ENCLOSURE

10 CFR 50.59 SUMMARY REPORT

Watts Bar Nuclear Plant
Safety Assessment and Safety Evaluation Summaries
Acronym List

ABMT	Auxiliary Boration Makeup Tank
ABSCE	Auxiliary Building Secondary Containment Enclosure
ACBP	Auxiliary Charging Booster Pump
ACP	Auxiliary Charging Pump
ACI	American Concrete Institute
ACU	Air Cleanup Unit
AFD	Axial Flux Difference
AFFFF	Aqueous Film Forming Foam
AHU	Air Handling Unit
AMSAC	Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC)
AOI	Abnormal Operating Procedure
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
AUO	Assistant Unit Operator
AVCS	Annulus Vacuum Control System
BE	Best Estimate
BFN	Browns Ferry Nuclear Plant
CAM	Continuous Air Monitor
CAP	Corrective Action Plan
CCD	Configuration Control Drawing
CCRS	Cable and Conduit Routing System
CCS	Component Cooling System
CCW	Condenser Circulating Water
CDCT	Cask Decontamination Collector Tank
CDWE	Condensate Demineralizer Waste Evaporator
COLR	Core Operating Limits Report
COMS	Cold Overpressure Mitigation System
CPDS	Condensate Polishing Demineralizer System
CPU	Central Processing Unit
CRDM	Control Rod Drive Mechanism
CRDMCs	CRDM Coolers
CREATCS	Control Room Emergency Air Temperature Control System
CRT	Cathode Ray Tube
CST	Condensate Storage Tank
CTB	Cooling Tower Blowdown
CVACU	Containment Air Vent Cleanup Unit
CVCS	Chemical Volume and Control System
DAW	Dry Active Radioactive Waste
DBA	Design Basis Accident
DBD	Design Basis Document
DBVP	Design Baseline and Verification Program
DCN	Design Change Notice
DD	Drawing Deviation

DWS	Demineralized Water System
ECCS	Emergency Core Cooling System
EDC	Engineering Document Change
EFPD	Effective Full Power Day
EHC	Electrohydraulic Control
EOC	End of Cycle
EOP	Emergency Operating Procedure
EQB	Engineering and Quality Building
ERCW	Essential Raw Cooling Water
ERFDS	Emergency Response Facility Data System
FDCT	Floor Drain Collector Tank
FHA	Fuel Handling Area
FHE	Fuel Handling Equipment
FHI	Fuel Handling Instruction
FHS	Fuel Handling System
FHSS	Fuel Handling and Storage System
FMBMS	Flood Mode Boration Makeup System
FPR	Fire Protection Report
GDC	General Design Criteria
GSC	Gland Steam Condenser
HCLC	High Crud, Low Conductivity
HCT	High Crud Tank
HD&V	Heater Drains & Vents
ICS	Industrial Control Systems
ICS	Integrated Computer System
IDI	Integrated Design Inspection
IFBA	Integral Fuel Burnable Absorber
LBLOCA	Large Break Loss of Coolant Accident
LCC	Lower Compartment Coolers
LCD	Liquid Crystal Display
LCHC	Low Crud, High Conductivity
LCO	Limiting Condition for Operation
LLRT	Local Leak Rate Test
LOCA	Loss of Coolant Accident
LONF	Loss of Normal Feedwater
LOOP	Loss of Offsite Power
LPMS	Loose Parts Monitoring System
LRPS	Liquid Radwaste Processing System
LWT	Legal Weight Truck
MFPT	Main Feedwater Pump Turbine
MCR	Main Control Room
MCRHZ	Main Control Room Habitability Zone
MDB	Modifications Building
MELB	Moderate Energy Line Break
MERITS	Methodically Engineered, Restructured and Improved, Technical Specifications

MFWP	Main Feedwater Pump
MMDP	Maintenance & Modification Department Procedure
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MTC	Moderator Temperature Coefficient
NADP	Nuclear Assurance Department Procedure
NCIG	Nuclear Construction Issues Group
NFRT	Neutron Flux Negative Rate Trip
NFPA	National Fire Protection Code
NFSR	New Fuel Storage Racks
NHI	Net Heat Input
NIS	Nuclear Instrumentation System
NRWP	Nonreclaimable Waste Pump
NRWT	Non-Reclaimable Waste Tank
NSAL	Nuclear Safety Advisory Letter
NSVR	North Steam Valve Room
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
OT	Over Temperature
PORV	Power Operated Relief Valve
PCT	Peak Clad Temperature
PER	Problem Evaluation Report
PRT	Pressurizer Relief Tank
PS	Process Specification
RBF&ED	Reactor Building Floor & Equipment Drain System
RCCA	Rod Cluster Control Assembly
RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RCW	Raw Cooling Water
RHR	Residual Heat Removal
RSO	River System Operations
RSW	Raw Service Water
RTD	Resistance Temperature Detector
RWST	Refueling Water Storage Tank
SCCW	Supplemental Condenser Circulating Water
SCV	Steel Containment Vessel
SDD	System Description Document
SFP	Spent Fuel Pool
SFPC	Spent Fuel Pool Cooling
SFPCCS	Spent Fuel Pool Cooling and Cleanup System
SFSR	Spent Fuel Storage Racks
SG	Steam Generator
SLB	Steam Line Break
SLP	Safe Load Path

SOI	System Operating Instruction
SPDS	Safety Parameter Display System
SQN	Sequoyah Nuclear Plant
SR	Surveillance Requirement
SSE	Safe Shutdown Earthquake
SSP	Site Standard Practice
SSPS	Solid State Protection System
TACF	Temporary Alteration Control Form
TAO	Temporary Alteration Orders
TGCPS	Turbine Generator Control and Protection System
TIR	Testing and Inspection Requirements
TMD	Transient Mass Distribution
TOPS	Turbine Overspeed Protection System
TPBAR	Tritium Producing Burnable Absorbers
TPS	Transmission Power Supply
TDCT	Tritiated Drain Collection Tank
TDMFP	Turbine Driven Main Feedwater Pump
TRM	Technical Requirements Manual
TSC	Technical Support Center
TSR	Technical Surveillance Requirement
TSS	Transmission System Studies
TSTF	Technical Specification Traveler Form
TVAN	Tennessee Valley Authority Nuclear
UCC	Upper Compartment Coolers
UFSAR	Updated Final Safety Analysis Report
UL	Underwriters Laboratories
UF	Under Frequency
UV	Under Voltage
USQ	Unreviewed Safety Question
WABA	Wet Annular Burnable Absorbers
WBF	Watts Bar Fossil Plant
WBN	Watts Bar Nuclear Plant
WDS	Waste Disposal System
WGDT	Waste Gas Decay Tank
WSD	Working Stress Design

Watts Bar Nuclear Plant
Safety Assessment and Safety Evaluation Summaries

SA-SE Number: Fire Protection Report Revision 11

Implementation Date: 02/26/1999

Document Type:	***Affected Documents:***	***Title:***
Fire Protection Report	Fire Protection Report Revision 11	Corrective action for WBPER980227, WBPER980971, and WBPER981057.

Description and Safety Assessments:

The revised sections of the Fire Protection Report are as follows:

Part II, Section 5:
Added list of inaccessible areas previously located in SSP-12.15.

Part II, Section 5:
Revised definition of Authority Having Jurisdiction due pending cancellation of STD 12.15.

Part II, Section 7.4:
Change title of Fire Protection Manager to Fire Protection Supervisor.

Part II, Section 8. 1 a:
Due to the pending implementation of standardized procedures, revised the wording to delete reference to specific procedure.

Part II, Section 9.3.b.2.g:
Provided allowance so that Incident Commander (IC) did not have to attend every drill during assigned shift. The IC's purpose is plant knowledge which is gained on the other assigned duties that qualified the IC to hold this position.

Part II, Section 14:
Revised page numbers on Table of Contents.

Part II, Section 14.2.1:
added subsection c. to cover planned situations when fire pump availability is reduced to only 2 electric fire pumps as will be the case in the Unit 1 Cycle 2 (U1 C2) outage.

This action will require the use of a backup fire pump of at least equal capacity of an electric fire pump. This ensures that an adequate water supply is available for the most demanding fire suppression activities in safety-related areas.

Part II, Section 14.2:
Revised table to add backup pump criteria.

Part II, Section 14.1:
Revised reference to P2500 computer to reflect generic name due to replacement of P2500.

Part II, Section 14.5:
Clarified 14.5.2 to address masking that cannot be cleared by jumpering requires an evaluation within the 8 hour time frame previously established to determine the effects of the masking on equipment operation and the implementation of appropriate compensatory actions if the masking condition cannot be cleared. The credible failure mode related to this change is:

> The evaluation of the extent of the masking condition that jumpering will not clear is inaccurate. In such a case the detection and, if applicable, the associated suppressing will not operate. With the compartmentation and other detection/suppression provided, the chance of fire spread that will affect both safe shutdown paths is not a reasonable assumption.

Watts Bar Nuclear Plant
Safety Assessment and Safety Evaluation Summaries

SA-SE Number: Fire Protection Report Revision 11 ***Implementation Date:*** 02/26/1999

Part II, Section 14.7.c, TIR:
Clarified that this is only a visual inspection.

Part II, Section 14.2.1 bases:
Revised the bases to cover the use of a backup pump when there are only 2 electric pumps available.

Part II, Section 14.2.1, TIR bases:
Clarified where the voltage is to be check when meeting this TIR requirement.

Part II, Section 14.8, bases:
Revised wording from equal to or greater than 20% to approximately 20% to prevent having to inspect some barriers twice. CFM - If too few barriers are inspected each year then at the end of the period for total inspection, a large number of barriers may have to be inspected. If due to resource constraints this last group could not be inspected then the uninspected ones could be declared inoperable and appropriate compensatory measures established in accordance with the FPR.

Part II, Section 14. 10 TIR & bases:
Added TIR 14. 10.m to verify that the Appendix R transfer switches function as intended by performing a continuity check.

Part II Table 14.8.1:
Revised the rating to reflect its physical characteristics that caused it to be evaluated since a UL assembly could not be purchased. The size and operational characteristics were such that a UL assembly was not available and so an evaluation was required to determine its equivalency.

Part II, Section 14, Table 14.8.2:
Removed dampers that are no longer in a regulatory fire barrier, added one damper inadvertently left out of the table and changed the system number for the dampers in the Diesel Generator Building to System 30.

Part II Section 14, Table 14. 10:
Revised to correct nomenclature of equipment listed.

Part II, Table 14.6: clarified note to not restrict when hose is to be removed from staged location.

Part III, Section 3, Table 3.2:
Revised to correct nomenclature of equipment listed.

Part VI:
Revised to ensure consistency between this part and the tables in Part II, Section 14.

The changes made by Revision 11 are expected to ensure the systems continue to operate within their design parameters prior to this revision. Thus the equipment operation as described in the FSAR will not be affected. This change is safe from a nuclear safety stand point and satisfies the FSAR and T/S. Therefore these changes do not constitute an unreviewed safety question.

SA-SE Number: FSAR Change Package 1570

Implementation Date: 12/17/1998

Document Type:	***Affected Documents:***	***Title:***
Procedure Change FSAR Change Package	Procedure No. 0-SI-79-1 R4 and PAI-708 R2 FSAR Change Package 1570	FSAR and Procedural Changes for Fuel Storage with Enrichments Greater than 4.3 wt%

Description and Safety Assessments:

This change is to allow new fuel with enrichments greater than 4.3 wt% to be stored in the new fuel vault and still meet the requirements of the Westinghouse Commercial Nuclear Fuel Division's "Criticality Analysis of the Watts Bar Units 1 & 2 Fresh Fuel Racks." Surveillance Instruction (SI) 0-SI-79-1, "Verification of Fuel Storage Configurations," and Plant Administrative Instruction (PAI) 7.08, "Storage of Material in the Spent Fuel Pool, Cask Pit & New Fuel Vault," both put administrative controls on ten fuel storage locations in accordance Figure 2 of the Criticality Analysis as well as Technical Specification Figure 4.3-2, which was added by Amendment 15 of the Technical Specifications.

FSAR Section 4.3.2-7 is being revised in accordance with FSAR Change Package 1570. This change allows the placement of ten insert plates into the ten restricted cells to provide a physical barrier to prevent inadvertent placement of fuel into any of the restricted locations. Also, these insert plates will be used to store unirradiated non-fuel bearing inserts which are thimble plugging assemblies, rod cluster control assemblies, burnable poison rod assemblies and tritium producing burnable absorber assemblies.

