation of $F_Q(Z)$, a parameter assumed as an initial condition in the accident analyses. All measured $F_Q(Z)$ values have been within specifications during Cycle 3 and no corrective actions have been necessary.

PER 99-015948-000 addressed the issue that the TS bases were not updated to reflect the need to monitor $F^W_Q(Z)$ in the expanded range early in life. The TS bases will not be revised for the remainder of Cycle 3. If the condition recurs in future reloads, the TS bases will be modified to reflect the appropriate monitoring needs. The changes to be made for Cycle 3 do not constitute an unreviewed safety question because:

Evaluations have been made of the effect of the core configuration specified for Cycle 3 upon the safety analyses described in the UFSAR. These evaluations included consideration of the mechanical design of the new fuel assemblies, nuclear and thermal-hydraulic design of the Cycle 3 core, and effects of the Cycle 3 core upon the LOCA and non-LOCA accidents discussed in the UFSAR. The implementation of the annular pellets results in a small break LOCA PCT assessment of +10 °F. Since all 10 CFR50.46 criteria, including significant margin to the regulatory limit of 2200 °F, continue to be satisfied, the margin of safety as defined in the Bases to the Technical Specifications is not reduced. Further, since the sum of the absolute values of all small break LOCA PCT assessments remains below 50°F, a schedule for reanalysis is not required. All conclusions presented in UFSAR were found to remain valid and no new credible failure modes have been created for the Cycle 3 reload.

The remaining two fuel design changes were made to improve quality of manufacturing and to increase resistance to snagging during fuel handling. Evaluations were performed which determined these changes did not affect the fuel assembly form, fit, or function.

There are no unreviewed safely questions or Technical Specifications changes identified as a result of the Watts Bar Unit 1, Cycle 3 core design. Therefore, the Cycle 3 reload design is licensable under 10 CFR 50.59, and requires no prior USNRC approval.

SA-SE Number: WBN Reload Cycle 3 Rev. 3

Implementation Date: 03/30/2000

Document Type:	***Affected Documents:***	***Title:***
Analysis	Watts Bar Unit 1 Cycle 3 Core Reload and Operation	Cycle 3 Reload Analysis Revision 3

Description and Safety Assessment:

The Reload Analysis considers reactor core reload and operation for Cycle 3 operation in all modes to a maximum cycle core average burnup of 20,500 MWd/MTU, including a power coastdown.

Revision 3 of the Reload Analysis relaxes the end-of-life (EOL) MTC limits in accordance with NRC approved TS. In addition, the BOL/ARO/HZP-MTC "as-measured limit" terminology specified is revised to "as-measured criterion." The Cycle 3 reload safety analysis was performed consistent with both the original EOL MTC limits and the relaxed limits to permit mid-cycle implementation of the relaxed limits after NRC approval was obtained.

Revision 3 of the COLR includes:

The EOL/ARO/RTP-MTC limit is revised to -4.50 x 10^{-4} $\Delta k/k/°F$ in accordance with TS Change 98-005 which was approved by NRC on March 14, 2000.

- The 300 ppm surveillance limit is revised to -3.75 x 10^{-4} $\Delta k/k/°F$.

- The 60 ppm surveillance limit is revised to -4.28 x 10^{-4} $\Delta k/k/°F$.

- The "as-measured limit" terminology is revised to "as-measured criterion" to be consistent with TS Bases Change 2000-01.

Evaluations have been made of the effect of the core configuration specified for Cycle 3 upon the safety analyses described in the UFSAR. These evaluations included consideration of the mechanical design of the new fuel assemblies, nuclear and thermal-hydraulic design of the Cycle 3 core, and effects of the Cycle 3 core upon the LOCA and non-LOCA accidents discussed in the UFSAR. The implementation of the annular pellets results in a small break LOCA PCT assessment of +10 °F. Since all 10 CFR50.46 criteria, including significant margin to the regulatory limit of 2200°F, continue to be satisfied, the margin of safety as defined in the Bases to the Technical Specifications is not reduced. Further, since the sum of the absolute values of all small break LOCA PCT assessments remains below 50° F, a schedule for reanalysis is not required. All conclusions presented in the UFSAR were found to remain valid and no new credible failure modes have been created for the Cycle 3 reload.

The remaining two fuel design changes were made to improve quality of manufacturing and to increase resistance to snagging during fuel handling. Evaluations were performed which determined these changes did not affect the fuel assembly form, fit, or function.

Based upon the preceding information and the following:
1) an End-of-Cycle (EOC) 2 burnup between 17,281 and 18,431 MWd/MTU (actual EOC 2 burnup was 18,066 MWd/MTU),
2) termination of Cycle 3 burnup at or before 20,500 MWd/MTU, including a power coastdown, and
3) adherence to plant protective and operating limitations given in the Technical Specifications and the Core Operating Limit Report (COLR),

There are no unreviewed safety questions or Technical Specifications changes identified as a result of the Watts Bar Unit 1, Cycle 3 core design. Therefore, the Cycle 3 Revision 3 reload design is licensable under 10 CFR 50.59, and requires no prior NRC approval.

SA-SE Number: WBN Reload Cycle 4

Implementation Date: 09/17/2000

Document Type:	***Affected Documents:***	***Title:***
Analysis	WBN U1 Cycle 4 Core Reload COLR, Revisions TS Bases Change TS-00-013	Cycle 4 Reload Analysis

Description and Safety Assessment:

This screening review/safety evaluation considers reactor core reload and operation for Cycle 4 operation in all modes to a maximum cycle core average burnup of 21,327 MWd/MTU, including a power coastdown.

Changes to be made for Cycle 4 include:

- Revised core configuration - 109 burned fuel assemblies will be discharged and replaced with 76 fresh Westinghouse Vantage +/Performance + (V +/P +) fuel assemblies, 33 Westinghouse Vantage 5H fuel assemblies discharged from previous cycles, and the remaining burned fuel assemblies will be shuffled. Fuel inserts including secondary sources, rod cluster control assemblies (RCCAs), and plugging devices will also be shuffled. Wet Annular Burnable Absorbers (WABA) will be utilized in selected core locations where discrete absorbers are required

- Revised COLR - The following changes will be made to the COLR:

    1. As-measured MTC criterion is changed to -0.30×10^{-5} $\Delta k/k/°F$ from -0.19×10^{-4} $\Delta k/k/°F$.
    2. CFQ is revised from 2.40 to 2.50 to reflect the new Best Estimate (BE) LOCA analysis implemented for Cycle 4.
    3. $F_{\Delta H}^{RTP}$ is revised from 1.60 to 1.65 to reflect the new BE LOCA analysis implemented for Cycle 4.
    4. The control bank insertion limits are revised to remove the more restrictive limits at reduced power that were required for Cycle 3.
    5. Table A.1 is revised to reflect the Cycle 4 specific required $F_Q^W(Z)$ penalty factor increases that are greater than 1.0200.
    6. K(z) function is revised to reflect the new BE LOCA analysis implemented for Cycle 4.
    7. AFD limits are revised.
    8. New values of W(Z) are provided. The W(Z) values are reload core specific and are changed every cycle. The Cycle 4 W(Z) values were calculated with the revised axial flux difference (AFD) limits noted above.

Changes to Fuel Assembly Design - The following changes will be made to the fuel assembly design of the fuel to be loaded in Cycle 4:

1. The IFBA B_{10} loading in the IFBA fuel rods is reduced from the loading used in previous fuel assemblies. The B_{10} loading for the Region 6 fuel assemblies is 2.198 mg/inch (1.4X) as compared to the 2.355 mg/inch (1.5X) loading used in the Region 4 and 5 fuel assemblies. The B_{10} loading was reduced to provide additional margin to the rod internal pressure limits.
2. The axial blanket enrichment is increased to 3.2 w% U_{235} from the 2.6 w% U_{235} used for the Region 4 and 5 fuel assemblies. The IFBA fuel rods contain annular pellets in the axial blanket region and solid axial blanket pellets in all other fuel rods similar to the Region 5 fuel assemblies.

3. Manufacturing process changes to the top nozzle: The top nozzle for the Region 6 fuel assemblies is now a cast design that does not include a plug in the instrument tube location. In addition, a new spring clamp is utilized that uses welds to retain the clamp to the nozzle, instead of a screw. The process changes reduce the number of components required to fabricate and assemble the top nozzle

4. Bead blasted Inconel-600 top nozzle spring screws: The top nozzle holddown spring screws are bead blasted in the thread to shank transition area to increase resistance to primary water stress corrosion cracking. This improvement is intended to prevent the fracturing of the screws that has been seen in burned fuel assemblies at WBN and other Westinghouse reactors.

5. Westinghouse Replacement Reconstitutable Top Nozzles (RRTN) are used on 33 burnt fuel assemblies. The original nozzles were replaced with the RRTNs on the 33 assemblies after fractures were found in their holddown spring screws. The RRTNs differ from the standard Vantage + nozzles in the following ways:

   - The groove profile of the top nozzle in the thimble hole has been elongated to accommodate variability in the nozzle manufacture, skeleton manufacture, and differential growth on irradiated thimbles.
   - The lead-in chamfer angle to the thimble hole bore is decreased to reduce the force required to install the nozzle.
   - The thimble hole bore diameter is increased slightly in order to reduce the force required to install the nozzle.
   - The top nozzle instrumentation plug is eliminated.

Change to the Technical Specification Bases - Technical Specification Bases Section SR 3.2.1.2 will be revised to change the core top and bottom 15% F_q exclusion region description to 10%. This change will ensure that anticipated Cycle 4 peak core powers will be measured and monitored.

The changes to be made for Cycle 4 do not constitute an unreviewed safety question because:

- Evaluations have been made of the effect of the core configuration specified for Cycle 4 upon the safety analyses described in the UFSAR. These evaluations included consideration of the mechanical design of the new fuel assemblies, nuclear and thermal-hydraulic design of the Cycle 4 core, and effects of the Cycle 4 core upon the LOCA and non-LOCA accidents discussed in the UFSAR. Conclusions presented in UFSAR were found to remain valid and no new credible failure modes have been created for the Cycle 4 reload.

- The fuel design changes were made to improve quality of manufacturing and to improve installation of the RRTNs to the burned fuel assemblies. Evaluations were performed which determined these changes did not affect the fuel assembly form, fit, or function.

Based upon the preceding information and the following:

1) an EOC 3 burnup between 18,509 and 19,657 MWd/MTU, and

2) termination of Cycle 4 burnup at or before 21,327 MTU, including a power coastdown, and

3) adherence to plant protective and operating limitations given in the Technical Specifications and the COLR,

There are no unreviewed safety questions or technical specifications changes identified as a result of the Watts Bar Unit 1, Cycle 4 core design. Therefore, the Cycle 4 reload design is licensable under 10 CFR 50.59, and requires no prior NRC approval.

SA-SE Number: WBN-TS-01-03-0

Implementation Date: 03/7/2001

Document Type:	***Affected Documents:***	***Title:***
Other Change	TRM Change No. WBN-TS-01-03 UFSAR Change Pkg. No. 1673	10 CFR 50.59 Rule Change

Description and Safety Assessment:

TRM Section 5.1, Technical Requirements Control Program, is being revised to be compatible with the recently revised rule for 10CFR50.59.

TRM Section 5.1.2 provides that TRM changes or additions may be made without prior NRC approval provided the change is not a candidate for inclusion in the Tech Specs [based on the criteria contained in 10 CFR 50.36(c)(2)(ii)] and provided the change does not involve an "unreviewed safety question." The term "unreviewed safety question", however, was eliminated from 10 CFR 50.59 in a recent CFR change, effective March 13, 2001. To accommodate this CFR change, TRM Section 5.1.2 is being revised to simply reference TRM changes made pursuant to 10 CFR 50.59 rather than those involving an "unreviewed safety question." An editorial wording enhancement is also made in Section 5.1.4. This change is administrative and has no affect on the review process for TRM changes.

In addition, several locations in the UFSAR are being revised by UFSAR Package No. 1673 to delete reference to the term "unreviewed safety question" and provide reference to NRC reviews pursuant to 10 CFR 50.59.

The changes to the TRM and UFSAR do not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, these revisions do not affect or involve plant operation, transient or accident analyses, or malfunctions of equipment. These changes are administrative and have no affect on the review process for TRM changes or reviews and evaluations discussed in the UFSAR. Hence, the proposed revision does not constitute an unreviewed safety question, require additional technical specification changes, or otherwise require NRC approval to implement.

SA-SE Number: WBOTSS-99-122-0

Implementation Date:01/07/2000

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 0-00-1-30 Rev. 0	Temporary Changes to Auxiliary Building HVAC

Description and Safety Assessment:

TACF 0-00-1-30 permits modification of the operational configuration of the Auxiliary Building heating, ventilation, and air conditioning (HVAC) system. In order to alter normal air flows during leak repair activities in the Fuel Transfer Canal (FTC) and during implementation of DCN D-50348 to upgrade fuel handling system located in the FTC, the following HVAC changes are necessary to eliminate the potential spread of contamination into the ventilation ductwork.

- Fire damper 0-ISD-031-3846 will be closed.
- All 36 of the HVAC openings around the FTC at Elevation 750' 1/2" will be blocked closed using herculite and duct tape.
- Two duct access doors will be opened into the fuel handling area exhaust system duct located on the refuel floor.

Activities in FTC are associated with implementation of the corrective action plan for WBPER99-007143-000 and implementation of DCN D-50348. During this activity, the FTC will not be used for transfer of fuel. There is currently no radioactive material stored in the FTC. However, the canal has been both wet and contaminated. Activities associated with leak repair in the canal or activities associated with implementation with upgrade of fuel handling system in the FTC could loosen and cause transport of contamination, including hot particles, into the ventilation ductwork. The intent of this TACF is to make temporary modifications to the ventilation system to ensure that if any contaminated material freed during these activities in the FTC will not be drawn into the permanent HVAC exhaust system.

During routine plant operation, one of the design functions of the fuel handling area exhaust system is to maintain the refuel floor at a negative pressure. The specified negative pressure is maintained using modulating dampers associated with the fuel handling area exhaust system. Opening 2 access doors into the fuel handling area exhaust duct and covering them with screens will ensure adequate airflow into the fuel handling area exhaust system to maintain the refuel floor at the required negative pressure.

During plant operation, if an Auxiliary Building Isolation (ABI) signal is initiated, the design functions of the fuel handling area exhaust system and the Auxiliary Building general ventilation system are to stop. Isolation of the Auxiliary Building during an ABI is provided by dampers and other components which are not impacted by this TACF.

During an accident, the Auxiliary Building negative pressure is maintained using the Auxiliary Building Gas Treatment System (ABGTS). Some portions of the fuel handling area exhaust ductwork, which are affected by the TACF, are used by the ABGTS as part of its suction-side ductwork. As described on the attached Table A, the blockage of air flow from around the FTC will not have a significant impact on the operation of the ABGTS because of the equalizing effects of the open access doors. Therefore, both systems (fuel handling area exhaust system and ABGTS) will continue to perform their design functions.

The fuel handling area exhaust system is non-safety-related, which stops on an ABI signal. The ABGTS primarily operates to assure that releases to the environment do not exceed 10CFR100 limits as a result of a loss of coolant accident, or a fuel handling accident, by maintaining the Auxiliary Building secondary containment enclosure (ABSCE) at a negative pressure and processing all exhaust air through the Air Clean up Units. The changes made by this TACF will not adversely impact the fuel handling area exhaust system, or the ABGTS from performing their design functions, during normal operation, and in the event of an accident, respectively.

Document changes have been evaluated for plant operability during the review process and found not to affect the plant operation. The credible failure modes for the system affected by these changes have been evaluated against the accidents identified in the UFSAR. It is concluded that they do not introduce a failure pathway different from those identified and evaluated in the UFSAR accidents. The applicable accidents and the equipment served by the affected safety system have been reviewed against these documentation changes and no new malfunction pathways will be introduced which have not previously been evaluated and identified. These changes will not increase the off-site dose rates to the public analyzed in UFSAR Chapter 15. No change will occur to the radiological consequences of accidents analyzed in the UFSAR.

The Technical Specification Bases have been reviewed to determine if any margins of safety are affected by these documentation changes. No margin of safety is identified in the Bases section which could be reduced by these changes.

SA-SE Number: WBOTSS-00-044-1

Implementation Date: 04/14/2000

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-00-4-030, Rev. 0	CRDM Cooler 1A-A Motor 1 Disabled while 1A-A Motor 2 remains operable

Description and Safety Assessment:

WO 00-005849-000, was initiated to troubleshoot the cause for Control Rod Drive Mechanism (CRDM) cooler 1A-A tripping off. CRDM Cooler 1A-A Motor 1 was found shorted to ground. During plant operation in Mode 1 or 2, this motor is not accessible for maintenance.

This TACF is initiated to determinate cable to permit CRDM Cooler 1A-A Motor 1 to be disabled while CRDM Cooler 1A-A Motor 2 will remain operable. This change will allow the CRDM Cooler 1A-A to meet the Watts Bar Nuclear Plant Fire Protection Report cooling requirements to maintain lower containment temperatures when operated in the bypass mode. Handswitches for associated dampers are positioned to ensure cooler is operated in the bypass mode only. This includes the Auxiliary Control Room, "C" handswitches, so as to ensure CRDM Cooler 1A-A will be aligned for BYPASS in the event of a MCR abandonment.

There are no WBN design basis Chapter 15 events for which the CRDM cooler system is required to operate. The CRDM coolers and associated dampers and duct are not safety related and are not required to perform a primary nuclear safety function.

The requirement in the Fire Protection Report is that either three Lower Compartment Coolers (LCCs) or two LCCs plus two CRDM coolers be functional for an Appendix R event for which a minimum total heat removal capability of 6.3M BTU/hr (MBH) must be available. During the bypass mode of operation, air is drawn from lower containment through the cooling coils and discharged back into Lower Containment. In the bypass mode, the air flow rate through the CRDM cooler with both Fan 1 and Fan 2 in service is approximately 37,000 cfm. At this flow rate the resulting pressure drop across the HVAC fittings is calculated to be 5.31 inches water gauge. The design air flow rate through the CRDM coolers in the bypass mode is 34,900 cfm +/- 10%. Based on the review of manufacturers fan performance curves, a single CRDM fan provides air flow rates of 32,000 cfm, at a pressure drop of 5.31 inches water gauge. This conservative method of evaluating system performance clearly identifies that a single fan provides design air flow rates when the CRDM system is in the bypass mode of operation. The minimum design flow is 31,410 cfm. The two fan flow design heat removal rate for CRDM Cooler 1A-A is maintained in the bypass mode of operation as this alteration maintains air flow above minimum design and cooling water design flow is not affected.

The CRDM cooling system is not addressed in technical specifications. The margin of safety defined in the basis for any technical specifications is not reduced.

The system itself is not safety related nor is it required for accident mitigation. As discussed in the previous evaluations, this activity involves placing the CRDM Cooler 1A-A in its operational mode which is required for the associated Appendix R function. This does not introduce any additional failure modes and does not increase the chances or consequences of accidents evaluated in the UFSAR. Therefore the TACF does not constitute a unreviewed safety question.

SA-SE Number: WBOTSS-00-057-0

Implementation Date: 6/19/2000

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-00-5-63	Recorder Installation for Cold Leg Accumulator No. 2.

Description and Safety Assessment:

A Yokogawa recorder will monitor and record the Cold Leg Accumulator (CLA) 2 level. Data from the recorder will be collected by the System Engineer and used to troubleshoot the CLA 2 out-leakage problem. Recorder leads are connected to test point 1 LP/952 in the safety injection system (SIS) Accumulator Tank 2 Level Loop. This temporary change is strictly a tool for collecting data and does not adversely impact the CLA 2 level control loop.

The temporary recorder will be connected to an existing level measurement loop on the SIS CLA 2 (loop 1-LP-63-109). This loop uses a differential pressure transmitter to drive a 10 to 50 milliamp current loop which contains alarm bistables and indicators. The signal conditioning portion of the current loop is located in instrument rack 1-R-14 in the Auxiliary Instrument Room. The loop is non-safety related and has no control functions. The current loop is designed with permanently installed test points which are resistors with test jacks designed for the purpose of connecting test instruments for monitoring loop current (and thus CLA 2 level) without disturbing the functionality of the current loop. The temporary recorder is a high impedance device which will have no impact on the current when connected to the test point. The test point, is designed with female banana jacks. The recorder input leads will be connected with banana type connectors.

1-LPL-63-109 is one of two redundant level measurement loops for CLA 2. It is used for Technical Specification Section 3.5.1 compliance verification of accumulator level via indicators in the MCR. Any failures of the 1-LPL-63-109 would be detected by the Operator through the loop Hi and Lo level alarms, or through the routine channel checks that are made between the two redundant level measurement loops. The balance of the loops in 1-R-14, which is a Control Group Rack, are non-safety related.

The recorder will monitor and record the CLA #2 level. The recorder leads are connected (plugged in) to a test point and power is supplied via a standard wall outlet. The recorder is not expected to monitor post accident conditions, and does not impact process instrumentation or ECCS logic as described in the UFSAR. Therefore, no unreviewed safety questions will exist. This temporary change will be removed upon completion of troubleshooting activities at Refueling Outage 3.

SA-SE Number: WBOTSS-00-079-0

Implementation Date: 06/26/2000

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-00-8-2, Revision 0	Temporary Alteration Disables Root Valve.

Description and Safety Assessment:

This TACF disables 1-RTV-2-385A, 1-PI-2-207 Root Valve, in the closed position by Furmanite injection to stop an external body-to-bonnet steam leak. 1-PI-2-207 is the non-quality related locally mounted main feedwater pump (MFP) A suction pressure indicator. A Work Requests will install a Furmanite fitting immediately upstream of the valve seat and a maximum amount of 6 sticks of compound will be injected to seal the body-to-bonnet joint leak to atmosphere.

This TACF changes the valve alignment by closing and disabling this normally open root valve. The position of this root valve is depicted in UFSAR Figure 10.4-7, TVA Drawing 1-47W804-1. However, since this is simply a non-quality related root valve to a local pressure indicator, and not a process flow valve, this change is not considered to significantly affect the description of the system as presented in the UFSAR.

The only type of accident that this TACF could possibly be associated with is "Minor Secondary System Pipe Breaks." However, the purpose of this TACF is to stop a steam leak in a 1-inch root valve to a non-quality related information only pressure indicator. Therefore, this activity should help minimize the probability of a secondary system pipe break by eliminating flow in this normal dead leg sense line. This 1 inch secondary pipe break accident is bounded by the analysis of major secondary pipe breaks and does not require further analysis.

This change involves a controlled injection of Furmanite compound into a small line segment on the upstream side of a leaking root valve. This valve controls a local pressure indicator which is not used in response to any plant transients or in any emergency response procedures. The local instrumentation is not discussed in the UFSAR nor is it a compliance instrument for the technical specifications. This line segment is not important to secondary line breaks and, by virtue of being on the secondary side, does not contribute to the potential for radiological releases even during SGTR events. This change therefore does not constitute a unreviewed safety question.

SA-SE Number: WBOTSS-00-108-0

Implementation Date: 09/06/2000

Document Type:	***Affected Documents:***	***Title:***
New Procedure	MI-61.010 Rev. 0	Ice Condenser servicing activities during fuel handling operations

Description and Safety Assessment:

The subject procedure identifies requirements for installation of hoses associated with Ice Condenser servicing. This includes temporarily installing hoses/piping in a containment penetration during an outage.

This maintenance instruction:

a) Allows hoses/piping to be installed in Containment Building penetration X-37 during Mode 5, 6, or core empty to remain temporarily installed during core alterations, handling of irradiated fuel inside containment or the fuel handling area, and mid-loop operations.

b) Allows the manual isolation valves on fluid filled hoses used for ice condenser melt tank draining through Containment Building penetration X-37 during Mode 5, 6 or core empty to remain open during core alterations, handling of irradiated fuel inside containment or the fuel handling area, and mid-loop operations provided a sufficient barrier has been verified intact between containment atmosphere and the Annulus. Prompt closure (within 15 minutes) of these manual isolation valves is required when this barrier integrity is lost, in the event of a loss of residual heat removal (RHR) shutdown cooling during mid-loop operations or a fuel handling accident inside containment or the fuel handling area.

c) Provide additional emergency closure actions to be performed should a loss of RHR cooling occur when Containment penetration are not intact during mid-loop operations. These steps require removal or cutting of hoses/piping in the penetration and the reinstallation of at least one blind flange in the Containment. This flange is testable and equipped with two seal 0-rings. All tools required to perform these actions shall be staged at penetration X-37. This emergency closure procedure is tracked by TI-68.002 when penetration X-37 are breached.

d) The penetration shall be restored to its normal configuration or the configuration required to support performance of a Containment Integrated Leak Rate Test (CILRT). Local Leak Rate Testing (LLRT) shall be performed on penetration X-37 prior to performance of CILRT.

e) As left LLRT shall be performed on penetration X-37 prior to Mode 4 entry or power ascension.

f) Dow Corning 3-6548 Silicone RTV Foam shall be used to seal all the affected penetration. A minimum depth of 12 inches is required in all penetration regardless of cables or hoses. A continuous nominal 1/4 inch bead of Dow Corning adhesive sealant 732, 96-081, 790, or 795 shall be injected at the foam to hose/sleeve interface on one side in X-37. Use of this sealant and adhesive for these penetration is documented in the following drawings:

These barriers are rated for 3 psi air or flood and 437°F using this sealant. No restrictions apply to the number of hoses or the separation of the hoses or cables in these applications. Maximum cable loading is 50% of the penetration area. As no Kaowool Boards will be installed at these penetration to ensure a 3-hour fire rating, installation of these temporary features in the penetration is considered a fire impairment that must be tracked and restored prior to Mode 4 entry.

g) Manual isolation valves shall be installed on all hoses which penetrate X-37 inside containment and the Annulus or Auxiliary Building. Configuration control of these valves shall be maintained in this instruction. During fuel handling or core alterations, personnel shall be assigned to close manual isolation valves on hoses inside containment and in the Annulus or Reactor Building which penetrate X-37 to ensure prompt closure upon notification of the event.

h) The worst case scenario for containment pressurization and heatup is the loss of RHR shutdown cooling. RCS pressurization is limited to 2 psig by the RCS vent paths available during mid-loop operations with minimal decay and residual heat. Thus, containment pressure should never achieve 2 psig or saturation temperature for 2 psig water due to core boiling of 229°F. All penetration altered by this instruction exceed these pressure and temperature conditions.

i) The temporary hoses/piping and related isolation valves and equipment that establish the barrier between Containment and the Annulus/Reactor Building are commercial grade and rated for at least 150 psig and 250°F. These components do not satisfy seismic Category I, TVA safety class 6 requirements. However, these components are not used to maintain containment integrity as required in Modes 1 through 4. The temporary piping which is installed in the steel containment vessel penetrations and which passes through the annulus is also seismically supported to ensure the piping does not fall on any other piping or equipment which may be located beneath the penetration as the result of a seismic event. Normal seismic Category I, TVA safety Class B barriers may be restored to their normal design configuration as determined by the Shift Manager. No Appendix R fires or internal missile hazards need to be considered for the use of this equipment as it will only be used during Modes 5 and 6 when moderate or high energy line breaks or fires affecting safe shutdown equipment are not postulated to occur. This breach will be tracked as a Fire Impairment for Mode 4 entry. An in service leak test of the hoses/piping shall be performed following alteration of X-37 and prior to declaring the breaches restored per TI-65 and TI-68.002.

j) During installation and restoration of the temporary alterations to these penetration, breaches shall be tracked for Containment penetration operability per Technical Specification 3.9.4 and NRC Generic Letter (GL)88-17.

The affected penetration shall only be altered in Modes 5 and 6 or no-mode when containment integrity and Shield Building integrity are not required. No risk of internal flood, moderate or high energy line break exists in this condition. No safety related cables penetrate the affected penetration. The altered penetration has no impact on Appendix R safe shutdown equipment in Modes 5 and 6. The interstitial spaces between the hoses will be filled with at least 12 inches of RTV foam per design drawings and consistent with Technical Specification 3.9.4 Bases. When the penetration is breached for installation and removal of hoses/piping, breaches to Containment Closure to ensure these breaches are closed prior to performing Core Alterations or movement of irradiated fuel inside containment or the fuel handling area. This instruction is considered an emergency closure procedure per GL 88-17 and Abnormal Operating Instructions (AOI)-14, Loss of RHR Shutdown Cooling and TI-68.002 when penetrations are breached during mid-loop operations. Configuration control shall be maintained over manual isolation valves when the altered penetration is sealed to ensure no path exists from Containment to the Auxiliary Building. Closure requirements are consistent with GL 88-17 definition. Upon notification of a Fuel Handling Accident or loss of RHR Shutdown cooling during mid-loop operations, both manual isolation valves on each hose shall be closed. Operability for containment integrity is established prior to Mode 4 entry by performance of LLRT's on testable penetration.

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

Implementation Date: 02/16/2001

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-01-1-030, Rev. 0	Control Rod Drive Mechanism Cooler 1B-B in Bypass Mode

Description and Safety Assessment:

February 1, 2001, 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 TACF will allow CRDM Cooler 1B-B motor 2 (1-MTR-030-0092/2-B) to be disabled while CRDM Cooler 1B-B motor 1 (1-MTR-030-0092/1-B) will remain operational. CRDM Cooler 1B-B motor will be disabled by removing the control power fuses for 1-BKR-030-0092/2-B (1-FU-212-B110/31 and 1-FU-212-B110/32) 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 TACF 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, 1-HS-30-94A and 94C, will be maintained in the OPEN position.

B. 1-TCO-30-93 CLOSED, it's associated handswitches, 1-HS-30-93A and 93C, will be maintained in the CLOSE position.

In this configuration annunciator 1-XA-55-5C, window 102A may alarm when CRDM Cooler 1B-B is operated due to less than design negative pressure at flow switch 1-FS-30-92A/B-B. 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 LCC, are required for safe shutdown per 10CFR50, Appendix R, to keep containment temperatures from exceeding operability (environmental qualification) limits on safe shutdown equipment inside containment.

The system itself is not safety-related nor is it required for accident mitigation. As discussed in the previous evaluations, this activity involves placing the CRDM cooler 1 B-B in its operational mode which is required for the associated Appendix R function. This does not introduce any additional failure modes and does not increase the chances or consequences of accidents evaluated in the UFSAR. Therefore the TACF does not constitute a unreviewed safety question.

SA-SE Number: WBOTSS-01-011-0

Implementation Date: 03/20/2001

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-01-03-047 Rev. 0	Installation of an alternate Main Turbine Oil Tank (MTOT) Vapor Extractor

Description and Safety Assessment:

This temporary alteration addresses the online installation of an alternate Main Turbine Oil Tank (MTOT) vapor extractor to be located downstream of the permanently installed vapor extractor. This is a non-quality related, non-seismic related system/component located in the Turbine Building. An air operated venturi type air mover, Coppus Jectair 6 or equivalent, will be attached to the vapor extractor 10-inch roof vent pipe (gooseneck). Flow testing of the installed vapor extractor sub-system determined that the vapor extractor flow is 830 cfm at 4 inches H_2O vacuum in the MTOT. The air mover is rated at 4500 cfm total flow in free air and is able to impose 5.6 inches H_2O vacuum under blockedtight conditions using 126 scfm of motive air at 80 psig. A 200-foot long 1-inch diameter air hose is capable of providing 220 scfm of motive air from the Station Air system (100 psig nominal) to the air mover on the roof at 80 psig. The frictional pressure drop in the 10-inch vapor extractor discharge piping between the MTOT and the roof vent at the operational flow rate of 830 cfm is approximately 0.8 inches H_2O. This air operated vapor extractor is capable of maintaining the desired negative pressure of approximately 4 inches H_2O at the MTOT (approximately 4.8 inches H_2O at the gooseneck) and the desired low rate of approximately 830 cfm in the event the installed vapor extractor fails. The vapor extractor discharge throttle valve should be throttled as necessary if the air operated vapor extractor is placed in service. Motive air will be supplied from the Station Service Air System. A manometer will be installed at a test connection on the MTOT Vapor Extractor suction piping to facilitate monitoring of MTOT pressure.

The subject MTOT Vapor Extractor is not described in any text of the UFSAR or in the technical specifications. However, the piping configuration of the vapor extractor is depicted on UFSAR Figure 10.4-1, TVA Drawing 1-47W807-1, FLOW DIAGRAM, TURBINE GRAINS AND MISCELLANEOUS PIPING. There are no DBAs involved with this activity nor are there any new credible failure modes. The function of the new air operated vapor extractor is to provide an alternate means of maintaining a slight negative pressure at the MTOT and main turbine bearing housings in the unlikely event of a failure of the installed vapor extractor. An alarm in the MCR provides indication of a failure in the existing vapor extractor. Operation would then take the appropriate steps to align and start the alternate vapor extractor as necessary. This does not conflict with nor does it affect the technical content of the UFSAR text.

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 does not affect the operation of any equipment important to nuclear safety, either directly or indirectly. This change will not increase the off-site dose rates to the public, as analyzed in the UFSAR Chapter 15, nor will it increase the radiological consequences of accidents analyzed previously.

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 an unreviewed safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLCE-99-015-0

Implementation Date: 10/14/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50007-A UFSAR Change Pkg. 1604 TRM Change # 99-012	Upgrade WBN Seismic Instrumentation Monitoring System.

Description and Safety Assessment:

The purpose of this Safety Evaluation is to review permanent modifications to the Seismic Monitoring System. The SE also supports corresponding updates made to Section 3.7.4 of the UFSAR and to Sections 3.3.4 and B3.3.4 of the TRM to reflect the configuration of the modified Seismic Monitoring System and changes in the logic used to determine the need for shutdown following the occurrence of a seismic event that exceeds the OBE design response spectrum.