The accidents analyzed in the "Criticality Analysis of the Watts Bar Units 1 & 2 Fresh Fuel Racks" are the introduction of water into the fresh fuel rack area (which is the worst case accident scenario) and the other postulated accidents are dropping a fuel assembly between the rack and the concrete wall or dropping a fuel assembly on top of the rack. For the latter two postulated accidents, the double contingency principle of ANSI N16.1-1975 is applied. Therefore, for these accident conditions, the absence of a moderator in the fresh fuel storage racks can be assumed as a realistic initial condition since assuming its presence would be a second unlikely event. For the introduction of water accident the center to center spacing of 21 inches is sufficient to ensure that the K_{eff} does not exceed 0.98 with fuel of 5 wt% in optimum moderation conditions from an aqueous foam or mist. The structural design of the racks preclude the insertion of fuel in other than the designed locations.

These proposed procedural changes and the placement of the inserts into the restricted cells for storage of fuel enrichments greater than 4.3wt% as required by the proposed FSAR change and the Technical Specification amendment is not an unreviewed safety question. This conclusion is based on the evaluation that there will be no increase in probability of a fuel handling accident, no increase in the consequences of a criticality event, or an increase of the offsite dose. Since there were no modifications due to these changes, there is no increase in the probability of occurrence of a malfunction of equipment important to safety or equipment important to safety previously evaluated in the FSAR. Currently, insert plates in the new fuel vault are in use with unirradiated burnable absorber rod assemblies inserted into them. The proposed placement of the insert plates into the ten restricted cells does not create an accident of a different type than any evaluated previously in the FSAR or create the possibility for a different type of malfunction that was previously evaluated in the FSAR. Also, the proposed administrative controls of the procedures and the placement of the insert plates into the restricted cells as designated in Technical Specification Figure 4.3-2 ensure that the margin of safety described in the criticality analysis is not reduced.

SA-SE Number: General Engineering Specification G29 Part B, Section 1, PS 4.M.4.4, Appendix F

Implementation Date: 03/30/1999

Document Type:	***Affected Documents:***	***Title:***
General Engineering Specification (G-Spec) Addendum	TVA General Engineering Specification G29, Part B, Section 1, Process Specification (PS) 4.M.4.4, Appendix F	Consolidation of Fasteners for Inventory Purposes

Description and Safety Assessments:

This activity allows the grouping of fasteners for inventory reduction. They are grouped together by composition and mechanical properties. Thus allowing their substitution in joints where the same type properties are needed. This revision limits partial replacement of fasteners on a joint, thus avoiding joints with mixed pedigrees.

The consolidation process allows the fasteners to be grouped by their mechanical and chemical properties. This process is not applicable to safety-related fasteners except for grouping only and there will be no substitution of safety-related fasteners. Safety-related applications will follow the applicable ASME code. For other activities, fasteners included in this evaluation will be equivalent to the existing fasteners in stress allowables and service requirements.

Considering the preceding and the fact that the consolidation effort only groups fasteners together for inventory reduction, safety will not be compromised. Therefore, this activity is acceptable from a nuclear safety standpoint and the probability of an equipment failure that has already analyzed will not be increased.

SA-SE Number: General Engineering Specification 55, Revision 8

Implementation Date: 05/15/1997

Document Type:	***Affected Documents:***	***Title:***
General Engineering Specification	TVA General Engineering Specification G55, Technical and Programmatic Requirements for the Protective Coating Program	Clarification and Addition of Technical Requirements

Description and Safety Assessments:

The revision to G55 incorporates all outstanding revision notices adding coating systems for the TVA nuclear sites, clarifies technical requirements, adds technical requirements dictated from ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE for additional inspection requirements, and includes administrative changes, including organizational changes.

The repair system will not alter the existing coating system. It has been design basis accident tested to assure that it will not lead to coating failure. This system will come in contact with the existing systems that have already been analyzed for possible failure. The area that is usually repaired is less than one square foot per 100 square feet. This amount will not significantly contribute to the existing analysis. Since these repair systems have been tested, they can withstand accidents. The repair systems will not create any new accidents or malfunctions because they will be applied by certified applicators and monitored b y qualifying agents. This assures that these repairs will be applied only in those areas where they were intended. This activity is acceptable because it does not alter or challenge the existing design basis accidents, nor does it alter the existing critical safety components or structures. Therefore, this specifications is not considered an unreviewed safety questions.

SA-SE Number: General Engineering Specification 55, Revision 9

Implementation Date: 03/30/1999

Document Type:	***Affected Documents:***	***Title:***
General Engineering Specification	TVA General Engineering Specification G55, Technical and Programmatic Requirements for the Protective Coating Program	Clarification of Instructions and Incorporation of Change Notices

Description and Safety Assessments:

G55 provides the technical and programmatic requirements for the protective coating program for TVA nuclear plants. This revision primarily clarifies some instructions and incorporates change notices that have been issued previously. The existing requirements are not to be altered as a result of this revision. The coating repair system are design basis accident qualified for use in primary containment. Therefore this activity is acceptable for nuclear safety.

The repair system will not alter the existing coating system. It has been design basis accident tested to assure that it will not lead to coating failure. This system will come in contact with the existing systems that have already been analyzed for possible failure. The area that is usually repaired is less than one square foot per 100 square feet. This amount will not significantly contribute to the existing analysis. Since these repair systems have been tested, they can withstand accidents. The repair systems will not create any new accidents or malfunctions because they will be applied by certified applicators and monitored by qualifying agents. This assures that these repairs will be applied only in those areas where they were intended. This activity is acceptable because it does not alter or challenge the existing design basis accidents, nor does it alter the existing critical safety components or structures. Therefore, this specifications is not considered an unreviewed safety questions.

SA-SE Number: Technical Instruction (TI) 100.006, Revision 2

Implementation Date: 06/16/1999

Document Type:	***Affected Documents:***	***Title:***
Technical Instruction (TI)	TI-100.006, Inservice Testing Program	Revisions to Incorporate Problem Evaluation Report (PER) Corrective Actions

Description and Safety Assessments:

This safety evaluation addressed Revision 2 to the Inservice Testing Program (IST), TI-100.006. The changes made by Revision 2 are part of the corrective actions identified in PER WBPER971242 which was written during preparation of the Basis Document for the Inservice Testing Program, TI-100.011, and documented conditions in which components were either inappropriately excluded or inappropriately included in the WBN IST program. Revision 2 to TI-100.006 removes the relief valve that was identified as being inappropriately included in the WBN Inservice Testing Program.

The valve being deleted is 1-RFV-62-688-S. This valve provides protection to the volume control tank during normal operation and non-accident related transients. The valve is not identified as active in either System Description N3-62-4001 or in FSAR Table 3.9-17 or Table 3.9-25. As described in System Description N3-62-4001, the valve relieves overpressure conditions in the VCT associated with normal operation and operational transients of the charging and letdown portion of the CVCS system. The VCT isolates from the charging pump suction and from the letdown system during accident conditions and is not required to function to mitigate the consequences of an accident or to achieve or maintain the cold shutdown conditions. No design basis accidents are associated with this valve.

Since 1-RFV-62-688-S is not required to function for accident mitigation or to achieve or maintain the cold shutdown, no creditable failure modes are identified for the deletion of this valve from the IST Program.

The proposed change is in compliance with the requirements of ASME Section XI in that the valve being removed from the ASME Section XI IST program is a valve that does not meet the requirements for inclusion in the IST Program. The valve is not required to function to mitigate the consequences of an accident or to achieve or maintain the cold shutdown condition. The deletion of this valve has been incorporated in FSAR Table 3.9-26 of the WBN FSAR by FSAR Change Package 1584 S01. Testing of this valve is not discussed in the WBN FSAR and is not part of the basis for the NRC's acceptance of the FSAR as documented in the SER. This change is consistent with the design bases documents in that 1-RFV-62-688-S is not described or listed as an active valve. The proposed change does not increase the probability of occurrence of accidents or malfunctions previously analyzed in the FSAR; it does not create the possibility of occurrence of an accident or malfunction of a different type than any previously analyzed in FSAR. It does not reduce the margin of safety for any Technical Specification. Therefore, based on the above justifications, the proposed change does not involve a unreviewed safety question and is acceptable from a nuclear safety standpoint.

SA-SE Number: TPS-ESO-OLR12

Implementation Date: 12/31/1998

Document Type:	*Affected Documents:*	*Title:*
Procedure	SPP-9.5, Revision 2 TPS-ESO-OL R12	Provision of the TPS Load Dispatcher guidelines for the use of TAOs.

Description and Safety Assessments:

This evaluation supports the exclusion of the Transmission Power Supply (TPS) TAO process from requiring a corresponding site Temporary Alteration Control Form (TACF).

The TAO process is used by the Power System Dispatcher in maintaining configuration control of temporary alterations of transmission system components. Electric Systems Operations-Operating Letter provides guidelines to the Power System Dispatcher for the use of temporary alteration orders (TAO) associated with non-permanent changes in wiring, control circuits, and mechanical changes that affect the operation of the electrical transmission system that supplies offsite power for SQN, WBN, and BFN. The Operating Letter states the following: "The Power System Dispatcher will make a predetermination that the alteration will not compromise the safe operation of the transmission system or offsite power supply requirements covered in TPS Operating Guides for Nuclear Plants. The dispatcher will make the appropriate notifications as necessary." Transmission System Studies (TSS) are the bases for determining compliance with NRC GDC-17 and identify parameters for preparing TPS Operating Studies for SQN, WBN, and BFN. The TPS Operating Studies are used to develop the TPS Operating Guides/Instructions for SQN, WBN, and BFN. Those TPS Operating Guides/Instructions are used by the TPS Dispatching Organization to operate the transmission system in accordance with limits determined by the TPS Operating Studies. Before a transmission system component can be placed under a TAO, the Power System Dispatcher has to make a predetermination of the Impact on the transmission system and offsite power requirements delineated in the TPS Operating Guides/Instructions and make appropriate notifications as necessary.

Issuance of Temporary Alteration Orders by the Power System Dispatcher is acceptable from a nuclear safety perspective and does not constitute a unreviewed safety question. In the situation where a proposed TAO that does adversely impact the transmission system and offsite power for the nuclear site, or challenges safety systems by creating transients that could decrease the margin of nuclear safety, or increase the potential for a unit trip, it cannot be approved without site notifications (Site Operations Shift Manager), which would require entering LCO actions and/or requiring a TACF, in accordance with SPP-9.5. In the unusual event, that a TAO resulted in degradation or loss of offsite power, the safety-related system has both loss of voltage and degraded voltage relaying that will automatically disconnect the offsite power circuits from the class 1 E buses, and start and connect the emergency diesel generators, which are fully qualified and capable of safely powering all required safety loads for all design basis events. The loss of offsite power scenario is currently recognized and evaluated in the FSAR.

SA-SE Number: Unit 1 Cycle 2 COLR

Implementation Date: 01/19/1999

Document Type:	***Affected Documents:***	***Title:***
Core Operating Limits Report	Unit 1 Cycle 2 COLR, Revision 2	COLR Revision to incorporate Tech Spec Change 97-008.

Description and Safety Assessments:

The purpose of the limits on $F_Q^W(Z)$ is to ensure that the value of the initial total peaking factor assumed in the accident analysis remains valid.

Technical Specification (TS) change number 97-008 was submitted to revise TS Surveillance Requirement (SR) 3.2.1.2(a) to replace the two percent penalty factor for $F_Q^W(Z)$ with a cycle specific, burnup-dependent factor in the Core Operating Limits Report (COLR). The NRC approved this TS change August 10, 1998 with an 30 day Implementation date, instead of an implementation date at the end of Cycle 2 as TVA requested, so the COLR was revised to include the penalty factor. For Cycle 2, the factor remains at two percent for the whole cycle. Therefore, the only change for Cycle 2 as a result of the TS change and the COLR revision is that the two percent penalty factor will come from the COLR instead of the TS.

The $F_Q^W(Z)$ margin calculated in TS Surveillance Requirement (SR) 3.2.1.2 will not change after this COLR revision. Since exactly the same number will be calculated for the $F_Q^W(Z)$ margin after this change, there will be no effect on any accident and no now credible failure modes are created.

For future cycles, the penalty factor will be larger than two percent in some burnup ranges if necessary, but will always be at least two percent as documented in the WCAP referenced in TS Section 5.9.5.b.3. Note that a larger penalty factor is more conservative.

The activity being evaluated is inclusion in the Cycle 2 COLR of the two percent penalty factor for $F_Q^W(Z)$ formerly located in TS Surveillance Requirement (SR) 3.2.1.2(a). The conclusion of the FSAR accident analysis remain valid.

SA-SE Number: Unit 1 Cycle 3 COLR, Revision 0

Implementation Date: 03/29/1999

Document Type:	***Affected Documents:***	***Title:***
Core Operating Limits Report (COLR)	COLR	WBN Unit 1 Cycle 3 Core Reload and Operation

Description and Safety Assessments:

This safety 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.

Changes to be made for Cycle 3 include:

Revised core configuration: - 84 burned fuel assemblies will be discharged and replaced with 84 fresh Westinghouse Vantage +/Performance + (V+/P+) fuel assemblies 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. The 32 Tritium Producing Burnable Absorbers (TPBAR) will be discharged.