DCN D-50007-A has been initiated to upgrade the Seismic Monitoring System by replacing obsolete Kinemetrics seismic monitoring instruments, removing from service all of the obsolete Engdahl instruments, and upgrading the recording capabilities of the system from analog to digital. Installation of upgraded seismic instrumentation will also permit deletion of seismic switches and triggers that are rendered redundant by the function of the upgraded system. The upgraded Seismic Monitoring System will greatly reduce the maintenance resources required to support system operation. The upgraded instrumentation will also provide seismic digital recording and analysis capabilities sufficient to permit the adoption of EPRI OBE Exceedance Criteria, which avoids unnecessary shutdown following non-damaging seismic events that exceed the OBE.

The Seismic Monitoring System is *not* safety-related; nor does it have any effect on any safety related system or equipment. The upgraded seismic instrumentation will improve the accuracy and reliability of the seismic monitoring system; while at the same time greatly simplifying the maintenance and surveillance resources required to support the system. The upgrades of DCN, D-50007-A enhance the functionality of the seismic monitoring system by the addition of the Kinemetrics Condor System, which-features a high degree of redundancy and is a one-to-one, or equivalent, replacement for the existing SMA-3/SMP-1 based recording and playback system. The inclusion of an onboard processing computer and strong motion analysis software as part of the Condor System facilitates timely evaluation of the recorded event. The removal from service of the obsolete Engdahl instruments greatly reduces the maintenance and surveillance requirements for the Seismic Monitoring System without any significant loss of data capability.

The use of EPRI OBE Exceedance Criteria ensures that the decision for a controlled shutdown is based on actual damage potential of the event, which reduces the risk associated with unnecessary shutdowns. UFSAR Section 3.7.4 and TRM Sections 3.3.4 and B3.3.4 have been revised to reflect the upgraded configuration of the Seismic Monitoring System and the adoption of EPRI OBE Exceedance Criteria for shutdown logic. The upgraded Seismic Monitoring System and use of EPRI OBE Exceedance Criteria maintain the UFSAR commitment to the intent of Reg Guide. 1.12, Revision 1.

The Seismic Monitoring System is *not* safety-related; nor does it have any effect on safety-related systems or equipment. The upgraded seismic instrumentation will improve the accuracy and reliability of the seismic monitoring system; while at the same time greatly simplifying the maintenance and surveillance resources required to support the system. The upgrades of DCN D-50007-A enhance the functionality of the seismic monitoring system by the addition of the new instrumentation, which features a high degree of redundancy and is a one-to-one, or equivalent, replacement for the existing analog based recording and playback system. The inclusion of an onboard processing computer and strong motion analysis software facilitates timely evaluation of the recorded event. The removal from service of the obsolete instruments greatly reduces the maintenance and surveillance requirements for the Seismic Monitoring System without any significant loss of data capability. The use of EPRI OBE Exceedance Criteria ensures that the decision for a controlled shutdown is based on actual damage potential of-the event, which reduces the shutdown risk associated with unnecessary shutdowns. Therefore, implementation of DCN D-50007-A and corresponding revisions to the UFSAR and TRM do not involve a unreviewed safety question.

SA-SE Number: WBPLCE-99-021-0

Implementation Date: 09/17/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50348-A UFSAR Chg Pkg 1623	Fuel Handling System Upgrade

Description and Safety Assessment:

The upgrade of the Fuel Transfer System will require the replacement of the control enclosures. The new control system will maintain the existing requirements for motion interlocks and other controls for the carriage and upenders. The control system will be computer based utilizing PLC to replace many of the existing electro-mechnical devices. The Spent fuel pit side console will act as the master controller for Transfer Operations.

A PLC based load weighing system with adjustable setpoints will be provided for the carriage. The load cell will enable cable overload setpoints to be set in the PLC logic to prevent cable and carriage damage while providing a digital display. The existing carriage drive mechanism (shaft and chain) will be replaced with a variable speed winch cable drive system.

The upgraded system replaces the carriage motor with a DC motor and associated motor controller which interlocks with the new PLCs. This allows indication of the Traverse carriage position and ensure accurate overtravel trip at each end of the transfer canal. This interlock will limit carriage travel and prevent the motor from hitting the hardstops.

The personal computer used for the database also supports touchscreen performance of the new monitor included as part of the Control console. This monitor will provide fuel handling operators with the ability to confirm or perform an action without having to enter the sequence manually for each -movement individually. Additionally, data logging is provided for logging fuel movement, travel override and interlock override. The present bridge, trolley and hoist motors and drives will be replaced with new AC brushless motors and drives. The new hoist, bridge and trolley motors provide improved operating characteristics.

The Fuel Transfer System is directly associated only with the refueling mode of operation and is not involved with plant startup, normal operation, or shutdown. The AC supply power is provided by the normal AC auxiliary power system, which is not a Class 1E power source, therefore no electrical separation or isolation is required except for that required between Node voltage levels. All cable and conduit routing will be in accordance with existing separation criteria for differing voltage levels.

Performance of this modification will require opening the fuel transfer tube gate valve in Mode 1. This valve is part of the Watts Bar Secondary Containment-Boundary because there is a vent from the fuel transfer tube to the annulus. This vent would normally be utilized as a bleed line to the annulus for any leakage from the primary containment. The Secondary Containment was designed to provide a positive barrier to all primary containment pathways during a LOCA. In order to ensure that a breach of the Secondary Containment/Auxiliary Building boundary does not occur, the fuel transfer tube vent to the Reactor Building annulus will be plugged prior to opening the transfer tube gate valve. This will assure compliance with technical specifications and will assure that there are no new credible failure modes.

The Primary Containment Boundary (fuel transfer tube double gasketed blind flange) will have to be breached in order to position the transfer cart in the reactor side upending position. WBN Technical Specification requires that primary containment integrity be maintained during Modes 1 through 4. In order to assure that no new credible failure modes are created, any modifications that require opening the fuel transfer tube blind flange shall be implemented only during Modes 5 or 6 (prior to flooding).

The DBA associated with the Fuel Handling and Storage System (FHSS) is the "Fuel Handling Accident (FHA)," documented in Chapter 15 of the UFSAR. This accident is defined as the dropping of a spent fuel assembly onto the spent fuel pit floor resulting in the rupture of the cladding of all the fuel rods in the assembly. The "Environmental Consequences of a Postulated Fuel Handling Accident" section of the UFSAR assumes that the highest powered assembly is involved and all of the rods in this assembly rupture. The proposed upgrade of the Fuel Handling System does not after any of the assumptions or conclusions of this analyses.

The scope of this modification has been reviewed and compared with the UFSAR and Technical Specifications. This change does not:

1. Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR. The FHSS is a non-nuclear safety system. The only accident evaluated in the UFSAR concerning this system is the FHA in Chapter 15 of the UFSAR. This modification will not affect the existing UFSAR evaluation and will not affect the environmental consequences of a postulated fuel handling accident,

2. Create the possibility of an accident or malfunction of a different type than evaluated previously in the UFSAR. These modifications will retain the required functional requirements of the fuel transfer and storage system as previously analyzed. The fuel transfer and storage system is physically isolated from systems important to safety during normal operation. The effects on the primary and secondary containment boundaries of this modification remain unchanged from the current configuration as evaluated in this safety evaluation. The changes in this modification do not create the possibility of an accident or malfunction of a different type than evaluated previously in the UFSAR.

3. Reduce the margin of safety as defined in the basis for any Technical Specification. Technical Specification 3.9, "Refueling Operations" governs the operation of the FHSS. Technical Specification 3.6, "Containment Systems" governs the operation of the primary and secondary containment. This analysis has determined that the margin of safety as defined in the bases for any technical specification has not been reduced. Since the system functional requirements and accident consequences remain unchanged, this modification will not decrease the margin of safety. Precautionary measures will be taken in accordance with site procedures and instructions for any primary or secondary containment penetrations. This will ensure that no containment breaches due to this modification will result in any reduction in the safety margin or affect radiological releases as addressed in 10 CFR 100.

For these reasons, this activity does not constitute a unreviewed safety question.

SA-SE Number: WBPLCE-00-003-0

Implementation Date: 3/30/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50397-A UFSAR Change Pkg 1622	Ice Condenser Replacement Baskets

Description and Safety Assessment:

UFSAR Change Package No. 1622 revises UFSAR Section 6.7 "Ice Condenser System" to address adding text which describes the 2 foot basket replacement option when damaged baskets need to be replaced in the ice condenser rows 1,2,8 and 9. Replacing 12 foot baskets for most locations in these outer rows in nearly impossible due to overhead interferences from overhanging equipment in the upper plenum. Also, text is being added to address alternate self-tapping sheet metal screws for all maintenance activities.

The UFSAR revision describes hardware alternates/options which have been either engineered or tested. A detailed evaluation for the 2 foot basket option was performed as part of the submittal package contained in Westinghouse Letter WAT-D-10716, Revision 2. Confirmatory lab tests were performed by TVA Central Laboratories Services for the alternate self-tapping sheet metal screws and the results evaluated and approved by TVA Corporate Metallurgical Staff for use as an alternate screw in the ice condenser.

The ice condenser system is a passive safety system designed primarily to absorb thermal energy during a high energy line break inside the Containment Building. The system ensures that a sufficient amount of ice is available to keep the maximum containment pressure below the steel containment vessel design limit during a large break LOCA. The system also limits the containment temperature response during a main steam line break inside containment to temperatures below the maximum equipment qualification limit. The ice condenser limits the radiological consequences of a LOCA by reducing the fission product iodine concentration of the post-LOCA containment atmosphere. The ice condenser is a source of borated water for the ECCS and the containment spray system during the recirculation phase of a LOCA. It contributes to the post-LOCA containment sump level which establishes a positive suction head for the ECCS and containment spray pumps. These safety features are credited in the long term and short term LOCA containment temperature analyses (UFSAR Sections 6.2.1.3.3 and 6.2.1.3.4), main steam line break inside containment temperature analysis (UFSAR Section 6.2.1.3.10), large break LOCA analysis (UFSAR Section 15.4.1) and the environmental consequences of a LOCA analysis (UFSAR Section 15.5.3). These are the ice condenser DBAs.

Since the engineering and testing results demonstrate adherence to the design basis requirements, this UFSAR revision activity does not alter the credible failure modes previously evaluated for the DBAs listed above. The failure mode assumed in each of these analyses (i e., loss of a single train of safety related diesel generator power) continues to be the limiting credible failure

The proposed ice condenser system UFSAR revision is based on existing system design requirements and does not involve system functional changes, hardware modifications (alternate/option hardware equivalent only) operational procedure revisions The revision does not alter, add or delete any safety related performance requirement for the ice condenser system or any other safety related system. The change is consistent with existing technical specification operability requirements. There are no changes to the assumptions, inputs, methodology or conclusions of the associated DBA analyses. As a result. the proposed change does not involve an unreviewed safety question.

SA-SE Number: WBPLCE-00-029-0

Implementation Date: 08/31/2000

Document Type:	***Affected Documents:***	***Title:***
UFSAR	UFSAR Change Package 1654	Application of Heavy Ice Baskets

Description and Safety Assessment:

Westinghouse has performed seismic analyses for the application of heavy ice baskets for the Watts Bar Ice Condenser System and the results have been reflected in Westinghouse WAT-D-10850.

The analysis methodology is generally the same as in the original seismic design qualification performed by Westinghouse and presented in the Watts Bar UFSAR. The set B seismic spectra were used in the analysis for the application of heavy ice baskets. This similar approach allowed comparison of the results of the design basis analyses and these new analyses.

From this seismic evaluation, Westinghouse has established a Watts Bar plant-specific ice basket maximum loading limit and configuration requirements. This evaluation has determined that the Watts Bar ice basket maximum average loading limits and configuration requirements presented in WAT-D-10850 are acceptable based on the seismic design allowables. The average is taken of the nine baskets (3x3 array) within each zone.

The following conclusions can be made based upon the analyses performed:

1. Lattice frames, cradles, ice baskets, lower support structure, embedments/collar studs are acceptable for this new weight limit.
2. Basket qualification was demonstrated (but, is limited to a maximum of 2 screws missing per basket, at the 6-foot elevation. All other elevations are still acceptable for up to 4 missing screws).
3. For the supplied time-histories (Set B, Evaluation Basis) safe shutdown earthquake (SSE) analysis controls over operating basis earthquake (OBE). Since the 2-mass OBE results were well within their allowables, additional OBE time-histories were not run.
4. The maximum single basket weight limit is 2090 lbs (ice plus basket). This limit includes 19 lbs toaccount for the concentrated mass at the lower support structure.
5. Neither the maximum average ice plus basket weight limit of 1809, 1909, and 2009 lbs for basket zones A, B and C respectively, nor the maximum single basket limit (2090 lbs ice plus basket weight) may be exceeded. Note that the maximum average limit also includes 19 lbs to account for the concentrated mass at the lower support attachment.

Calculation WBN-OSG4-091, "Maximum Containment Water Level" was reviewed to assess the potential of impact of increased ice mass from this change. This calculation showed that the peak flood level is represented by the transient analysis. That analysis will not be affected by the increased ice mass since the peak transient level will occur well before completion of the ice melt. The rate of ice melt is governed by the channel configuration which remains unchanged. The final flood level is three feet, or 25%, below the peak transient level, providing ample margin for the potential maximum of increase in ice mass volume of 7%. Additionally, WAT-D-8902, Revision 1 was reviewed. A Westinghouse analysis was performed to assure that the sump pH is maintained above the minimum of 8.0. This analysis conservatively assumed a low ice mass volume. The technical specification requirement for the ice pH is between 9.0 and 9.5, therefore the additional ice mass will only tend to assure that the sump pH will remain above the minimum.

The proposed ice condenser system UFSAR revision is based on existing design requirements and limits, and does not involve system functional changes, or hardware modifications. The revision does not alter, add, or delete any safety related performance requirement for the ice condenser system or any other safety related system. The change does not affect existing technical specification operability requirements. There are no changes to assumptions, inputs, methodology or conclusions of the DBA analyses. Therefore, the proposed UFSAR change is not an unreviewed safety question.

SA-SE Number: WBPLEE-98-065-0

Implementation Date: 1/20/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN W-39833-A	Replacement Breakers from Spare Breakers

Description and Safety Assessment:

DCN 39833-A identifies acceptable replacement breakers for non-safety related, 480V switchgear load/feeder breakers. Specifically,

(1) for 480V Intake Pumping Station Board, adds a note identifying that DS-632 breakers (non-qualified) from the Unit 2 Turbine Building Boards or the Service Building Main Boards may be used to replace the DS-632 breakers (1LB qualified) in the 420V Intake Pumping Station Board,

(2) for 480V Intake Pumping Station Board compartment 3D breaker. changes the amptector type from LI to LS and adds a note to permit an electrically operated breaker to be exchanged with the mechanically operated breaker in compartment 2D,

(3) for 480V Turbine Building Common Board compartment 2D breaker, replaces the 150A CT sensor with a 800A CT sensor, and adds a note that this breaker can be exchanged with the breaker in compartment 6C,

(4) for 480V Auxiliary Building Common Board compartment 12D, adds a note to permit an electrically operated (EO) breaker to be exchanged with the mechanically operated breaker in compartment 2D,

(5) for 480V Service Building Main Board compartment 2B, replaces the 300A CT sensor with a 600A CT sensor and replace the LI Amptector with a LS Amptector, adds a note that this breaker can be exchanged with breakers in compartments 2A, 2C, 12A, & 12C,

(6) for 480V Service Building Main Board compartment 2B, replaces the 300A CT sensor with a 600A CT sensor, and adds a note that this breaker can be exchanged with the breaker in compartment 8D,

(7) for 480V Unit Board 1A compartment 8A replaces the 100A CT sensor with a 150A CT sensor and replaces the LI Amptector with a LS Amptector, adds a note that this breaker can be exchanged with the breaker in compartment 11A,

(8) for 480V Unit Board 2B compartment 10D, replaces the 100A CT sensor with a 400A CT sensor and replaces the LI Amptector with a LS Amptector, adds a note that this breaker can be exchanged with breakers in compartments 10A, 480V Unit Board 1A compartment 9C, 480V Turbine Building Common Board compartments 3B and 3C, and 480V Service Building Main Board compartment 12D, and,

(9) for 480V Unit Board 2B compartment 10D, replaces the 100A CT sensor with a 600A CT sensor and replaces the LI Amptector with a LS Amptector, adds a note that this breaker can be exchanged with the breaker in compartments 10B, 480V Unit Board 1A compartment 8C, 480V Unit Board 2B compartment 10B, 480V Turbine Building Common Board compartments 1A, 1B, 5D, 8D, and 13A.

This change to non-safety related equipment does not result in a special test or experiment and has no affect on the DBA or credible failure mode. The licensing basis is unaltered by the change.

Revised TVA drawing 1-45W731, 480V Auxiliary Building Common Board is also UFSAR Figure 8.3-15. This change is minor and does not change the UFSAR text. The change is not an unreviewed safety question because the probability/possibility of occurrence or consequences of an accident or malfunction is unaffected. This change is determined to be acceptable from a Nuclear Safety Standpoint.

SA-SE Number: WBPLEE-98-081-0

Implementation Date:3/30/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN W-39988-A	ERCW Pressure Indicators

Description and Safety Assessment:

DCN W-39988-A replaces 0-PI-67-178 and 0-PI-67-18B (panel 1-L-143, IPS, El 722) , which were electronic pressure indicators, with direct reading pressure indicating gages meeting ASME XI accuracy requirements for testing. These gages are being installed in parallel with 0-PT-67-17 and 0-PT-67-18, respectively. These gages are used to measure the pressure on ERCW Headers A and B. The installed instrumentation did not meet accuracy requirements for ASME Section XI testing. Previously, temporary installation of maintenance and test equipment (M&TE) was necessary to meet the requirements.

DCN W-39988-A also adds differential pressure indicating gages 2-PDI-67-61 and 2-PDI-67-62 (1-12 ERCW tunnel, el 715. 1), in parallel with transmitters 2-FT-67-61 and 2-FT-67-62, respectively. These gages will meet ASME Section XI accuracy requirements for test instrumentation. These gages will be used to determine the flow in ERCW Headers A and B. Previously, temporary installation of M&TE was necessary to measure this parameter.

The instruments being installed by this DCN are Technical Specification compliance instruments since ASME testing is required by the Technical Specifications. The instruments are being installed in parallel with trained or train-associated instruments. The indicating function of the test gages is non-safety related, has no control function, and does not interface with any plant safety related system or function. The safety function of the gages is seismic 1(L)A - to maintain structural and pressure boundary retention during a seismic event. This mechanical pressure boundary interface is the only interface with other plant safety related equipment.

There is no adverse impact on nuclear safety. The instruments are being installed through use of a piping "T" to make the parallel connection across the existing instruments as described above. The installation will be made downstream of the panel isolation valves. During installation, the isolation valves will be closed to assure ERCW integrity. The new equipment and necessary piping and valves have been purchased in accordance with the requirements WB-DC-40-36 and N3E-934, which assures that the materials are qualified for their application. The purchasing documents also required instrumentation which would meet the ASME accuracy requirements. The materials are compatible. The instruments will be installed in accordance with N3E-934, which will assure that structural and pressure boundary integrity is maintained. The instruments will be tested in accordance with N3E-934 (visual in-service leak test) which will meet the requirements of ASME Section XI testing for instruments installed in 1 inch or under lines. This will verify that there is no leakage prior to the equipment affected being returned to service. Since the instruments have been purchased, installed, and tested to meet the safety function requirements (seismic Category 1(L)A) and ASME requirements (accuracy greater than or equal to 2% of span), and since the portion interfacing with other plant equipment is qualified for the interface, there is no adverse impact on nuclear safety.

The pressure indicators and differential pressure indicators being installed have no safety function and no control function. They are used only for testing. The only interface that these pressure indicators and differential pressure indicators have with other plant systems or components is the pressure boundary interface with the safety related ERCW system. Since the procurement, installation, and testing of the equipment has been accomplished with the necessary procedures to assure that structural and pressure boundary integrity is maintained, there is no degradation to the safety related ERCW system and no increase in probability of accident probability or equipment malfunction. The gages are purchased and installed to meet the accuracy requirements of ASME Section XI for testing of ERCW pump performance, which assures that testing of the ERCW system to technical specification requirements will be adequate and there will be no increase in the probability of equipment malfunction. The functional configuration of the safety related ERCW system is not modified by this change, the gages have no control function, and there is therefore no increase in probability of a different type of accident or malfunction of equipment. The safety related ERCW system is not adversely impacted by the modification and any potential malfunctions of the ERCW system or its equipment (pressure boundary loss) have been previously evaluated. Therefore this change does not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-98-084-1

Implementation Date: 09/27/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39928-B	Permanent cable installation for use in Containment during Refueling Outages.

Description and Safety Assessment:

This design change installs fiber optic primary containment electrical penetration feed-throughs in the penetrations for upper containment and for lower containment. The fiber utilized for the feed-throughs has a step index profile with an all silica glass core which is radiation resistant. The fiber has a polyimide (Kapton) buffer which is also very radiation resistant. The fiber optic feed-throughs provide permanent circuit paths into containment for video, audio, and dosimetry data feeds. These circuits are non-Class 1E, non 10CFR50.49 and, therefore, attenuation due to accident radiation is not considered a problem, since the devices using the feed-throughs are not required during a DBE. However, the fiber and buffer selected for this application has passed environmental qualification (EQ) testing which proves that the installation of the fiber feed-throughs will not degrade the penetration's ability to maintain containment pressure boundary.

DCN M-39928-B also makes provisions for fiber optic multiplexers which will be mounted temporarily in primary containment and remote locations outside containment by RADCON during outages for closed circuit television (CCTV), audio, and dosimetry. The multiplexers are powered by existing lighting receptacles and utilize fiber optic feed-throughs for the circuits. In order to connect the multiplexers to the feed-throughs, permanent fiber optic interconnection boxes are installed in the annulus and primary containment near the penetrations with permanent interconnecting fiber optic cables between the feed-throughs and interconnection boxes. Temporary interconnecting fiber optic cables are provided for circuit connection between the interconnection boxes and multiplexers. Temporary installation of the video cameras, monitors, audio systems, dosimeters and copper cables is used to connect those devices to the multiplexers.

DCN M-39928-B installs permanent interconnection boxes in the annulus and in upper containment to provide copper cable communications circuits between the primary containment and areas outside the Reactor Building. An existing feed-through is used for these circuits and permanent copper cables are installed between the feed-through and the interconnection boxes. Temporary cables at the interconnection boxes are installed in containment and the annulus during outages for communications.

Since the fiber optic feed-throughs are permanent EQ qualified portions of existing primary containment electrical penetrations, the consequences of a fuel handling accident in containment would be mitigated more quickly than it would be with the temporary copper cables through the mechanical pipe sleeve used for outages in the past because the copper cables would have to be cut and enough of the cables remaining in the sleeve would have to be removed to allow the installation of the blind flange to seal the mechanical penetration. Additionally, because the interconnection boxes and conduits are seismic Category 1(L) supported, the installation would not endanger any safety-related equipment during a seismic event. The new electronic equipment was purchased to the applicable requirements of TVA Standard Specification, "Electromagnetic Interference (EMI) Testing Requirements for Electronic Devices." This specification requires testing in accordance with EPRI guidelines for emissions for analog and digital electronic equipment. Additionally, the use of fiber optic cabling eliminates the potential for conducting emissions to remote locations.

The fiber optic feed-throughs are EQ qualified for accident conditions, the permanent interconnection boxes and conduits are seismically supported, the temporary multiplexers are only installed during refueling outages and the circuits utilize non-1E penetrations which are separated from Class 1E circuits. Therefore, the evaluation above shows that:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR was not increased and

2. The possibility of an accident or malfunction of a different type than previously evaluated was created and

3. There was reduction in a margin of safety as defined in the basis for any technical specification.

Based on these review results, it can be concluded that the proposed activity does not create an unreviewed safety question.

SA-SE Number: WBPLEE-98-085-0

Implementation Date: 08/29/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50017-A	Drawing Discrepancy DD 98-0053

Description and Safety Assessment:

EDC E-50017-A is a documentation only change to resolve DD 98-0053. The changes deletes annunciator multiplexer 12, channel 119 from drawings and corrects the terminals on the 24VDC codes, alarm, and paging charger.

EDC E-50017-A is also a documentation only change to the drawings for the 480 Volt transformer room exhaust fans 1A4 (1-MTR-30-244J) and 2B4 (2-MTR-30-246J). These are not safety related and are not powered from 1E power source like other fans in rooms 772.0-A6 and Al 1. Electrical control diagrams) indicate the fans 1A4 and 2B4 receive non-divisional power, but also include either an "A" or "B" (for example 1 A4-A and 2B4-B) . The "A" and "B" are not correct and are being removed from the drawings.

There are no physical change to the plant as a result of this change.

The components within this change are not important to safety and are not required to mitigate any DBA, therefore, this change does not affect nuclear safety. Although these items involve communications equipment and non-safety vent fans that are described in the UFSAR, none of these changes degrades any safety equipment below the design basis nor increases challenges to safety systems assumed to function in the accident analyses. All regulatory and industry requirements for the equipment have been met. Therefore, this change does not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-98-104-0

Implementation Date: 02/03/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50027-A	Drawing Discrepancies

Description and Safety Assessment:

This change, EDC E-50027-A, is a documentation only change to resolve drawing discrepancies.

EDC E-50027-A revises wiring drawing to correct wire numbers from 1SK69 and 1CK69 to 2SK69 and 2CK69. These wire numbers are in the circuit that annunciate the Sequoyah Line Transfer Trip Carrier low signal level and signal abnormal. Also this change corrects annunciation window number from 5068 to 51 GE and changed reference drawing. This change is in agreement with drawings.

EDC E-50027-A revises schematic diagram per DD 98-0070 to correct fuse numbers FS117 and FS118 to FS15 and FS16 respectively. There fuses are in the circuit for the 5th Vital Battery Room Temperature Transmitter 0-TT-31-487. This change is in agreement with drawings. Also this change corrects the Master Equipment List (MEL) for fuse identification.

EDC E-50027-A revises flow diagram, piping drawing and control drawing to delete temperature well. Purchase request W-6948 procured temperature control valve with a temperature element without a well.

These changes are documentation only to drawings per Drawing Deviation Program.

These changes do not involve any physical modifications to the plant, modify the safety function of any equipment, or affect. fission product barriers. The changes do not alter any DBA or operational transient analyses previously performed, and no new accidents or equipment malfunction failures are created. The changes do not affect setpoints or safety limits and, therefore, does not reduce any margins of safety as defined in the technical specifications. Therefore, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number: WBPLEE-99-019-0

Implementation Date: 08/13/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50150-A	Drawing Discrepancies

Description and Safety Assessment:

Engineering Document Change (EDC) E 50150-A resolves two DDs so that the drawings more accurately reflect the as-constructed plant configuration and to ensure that all design documents are consistent. No hardware or functional changes are being made by this EDC. This EDC corrects drawing discrepancies in accordance with DD 99-0002 and DD 994003. Specifically, this EDC makes the following changes:

Revise control diagram (Fire Protection Plan Figure 11-14) to change the functional representation of the High Pressure piping arrangement in the Office Building and Service Building. A Tee connection is added for the water supply to the Office Building Chillers and Air Conditioning Units (ACU's) and the Office Building raw service water connections. The water supply to the Service Building Chillers and ACU's is shown an a separate line from the header rather than on the same line as the sprinkler heads in the Power Stores.

Revise logic diagram (Fire Protection Plan Figure 11-24) to correct the orientation of the functional representation of check valves 0-25-561 and 0-25-650. These check valves are in the line from the strained raw cooling water header to the fire protection system in the Turbine Building. The check valve is reversed by the change to be in agreement with the direction of flow in the header. The upper tier document that is referred to for flow piping arrangement and flow direction is the flow diagram (Fire Protection Plan Figure 11-1). EDC E 50150-A revises the logic diagram to be consistent with the flow diagram.

Since these are documentation changes only and do not represent any functional, operational, or physical change to the plant, the minor changes to the above UFSAR figures by this EDC do not impact the probability of occurrence or consequences of any accident or equipment malfunction currently evaluated in the UFSAR. In addition, the documentation changes do not create the possibility of an accident or equipment malfunction of a different type than previously evaluated and does not reduce the margin of safety as defined in the basis for any technical specification. Therefore, the changes made by EDC E-50150-A do not constitute a unreviewed safety question.

SA-SE Number: WBPLEE-99-026-1

Implementation Date: 10/01/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50190-A UFSAR Change Pkg. 1595	Tornado Dampers

Description and Safety Assessment:

This change, DCN D-50190-A, (1) Provides design output authorization for Operations current normal operating practice of removing electrical power from the tornado dampers in the open position during normal/non-tornado operation, (2) justifies Operations capability to restore electrical power to the tornado dampers as a result of a Tornado Warning, and (3) adds a warning nameplate above 0-HS-31-34-A. 0-HS-31-34-A is a hand switch located on the HVAC panel 1-M-9, located in the MCR. 0-HS-31-34-A is a hand switch which operates (open or close) the tornado dampers (0-FCO-31-21 -A and -34-A). The two MCR Emergency Pressurizing fans take suction from a-common duct which is connected to the outside air through two separate intake points. These dampers have a safety function to protect the MCRHZ from the effects of a tornado by being closed by a manual action during tornado warnings and to provide an open suction pathway for MCR Emergency Pressurization as part of the Control Room Emergency Ventilation System (CREVS) after a Control Room Isolation (CRI) signal. The dampers are left open during normal operation (non-tornado warnings) with power removed since a failure of the switch could inadvertently close these dampers if they were left energized. Closing 0-FCO-31-21-A and -34-A dampers defeats both trains of MCR Emergency Pressurization, however this is allowed by the Technical Specifications for up to eight hours in the special situation of a tornado warning.

The UFSAR is being revised by UFSAR Change Package 1595 to identify the hand switch as a potential single failure and describe the operator actions to remove power to the dampers in the open position during non-tornado operation and to restore power during tornado warnings.

The failure of both trains of the MCR Emergency Pressurization Fans which could be caused by a single failure of hand switch 0-HS-31-34-A, which controls the tornado dampers, is not analyzed as a DBE. This DCN identifies the operational requirement to remove power to tornado dampers in the open position during normal operations and restore electrical power to the tornado dampers as part of preparation as a result of a Tornado Warning. This change ensures that the plant configuration is consistent with the accident analysis and equipment malfunctions analyzed in the UFSAR. Therefore, these changes do not impact the probability of occurrence or consequences of any accident or equipment malfunction currently evaluated in the UFSAR, nor does this change create the possibility of an accident or equipment malfunction of a different type than previously evaluated and does not reduce the margin of safety as defined in the basis for any technical specification. Therefore, the changes made by DCN D-50190-A do not constitute a unreviewed safety question.

SA-SE Number: WBPLEE-99-033-0

Implementation Date: 01/31/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50243-A UFSAR Change Package 1593	Thimble Tube Plugging Method.

Description and Safety Assessment:

The Incore Flux Mapping Subsystem consists of 58 incore flux thimbles which permit measurement of the axial neutron flux distribution within the reactor core by insertion of the incore instrumentation movable miniature detectors. The Bottom Mounted Instrumentation (BMI) thimbles are inserted into the reactor through thimble guide tubes mounted on the bottom of the reactor vessel. The thimbles serve as the pressure boundary between the reactor water pressure and the atmosphere. These thimbles are subject to wear causing thinning of the thimble tube walls. This change, EDC-E-50243-A, provides a method of evaluating thimble tube wall loss and a detail for plugging tubes when it is determined that they are to be taken out of service.

The BMI thimbles are in the active region of the core during normal operation and are exposed to extreme conditions of heat, RCS flow velocities and hard radiation as well as a high neutron flux. These conditions can cause wear of the thimbles which are an ANS Class II item and are a RCS pressure boundary. As such they must be monitored for wear at each cycle outage. A high degree of safety is inherent in the wall thickness of the thimbles and they will withstand RCS pressure to a wear that is equivalent to a 90% loss of wall. The amount of wear may be determined by the formula that exists. The wear formula contains an exponent that defines the shape of the wear curve that is used in the predictive life of the thimbles. As the thimbles wear in accordance with the flow, flux and temperatures ranges of the plants unique core, the curve flattens out and the percentage wear for each cycle becomes smaller. This necessitates a new value for "n" that more accurately predicts the wear for the upcoming cycle. A method of determining this new "n" factor is found in Westinghouse WCAP. The calculation of a new "n" exponent does not involve a mechanism for personnel safety to be decreased nor is the plant equipment degraded in any way. Due to the ability of a thimble tube leak to be isolated and per the leak rates reported in WCAP, a leak would not be reportable as a small break LOCA.

This change provides a detail to be used if a thimble tube is determined to require plugging. A compression-type cap will be installed on the thimble tubing just above the seal table, below the manual isolation valves. The cap is not subject to leakage the way a valve stem or packing may leak, thus providing more certain isolation in case of a thimble tube rupture. The tubing that has been disconnected from the seal table to allow for the cap installation will be left in place to act as a travel stop to impede thimble tube ejection in the unlikely of a mechanical failure. The tubing that is left in place may have a fitting (e.g. cap, plug, bolt) installed to provide more mass to aid in prevention of a thimble tube ejection.

The passive failure of the thimble is not analyzed as a DBE. This change does not change the likelihood or the consequences of a pressure boundary failure. Therefore, this change does not create or affect any credible failures.

There is no change in the function, operation, testing, maintenance or surveillance of the Incore Instrument System. The system will operate as before. The use of a compression-type cap is in accordance with design basis documents. Therefore, there is no increase in probability or consequences of evaluated accidents and malfunctions, no possibility that a different type of accident or malfunction than those previously evaluated has been created, and no reduction in technical specification safety margin has occurred. Therefore, it can be seen that this change does not involve an unreviewed safety question.