Revised Core Operating Limit Report (COLR) - The following changes will be made to the COLR:

1. As-measured MTC limit will be changed to -1.9×10^{-5} Δk/k°F from -2.1×10^{-5} Δk/k°F.
2. The control bank insertion limits will be revised and made burnup dependent.
3. Table A.1 will be included to provide F_Q margin decreases that are > than 2% per 31 EFPD.
4. AFD limits will be revised.
5. New values of W(Z) will be included.

Changes to Fuel Assembly Design - The following changes will be made to the fuel assembly design of the fresh fuel to be loaded in Cycle 3:

1. Annular axial blanket fuel pellets will be incorporated for IFBA fuel rods:
2. Manufacturing process change in the protective grid.
3. Manufacturing change in the Inconel grid inner-to-outer strap joint.

Evaluations were performed for industry events concerning rod internal pressure issue, Jedinstro debris filter bottom nozzle issue, Vantage 5H flow vibration issue, control rod insertion, axial offset anomaly, underbent mid-grid vane angle, negative flux rate trip (deleted at WBN for Cycle 3). EOL MTC requirements and boration system requirements. This review ensured that all safety limits and safety analyses for Cycle 3 were consistent and the margin of safety was not affected.

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 FSAR. 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 FSAR. The implementation of the annular pellets results in a small break LOCA PCT

SA-SE Number: Unit 1 Cycle 3 COLR, Revision 0 ***Implementation Date:*** 03/29/1999

assessment of +10 °F. Since all 10 CFR 50.46 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 FSAR 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 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 reload design is licensable under 10 CFR 50.59, and requires no prior NRC approval.

SA-SE Number: Unit 1 Cycle 3 COLR, Revision 1

Implementation Date: 09/01/1999

Document Type:	***Affected Documents:***	***Title:***
Core Operating Limits Report (COLR)	COLR	WBN Unit 1 Cycle 3 Core Reload and Operation

Description and Safety Assessments:

This safety assessment/screening review/safety 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.

Changes to be made for cycle 3 include:

- Revised core configuration - 84 burned fuel assemblies will be discharged and replaced with 84 fresh Westinghouse Vantage +/Performance + (V +/P +) fuel assemblies 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. The 32 Tritium Producing Burnable Absorbers (TPBAR) will be discharged.

- Revised Core Operating Limit Report (COLR) - The following changes will be made to the COLR:

  1. As-measured MTC limit will be changed to -1.9×10^{-5} Δk/k/°F from -2.1×10^{-5} Δk/k/°F.
  2. The control bank insertion limits will be revised and made burnup dependent.
  3. Table A.1 will be included to provide F_Q margin decreases that are > than 2% per 31 EFPD.
  4. AFD limits will be revised.
  5. New values of W(Z) will be included.

  Revision I COLR Changes:

  - *CFQ is revised from 2.50 to 2.40.*
  - *AFD limits are reduced.*
  - *New values of W(Z) are provided (calculated at the reduced AFD limits).*
  - *Table A. I is deleted and replaced with a constant 1. 02 factor which is consistent with the new AFD limits and W(Z) values.*

- Changes to Fuel Assembly Design - The following changes will be made to the fuel assembly design of the fresh fuel to be loaded in cycle 3:

  1. Annular axial blanket fuel pellets will be incorporated for IFBA fuel rods:
  2. Manufacturing process change in the protective grid.
  3. Manufacturing change in the Inconel grid inner-to-outer strap joint.

The changes to be made for cycle 3 do not constitute an unreviewed safety question because:

SA-SE Number: Unit 1 Cycle 3 COLR, Revision 1 ***Implementation Date:*** 09/01/1999

- Evaluations have been made of the effect of the core configuration specified for cycle 3 upon the safety analyses described in the FSAR. 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 FSAR. The implementation of the annular pellets results in a small break LOCA PCT assessment of +10 °F. Since all 10 CFR 50.46 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 FSAR 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 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 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 reload design is licensable under 10 CFR 50.59, and requires no prior NRC approval.

SA-SE Number: WBN Offsite Dose Calculation Manual

Implementation Date: 04/4/1999

Document Type:	***Affected Documents:***	***Title:***
Offsite Dose Calculation Manual	ODCM Revision 1	ODCM revision.

Description and Safety Assessments:

This ODCM revision deletes the requirement to sample weekly for noble gases released through containment venting and adds all applicable sampling footnotes previously applied to containment venting to the sampling requirements for the Auxiliary Building Exhaust (ABE) to implement DCN 50165. This DCN installs filter trains on the containment pressure relief (vent) line and changes the operation of this system such that the venting of containment will now be a continuous (instead of a batch) release into the Auxiliary Building Exhaust release header. This change was reviewed in a separate Safety Evaluation, and will not be evaluated in this document. Since the release will be continuously feeding into the ABE, it cannot be quantified separately from the ABE, since any samples obtained at the ABE radiation monitor will contain activity from both the ABE ventilation system and the containment pressure relief system. Because it will not be possible to associate the activities with their respective sources, the requirement to sample separately for the containment vent is being deleted. Samples obtained at the ABE will contain activity from both sources and will be used to determine the amount of radioactivity released through the ABE. A small amount of the total activity released may no longer be identified and accounted for since the vented activity will be sampled after dilution by the Auxiliary Building ventilation. Releases from venting of containment were reviewed for the years 1996 through 1998, and it was determined that concentrations of activity in containment which result in significant releases (> 10% of the total site release/dose as recommended in Regulatory Guide 1.109) will be well above the levels of detection after they are diluted. Thus, the level of effluent control required by 10 CFR Part 20.1302, 10 CFR Part 50, and 40 CFR Part 190 will not be lessened as a result of this change.

This ODCM change revises the Steam Generator Blowdown maximum flow rates in Section 6.0 and on Figure 6.3 to match the WBN UFSAR as a corrective action to PER 99-001837-000. It also adds wording on Action E of Table 1.1-2 to match compensatory action requirements for other noble gas monitors (contained in Action C of that same table), clarifies that Table 2.2-2 Note 4 applies only during releases from these points, and changes a reference to a cancelled site procedure in Appendix C. These latter four changes are considered non-intent changes as defined in ODCM Appendix C and will not be evaluated further in this analysis.

There are design basis accidents or credible failure modes associated with this change.

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. This change also does not affect the way in which any plant equipment is operated. 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. Implementation of this change will not create the possibility of a new type of accident or malfunction not previously evaluated in the FSAR, will not increase the probability or consequences of any accident or malfunction previously evaluated in the FSAR, and will not reduce the margin of safety of any Technical Specification. It is concluded that this change does not involve an unresolved safety question, and that it is acceptable from a nuclear safety perspective.

SA-SE Number: WBLMGR-98-003-0

Implementation Date: 11/25/1998

Document Type:	***Affected Documents:***	***Title:***
Tech Spec Bases	Tech Spec Change 98-015, Revision 21 Procedure 1-SI-61-1	The Change Package is to correct/clarify four separate issues which affect the TS Bases sections.

Description and Safety Assessment:

Tech Spec (TS) Change Package is to correct/clarify four separate issues which affect the TS Bases sections for (1) ACTIONS B. 1 and B.2 and C1 of TS 3.6.12, (2) SR 3.6.11.4, (3) SR 3.6.11.5, and (4) SR, 3.6.12.6.

(1) TS 3.6.12, Ice Condenser Doors, ACTION Condition C states, "Required Action and associated Completion Time of Condition B not met" TS 3.6.12, ACTION Condition D states "Required Action and associated Completion Time of Condition A or C not met," Thus, Condition C entry is the next allowed progressive step for any failure to comply with Condition B Required Actions or Completion Times. While the Bases for Tech Spec 3.6.12 do require the same as above, they also state in Section B.1 and B.2 "If the maximum ice bed temperature is > 27 °F at any time, the situation reverts to Condition C and a Completion time of 48 hours is allowed to restore the inoperable door to OPERABLE status or enter into Required Actions D.1 and D.2. Ice bed temperature must be verified to be within the specified Frequency as augmented by the provisions of SR 3.0.2. If this verification is not made, Required Actions D.1 and D.2, not Required Action C.1 must be taken." [The last sentence is the problem.], and in Section C.1 "Condition C is entered from Condition B only when the Completion Time of Required Action B.2 is not met or when the ice bed temperature has not been verified at the required Frequency." The revision to these Bases deletes the above Section C.1 statement altogether, and revises its first sentence to read, "If Required Actions or Completion Times of B.1 or B.2 ..." Section B.1 and B.2 is changed to read, "If the maximum Ice bed temperature is > 27 °F at any time, or ice bed temperature is not verified to be within the specified Frequency as augmented by the provisions of SR 3.0.2, the situation reverts to Condition C and a completion time of 48 hours is allowed to restore the inoperable door to OPERABLE status or enter into Required Actions D.1 and D.2. "

(2) TS SR 3.6.11.4 is a visual inspection for ice or frost buildup in excess of 0.38 inches on structural members comprising at least two flow channels per ice condenser bay. The associated TS Bases states, "More than one discrepant flow channel in a bay is not acceptable, however." With no further guidance provided, one may conclude that the only alternative is to clear some flow channels and reperform the SR. However, per Westinghouse letter WAT-D-10549 the intent was to initiate a more detailed inspection to assess whether or not accident analysis limits were being exceeded. Thus, the TS Bases for SR 3.6.11.4 is revised to read, "More than one discrepant flow channel in a bay requires a more detailed evaluation to assess actual ice and frost blockage in regards to allowed accident analysis limits and the need for ice/frost removal."

(3) TS SR 3.6.11.5 states, -"Verify by chemical analyses of at least nine representative samples of stored ice:

a. Boron concentration is ≥ 1800 ppm; and
b. pH is ≥ 9. 0 and ≤9.5."

To clarify the sampling and analysis requirements of this SR, its associated TS Bases are revised to read as follows, "This is accomplished by obtaining at least nine representative samples. Representative samples are those taken approximately one foot from the top of each selected ice basket, with the selected baskets being distributed throughout the ice condenser. If the initial analysis results in an average pH value or an average boron concentration outside prescribed limits, 55 additional randomly selected samples shall be analyzed. If the average pH value or average boron concentration of the expanded sample is outside their prescribed limit(s), then entry into ACTION Condition A is required." Using average boron concentration and the average pH value to meet acceptance criteria is based on information provided in Westinghouse letter WAT-D-10554. The sample collection criteria is based on Westinghouse Equipment Operating Procedure EOP-2, Analysis of Ice. The expanded sample size is based on a management decision established to give a very high confidence in the status of the ice condenser as a whole, and as concurred with by the NRC for a corrective action taken when the issue of "representative" samples was first identified and a single sample failed to meet boron concentration requirements.

(4) TS SR 3.6.12.2 is a weekly visual inspection of the ice condenser intermediate deck doors (IDDs) Performed during Operating modes to verify the doors are closed and not impaired by ice, frost or debris, thus providing assurance the doors have not been left open or obstructed. TS SR 3.6.12.2 is an 18 month (outage) inspection of the IDDs for structural deterioration, free movement of their vent assemblies (curtains), and free movement of the doors themselves. The latter is determined by measuring each doors "lifting force." Questions raised about the applicability of the lift tests to SR 3.6.12.2 led to incorporating the performance into the performance instruction for SR 3.6.12.2. In response to TVA's questions on this matter, Westinghouse letter WAT-D-10547 stated that failure to pass the lift test was not an OPERABILITY issue. Based on this and additional information provided in WAT-D-10547, the TS Bases for SR 3.6.12.6 is revised as follows to clarify the purpose of the lift test, and thus its applicability to only SR 3.6.12.6:

SR 3.6.12.6

The above test lifting forces were established based upon test results gathered on newly manufactured Intermediate Deck Doors set up in fixturing to simulate plant installation tolerances. The lifting force values developed were to account for and envelope expected door panel variations in weight and hinge friction and alignments. The intent of the surveillance is to establish a method of detecting abnormalities or deteriorating conditions of the door panels or hinges after completion of refueling outage maintenance activities.

1-SI-61-1, Determination of Boron and pH on Ice Condenser Ice," is revised to incorporate the detailed methodology for complying with the above change to the TS Bases for SR 3.6.11.5. Specifically, the SI did not provide the guidance necessary to ensure the selection of ice samples were sufficiently distributed throughout the ice condenser to yield results representative of the ice condenser ice as a whole. Also, the SI did not provide guidance for when an individual sample did not meet the boron and/or pH acceptance criteria. To accomplish the above, the SI now requires at least three randomly selected ice baskets from each ice condenser quadrant (six bays per quadrant), giving a minimum total of 12 baskets sampled across the entire ice condenser. The boron and pH analysis results for each ice sample are then averaged to determine their values for the ice condenser as a whole. If either averaged boron or pH value does not meet its respective SR acceptance criteria, then 55 additional randomly selected baskets are sampled. Should either averaged boron or pH value from this expanded sample not meet their acceptance criteria, then the Ice Condenser Ice Bed is declared inoperable, and the appropriate actions of TS 3.6.11 are entered. These changes are consistent with Westinghouse letter WAT-D-10554, Westinghouse Equipment Operating Procedure WAT-EOP-2, "Analysis of Ice," and sampling and analysis methods used by other ice condenser nuclear plants.

Accident Evaluation

The ice condenser is an engineered safety feature provided to minimize the pressure and temperature excursions in containment following a loss of coolant accident (LOCA) or a steam line break (SLB). The ice condenser is designed to remain functional following a design basis earthquake, and thus is not postulated as being susceptible to structural failures. If not property maintained, it can fail to provide sufficient air and steam flow through it due to ice blockage of its flow paths through the ice bed region or from physically restrained doors that are frozen shut Proper boron concentration and pH of the ice condenser ice is necessary for ensuring reactor core shutdown margin and hydrogen generation in containment are within LOCA analysis limits. Changing the TS Bases and 1-SI-61-1, as described above, does not affect the ability of the ice condenser to perform its design function during a LOCA or SLB, as the changes do not involve, nor are they the result of any physical or operational changes to the ice condenser.

SA-SE Number: WBLMGR-98-003-0 ***Implementation Date:*** 11/25/1998

Except for the changes to the ACTION Conditions of the TS Bases for TS 3.6.12, all other changes are supported by a Westinghouse WAT-D letter. Additionally, the sampling changes to 1-SI-61-1 exceed the requirements of TS SR 3.6.11.5.

The TS Bases changes and changes to 1-SI-61-1 are not the result of a physical modification to the ice condenser, or license maintenance or operational requirements. As the TS Bases serve to provide clarification of requirements provided by the Tech Specs, changing the TS Bases to be consistent with TS requirements or to contain enhancements and/or clarifications can only serve to preclude possible TS noncompliance, or over compliance, due to a misunderstanding of the requirements. Except for deletion of the contradictions in the TS Bases for TS 3.6.12, all other changes are supported by Westinghouse documentation. Based on the above, this TS Bases change does not constitute an unreviewed safety question or required a TS change.

SA-SE Number: WBLMGR-99-001-0

Implementation Date: 02/05/1999

Document Type:	***Affected Documents:***	***Title:***
Tech Spec Bases	Tech Spec Change 98-026, Revision 25	TS Bases Change 98-026 administratively deletes a contradictory statement in the Tech Spec Bases for LCO 3.4.11, Pressurizer PORVs.

Description and Safety Assessments:

Tech Spec Bases Change 98-028 administratively deletes a contradictory statement in the Bases for the ACTIONS of LCO 3.4.11, Pressurizer PORVs, (Page B 3.4-53) which states: "Testing is not performed in lower MODES." This sentence is incorrect and inconsistent with the Tech Specs and Bases for Surveillance Requirements (SR) 3.4.11.2 and SR 3.4.12.8. SR 3.4.11.2 and its associated Bases currently specify the performance of an 18 month valve cycle test for the PORVs with no Mode restrictions on test performance. SR 3.4.12.8 and its associated Bases currently specify the performance of an 18 month COMS channel calibration including PORV actuation with no Mode restrictions on test performance. The associated surveillance instructions for these SRs allow performance of PORV testing in Modes 4-6 for SR 3.4.11.2 and in "All Modes" for SR 3.4.12.8.

The statement, "lower modes" is not defined but apparently means Modes 4-6, and was carried over from the Standard MERITS Tech Spec which has the same statement. Based on discussion with industry personnel involved with the development of the generic MERITS Tech Specs, the statement apparently was an inference to the LCO 3.0.4 exception contained in LCO 3.4.11. During development of the MERITS Tech Spec, several plants had existing Tech Spec requirements to perform PORV testing in Mode 3 (instead of "lower modes" which would require an exception to LCO 3.0.4 in order to perform the test. Since neither SR 3.4.11.2 or its BASES discuss a Mode restriction on the PORV surveillance testing, the obscure note evidently went undetected during development of the WBN Tech Spec and related surveillance instructions. Therefore, in accordance with the WBN corrective action program, the subject statement should be deleted under TS Bases Change 98-026.

The subject change to the TS Bases is administrative in that it deletes contradictory information in the TS Bases. The current sentence, if not deleted, gives the appearance of restricting PORV testing to "higher" Modes (e.g., Modes 1-3), where no such restriction actually exists. The change has no effect on any design, hardware, testing aspects, or operational characteristics for the plant and therefore can have no effect on any accident previously evaluated in the SAR or create any potential for a new accident not previously evaluated in the SAR.

The subject change to the TS Bases is administrative in that it deletes contradictory information in the TS Bases. Since the proposed change does not alter the testing requirements of the Tech Specs, and has no effect on any design, hardware, testing aspects, or operational characteristics of the plant, and has no effect on any accident or malfunction, the proposed change does not constitute an unreviewed safety question.

SA-SE Number: WBOCEM-98-001-0

Implementation Date: 05/11/1998

Document Type:	***Affected Documents:***	***Title:***
Procedure	ODCM Revision 9	ODCM Revision

Description and Safety Assessments:

This ODCM revision increases the referenced maximum design flow rate for the release of a Waste Gas Decay Tank (WGDT) from 55 cfm to 100 cfm. The change addresses concerns raised in WBPER970332 regarding the sources and uses of design output. This PER identified that the maximum flow rate value of 55 cfm specified in the ODCM could not be located in any design output. The 100 cfm value was obtained from the system description for the radwaste treatment system.

One of the design accidents described in Chapter 15 of the UFSAR is a ruptured WGDT and subsequent release of the tank contents to the atmosphere. This is the only described accident/malfunction which is evaluated for potential impact by this change. The revision will allow the routine release of the WGDT at higher flow rates than previously allowed. The change would not increase the probability for a WGDT rupture from an infrequent to moderate frequency, nor affect the operation of any equipment important to safety, therefore it would not increase the probability of an accident previously evaluated or malfunction of equipment important to safety. The change would allow a routine release at higher flow rates but would not increase the total amount of radioactivity therefore consequences of an accident would not increase. The change does not impact the technical specifications therefore, would not reduce the margin of safety."

SA-SE Number: WBOCEM-98-023-0

Implementation Date: 09/11/1998

Document Type:	***Affected Documents:***	***Title:***
Offsite Dose Calculation Manual	Offsite Dose Manual, Revision 10	Wording is changed to reflect the upcoming switch from Site Standard Practices to Standard Programs and Processes

Description and Safety Assessments:

The wording in Section 5.3 regarding the requirements for how to change the ODCM is revised to reflect the upcoming switch from Site Standard Practices to Standard Programs and Processes, and the list of specified organizations which must review all ODCM changes is deleted. This change is administrative and will not be evaluated further in the Safety Assessment/Evaluation. This ODCM revision also changes the equipment ID for the Condensate Demineralizer Effluent Flow Rate Measurement device listed in Tables 1.1-1 and 2.1-1 from FI-14-456A/B to FI-14-456 to implement DCN 39302. A separate safety assessment/evaluation was performed to evaluate the equipment change (WBPLMN-97-084-1) and SAR change, therefore this change will not be evaluated for safety impacts in this document. The ODCM revision clarifies that the compensatory actions for inoperable noble gas and iodine/particulate samplers (Table 1.1-2, ACTIONS C and D) on effluent release points defined in the ODCM are not required to be performed during periods when no releases are being made. This is implied by the statement that "effluent releases via this pathway may continue provided that ... " and will now be stated directly in the compensatory action. The requirement for the compensatory actions taken when the isokinetic sampler or the heat trace are inoperable (Table 1.1-2 ACTION G) on the Shield Building Exhausts are revised to add that the continuous iodine/particulate sampling is discontinued during the time and that releases from those points are suspended. This change makes it clear that iodine/particulate sampling cannot be performed during periods when this equipment is inoperable. This is because virtually all of the radioactivity will plate out on the sample lines, rendering the samples invalid.

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 clarifies the meaning of compensatory action statements for inoperable effluent monitoring equipment. 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.

SA-SE Number: WBO-MET-99-011-0

Implementation Date: 09/23/1999

Document Type:	***Affected Documents:***	***Title:***
Maintenance	Maintenance Procedure 99-12996-00 Maintenance Procedure 98-16111-00	Spent fuel pump removed from service to replace diaphragm.

Description and Safety Assessments:

The requested action is to remove from service the spent fuel pit cooling pumps for the duration required to replace a diaphragm in the 0-ISV-078-516-S, Spent Fuel Pit Cask Load Area Supply Isolation Valve and repair a body-bonnet leak on 0-THV-78-512, Spent Fuel Pit Heat Exchanger B Outlet. The pumps must be tagged out of service to isolate these valves for maintenance. The work and tag out should be performed within 16 hours but 24 hours will be considered to be the maximum time for the pumps to be out of service. Assuming spent fuel pit water is 90°F at beginning of work, the temperature after 24 hours would be approximately 104°F. The pumps are designed to remain functional for the design basis earthquake and within the required stress limits for the operational basis earthquake. The most serious failure of this system would be complete loss of water in the storage pool. The system is designed such that loss of water cannot occur to the extent of the spent fuel becoming uncovered.

Chapter 15 of the FSAR discusses the causes and consequences of a fuel handling accident, including the dropping of an irradiated fuel assembly in the spent fuel pool or on the fuel handling area floor. None of the provisions discussed in the FSAR for response to a fuel handling accident are altered or affected by the work associated with either work order considered in the review.

The FSAR also discusses use of the spent fuel cooling system as a means of removing residual heat from the reactor core during a flood above plant grade. This means of operation is accomplished by installing temporary, pre-constructed spool pieces between the spent fuel pool cooling system and the Residual Heat Removal (RHR) system. The spool piece, in conjunction with valve realignments, allows water to be diverted from the spent fuel cooling system to the core via the existing RHR system piping. Return to the spent fuel pool cooling system during flood mode cooling is via the fuel transfer tube to the fuel transfer canal, then through the slot between the fuel transfer canal and the spent fuel pool. This open core cooling mode of operation is only utilized if a flood occurs during refueling operations. For a flood that occurs with the unit on line, a different means of removing residual heat from the reactor core is utilized that requires cross-connecting the high pressure fire protection pumps to the auxiliary feedwater supply lines to provide feedwater and use of the steam generator power operated relief valves to provide a steam removal pathway. This alternative means for use during power operation does not rely upon operation of the spent fuel pool cooling system.

Two failure modes have been identified and considered:

> First, the work activity may stretch beyond the 24 hours allowed. This work stretch out may be due to discovery that the parts available for the work are not the correct parts, or by damage to or discovery of damage to valve parts during disassembly. This is probably the most credible failure mode for the work considered.
>
> Second, the system may not be capable of return to service due to failure of some system component or components. For example, a valve in the hold order boundary may stick in position and prevent reestablishment of flow or all three pumps may fail to restart. Since the valves involved are all manual valves which are highly reliable and will be exercised by the action of placing them in position for the hold order, and since the pump controls are redundant [i.e., only 1 of the 3 installed pumps has to restart to reestablish cooling] and simple (i.e., a manual push button to start with no automatic interlocks or prohibits] this is considered a very highly improbable failure mode.

SA-SE Number: WBO-MET-99-011-0 ***Implementation Date:*** 09/23/1999

This activity involves using approved administrative controls (hold order) to prevent the operation of the spent fuel pit pumps. The pumps are being removed from service to allow routine maintenance of valves in the spent fuel pool cooling system. The FSAR specifically permits removal of the spent fuel pool cooling system from service to perform maintenance of system components. This hold order will be in place a maximum of 24 hours. During this time the spent fuel pool water temperature will increase approximately 14°F. This minimal increase in temperature will not place the plant in an unanalyzed condition or create an unreviewed safety question.

SA-SE Number: WBORAD-97-001-0

Implementation Date: 03/27/1998

Document Type:	***Affected Documents:***	***Title:***
FSAR Change Package 1490	FSAR	Designation of Concrete Pad as Storage Area for Radioactive Material and a Temporary Storage Area for Dry Active Radioactive Waste

Description and Safety Assessments:

A concrete pad has been constructed (DCN R-38684-A) on the northeast side of the Condensate Demineralizer Waste Evaporator (CDWE) Building, external to permanent plant buildings for storage of radioactive material (contaminated equipment, tooling, etc.) and temporary (interim, i.e. less than 5 years) storage of dry active radioactive waste (DAW). Outside radwaste storage is addressed in FSAR Section 11.5.5.2.