SA-SE Number: WBPLEE-99-046-0

Implementation Date: 06/28/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50284-A	Removal of Obsolete Signal Converters.

Description and Safety Assessment:

Design Change Notice, DCN D-50284 deletes signal converters 1-TM-24-73D and 1-TM-24-74D from the instrument loops controlling the seal oil air side and seal oil hydrogen side heat exchanger temperature. Temperature control valves, 1-TCV-24-73 (Air Side HX) and -74 (hydrogen side), are installed in the raw cooling water (RCW) discharge piping and are designed to regulate flow to maintain the seal oil temperature at a preset value. These signal converters, which provided an interface with the ICS, have failed and are obsolete. An evaluation of the circuits determined that the signal to the ICS computer can be produced from 1-TM-24-73A and 1-TM-24-74A across a 100 ohm resistor that is installed on temperature controllers, 1-TIC-24-73 and 1 -TIC-24-74.

The RCW system is the heat sink for cooling non-safety related plant components. The RCW components located in the Turbine Building do not require any seismic evaluation. The seal oil system is used to provide a seal barrier to prevent the escape of hydrogen gas from the generator where the rotor shaft ends of the generator extend out of the housing. The oil is maintained at a pressure higher than the hydrogen gas. The associated seal oil cooler heat exchangers maintain the oil temperature within acceptable limits. The generator seal oil system is non-safety related.

The change complies with system safety and functional requirements specified in the design criteria and system description. The removal of the subject converters does not affect proper operation of the generator seal oil cooling and the RCW system. Therefore, this change is safe and does not involve an unreviewed safety question.

SA-SE Number: WBPLEE-99-049-0

Implementation Date: 11/20/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50287-A	Drawing Deviations (DD) 99-0025

Description and Safety Assessment:

This change, EDC E-50287-A revises control diagram drawings to depict the correct annunicator window, multiplexer and channel identification for the diesel generator starting air system.

DD 99-0025 was written in order to correct drawing deficiencies on UFSAR figures for the diesel generator starting air system. This system has pressure switches that provide an alarm at each air-starting valve and on each air receiver. These switches close on low air pressure to alarm the operator of possible air starting system problems on MCR Panel 0-M-26, ACR Panel 0-L-4 and locally.

Also, EDC E-50287-A adds enhancement to control drawings to depict the annunciator window number for the Diesel Generator Starting Air alarms on ACR Panel 0-L-4.

These changes associated with this EDC are drawing revisions only and do not require any field work.

Since these are documentation changes only and do not represent any functional, operational, or physical change to the plant, the minor changes to the above UFSAR figures by this EDC do not impact the probability of occurrence or consequences of any accident or equipment malfunction currently evaluated in the UFSAR. In addition. the documentation changes do not create the possibility of an accident or equipment malfunction of a different type than previously evaluated and does not reduce the margin of safety as defined in the basis for any technical specification. Therefore, the changes made by EDC E-50287-A do not constitute a unreviewed safety question.

SA-SE Number: WBPLEE-99-057-0

Implementation Date: 8/13/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50291-A	Drawing Discrepancies.

Description and Safety Assessment:

DD 99-0032 was written in order to correct drawing deficiencies on drawing (UFSAR Figure 10.2-4) for the Control Start Temperature Recorder (1-TR-47-2). This recorder has 24 pens, with pens 3, 4, 7, 8, 17, 19, and 21 associated with the incorrect temperature elements. Also ICS points T2051A, and T2052A are labeled incorrectly as Y2051A and Y2052A.. These ICS points appear to be typographical errors.

DD 99-0037 was written in order to correct drawing deficiency on drawing to show correct point/wire numbers on red/green light circuit for high pressure stop valve.

DD 99-0035 was written in order to show that a load (FS-70-190) was deleted from 1-BKR-235-3/24-F on drawing. The load was previously deleted by a DCN.

This EDC change, revises control diagram to depict the correct temperature element associated with the correct pens for the Control Start Temperature Recorder (1-TR-47-2). This change is in agreement with plant Scaling and Setpoint Document (SSDs) and Turbine Instrumentation connection drawings.

This EDC change, also revises control diagram to depict the correct ICS computer point for (T2051A) and (T2052A). These ICS points were obvious typographical errors and Computer Termination and input/output (I/O) drawing list along with the plant SSDs were used to verify the correct points. Also discovered and corrected during the issue of this EDC is ICS point T2004A; this point was labeled incorrectly as T2004. This point is also a typographical error and Computer Termination and I/O drawing list along with the plant SSD was used to verify the correct point.

This EDC change, revises schematic diagram to correct point/wire number on red/green light circuit for high pressure stop valve FCV-1-36. This change is in agreement with the Main Feedwater Pump & Turbine 1A connection diagram.

This EDC change, revises wiring diagram 120V AC Vital Instrument Power Board 1-111 loading sheet to show load FS-70-190 as deleted from breaker 24. This load was deleted in a DCN and this drawing was inadvertently omitted from the change paper.

The changes associated with this EDC are documentation drawing revisions only and do not require any field work.

The only part of the change which affects information presented in the UFSAR is the documentation change to drawing (UFSAR Figure 10.2-4).

Since these are documentation changes only and do not represent any functional, operational, or physical change to the plant, the minor changes to UFSAR figure 10.2-4 do not impact the probability of occurrence or consequences of any accident or equipment malfunction currently evaluated in the UFSAR. In addition, the documentation changes do not create the possibility of an accident or equipment malfunction of a different type then previously evaluated and does not reduce the margin of safety as defined in the basis for any technical specification. Therefore, the changes made by the EDC do not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-99-067-0

Implementation Date: 02/25/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50311-A	Replacement of Air Conditioning System relays.

Description and Safety Assessment:

The DCN replaces AR relays with relays of a different type. The relay function remains unchanged as does the host equipment (Shutdown Board Room and Electric Board Room Chillers) operation. The ability to maintain rooms as originally designed will improve.

PERs WBPER970723 and WBPER980333 document problems with HVAC System. A sub-component, AR relay style 1457C80, does not function as needed.

A walkdown of the 6.9KV and 480V Shutdown Boards identified ten critical AR relays which will be replaced, two unused relays which will be removed, and four non-critical Fire Pump, annunciation relays which will continue to be used.

DCN documents this relay replacement and revises four affected UFSAR figures (8.3-20, -21A,-22, and -23) to: (1) delete unused relays, (2) show component identification and (3) revise the number of available relay contacts. The new relay performs the same system function and has the same failure modes as the original relay. Coil failure and contact failure are the two common causes of failure. Physical attributes, such as weight, size, and electrical burden, are slightly different, but will not adverse affect the failure mode of the host equipment. The relays are physically located in a essentially mild environment with no radiological impact.

The chiller must be operable for the following events: LOCA, SSE, Loss of Offsite Power (LOOP), Tornado, Flood, Airborne Radioactive Contamination, and FHA Outside Containment. For the SSE and LOOP, the replacement relay passed seismic qualification testing; in addition, the new relay operating characteristics envelope the original relay characteristics. Other events will not be affected by this relay replacement.

No text or table discusses or refers to the subject relays. This change results in no modification to the described facility, procedures, technical specifications, or commitment to NRC, and does not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-99-068-0

Implementation Date: 5/9/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50084-A	Removal of Abandoned Instrumentation in Unit 1 TD AFW Pump Room

Description and Safety Assessment:

DCN D-50084-A will authorize the permanent removal of previously abandoned instruments 1-PS-3-121A, B, and D, (formerly Train A), and 1-PS-3-125A, B, and D, (formerly Train B) and their associated sense lines and cables. The instruments were disconnected electrically and isolated mechanically and then abandoned in place under the scope of DCN M-38697-A prior to initial licensing of WBN Unit 1. The original design function of the abandoned pressure switches was to detect low pressure in the Unit 1 Turbine Driven Auxiliary Feedwater (TDAFW) suction line, and initiate a swapover of the suction from the Condensate Storage Tank to ERCW, using a two out of three logic for each train. Separate and redundant instrument loops (1-PS-3-139A, B, D, Train A, and 1-PS-3-144 A, B, D, Train B) now perform those functions previously intended for the abandoned instrument loops. The safety evaluation included within DCN M-38697-A evaluated the abandonment of the above listed instruments, and the use of the Motor Driven AFW (MDAFW) switches for the swapover logic, and concluded that no unreviewed safety exists.

Since the abandoned instruments described above had been previously isolated from the process and power circuit systems under the scope of DCN M-38697-A, there is no longer any DBE listed in the accident analysis which would be associated with the components to be removed by DCN D-50084-A. The only credible failure mode which could be associated with the abandoned instruments and sense lines is due to the fact that the components were abandoned in place in a Seismic Category 1 structure, in an area of other Safety-Related equipment. However, no support modifications were performed under DCN M-38697-A, and the instruments and sense lines continued to be restrained and supported per Seismic Category 1 criteria after the abandonment. The postulated failure of seismically supported components due to a DBE seismic event is not considered a credible failure, and therefore the abandoned components did not, and do not now pose a threat to the safety-related equipment located in the area.

The proposed design change introduces no increased probability of an accident or malfunction of equipment important to safety, or create the possibility for an accident or malfunction of a type different than any evaluated previously in the UFSAR since these instruments have been isolated and do not perform any type of function. The change introduces no increased radiological consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR. The permanent removal of the components does not create any new credible failure modes, and it can be concluded that the proposed change is acceptable from a Nuclear Safety standpoint, and no unreviewed safety question exists.

SA-SE Number: WBPLEE-99-085-0

Implementation Date: 12/18/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50341-A	Stator Ground Fault Protection

Description and Safety Assessment:

This DCN provides a non-safety related microprocessor based relay (Beckwith M-3430) to replace the existing electro-mechanical relay for detecting generator stator ground faults. This protective function is not a class IE safety function nor is it required to support the operation of equipment required for the safe shutdown of the plant.

In the Watts Bar Switchyard Review dated May 6,1997, Transmission Power Services recommended that neutral 100% ground fault protection be added to the main turbine generator. The existing WBN turbo generator protection scheme currently protects approximately 95% of the stator winding from ground faults and does not provide 100% stator protection. The current protection scheme will not detect faults at or near the generator neutral.

The basic protective functions and plant interfaces remain unchanged for this new equipment. The turbine will still be tripped due to relay operation which will lead to a reactor trip when operating at full power. This is the same scenario that exists for the existing protective relays. The loss of external electrical load and/or turbine trip has been previously analyzed in UFSAR Section 15.2.7 and found to be acceptable. This has been categorized as a Condition II event, which by definition, will not propagate to cause a more serious fault.

This non-safety related equipment does not provide any safety function nor is required to support the operation of any safety related equipment. The existing relay protection scheme relies on a one out of one trip logic for stator ground fault protection. This is the only protective function (stator ground fault) for which the new microprocessor relay is utilized. The failure mode of the new relay is that the relay either operates falsely or fails to operate at all. While undesirable from an economic position, these failure modes are acceptable from a safety standpoint. The current UFSAR analysis is bounding for the worst possible scenario that could be postulated from this proposed activity and this activity will not result in any new accidents or malfunctions of a type than those previously analyzed.

The relay has shown to be reliable based on its successful operation and track record with TVA's Hydro Units. The new relay has been proven not to fail due to any internal faults in such a way as to cause a false trip. In addition, loss of one or more of the 120 volt AC metering pressure transmitter (PT) sensing inputs will not generate a false trip signal. The relay is designed to detect stator ground faults on 60 Hz overvoltages or loss of third harmonic voltages and will provide a trip signal for these conditions. If the generator neutral voltage sense circuit were lost due to failure of the neutral transformer or an open circuit for any reason, then the generator ground fault protection relay would no longer be able to detect ground faults and the relay would generate a trip signal. The probability of such an occurrence is very small and this action is deemed acceptable. The protection of the generator is the primary function of this relay and its operation on loss of this circuit assures this valuable piece of equipment is not left unprotected. The current UFSAR analysis is bounding for the worst case scenario that could be postulated from the proposed activity and will not result in any new accidents or malfunctions of a type than those previously analyzed.

This DCN replaces a non-safety relative generator ground fault protective scheme that provides 95% stator ground fault protection with a microprocessor based relay protective scheme that will provide 100% stator protection. The new relay has a proven track history at TVA's Hydro Plants and the procurement package for this relay imposed suitable requirements to assure a quality product produced under an acceptable vendor's Quality Assurance (QA) Program. The addition of the new relays will not change the protective functions previously required by design basis documents. The possible failure modes have been previously analyzed in the UFSAR. This DCN will improve the level of protection for the non-safety related generator and does not involve an unreviewed safety question.

SA-SE Number: WBPLEE-99-111-0

Implementation Date:* *12/20/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50457-A	Replace Failed Positioner on Valve

Description and Safety Assessment:

DCN D-50457-A replaces failed positioner 1-POS-24-48 on valve 1-TCV-24-48. The installed positioner, a Moore Products P/N 750E232GPNNF electro-pneumatic device with an input of 10-50 MADC, is obsolete, and there are no replacement parts available for repair. The replacement is a PMV P-1220 pneumatic positioner with a 3-15 psig input. A Masoneilan P/N 8005N UP transducer with a Type 77-40 pressure regulator will also be installed to convert the 10-50 MADC control signal to 3-15 psi required by the replacement positioner.

The purpose of valve 1-TCV-24-48 is to control the flow of cooling water through the generator hydrogen heat exchangers. Replacement of the obsolete positioner will not affect the required function of the valve. The basic characteristics of the replacement positioner are unchanged except the input control signal is 3-15 psig. The additional VP transducer, 1-TM-24-48B, provides the conversion from the 10-50 MADC control signal to 3-15 psig.

Existing cables connected to 1-POS-24-48 will be disconnected and reconnected to the I/P transducer input. The mounting bracket for the existing positioner will be removed and replaced with a mounting bracket supplied with the new positioner. The existing pressure regulator will be utilized for the new positioner and relocated as recommended by the valve manufacturer. Interconnecting tubing will also be replaced as necessary to accommodate the new positioner.

The new VP transducer and associated pressure regulator, 1-PREG-024-0048/A2, will be installed on the existing control air supply tubing support located near the valve.

These instruments are not used for accident mitigation or post accident monitoring. The control and operation of tile affected valve, 1-TCV-24-48, is shown in Figure 9.2-37 but is not discussed in the UFSAR. This change will not diminish the capability of the RCW System to perform its design function. The generator hydrogen temperature control function performed by 1-TCV-24-48 is unaffected by the replacement of the failed positioner except that equipment reliability should be enhanced.

The equipment affected by this DCN performs no safety function, and does not interface with any equipment important to safety. The proposed design change does not increase the probability of an accident or the occurrence of a malfunction of equipment important to safety. The consequences of an accident or a malfunction of equipment will not be increased. No accidents or malfunctions of a different type than previously evaluated in the UFSAR are created. The proposed design change does not affect any technical specification or margin of safety identified in the Technical Specification Bases. Therefore, this change does not involve any unreviewed safety question.

SA-SE Number: WBLEE-99-115-0

Implementation Date: 02/24/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50467-A	Main Control Room Panel 0-M-27B Modifications

Description and Safety Assessment:

DCN D-50467-A changes the following items on MCR panel 0-M-27B (1) replaces the power disconnect nametag "PDO" to "PDC" on 2-HS-70-3A, (2) adds mimic line on 0-M-27B for handswitch 1-HS-65-10-A, and (3) revises the control and logic for 2-FCV-70-3 to show the valve symbol to be normally closed rather than open. This will make the control and logic drawing for the normal valve position of 2-FCV-70-3 match the mechanical flow diagram. The DCN also revises the mechanical flow diagram to add a note regarding the normal position of the valve. The main control room attributes of the switches for valves involved in this change are not discussed in the UFSAR, but the changes associated with the valves (notes and symbol) do appear on UFSAR Figures 9.2-21, SH A, 9.2-24, and 9.2-19.

PER 99-12106-000 identified that the nametag which shows the operator which position a device is in when power is removed was incorrect. This nametag is helpful to explain why both lights are not lit in normal operation.

PER 99-13147-000 identified that a very small piece of mimic related to handswitch 1-HS-65-10A was missing from panel 0-M-27B. The mimic represents the flow path and the relationship of the component which 1-HS-65-10-A controls with the flow path. The mimic will be added to the panel by this DCN.

The RHR, Emergency Gas Treatment System (EGTS) and Component Cooling System (CCS) are safety-related and are used in the mitigation of any accident. None of the nameplate changes made by this DCN affects instrumentation used for post accident monitoring, sampling, or Technical Specification compliance. This change will therefore not degrade the performance of any safety-related equipment. There are no design bases accidents affected by this DCN. Correction of the operator aids will improve plant operational efficiency because potential equipment misuses are eliminated. Even though the nameplate and mimic changes are related to safety related equipment, the changes do not affect the operation of the safety related equipment. Therefore, there will be no credible failure modes introduced by this DCN.

The operator aids (nameplates and demarcation) on panel 0-M-27B do not interfere with the operator's ability to operate safety related equipment. The proposed design change does not increase the probability of an accident or the occurrence of a malfunction of equipment important to safety. The consequences of an accident or a malfunction of equipment will not be increased. No accidents or malfunctions of a different type than previously evaluated in the UFSAR are created. The proposed design change does not affect any technical specification or margin of safety identified in the Technical Specification Bases. Therefore, this change does not involve any unreviewed safety question.

SA-SE Number: WBPLEE-99-117-0

Implementation Date: 10/26/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50393-A	Moisture Separator Reheater Level Gages

Description and Safety Assessment:

This change, DCN D-50393-A, replaces the glass sight gages and the float-type level switches for the Feedwater Heaters (Nos. 1, 2, and 4) and the MSR drain tanks. The replacement level measuring system consists of a metal-tube magnetically coupled level indicator/switch assembly. This replacement level measurement system eliminates the glass fracture failures with the existing glass sight gages and the numerous failures associated with the Mercoid float-type level switches. The magnetically coupled level switches are mounted on a separate rod located along side of the metal tube. This rod provides a mounting fixture for the level switches. These switches are magnetically actuated when the float (which travels within the metal tube) is positioned within close proximity. This design allows the switches to be de-coupled from the process temperature and thus extends the operating life of the switches.

UFSAR Sections 10.2 and 10.4.7 describes the Heater, Drains, and Vents System and the Condensate System. This change does not affect the UFSAR text or tables. However, UFSAR Figures 10.4-29, -30, -31, and -33 are affected by this change since the level measurement system is configured differently than the existing system.

The replacement metal tube/magnetically coupled level measurement system has the same failure modes as the existing level measurement system. Potential failures include 1) pressure boundary breach, 2) float malfunction (e.g. sticking, filling with water, etc.), and 3) switch contact failure (i.e., failure to close or open). The existing level switches (Mercoid) uses a float/rod assembly to actuate switch contacts. The main difference is the replacement system uses magnetic coupling between the float and the switch assembly to provide contact actuation. This design concept is commonly used with many years of reliable service. The main improvement of the replacement level measurement system is the location of the switch assembly. This switch assembly is mounted on a rod which is insulated from the process temperature. This design feature will extend the operating life of the switch mechanism. Therefore, no new failure modes are created by this change.

The Feedwater Heaters (Nos. 1, 2, and 4) high-high level switch provides an actuation signal to isolate condensate flow and extraction steam flow. This Feedwater String Isolation event would result in a loss of the condensate supply to the feedwater system. The DBEs associated with the condensate system flow involve 1) LONF, and 2) excessive heat removal due to feedwater system malfunction. Both events are incidents of moderate frequency, or Condition II events. (The Feedwater Heater level measurement system would have no impact on an excessive heater removal event). A LONF results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. The reactor trip on low-low water level in any steam generator provides the necessary protection against this event. The AFW system is used to remove stored and residual heat needed to prevent reactor coolant system over- pressurization or loss of water from the core. The subject bypass valve does not interact with the reactor protection system used to detect this event or the AFW system used to mitigate its consequences.

DCN D-50393-A, does not affect any UFSAR evaluations (accident analysis) previously performed. The consequences and probability of accidents previously performed and malfunctions of equipment important to safety are not affected. This change does not create any new failure modes. Also, 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 unreviewed safety question exist.

SA-SE Number: WBPLEE-99-118-0

Implementation Date: 10/03/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50446-A	Replacement of pressure transmitter and pressure switches.

Description and Safety Assessment:

This change, DCN-D-50446-A, replaces the existing obsolete Turbine Impulse Pressure transmitter, 1-PT-47-13, with a Rosemount Model 1151 transmitter and a power supply, 1-PX-47-13 for the transmitter. It also replaces the existing Turbine Impulse Pressure Switches, and the local switch status indicators with an electronic panel meter with local indication and four output relays. The pressure switches measure turbine impulse pressure and provide a permissive for the turbine runback circuit when turbine power (as represented by the turbine impulse pressure) exceeds the setpoint. All of the setpoints remain the same with respect to percent of turbine power. There is no change to the system function or logic.

The Turbine Impulse Pressure Transmitter, is used for electrohydraulic control (EHC) system control when "Load Control - Imp In" is selected by the control room operator. The existing Hagan transmitter is obsolete. This change will replace it with a Rosemount model 1151 GP transmitter. Currently, there is no control room indication for this transmitter loop. An input from the transmitter to the ICS is being added. None of these components perform a primary safety function. UFSAR text is not impacted by this change. However, this change does impact TVA drawings which are UFSAR figures.

This change is associated with the Turbogenerator Control System and interfaces with the MFW System and the Heater, Drains and Vents System. UFSAR Chapter 15 accident analysis identifies two Condition II events associated with the feedwater system 1) LONF, and 2) Excessive Heat Removal Due to Feedwater System Malfunction. The LONF 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 the MFW System, Heater Drains and Vents System, or Turbogenerator Controls. These systems are not credited with mitigating any UFSAR Chapter 15 events. The Turbine Runback logic has not been revised. Only the components have changed. The use of the instrument contacts remain the same. This change does not create any additional failure mechanisms. The existing Turbine Runback logic could be inadvertently actuated or fail to be actuated when needed due to random component failure. The Turbine Runback logic with the new components exhibits these same potential failure modes.

DCN-D-50446-A does not affect any UFSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. 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 unreviewed safety question exist.

SA-SE Number: WBPLEE-00-004-0

Implementation Date: 03/30/2000

Document Type:	***Affected Documents:***	***Title:***
UFSAR	UFSAR Pkg. 1617	Radiation Monitors Channel Operational Test (COT)

Description and Safety Assessment:

UFSAR Sections 12.3.4.1.3 and 12.3.4.2.6 details specific channel operational test (COT) intervals for the non-safety related area radiation monitors and the airborne particulate monitors, respectively. This level of detail is not appropriate for the UFSAR, as it is not important to the description of the plant or to the presentation of its safety analysis and design bases. The information concerning the COT intervals is given in the technical specifications, the ODCM, and the TVA Calibration Program procedure and maintenance instructions. This change does impact the Safety Evaluation Report. Section 12.4 of Supplements 10 and 12 are affected. A search of docketed correspondence turned up no additional commitments to the NRC.

This change, UFSAR Change Request 1617, removes the specific interval and instead refers to the technical specifications, the ODCM, the TVA Calibration Program procedures. This is consistent with the UFSAR treatment of calibration of other radiation monitors and instruments (e.g., sections 7.5.1.7.1, 7.6.7, and 11.4.4). The intent of this change is to allow an interval of 18 months between COTs for the radiation loops (radiation monitoring flow loops will continue to have a COT interval of three months). The interval is currently 3 months. The change substitutes a reference to the technical specifications, etc., for the numerical value of the COT interval. The COT will be performed at a frequency required by the technical specifications, the ODCM, and the TVA Calibration Program procedure and maintenance instructions.

The affected monitors do not perform a primary safety function. They are utilized for personnel protection. If a monitor detects high radiation, an alarm is generated to warn personnel in the area. The monitors can also be used by the operators as an aid in diagnosing leaks of radioactive material in the plant. They are not required to mitigate a DBA. Their failure as a result of a DBA is acceptable and does not threaten nuclear safety.

The affected monitors do not perform a primary safety function. They are not required to mitigate a DBA. This change, therefore, does not constitute a unreviewed safety question.

SA-SE Number: WBPLEE-00-006-1

Implementation Date: 11/20/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50392-A	Revise Turbine Runback logic on No. 3 Heater Drain Tank

Description and Safety Assessment:

DCN D-50392-A, revises the Turbine Runback Logic initiated by high level in the No. 3 HDT. The existing Turbine Runback logic consists of a high level condition in the No. 3 HDT coincident with a Turbine power level of 85% or greater. The HDT high level signal is initiated from a full open limit switch on the condenser bypass valve, 1-LCV-6-105B. This change replaces the valve limit switch signal with an actual HDT high level signal generated from level loop 1-L-6-26, and adds a HDT Pump discharge low flow signal permissive from existing flow loop 1-F-6-107. (A time delay is included to filter potential spurious low flow or high tank level actuation signals). The revised Turbine Runback initiation signal requires the following coincident conditions: 1) No. 3 HDT high level, 2) No. 3 HDT Pump discharge low flow, and 3) Turbine Power 2:85%. This revised Turbine Runback logic will provide a more direct measurement signal (i.e., actual No. 3 HDT level) and provide a permissive signal (low No. 3 HDT discharge flow) that minimizes potential spurious Turbine Runback condition.

This Turbine Runback circuit is part of a non-safety related feature. The equipment is located in the Turbine Building except the low flow alarm unit and the time delay relay that are mounted in the Auxiliary Instrument Room (Control Bldg Elev. 708).

UFSAR text is not impacted by this change. However, this change impacts UFSAR Figure Nos. 10.4-30, -31, -34; 10.2-3, respectively.

UFSAR Chapter 15 accident analysis identifies one Condition I event associated with Load Rejection and two Condition II events associated with the feedwater system 1) Loss of Normal Feedwater, 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, DCN D-50392-A, does not affect any UFSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. 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 unreviewed safety questions exist.

SA-SE Number: WBPLEE-00-008-0

Implementation Date:03/06/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50498-A	Drawing Discrepancies

Description and Safety Assessment:

This change, EDC E-50498-A, resolves DD 99-0096 and DD 00-0005. Each is described below:

DD 99-0096 identifies that a discrepancy exists for various UNID numbers between primary drawings and the MEL. These numbers are for DC oil pumps and pressure switches associated with the emergency diesel generators. EDC E-50498-A changes the primary drawings to match MEL. Affected drawings are 1-45W760-82-6, -6A, -6B and -6C. These are FSAR Figures 8.3-29B, -29C, -29D, and -29E, respectively.

DD 00-0005 identifies that drawing 15E500-2 (FSAR figure 8.1-2A) incorrectly depicts breakers feeding the unit 2 pressurizer heater boards as normally closed rather than normally open. This drawing is the Key Diagram for the station auxiliary power system. These breakers provide the interface points between Unit 1 and Unit 2 and are maintained open as depicted on their single line drawings (FSAR figures 8.3-18, and -19). These single line drawings indicate in a drawing note that these breakers are "...lowered to the floor (racked out) and have seismic restraints installed or removed from the shutdown board room...". Drawing 15E500-2 correctly shows these breakers as Unitl/Unit2 interface points and their downstream loads under crosshatch. The Unit 1 counterparts, heaters A-A and B-B, have no "normal" position. Rather they are positioned by operational needs at any given time, which could be either open or closed. As such, the designations for the Unit 1 and Unit 2 heaters are being removed.

There are no accidents evaluated in the FSAR which may be affected by this change and there are no credible failure modes associated with this change. Therefore, there is no unreviewed safety question.

SA-SE Number: WBPLEE-00-010-0

Implementation Date: 11/16/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50484-A	Abandons in Place SSCW Flow Transmitter and Temperature Element

Description and Safety Assessment:

DCN M-39816-B, previously issued and closed, provided for the design and installation of the Supplemental Condenser Circulating Water (SCCW) sub-system addition to the Condenser Circulating Water system. The SCCW system supplies water from the Watts Bar Reservoir to provide a source of cooler water to the existing Unit 1 cooling tower discharge flume. SCCW discharge line flow and temperature are monitored and recorded via a solar powered data logger which records the output from a flow/temperature monitoring annubar assembly mounted in the discharge line. This data is recorded to provide proof of environmental compliance of the discharge flow.

DCN D-50437-A, previously issued and closed, addressed the design and implementation of the changes to the ICS that were necessary to receive the SCCW thermal discharge monitoring system data from the Environmental Data Station (EDS) computer over the existing EDS to ICS computer data link, store the date, and display the data. One of the monitoring stations is the SCCW effluent monitoring station (Station 31) which provides temperature and flow data from the SCCW discharge line annubar sensor.

This change, DCN D-50484-A, abandons in place the SCCW discharge line flow transmitter and temperature element, and removes the solar panel and battery box. This DCN also modifies the ICS NPDES view to remove the reference to the SCCW Effluent Station, temperature element TE-27-111, and flow element FE-27-111. The ICS will continue to receive the SCCW effluent temperature and effluent flow data over the EDS to ICS data link: the difference being that the data comes from instrumentation at the Glory Hole (Station 32) which is outside configuration control.

The SCCW sub-system does not perform any safety related function. There are no design bases accidents affected by this DCN. The SCCW discharge flow and temperature data will continue to be recorded via instrumentation outside design control. It is unlikely that the flow/temperature monitoring and recording system would be down for an extended period and it is unlikely that a potential violation of the permit would occur. The risk of this unlikely occurrence is considered acceptable. Therefore, there will be no credible failure modes introduced by this DCN.

The equipment being removed from configuration control performs no safety function, and does not interface with any equipment important to safety. The proposed design change does not increase the probability of an accident or the occurrence of a malfunction of equipment important to safety. The consequences of an accident or a malfunction of equipment will not be increased. No accidents or malfunctions of a different type than previously evaluated in the UFSAR are created. The proposed design change does not affect any technical specification or margin of safety identified in the technical specification bases. Therefore, this change does not involve any unreviewed safety question.

SA-SE Number: WBPLEE-00-015-0

Implementation Date: 2/14/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50526-A	Diesel Generator Lube Oil Circulating Pump

Description and Safety Assessment:

This change replaces the motor for the Diesel Generator 1A-A Lube Oil Circulating Pump (1-MTR-82-Al -A) which is defective and obsolete. This change affects two safety-related systems: standby diesel generator system and diesel auxiliary power system.

The replacement motor is the same horsepower (1) and frame (145T) as the obsolete motor. However, the full load current increased from 1.95 amperes to 2.05 amperes and the locked rotor current increased from 8.3 amperes to 11.28 amperes. As a result of these increases in the currents, the trip setting for breaker 1-BKR-82-A1/1-A (located in compartment 2D of 480V Diesel Auxiliary Board 1A1-A) is increased from 17 amperes to 23 amperes. Additionally, there is a slight increase in motor RPM from 1135 to 1145.

The safety function of the motor is to circulate warm oil through the oil system to keep the engine in a state of readiness for an immediate start. Identical replacement motors have previously been installed on 7 of the 8 diesel generator engines for their lube oil circulation pumps. There are no Chapter 15 DBAs which may be affected by the proposed activity. The credible failure modes of this motor is not to operate.

A unreviewed safety question does not exist because the replacement motor is the same horsepower as the replaced motor and is fully qualified for safety related use.

SA-SE Number: WBPLEE-00-018-0

Implementation Date: 03/07/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50418-A	Diesel Generator Air Start Tank Pressure Switches

Description and Safety Assessment:

This change, EDC E-50418-A will allow the installation of sense line test connections on the diesel generator air start tank pressure switches (16 total). This change will add an additional fitting to the current installation. The addition of this passive component will not affect the ability of the diesel generators to perform their safety-related function as it applies to the Plant. The installation will be performed as a Safety Related Seismic 1, "Instrument Class" installation. The diesel generator air start tanks have been designed to have an operating pressure of 260 psig, with a 266 gal capacity. The new test tee, reducing bushing, connector and plug are seismically qualified, and have been tested to withstand greater than 4000 psig. The installation of the test tee connections will reduce damage/breakage of the pressure switches, and reduce future involvement for maintenance with 120V AC power supply. The installation will supply a test tee with connector and plug to allow ease of maintenance for periodic calibrations. This design change will increase the life of the pressure switches, and decrease calibration times associated with the pressure switches.

The pressure switches monitor pressure in the diesel generator start air tanks (16 total). The pressure is monitored locally, and in the MCR and Auxiliary Control Room (ACR). The diesel generator start air tanks shall maintain a minimum of 190 psig to be in compliance with Technical Specifications. The addition of a passive failure component will not affect the operation or function of the diesel generator start air tanks.

The diesel generators are required to mitigate the effects of accidents. This change to the diesel generators does not create a new failure mode, nor does it affect any credible failure modes of the diesel generators. The addition of the passive fitting is an extension of the current pressure boundary of the line. The installation of test connections does not impair the diesel generator's ability to start and supply emergency onsite power to the safety related loads.

In this safety evaluation the credible DBEs associated with the diesel generators ability to perform their required safety related function will not be discussed. Instead, the ability of the passive fitting to retain its position and pressure for which it is designed will be evaluated. It is understood that the diesel generators are imperative to mitigating the effects of accidents such as an SI signal, a loss of voltage signal, or a degraded voltage signal.

UFSAR Sections 9.0, 9.5.6, and 9.5.6.2 describe the Auxiliary power system along with a description for the diesel generator starting system. This change does not affect the UFSAR text or tables. However, UFSAR Figure Nos. 9.5-25, 9.5-25A, 9.5-25B and 9.5-25C are affected by this change. The addition of a note to allow the installation of a test connection will be added to these figures. A UFSAR Change Package is not required for UFSAR Figure changes that are also TVA issued drawings.

The addition of this passive component does not create any additional failure modes. The only credible failure mode associated with this installation is a pressure boundary breach. Each air start tank is designed and fabricated per ASME Boiler and Pressure Vessel Code Section VIII, tested to 375 psig, with a working pressure of 260 psig and 266 gallon capacity, and located on the auxiliary skid adjacent to each diesel generator. The fittings that may be installed have been seismically qualified and tested to greater than 4000 psig.