Appropriate isotopic distributions are used to determine the radiological impact of temporary storage of rad material and radioactive waste on the pad. The concrete pad is enclosed by an eight foot locked fence to provide: physical security, restrict personnel access, and radioactive material control. The area will become part of the RADCON surveillance program to include: radiological surveys, postings, and visual inspections to detect containers integrity. All radioactive material/waste will be enclosed in weather resistant containers. RADCON calculation, "Calculation for the Use of the CDWE Concrete Pad for the Interim Storage of Rad Material and Radwaste," evaluated the: impact of creating a direct radiation pathway, impact from other initiating events (e.g., tornado, missiles, seismic), impact on the accident mission dose calculations, impact on creating new radioactive effluent. Based on this evaluation, DAW that is stored on the concrete pad will be limited to:

DAW 3412 Ci

In addition, the following dose rates must be administratively controlled:

- < 500 mrem/yr on the outside of the concrete pad chain link fence (SSP-5.01/RCI- 103)
- < 100 mrem/hr along the North wall of the CDWE to the Shield Building Vent Radiation Monitoring room and through the North gate along the railroad tracks (WBN-APS3-049 mission dose routes) (RCI- 103)
- 1 R/hr at 30 cm from any item on the pad
- All rad material/waste on the pad will be enclosed in weather-resistant containers (RCI-103)
- Interim/temporary storage on pad will be limited to less than 5 years (RCI-103).
- Resin waste and contaminated oil/liquids will be limited to that which is being staged for imminent shipment

FSAR Section 11.5.5.2 will be revised to delete "The contact dose for containers stored outside is in accordance with 49 CFR 173.441," the radiation level limit criteria for shipping radioactive material, since this requirement is addressed in Chapter 12 of the FSAR.

Spent resins that have been dewatered and contaminated oil are not apart of this evaluation. If spent resin or contaminated oil are placed on this concrete pad, they are there for imminent shipment offsite and not there for storage.

FSAR Section 12.3, will be revised to incorporate outside temporary storage of radioactive materials and radioactive waste (DAW).

SA-SE Number: WBORAD-97-001-0 ***Implementation Date:*** 03/27/1998

Storage of radioactive material and temporary storage of DAW in areas exterior to permanent plant buildings does not impact any FSAR evaluations (accident analysis or equipment malfunction failures) previously evaluated. No new accidents or equipment malfunction failures are created by this proposed change. Technical Specifications are not affected. This proposed change has no effect on Structures, Systems, or Components (SSCs) or radwaste treatment systems. Therefore, on the basis of the RADCON calculation, "Calculation for the Use of the CDWE concrete pad for the Interim Storage of Rad Material and Radwaste," it is concluded that the storage of radioactive material/waste outside is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number: WBORAD-98-007-0

Implementation Date: 10/09/1998

Document Type:	***Affected Documents:***	***Title:***
FSAR Change Package 1548	FSAR	Evaluation of changes to the WBN FSAR Report Chapter 12

Description and Safety Assessments:

This is an evaluation of changes to the WBN Final Safety Analysis Report, Chapter 12, Sections 12.4 and 12.5 as described in SAR Change Package Number 1548.

Changes to Section 12.4 (Dose Assessment) involves updating annual person-rem exposure estimates. Previously, estimates of annual person-rem exposures were based on conservatively assumed radiation sources, design shielding, calculated design dose rates, and estimates of projected work functions and occupancy times. FSAR Change Package Number 1548 calculates estimates of annual person-rem exposures based on methodology provided in Regulatory Guide 8.19, which focuses principally on procedural and administrative considerations. Dose estimates are formulated using actual plant maintenance and operations occupancy times and collective exposures, and dose corrected for projected unplanned maintenance and operational occurrences.

Changes to Section 12.5 (Radiological Control (RADCON) Program) involves updating radiation protection program descriptions regarding control of access to high radiation areas to ensure consistency in terminology and program requirements between Technical Specifications 5.11 and FSAR Program description. Previously, the FSAR attempted to summarize requirements outlined in the Technical Specifications. This summary fell short of effectively capturing essential elements for access control, monitoring, surveillance and posting requirements for high and very high radiation areas. FSAR Change Package Number 1548 eliminates the summarization of high radiation area access controls and provides a direct reference to Technical Specification 5.11 to ensure consistency in all program elements. Clarification is also made for access controls to very high radiation areas which specify that the elements outlined for access control to very high radiation areas is in addition to the requirements for access controls to high radiation areas in accordance with 10CFR20.1602.

Changes identified in FSAR Change Package Number 1548 are administrative in nature. These changes do not perform any safety function and are not used nor required to mitigate any accident. FSAR Chapter 15 accident analyses does not identify any failures associated with occupational radiation dose nor control of access to high or very high radiation areas. This change is not associated with increasing the consequences of an accident previously evaluated. This change does not change any system or the logic or function of any system that is important to safety. This change is not associated with any protective feature used to detect or mitigate the effects of a design basis accident. A review of the detailed changes leads to the conclusion that this change is safe and does not constitute an unreviewed safety question.

The FSAR does not identify any equipment or system failure modes which could occur as a result of the administrative changes to the plant's annual person-rem exposure estimates or the radiation protection program controls for access to high or very high radiation areas. No new potential single failure of an existing components or system will occur as a result of these administrative changes. These changes do not cause any system important to safety to fail to fulfill it functional requirements."

SA-SE Number: WBOTSS-97-241-1

Implementation Date: 02/19/1999

Document Type:	*Affected Documents:*	*Title:*
Procedure Change	MI-88.003, Revision 3 AOI-29	Temporary penetration sealing requirements, compensatory controls and configuration control.

Description and Safety Assessments:

The subject procedure revision identifies requirements for sludge lancing during core alterations and the installation of hoses and cables associated with Steam Generator (SG) maintenance. This includes temporarily installing hoses and cables in a containment penetration during an outage (unit in Mode 5, 6 or core empty). This Safety Evaluation demonstrates that these penetrations can be in an altered configuration and the selected temporary fluid and gas filled hoses can remain in service during core alterations and provide a barrier between containment and the outside atmosphere in the event of a fuel handling accident or loss of residual heat removal system (RHR) shutdown cooling precluding and unfiltered release of radioactive material to the public.

Sludge lancing uses an equipment system that temporarily locates equipment both inside and outside containment and requires a containment penetration for connection hoses and cables. Also compressed air hoses (with air flow outside to inside) are required to power the sludge lance return pumps and is also used to cool the eddy current equipment and power other SG related activities. The sludge lance connecting hoses contain 1) high pressure cleaning water entering containment, 2) cleaning water (with entrained sludge and air) exiting containment, and 3) air (from the surge tank and holding tank) vented back into containment. The sludge lance cables are for equipment control wiring and communications. The eddy current cables are for platform cameras, communications, equipment power and control, and to transmit the eddy current data out of containment.

Compensatory measures are established for the following conditions:

a) This procedure revision is an emergency closure procedure if containment penetrations X-108 and X-109 are breached during the performance of this instruction. Emergency closure is controlled by TI-58.002 if AOI-14 is entered for a loss of RHR shutdown cooling during mid-loop operations. Estimated closure time is 30 minutes. Emergency closure is directed by AOI-29 in the event of a fuel handling accident inside containment or on the refueling floor.
b) This procedure revision maintains configuration control over sludge lance equipment and related manual isolation valves at penetrations X-108 and X-109 once containment closure is established by this instruction. Personnel are assigned to close manual isolation valves inside containment and the annulus when these valves are open during sludge lancing or maintenance of SG's.
c) This procedure revision directs installation of blind flanges on penetrations X-108 and X-109 within 4 hours following a fuel handling accident inside containment or core boiling due to a loss of RHR shutdown cooling during mid-loop operations to ensure seismic design requirements are met during the post accident period.
d) This procedure revision directs restoration of penetrations X-108, X-109, and X-118 within 27 hours of notification of a flood per AOI-7.01 as these altered penetrations are below the flood plane are not rated for design basis flood pressures.
e) This instruction obtains Shift Manager direction for closure of manual isolation valves and removal of temporary hoses and cables installed at penetrations X-108, X-109, and X-118 in the Unit 1 Additional Equipment Building upon notification of a tornado watch/warning per AOI-8 to ensure that no missile hazards will create an ABSCE or containment breach, if required.

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These measures ensure design basis requirements and Tech Spec requirement are met or exceeded for floods, tornadoes, earthquakes, fuel handling accidents, and loss of RHR shutdown cooling.

The WBN Unit 1 Technical Specification Bases states:

"In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as 'containment closure' rather than 'containment OPERABILITY.' Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required....

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling...

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be NRC approved and may include use of a material that can provide a temporary, atmosphere pressure, ventilation barrier for the other containment penetrations during fuel movements..."

The affected penetrations 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 penetrations. These altered penetrations have no impact on Appendix R safe shutdown equipment in modes 5 and 6. The interstitial spaces between the hoses and cables will be filled with at least 12 inches of RTV foam per design drawings and consistent with Tech Spec 3.9.4 bases. When penetrations are breached for installation and removal of cables and hoses, breaches to Auxiliary Building Secondary Containment Enclosure (ABSCE) and Containment Closure are tracked per TI-65 and TI-68.003 respectfully 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 per 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 sludge lancing equipment when the altered penetrations are sealed to ensure no path exists from containment to outside atmosphere. 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. Within 4 hours following a fuel handling accident, the affected penetrations will be restored using one permanent flange to assure no unfiltered radioactive release path is created due to a seismic event. Operability for containment integrity is established prior to Mode 4 entry by performance of LLRT's on testable penetrations. Contingencies have been developed to restore altered penetrations to normal upon notification of other design basis events such as flooding and tornadoes which could adversely impact the altered penetration integrity.

The probability of an accident or the consequences of an accident is not increased by the procedure revision because fuel movement activities and fuel handling equipment and procedures are independent from this instructions. The radiological consequence analysis in FSAR for the fuel handling accident remains valid and no revisions to the analysis are required. Installation of permanent closure blind flanges ensure closure integrity is maintained. The probability of occurrence of a malfunction of equipment important to safety will not be increased. The primary penetrations opened by this procedure will be leak tested prior to Mode 4 to ensure containment operability during Modes 1 through 4. Containment closure will be maintained as previously shown, therefore no increase in the release of unfiltered radioactive material will occur. The possibility of an accident or malfunction of different type will not be increased and the margin of safety will not be reduced. Technical Specification Bases 3.9.4 states that during the Fuel Handling Accident no pressurization of containment is credible. Without pressurization, the margin of safety is simply containment closure. Therefore, there is no unreviewed safety question.

SA-SE Number: WBOTSS-98-005-0

Implementation Date: 03/11/1998

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-98-01-030	Control Rod Drive Mechanism (CRDM) Cooler A-A Motor Failure

Description and Safety Assessments:

January 1, 1998, a Work Document, WO 98-000002-00, was initiated to troubleshoot the cause for CRDM cooler A-A tripping off. CRDM Cooler A-A Motor 1 (1-MTR-030-0083/1-A) was found shorted to ground. During plant operation at power levels greater than 0, this motor is not accessible for maintenance. TACF 1-98-01-030 is initiated to determinant cable to permit CRDM Cooler A-A Motor 1 (1-MTR-030-0083/1-A) to be disabled while CRDM Cooler A-A Motor 2 (1-MTR-030-0083/2-A) will remain operable. This change will allow the CRDM Cooler A-A to meet the 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. Annunciator 1-XA-55-5C, window 102A may alarm when CRDM Cooler A-A is operated due to less than design negative pressure at flow switch 1-FS-30-83A/B-B and 1-FS-30-83B/A-B.

There are no design basis 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. CRDM Coolers B-B, C-A and D-B and their associated dampers are not impacted by this TACF. CRDM Cooler A-A will be capable of providing cooling as required by the Fire Protection Report. CRDM Cooler A-A will provide cooling when operated in the bypass mode, which is the mode needed to cool lower containment.

The requirement in the Fire Protection Report is that either 3 Lower Compartment Air Coolers (LCACs) or 2 LCACs plus 2 CRDM Coolers be functional for an Appendix R event for which a minimum total heat removal capability of 6.3 M 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 heating, ventilating, and cooling (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 review of manufacturers fan performance curves, a single CRDM fan will provide 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 in the bypass mode of operation. The minimum design now is 31,410 CFM. The two fan flow design heat removal rate for CRDM cooler A-A is not degraded in the bypass mode of operation as this alteration maintains air now above minimum design and cooling water design flow is not affected.

One design function of this cooler is to take suction from the CRDM shroud during normal reactor operation. CRDM cooler A-A will not be capable of performing this function. This is not a safety related function or an Appendix R fire related function. The remaining 3 CRDM coolers will be capable of removing the required air flow from the CRDM shroud. The design basis function of these coolers, per the Fire Protection Report, is to provide cooling to maintain lower containment temperatures when operated in the bypass mode. The design basis function of this system is not adversely impacted because all 4 CRDMs are capable of cooling lower containment during Appendix R fire conditions. The implementation of TACF 1-98-01-030 does not introduce different failure modes which impact the Appendix R function from the existing CRDM system configuration. 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.