The pressure switches associated with the start air tanks monitor the air pressure in the tanks. The tanks provide the required air pressure to allow the pinion gear to engage in the starting motor for the diesel generators. The addition of the test connection is a passive component and does not affect the diesel generator from performing its design function or ability to mitigate accidents.

SA-SE Number: WBPLEE-00-018-0 ***Implementation Date:*** 03/07/2000

This change, EDC E-50418-A, does not affect any UFSAR evaluations (accident analysis) previously performed. The consequences and probability of accidents previously performed and malfunctions of equipment important to safety are not affected. This change does not create any new failure modes. Also, the technical specification is not affected. This change is in compliance with safety requirements as specified in the design basis documents. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed changes is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number: WBPLEE-00-24-0

Implementation Date: 4/12/2000

Document Type:	***Affected Documents:***	***Title:***
ODCM	ODCM Revision 3	Radiation Monitors Channel Operational Tests Frequency

Description and Safety Assessment:

ODCM Tables 2.1-1 and 2.1-2 detail specific COT intervals for the liquid and gaseous effluent monitors, respectively. The current interval between tests is 3 months. This change, Revision 3 of the ODCM, will revise the interval to nine months. According to the Radiation Monitoring Design Criteria, WB-DC-40-24, none of the monitors perform a primary safety function are required by the Technical Specifications.

Liquid Effluent Monitors
O-RE-90-122 - Waste Disposal System Liquid Monitor
O-RE-90-133, -134, -140 -141 - Essential Raw Cooling Water Effluent Monitors
O-RE-90-212 - Turbine Building Sump Discharge Monitor
O-RE-90-225 - Condensate Demineralizer Regenerant Waste Discharge Monitor
1-RE-90-120, -121 - Steam Generator Blowdown Liquid Monitors

Gaseous Effluent Monitors
O-RE-90-101 - Auxiliary Building Vent Monitor
O-RE-90-118 - Waste Disposal System Gas Effluent Monitor
O-RE-90-132 - Service Building Vent Monitor
1-RE-90-119 - Condenser Vacuum Pump Exhaust Normal Range Noble Gas Monitor
1-RE-90-400 - Unit 1 Shield Building Vent Monitor
2-RE-90-400 - Unit 2 Shield Building Vent Monitor

The subject radiation loops are comprised of several components. The loops include ratemeters with local and/or main control room indication, recorders, local audible alarms, main control room alarms, etc. No part of these loops performs a primary safety function. The demonstrated accuracy calculation for each of these radiation loops has determined that in order to maintain the accuracy of the radiation monitors, they must be calibrated every eighteen months (plus 25%). There are no other calibrations or functional checks required by the calculation. Changing the COT interval to nine months is conservative with respect to the demonstrated accuracy calculations. Additionally, calculation performed a review of radiation monitor loops to determine the reliability of the components. None of the components of the radiation loops were found to be outside the "as found" values. In fact none of the components of the radiation loops were found outside the "as left" values.

An evaluation of the performance of the radiation monitor loop components was performed. Of the components in the radiation loops that were found outside the "as found" values, a majority of them were related to iodine detection channels of the Auxiliary Building Vent Monitor, the Containment Atmosphere Monitors and the Service Building Vent Monitor. Iodine channels are not a subject of this change. Of the remaining components that were outside the "as found" values, most were associated with the local indicators or the loop recorders. The local indicators are not typically used for determining radiation in an area, and the recorders are used for trending. Chemistry uses the ICS computer (unless unavailable) to obtain any required radiation monitor reading. Any inaccuracy in the local indicator or recorder will not affect the function of the monitors as they are utilized for monitoring effluents. The utilization of the recorders does not require a precise value in order to provide trending.

For these radiation monitors the most important attribute is the ability to alarm for a high radiation signal and perform any required control function to prevent an improper effluent release. None of the alert or high radiation alarms were found outside the "as found" values. The computer was found to be outside the "as found" value only once in 240 COTs. For the most important attribute of these effluent radiation loops, this evaluation determines that the gas and liquid effluent monitors within the scope of this change are and will perform reliably. The radiation monitors will continue to function as required to enable the plant to monitor the effluent release pathways. The radiation monitoring system will function in the same manner as it did before this change. The affected monitors do not perform a primary safety function. They are not required to mitigate a DBA. This change, therefore, does not constitute a unreviewed safety question.

SA-SE Number: WBPLEE-00-029-0

Implementation Date: 10/19/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50569-A UFSAR Chg Pkg 1634	Elimination of LO-LO Level Turbine Trip Function

Description and Safety Assessment:

DCN D-50569-A eliminates the LO-LO level turbine trip function of 1-LS-47-105. The level switch currently provides a turbine trip on LO-LO MTOT level, annunciation in the MCR and ICS computer point indication. This DCN will eliminate the turbine trip function, but will retain the alarm and ICS input. This change is being implemented to reduce spurious BOP single point trip failures of the plant and associated challenges to safety systems.

This change makes the necessary wiring changes in the Instrument rack, 1-R-71, to delete the turbine trip function from the associated relay. It will also make changes to the in/out list nomenclature and re-program the computer points as applicable. As a result of eliminating the turbine trip function for the level switch, the alarm will be re-located from the first out annunciation panel on 1-M-4 and re-located to the Turbo-Generator Grouping panel on 1-M-2 annunciators.

UFSAR text in Section 10.2.4 is affected by this change and is contained in UFSAR Change Package 1634. UFSAR Figures 10.2-1, 10.2-2 and 10.2-3 (TVA Drawing Nos. 1-45W600-47-2, 147W610-47-1 and 1-47W610-47-2; respectively) are also affected by this change and are also located in the subject DCN.

DCN D-50569-A, does not affect any UFSAR evaluation (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunctions are created. Technical specifications are not affected. This change is in compliance with safety requirements as specified in the 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 unreviewed safety questions exist.

SA-SE Number: WBPLEE-00-035-0

Implementation Date: 08/11/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50543-A UFSAR Chg. Pkg. 1631	This change allows the Steam Generator Blowdown (SGBD) flow to be reduced.

Description and Safety Assessment:

DCN D-50543-A, replaces the SGBD flow control valve, 1-FCV-15-43, and valve controller and makes changes to associated flow instrumentation. The low SGBD flow is needed to reduce condensate inventory loss since the discharge flow path is aligned to the Cooling Tower Blowdown (CTB) system. The SGBD is used for controlling the water chemistry in the secondary side of the steam generators. The SGBD system removes contaminants introduced into the steam generators by the feedwater system or fission products that may leak into the steam generators via steam generator tube leaks. This system operational alignment requires the existing flow control valve to drop pressure in excess of its design limits prior to cavitation. The replacement valve is designed to handle large pressure drops without incipient cavitation damage.

The SGBD flow control valve will be positioned using a manual loading station controller instead of the existing automatic/manual (setpoint) controller. The use of a manual controller is acceptable since the SGBD flow will remain constant since the valve's upstream pressure (steam generator pressure) and downstream pressure (CTB) are relatively constant during normal operation. Additionally, the SGBD individual flow transmitters will be re-scaled to a span of 0-95 gpm instead of the current span of 0-120 gpm. This is necessary to provide higher resolution at lower flow values. Also, the associated orifice plates will be replaced with ones that provide a higher differential pressure (for any given flow value). The local SGBD individual flow indicators will be removed since the associated transmitters will provide all needed SGBD flow information to the ICS. The existing SGBD common header flow transmitter will be replaced with a local flow indicator with the same scale of 0-400 gpm. This transmitter is not required since total SGBD flow can be obtained from the ICS by summing the individual flow values. An ultra-sonic flow indicating transmitter will replace local flow indicator to provide SGBD effluent flow information to the main control room via the ICS. This new flow channel will provide plant personnel with convenient, continuous effluent flow information needed to meet ODCM monitoring requirements.

UFSAR Sections 10.3.5 and 10.4.8 will be revised to remove the normal SGBD flow values. The SGBD flow values will vary to meet system operational needs, therefore to specify a single normal flow value would not represent all plant operating conditions. The maximum and minimum SGBD flow values will remain specified. This change also impacts UFSAR Figure Nos. 10.3-2, 10.3-3, 10.3-3 Sheet A, and 10.4-24.

UFSAR Chapter 15 accident analysis identifies two accidents related to SGBD flow rate assumptions. These accidents are Postulated Loss of AC Power to the Plant Auxiliaries and Postulated Steam Line Break. An assumed SGBD flow rate is established for purposes of determining environmental consequences of these accidents. The established flow rate (i.e., 25 gpm) is a nominal value and does not dictate an actual flow requirement. Therefore, this change that allows the SGBD flow rate to be lowered to 50 gpm total does not adversely affect this assumed value.

This change does not create any additional equipment failure modes that affect SGBD operation. The replacement flow control valve fails closed on loss of motive power. This is the same failure mode as the existing flow control valve. This change does not involve any safety related electrical equipment or affect any safeguards equipment. The SGBD flow measuring channels do not perform any control actions. The orifice plate replacement associated with the individual SG blowdown lines are part of a safety related, seismic Category I piping system. The replacement orifice plate fabrication and installation requirements are in compliance with the nuclear safety-related quality assurance program. The orifice plate replacements do not create any new failure modes. Technical specification is not affected. This change is in compliance with safety requirements as specified in design basis documents and process flow monitoring requirements specified in the ODCM. 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 unreviewed safety question exists.

SA-SE Number: WBPLEE-00-037-0

Implementation Date: 08/29/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50483-A	Delete Continuous Air Monitors and Install Portable CAMs

Description and Safety Assessment:

This change, DCN-D-50483-A, deletes the following Continuous Air Monitors (CAMs): O-RE-90-16, O-RE-90-138, 1-RE-90-14, and 1-RE-90-62; and installs portable CAMs in the Unit 1 Hot Sample Room and the Waste Packaging Area. These monitors are non-safety related. The CAMs do not perform a primary safety function and are not required to mitigate a DBA.

O-RE-90-16 provides real-time detection capability for particulate radioactivity in the Decontamination Room. O-RE-90-138 provides real-time detection for the Waste Packaging Area. 1-RE-90-14 provides the same capability for the Unit 1 Hot Sample Room and 1-RE-90-62 provides it for the Reactor Building Lower Compartment Instrument Room.

The monitors are used for personnel protection in their respective rooms. The monitors have beta detectors that will generate a local alarm and an alarm in the MCR on high radiation. The local alarm notifies any personnel in the area of the high radiation and the MCR alarm allows the operator to take the appropriate action.

The Decontamination Room is normally unoccupied and has no ongoing work activity that would require the continuous coverage of a permanently installed airborne monitor. There are no processes in the room that would contain radiation such that there could be a breach of the pressure boundary causing a high radiation alarm. Site RADCON would monitor, as required, when site personnel are working in this room in order to maintain as low as reasonably achievable (ALARA).

The Waste Packaging Area is normally unoccupied. Personnel enter the area to perform work activities related to packaging and processing of radioactive waste. The monitor serves to warn personnel working in the area that there is a high level of airborne radioactive material. It also serves to notify the control room of a high radiation condition when the sample room is unoccupied. This monitor provides compliance, in part, with 10 CFR 50, Appendix A, General Design Criteria (GDC) 63. In order to maintain ALARA principles and compliance with GDC 63, a portable CAM will be installed in the Waste Packaging Area to replace the deleted permanent in-plant monitor. The portable CAM, in conjunction with personal electronic dosimeters, would provide adequate protection from high radiation occurrences.

The Unit 1 Hot Sample Room, although normally unoccupied, routinely has personnel entering the room to perform work activities related to the sample panels in the room. The monitor serves to warn personnel working in the sample room that there is a high level of airborne radioactive material. It also serves to notify the control room of a high radiation condition when the sample room is unoccupied. In order to maintain ALARA principles, a portable CAM will be installed in the Unit 1 Hot Sample Room to replace the deleted permanent in-plant monitor. The portable CAM, in conjunction with personal electronic dosimeters, would provide adequate protection from high radiation occurrences.

The Reactor Building Lower Compartment Instrument Room is also normally unoccupied but has periodic entries for work activities. The monitor is used to aid in determining if the Reactor Building can be entered safely. The Reactor Building is not normally accessible to plant personnel. Entry into the Reactor Building is closely controlled by RADCON, such that ALARA principles are maintained.

The affected monitors do not perform a primary safety function. They are not required to mitigate a DBA, nor are they involved in the prevention or limitation of off-site releases. This change, therefore, does not constitute a unreviewed safety question.

SA-SE Number: WBPLEE-00-062-0

Implementation Date: 02/07/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50454-A	Electrical Services to Main Office Building Modifications

Description and Safety Assessment:

DCN D-50454-A provides documentation for an interface with permanent plant facilities restating front modifications to the Main Office Building. This DCN will provide electrical service for two new prefabricated buildings to be installed and to a new security guard watch tower located on the roof of the Main Plant Office Building. The two prefabricated buildings are detached from plant structures and will be installed under the Facilities Work Request process per Business Practice (BP)-239. One of the new buildings is to serve as a meeting area and the other new building is to serve as a vending area. The Main Office Building is listed in Appendix A of BP-239.

The electrical service for the two new prefabricated buildings will be supplied from the Service Building Main Board through a new transformer and new distribution panel located outside the Main Office Building. The Service Building Main Board is not addressed as an exclusionary system as defined by BP-239 and will therefore require design output and configuration control to support the modifications being made by the Facilities Work Request process.

The new security guard building electrical service is to be supplied from Office Building Vent Board Number 2 through a separate transformer located in the Main Office Building at column D3, Elevation 744.0. An additional new transformer and lighting distribution panel, located at column D3, Elevation 744.0 and supplied from Office Building Vent Board Number 2, will be utilized to provide electrical service for future Main Office Building lighting and receptacle upgrades. The Main Office Building is an exclusionary structure and these future modifications will be performed under the BP Facilities Work Request Process. However, the Office Building Vent Board No. 2 is not listed as an exclusionary system by BP-239 and requires design output under the Site Procedures and Process (SPP)-9.3 process.

Communications circuits to the new prefabricated buildings are controlled and installed by Telecommunications personnel in conjunction with operations. These communications circuits and their cables are not documented nor entered into the WBN Computerized Cable and Conduit Routing System in accordance with BP-239. This DCN provides for the installation of unscheduled conduits for these communications circuits.

The WBN UFSAR Section 9.5.2, Plant Communications System, commits to the installation of redundant code, alarm, paging speakers with redundant signal and power sources ("A" and "B" sources) throughout the plant and to a reliable three hour power supply for both sources. The code, alarm, and paging intraplant communications system will be extended into the new vending area and conference room buildings. An "A" source and a "B" source speaker will be installed in each building to provide accountability/evacuation alarms. The signals for these speakers will be derived by splicing into signal circuits for existing speakers 0-CSPR-252-11 ("A" source) and 0-CSPR-252-19 ("B" source). The power supplies for the speakers will be taken through an uninterruptible power supply (UPS) which will meet the requirement to provide a three-hour backup power supply upon loss of normal ac power. The UPS will be powered from a spare breaker in the local 120 VAC distribution panel in the new vending building.

The only safety concerns associated with this design change is in relation to personnel safety and equipment damage. Plant practices and procedures are designed to comply with applicable OSHA and other industry standards. These codes and standards are intended to ensure personnel or equipment hazards are minimized. The proposed modifications do not address any design or functional requirements of any safety or quality related components addressed in the UFSAR. The proposed modification revised UFSAR Figures 2.1-5 and 9.2-29A. The changes being made to this figures affect interfaces with exclusionary areas as defined by BP-239. The proposed modification adds a note to Figure 2.1-5 to reference the vendor contract number and drawing series.

SA-SE Number: WBPLEE-00-062-0 ***Implementation Date:*** 02/07/2001

For UFSAR Figure 9.2-29A, a potable water line is being added to the line that currently provides water to the emergency shower and eyewash in the caustic chemical area in the yard. The UFSAR commitment for code, alarm, and paging speakers with redundant signal and power source with a reliable three hour power supply is met. Additionally, the proposed modification will provide additional structures, meeting space, vending area and electrical distribution capabilities for future modifications to facilities defined as exclusionary in the BP or as determined by the appropriate TVA organizations and are not a requirement of any Design Basis Documents. Since these components do not involve any structures, systems, or components required for safe shutdown or DBAs, an unreviewed safety question does not exist.

SA-SE Number: WBPLEE-00-113-0

Implementation Date: 11/21/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50861-A UFSAR Chg Pkg 1666	NRC Emergency Notification System (ENS) Phone Lines

Description and Safety Assessment:

The WBN site telecommunications group is rewiring the connection of the NRC ENS phones from the connection to the NRC supplied direct lines and is connecting them to TVA's telephone switching system. This is being done as a cost saving measure for the NRC per NRC Regulatory Issue Summary (RIS) 2000-11, "NRC Emergency Telecommunications System." The only change for the NRC will be new telephone numbers assigned to the WBN ENS lines. There will be one operational change to the phones. The old system required the operator to lift the telephone and dial an area code and number to reach the NRC. With the revised system the operator will dial 9-1-area code and number to reach the NRC. The only physical work required is lifting the phone lines coming to the Node 2 communications building at the northwest corner of the protected area from the Bell Hut near the warehouse facility and re-landing them on TVA's telephone switching frame.

EDC E-50861-A provides documentation for the functional connection of the NRC ENS phones to TVA's telephone switching system. The existing telephone system circuits at WBN are not under configuration control and the system configuration is controlled by the site telecommunications group. The ENS phone system is to be configured and maintained by the Telecommunications group similar to the regular plant telephone system.

This change includes UFSAR Change Package 1666 which changes the ENS description in the WBN UFSAR. Additionally, this EDC will revise Design Criteria, WB-DC-00-3, Technical Support Center (TSC), and N3-250-4003, Automatic, Manual, And Public Telephone System, to reflect this updated Telecommunications System for the NRC. It was also noted in the review of WB-DC-00-3 that it was implied that the TSC phones had the ability to access the plant voice paging system. This is not true and the document was revised to say that "some" of the phones had this capability. Finally, a new figure will be created to document the functional connections of this system and added to N3-250-4003.

The WBN telephone system performance is unaltered by this change, it will maintain its intended function 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, from a Nuclear Safety standpoint this modification is acceptable.

Revising the NRC ENS phone lines to use TVA's phone system instead of the NRC provided leased lines does not involve an unreviewed safety question. The telephone system is not an accident or anticipated operational transient initiator nor is it credited for mitigating any accidents or malfunctions. This system was installed to provide a reliable communication path to the NRC during emergency conditions. TVA's phone system has proven reliable in daily operation since 1989. This system is available for the same events as the current NRC leased lines.

SA-SE Number: WBPLEE-01-004-0

Implementation Date: 03/29/2001

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50902-A	Fire Protection Report (FPR) Revision 15

Description and Safety Assessment:

EDC E-50902-A involves revising design output drawings to reference the appropriate sections of the FPR. The Fire Protection Report, Revision 15 is being issued in conjunction with this EDC. EDC 50902-A expands the Multiple High Impedance note to reference operational restrictions. This note applies to each of the originally identified breakers.

GL 86-10 states "... simultaneous high impedance faults (below the trip point for the breaker on each individual circuit) for all associated circuits located in the fire area should be considered in the evaluation of the safe shutdown capability." To show compliance, a calculation for Appendix R - High Impedance Fault Analysis, was issued.

Multiple high impedance fault analysis found that GL 86-10 could be met, without restrictions if several 480V Shutdown and Reactor MOV Board breakers were placed in the de-energized position. A note was added to applicable single lines drawings to align these breakers in the open (de-energized) position. No compensatory measures or alternative plans were identified.

Because of the additional alternate feeder issue, their use, although permitted, must follow the existing technical specification limitations. The five breakers that may be energized are for the 480V plug receptacles (outlets) and crane. Energizing these breakers was and is acceptable provided the duration is short and/or compensatory measures are taken. Based on the FPR, it was determined that these breakers could be energized with operational limits.

Design Bases Accident is an Appendix R fire which is discussed in the FPR. Credible Failure Modes is loss of shutdown capability due to a Multiple High Impedance Fault from an Appendix R fire engrossing either or both redundant shutdown paths. To ensure availability of both redundant paths, an analysis was successfully performed based on certain identified breakers being open (de-energized). With the breaker closed (energized), specific operating requirements apply. The existing FPR documents this operating requirement. In noting on the single lines that energizing identified breakers may be occasionally required, the operating requirement was not referenced. This EDC does not change any credible failure mode, but identifies that an operational limit existed in the issued FPR.

The DBA of concern is an Appendix R fire which is discussed in the FPR. EDC E-50902-A involves revising the FPR and design output drawings to cross reference each other. It was previously documented that the use of loads on a temporary, limited basis, e.g., maintenance activities is permitted.

This EDC does not entail a special test or experiment. It does not involve new procedures or instructions or revisions to procedures or instructions that are addressed in the UFSAR. No UFSAR text, tables, or graphs will be impacted by this change. UFSAR Figures 8.3-20, -20A, -21, -21A, -22, -22A, -23 and -23A will be updated through normal UFSAR drawing update. Also, it does not impact any requirements or capability of any system protecting safety related structures, system or components. There are no associated UFSAR, safety evaluations, design criteria, technical specifications or technical requirement impacts affected by this change. This change is acceptable from a nuclear safety standpoint and therefore, does not involve an unreviewed safety question.

SA-SE Number: WBPLEE-01-008-0

Implementation Date: 03/22/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50957-A	Revise setpoint of temperature indicating controller.

Description and Safety Assessment:

DCN D-50957-A, changes the setpoint for temperature indicating controller (TIC), 1-TIC-2-329A. The existing control setpoint is 265 °F. This DCN will revise the setpoint to a value of 230 °F. This controller is associated with flow control valve 1-FCV-2-329A. This valve is used to regulate condensate flow through the 1st Stage SGBD heat exchanger. The subject controller monitors condensate discharge temperature and sends a pneumatic signal to the control valve to maintain the condensate discharge temperature at the established setpoint.

The 1st Stage SGBD heat exchanger condensate control valve is used to regulate the condensate flow through the heat exchanger. This is accomplished by monitoring the condensate discharge temperature and providing a control signal to maintain the discharge temperature at the desired setpoint. This change lowers the setpoint from 265 °F to 230 °F. This change will cause an increase in condensate flow for a given SGBD flow and will cause a slight reduction in heat transfer to the condensate system. The DBEs associated with the condensate system flow involve: 1) loss of normal feedwater, and 2) excessive heat removal due to feedwater system malfunction. Both events are incidents of moderate frequency, or Condition II events. The subject condensate valve setpoint change does not cause or affect either of these events. Also, the subject setpoint change does not introduce any new failure modes or affect any existing failure modes. The excessive heat removal event is caused by a malfunction of one or more feedwater regulating valves. The condensate valve does not interact electrically or mechanically with the feedwater regulating valves. However, a failure state that results in a closed condensate valve is a low risk for causing a loss of normal feedwater event since a small portion of the total condensate flow is routed through this path (maximum flow is approximately 1650 gpm out of a total of 19,000 gpm). A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. The reactor trip on low-low water level in any steam generator provides the necessary protection against this event. The AFW system is used to remove stored and residual heat needed to prevent RCS over-pressurization or loss of water from the core. The subject condensate valve does not interact with the reactor protection system used to detect this event or the AFW system used to mitigate its consequences.

This change, DCN D-50957-A, does not affect any UFSAR evaluations (accident analysis) previously performed. The consequences and probability of accidents previously performed and malfunctions of equipment important to safety are not affected. This change does not create any new failure modes. Also, the technical specification is not affected. This change is in compliance with safety requirements as specified in design basis documents. The changes to the plant thermal cycle for various SGBD operational modes is within the rating thermal rating of the turbo-generator. 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 unreviewed safety question exist.

SA-SE Number: WBPLMN-96-115-0

Implementation Date: 10/27/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-38924-A Fire Protection Report	In-situ Combustible Loads

Description and Safety Assessment:

This change accounts for items such as Security/Operations Workstations, trash cans/bags, Hazardous Material Spill Kits, ladders, cabinets, etc. that are present during normal plant operation and considers them to be in situ combustibles. The incorporation of the combustible loading of these items into drawing 45W893-35 will eliminate the need to track these components through the transient combustible load permit program. This is an administrative change only and does not involve any field work. This change maintains WBN's commitment to account for in situ and transient combustibles. The additional combustibles resulted in the fire severity of some rooms to change from insignificant to low and this causes the Fire Protection Report to be changed. This change does not impact the fire hazards analysis nor result in any additional fire protection measures.

The change in value to the in situ combustible was small and did not require any additional fire protection measures. The only rooms that changed fire severity rating (i.e., from "insignificant" to "low') are located in the Auxiliary Building where the fire barriers are minimum 2-hour rated. None of the increases in combustible loads caused the fire severity to exceed a one hour fire load. The one room in the Diesel Generator Building that changed rating was the toilet which is considered as part of the Corridor. The barrier rating of the Corridor is 3-hour and the combustible load in the toilet is approximately 6 minutes. Therefore, the increase in in-situ combustible loads are minor; does not impact the fire hazards analysis or fire safe shutdown; and does not require any additional fire protection measures.

Appendix R requires that a fire be postulated even if there are no combustibles in the room and that everything in the room of fire origin be considered damaged. The quantity of combustibles does not increase the probability of a fire or malfunction of equipment. Therefore, there is no increase in the consequences of a fire previously evaluated. Since a fire is already postulated, this does not create a possibility for a new or different kind of accident previously evaluated. This does not affect the technical specifications, therefore there is no reduction in the margin of safety and as such does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-98-040-0

Implementation Date: 11/03/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39826-A	Makeup Water Treatment System Pump and Motor Replacement

Description and Safety Assessment:

This safety evaluation addresses the replacement of the existing Acid Pumps and Motors and Caustic Pumps and Motors with new pumps and motor assemblies (Goulds Pump, Model No. 3171, size: 1" outlet X 1-5" inlet X 6" impeller, each pump supplied with a General Electric Motor, Model No. 5KS213SSP101 D9). These pumps are used to supply acid and caustic to the Makeup Water Treatment System.

Revised flow diagram to show the design requirements for the Acid and Caustic Pumps (25 gpm @ 115 Ft hd).

Revised the wiring diagrams to show the actual Acid and Caustic motor horse power rating.

Revised the System Description N3-28-4002, Rev.1 'Makeup Water Treatment Plant" Sections 3.2.17 and 3.2.18 to describe the Acid and Caustic Supply pumps.

Replacement parts for the existing pumps are not compatible after 1/1/98 for Goulds Pump Model No. 3171.

This system's associated components, piping, and valves are located in the Chemical Storage Building. This equipment does not perform a primary safety function, are installed in a non seismic structure, and are not used during any accident. The Chapter 15 accident analyses does not identify any failure that is associated with revising drawing 1-47W834-2. This change and equipment are not associated with increasing the consequences of an accident previously evaluated, and is bounded by the existing analyses. This DCN does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the conclusions that this change is safe and does not constitute an unreviewed safety question.

UFSAR does not identify any equipment faults which could occur as a result of this change. Also, this change is not associated with the protective features used to detect and mitigate the effects of any events. The equipment involved in the change does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the UFSAR. This DCN does not change or affect the design basis for any system that is important to safety,

These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBA, all safety related equipment is expected to operate as designed to limit the consequences of the DBA. No new potential single failures of existing components will occur as a result of this documentation change only DCN. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. These system's associated components, and piping do not perform any accident mitigation function. This equipment is not used in the mitigation of any accident/malfunctions and does not change the radiological consequences of an accident previously evaluated 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.

The performance requirements are not impacted by these changes, the system design and functional requirements will remain the same, no UFSAR text or tables are affected by this change. None of the changes are associated with nuclear safety, either implicitly or explicitly. This change does not involve procedures that affect system operational characteristics described in the UFSAR. The change does not impact compliance with the technical specification. The change does not conflict with or affect a process or procedure outlined, summarized, or described in the UFSAR. No other nuclear safety considerations are impacted. Therefore, there are no unreviewed safety questions.

SA-SE Number: WBPLMN-98-044-1

Implementation Date: 03/01/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39947-A UFSAR Change Pkg. 1520 S2	ABSCE Permissible Leakage

Description and Safety Assessment:

Revision 1 added an item to the UFSAR Change Package 1520 S02 Section 6.2.3.2.1, which inadvertently stated ABSCE permissible leakage as 9300 cfm instead of 9900 cfm. UFSAR change corrected this number.

This safety evaluation addresses UFSAR changes submitted under the UFSAR Change Package 1520, and design document changes effected by DCN S-39947-A. These changes were identified during the UFSAR re-review project to provide consistency among the UFSAR, the system description documents, the design criteria, calculations, and design drawings. The changes also streamline text to improve its readability; delete repetition within and among documents; correct the obvious grammatical errors and omissions; and delete any unnecessary, or superfluous information. None of the changes affect the design bases of systems/equipment, or their functional/operational characteristics. These are documentation-only changes, which do not impact the physical plant or any operating procedures. Most of the these documentation changes are minor changes, which are implemented from marked copies of the UFSAR, system descriptions, etc., and not specifically listed in WBPLMN-98-044-0. However, many of the changes, which are not deemed "minor" are specifically listed and evaluated in the Safety Assessment section. Any non-minor discrepancies that were discovered during the UFSAR re-review process were either corrected, as described above, or addressed by other programs (e.g., Corrective Action Program, etc.). Since the UFSAR changes and/or the implement documentation-only changes, not affecting the design bases of systems/equipment. or their functional/operational characteristics, these changes do not constitute a unreviewed safety question.

SA-SE Number: WBPLMN-98-104-0

Implementation Date: 08/31/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50022-A	Drawing Discrepancy 98-0060

Description and Safety Assessment:

DD 98-0060 has identified a misleading component description for the 480 Volt Transformer Rooms. These exhaust fans are labeled as "Exhaust Fans 1A-A", which implies that the fans are backed by trained power. All of the fans are safety related and backed by trained power except for 1-FAN-030-0244J, in Room 1A and 2-FAN-030-0246J, in Room 2B, which are not safety related and are not powered by backed power.

The flow diagram is being revised by ECN E-50022-A to correct the fan designations for the four 480 Volt Transformer Room. Correcting these designations will not affect the identification of these fans nor will it affect the safety class designation.

The MEL will be revised to remove the "-A" and "-B" from 1-FAN-030-0244J and 2-FAN-030-0246J UNID description, respectively. These exhaust fans are not safety related and are not backed by trained power.

There are no physical changes to the plant as a result of this design change.

With the exception of the fans which cool the common board transformers, (1-FAN-030-0244J and 2-FAN-030-0246J) the 480V transformer room ventilation system is a primary safety-related system and is designed to operate during and after all DBEs and DBAs except loss of all AC power.

The primary safety function for these fans is to maintain the room temperatures within the specified environmental limits to ensure that the primary safety-related equipment located in the rooms is operable after a DBE. There are no credible failure modes affected by these changes.

The subject documentation changes are strictly administrative and do not directly or indirectly impact any safety analysis that forms the basis for any Technical Specification. Therefore, no technical specification changes will be required due to implementation of EDC E-50022-A. Due to the administrative nature of the subject changes, it has been determined that they:

- will not create the possibility of a new type of accident or equipment malfunction not previously analyzed in the UFSAR or change the frequency category of any analyzed event to a higher frequency category,
- will neither increase the probability nor the consequences of an accident or equipment malfunction already evaluated in the UFSAR,
- do not infringe on any margin of safety defined in the technical specifications, and
- do not involve modifications to any radwaste system or involve any special tests or experiments.

Based on the results of this safety evaluation, it is concluded that the subject modifications do not involve an unreviewed safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-98-106-0

Implementation Date: 10/27/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50040-A	Resolves discrepancies for sixteen valves in the Chemical and Volume Control System.

Description and Safety Assessment:

This safety evaluation addresses the scope of the EDC E-50040-A, which resolves DD 98-0062.

The DD identified discrepancies between the flow diagram, the walkdown data, and the vendor drawings for valve 1-DRV-62-632. While correcting this specific valve, fifteen other valves were found to be in error on the flow diagram.

The EDC identifies the discrepancies between the flow diagram (1-47W809-1), walkdown data, and the vendor drawings for sixteen valves in the CVCS. These valves are shown as diaphragm valves on the flow diagram and as globe valves on the vendor drawings, except for one which was shown as a needle globe on the vendor drawings. Based on the walkdowns of the as-built piping system, the valves were verified to be globe valves as stated on the vendor drawings. Serial numbers taken from the walkdown data were used to find the correct NPV-1 Code data report to determine the valve drawing numbers and the valve type.

Drawing 1-47W809-1 is being revised to correct the type of valve shown on the drawing. In addition the data for each of the valves listed will be updated and corrected as necessary in the MEL. No modifications, re-tagging, or other field work is required for any of the above changes.

The proposed documentation changes have been evaluated against the applicable design bases and technical specification requirements and determined not to degrade CVCS or any other safety-related systems below their design bases nor increase any challenges to systems assumed to function in the accident analysis. The subject documentation changes do not directly or indirectly impact any safety analysis that forms the basis for any technical specification and no technical specification changes will be required due to implementation of EDC E-50040-A.

It has been determined that implementation of the subject documentation changes:

- will not create the possibility of a new type of accident or equipment malfunction not previously analyzed in the UFSAR because the changes do not introduce any new accident scenarios, malfunction initiators, malfunction pathways or failure pathways not previously considered in the UFSAR and do not pose an increased challenge to any safety-related system;

- will neither increase the probability nor the consequences of an accident or equipment malfunction already evaluated in the UFSAR because the changes do not alter the frequency class of any analyzed event to a higher frequency class; do not adversely affect the safety function of any equipment which could cause, intensify, or mitigate the consequences of any DBE; do not cause any undesirable interactions with other systems important to safety; do not affect 10 CFR 100 compliance; and does not introduce any new equipment malfunction pathways;

- do not infringe on any margin of safety defined in the technical specifications; and

- do not involve modifications to any radwaste system or involve any special tests or experiments.

Based on the results of this safety evaluation, it is concluded that the subject documentation changes do not involve an unresolved safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-98-122-0

Implementation Date: 10/19/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39197-A	Condensate Polishing Demineralizer System Blowdown

Description and Safety Assessment:

DCN S-39197-A permits release of CPDS when the hydro units are not operating provided the required prerequisite is satisfied. The Design Criteria, "Heat Rejection System," and SDD, "Condensate Polishing Demineralizer System (CPDS)," currently state that the CPDS may be released with the CTB dilution flow < 20,000 gpm provided the tank is sampled and the activity is $\leq$ to the LLD as defined in ODCM, (Table 2.2-1). In addition, the SDD states that Operations shall (1) have the tank sampled prior to each release when CTB dilution flow < 20,000 gpm and (2) not make any additions to the tank that is being released after it has been sampled and found acceptable for release. This change clarifies that in addition to the LLD, if the gross gamma radioactivity is $< 5 \times 10^{-7}$ μ Ci/cm^3, the CPDS may be released with the CTB dilution flow < 20,000 gpm.

The CPDS, its associated components, piping, and valves are located in the Turbine Building. The CPDS is normally non radioactive, non-safety-related, installed in a non-seismic structure, and is not used during any accident. The CPDS does have the potential to be radioactive in the unlikely event of a large primary to secondary leak. However, these changes permit a release from the CPDS only when the CTB dilution flow is < 20,000 gpm and the activity is $< 5 \times 10^{-7}$ μ Ci/cm^3. The activity is verified with sampling and analysis prior to release. These changes are within the existing design basis limitations of the ODCM and therefore, do not represent a change to radioactive release criteria or result in higher discharge concentrations (non radioactive).

This change does not alter the system design from a functional perspective since the CPDS may already be released if the activity is below the LLD limits. The prerequisites that have been added assure the liquid to be discharged is non radioactive and that no simultaneous additions are being made to the discharge flow path. This DCN does not affect the design basis for any system that is important to safety. No additional components have been added by this change. No new potential single failure of existing components has been anticipated to occur. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. This change does not affect the radioactive releases in excess of those established by 10 CFR 20 and 10 CFR 100. This change does not reduce the margin of safety identified in the technical specifications.

Based on the results of this safety evaluation, it is concluded that the subject documentation changes do not involve an unreviewed safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-98-123-1

Implementation Date: 12/01/1999

Document Type:	*Affected Documents:*	*Title:*
Design Change	EDC E-50079A	Resolve Drawing Discrepancies (DDs)

Description and Safety Assessment:

EDC E-50079-A resolves the discrepancies identified in DDs 98-0075 and 98-0082. DD 98-0075 identifies the following equipment, 0-HOSE -77-1816, 0-HOSE -77-1817, O-CASK-77-1818 and, O-PMP-77-1819, that should be removed from TVA drawing 1-47W830-3 and the MEL since the equipment is contained within the vendor-controlled equipment boundary of the spent resin packaging system (SRPS). DD-98-0082 identifies valves which should be labeled as diaphragm valves. This EDC revised flow diagram, 1-47W856-1, to show the valves as diaphragm valves which eliminated the discrepancy between the drawing and Bill of Materials. Also, DD 98-0082 identifies the TVA Drawings, 1-47W609-3 and 1-47W610-62-4, which was revised to show valve 1-62-949, the manual isolation valve for the RCDT, to be in the normally open position. This change will eliminate the discrepancies with SOI-62.06 and the drawings.

EDC E-50079-A revised the flow diagram, TVA drawing 1-47W830-3, "Mechanical Flow Diagram Waste Disposal System" to (1) add resin sampler with a component identification (CID) (0-SMPL-77-1817), (2) delete CIDs O-CASK-77-1818 and O-PMP-77-1 819 which are contained within the boundaries of the vendor-owned equipment of the SRPS, (3) and correct the vent flow path. Drawing 47W560-20 was also revised to show the two hoses (0-HOSE-77-1816 and 0-HOSE-77-1817). The cask and Pump are within the vendor-controlled equipment boundary of the SRPS. Installation of this equipment, and other removable vendor components is coordinated between, RADCON and the vendor as required for the operation of the plant in accordance with the TVA partnership agreement for radwaste with Chem-Nuclear Systems, Inc. Configuration control of the vendor components within the vendor boundary is controlled by the applicable approved site / vendor procedures. Therefore, deletion of the subject equipment CIDs is considered a documentation enhancement which will eliminate potential conflicts between the affected drawings and other associated documents.

This change is technically acceptable since valve 1-62-949 only provides manual isolation of the waste disposal system (WDS) RCDT discharge flow into CVCS HUT A. Normal alignment is to either the CVCS HUT or tritiated drain collector tank (TDCT) per Section 3.1.3.1 of System Description N3-77C-4001. The current operating instruction (SOI-62.06) uses the CVCS HUT as the normal alignment. Flow to the CVCS holdup tank is provided by RCDT pumps A and B which energize in response to RCDT level switch 1-LS-77-1. Containment isolation valves 1-FCV-77-9 and 1 -FCV-77-10, located between the RCDT pumps and valve 1-62-949, isolate flow to CVCS holdup tank A in response to MCR hand switches. Valves 1-FCV-77-9 and -10 close automatically upon receipt of a phase A containment isolation, or from a HI radiation signal generated from radiation elements I-RE-90-275 and 1-RE-90-276 respectively. The primary safety function of the line is to assure the integrity of primary containment by automatically isolating during DBEs which generate either a phase A containment isolation, or high radiation signal. This safety function is not impacted by the proposed realignment of manual valve 1-62-949 to normally open.

The components that are impacted by the documentation only change and its associated components are located in the Auxiliary Building. Also, components have not been deleted, added or altered in any way. No margin of safety is identified in the bases section of the technical specifications which could be reduced by these changes. These Changes do not prevent any component from performing its function as described in the technical specifications. This is a documentation only change. Therefore, EDC E-50079-A does not change or affect the design basis for any component important to safety; therefore, the change does not increase the probability of occurrence of an accident or malfunction as described in the UFSAR, will not increase the consequences of an accident or malfunction previously evaluated in the UFSAR, does not create the possibility for a new accident or malfunction evaluated in the UFSAR nor reduce the margin of safety as defined in the UFSAR.

SA-SE Number: WBPLMN-99-005-0

Implementation Date: ***11/03/1999***

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50158-A UFSAR Change Pkg. 1590	Alternate method to inject Steam Generator layup chemicals.

Description and Safety Assessment:

The Steam Generator Wet Layup Recirculation System (WLRS) protects the Steam Generator internals from corrosion during periods of cold shutdown. The WLRS achieves this by assuring thorough mixture of corrosion control chemicals that are injected into steam generator layup water. In lieu of using the WLRS, an alternate method of injecting and mixing the layup chemicals may be use as described below.

EDC E-50158-A allows for an alternate method to inject the layup chemicals into the steam generator, and complete mixing would be accomplished by bubbling nitrogen (N_2) through the bottom of the steam generator during an outage (cold shutdown). Site Chemistry method will be to inject the layup chemicals into the steam generator by back flowing through the steam generator sample lines. The design basis document currently describes the use of N_2 for mixing; that is, the design basis document states for each steam generator a pressure regulator, pressure gauges, and temporary hose connection are provided for sparging N_2 up through the layup water for mixing. Therefore, there are no changes to the design for the N_2 sparging. In addition, the layup chemicals are further agitated (mixed) by injecting them either when the water level is low or during initial filling process. The layed up steam generator are sampled to ensure that the chemicals are mixed.

These system's associated components, piping, and valves are located in the Auxiliary Building and yard. This equipment does not perform a primary safety function (except for containment isolation which has not been affected), are installed in a Seismic Category I structure, and are not used during any accident. The Chapter 15 accident analysis were review with no impact. This system and equipment is not associated with any accident, does not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis. This EDC does not change the logic or function of any system that is important to safety.

This change is not associated with the protective features used to detect and mitigate the effects of any event. The equipment associated with this EDC does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the UFSAR. This EDC does not change or affect the design basis for any system that is important to safety.

These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBA, safety-related equipment is expected to operate as designed to limit the consequences of the DBA. The previously evaluated malfunctions of components were reviewed and there is no increase of the consequences of these malfunctions. This change does not result in a radioactive release in excess of those established by 10 CFR 20 and 10 CFR 100 and does not create a new radioactive liquid or gaseous effluent release pathway as defined in ODCM. No new potential single failures of existing components will occur as a result of this EDC. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The equipment that is associated with this change is not used in the mitigation of any accident/malfunctions and does not change the radiological consequences of an accident previously evaluated 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. The ODCM limits for releases from either the Liquid Radwaste Processing System (LRPS) or the Gaseous Waste System are not revised or challenged by these changes. Based on the results of this safety evaluation, it is concluded that the subject documentation changes do not involve an unreviewed safety question and are acceptable from a nuclear safety perspective

This EDC does not affect the design basis for any system that is important to safety. No additional components have been added by this change. No new potential single failure of existing components has been anticipated to occur. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. This change does not affect the radioactive releases in excess of those established by 10 CFR 20 and 10 CFR 100. This change does not reduce the margin of safety identified in the Technical Specifications 5.7.2.3 or 5.7.2.7.

SA-SE Number: WBPLMN-99-011-0

Implementation Date: 10/28/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50177-A UFSAR Change Pkg. 1591	Temper Bead Weld Repairs

Description and Safety Assessment:

UFSAR Table 3.2-7 and Design Criteria WB-DC-40-36, Table 3.1-8 are being revised in accordance with UFSAR Change Package 1591 and EDC E-50177-A to add ASME Code Section III 1989 Edition Paragraph NB-4622.9 for performing temper bead repair of carbon and alloy steel base materials for ASME Code Class I components. The Code of Federal Regulations, 10CFR50.55a, identifies the 1989 Edition of Section III as being approved by the NRC. ASME Code Section XI paragraph IWA-4120 states that "Repairs shall be performed in accordance with the Owner's Design Specification and the original Construction Code of the component or system. Later Editions and Addenda of the Construction Code or of Section III, either in their entirety or portions thereof, and Code Cases may be used…." If these cannot be used ASME Section XI also contains a temper bead repair process.

The original Construction Code or Code of Record for ASME systems is the 1971 Edition thru Summer 1973 Addenda of Section III which contained a temper bead repair process, Paragraph NB-4642. However, it only allowed a repair of 3/8 inch or 10% of the bass material thickness, which ever is less, and no more than 10 square inches of surface area by the temper bead process. The 1989 Edition allows 100 square inches and 1/3 the base material thickness. This is less restrictive than the Code of Record, however, this later Code is more restrictive in other areas such as Post Weld Soak and procedure qualification. Both Codes required the procedure to be qualified to the requirements specified in that Year Edition of the Code. Therefore, since qualification to the later requirements is required, it is equally as good as Code of Record. This change will have no impact on the function of the components nor design bases of the system. It allows the repair of defects such as steam cuts, corrosion pits, etc. in ASME Code Class I components such as the steam generator, pressurizer, or reactor vessel, without Post Weld Heat Treatment (PWHT) when PWHT is impractical or cannot be performed.

This change is a documentation change only which adopts a later Code for temper bead repair of base material which has been approved by NRC. The procedure qualification requirements of the later Code along with its required non-destructive examination makes the repair equivalent to the original Code requirement. Therefore this change will not result in an increase in the probability of an accident or malfunction of equipment. Consequently, there will be no increase in the consequences. Since the repair is equivalent to the original Code requirements, there will be no change in margin between the original Code and the later Code requirements. No margin of safety is affected by this change. Therefore, this change does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-99-019-0

Implementation Date: 10/27/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50211-A	Fire Dampers Required for Unit 1 Operation

Description and Safety Assessment:

This EDC is issued to change flow diagram 1-47W866-11 to show fire dampers 2-ISD-031-3880, -3881, -3865, and -3866 as being required for Unit 1 operation.

Current configuration shows the subject fire dampers as included in the Unit 2 area not required for Unit 1 operation. The parent duct work is not required for Unit 1 but the fire dampers are part of the compartmentation requirements, drawings 47W240-1 and 47W240-3, which must be maintained for Unit 1. These fire dampers are currently within the scope Fire Protection Report.

Also, these fire dampers are currently within the scope of 0-FOR-304-3, "Fire Damper (Internal) Visual Inspection-Auxiliary, Control and Diesel Generator Building" for initial and surveillance testing, such that the design and licensing have been maintained and are current (functional and available for design basis).

This change and equipment are not associated with increasing the consequences of an accident previously evaluated and is bounded by the existing analyses. This EDC does not change the logic or function of any system that is important to safety.

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.

Document changes have been evaluated for plant operability during the review process and found to not affect the physical plant. These "documentation only" changes will not increase the dose to the public analyzed in the UFSAR Chapter 15. No change will occur to the radiological consequences of accidents analyzed as a result of these changes. The credible failure modes for the systems affected by these changes have been evaluated against the accidents identified in the UFSAR, and it is concluded that they do not introduce a failure pathway different from those identified and evaluated in the UFSAR accidents. The applicable accidents and the equipment served by the affected safety systems have been reviewed against these documentation changes, and no new malfunction pathways will be introduced which have not previously been evaluated and identified. All document 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 affected systems. This safety evaluation was prepared solely because implementation of the subject EDC will have an impact on UFSAR Figure 9.4-14.

Therefore; based on the above evaluation, implementation of the changes under EDC E-50211-A:

- Will not create the possibility of a new type of accident or equipment malfunction not previously analyzed in the UFSAR;
- Will neither increase the probability nor the consequences of an accident or equipment malfunction already evaluated in the UFSAR;
- Does not infringe on any margin of safety defined in the technical specifications; and
- Does not involve modifications to any radwaste system or involve any special tests or experiments.

Based on the results of this safety evaluation, it is concluded that the subject documentation changes do not involve an unresolved safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-99-020-0

Implementation Date: 8/16/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50202-A	New Unit 1/Unit 2 Interface Point

Description and Safety Assessment:

Sight flow glasses are used to visually observe flow in a piping system. Sight flow glass was determined to be corroded and no longer usable. To replace this Unit 1 sight glass, a request for a Unit 2 component was initiated. Sight flow glass 2-FG-14-005 was approved for removal and use in Unit 1. Sight flow glass 2-FG-1 4-005 was believed to be on the Unit 2 side of the Unit 1/Unit 2 interface prior to its removal. After the 1-FG-14-005 was replaced with 2-FG-14-005, it was observed that the previous location of 2-FG-14-005 was on the Unit 1 side of the Unit 1/Unit 2 interface. Flow diagram and design calculation shows the 2-FG-14-005 to be inside the Unit 1 boundary. The piping upstream of 2-FG-14-005 does not pass any flow since the Unit 2 polishers are not operational. A pancake flange was located on the downstream side of the 2-FG-14-005. Since the piping upstream of 2-FG-14-005 was not being used, the pancake flange prevented back flow from the discharge piping of the Unit 1 polishers from entering the piping upstream of 2-FG-14-005 prior to it going to the High Crud Tanks (HCT). However, this pancake flange was not identified on any drawings. PER 99-000827-000 identified these discrepancies. DCN D-50202-A implements the corrective actions for PER 99-000827-000.

DCN D-50202-A installs a carbon steel blind flange downstream of the initial location of 2-FG-14-005. This DCN revises all associated design documents to reflect the installation of the carbon steel blind flange and the new Unit 1/Unit 2 interface point as being at the new carbon steel blind flange. In addition, it was discovered that the flow diagram has the 3 inch by 4 inch reducer downstream of the original location of sight flow glass 2-FG-14-005 shown in the wrong direction. This DCN corrects this drawing error. The design documents to be revised include the UFSAR, drawings, and certain mechanical calculations.

Installing a carbon steel blind flange downstream of the initial location of 2-FG-14-005 and moving the Unit 1/Unit 2 interface point to the carbon steel blind flange will not change the function of the condensate demineralizer system as it is described in the UFSAR. There are no DBAs 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 functions. Therefore, this modification will not affect any DBAs or anticipated operational transients.

Catastrophic blind flange failure which would allow any back flow from the discharge of the Unit 1 polishers to flow out of the piping and on to the floor would be a failure mode for this activity. However, this would not be a credible failure mode because this is analyzed and dead weight supported piping. Failures of the carbon steel blind flange 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 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 condensate demineralizer system does not reduce the margin of safety of any basis for any technical specification. For these reasons, this activity does not constitute a unreviewed safety question.

SA-SE Number: WBPLMN-99-029-0

Implementation Date:10/25/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50228-A UFSAR Chg Pkg. 1614	Environmental Data Drawing Revision to Specify Minimum LOCA Temperatures

Description and Safety Assessment:

The minimum LOCA temperatures for the Auxiliary Building 480V Transformer Rooms 772.0-A6 and -All and HVAC Equipment Rooms 737.0-A3 and -A12 are not addressed in the technical specification. The equipment and components located in these rooms are unaffected by these minimum space temperatures in such a way as to adversely impact the ability of the plant to cope with an accident situation. UFSAR Section 9.4.3.2.6 is revised to lower the minimum LOCA temperature for the Auxiliary Building 480V Transformer Rooms 772.0-A6 and -All from 32°F to 19°F. Table 9.4-8 is revised to reflect the additional HVAC equipment failure modes as postulated by the revised design calculations. All these documentation changes are being made under UFSAR Change Package 1614.

All document changes have been evaluated for plant operability during the review process and found not to affect the physical plant. These changes will not increase the off-site dose rates to the public as analyzed in the UFSAR Chapter 15. No change will occur to the radiological consequences of accidents analyzed as a result of these changes. The credible failure modes for the systems affected by these changes have been evaluated against the accidents identified in the UFSAR. It is concluded that they do not introduce a failure pathway different from those accidents which have been identified and evaluated in the UFSAR accident analyses. The applicable accidents and the equipment served by the affected safety systems have been reviewed against these documentation changes, and no new malfunction pathways will be introduced which have not previously been identified and evaluated. The Technical Specification Bases have been reviewed to determine if any margins of safety are affected by these documentation changes. No margin of safety is identified in the Bases which could be reduced by these changes.

SA-SE Number: WBPLMN-99-039-1

Implementation Date: 10/02/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50305-A	Turbine Generator Control and Protection System (TGCPS).

Description and Safety Assessment:

The EHC fluid supply system provides high-pressure fluid for motive force to the servo-actuators which position the turbine steam valves in response to electrical signals from the electronic controller. The EHC skid is located in the Turbine Building and is TVA Class H, non-seismic. The fluid control medium is tri-aryl phosphate ester which possesses qualities for fire resistance and fluid stability. This system consists of a fluid reservoir, tubing, controls, pumps, motors, filters and heat exchangers. The components are arranged in two duplicate sets. When one set is in operation, the other one is in standby status and will automatically start functioning if a need arises. During normal turbine operation with control system latched, one motor-pump combination is sufficient to supply fluid requirements. The pump discharge is directed to the system (accumulators and supply header) until it reaches a pressure of 2200 psi, at which point the unloader valve "unloads" and pump discharges directly to reservoir. The accumulators supplies the system fluid requirements until the header pressure drops to 1750 psi, at which point the unloader valve "energizes" and directs pump discharge to the system. Therefore, the system continually cycles between 1750 psi and 2200 psi, and a check valve is provided in the line after the unloader valve to prevent backflow from the system. This constant pressure cycling causes extra stress on the system components, increases the potential for initiating either low pressure alarms or opening the downstream relief valve, and increases the potential for unloader valve instability and EH fluid overheating.

Westinghouse Availability Improvement Bulletin 9107 recommends enhancement of the EHC system by replacing the existing fixed displacement vane pumps with constant pressure pumps equipped with flex hose connections which will isolate the EHC piping from the pumps to reduce vibrations and help prevent leaks. This modification will enhance control of the EHC system pressure, provide increased troubleshooting capabilities, and minimize potential for fatigue of tubing and components. This upgrade will incorporate pumps that vary their output capacity to meet the given flow demands of the system, thereby, maintaining nearly constant pressure as set by their own pressure compensators.

In addition to the above changes, the instrument tubing on the EHC skid will be reworked to position the pump discharge pressure gauges' sensing points upstream of the pump discharge filters to match Westinghouse drawing series 797J034. According to the Siemens-Westinghouse representatives, the present EHC pump discharge pressure gauges' sensing points are downstream of the pump discharge filters.

The TGCPS is not required to function to mitigate the consequences of any DBA or DBE. The TGCPS is required to provide turbine overspeed protection and mitigation for the purposes of minimizing the potential for generation of turbine missiles. The TGCPS is not required to operate during a loss of offsite power or during other unusual events or conditions. This modification will not change the function of the TGCPS as it is 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 TGCPS does not perform any reactor safety-related function, nor will it compromise the ability of safety-related systems to perform their intended functions. Therefore, this modification will not affect any DBAs or anticipated operational transients.

This change will enhance control of the EHC system pressure, provide increased troubleshooting capabilities, and minimize potential for fatigue of tubing and components. Therefore, the proposed activity will not reduce the margin of safety as defined in the basis for any technical specification. For these reasons, this activity does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-99-045-1

Implementation Date: 11/09/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50220-A	Documentation Changes to Essential Raw Cooling Water System.

Description and Safety Assessment:

Revision 1 to this safety evaluation adds additional background information to the problem description to establish the safety evaluation as a stand alone document. (PER 99-014969-000).

During the review of UFSAR Section 6.2.4.3.1 "Possible Leakage Paths", Type E-leakage Path as it applies to ERCW, Note 3, "A water seal at greater than peak containment accident pressure is used to prevent bypass leakage in certain lines (such as the safety injection pump discharge). The seals are available for at least 30 days after a DBE (see Table 6.2.6-2b)," could not be verified in the design basis calculations to determine if containment integrity is maintained. PER WBPER980808 was written and a corrective action was developed. The maximum air leakage through the ERCW piping penetrations loop seals and traps was documented in a calculation and the system description was changed to reflect the results. This demonstrated continued compliance with the licensing basis and resolved this condition adverse to quality.

The safety evaluation addressed documentation changes to show that the WBN Unit 1 ERCW system is capable of preventing containment air from leaking though a Type E leakage path. Type E leakage is defined as a path from the containment that bypass the annulus and leak directly past a clean up system to the outside environment using either the Train A or Train B upper or lower compartment supply or return flow paths under a LOCA conditions for a period of 30 days.

The UFSAR (Section 6.2.3.1 and Table 6.2.4-3) and System Description (N3-67-4002) were changed in Amendment 1 to document the results of the WBN Calculation. The wording to explain the methods utilized to prevent leaking through a Type E leakage path was enhanced; however, this did not result in a change to the plant. This is a documentation only change to justify that the existing ERCW system is capable of containing bypass leakage in a Type E leakage path. Also, the limiting conditions that were developed in calculation was added to the system description

There are no nuclear safety accident scenarios or new failure modes involved with this design changes. EDC E-50220-A is required to implement the changes addressed as a part of WBPER980808. The safety evaluation addressed the documentation only changes to show that the WBN Unit 1 ERCW system is capable of preventing a Type E leak of containment air to the outside environment using either the Train A or Train B upper or lower compartment supply or return flow paths under a LOCA conditions for a period of 30 days.

Therefore:

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR will not increase as a result of the documentation only changes proposed in EDC E-50220-A.

- The probability of an accident or malfunction of a different type previously evaluated in the UFSAR will not be created as a result of the documentation only changes proposed in EDC E-50220-A.

- The margin of safety as defined in the basis for any technical specification will not be reduced as a result of the documentation only changes proposed in EDC E-50220-A.

The proposed change does not involve any unreviewed safety question.

SA-SE Number: WBPLMN-99-054-0

Implementation Date: 11/27/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50306-A UFSAR Chg. Pkg. 1606	Fifth Vital Battery Room Heating and Ventilation System Modification

Description and Safety Assessment:

Safety Evaluation Report Supplement 13, Section 8.3.2.7 states that the Fifth vital battery has its own dedicated heating and ventilation system with redundant intake and exhaust fans with each set powered from separate auxiliary electrical trains. Meanwhile, the UFSAR Sections 8.3.2.1.1 and 9.4.3.2.5 state "Vital Battery Room V has a dedicated heating and ventilation system with a backup", and "ventilation air is drawn directly from the outside and is exhausted directly to the outside," respectively. UFSAR Change Package 1606 has been prepared to revise those sections to accurately depict the new configuration of the FVBR ventilation system, which also revises Table 9.4-5, "Failure Modes and Effects Analysis for Active Failures in Auxiliary Board Rooms Air Conditioning System." This Table includes analysis for the FVBR intake and exhaust fans, their flow controllers and tornado isolation dampers. The technical specification is not impacted. Heating, cooling, and ventilation calculations have been revised to verify that the change will not adversely affect the ability to maintain the room ambient temperatures within the required limits, and continue to provide effective hydrogen removal capability. for the vital battery rooms and the 480V board rooms. Therefore, the changes effected by DCN D-50306-A do not challenge the ability of the HVAC systems in these rooms to maintain room temperatures within required limits, and provide effective hydrogen removal function. Therefore, the ability of the plant to cope with an accident situation is not adversely affected.

The change has been evaluated for plant operability during the review process and precautionary measures have been specified during the installation of this modification. These measures include a restriction on the use of the Fifth vital battery as a substitute for any one of the main vital batteries during the implementation of DCN D-50306-A, and provide temporary ventilation for hydrogen removal.

This change will not increase the Off-Site dose rates to the public, as analyzed in UFSAR chapter 15, nor will it cause an increase to the radiological consequences of accidents analyzed previously. The credible failure modes for the systems affected by this change have been evaluated against the accidents identified in the UFSAR. Any new failure modes identified are analyzed by the Auxiliary Buiding HVAC failure modes and effects analysis calculation, the results of which will be reflected in the revision to UFSAR. The results of these analyses indicated that no failure pathways different from those identified and evaluated in the UFSAR accidents existed.

The applicable accidents and the equipment served by the affected safety-related systems have been reviewed against this change, and it was concluded that no new malfunction pathways, not previously identified and evaluated, will be introduced. The Technical Specification Bases have been reviewed to determine if any margins of safety are introduced by this change, and none were identified. Therefore, this change, effected by DCN D-50306-A, does not constitute a unreviewed safety question.

SA-SE Number: WBPLMN-99-059-0

Implementation Date: 10/22/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50431-A	Installation option of pre-filters at air intakes.

Description and Safety Assessment:

EDC E-50431-A revises Turbine Building and CCW Pumping Station ventilation systems' ductwork drawings and the Turbine Building airflow diagram to give an option to the plant to install prefilters at the air intakes to keep out insects and atmospheric pollutants, such as dust and pollen, etc. The intakes in consideration are the North Fan Room, South Fan Room, louvered intake openings in the two sides of the Turbine deck, and the two CCW Pumping Station supply fans. The prefilters are specified to be the low-efficiency type throwaway filters with an approximate dust removal efficiency range of 10%- 15%.

The addition of a note to design documents to give the plant the option to install prefilters at the Turbine Building and the CCW Pumping Station air intakes, if elected, to keep out insects and atmospheric pollutants, does not constitute a unreviewed safety question; because, the Turbine Building and the CCW Pumping Station ventilation systems are not safety-related, and have no interface with any safety-related equipment or systems. The only accident which will be addressed in this safety evaluation is LOCA, since the Control Building LOCA temperature calculations assumed a boundary condition of 120°F for the Turbine Building spaces. During a LOCA, calculated space temperatures in the Control Building will remain within the required limits if the Turbine Building temperature does not exceed 120°F. Implementation of plant instruction will ensure that the Turbine Building space temperatures remain below 120°F. The CCW Pumping Station is not adjacent to any Category I structure. Therefore, any temperature increases in the CCW Pumping Station, which might be caused by the installation of prefilters at the two intakes, would not impact space temperatures in any safety-related equipment area.

Installation of the filters in non-safety related ventilation equipment could not cause an accident, an accident of a different kind or decrease the margin of safety as this equipment is not addressed in the technical specification. Therefore the change does not involve an unreviewed safety question.

SA-SE Number: WBPLMN-99-060-1

Implementation Date: 10/19/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50442-B	Condensate Booster Pump Drain Fillings

Description and Safety Assessment:

DCN D-50442-B revised the applicable design to replace the drain fittings, piping and drain valve with a drain plug.

The CBPs (A, B, and C) volute casing drain is leaking at the nipple that threads into the casing. The carbon steel drain contains fittings, piping and a drain valve. This volute casing drain is recessed back into the pump housing and is very difficult to access and repair. The nipple can not be seal welded to the casing to stop the leak. The maintenance on the pump does not require that the volute casing be drained that frequently (approximately 10-15 years). The drain should be replaced.

The Condensate System , its associated components, piping, and valves are located in the Turbine Building. The Condensate System is normally non-radioactive, non-safety-related, installed in a non-seismic structure, and is not used during any accident. The Condensate System does not have the potential to be radioactive. These changes are within the existing design basis limitations of the ODCM and therefore, do not represent a change to radioactive release criteria or result in higher discharge concentrations (non radioactive).

This change does not alter the system design from a functional perspective since the CBPs already may be drained as necessary. This DCN does not affect the design basis for any system that is important to safety. There are no failure modes associated with this change. The existing threaded drain nipple and piping are being replaced with a threaded drain plug which does not introduce an equipment failure of a different type. No new potential single failure of existing components has been anticipated to occur. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. This change does not affect the radioactive releases in excess of those established by 10CFR 20 and 10 CFR 100. This change does not reduce the margin of safety identified in the Technical Specifications 5.7.2.3 or 5.7.2.7.

Based on the results of this safety evaluation, it is concluded that the subject documentation changes do not involve an unresolved safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-99-070-0

Implementation Date: 10/06/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50366-A UFSAR Chg Pkg 1613	Re-gearing Limitorque Actuators on Motor-Operated Valves (MOVs)

Description and Safety Assessment:

DCN D-50366-A addresses the re-gearing of Limitorque actuators on MOVs 1-FCV-63-25 and 26 to ensure the actuators will satisfactorily perform their intended safety functions following.

There are several issues driving the proposed modification. The original analysis on valves 1-FCV-63-25 and-26 failed to consider a postulated but credible LOOP with concurrent Large Break LOCA and resulting SI which would be the worst case. The pressures from this case were used to analyze the opening forces which increased the required thrust, therefore, decreasing the available margin. Design Standard 18.2.21 imposes more stringent torque and thrust requirements on MOVs within the GL 89-10 population, effectively increasing the thrust required to open/close the valves. Limitorque Technical Update 98-01 revises Limitorque's methodology for calculating the actuator torque output, including gear efficiency and motor output, which decreases the actuator capability and, consequently, reduces the thrust available to open/close the valves. The cumulative effects of these changes reduce the overall thrust margin. These new requirements caused the margin of FCV-63-25 and -26 to decrease to a point that any degradation in the valves would have caused the valves to be inoperable and the testing frequency was increased significantly. To restore adequate thrust margin in 1-FCV-63-25 and -26 valves, the drive ratio of the valve actuators will be modified by the installation of new motor and worm shaft pinion gears. The re-gearing of these actuators will increase the available torque but, due to a reduction of worm shaft rotational speed, will also increase the valve stroke times.

Although the existing thrust margins are still positive (and therefore are not an operability issue), the existing design makes no provision for future degradation and does not optimize the testing frequency mandated by GL 96-05. The proposed modifications will incorporate an allowance to account for anticipated actuator degradation consistent with that currently in use within the nuclear industry. This allowance is expected to minimize future impacts once GL 96-05 testing is implemented for WBN. A potential benefit of increasing the thrust margins for the affected MOVs will be a reduced inspection and testing frequency under the GL 96-05 program.

The SI system operates to help mitigate the consequences of steamline and feedline break events. The licensing basis steamline break analyses for Watts Bar include the following: Accidental Depressurization of the Main Steam System, Major Rupture of a Steam Pipe, Steamline Break Mass and Energy Releases Inside Containment, and Steamline Break Mass and Energy Releases Outside Containment. The Steamline Break with Coincident Rod Withdrawal at Power event is a fast transient which reaches the most limiting point and turns around due to reactor trip before safety injection can occur. The limiting factor in reaching full flow for SI is the stroke times of the Refueling Water Storage Tank (RWST) and VCT valves. The total stroke time for these valves is 25 seconds. The valves associated with this modification start at the beginning of that time. Increasing the stroke time from 10 to 12 seconds will not impact the ability of the SI System to be at full flow in the analyzed time frame.

There are no credible equipment or system failure modes introduced by the subject modifications which (a) have not been previously accounted for in the design of the affected equipment or systems, or (b) would prevent the affected equipment or systems from performing their intended safety functions. The subject modifications do not introduce any new malfunction initiators or other failures not previously considered in the UFSAR. The subject modifications will not introduce any new malfunction of equipment which will either directly or indirectly, affect any nuclear safety-related systems, structures, or components. The accompanying increases in valve stroke times have been evaluated against the applicable design bases and technical specification requirements and determined not to degrade the response time or performance of the associated safety systems below their design bases nor increase any challenges to those or other safety-related systems assumed to function in the accident analysis.

SA-SE Number: WBPLMN-99-070-0 ***Implementation Date:*** 10/06/2000

The subject modifications do not directly or indirectly impact any safety analysis that forms the basis for any Technical Specification. It is concluded that no technical specification changes will be required due to implementation of DCN D-50366. Therefore, the subject documentation changes will not reduce the margin of safety as defined in the basis for any Technical Specification. This change does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-99-074-0

Implementation Date: *12/06/1999*

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50464-A	Drawing note correction

Description and Safety Assessment:

This DCN revises Note 18 on drawing 1-47W844-1. The note states "The disc on these valves is made of Austenitic Gray Iron which is not suitable for raw water service. Repair or replacement of these valves requires the use of materials which are not incompatible with raw water service. Specifically prohibited are Austenitic Gray Iron, C95400, Al-Bronze not properly heat treated, or other materials not compatible with raw water or compatible only when in the proper heat treated condition (unless the proper heat treatment is specified in the procurement document)."