This activity does not constitute an unreviewed safety question. The CRDM cooling system will continue to be capable of meeting its Appendix R function to cool lower containment.

SA-SE Number: WBOTSS-98-052-0

Implementation Date: 03/02/1999

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration Control Form (TACF)	TACF Number 1-98-020-024 Revision 1 Work Order (WO) 990014400	Temporary Water Supply to EHC Heat Exchanger

Description and Safety Assessments:

TACF 1-98-020-024, Revision 1 and WO 990014400 temporarily installed supply and discharge hoses to make Electrohydraulic Control (EHC) Heat Exchanger 1A operable to support placing the main turbine on turning gear during the second refueling outage while the raw cooling water (RCW) system is out-of-service for various modifications/maintenance. The temporary water supply will be provided from the raw service water (RSW) system. The required flow of RSW to the EHC Heat Exchanger 1A is 20 gpm of continuous flow.

A rubber hose will be routed from a fire hose "tee" or "Y" to a strainer in to EHC Heat Exchanger 1A supply piping. A rubber hose will be routed from the test connection on the discharge of EHC Heat Exchanger 1A to the floor drain or equipment drain. The supplier of water to the RSW system is the Fire Protection system which will remain in service during the second refueling outage. Therefore, a dependable supply of water to EHC Heat Exchanger 1A will be provided. RSW will be isolated from the Fire Protection System in the event of a fire and thus will not add unanalyzed loads to the Fire Protection System during a fire.

The above described change will occur while the plant is shutdown for the second refueling outage and will not be in affect after the plant is in operation. The change does not affect safety-related equipment or place safely-related equipment in a condition that is adverse to safety. This change does not increase the probability of or consequences of an accident nor does it introduce different types of accidents. This change does not affect equipment that is defined in to Technical Specifications. Considering this, implementation of this TACF does not constitute a unreviewed safety question.

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Implementation Date: 05/07/1998

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 0-98-1-14	Replace acid bleed valve 0-FCV-14-141 with a manual isolation valve.

Description and Safety Assessments:

The change of the 0-FCV-14-141 from a 0.5 inch air operated valve made with Carpenter 20 material to a manual valve 0.5 inch stainless steel (Type 316) ball valve rated at 600 psi. This material is rated as good per the 'Corrosion Resistance Tables' for use with concentrated sulfuric acid. It has socket weld ends and a Teflon seat. To provide for proper engagement of the pipe and valve, a coupling may be installed on either side of the valve.

A field walkdown of this valve was performed. The valve is located in the Acid Reclaim Tank Room. There is a spray shield around this valve to deflect any potential leakage to the floor. There is no safety related components in this isolated room.

The SOI-14.02 will be revised to include instructions to operate this valve manually as opposed to the automatic/handswitch operation. The control air which originally was used to operate the original valve will be isolated. The function of the valve to flow acid has not been changed. In addition to the SOI revision, a TACF tag will be installed at the handswitch to identify the temporary configuration change.

The change in material does not affect the operational characteristics, processes, or procedures as described in the UFSAR. The 0-FCV-14-141 valve is not discussed in the UFSAR, but does appear in Figure 10.4-36B. Closure of the air valve, therefore, does not impact the UFSAR.

The Watts Bar Safety Evaluation Report (SER) and its Supplements have been evaluated for impacts. The changes of this TACF do not impact the NRC's understanding of the design and operation of Watts Bar Nuclear Plant as described in the FSAR.

The changes being made by this TACF do not affect any UFSAR Chapter 15 fault or operational transient evaluations. The 0-FCV-14-141 valve does not perform any safety-related functions and is not required for the orderly shutdown of the reactor. Therefore, the probability of occurrence or consequences resulting from a previously evaluated accident or equipment malfunction is not increased. In addition, since the 0-FCV-14-141 valve does not have the potential to adversely impact any other safety related equipment, the possibility for an accident or equipment malfunction of a type different than any other evaluated previously in the UFSAR is not increased. Since the Technical Specifications do not address the condensate polishing demineralizer system and the changes being implemented by this TACF do not have the potential to indirectly affect Technical Specification systems, the safety margins defined in the Technical Specifications Bases are unaffected. Therefore, on the basis of the evaluation of effects, it is concluded that TACF 0-98-1-14 is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

The design basis accidents and anticipated operational transients of UFSAR Chapter 15 have been reviewed with respect to the changes performed by this TACF. The specific design basis accident and operational transient considered are a steam generator tube rupture and a steam generator tube leak. No credit is taken in UFSAR Chapter 15 for operation of 0-FCV-41-141 valve to mitigate a steam generator tube rupture event. The condensate demineralizer system, of which this valve is a part, is designed to function during normal plant operation. Changing the 0-FCV-14-141 valve out to manual operation has no affect on operation of the condensate demineralizer system in support of the plant with a steam generator tube leak.

SA-SE Number: WBOTSS-98-079-0

Implementation Date: 04/15/1999

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF Number 1-98-008-006	Furmanite Repair of Steam Leak - Heater Drains and Vents System

Description and Safety Assessments:

This temporary alteration (TA) will close 1-PIPE-6-H at the lower pipe tee to 1-LS-6-85B,. MSR B-2 HP DRAIN TANK LOW LEVEL, switch. This work will be done in accordance with WO 97-014755-001 by injecting Furmanite compound upstream of the leak per Furmanite procedure N-97220

This TA will also configure the root valves to the subject level switch, (1-RTV-6-1652A and 1-RTV-6-1653A). in the closed position and the field wires at 1-LS-6-95B will be lifted, lugged, and taped-to prevent common abnormal level alarm window XA-55-2B-33A from erroneously alarming.

UFSAR section 10.4.10.2. describes the subject MSR drain tank level alarm function: "Low level alarm is also annunciated if the level drops below the normal control range". This TA defeats the subject alarm, conflicting with the reference.

The only type of accident that defeating this alarm could be associated with is "Minor Secondary System Pipe Breaks". This is only possible if: (1)a level control failure results in a steam/water flow condition in the drain pipe, and (2) the condition goes undetected long enough for flow accelerated corrosion to erode the pipe wall resulting in a through wall steam leak and subsequent pipe break. This 6 inch secondary pipe brake accident is bounded by the analysis of major secondary pipe breaks and does not require further analysis.

Operation of the normal and by pass level control valves are not affected by this TA. In the event of a failure, both the normal and by pass level control valves have valve position limit switches to give light indication in the MCR if the valve is either full open or full closed. Additionally there is a local sight glass to indicate drain tank level. If the normal LCV failed open due to a malfunction, steam flow would increase through the HP tube bundle and a steam and liquid mixture would pass through the drain line to the number 1 condensate heaters. The immediate adverse impact is principally a reduction in plant efficiency due to higher steam flow. The longer term degradation and failure potential is due to the affects of flow accelerated corrosion in the drain line. The increased flow condition would be detected by investigating valve position limit lights, monitoring the level sight glass or by system thermal performance monitoring within a shift to a few days.

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Implementation Date: 10/23/1998

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration Procedure	TACF Number 0-98-2-28 Revision 1 SOI-14.02, 14.03, 28.01, 1-PI-OPS-1-OS, AOI-7.01	Install new heads over the existing heads in the acid storage tank.

Description and Safety Assessments:

DCN W-39361-A will install new heads over the existing heads of the acid storage tank. In order to perform this task, the tank must be drained of its contents for an extended period of time. This tank supplies acid to the condensate demineralizer system which is in continuous operation while the unit is producing power. To enable the demineralizer system to remain in service, a temporary supply of acid is necessary. TACF 0-98-2-28, Revision 1, will provide for a tanker truck as a storage tank with a temporary pump and hoses which are routed into the Acid and Caustic Storage Building. (A later revision of this TACF added operator action to manually isolate the temporary pump from the tanker when acid flow is stopped to prevent gravity draining of the tanker.) A 90° elbow will be removed from the permanent acid piping and the temporary hose will be connected to the permanent pipe with a flange. The power cables to Acid Supply Pump B will be disconnected from the motor. A temporary jumper will be spliced to the existing cables to supply power to the temporary pump. During this time, the acid tank level instrumentation will be out of service. Level will need to be verified manually.

Revisions will be necessary for System Operating Instruction (SOI)-28.01, 14.02, and 14.03. The changes will reflect the fact that a tanker instead of the tank will be used to supply acid. They will also ensure the operator is aware that a temporary pump will be starting when the permanent pumps handswitch is manipulated. In addition, 1-PI-OPS-1-08 will need to delete the requirement to obtain acid level from the tank. Abnormal Operating Instruction (AOI)-7.01 will need to provide steps to remove the tanker from the area in the event of a maximum probable flood.

FSAR Sections 9.2.3 and 10.4.6 have been reviewed and the functional requirements of the condensate demineralizer system and the makeup water treatment plant have not changed. This TACF will temporarily affect 1-47W834-2 which is FSAR Figure 9.2-27. The only change to the drawing will be that there is a temporary tie in from a tanker to supply acid to the condensate demineralizer system instead of the acid storage tank. Section 10.4.6 of the FSAR specifically states that acid is supplied to the condensate demineralizer system from the acid storage tank. This tank will not supply acid for the duration of the TACF. However it should be emphasized that although the source of acid might have changed, the type of acid to be used, the amount to be used, and the reason for use have not changed.

The Watts Bar Safety Evaluation Report, (SER) and Supplements 1-20 have been evaluated for impacts. The change due to TACF 0-98-2-28 Revision 1 does not impact the NRC's understanding of the design and operation of Watts Bar Nuclear Plant as described in the FSAR.

The design basis accidents and anticipated operational transients of FSAR Chapter 15 have been reviewed with respect to the changes performed by TACF 0-98-2-23 Revision 1. The specific design basis accident and operational transient considered are a steam generator tube rupture and a steam generator tube leak. No credit is taken in FSAR Chapter 15 or in Design Basis Events Design Criteria, WB-DC-40-64, for operation of the acid storage tank and associated piping to mitigate a steam generator tube rupture event. The makeup water treatment system (the portion which contains the acid storage tank) and the condensate demineralizer system are designed to function during normal plant operation with a steam generator tube leak. Changing the source of acid has no affect on operation of the condensate demineralizer system in

SA-SE Number: WBOTSS-98-087-3 ***Implementation Date:*** 10/23/1998

support of the plant with a steam generator tube leak. Other design basis events considered were tornadoes, floods, and earthquakes. It is conceivable that a tornado could cause the tanker to destroy a portion of the switchyard. However, the switchyard was not built to withstand a tornado so there would be no added destruction due to a tanker entering the switchyard at that time. In the event that prior notice of an event such as a 100 year flood is given, provisions can be made to remove the tanker from the immediate area.

This TACF does not affect any FSAR Chapter 15 fault or operational transient evaluations. The acid storage tank and associated piping do not perform any safety-related functions and are not required for orderly shutdown of the reactor. Therefore, the probability of occurrence or consequences resulting from a previously evaluated accident or equipment malfunction is not increased. In addition, since the acid storage tank and associated piping do not have the potential to adversely impact any other safety-related equipment, the possibility for an accident or equipment malfunction of a type different than any other evaluated previously in the FSAR is not increased. Since the Technical Specifications (TSs) do not address either the condensate demineralizer system or the makeup water treatment plant and the changes of TACF 0-98-2-28, Revision 1, do not have the potential to indirectly affect TS systems, the safety margins defined in the TS Bases are unaffected. Therefore, on the basis of the evaluation of effects, it is concluded that TACF 0-98-2-28, Revision 1, and any associated procedure changes are acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number: WBOTSS-98-193-0

Implementation Date: 12/13/1998

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-98-18-61, Revision 0 Procedure ARI-138-144 Revision 4	Ice Condenser Inlet door Circuit MCR Annunciation Alarm

Description and Safety Assessments:

The Ice Condenser Inlet Door Position Monitoring System involves six (6) sets of status indication lights (green and red) located in the main control room (MCR) on panel 1-M-10. These status lights provide inlet door position information. Each green/red status light set monitors a zone. There are six zones (A through F) with eight doors in each zone for a total of 48 doors. Inlet door mounted zone switches provide the necessary switching action (i. e. contact action - open/closed) to complete the circuit and illuminate the appropriate status light. The zone switch contacts are configured for each status light set to illuminate a green light if all 8 doors are closed (the 8 "closed" contacts are wired in a series configuration) and to illuminate a red light if any of the 8 doors are open (the 8 "open" contacts are wired in a parallel configuration).

The MCR Inlet Door Annunciation Circuit consists of a parallel contact arrangement of all 48 inlet door zone switches. This Inlet Door Annunciation Circuit and the above described Inlet Door Position Monitoring System uses separate inlet door zone switches and interrogation supply power. With all inlet doors closed, the zone switch contacts are open, thus, creating an equivalent open circuit. If one or more inlet door zone switch(s) detect an open door, the associated contact(s) closes, thus providing a closed circuit (electrical current flow) and a corresponding annunciation in the MCR (window number 144-A).