The note requires the disc to be replaced with a raw water compatible material when the valves are repaired or replaced. Some of the valves have been repaired or replaced with suitable material and the note is still referenced to those valves implying they still have the Austenitic Gray Iron disc. The note is being revised to delete the reference to the original disc and only refer to what is required to be installed. The change deletes the historical information and the reference to heat treatment which is a design function to be specified or approved and references the design criteria. None of the requirements which are pertinent are being changed. This is a clarification and does not constitute a unreviewed safety question.

SA-SE Number: WBPLMN-99-079-0

Implementation Date: 3/10/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50416-A	Cask Loading Pit Gate

Description and Safety Assessment:

This safety evaluation addresses design changes per the scope of EDC E-50416-A. These changes clarify WBN's design bases. The changes have been evaluated for plant operability during the review process and do no affect the physical plant configuration or change the operating parameters of the affected systems. This EDC will ensure that the design documents are consistent with each other.

In summation, the changes:

- clarify WBN's design bases regarding the Cask Loading Pit Gate and are intended to maintain accuracy and consistency between the UFSAR and other affected design documents with respect to the as-built configuration of the plant. This change revises design documents to incorporate the following requirements: (1) the Cask Loading Pit Gate will not be utilized to drain the Fuel Cask Loading Pit; and (2) the Cask Loading Pit Gate will be installed in its stored position only.

- 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 of WBN's Cask Loading Pit Gate.

- 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 Spent Fuel Pool Cooling and Cleaning System (SFPCCS) and the FHSS from performing their intended safety function.

- do not increase any challenges to safety-related systems 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 or 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 affected systems 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 affected systems and determined not to introduce any new accident scenarios or failure pathways.
- do not increase the probability of any analyzed accident described in UFSAR Chapters 6 and 15.
- do not involve any new single failures.
- 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.
- will neither increase the probability nor the radiological consequences of an accident or equipment malfunction important to safety previously evaluated in the UFSAR due to the clarification of the stored position of the Cask Loading Pit Gate. The post-accident operation of the SFPCCS and the FHSS is not impacted.
- do not infringe on any margin of safety defined in the basis for any technical specifications. The technical specifications have been evaluated with no impact identified.
- do not involve modifications to any radwaste system or involve any special tests or experiments.

Based on the results of this safety evaluation, it is concluded that the subject changes do not involve an unreviewed safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-00-001-0

Implementation Date: 09/20/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50378-A and Procedure AOI-12, AOI-29, ARI-151-158, MI-61.06, MI-88.001, PAI-2.03, SOI-30.02, SOI-30.0-3, SOI-61.02 UFSAR Chg. Pkg. 1620	Ice Condenser System Ice Blowing and Negative Return Line

Description and Safety Assessment:

DCN D-50378-A implements design changes and modifications to the Ice Condenser System Ice Blowing and Negative Return lines to permit ice blowing to occur concurrently with fuel handling and core alteration activities during Mode 6 operation. These modifications install some permanent piping on the Additional Equipment Building and Auxiliary Building roofs and also simplify the routing of the ice blowing and negative return line piping. This is being done to facilitate operation of the ice blowing system. reduce the likelihood of ice blockages occurring, and to reduce the effort required to remove ice blockages during operating Modes 5 and 6, and during defueled plant conditions.

Currently, one of the two personnel access doors in the Containment Access Air Locks must be closed during Mode 6 fuel handling operations. Access to and from the containment would be facilitated if both doors could be open concurrently. The existing design of the Containment Air Lock access doors operating mechanism permits both doors to be open at the same time. However, existing design documentation and Technical Specifications require one of the doors to be closed during fuel handling activities in Mode 6 operation. Therefore, an analysis was performed to determine the acceptability of having both doors open during Mode 6 fuel handling operations. and to identify any special requirements for this operating configuration. This design change does not require any equipment modifications.

The permanent modifications being implemented by this DCN are the rerouting of the ice blowing and negative return lines on the Additional Equipment Building and Auxiliary Building roofs and the negative return line in the Auxiliary Building. These modifications do not adversely affect any DBEs.

The temporary configurations of the ice blowing and negative return lines which are being implemented to permit ice blowing activities to occur concurrently with fuel handling activities inside containment or core alterations, will only be in effect during Modes 5 and 6 operation. Subsequently, the only accident which would be likely to occur during these activities would be a fuel handling accident. The temporary configurations of the containment penetrations, 1-PENT-304-0079A and 1-PENT-304-0079B, which are used for ice blowing activities will contain piping and be sealed with a temporary silicone seal. The Shield penetration, 0-SLV-304-R1S023, will contain both the ice blowing and negative return piping and be sealed with a temporary silicone seal. Since a fuel handling accident will not result in a noticeable rise in containment pressure, the consequences of a fuel handling accident are not affected. Administrative controls will ensure the timely closure of one of the valves in each line, and the subsequent reinstallation of the blind flanges on the subject penetrations when radiological dose limits permit, for long term recovery from the accident.

The allowance to have both doors of the containment personnel airlocks open under administrative controls will also only affect Mode 6 fuel handling activities inside containment, or core alterations. As discussed above, the only accident which would be likely to occur during these activities would be a FHA. Administrative controls will ensure the timely closure of one of the doors in each airlock.

Administrative controls will ensure that the penetrations which have the temporary ice blowing and negative return piping installed will be returned to their original configuration upon notification of other DBEs, such as a tornado. which could adversely impact the integrity of the altered penetration. DCN D-50378-A also specifies the administrative controls which will permit both doors of the Containment personnel airlocks to be open during Mode 6 refueling operations. The controls involve assigning personnel to ensure closure of one of the doors in each airlock subsequent to a fuel handling accident.

SA-SE Number: WBPLMN-00-001-0 ***Implementation Date:*** 09/20/2000

UFSAR Change Package 1620 provides the changes necessary to address the opening of both doors of the Containment Personnel Airlocks during Mode 6 refueling activities, and describes the temporary configurations of penetrations X-79A and X-79B that may exist during Modes 5 and 6 operation. as well as Mode 7 (defueled) plant conditions.

Technical Specification Change Tracking Number 99-009 provides the Significant Hazards Evaluation to support the changes to Technical Specification 3.9.4 to permit the temporary alterations to the containment penetrations 1-PENT-304-0079A and 1-PENT-304-0079B and the associated administrative controls to permit ice blowing activities to occur during fuel handling operations. The Technical Specification Change also provides the bases for allowing, both doors on the Containment Personnel Airlocks to be open during Mode 6 refueling activities.

There are not any accidents which have been evaluated in the UFSAR that may be affected by DCN D-50378-A. A fuel handling accident is the only accident which this modifications would have any potential of affecting. There are no credible failure modes added or changed as a result of these changes to the UFSAR.

Therefore, the probability of occurrence or the consequence of any accidents or equipment malfunction is not increased. In addition, the changes do not affect the consequences of any previously evaluated accidents because the ability to maintain the integrity of the Steel Containment Vessel and Shield Building penetrations has not been adversely impacted. The changes do not have the potential to create a new accident or equipment malfunction: nor do the documentation changes to the UFSAR affect the margin of safety defined in the Technical Specification Bases.

SA-SE Number: WBPLMN-00-002-0

Implementation Date:03/28/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50502-A UFSAR Pkg. 1618	Radioactive Liquid Release Analysis

Description and Safety Assessment:

The SGBD System is currently being returned to the Condensate System via the CPDS. As an alternate, the SGBD may be released via the CTB piping. The SGBD may be released provided the CTB dilution flow > 20,000 gpm. If flow is < 20,000 gpm passing through the CTB line, a valve (1-FCV-15-44) in the discharge line from the SGBD is automatically closed. The flow is measured with a flow element (0-FE-27-98) in the lines to the diffuser valves (0-FCV-27-100 and 0-FCV-27-101). A flow signal from the hydro units of the upstream dam that is <3,500 cfs results in cessation of the CTB discharge to the river by closing the diffuser valves and opening valve (0-FCV-27-97). When the diffusers valves are closed, the CTB flow rate goes to zero. Therefore, the SGBD can not be processed on a continuous bases.

DCN D-50502-A revises the analysis for the Annual Radioactive Liquid Release to support releasing SGBD, on a continuous basis, to the CTB without treatment. The configuration control drawing (CCD) 1-47W801-2 is revised to show flow control valve 1-FCV-15-44 as "Normally Open". This change revises Design Criteria, WB-DC-40-37 "Heat Rejection System," and System Description Document (SDD), N3-15-4002 "Steam Generator Blowdown System, " to state that the SGBD may be released, on a continuous basis, with the CTB dilution flow < 20,000 gpm provided the activity for gross gamma is ≤ 5 E-7 micro Ci/cc and tritium is ≤ 1E-5 micro Ci/cc, and the maximum flow is 220 gpm. SDD, N3-77C-7001, "Liquid Radwaste Processing System" is revised to update the annual releases. The Design Criteria WB-DC-40-24, "Radiation Monitoring" is also being revised to clarify an additional function for the radiation monitors. The activity for gross gamma (≤ 5E-7 micro Ci/cc) and for tritium (≤ 1 E-5 micro Ci/cc) are equal to the Low Level of Detection (LLD) as defined in the ODCM. The sensitivity of our instruments are such that gross gamma and tritium in the SGBD may be measured at or below the LLD. Therefore, the SGBD gross gamma radiation monitors will be set below the LLD to ensure that no liquid that has gamma activity is processed to the yard holding pond. Note, when the upstream dam is not flowing, the SGBD will be diverted via the CTB to the Yard Holding Pond and may contain low levels of tritium. The tritium stays entrained in the liquid and will be processed from the Yard Holding Pond to the river when the upstream dam starts flowing. Therefore, this amount is not considered significant at levels at or below the LLD.

The DCN revises the control logic for valve 1-FCV-15-44. Previously, the control switch had an "OPEN" (spring return to "AUTO"), a "CLOSE" and an "AUTO" position. The "OPEN" position was used only to initially open the valve, and the switch returned to the "AUTO" position when the switch operator was released. The "AUTO" position was interlocked with CTB dilution flow and radiation level setpoints (1-RE-90-120 and 121) such that if CTB dilution flow was < 20,000 gpm or if radiation was above the setpoint as established using setpoint scaling documents, the valve would close. A maintained "OPEN" position replaces the spring return "OPEN" position, which is independent of the CTB dilution flow. This maintained "OPEN" position is accomplished by replacing the existing control switch, 1-HS-15-44, with a three position key operated selector switch. The new control switch will be maintained in all three positions and will be mounted in the same junction box (JB 1774) as the existing switch. One of the three spare conductors in cable 1V4140 will be utilized for this modification in order to avoid running a new cable for the valve circuit. Minor rewiring of the contacts for relays KIR and XY (both located in rack 1-R-71) is required. The "OPEN" position of 1-HS-15-44 will be interlocked with the radiation level setpoint only such that SGB releases can be made with low dilution flow provided overall activity levels for 1-RE-90-120 and 121 are below those established in the applicable setpoint scaling documents. The switch will be locked (key removed) to prevent inadvertent movement to the "OPEN" position. Administrative controls shall be by Operations while using the "OPEN" position to assure that radiation greater than expected is not released from the plant. Additionally, an alarm will be provided in the MCR to alert the operators if valve 1-FCV-15-44 should inadvertently close. The addition of the MCR alarm requires a cable from the valve to a multiplexer in the Turbine Building.

There are no accidents associated with the SGBD or the CTB line. These changes do not create new failure modes of equipment involved in termination of releases due to potential activity above the 10 CFR 20 limits. The addition of the hand switch position which is independent of CTB dilution flow could allow release at low dilution flow if the switch contacts shorted. Any releases should be less than the 10 CFR 20 limits even if this were to occur because the only time the "OPEN" position of the switch would be used when the SGBD gross gamma is ≤ 5 E-7 micro Ci/cc and tritium is ≤ 1 E-5 micro Ci/cc or essentially non-detectable radioactivity. Additionally, the release is independent of the CTB flow and is still dependent on radiation level as determined by monitor 1-RE-90-120 and 121. These changes do not create any unacceptable equipment failure modes that would cause the SGBD to be unable to perform its function of processing condensate water, nor do these changes affect failure modes of equipment that is important safety. This DCN does not add any additional or different types of failure modes that have not been addressed in the UFSAR.

The Safety Evaluation Report (SER) has been reviewed though Supplement 20. The only identified potential impact is in Supplement 16, page 11-3, the staff assumed that the SGBD would be cooled, processed by the CPDS and not continuously released to the CTB. However, the potential impact should not affect NRC final conclusions regarding radioactive releases.

Based on the results of this safety evaluation, it is concluded that the subject documentation changes do not involve an unreviewed safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-00-007-0

Implementation Date: 09/25/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50450-A UFSAR Chg Pkg 1621	Condensate Booster Pump Modifications to prevent Water Hammer and Vibrations

Description and Safety Assessment:

For minimum flow protection, WBN employs the Yarway Automatic Recirculation Control (ARC) Valves to provide cooling flow through the Condensate Booster Pumps (CBPs). The ARC valves also serve as a check valve to prevent reverse flow through these pumps. Numerous problems have occurred with the valves resulting in valve leakage to the main condenser during full power operation and backflow through the pumps. The leakage from the CBPs is not only a source of lost megawatts, but the leakage flow (hot water and flashing steam) has set up unacceptable water hammer/vibrations in the downstream piping routed to the main condenser. Reverse flow through the failed check disk portion of the valve has induced backward rotation of the pump resulting in pump bearing failure. These ARC valves are fully mechanical in nature and operate off system pressure, flow, and hardware linkages (i.e., no electrical or pneumatic controls).

To address these problems, this DCN D-50450-A reroutes the CBPs' ARC valves' recirculation line to the condensate piping upstream of the pumps. This line is presently routed to the main condenser. The reroute of the piping is shown on TVA Condensate Flow Diagram 1-47W804-1, coordinate G-5, et al. This DCN also revises the minimum flow requirement for the CBPs. The minimum flow requirement has been analyzed by the pump designer/manufacturer and addresses three modes of operation: continuous unrestricted operation in the recirculation mode (min. flow of 5189 gpm), intermittent operation in the recirculation mode in excess of 60 hours but not more than 1500 hours between overhauls (min. flow of 3632 gpm), and operation in the recirculation mode of a total accumulation of not more than 60 hours between overhauls (min. flow of 1224 gpm). The ARC valves are receiving new bypass orifices to compensate for the change in flow coefficient requirements caused by the reroute of the discharge piping from the main condenser to the CBP suction piping and the revised minimum flow requirement.

The piping being rerouted is modeled in CHECWORKS as part of the Flow Accelerated Corrosion (FAC) Program. Even though it is not is not a priority to be replaced during the next outage, the piping used to reroute the recirculation piping will be 2-1/4% Chrome 1% Moly, as used in the other DCNs associated with the FAC Program. This material is more resistant to erosion than the carbon steel material originally installed. Chrome-moly has comparable strength values to carbon steel piping, and the materials have the same coefficient of thermal expansion.

The accident analysis described in UFSAR Section 15.4.2, Major Secondary System Pipe Rupture, includes a trip of the CPDs. This revision to the system does not affect anything described in the analysis nor any of the conclusions arrived at by the analysis. The margin of safety is not reduced because the material replacement enhances the ability of the piping system to maintain its pressure retention properties. Therefore this change is not considered an unreviewed safety question.

SA-SE Number: WBPLMN-00-009-0

Implementation Date: 03/01/2000

Document Type:	***Affected Documents:***	***Title:***
UFSAR	UFSAR Change Pkg. 1624	UFSAR Review of Sections 9.3.2 and 9.3.3

Description and Safety Assessment:

This safety evaluation addresses document changes identified as part of the UFSAR Review/Verification Program. Specifically addressed are UFSAR Chapter 9.3.2 and 9.3.3 and the associated system descriptions and calculation changes which were identified as a result of that review. Regulatory Guide 1.70, November 1978, "Standard Format and Content of Safety Evaluation Reports for Nuclear Power Plants" was utilized in this verification and content effort. Any discrepancies between the UFSAR and the design documents such as system descriptions, design criteria, calculations, and drawings were investigated. This action partially implements the corrective action. Most changes to the UFSAR are minor in nature type changes with no impact on WBN's design bases, do not alter the operational characteristics of systems involved, nor do they differ from the processes or procedures described in the UFSAR. Document changes have been evaluated for plant operability during the review process and found not to affect the physical plant.

The UFSAR is the only document affected. These changes do not change the intent of the UFSAR text. For example, (1) deleting text for equipment that is not required for Unit 1 operation and this equipment has been abandoned in place in the plant, and (2) correcting the temperature and pressures to agree with other documents contained in the UFSAR. There are no technical changes associated with this safety evaluation.

The proposed minor changes to the UFSAR will not increase the likelihood of the DBAs occurring. These changes to the UFSAR do not effect physical changes to the plant, nor do they involve any plant procedures. These minor changes will not increase the dose to the public analyzed in the UFSAR Chapter 15. No change will occur to the radiological consequences of accidents analyzed as a result of these changes. The proposed changes do not effect any changes to the plant design bases, operating procedures, or the physical plant. The credible failure modes for the systems affected by these changes, have been evaluated against the accidents identified in the UFSAR and concluded they do not introduce a failure pathway different from those identified and evaluated in the UFSAR accidents. The applicable accidents and the equipment served by the affected safety systems have been reviewed against these documentation changes, and no new malfunction pathways will be introduced which have not previously been evaluated and identified.

Based on the results of this safety evaluation, it is concluded that the subject documentation changes do not involve an unreviewed safety question and are acceptable from a nuclear safety perspective.

Table1: GENERAL OVERVIEW OF CATEGORY 1, 11, 111 CHANGES

UFSAR-SECTION, SYSTEM DESCRIPTION, OR CALCULATION	DESCRIPTION of REVISION	JUSTIFICATION OF REVISION
UFSAR 9.3.2, "Process Sampling System"	Deleted "... evaporator condensate demineralizer samples	The UFSAR is for Unit 1 only operation. Since this equipment is not requjred for Unit 1 operation and is abandon in place, the text is being deleted.
	Deleted boron analyzer	The UFSAR is for Unit 1 only operation. Since this equipment is not required for Unit 1 operation and is abandon in place, the text is being deleted.
	Deleted hydrogen and referred to Section 11.3.2".	This change is a clarifications of information that is already contained in other sections of the UFSAR.
	Deleted " boric acid and waste evaporator	The UFSAR is for Unit 1 only operation. Since this equipment is not required fcr Unit 1 operation and is abandon in place, the text is being deleted.
UFSAR Table 9.3.2, "Process Sampling System"	Deleted " evaporator condensate derneralizer samples	The UFSAR is for Unit 1 only operation. Since this equipment is not required for Unit 1 operation and is abandon in place, the text is being deleted.
	Deleted boron analyzer	The UFSAR is for Unit 1 only operation. Since this equipment is not required for Unit 1 operation and is abandon in place, the text is being deleted.
	Deleted boric acid and waste evaporator	The UFSAR is for Unit 1 only operation. Since this equipment is not required for Unit 1 operation and is abandon in place, the text is being deleted.
	Revised temperatures and pressures.	This change is a clarifications of information that is already contained on the applicable figures in the UFSAR.
	Deleted "WTS" as a sample system.	The UFSAR is for Unit 1 only operation. Since this equipment is not required for Unit 1 operation and is abandon in place, the text is being deleted.
	Deleted "RHR" Unit 2 as a sample system.	The UFSAR is for Unit 1 only operation. Since this equipment is not required for Unit 1 operation and is abandon in place, the text is being deleted.
	Deleted "SIS" Unit 2 as a sample system.	The UFSAR is for Unit 1 only operation. Since this equipment is not required for Unit 1 operation and is abandon in place, the text is being deleted.
UFSAR 9.3.3.3, "Drains - Reactor Building"	Deleted "10 inch" butterfly.	This information was considered unnecessary or contains non-contributory details, and was deleted since this information can be found in other design

		output documents.

SA-SE Number: WBPLMN-00-010-0

Implementation Date:02/12/2000

Document Type:	***Affected Documents:***	***Title:***
Procedure Change	0-SI-65-8-B, Revision OTO	One Time Only Procedure Change

Description and Safety Assessment:

This safety evaluation addresses a one time only (OTO) change to surveillance instruction, 0-SI-65-8-B to allow a flow path to be established through open test ports and/or access doors in the Reactor Building.

WBN PER 00-004077-000 documents that during performance of the charcoal adsorber inplace test Section 6.6 of 0-SI-65-8-B, the background readings of the air stream through the EGTS filter train were too high to perform the required tests. The value measured was 2.4 ppm at both the air clean-up unit's inlet and outlet locations. This implies that the charcoal beds were saturated by the gas molecules. The background gas is believed to be R-22 which may have leaked into the annulus from the Auxiliary Building due to the relative differential pressure between the two areas during normal operation and in consideration of penetrations through the Shield Building wall.

Precautions within the test precludes this test from being performed with any fuel movement in the Auxiliary Building prevents any releases should there be a FHA. Therefore there is no possibility of any releases during fuel movement.

This procedure change will be performed while the plant is in LCO 3.6.15, and the affected plant features returned to their normal design configuration before exiting the LCO. The primary impact of the change is to the Shield Building in that its pressure boundary integrity is breached. However, the automatic containment isolation function, including the Shield Building isolation, remains intact. Therefore, in the event of an accident, on a containment vent isolation (CVI) signal, the containment and the Shield Building integrity will be automatically maintained.

In addition, the EGTS subsystems will be manually started, but the fans' handswitches will be left in "Auto". This ensures that, while in LCO 3.6.15, should there be an accident requiring the automatic actuation of both EGTS Trains, this functional integrity is maintained.

ABSCE boundary integrity is not breached by the open valves 1-FCV-30-61 and -62, which effectively connect the annulus volume to the ABSCE since the ABGTS capability includes the entire volumes Containment, Annulus, and the Auxiliary Building.

These temporary changes will not increase the off-site dose rates to the public as analyzed in UFSAR Chapter 15.1. No change will occur to the radiological consequences analyzed as a result of these changes. The credible failure modes for the systems affected by these changes have been evaluated against the accidents identified in the UFSAR. It is concluded that they do not introduce a failure pathway different from those identified and evaluated in the UFSAR accidents. The applicable accidents and the equipment served by the affected safety systems have been reviewed against these temporary changes and no new malfunction pathways will be introduced which have not been previously evaluated and identified. The Technical Specification Bases have been reviewed to determine if any margins of safety are affected by these changes. No margin of safety is identified in the Bases section which could be reduced by these changes.

SA-SE Number: WBPLMN-00-016-0

Implementation Date: 09/27/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50550-A	Fifth Vital Battery Room Ventilation and Fire Damper

Description and Safety Assessment:

DCN D-50550-A installs an opening, in the T-line wall of the Fifth Vital Battery Room (FVBR), on El. 772.0 floor, between the column lines A4 and A5. This DCN also installs a closed 3-hour-rated fire damper in the opening to maintain the integrity of the T-line wall as a 2-hour regulatory barrier. An evaluation of the wall structural integrity has shown that the new opening, equipped with a fire damper, a balancing damper, and a grille, will not affect the structural integrity of the T-line wall under any applicable load condition. This ensures that the integrity of the T-line wall as a 2-hour regulatory barrier is maintained, to allow the independent operations of the 480 volt board room 1A air conditioning system and the FVBR ventilation system. The next revision of the Fire Protection Report will document the presence of this opening and the fire damper in it. Therefore, this safety evaluation will address only the DCN D-50550-A, which installs passive features, i.e., ventilation opening in the T-line wall, a closed fire damper therein, a balancing damper and a grille.

The T-line wall, with the HVAC opening, equipped with a closed fire damper, balancing damper and a grille, is not addressed in the Technical Specification. The equipment and components, located in the two rooms adjacent to this wall, are unaffected by the proposed changes to the wall since the wall structural integrity is maintained for all load cases, and the wall's functional integrity, as a 2-hour regulatory barrier, is maintained in conformance with WBN's commitments to 10CFR50 Appendix A requirements. The FVBR ventilation and the 480 volt Board Room 1A air conditioning systems are not impacted since the T-line wall will continue to serve as a physical barrier to preserve the independent operation of these two systems. Therefore, the ability of the plant to cope with an accident situation is not adversely affected.

This change has been evaluated for plant operability during the review process and found not to adversely affect the physical plant. This change will not increase the off-site dose rates to the public, as analyzed in UFSAR chapter 15.1, nor, will it increase the radiological consequences of accidents analyzed previously. The credible failure modes for the systems affected by this change have been evaluated against the accidents identified in the UFSAR. It is concluded that they do not introduce a failure pathway different from those identified and evaluated in the UFSAR accidents. The applicable accidents and the equipment served by the affected safety systems have been reviewed against this change. and it was concluded that no new malfunction pathways will be introduced, not previously identified and evaluated. Technical Specification Bases have been reviewed to determine if any margins of safety are reduced by this change; and, none was identified.

SA-SE Number: WBPLMN-00-017-0

Implementation Date: 11/03/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50497-A	Revise System Description for Diesel Generator and Essential Raw Cooling Water to Indicate Locked Open Valves

Description and Safety Assessment:

Engineering Document Change (EDC) E-50497-A is issued to revise system descriptions N3-82-4002 (Standby Diesel Generator) and N3-67-4002 pertaining to ERCW header supply valves 1 and 2-FCV-67-66 and 1 and 2-FCV-67-67. With an emphasis on reducing the thermo-lag protection, DCN S-34175-A administratively locked open these valves. System description N3-67-4002 regarding these valves was revised accordingly; but when reviewing system description N3-82-4002, it was discovered that Sections 2.2.9.a and 3.1.2 currently reflect the existence of valve logic for these valves to open automatically upon receipt of a diesel generator start signal. DCN S-34175-A failed to revise the applicable sections of N3-82-4002 to reflect the administratively locked open position of these valves.

The changes made by EDC E-50497-A do not affect the design of the plant, physically modify any equipment in the plant, or affect how the plant is operated. Previously issued DCN S-34175-A, eliminated a potential failure mode by administratively locking open valves 1 and 2-FCV-67-66 and 1 and 2-FCV-67-67. The change to UFSAR Table 9.2-2 credits the open valves in demonstrating the ability of ERCW to supply cooling water to the diesel generators. The proposed changes do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR. A possibility of an accident or malfunction of a different type than evaluated previously in the UFSAR is not created. These changes will not reduce any margin of safety as defined in the basis for any Technical Specifications. Therefore, the changes do not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-00-018-0

Implementation Date: 04/07/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50372-A	Documentation for ERCW Valve Replacement

Description and Safety Assessment:

EDC E-50372-A addresses a replacement valve on the ERCW System in the Intake Pumping Station. This EDC documents the review and revision of various licensing basis and design basis documents as required.

The existing valve (1-FCV-67-9A) is an obsolete, Limitorque SMB-000-2 motor operated, 4-inch, flow control, Energy Products Group (EPG - a Division of Gulf & Western) ball valve and the body of the valve was replaced under a material equivalency or "like for like" procurement protocol with an E-Series Standard Port, 4 inch Anchor/Darling (A/D) ball valve. This protocol requires that the replacement component in fact be essentially an identical, or better, unit in all practical respects, e.g. weight, center of gravity, material, function, etc. During the initial procurement phase, differences were noted between the old valve body and the new valve body in that the material of construction was different. stainless steel versus carbon steel which is a significant enhancement to the ERCW System, which has shown over 20 years of system operation to not hold up to the use of carbon steel, and that the weight and corresponding center of gravity varied from the original assembly, but remained within acceptable limits, which was required if the EPG valve qualification was to be maintained or improved regarding the original calculated Frequency/Mass Distribution and Stiffness Data for seismic qualification. Therefore, the A/D valve would be an acceptable replacement valve.

The ERCW provides a safety grade water barrier (between radioactive and non-radioactive systems) while performing it's primary safety function, that of providing cooling water (heat sink) for those safety grade systems that provide plant safe shutdown capability, which contains radioactive material, or the potential to be radioactive. This water barrier also acts as the preferred heat sink for the plant safe shutdown equipment, especially the safety grade heat exchangers. As such, the ERCW System is required to mitigate the consequences of any of the plant DBEs as identified in the plant's licensing bases documents. Review of the UFSAR Chapter 15 Safety Analyses indicates that for any of the DBAs and anticipated operational transients which would require the primary safety function of the ERCW System, that respond on demand is neither diminished nor enhanced by replacement of the 1-FCV-67-9A-A valve body, and therefore, it is concluded that it is acceptable to revise the subject design documents.

The new valves will be subjected to the same environmental and service conditions as the original valves and are sufficiently similar such that the failure mechanisms remains the same as those for the old valve. Therefore, replacing the existing EPG ball valve with an A/D valve will not introduce any new, credible, equipment failure modes not previously evaluated in the UFSAR Chapter 9 failure modes and effects analyses. Therefore, the replacement valves are designed and manufactured to the ASME Section III Code, the same as the original valves, and therefore, loss of system pressure boundary as a result of the total failure of a valve bodies' pressure-retaining capability or of the flanged joint at the valve body to attaching piping is extremely unlikely due to the design, construction, qualifications, and testing of these valves. Further, The FMEA calculation referenced above indicates the redundancy designed into the ERCW Intake Pumping Station is such that the total failure of any one of either the 1A, 2A, IS, or 2B headers are backed up by at least one of two header sets for accident mitigation purposes. A total failure of one header set bounds the total single or passive failure of any one pump, piping, and valving pressure boundary, and therefore, the ERCW design meets the requirements of the ANSI/ANS-58.9, 'Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems,' and GDC 17. Specifically, these standards provide criterion for the designer which interprets the requirements of 10 CFR 50, Appendix A, with respect to design against both active and passive single failures in safety-related light water reactor fluid systems. The subject valve replacement is considered a system enhancement designed to improve maintainability, and therefore reliability, and will not adversely affect the physical form, or structural integrity of the ERCW System, or associated components in any way, nor the mechanical functioning of the ERCW system to perform its primary accident mitigating safety function.

In addition, the material breaks for the instrument line connections from the strainer flush line and the strainer back flush line were also revised (a minor editorial change).

EDC E-50372-A addresses the replacement of valve body I-FCV-67-9A. The original valve body was a carbon steel ball valve. The replacement valve is a stainless steel ball valve. The type and function of the valve is unchanged, the material actually offers enhanced system performance in that it is far more corrosion resistant than the original carbon steel valve and thus combats system malfunction related to corrosive degradation. In summary, changing the UFSAR Figure 9.2-1 to reflect a change of the carbon steel to stainless steel interface, does not increase or decrease the probability of occurrence of any of the previously analyzed events and equipment malfunctions for the ERCW System because the design and licensing bases normal position for the subject valve has not been changed. The consequences of any of the previously analyzed events and equipment malfunctions is not increased because the "post accident" and "power failure" position for these valves has not been changed. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not increased because the manner in which it is operated in support of the system's function is also unchanged. No technical specification margins of safety are reduced. This change does not affect the operation of any equipment important to nuclear safety, either directly or indirectly. Therefore, implementation of EDC E-50372-A is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number: WBPLMN-00-020-0

Implementation Date: 4/17/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50561-A DD 00-0021	ERCW Design Documents vs. As-Built Configuration

Description and Safety Assessment:

This safety evaluation addresses design changes per the scope of EDC E-50561-A. Changes have been performed to the ERCW System flow diagrams to correct specified piping line sizes at various locations. In addition, another ERCW flow diagram has been revised to depict the existence of a section of piping between two valves. These changes clarify WBN's design bases. 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 ERCW System. This EDC will ensure that the design documents are consistent with each other.

In summation, the changes:

- clarify WBN's design bases regarding the ERCW System and are intended to maintain accuracy and consistency between the UFSAR and other affected design documents with respect to the as-built configuration of the plant. This change revises the ERCW Flow Diagrams to ensure that the design documents and the as-built configuration of the plant agree.
- 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. The changes do not change the existing operation of the ERCW System.
- are not expected to adversely affect the NRC's understanding of the design, configuration, or operation of WBN's ERCW System.
- 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 ERCW System from performing its intended safety function.
- do not increase any challenges to the safety-related ERCW System assumed to function in the accident analysis such that the ERCW 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 or DBE 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 ERCW 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 ERCW System and determined not to introduce any new accident scenarios or failure pathways.

- do not increase the probability of any analyzed accident described in UFSAR Chapters 6 and 15 do not involve any new single failures as the Failure Modes and Effects Analysis 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.

- will neither increase the probability nor the radiological consequences of an accident or equipment malfunction important to safety previously evaluated in the UFSAR due to the revision to the ERCW Flow Diagrams, 1-47W845-1 and 1-47W845-4, to ensure that the design documents and the as-built configuration of the plant agree. The post-accident operation of the ERCW System is not impacted.

- do not infringe on any margin of safety defined in the basis for any technical specifications. The technical specifications have been evaluated with no impact identified.

- do not involve modifications to any radwaste system or involve any special tests or experiments.

Based on the results of this safety evaluation, it is concluded that the subject changes do not involve an unreviewed safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-00-024-0

Implementation Date: 04/12/2000

Document Type:	***Affected Documents:***	***Title:***
UFSAR	UFSAR Change Package 1628	Large Break LOCA Computer Code - LOCBART

Description and Safety Assessment:

Westinghouse has identified an error in the current safety analysis of record for the Large Break Loss of Coolant Accident (LBLOCA). The error involves the LOCBART computer code. After accounting for this error, Westinghouse has determined that a reduction in the allowable heat flux hot channel factor (F_Q), is necessary to ensure that the PCT during a postulated design basis LBLOCA remains within the regulatory limit of 2200 °F. Revision 1 has been issued to the current Cycle 3 COLR to change the required core operating limits, including a F_Q reduction from 2.5 to 2.4. UFSAR Change Package 1628, Supplement 0 is being performed to provide a reference to the COLR at locations within the UFSAR that a specific value is stated for F_Q. The changes will clarify that the COLR provides the cycle specific value for F_Q. Table 4.1 -1, Table 4.4-1, Section 15.4.1.1.5, and Table 15.4-19 are revised by UFSAR Change Package 1628.