This is a change to the equipment involved with providing the MCR Inlet Door Annunciation Circuit providing a MCR alarm for an open Ice Condenser Lower Inlet Door. Rather than the annunciator system monitoring the 48 parallel connected door limit switches directly, a 24V dc relay will be interposed between the annunciator input and the limit switches. This relay will be powered by a power supply separate from the annunciator. Failure of this power supply will be alarmed. The circuitry inside containment is experiencing some low resistance values that is causing the annunciator to alarm falsely. This failure appears to be the result of cable degradation between the sense wires and the cable shield which is causing current leakage paths resulting in false RONAN alarms. By isolating the leakage paths in the sensing circuit from RONAN, the change will allow the MCR alarm to function properly.

Two monitors are installed to alarm the opening of ice condenser lower inlet doors. These monitors aid Operations in assuring the lower inlet doors remain closed and ice bed integrity is maintained. The primary system is the inlet door monitoring system. The secondary system is the inlet door annunciator system. The annunciator system consists of 48 door limit switches which are wired such that opening of any door results in an alarm. The ice condenser door position annunciator system has caused false alarms due to low resistance between the sense circuit and the cable shielding inside containment. This situation has led to false indication in the Ronan annunciator system and has degraded the ability to perform channel operability confirmation on the lower inlet door monitoring system.

A change to the equipment involved with providing a MCR alarm for an open Ice Condenser Lower Inlet Door, is proposed. Rather than the annunciator monitoring the 48 inlet door limit switches directly, a 24V dc relay will be interposed between the annunciator input and the limit switches. This relay will be powered by a power supply separate from the annunciator. Failure of this power supply will be alarmed. This change will allow the MCR alarm to function properly. By restoring the functionality of the annunciator circuit, the system will continue to function as described in the UFSAR and the TRM bases and no unreviewed safety question will exist. This temporary change will be removed upon completion of maintenance to the wiring system at the refueling outage.

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

Implementation Date: 04/01/1999

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF Number 1-99-001-081 Revision 0	Provide Temporary Demineralized water supply to the Primary Water System.

Description and Safety Assessments:

A temporary Demineralized Water supply to the Primary Water System will be provided by this TACF. The Demineralized Water supply is required to ensure that the Primary Water System is capable of supplying required Primary Water loads during the second refueling outage while the Primary Water Storage Tank is out-of-service for maintenance. The anticipated Primary Water loads are listed below:

- Waste Gas Compressor A - 5 gpm, intermittent flow.
- Waste Gas Compressor B - 5 gpm. intermittent flow.
- Bed Transfer - 40 gpm, intermittent flow.
- Makeup to the Reactor Coolant System - 70 gpm.

The required maximum flow of Demineralized Water to the Primary Water System to approximately 120 gpm. A hose will be routed from the Demineralized Water flanged connection near valve 1-ISV-59-519 to the Primary Water connection near 1-DRV-81-536. The hose shall be 4 inch or larger and constructed of rubber with a pressure rating of 125 psi or greater and a temperature, rating of 85°F or greater. The required length of the hose is approximately 200 feet. The hose shall be routed out of the main thoroughfare to prevent a tripping hazard to plant personnel and secured to prevent falling on plant equipment/personnel. The hose should be routed on the floor as much as possible. When routed overhead, the hose shall be sufficiently supported to prevent sagging. Maintenance & Modification Department Procedure (MMDP) 2 and Site Standard Practice (SSP) 7.04 provide requirements for rigging of the hose. The Primary Water connection will be made using the installed connector at valve 1-DRV-81-536. Valves 1-DRV-81-536 and 1-ISV-59-519 must be opened to provide flow to the Primary Water System. Back leakage of Primary Water into the Demineralized Water System was evaluated. It was determined that the pressure head provided by the Demineralized Water Tanks is greater than that of any of the Primary Water loads. The Demineralized Water Tanks provide a constant pressure head on the Demineralized Water System and Primary Water System. The Demineralized Water Tanks are designed with low level annunciation to warn the Operators of low tank levels and thus prevent emptying the tanks. The accident scenarios for Modes 5 and 6 include Fuel Handling and Residual Host Removal (RHR). Installation of the temporary hose does not interface with Fuel Handling or RHR equipment nor does it affect equipment necessary for Fuel Handling and RHR operation.

This change provides a temporary water supply to the Primary Water System. The water supply is required to ensure that Primary Water loads receive water during the second refueling outage while the Primary Water Storage Tank is out-of-service for maintenance. The temporary water supply will be provided from the Demineralized Water System. The required flow of Demineralized Water to the Primary Water System is approximately 120 gpm maximum. The flow is intermittent. A rubber hose will be routed from the Demineralized Water System to a connection in the Primary Water System. Thus, a dependable supply of water to the Primary Water System will be provided.

The above described change will occur while the plant is shutdown for the second refueling outage (Modes 5 and/or 6) and will not be in affect after the plant is in operation. The change does not affect safety-related equipment or place safety-related equipment in a condition that is adverse to safety. This change does not increase the probability of or consequences of an accident nor does it introduce different types of accidents. This change does not affect equipment that is defined in the Technical Specifications. Thus this change does not constitute a unreviewed safety question.

SA-SE Number: WBOTSS-99-043-0

Implementation Date: 03/11/1999

Document Type:	***Affected Documents:***	***Title:***
TACF	TACF Number 1-99-4-215 Revision 0	Blocked fire barrier door for temporary power cables.

Description and Safety Assessments:

This change (TACF 1-99-4-215) installs temporary power cables from 480 volt power outlets 1-PO-215-1 and 1-PO-215-2 to O-BKR-39-37 and 1-DXF-215-A respectively. It also installs a temporary power cable between 1-BKR-215-A001/5F2-A and 2-BKR-215-A001/5F2-A. One of the temporary cables installed under this change blocks fire barrier door D024 open between Diesel Generator Building 480 Volt Board Rooms 1A and 2A. Without having the capability to close these doors, the design $C0_2$ concentrations versus time may not be achieved per Section 12.3.3 of the Fire Protection Report. The purpose of this change is to provide power to the Diesel Generator Building C02 Refrigeration Unit and Lighting Cabinet 45 during an outage on the 1A-A 480 Volt Diesel Auxiliary Boards. Failure to maintain power on the Diesel Generator Building $C0_2$ Refrigeration Unit would lead to increases in temperatures and pressures in the associated $C0_2$ storage tank which has the potential to cause the tank's relief valve to lift. Implementation of this change insures $C0_2$ inventory is maintained and reduces the hazard to personnel in the area of a potential release. Maintaining power to Lighting Cabinet 45 is required to support outage work activities, fire protection, plant security, flood protection, and personnel safety. The affected loads are electrical receptacles and lighting in the 1 A-A Diesel Building rooms fire protection piping heat trace circuits, the sump pump in safety related manhole 25, the Diesel Building $C0_2$ evacuation alarm strobe lights, the intrusion door alarm unit D 18 signal lights and relay, and other miscellaneous loads.

There are no design basis accidents that may be affected by this temporary change because the temporary alteration is in place concurrent to an outage on the 1A Shutdown Board. In accordance with Technical Specification 3.8.2 and 3.9.5, this change will only be in place with the reactor vessel de-fueled or with the plant in Mode 6 with the water level ≥ 23 feet above the reactor vessel flange and required B-train equipment must be operable. This configuration is in place with A-train equipment inoperable but, with an operable B-train bus, does not exceed the Technical Specification operability requirements before, during and after implementation and the credible failure modes of the temporary change are the same as before the change.

Providing temporary power from 2-MCC-215-A001 -A to loads normally supplied from 1-MCC-215-A001-A does not involve an unreviewed safety question. Only A-train shutdown power is affected by the change and B-train shutdown power remains Technical Specification operable before, during and after the change.

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

Implementation Date: 02/13/1999

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF Number 1-99-6-2 Revision 0	Gland Steam Condenser Drain System temporary pump.

Description and Safety Assessments:

This temporary alteration (TA) modifies the drain system for the Gland Steam Condenser. There is a suspected tube leak in the condenser that is outrunning the capacity of the as-designed gravity drain system causing the condenser to flood up and lose vacuum. The loss of vacuum itself in the Gland Steam Condenser is not detrimental to plant operations; however, the effects are. The effects are seen at the LP Turbine seals where steam is blowing to the atmosphere. This blowing steam is drawn into the turbine rotor bearing housings which are under a slight negative pressure, resulting in water accumulation in the turbine bearing oil. To reestablish Gland Steam Condenser vacuum, a temporary pump is connected to the Gland Steam Condenser drain system to increase the drain flow. In order to maintain a water solid suction to the pump, the normal drain flow path is isolated and a water seal is formed at the drain overflow discharge line. These temporary modifications have no effect on the design basis accidents evaluated in the UFSAR. There is no effect on any design basis credible failure modes. There are no new system functions introduced by this TA. No unmonitored release paths are created by this TA. Therefore, this TA does not constitute a unreviewed safety question.

SA-SE Number: WBPTSS-99-051-0

Implementation Date: 02/27/1999

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration Change Form (TACF)	TACF 0-98-005-024, Revision 1	Temporary Water Supply to the Auxiliary Boiler Blowdown Tank

Description and Safety Assessments:

This TACF provides a temporary water supply to the Auxiliary Boiler Blowdown Tank. The water supply is required to ensure that the Auxiliary Boiler is operable during the second refueling outage while the Raw Cooling Water (RCW) system is out-of-service for various modifications/maintenance. The temporary water supply will be provided from the Raw Service Water (RSW) System. The required flow of RSW to the Auxiliary Boiler Blowdown Tank is 30 gpm. The flow is intermittent. A hose will be routed from 0-DRV-25-574 to a pipe nipple in the Auxiliary Boiler Blowdown Tank supply piping. The supplier of water to the RSW system is the Fire Protection system which will remain in service during the second refueling outage. Thus, a dependable supply of water to the Auxiliary Boiler Blowdown Tank will be provided. RSW will be isolated from the Fire Protection System in the event of a fire and thus will not add unanalyzed loads to the Fire Protection System during a fire.

The above described change will occur while the plant is shutdown for the second refueling outage and will not be in affect after the plant is in operation. The change does not affect safety-related equipment or place safety-related equipment in a condition that is adverse to safety. This change does not increase the probability of or consequences of an accident nor does it introduce different types of accidents. This change does not affect equipment that is defined in the Technical Specifications. Therefore, this change does not constituted a unreviewed safety question.

SA-SE Number: WBOTSS-99-053-0

Implementation Date: 03/03/1999

Document Type:	***Affected Documents:***	***Title:***
Work Order (WO)	WO 99-002592-000	Manipulation of a Damper on Train A of the Main Control Room (MCR) Air Handling Unit (AHU)

Description and Safety Assessments:

This WO fails open the damper on the suction duct of Train A of the Main Control Room (MCR) Air Handling Unit (AHU) by closing the control air supply to the damper's actuator. This damper will be failed in this position during the transition time for placement of Train B of the MCR Heating, Ventilating, and Air-conditioning (HVAC) system in service. The opening of the damper will ensure the operability of the Train A system during the transition period.

This action is necessary because the damper on the suction duct of Train A was found to be only able to open to 45 degrees open with the Train A AHU in service during the performance of 0-SI-31-31-A, "Control Room Emergency Air Temperature Control System (CREATCS) Train A Operability Test." CREATCS is required to maintain temperatures in the Main Control Room Habitability Zone (MCRHZ) during operating Modes 1-6 and during movement of irradiated fuel assembles. The opening of the damper will ensure that Train A of the MCR HVAC system will function, if Train B of the system fails to maintain temperatures below 104 degrees F. Exceeding 104 degrees in the MCRHZ may require evacuation of the Control Room.

This activity does not affect any of the FSAR evaluations (accident analysis or equipment malfunction failures) previously performed, since the CREATCS system does not have any affect on components or areas involved in the initiating of accidents describe in the FSAR. No new accidents or equipment malfunction failures are created. This activity ensures that the required heat removed capacity as discussed in the Technical Specification will be available if required. Therefore, this evaluation concludes that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number: WBO-WBD-98-006-0

Implementation Date: 03/30/1999

Document Type:	*Affected Documents:*	*Title:*
FSAR Figure	FSAR Figure 13.5-1 FSAR Change Package Number 1542	Updated FSAR Review - Section 13.5

Description and Safety Assessments:

This is a safety assessment for the revision to FSAR Section 13.5, Plant Instructions, which is a complete revision to standardize toward Sequoyah Nuclear Plant's FSAR 13.5, Site Instructions and Section Instruction Letters.