The Westinghouse Reload Safety Evaluation has been revised to document that all safety parameters and analyses remain valid with the reduced F_Q. UFSAR Change Package 1628, Supplement 0 is simply being performed to provide a reference to the COLR at locations within the UFSAR where a specific value is stated for F_Q. This minor change will assure consistency among licensing basis documents and does not constitute a unreviewed safety question.

SA-SE Number: WBPLMN-00-027-0

Implementation Date: 10/02/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50549-A	Moisture Separator Reheater Isolation and Drain Valves Installation

Description and Safety Assessment:

Install isolation valves and drain valves in all six MSR high pressure operating vents near the No. 1 extraction header. These isolation valves used in conjunction with the FCV's in the high pressure operating vents will allow repair of the operating vents at power (if necessary). The drain valves will allow personnel to drain the operating vent during maintenance, and will allow verification that there is no leakage through the isolation valves without safety risk to personnel. Drain valves will be capped for double isolation. Carbon steel piping between the new isolation valves and the No. 1 extraction header will be replaced with chrome-moly steel to ensure that all potential leaks in carbon steel piping can be isolated.

Adding these isolation valves and drain valves to the MSR high pressure operating vents will not change the function of the heater drains and vents system as it is described in the UFSAR. There are no DBAs or operational transients in Chapter 15 of the UFSAR associated with the proposed modifications. There are no Appendix R components or equipment, or any nuclear safety related systems or portions of systems affected by the proposed modifications. The heater drains and vents system does not perform any safety related function, nor will it compromise the ability of safety related systems to perform their intended functions. Therefore this modification will not affect any DBAs or anticipated operational transients.

This DCN will not change credible failure modes of the heater drains and vents system. Credible failure modes associated with the implementation of this DCN are:

1. Failure of any of the new isolation valves to change position when being manually operated.
2. Leakage or other failure of the drain valves.
3. Catastrophic valve failure, rendering the valves useless to isolate flow.

The possible failure of the drain valves is minimized by the presence of nipples and caps, which provide a second isolation. The possible failure of the isolation valves is reduced by selection of valve materials (body of isolation valves to be chrome-moly) compatible with the conditions in the operating vent lines. These valves are selected to accommodate pressures and temperatures found in their respective portions of the heater drains and vents system. Therefore there are no new credible failure modes associated with the implementation of this DCN.

This modification 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 malfunction previously evaluated in the UFSAR, or create the possibility of an accident or malfunction of a different type from those previously evaluated in the UFSAR. This modification does not reduce the margin of safety of any basis for any technical specification. For these reasons, this activity does not constitute a unreviewed safety question.

SA-SE Number: WBPLMN-00-029-0

Implementation Date: 6/01/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50618-A	Leaking Fuel Oil Pipe and Interface Boundary Valve

Description and Safety Assessment:

DCN D-50618 B-A provides design details for the repair of a section of the non-safety-related portion of the fuel oil supply piping which has been determined to have a leak in it. The section of pipe with the leak runs from the fuel oil storage station and under the liquid nitrogen storage station.

The fuel oil supply piping to the diesel generators was suspected of leaking when fuel oil was found in the plant storm drain system. Various sections of the supply piping were tested to determine the location of the leak. The section with the leak was identified by digging a hole in the ground between the liquid nitrogen storage station and the Fire Hall to expose the fuel oil supply piping to the diesel generator. A section of pipe was cut out and a pressure test plug was installed in the open end of the piping. A valve was removed from the piping in the storage tank station and a pressure test flange was installed in that end of the piping. The piping was pressurized to approximately 98 psig. The pressure dropped approximately 50 psig in approximately 10 minutes. Other sections of piping were successfully tested with no additional leaks identified.

Since this piping runs through the bank surrounding the fuel oil storage station, under the nitrogen storage station, a concrete slab used to park truck trailers, and a set of railroad tracks; replacing the subject piping is not practical. Rather than digging a new trench to reroute the subject piping, a smaller pipe will be routed through the existing pipe and connected to the fuel oil supply piping outside of the damaged area of piping.

The fuel oil supply piping from the storage tanks to the 7-Day fuel oil supply tanks in the Diesel Generator Building is non-safety-related. It is TVA Class G (Seismic Category 1L (Q) inside the Diesel Generator Building, and TVA Class H (Non-Seismic) in the portion of piping buried in the yard. The valve 0-DRV-018-0681 is being included in the U1/U2 interface boundary program. This valve is in the Additional Diesel Generator Building fuel oil system, which is not required for Unit 1 operation. This valve has its discharge routed to the same Unit 1 drain funnel as valve 0-VTV-018-685 which is a U1/U2 interface point.

The repair of the leak in the fuel oil supply line to the diesel generators is being made in the non-safety related portion of the fuel oil supply system. This line is not required by the fuel oil system in order for the system to meet its DBA mitigation requirements.

There are not any accidents which have been evaluated in the UFSAR that may be affected by DCN D-50618-A. There are no credible failure modes added or changed as a result of these changes to the UFSAR. The change to the fuel oil supply piping is being made in the non-safety related portion of the fuel oil system. The valve 0-DRV-018-681 isolates the Additional diesel generator fuel oil system from a Unit 1 drain funnel. These modifications do not affect the safety related portions of the fuel oil system, or its ability to mitigate the radiological consequences of an accident which is evaluated in the UFSAR. Therefore, the probability of occurrence or the consequence of any accidents or equipment malfunction is not increased. In addition, the changes do not affect the consequences of any previously evaluated accidents because the ability to transfer fuel oil to the diesel generators has not been adversely impacted. The changes do not have the potential to create a new accident or equipment malfunction; nor do the changes to the UFSAR figures affect the margin of safety defined in the Technical Specification Bases. Therefore, this change does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-00-031-0

Implementation Date: 09/22/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50506-A UFSAR Chg Pkg 1633	Positive Displacement Pump Abandoned

Description and Safety Assessment:

DCN D-50506-A abandons in place the Unit 1 PD pump and associated components. The pump at WBN has had a history of operational problems and excessive maintenance. As a result, the modification implemented by DCN D-50506-A will mechanically and electrically isolate the pump from interfacing plant systems. The PD pump was designed to deliver charging flow to the reactor coolant system and reactor coolant pump seals during normal power operation. The PD pump is not relied upon to safely shutdown the plant and maintain it in a safe shutdown condition or prevent or mitigate a DBE. Since the pump has not operated properly in the past and because of pressurizer overfill concerns, the pump has been tagged out of service as described in UFSAR Section 15.2.14. Either of the two centrifugal charging pumps (CCP) normally supplies the charging flow. The PD pump is not aligned to nor does it receive a SIS signal to start on the initiation of SIS operations. The CCPs are designed to provide the ECCS safety function of the Chemical and Volume Control System (CVCS) during a DBE. The pump abandonment prevents the PD pump from continuously operating during an inadvertent SI signal which alleviates the pressurizer overfill concerns of the UFSAR analysis associated with PD pump operation. The pump and components associated with the abandonment retain their seismic I classification so that the abandoned equipment will not compromise the reactor coolant pressure boundary. The margin of safety as defined in the basis for any Technical Specification is not reduced.

The CCPs are designed to provide a portion of the ECCS safety function during a DBE. Implementation of DCN D-50506-A does dictate that the CCPs now also operate for normal charging and seal injection functions which were performed by the PD pump therefore resulting in additional operation of the CCPs. Because of their use during normal operation, the centrifugal charging pumps are expected to experience increased wear. However, this increased wear is evaluated through periodic inspection, maintenance, and monitoring of the CCPs.

The main concern with maintaining the CCPs in an operable condition has been the occurrence of shaft failures. CCP shaft failure has been identified as a generic industry problem. However, WBN has not experienced any shaft failure problems to this date. As a result of CCP problems at other utilities, WBN has initiated steps to reduce the potential for shaft failures. Previously, WBN implemented a design change, DCN M-20638-A, to continuously vent hydrogen gas from the CCP suction flowpath back to the Volume Control Tank. This change eliminated gas intrusion into the pump as one of the contributors to pump failure. In addition, WBN replaced the CCP 1A-A pump shaft during the Unit 1 Cycle 2 Refueling Outage. The 1B-B CCP shaft is scheduled to be replaced during the Unit 1 Cycle 3 Refueling Outage. The shaft material for CCP 1B-B may be upgraded based on information collected from other utilities. The CVCS system description will be revised to state that "the CCP shaft will be evaluated for replacement on a scheduled frequency consistent with the increased use of the pumps for normal operation." Future shaft replacements will be implemented based on the results of ongoing surveillance and inspection activities, as well as a review of industry experience at other utilities. WBN's actions to replace the CCP shafts, and the CCP inspection, maintenance, and surveillance programs ensure that the CCPs will be able to perform their design basis functions.

The Technical Specification limits imposed on the CCPs remain valid. The removal of the PD pump from the licensing and design basis does not impact the normal charging operation of Watts Bar. This design change does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-00-032-0

Implementation Date: 2/05/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50107-A UFSAR Change Pkg 1632	Reactor Coolant System Leak Detection Methods

Description and Safety Assessment:

UFSAR Section 5.2.7.8 currently requires a correlation of the containment pocket sump level monitor, the containment air particulate radiation monitor, and the containment air gas radiation monitor for RCS leak detection purposes per the guidelines of NRC Regulatory Guide 1.45.

The primary purpose of the RCS leak detection requirements is to identify a relatively small breach in the RCS early so that actions can be taken by Operations to prevent a more serious loss of reactor coolant. Inability of the radiation monitor gas channel to detect a 1 gpm RCS leak in 1 hour during times of low RCS source terms does not compromise this function. Regulatory Guide 1.45 recognizes that the radiation monitors will be of limited value during times when the RCS source terms are low, and indicates additional supplementary methods to support leak detection should be available during these times. This change establishes design and operational radiation monitor setpoints for the particulate and gas channels, and requires a comparison of the sump rate of rise and performance of an RCS mass balance and/or chemical analysis of the RCS to confirm the existence of a RCS leak. DCN D-50107-A establishes this methodology in design output, which will be translated to Operations procedures. In addition, for consistency with Regulatory Guide 1.45 and the Bases for Technical Specification, the required sensitivity of the noble gas channel is changed by DCN D-50107-A from 2.09×10^{-5} μCi/cc to 1.12×10^{-5} μCi/cc. This methodology will serve as an indirect method of correlating the response of the radiation monitors and pocket sump.

As discussed in UFSAR Section 5.2.7, in addition to the two radiation monitors and the containment pocket sump, RCS leak detection can be supported indirectly by containment temperature monitoring and humidity monitoring. Further, the changes implemented by DCN D-50107-A require a RCS mass balance be performed to confirm indications (from the sump level monitor or the radiation monitors) of an RCS leak. This methodology will provide an additional method of RCS leak detection in addition to providing an indirect method of correlating the response of different leak detection methods. Since both the particulate and gas channels of the radiation monitors, and the containment pocket sump level monitor will detect a 1 gpm RCS leak within 1 hour when the design basis source terms are present in the RCS, the design. basis, UFSAR, and Technical Specifications requirements are met. It is therefore concluded the function of early RCS leak detection per the requirements of the design basis, UFSAR, Technical Specifications, and Regulatory Guide 1.45 are satisfied.

Both the gas and particulate channels of the containment air monitors will detect a 1 gpm RCS leak within 1 hour with design basis source terms present in the RCS. Consequently, these monitors meet the design basis document requirements. As an enhancement, DCN D-50107-A includes a revision of RCS System Description to include the requirement for Operations to perform an RCS mass balance if an alarm is received on the particulate or gas monitor in conjunction with an increasing rate of rise on the pocket sump monitor, or if an alarm is received on the pocket sump monitor. This section is also revised to indicate a chemical analysis of the RCS may become necessary to confirm the RCS source term has increased if radiation monitor response is increasing without a corresponding increase in RCS leakage. Consequently, this change is safe from a design basis document standpoint.

Integrity of the reactor coolant pressure boundary is not adversely affected by this change since the existing RCS leakage detection equipment will meet the design and licensing basis requirements as defined in the RCS design criteria and the UFSAR, and additional changes made by DCN D-50107-A enhances detection of a 1 gpm RCS leak within 1 hour during times when the RCS source terms are less than the design and licensing basis source terms.

Revising radiation monitor setpoints and requiring an RCS mass balance introduces no new failure modes since no new equipment is added by this change. Radiation monitors 1-RE-90-106 and 112 are not primary safety function monitors and therefore not redundant, trained monitors.

The changes made by DCN D-50107-A do not constitute a unreviewed safety question since the pocket sump monitor and both the gas and particulate radiation monitors will detect a 1 gpm RCS leak within 1 hour with the design and licensing basis RCS source terms as evaluated in the UFSAR. The changes implemented by DCN D-50107-A do not result in an increase in the probability or consequences of accidents or malfunctions analyzed in the UFSAR, do not reduce Technical Specification margin of safety, and do not create the possibility of a new accident or malfunction. This change is adding no new equipment and is not introducing new failure modes of existing equipment. Changes made by DCN D-50107-A will enhance detection of a 1 gpm RCS leak within 1 hour during times when the RCS source terms are less than the licensing basis source terms (UFSAR Table 11.1-7). Operations ability to detect a 1 gpm RCS leak within 1 hour will be enhanced, thereby facilitating actions to limit RCS leakage before such leakage can propagate to a larger loss of coolant. In accordance with Regulatory Guide 1.45, other indirect methods of RCS leak detection, such as temperature monitoring and humidity monitoring, are available during times when the RCS source terms are less than the design and licensing basis source terms. Consequently, the RCS leak detection monitors meet design, licensing, and technical specification requirements.

SA-SE Number: WBPLMN-00-035-1

Implementation Date: 6/20/2000

Document Type:	***Affected Documents:***	***Title:***
UFSAR	UFSAR Change Pkg. 1635	Barge Shipments Past WBN Site

Description and Safety Assessment:

UFSAR Change Package No. 1635, is being performed to revise Table 2.2-1 "Waterborne Hazardous Material Traffic (Tons)" to add a note explaining that recently initiated ethanol barge shipments past the WBN Site have been evaluated within the design basis. The design basis evaluation demonstrates that no explosion hazard exists due to the ethanol shipments. In addition, the table is classified as Historical Information by UFSAR Change Package 1635. Also as part of UFSAR Change Package 1635, minor corrections of data such as addition errors are corrected.

The design basis calculations which demonstrate the no explosion hazard exists also ensure that the possibility for an accident or equipment malfunction of a different type is not created. Shipment of hazardous materials past the plant is not the subject of any technical specification. In addition since design basis calculations have been performed to demonstrate that the risk of significant damage due to an ethanol explosion is sufficiently low to render further analysis unnecessary, the margins of safety for equipment that is addressed by the Technical Specifications are also not reduced due to indirect effects. Table 2.2-1 is classified as Historical information by UFSAR Change Package 1635. Also as part of UFSAR Change Package 1635, minor corrections of data such as addition errors are corrected. The change to classify Table 2.2-1 as Historical Information is consistent with the Guidance provided in Nuclear Energy Institute (NEI) 98-03, Revision 1. Appendix A3 of NEI 98-03 does not require licensees to update historical information to reflect minor changes in the site environment, such as for nearby facilities. However NEI 98-03 does require that the design basis be maintained up to date. The revision to the calculation which has been performed in support of this activity serves to maintain an updated design basis. The corrections to the tabular data of Table 2.2-1 are considered editorial in nature. Therefore, the changes of UFSAR Change Package 1635 are safe from a nuclear safety perspective and do not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-00-038-0

Implementation Date: 5/27/2000

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 0-00-002-043	Waste Gas Analyzer Compressors

Description and Safety Assessment:

The waste gas analyzer sequences through various samples for Unit 1 and 2. The analyzer system consists of an analyzer (oxygen (O_2)), two Waste Gas compressors in series, and a multipoint sampling header. The O_2 analyzer (0-02E-043-0450) is located in the Unit 2 Hot Sample Room and is 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 reactor coolant drain tank (RCDT), CVCS volume control tank (VCT), spent resin storage tank (SRST), pressurizer relief tank (PRT), waste gas decay tanks (WGDTs), and CVCS Holdup Tanks (HUTs). If the analyzer is inoperable for more than 30 days a report is required to be filed with the NRC. The samples have their pressure reduced, if required, to approximately 5 psig, and then compressed with two compressors to approximately 37 psig before entering the analyzer. The analyzer pressure control valve (PCV) currently controls the pressure to 20 ± 2 psig, flow thru the analyzer to 344 ± 50 cc/min, and a minimum flow thru the bypass of 3,000 cc/min.

A radioactive gas leak was found in Waste Gas Analyzer Room, PER was written, and the source was determined to be the second compressor. Since this compressor has failed several times, a trouble shooting WO was initiated to determine the cause at the failure and to obtain additional data using only one compressor (flow rates and pressure). The WO showed that PCV on the WGDT was not reducing the pressure to approximately 5 psig, and the PCV an the analyzer was not controlling pressure. Both PCVs were either repaired or replaced. The data was reviewed, and determined that one compressor was sufficient to supply the required flow and pressure. TACF 0-00-002-043 revises the CCDs to show that only one of the two waste gas analyzer compressors will be used, and clarifies the pressure and flow requirements for the analyzer.

This TACF does not create new failure modes of equipment involved in termination of releases due to potential activity above the 10 CFR 20 limits. The use of only one of the two compressors could allow gaseous release in the Auxiliary Building if pressure boundary was lost. Any releases should be less than the 10 CFR 20 limits even if this were to occur radiation level would be detected by an area monitor (2-RE-90-007) which alarms in the MCR and a portable continuous air monitor supplied by Site RADCON that alarms locally. These changes do not create any unacceptable equipment failure modes that would cause the Waste Gas Analyzer to be unable to perform its function of processing sample gases, nor do these changes affect failure modes of equipment that are important to safety. This TACF does not add any additional or different types of failure modes that have not been addressed in the UFSAR.

No new potential single failures of existing components will occur as a result of rerouting the discharge of the first compressor to the inlet of the heat exchanger supplying the O_2 analyzer. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The Waste Gas O_2 analyzer, its associated components, and piping do not perform any accident mitigation function. These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBE, all safety-related equipment is expected to operate as designed to limit the consequences of the DBE.

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. The ODCM limits for releases from the Waste Gas Disposal System are not revised or challenged by these changes. Based on the results of this safety evaluation. it is concluded that the subject changes do not involve an unreviewed safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-00-039-1

Implementation Date: 12/14/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50373-A UFSAR Chg Pkg 1637 TS Bases Change 00-012	Removes Requirement for Periodic Testing of ABGTS fans.

Description and Safety Assessment:

EDC E-50373-A implements a change to System Description for Auxiliary Building - HVAC System, to remove the requirements for the periodic testing of the ABGTS fans to draw down the ABSCE to a minimum negative pressure of 0.25-inch wg within 4 minutes after the receipt of an Auxiliary Building isolation (ABI) or high radiation signal, with one Auxiliary Buildling general supply fan operating, and its isolating damper open. The time limit has not been removed. This EDC also adds to the system description, periodic inspection of the ABI damper seals for the Purge Air Supply, and the Auxiliary Building General Vent Supply to ensure continued performance as demonstrated in the preoperational and previous surveillance testing.

The ABGTS, its associated components, duct work, and fans are located in the Auxiliary and Reactor Buildings. The ABGTS is a fully redundant air cleanup system which is provided to reduce radioactive releases from the ABSCE to the environment during an accident to levels sufficiently low to keep the site boundary dose rates below the requirements of 10 CFR 100. This is accomplished by exhausting air from the ABSCE to maintain a negative pressure within the boundary. Exhaust air leaving the ABSCE is processed by the ABGTS filter train before it is discharged to the outside. The duct work associated with either Containment Purge Air, Incore Instrument Room, or Auxiliary Building General supply does not allow additional leakage from the ABSCE, but would increase clean air coming into the ABSCE.

The Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decay Tank or associated piping. This modification will not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis of the Waste Gas Decay Tank. The EDC does not change the logic or function of any system that is important to safety.

The ABGTS has the capabilities needed to preserve safety in accidents as severe as a LOCA. This was determined by conducting functional analyses of the system to verify that the system has the proper features for accident mitigation which consist of a failure modes and effects analysis, a review of Regulatory Guide 1.52 sections to assure licensing requirement conformance, and a performance analysis to verify that the system has the desired accident mitigation capabilities. A detailed failure modes and effects analysis is presented in the UFSAR Table 6.2.3-3. The equipment involved in the change does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the UFSAR. This EDC does not change or affect the design basis for any system that is important to safety.

No new potential single failures of existing components will occur as a result of this change. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. This change does not affect any equipment required for safe operation or shutdown. In the event of a DBA, all safety related equipment is expected to operate as designed to limit the consequences of the DBA.

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. The ODCM limits for releases from the Waste Gas Disposal System are not revised or challenged by these changes

Based on the results of this safety evaluation, it is concluded that the subject changes do not involve an unreviewed safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-00-040-0

Implementation Date: 07/18/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50490-A UFSAR Change Package 1645	ABSCE Inleakage Rate

Description and Safety Assessment:

EDC E-50490-A is issued to make three changes to System Description N3-30AB-4001 as follows: (1) change the air flow range associated with 1-AHU-031-0475-B in Table 9.5 from "+10% / -0%" to "+10%/-15%"; (2) clarify Table 9.8 to state "Train A AFW" for cooler 1A-A and "Train B AFW" for cooler 1B-B in the Unit 1 ABGTS room, instead of "Train A and B AFW" as currently stated for both coolers; and (3) revise Section 6.0 to simplify the acceptance criteria for the ABGTS pressure test to require measurement and verification that minimum vacuum relief flow rate is achieved, in place of calculating and verifying the maximum building in-leakage rate based on measured ABGTS fan flow rate and vacuum relief flow rate. Item (3) above will require a change to the UFSAR and is the subject of this evaluation.

This change to Section 6.0 of the System Description will provide a more direct method of verifying during testing that ABSCE in-leakage rate has not exceeded design requirements. The existing requirements for the ABGTS pressure test are that total ABGTS fan flow rate be measured and verified to be between 9300 cfm and 9900 cfm, that the total vacuum relief make-up flow rate be measured, and that a total in-leakage value be calculated based on the difference between the measured ABGTS fan and vacuum relief flow rates. This difference between the measured values is then verified to be less than or equal to 7930 cfm. The proposed change to the test requirements will require comparison of the measured vacuum relief flow rate to a calculated acceptance criteria of 1370 cfm minimum, and will no longer require verification that the in-leakage rate be less than or equal to 7930 cfm. This proposed method of verification is based on the current design, and offers a simpler, direct comparison to measured values. The current maximum value for ABSCE in-leakage rate, 7930 cfm, is derived from the difference between the minimum acceptable ABGTS air flow rate of 9300 cfm, and the maximum postulated in-leakage due to a compressed air line break of 1370 cfm. Verification that the vacuum relief flow rate is greater than 1370 cfm demonstrates that sufficient margin exists to overcome the effects of the postulated rupture of a compressed air line leaking into the ABSCE boundary.

The ABGTS is designed to operate during and after any DBE except the loss of all AC power. The system is required to maintain air pressure in the ABSCE below atmospheric during an accident to prevent unfiltered releases, to reduce the concentration of radioactive nuclides in releases to the environment, and to minimize the spread of airborne radioactivity in the Auxiliary Building following an accident. The ability of the ABGTS to perform as required during an accident is unaffected by this proposed change, since the existing design values for ABGTS air flow rate and ABSCE postulated in-leakage rate are not affected. The integrity of the ABSCE is maintained, since no additional in-leakage is permitted as a result of this change.

System Description will require the measurement of vacuum relief flow rate during ABGTS testing to verify that a minimum flow of 1370 cfm is achieved. The verification that the vacuum relief flow rate is greater than 1370 cfm demonstrates that sufficient margin exists to overcome the effects of the postulated rupture of a compressed air line leaking into the ABSCE boundary. Current testing requirements specify the verification that ABSCE in-leakage rate does not exceed a predetermined value of 7930 cfm (based on the minimum ABGTS flow of 9300 cfm - 1370 cfm). This change simplifies the current verification method by making a direct comparison to the measured vacuum flow rate in place of calculating in-leakage rate. The change is based on flow rate values currently included in existing design calculations, so there is no effect on the original design parameters for the operation of the ABGTS or to the in-leakage values for the ABSCE. There is no physical modification required as a result of this change. There is no impact to the technical specifications as a result of this change. This change will not increase the off-site dose rates to the public as analyzed in UFSAR Chapter 15, nor will it increase the radiological consequences of accidents analyzed previously. The applicable accidents and equipment served by the affected safety systems have been reviewed against this change, and it is concluded no new malfunction pathways will be introduced that were not previously identified and evaluated. Based on the above, this change does not constitute a unreviewed safety question.

SA-SE Number: WBPLMN-00-041-0

Implementation Date: 09/08/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50433-A UFSAR Chg. Pkg. 1640	Review of Valves Locked by Design

Description and Safety Assessment:

Design change E-50433-A implements a documentation review of valves locked by design. The review involved the review of all pertinent system flow diagrams to ensure all by design locked valves are consistently and accurately depicted on the flow diagrams. Associated system descriptions were also scrutinized and some revised to minimize duplicated information. The revisions to various flow diagrams and related system descriptions do not alter the depicted position of any valve or damper with the exception of damper 0-31-2114. The depicted position of this damper was revised to conform with System Description N3-30CB-4002 and UFSAR Section 9.4.1.2. Some UFSAR figures were revised but the changes do not increase the likelihood of a postulated accident or malfunction, introduce a new accident or malfunction, or increase the radiological release associated with any postulated accident or malfunction. No margin of safety is affected by the documentation changes. Thus no unreviewed safety question is associated with this change.

SA-SE Number: WBPLMN-00-043-0

Implementation Date: 07/10/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50413-A	Balancing Damper above Main Control Room Chiller

Description and Safety Assessment:

EDC E-50413-A is issued to allow the adjustment of balancing damper, as required, to reduce the flow of ventilation air to the area above MCR Chiller A, located in the Auxiliary Building on Elevation 737.0, Room A1. MCR Chiller A has a water spray shield constructed above it, and ventilation supply air from the non-safety related Auxiliary Building General Ventilation System is ducted directly through an opening in this spray shield. When the outdoor temperature is between 40°F and 60°F, the outdoor supply air is not heated or cooled by the supply coils, resulting in unconditioned air being supplied through the system. During the Spring and Fall seasons, supply air from the General Ventilation system may become cold. When this cold air is ducted directly on the chiller, the entire chiller, including the refrigerant oil, may be cooled to temperatures which are below the manufacturer's recommended start-up temperatures. This change is to permit the adjustment of balancing damper 1-BLD-031-3085, located directly above the chiller at the supply grille, to reduce or stop the flow of ventilation air, as needed, and prevent the excessive cooling of the water chiller.

During a DBA, the Auxiliary Building General Ventilation system automatically stops, air flow patterns are established and maintained by the ABGTS, and cooling in safety related equipment areas is accomplished by the ESF pump room and area coolers. The Auxiliary Building General Ventilation System does not operate during a DBA. There are no credible failure modes introduced that are associated with the adjustment of manual damper 1-BLD-031-3085, which is a passive device.

The adjustment of the balancing damper above Main Control Room Chiller A will prevent the possibility of cooling the chiller package to below the manufacturers recommended start-up temperature while on stand-by. This adjustment will reduce air flow to one grille in the non-safety related Auxiliary Building General Ventilation System, but will cause no substantial change in the cooling air supplied to room 737.0-Al, where the chiller is located. There is no impact to the technical specifications and no change to the Design Basis for the General Ventilation system. This change was evaluated for plant operability during the review process and found not to adversely affect the physical plant. This change will not increase the off-site dose rates to the public, as analyzed in UFSAR Chapter 15, nor will it increase the radiological consequences of accidents analyzed previously. The applicable accidents and equipment served by the affected safety systems have been reviewed against this change, and it was concluded that no new malfunction pathways will be introduced that were not previously identified and evaluated. Based on the above, this change does not constitute a unreviewed safety question.

SA-SE Number: WBPLMN-00-046-0

Implementation Date: 10/24/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50660-A UFSAR Chg Pkg 1644	Solid Radwaste Disposal System (SRDS)

Description and Safety Assessment:

The Solid Radwaste Disposal System (SRDS) processes and packages, for offsite shipment and disposal, the dry and wet solid radioactive waste produced through power generation. The dry active waste (DAW) consists of compactible and noncompactible material. The wet active wastes (WAW) consist of spent resins generated as a result of the operation and maintenance of WBN. This radwaste consists of CVCS damineralizer resins which processes RCS water, and Spent Fuel Pool demineralizer resins. These resins are temporarily stored onsite in the SRST, and after accumulation of sufficient resin for off site shipment, the SRST contents are normally processed to a shipping container. The shipping container consists of an inner disposable high integrity container (HIC) with an outer returnable shield. Filter elements are mounted inside the liner near the bottom and are connected to a hose connection outside the shield to facilitate dewatering spent resins. The container also has fill and vent connections.

WBN has identified two issues with SRDS. The first is that the UFSAR Section 11.5.3.1 describes transfer of spent resin to a shipping container licensed pursuant to the general license provisions of Paragraph 71.12 (b) 10 CFR Part 71. Contrary to this requirement spent resin were transferred from the SRST contents to a shipping container that is no longer licensed by the NRC. PER 00-007771-000 has been written to address this problem. The second issue deals with the need to transfer the SRST contents to another location so that repairs can be made on the tank. This need is due to a management decision not to ship to the Barnwell Waste Management Facility, and WBN is waiting on NRC's approval for the radioactive waste storage modules at Sequoyah Nuclear Plant to be licensed to receive radioactive waste from WBN.

The transfer of the SRST contents to the Rad-Vault is the only issue that is addressed any further in this Safety Evaluation since the change for the shipping container requirements has been approved by NRC in 10 CFR Part 71. The DBAs and anticipated operational transients identified in UFSAR Chapter 15 is the potential increase of off site dose to the environment beyond 10 CFR 20 and 10 CFR 100 limits. This event is most closely approximated by the Infrequent Fault of Waste Gas Decay Tank Rupture (UFSAR Section 15.3.5).

This EDC does not create new failure modes of equipment involved in termination of releases due to potential activity above the 10 CFR 20 and 10 CFR 100 limits . Any releases will be less than the 10 CFR 20 and 10 CFR 100 limits even if this were to occur. No single failure of a different type has been created by this design change. These changes do not create any unacceptable equipment failure modes that would cause the SRDS to be unable to perform its function of processing solid waste, nor do these changes affect failure modes of equipment that are important to safety. This EDC does not add any additional or different types of failure modes that have not been addressed in the UFSAR. If a leak were to occur that became airborne, increased radiation levels would be detected by a CAM in the Auxiliary Building Railroad Bay that alarms locally, and by the Auxiliary Building Vent Radiation Monitor which alarms in the MCR.

The SRDS associated components, piping, and valves are located in the Auxiliary and Reactor Buildings. This system is not safety related, is installed in a seismic structure, and is not used during any accident. The Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a WGDT as a consequence of a failure of a single WGDT or associated piping. This modification will not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis of the WGDT. The EDC does not change the logic or function of any system that is important to safety.

UFSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, include fuel cladding defects in combination with malfunction in Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. The EDC is not associated with the equipment that could cause these events. Also, this change is not associated with the protective features used to detect and mitigate the effects of these events. The equipment involved in the change does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the UFSAR. This EDC does not change or affect the design basis for any system that is important to safety.

No new equipment has been added by this change, and no new potential single failures of existing components will occur as a result of processing the SRST contents to the Rad-Vault. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. This system, its associated components, and piping do not perform any accident mitigation function. These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBE, all safety related equipment is expected to operate as designed to limit the consequences of the DBE.

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 an unreviewed safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-00-051-0

Implementation Date: 08/22/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50649-A and UFSAR Change Package 1647	Use of Existing New Fuel Elevator

Description and Safety Assessment:

This change (DCN D-50649-A) documents the requirements for the use of the existing New Fuel Elevator (NFE) (WBN-O-ELEV-079-0001) to repair irradiated fuel assemblies. The NFE is located on Elevation 757.0 of the Auxiliary Building and is part of the Fuel Handling and Storage System (FHSS). The NFE is normally used to lower a new fuel assembly into the FTC where it is transported to the Fuel Transfer System.

The use of the NFE is necessary as a result of notification by Westinghouse Nuclear Safety Advisory Letter, NSAL-99-004, concerning fuel assembly top nozzle spring screw failure, subsequently PER 99-014580-000 was written at WBN. As part of the PER corrective action the top nozzles will be replaced on all fuel assemblies identified with fractured spring screws. The probability of the automatically and administratively safeguards failing to keep the irradiated fuel assembly below the safe shielding depth is very low, less than 1 E-6. The mechanical hard stop developed for use on the NFE is a fail-safe physical stop. The stop provides a level of assurance comparable to the bridge crane that irradiated fuel will not be lifted out of the water. By repairing the nozzle spring screws the potential for a future fuel handling problem has been reduced. A similar successful repair of fuel assemblies has occurred at R. E. Ginna using the NFE as a work platform.

A fuel handling accident is the only accident which this change would have any potential of affecting. There are no credible failure modes added or changed as a result of this change to the UFSAR. The NFE will be utilized to facilitate repair of the defective fuel assemblies as described in Westinghouse Nuclear Safety Advisory Letter, NASL-99-004. Dropping a fuel assembly has been evaluated in UFSAR section 15.5.6. As with all operations that involve the handling of irradiated fuel assemblies, the potential for a fuel handling accident exists if the NFE is used to raise and lower irradiated fuel assemblies. The population of fuel assemblies transported in the NFE is expected to be small compared to the total population of fuel assemblies handled by the bridge crane. The NFE is designed to operate with only one fuel assembly at a time. This change does not modify the elevator capacity. The existing accident analyses for a fuel assembly drop bounds the potential drop of an irradiated fuel assembly from the NFE. The potential off-site doses will remain within the current off-site dose limits and the corresponding 10 CFR 100 guidelines.

Therefore, the probability of occurrence or the consequence of any accidents or equipment malfunction is not increased. In addition, using the NFE to raise or lower irradiated fuel assemblies to facilitate maintenance on the fuel assemblies, does not affect the consequences of any previously evaluated accidents. The change does not have the potential to create a new accident or equipment malfunction; nor does the documentation change to the UFSAR affect the margin of safety defined in the Technical Specification Bases.

Therefore, this change does not constitute a unreviewed safety question.

SA-SE Number: WBPLMN-00-058-0

Implementation Date: 10/25/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50534-A UFSAR Chg Pkg 1649	Use of Hydrazine as Oxygen Scavenging Agent

Description and Safety Assessment:

EDC E-50534-A revises the UFSAR and the SDD to clarify that hydrazine may be employed as a oxygen scavenging agent before or during cooldown prior to placing the RCS on RHR. Hydrazine is currently being added to CVCS during heatup when the RHR is being used to circulate the RCS.

The RHR associated components, piping, and valves are located in the Auxiliary and Reactor Buildings. This design change does not change the logic or function of any system that is important to safety. The DBAs and anticipated operational transients identified in UFSAR Chapter 15 have been evaluated with respect to the potential increase of off site dose to the environment beyond 10 CFR 20 and 10 CFR 100 limits. The event requiring consideration is the Limiting Fault of LOCA. This change does not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis. The EDC does not change the logic or function of any system that is important to safety.

The limiting equipment RHR fault which could occur, is the RHR pump seal failure. The water will spill out on the floor in a shielded compartment. The water spillage will drain to the sump. This change does not affect this equipment fault. No new permanent equipment has been added by this change. However, the injection equipment is not safety related and is not credited in the event of a DBE. During a DBE, manual actions are relied on to manually isolate the injection equipment from RHR. An RHR train cannot be credited as TS operable when the isolation valve is open for hydrazine injection. In the event of a DBE, all safety related equipment is expected to operate as designed to limit the consequences of the DBE. This EDC does not change or affect the design basis for any system that is important to safety.

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 an unresolved safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-00-67-0

Implementation Date: 01/10/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	UFSAR Change Package 1650	Deletion of "Zero Leakage to Atmosphere" in UFSAR

Description and Safety Assessment:

Table 6.3-6 will be revised to delete the footnote that attempts to justify zero leakage to the atmosphere from the RHR pump, SIS pump, and CCP mechanical seals. Table 6.3-6 footnote will be supplemented to explain that although minimal seal leakage may develop periodically, infrequent minor pump seal leakage is bounded due to the large margin that exists between the total realistic ECCS recirculation look leakage and the total evaluated leakage of 3760 cc/hr. Table 6.3-6 will also be revised to specify zero leakage to the drain tank from the ECCS pumps.

This change is necessary to reflect actual plant conditions. It was originally expected that the ECCS pumps would be supplied with tandem seals that would essentially eliminate pump leakage. However, when the final NSSS design supplied pumps with single mechanical seal faces, the UFSAR table was not revised accordingly. Also, minimal leakage from the mechanical seals may develop periodically due to normal wear associated with the operation of rotating equipment.

The infrequent presence of minor mechanical seal leakage from the ECCS pumps does not increase the probability of occurrence of any of the accidents currently analyzed within chapter 15 of the UFSAR. The proposed change will not increase the probability of occurrence of a malfunction of the ECCS pumps. Minimal leakage from the mechanical seals may develop periodically due to the normal wear associated with the operation of rotating equipment. However, infrequent minor pump seal leaks are not indicative of impending gross leakage from the seals or complete failure of the seals. Minor pump seal leaks are monitored and repaired within the plant's normal maintenance program. The equipment continues to operate as designed and the ECCS pumps will continue to be able to perform their required design basis function. The proposed change does not increase the consequences of an accident previously evaluated in the UFSAR. Offsite and Control Room dose subsequent to a LOCA is presently analyzed concurrent with a total ECCS recirculation leak to the Auxiliary Building atmosphere of 3760 cc/hr. Although the dose analysis is based on 3760 cc/hr leakage to the atmosphere, it also determines a realistic leakage of 94 cc/hr from all components (excluding pump seals) within the recirculation loop external to containment. Therefore, there is sufficient margin in the dose analysis to accommodate infrequent minor seal leakage from the ECCS pump seals such that calculated Offsite and Control Room dose will not be affected. The changes to UFSAR Table 6.3-6 to clarify their seal configuration and to acknowledge that infrequent minor seal leaks may occur does not have the potential to increase the consequences of malfunctions UFSAR. The proposed change does not have the potential to create a possibility for a malfunction or accident of a different type. No new failure mechanisms are created. The proposed change does not reduce the margin of safety as defined in the basis for any technical specification. Technical specification controls will remain in place to minimize leakage from RHR system and CVCS and to require preventive maintenance and integrated leak testing. The proposed change to Table 6.3-6 will acknowledge minimal leakage from the mechanical seals of the RHR, SIS and CCP pumps due to the normal wear associated with the operation of rotating equipment. This will not reduce the margin of safety as the total recirculation loop leakage external to containment from all contributors does not change.

In conclusion, the changes of UFSAR Change Package 1650 do not constitute an unreviewed safety question and are safe to implement from a nuclear safety perspective.

SA-SE Number: WBPLMN-00-068-0

Implementation Date: 09/27/2000

Document Type:	***Affected Documents:***	***Title:***
Other	Fire Protection Report, Revision 13	Fire Protection Report Discrepancies

Description and Safety Assessment:

The proposed change to the FPR will correct the discrepancies identified in PER 00-003832-000, PER 00-004862-000, and PER 00-07596-000. This revision to the report will correct the fire severity classification in Part VI for room 692.0-A31 to agree with the combustible loading calculation. Also, Parts II and V of the Report will be revised to remove the Shift Technical Advisor from the listing for the minimum crew to agree with the information in the INDMS-SSA and AOI-30.2. In addition, this revision will list the manual actions required in Part V1, Section 3.49.2 for 1-FCV-63-118 and Section 3.22.2 for 1-FCV-63-80-A, 98-B, and 118-A that are required by the existing Safe Shutdown Analysis.

There is no physical modification or revision to any calculation or analysis required by this change. There is no impact to the Technical Specifications as a result of this revision, and no margins of safety are affected or introduced. The change will not increase the off-site dose rates to the public as analyzed in UFSAR Chapter 15. 1, nor will it increase the radiological consequences of accidents analyzed previously. The change to the Report does not create additional malfunction pathways that have not previously been identified and evaluated. The identified changes to the FPR reflect the current established requirements in the Safe Shutdown Analysis, which evaluates the effects of an Appendix R fire on plant systems. Based on the above, this change does not constitute a unreviewed safety question.

SA-SE Number: WBPLMN-00-070-0

Implementation Date: 09/11/2000

Document Type:	***Affected Documents:***	***Title:***
UFSAR	UFSAR Change Package 1656	Heatup and Cooldown Rates

Description and Safety Assessment:

UFSAR Change Package 1656 revises the UFSAR to resolve discrepancies between the UFSAR and plant operating procedures regarding the cooldown rate of the reactor during unit shutdown. As described in PER 00-011420-00 below, plant operating instructions specify administrative cooldown rates which are different from those described in the UFSAR.

PER 00-011420-00 PROBLEM DESCRIPTION

> "During review of the GO-6 revision for permitting P-12 Bypass, it was determined that the target RCS cooldown rate of 60°F/hr allowed by GO-6 is not consistent with WBN UFSAR, Section 5.4.1.2 which states, "The heatup and cooldown rates imposed by plant operating limits are 50°F/hr for normal operations and 100°F/hr under abnormal or emergency conditions",. AND Section 5.2.1.5, which, with regard to heatup and cooldown rates, states, "The expected normal rates are 50°F/hr." Section 5.4.2.3 of the UFSAR states, "The heatup and cooldown curves are given in the PTLR required by the technical specifications." The PTLR allows RCS cooldown up to 100°F/hr. "

UFSAR Change Package 1656 revises the UFSAR to remove references to specific administrative heatup and cooldown temperature limits and specify that the maximum rate will be in compliance with the design requirements specified for RCS cooldown, as specified in the Pressure and Temperature Limits Report (PTLR). The PTLR is attached to the RCS System Description Document. This ensures compliance with 10CFR50 Appendix G.

This change resolves discrepancies between plant operating instructions and the UFSAR regarding the administratively controlled heatup and cooldown rate established for the RCS. The UFSAR is being changed to delete references to a specific heatup/cooldown rate and to state that the maximum rate will be in compliance with the design requirements. The heatup/cooldown rate utilized by the plant operating procedures (currently 60°F per hour) is still well within the design limits for the RCS and therefore does not impact the design or operating requirements specified by the vendor, or by the RCS PTLR developed by TVA.

The change to replace specific references to administrative/expected heatup and cooldown limits with a reference to the PTLR does not impact the analysis of any DBA and does not increase the probability of any UFSAR analyzed event. The PTLR limits are not derived from DBA analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Although the PTLR limits are not derived from any DBA, the PTLR limits are acceptance limits since they preclude operation in an unanalyzed condition. The design limit for the reactor heatup/cooldown rate specified in the PTLR and Westinghouse Design Specifications is 100°F per hour. Heatup and cooldown rates less than 100°F/hour are administratively established to provide operating margins that ensure the plant is operated within the design limits for the RCS. No accident is made more severe by this change in the UFSAR to reference the PTLR rather than specifying exact heatup/cooldown rates. All relevant design features which perform accident mitigation functions, including systems which operate to prevent contamination release, are available for the challenges that are possible during and after DBAs.

The change in the UFSAR to reference the PTLR for heatup/cooldown limits does not affect the integrity of any fission product barriers. The fuel, cladding, RCS pressure boundary and containment are not degraded by this change. Heatup and Cooldown will be controlled in the same manner as before such that no design limits or operational limits are challenged. Technical Specification 3.4.3 is concerned with two elements: assurance that the limits for heatup, cooldown, and inservice leak and hydrostatic testing are maintained and the limits on the rate of temperature change are maintained. Assurance of these limits are unaffected by the UFSAR change to reference the PTLR rather than specifying an exact heatup/cooldown rate. There are not any physical modifications being implemented in support of this change. Nor are any operating procedures affected. The existing operating procedures ensure that the design of the RCS is not impacted by the established heatup and cooldown rate. In conclusion the UFSAR change does not adversely impact nuclear safety and does not involve an unreviewed safety question.

SA-SE Number: WBPLMN-00-076-0

Implementation Date: 10/05/2000

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50797-A UFSAR Chg Pkg 1661	Centrifugal Charging Pumps (CCPs) Performance Data

Description and Safety Assessment:

EDC E-50797-A serves to evaluate and revise the appropriate design documents to reflect the as-found performance data for CCPs 1A-A and 1B-B.

During the WBN Unit 1 Cycle 3 refueling outage, the rotating element in CCP 1B was replaced. During post-modification testing of the pump, the pump developed a total head of 2196 feet at the test flow value of 537 gpm. Testing of CCP 1A per the same procedure showed that the 1A pump developed a total head of 2163 feet at the test flow rate of 536 gpm. These head values slightly exceed the value depicted in the maximum composite pump performance curve provided in System Description as the pumps approach the runout speed.

The potential result of the pumps operating above the maximum allowable pump curve is that during post-LOCA recirculation, the pumps could operate beyond their runout limit of 560 gpm assumed in the ECCS Analysis. However, the plant can utilize the system resistance to verify that the plant system setup is within ECCS analysis limits. The ECCS system resistance is within the required acceptance band as specified in the ECCS flow analysis. Using plant test data to model the ECCS, Westinghouse has determined that the maximum pump flow rate during post-LOCA recirculation would be less than 550 gpm, which is consistent with the ECCS analysis.

Therefore, it is determined that the test data indicating the potential for an increase in runout flow does not represent a condition adversely affecting the capability of the CCPs to perform their normal charging function or their post-accident function of safety injection following certain DBAs.

Although test data collected during the performance of system testing indicated that the CCPs may be operating at a slightly higher developed head value near the pump runout limits, potentially affecting the ECCS analysis, additional evaluation based upon plant data indicates that the pumps would operate at a flowrate within the acceptable limits of the ECCS analysis. It was determined that the pumps will operate within allowable flow and horsepower limits, and that the plant will remain within the limits evaluated in the UFSAR. This change does not represent an unreviewed safety question.

SA-SE Number: WBPLMN-00-077-0

Implementation Date: 01/19/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	UFSAR Change Package 1660 and Design Criteria WB-DC-40-70 R3	Chapter 15 Accident Analyses Revisions

Description and Safety Assessment:

Three revisions to the UFSAR accident analyses are made. The first is associated with assumptions in the Inadvertent ECCS at Power event. Additional conservatism is included in this analysis for the pressurizer filling analysis. The second is associated with the Main Steam line Break Core response non-LOCA model. For this mode, the RCS flowrate was corrected to be consistent with other non-LOCA analyses in the UFSAR. The third is associated with including the previously analyzed post-LOCA design basis hydrogen accumulation predictions into the UFSAR containment peak pressure analysis. The non-LOCA analyses demonstrated that the basic Fuel rod protection criteria were met.

The Inadvertent ECCS analysis predicts pressurizer filling prior to operator termination of the event but does not predict safety valve relief even with the PORVs blocked to maximize the challenge to the pressurizer safeties. Therefore, the potential for a loss of coolant as a result of the pressurizer safeties sticking open remains low. The analysis of record for the departure from nucleate boiling (DNB) event of inadvertent SI injection remains unchanged, however the overfill event is revised.

The steam line break event is not significantly altered by the flow rate assumption correction. Other than some minor UFSAR corrections, the analysis remains essentially unaffected.

The hydrogen accumulation analysis and the peak containment pressure analysis are both currently presented in the UFSAR. An evaluation of the hydrogen impact on the peak containment pressure is now quantified and presented in the UFSAR. The two separate analyses are now quantified concurrently for the peak containment pressure. The impact is small and the total containment pressure at the maximum peak with the hydrogen effect included remains below the design limit.

For all three of these events, the defense in depth is not reduced, the core limits are not impacted, and the containment design is not challenged. DNB response does not change for the non-LOCA transients (Inadvertent SI and main steam line break core response). For the containment LOCA response, the containment pressure remains below the design pressure which is well below the ultimate failure pressure of the containment. Margin to failure is therefore preserved consistent with the conservative nature of the assumptions used in the UFSAR safety analyses.

Therefore, the changes do not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-00-90-0

Implementation Date: 11/17/2000

Document Type:	***Affected Documents:***	***Title:***
Other	UFSAR Chg Pkg 1667 WB-DC-40-66 R2	Containment Penetration Figures UFSAR Section 6.2.4

Description and Safety Assessment:

PER 99-013164-000 was initiated to document the discrepancies in UFSAR Subsection 6.2.4. The drawings of the steel containment vessel (SCV) penetration and the Shield Building penetration shown in UFSAR Subsection 6.2.4 require updating. Details such as dimensions, weld details, construction notes have been deleted. These details are not necessary for the UFSAR Figures.

UFSAR Subsection 6.2.4 was corrected for typographical errors. The text which describes penetration Type IX was incorrect. Penetration Type IX is similar to penetration Type VII. The text which describes penetration Types XI and XII, was incorrect. Containment sump water flows out of the containment through two Type XII penetrations which are shown in Figure 6.2.4-9. Other minor grammatical errors were corrected as well.

The following UFSAR Figures were updated to remove excess details such as dimensions, weld details, and construction notes:

Figure 6.2.4-3
Figure 6.2.4-4
Figure 6.2.4-5
Figure 6.2.4-6
Figure 6.2.4-7
Figure 6.2.4-8
Figure 6.2.4-9
Figure 6.2.4-10
Figure 6.2.4-17A
Figure 6.2.4-17E
Figure 6.2.4-23

The following UFSAR Figures were updated to remove excess details such as dimensions, weld details, and construction notes in addition to what is indicated below:

Figure 6.2.4-1 Figure 6.2.4-2	The bellows shown outside the shield wall was removed to be current with the "as constructed" drawing. The bellows used in the original design has been replaced with a boot seal that is rated for fire and as an Auxiliary Building secondary containment enclosure (ABSCE) boundary. Use of the boot seal is consistent with TVA design criteria requirements and is consistent with the existing UFSAR text that describes these penetrations.
Figure 6.2.4-11	Excess details were not shown. The figure has been redrawn for legibility.
Figure 6.2.4-12	The dogging beam and three leg lifting bridle details were removed to be consistent with the "as constructed" drawing.
Figure 6.2.4-13	The previous figure in the UFSAR was based on the "as designed" drawing. The figure has been updated to agree with the "as constructed" drawing.
Figure 6.2.4-17	The figure has been updated to provide a general description of electrical penetrations. The typical module length shown has been increased to be more consistent with electrical penetration drawings.
Figure 6.2.4-17B	Excess details were not shown. The figure has been redrawn for legibility.
Figure 6.2.4-17C	The weld detail was removed. Notes to show the containment penetration nozzle and the distance to the center line of the Reactor Building were added.

Design Criteria WB-DC-40-66 Figures are based on the UFSAR Figures. The Design Criteria Figures were updated to be consistent with the UFSAR.
Figure 9-la is based on UFSAR Figures 6.2.4-1 and 6.2.4-3.
Figure 9-1b is based on UFSAR Figures 6.2.4-4 and 6.2.4-5.
Figure 9-1c is based on UFSAR Figure 6.2.4-8.
Figure 9-1d is based on UFSAR Figure 6.2.4-9.
Figure 9-1e is based on UFSAR Figure 6.2.4-16.

Containment penetrations exist to limit radiological releases following DBEs. None of the changes in the UFSAR affect the ability of the penetrations to perform this function. This is an editorial change that resolves discrepancies in the UFSAR concerning the configuration of the penetrations. The figures have been updated to show the "as constructed" configuration of the penetrations. Removal of the depiction of the bellows shown outside the shield wall for Figures 6.2.4-1 and 6.2.4-2 does not impact ABSCE boundary integrity. The bellows have been replaced with a boot seal that is rated for fire and as an ABSCE boundary. Use of the boot seal is consistent with the requirements of the design criteria and is consistent with the existing UFSAR text that describes these penetrations. There are no new credible failure modes added or changed as a result of these changes to the UFSAR. Therefore, the probability of occurrence or the consequence of any accidents or equipment malfunction is not increased. In addition, the UFSAR changes do not affect the consequences of any previously evaluated accidents because the ability to maintain the integrity of the Steel Containment Vessel and Shield Building penetrations has not been adversely impacted. The UFSAR changes to correct the depiction of the penetrations do not have the potential to create a new accident or equipment malfunction; nor do the editorial changes to the UFSAR affect the margin of safety defined in the Technical Specification Bases. Consequently, this change does not constitute a unreviewed safety question.

SA-SE Number: WBPLMN-00-093-0

Implementation Date: 01/24/2001

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50506-A DCN D-50550-A	Fire Protection Report (FPR) Revision 14

Description and Safety Assessment:

The FPR is being revised due to DCNs D-50506-A and D-50550-A. DCN D-50506-A abandoned the PD Pump in place. Before this change, an Appendix R fire in certain areas of the plant had the possibility of causing a fire induced spurious start of the PD Pump. To prevent this, a local manual action was required to be performed within 15 minutes.

DCN D-50506-A eliminates the possibility of the PD Pump spuriously starting and therefore eliminates the need for the manual actuation. No credible failure modes were found for the abandonment of the PD Pump. The pump is not required to operate to safely shutdown or maintain the plant in a safe condition, nor to mitigate the consequences of a DBE.

A second design change, DCN D-50550-A, added a ventilation opening in the wall between rooms 772.0-A1 (Fire Area 32) and 772.0-A8 (Fire Area 39). The opening is provided with an approved fire damper to maintain the required fire barrier rating of the wall. The safety evaluation review for D-50550-A addresses the adequacy of this change and will not be repeated in this evaluation, except for the following clarification. DCN D-50550-A evaluated the fire damper in the closed position until DCN D-50306-A could be completed. DCN D-50306-A placed the fire damper in the open position. During normal operation the fire damper is held open by a fusible link. The fire hazards analysis for the two rooms always assumes the same as with the damper in the closed position. The FPR requires a revision to document these changes and since the FPR is the licensing basis for WBN compliance to Appendix R, this change to the FPR requires a safety evaluation.

The current Appendix R Fire Safe Shutdown Analysis (FSSD) requires the PD Pump to not start during a fire. It was conservatively assumed that a fire could cause damage to circuits associated with the PD Pump and result in an unwanted spurious pump start. To prevent this unwanted spurious pump start, a local manual operator action had to be performed within 15 minutes. This change eliminates the possibility of a fire induced spurious pump start which satisfies the requirement of the FSSD analysis and also eliminates a manual action. In addition, this change also eliminates the PD pump as a potential ignition source since the pump is electrically isolated. There are no additional consequences due to a malfunction of equipment since the PD pump is electrically and mechanically isolated. The radiological consequences of the accidents evaluated in the UFSAR are not increased since the PD pump was not required for achieving and maintaining safe shutdown in the event of a fire or accident. Therefore, this change to the FPR does not involve a unreviewed safety question.

ENCLOSURE 2

INADVERTENTLY OMITTED SAFETY EVALUATIONS
10 CFR 50.59 SUMMARY REPORT
DATED OCTOBER 18, 1999

SA-SE Number: WBOTSS-98-105-0

Implementation Date:01/30/1999

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-98-012-006	Furmanite Injection for Temporary Leak Repair

Description and Safety Assessment:

This TACF disables 1-RTV-6-374A, 1-PI-6-18 Root Valve, in the closed position by Furmanite injection to stop an external body-to-bonnet steam leak. 1-PI-6-18 is the non-quality related locally mounted Feedwater Heater Al Drain Line Pressure indicator. Work Order (WO) 98-010622-001, using Furmanite Procedure C-N-98236, will install a Furmanite fitting immediately upstream of the valve seat and a maximum amount of 4 sticks of compound will be injected to seal the body-to-bonnet joint leak to atmosphere.

This TACF changes the valve alignment by closing and disabling this normally open root valve. The position of this root valve is depicted in UFSAR Figure 10.4-27, TVA Drawing 1-47W805-1. However, since this is simply a non-quality related root valve to a local pressure indicator, and not a process flow valve, this change is not considered to significantly affect the description of the system as presented in the UFSAR.

The only type of accident that this temporary alteration could possibly be associated with is "Minor Secondary System Pipe Breaks." However, the purpose of this TACF is to stop a steam leak in a ½-inch root valve to a non-quality related information-only pressure indicator. Therefore, this activity should help minimize the probability of a secondary system pipe break by eliminating flow in this normal dead leg sense line. This ½-inch secondary pipe break accident is bounded by the analysis of major secondary pipe breaks and does not require further analysis. This does not affect technical specification equipment, therefore, this TACF does not constitute an unreviewed safety question.

SA-SE Number: WBOTSS-98-114-1

Implementation Date: 12/01/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	TACF 1-98-14-212, Rev. 1	Removal of Unit 2 480V Shutdown Board Alternate Feeder Breaker

Description and Safety Assessment:

This safety evaluation is for Revision 1 of TACF 1-98-14-212. Revision 0 of this TACF removed the Unit 2, 480 Volt Shutdown Board Alternate Feeder Breakers (4 total). These Westinghouse DS-632 breakers were shipped to Sequoyah Nuclear Plant so that parts would be available for use in their Westinghouse DS-532 breakers which were thought to have been damaged. These materials were not available from the vendor nor other TVA facilities in a time frame which supported Sequoyah's Unit 1 Cycle 9 refueling outage schedule. It is was the intent of Sequoyah to refurbish Watts Bar's breakers with qualified replacement parts procured from Westinghouse by the end of September 1998 if required. Once Sequoyah checked out their breakers, it was determined that no parts from Watts Bar's breakers were needed. The breakers were never removed from their shipping crates at Sequoyah and have since been returned to Watts Bar. During normal plant operation the 480 Volt Shutdown Boards are supplied via their Normal Feeder Breakers and the Alternate Feeder Breakers are in the open position. The only time the Alternate Feeder Breakers are required to be closed is when maintenance is being performed on the 6.9kV Shutdown Board Normal Feeder Breaker, the associated 6.9kV/480V transformer, the 480 Volt Shutdown Board Normal Feeder Breaker, or the associated cables/bus. Watts Bar has one spare Westinghouse DS-632 breaker which can be utilized at any time (for the duration of this TACF) as an alternate feeder breaker should there be a need. The Unit 1, 480 Volt Shutdown Board Alternate Feeder Breakers could also be utilized as a contingency.

There are no DBAs that may be affected by this temporary change because no credit has been taken for the alternate feeds in their evaluations. The boards remain energized before, during, and after their removal. There is no automatic transfer between the normal supply and the alternate supply breakers. Their purpose is to facilitate maintenance and to provide a quick path to switch in a spare transformer in the event of a primary transformer failure or a 6900V Shutdown Board feeder breaker failure. The credible failure modes of the temporary change is the same as before the change.

This temporary change removes the Unit 2 (required for Unit 1), 480V Shutdown Board alternate feeder breakers (four total) from service while they are being used at Sequoyah Nuclear Plant. During normal plant operation, the 480V Shutdown Boards are supplied via their normal feeder breakers. The alternate feeder breakers are used only for maintenance purposes. There is no automatic transfer capability between the normal and alternate breakers. If needed a spare breaker could be moved to the alternate breaker compartment and used as the alternate breaker. There are no DBAs that may be affected by this temporary change because no credit has been taken for the alternate feeds in their evaluations. There is no change in failure modes before, during, and after removal of the 480V Shutdown Boards alternate feeder beakers.

SA-SE Number: WBOCEM-99-001-0

Implementation Date: 02/10/1999

Document Type:	***Affected Documents:***	***Title:***
Procedure	Offsite Dose Calculation Manual Cancellation of PAI-4.01	Offsite Dose Calculation Manual, Revision 0

Description and Safety Assessment:

Plant Administrative Instruction (PAI)-4.01 is being canceled and the ODCM is being re-issued as a stand-alone manual. Attachment 1 to PAI-4.01 was relocated into this manual with the following changes: the Source Notes were relocated to Section 11.0 of the new ODCM; the cover sheets for each of the ODCM sections were deleted; the table of contents was revised to reflect the changes in page numbering; a new Appendix C was added which contains the administrative process to be used to revise the ODCM.

This ODCM change is acceptable from a nuclear safety perspective. The revision does not affect any calculation methodology described in the ODCM; therefore, it does not affect the accuracy or reliability of effluent, dose, or setpoint calculations. The change also does not affect the way in which the effluent monitoring system is operated, it only provides an administrative process to perform future revisions to the manual. Since no dose or setpoint determinations are affected, and no equipment operational requirements are changed, this revision will not lessen the level of effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, or Appendix I to 10 CFR 50. Therefore, there is no unreviewed safety question involved.

SA-SE Number: TACF-1-99-002-261

Implementation Date: 04/01/1999

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-99-002-261	Temporary Indications During Plant Computer Replacement

Description and Safety Assessment:

Temporary alteration TACF 1-99-002-261 will provide Main Control Room (MCR) indication of selected plant parameters during the Refueling Outage 2 plant computer replacement under DCN M-39911-A. The temporary monitoring equipment is not safety-related, the data is not used to perform any safety-related or safe shutdown or accident mitigation functions, nor is the data used to perform functions essential to the health and safety of the public. Though this temporary monitoring equipment may alert the operator that an abnormal condition exists, operators cannot procedurally take inappropriate safety-related action based solely an the information provided by this TACF. This temporary alteration introduces no increased probability of an accident or malfunction of equipment important to safety, or create the possibility for an accident or malfunction of a type different than any evaluated previously in the UFSAR. This-alteration introduces no increased radiological consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

The credible failure modes associated with TACF 1-99-002-261 will be no different than those associated with the existing P2500 Plant Computer, or the new Integrated Computer System (ICS) computer system being installed under the scope of DCN M39911-A. The credible failure modes of (1) Loss of indication, or loss of one or more data sources; and (2) Display of incorrect information, remain the same as with the existing equipment. Neither the P2500 Plant Computer, nor the ICS computer are defined as Safety-Related, and are not required to meet the single failure criterion or be qualified to IEEE criteria for Class 1E equipment. The temporary monitoring equipment being installed for TACF 1-99-002-261 does not need to be qualified to a higher safety class than the equipment it is temporarily replacing. The temperature elements and level transmitter instrument loops, whose readouts will be displayed on a MCR computer temporarily dedicated for this TACF, are not safety-related instrument loops, and do not perform any technical specification or compliance functions.

The implementation and testing phase of the TACF may be performed during plant operating Modes 1 through 6, and will involve panels I-R-104, 1-R-155 and I-R-156 located in the Computer Room, and panels I-R-111. I-R-112, and 1-R-177 located in the Unit 1 Auxiliary Instrument room. However, there are no safety-related / IE powered components mounted in any of these panels. Also the installation of the temporary jumpers from the panels to the fluke dataloggers will not alter the loop impedance values, and thus will have no impact on the readout or performance of the existing P2500 plant computer as the connections are made.

This temporary alteration does not create any new accidents of any type, nor does it create any new credible failure modes of any type that would represent an unreviewed safety question.

Therefore the temporary monitoring equipment to be installed for TACF-1-99-002-261 does not result in an unreviewed safety question.

SA-SE Number: WBOTSS-99-111-0

Implementation Date: 09/29/1999

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-99-15-002	Furmanite Injection for Leakage Control of Valves

Description and Safety Assessment:

TACF 1-99-15-002 documents temporary leak sealing activities performed on two valves: 1) 1-BYV-002-0779, MFP IB Suction Valve (1-FCV-2-224) Bypass valve; temporary on-line non-code repair of 1 inch 8000 carbon steel, ASTM A-105, Hancock Model 5500W-1 globe valve body-to-bonnet leak using Furmanite, procedure C-N-99215. Repair consists of 'Killing' the valve in the normally closed position by drilling the valve body and injecting Furmanite compound upstream of the valve seat and, if necessary, downstream of the valve seat and 2) 1-RTV-006-1628A, moisture separator reheater (MSR) B2 Low Pressure (LP) Drain Tank Level Gage (1-LG-6-81) upper root valve, temporary on-line non-code repair of a 1-inch Hancock Model 5500W-1 socket welded carbon steel bolted bonnet globe valve, rated for 600 pounds at 910°F, body-to-bonnet steam leak to atmosphere using Furmanite procedure C-N-99210. This valve is the normally open upper root valve to MSR 82 LP Drain Tank Level Gage 1-LG-6-84. Repair consists of cutting off the valve handwheel with the valve left in the normally open position and encapsulating the valve in an enclosure.

Disabling these valves, 1-BYV-2-779 and 1-RTV-6-1628A, in their normal positions, closed for 1-BYV-2-779 and open for 1-RTV-6-1628A, as depicted on the associated flow prints 1-47W804-1 (UFSAR Figure 10.4-7) and 1-47W805-3 (no associated UFSAR figure) respectively, also agrees with the normal position for the 779 valve as described in UFSAR Figure 10.4-7.

MSR 82 LP drain tank level gage 1-LG-6-81 was formerly a glass type gage. It was replaced during Refueling Outage 2 with a magnetically coupled gage which consists of a titanium float in a stainless steel tube. The float is magnetically coupled to an external indicator. This gage is for indication only. High and low alarm only level switches are also provided for this tank and incorporate dedicated isolation / root valves for each switch. The purpose of 1-RTV-6-1628A is to provide isolation capability for corrective maintenance. Given the robust design of this new level gage (no glass) and its age, the likelihood of a downstream leak / the need to isolate it on-line is not considered credible.

Normal operation and shutdown of the main and standby feedwater pumps are not affected by this TACF. Disabling 1-BYV-2-779 in the normally closed position does have an affect on System Operating Instruction (SOI)-20. 1, condensate and feedwater system. Specifically, this valve is used during the start of main feed pump 1B to pressurize and warm the pump casing prior to opening the associated suction isolation valve, 1-FCV-2-224. This SOI will be revised to specify an alternate means to accomplish this task in the unlikely event the 1B MFP is removed then returned to service prior to the replacement of this valve. The Furmanite repair consists of 'Killing' the valve in the normally closed position by drilling the valve body, installing a Furmanite adapter(s), and injecting Furmanite compound upstream of the valve seat and, if necessary, downstream of the valve seat. The materials, maximum allowable injection pressures, and maximum volume of compound to be used which are specified in the approved Furmanite procedure are suitable for the temperature and pressure of this application. These safeguards ensure that no extraneous Furmanite compound is carried forward into the feedwater system.

There are no DBAs that could be associated with the activity of killing MFP 1B's suction isolation bypass valve in the closed position.

The affect of this Furmanite injection/valve kill is that the valve external body to bonnet leak is temporarily repaired and that the valve cannot be opened. The potential credible failure modes of this 1-inch non-quality-related non-seismic bypass valve are external leakage, internal seat leakage, failure to open, and failure to close. By tailing the valve in the closed position, this activity effectively eliminates the other three failure modes: external leakage, internal seat leakage, and failure to close. This activity does not introduce any new credible failure modes.