Figure 13.5-1, System of written procedures has been revised to reflect the procedure types due to the new procedure hierarchy. Figure 13.5-2, Unit 1 Control Room Operating Area, has been revised due to differences between the FSAR Figure and a Figure in SSP-12.01, Conduct of Operations. The SSP Figure replaced the FSAR Figure.

This change is an administrative change only which will not increase the probability of an accident previously evaluated in the FSAR nor create a different type of accident. The probability of occurrence of a malfunction nor a malfunction of a different type of equipment important to safety has not been increased by the administrative changes. The administrative changes do not increase the consequences of an accident nor decrease the margin of safety. Therefore, these administrative changes do not constitute an unreviewed safety questions.

SA-SE Number: WBO-WBD-98-007-0

Implementation Date: 03/30/1999

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Section 13.6 FSAR Change Package Number 1543	Updated FSAR Review - Section 13.6

Description and Safety Assessments:

This is a safety assessment for the revision to FSAR Section 13.6, Plant Records, which is a complete revision to standardize toward Sequoyah Nuclear Plant's FSAR 13.6, Plant Records.

There will be no work to t he plant or systems due to this revision and there are no Design Change Notices (DCN's) involved.

This change is an administrative change only which will not increase the probability of an accident previously evaluated in the FSAR nor create a different type of accident. The probability of occurrence of a malfunction nor a malfunction of a different type of equipment important to safety has not been increased by the administrative changes. The administrative changes do not increase the consequences of an accident nor decrease the margin of safety. Therefore, these administrative changes do not constitute an unreviewed safety questions.

SA-SE Number: WBPLCE-98-017-1

Implementation Date: 04/16/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number M-39816-B	Supplemental Condenser Circulating Water System.

Description and Safety Assessments:

The Supplemental Condenser Circulating Water (SCCW) system is being added as a sub-system of the existing CCW system (reference WB-DC-40-27). As the name implies, the SCCW system provides supplemental cooling water to the CCW. The supplemental cooling water is cooler than the return CCW water from the Unit 1 cooling tower. The cooler water improves the thermal efficiency of the Unit 1 condenser and therefore the plant performance. Annual benefits from this improved efficiency will be approximately 65,000 Mega-Watt Hours of electrical energy.

The new SCCW system will operated as a passive system providing a continuous supply of supplemental cooling water to the Unit 2 cooling tower basin and flume, except when the system is shutdown during refueling outages or for modification/maintenance. With only the Unit 1 CCW pumps operating, the SCCW flow passes through a new weir/stop log in the common wall between the Unit 1 and Unit 2 flumes. Then the SCCW flow mixes with the Unit 1 tower CCW return flow and enters the condenser via the CCW pumps and supply conduits. The mixed flow returns via the CCW return conduits to the Unit 1 cooling tower. Approximately the same SCCW system flow rate is returned to the Watts Bar Fossil (WBF) discharge channel through the 78 inch SCCW discharge line and energy dissipator structure. During winter operation approximately 40% of the supply line flow is diverted through a 42 inch bypass line to satisfy environmental permit requirements for the WBF channel discharge.

The flow through the SCCW system is set by appropriately throttling the 90 inch supply line flow to the Unit 2 cooling tower via a new SCCW supply line butterfly valve (WBN-0-FCV-027-0112). The flow is set at 135,000 gpm with the Watts Bar lake level at nominal summer pool (elevation 740.5 feet). With the supply valve remaining in that position the flow under flooding conditions (Watts Bar lake elevation 745.0 feet) be approximately 166,000 gpm. Substantially lower flow will occur in the winter months when the Watts Bar lake level approaches elevation 735.0 feet, The 166,000 gpm flood flow rate is used as a limiting condition in the functional and structural design basis of the SCCW system. The supply line butterfly valve is open to the throttled position when the SCCW system is operating and it is fully closed when the system is not operating.

Flow through the new bypass line during winter operation is throttled by a 42 inch bypass line butterfly valve (WBN-0-FCV-027-01 10) to divert approximately 40% of the supply line flow to the discharge line, without passing through the condensers. This mitigates the temperature differentials between the river and the discharge flow to the river during winter operation. The bypass valve is fully closed during summer operation. It is open to the throttled condition during winter operation. When the SCCW system is not operating but the system is pressurized upstream of the supply line valve, the bypass valve is fully closed.

Water level in the Unit 1 and Unit 2 cooling tower basins is maintained at an appropriate level to assure that adequate net positive suction head is maintained at the CCW pumps. This is accomplished by new SCCW weirs at the inlet to the Unit 2 basin and the outlet to the Unit 1 tower basin. The existing blowdown weirs are lowered by about six inches to ensure adequate blowdown flow during winter operation when the SCCW flow to the towers is low. During summer operation the bypass valve is closed and the SCCW flow to the towers is between 135,000 gpm and 166,000 gpm. Excess blowdown flow during summer operation is avoided by closing a new sluice gate at the Unit 2 blowdown line entrance. This gate is closed and opened each spring and fall at the same time the by-pass valves are closed and opened. There is also a stop log between at the flume wall weir which can be closed for cleaning and desilting operations of each tower basin, separately.

FSAR Change Request 1517 has been prepared to update sections 1.2.2 and 10.4.5 of the FSAR as required to describe this modification of the CCW system.

SA-SE Number: WBPLCE-98-017-1 ***Implementation Date:*** 04/16/1999

Failures of the CCW system or SCCW sub-system in any way associated with this design change systems do not contribute to or initiate any of the accident scenarios in the SAR; therefore, this modification will not increase the probability of occurrence or the consequences of an accident or malfunction previously evaluated in the FSAR or create the possibility of an accident or a malfunction of a different type from those previously evaluated in the FSAR. This modification to these systems does not reduce the margin of safety in the basis for any Technical Specification.

SA-SE Number: WBPLCE-98-020-0

Implementation Date: 01/12/1999

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Change Package 1532	UFSAR Section 3.8 Revision

Description and Safety Assessments:

The following are specific UFSAR sections/tables of Section 3.8 which are being revised by Change Package Number 1532. (Note: Page number references are approximate UFSAR page numbers which are to aid in locating the discrepancy. Sequential alpha-numeric are for reference purposes only in this Safety Evaluation. T= Typographical Changes, C = Clarifications, E = Editorial).

			Typographical Changes	
	Section	Page	Current FSAR	Changes for Update
1T	Section 3.8. 1.1	3.8-1	Value of 155 feet should be 150 feet	Correct value to approximately 150 feet
2T	Section 3.8.2.4.8	3.8.2-15	Word "shall" should be "shell"	Change the word to "shell"
3T	Section 3.8.2.4.6	3.8.2-16	Word "shall" should be "shell"	Change the word to "shell"
4T	Section 3.8.3.1.4	3.8.3-4	The Azimuth location of 114' should not have units of feet and inches	Correct units to minutes and seconds
5T	Section 3.8.3.1.10	3.8.3-8	Azimuth 350° should be 305°	Correct azimuth to 305°
6T	Section 3.8.4.1.3	3.8.4-13	Erroneous reference to a term "tornado Category I"	Delete the reference to "tornado Category I"
7T	Section 3.8.4.2.1	3.8.4-20	ASTM 1975	Correct to ASTM 1971
8T	Section 3.8.4.3.1	3.8.4-25	1 psi	1.25 psi
9T	Section 3.8.4.4.1	3.8.4-27	percentage of .001	percentage of .1
10T	Section 3.8.4.4.1	3.8.4-29	Incorrect term in eqn.	Correct eqn. By deleting term
11T	Section 3.8.4.4.1	3.8.4-29	Incorrect reference to FSAR Section	Delete reference to FSAR Section
12T	Section 3.8.4.4.1	3.8.4-29	Flued head typographical error	Correct typographical error
13T	Section 3.8.4.4.1	3.8.4-30	typographical error "slat"	correct error to "slab"
14T	Section 3.8.5.1.1	3.8.5-1	The word screens should be screeds	Correct word to screed
15T	Section 3.8.5.1.2	3.8.5-2	Elevation 723.25 should be 725.25	Correct elevation to 725.25
16T	Table 3.8.1-2	Page 2	Typographical error make K and Cc uppercase	Revise K and Cc to be uppercase
17T	Table 3.8.3-3	Page 4	Typographical error make K and Cc uppercase	Revise K and Cc to be uppercase
18T	Table 3.8.3-7	Page 2	Typographical error make K and Cc uppercase	Revise K and Cc to be uppercase
19T	Table 3.8.4-01	Page 2	Typographical error load case IA and lb. Duplicate load case V	Revise load case IA to la Revise load case lb
20T	Table 3.8.4-01	Page 3	Load case V referenced	Delete reference to load case V.
21T	Table 3.8.4-02	Pages 2 and 3	Typographical error omitting eqn. Identification	Correct typographical error by adding eqn identification.

SA-SE Number: WBPLCE-98-020-0 **Implementation Date:** 01/12/1999

22T	Table 3.8.4-03	Page 1	Typographical error make K and Cc uppercase	Revise K and Cc to be uppercase
23T	Table 3.8.4-04	Page 2	Typographical error make K and Cc uppercase	Revise K and Cc to be uppercase
24T	Table 3.8.4-04	Page 1	Typographical error	Correct by adding Live Load value
25T	Table 3.8.4-05	Page 2	Typographical error make K and Cc uppercase	Revise K and Cc to be uppercase
26T	Table 3.8.4-06	Page 1	Typographical error make K and Cc uppercase	Revise K and Cc to be uppercase
27T	Table 3.8.4-07	Pages 1 through 5	Typographical error make K and Cc uppercase	Revise K and Cc to be uppercase
28T	Table 3.8.4-07	Page 1	Typographical error column heading	Correct "No."
29T	Table 3.8.4-11	Page 2	Typographical error make K and Cc uppercase.	Revise K and Cc to be uppercase
30T	Table 3.8.4-13	Page 2	Typographical error make K and Cc uppercase	Revise K and Cc to be uppercase
31T	Table 3.8.4-17	Page 1	Typographical error in Load Case VII	Delete one of the duplicated "L" terms.
32T	Table 3.8.4-20	Page 1	Typographical error omitting "Foundation" from the table title	Revise to add "Foundation" to the title.
33T	Table 3.8.4-22	Page 4	Typographical error	Correct "S" allowable load term
34T	Table 3.8.4-23	Page 1	Typographical error make K and Cc uppercase	Revise K and Cc to be uppercase
35T	Appendix 3.8B	3.8B-2	The word "an" should be "and"	Correct the word to "and"
36T	Appendix 3.8B	3.8B-4	The word "be" should be the letter "b." Capitalize "Z" for length constant.	Correct the word to the letter "b." Use capital "Z" for length constant.
37T	Appendix 3.8B	3.8B-4	The second equation from the bottom of the page should have a "less than" sign	Correct signage
38T	Appendix 3.8B	3.8B-8	Heading "PHERICAL SHELLS" should be SPHERICAL SHELLS	Correct the word to Spherical
39T	Appendix 3.8C	3.8C-4	The first symbol μ_θ should be μ_ϕ.	Correct the symbol.
40T	Appendix 3.8E; Section 3.8.4	3.8E-7	Equation (4) and (10) have the letter designation "N" when it should be E'	Correct letter designation to E'
41T	Appendix 3.8E; Table 3.8E-1	Page 1	Formula A has the effective length factor, K, and length, I, multiplied by the radius of gyration, r, when KI should be divided by r	Correct formula
42T	Appendix 3.8E; Table 3.8E-1	Page 1	Formula B has π raised to the 2E power when it should be π^2 multiplied by E. Delete semicolon after "I."	Correct formula

SA-SE Number: WBPLCE-98-020-0 **Implementation Date:** 01/12/1999

43T	Appendix 3.8E; Table 3.8E-1	Page 2	Misspelled word "the." Actual unbraced length should be represented by "I" rather than "R"	Correct spelling and change "R" to "I."
44T	Section 3.8.4.1.2	3.8.3-12	Space in text	Delete space in text
45T	Appendix 3.8B	3.8B-4, -5	Some formulas have italic text instead of regular text as the other formulas.	Revise to make text normal and not italics.
All of the above typographical errors (1T through 45T) were reviewed and do not change any results. These changes are minor and do not affect the outcome of any evaluations.				

	The following sections are being revised to update the FSAR for Clarification and Consistency.			
	Section	Page	Current FSAR	Changes for Update
1C	Section 3.8.1.3	3.8-6	Heading for Snow should be the heading for Live Load with snow load included as a part of the definition.	Change Heading to show Live Load. Revised to match design criteria as reviewed and accepted by the DBVP.
2C	Section 3.8.1.4	3.8-11	Values were never updated to reflect values in calculation.	Update values to current calculation values.
3C	Section 3.8.2.3.2	3.8.2-10 through -12	Loading combinations do not match WB-DC-20-3.	Revise load combinations to match WB-DC-20-3 as reviewed and accepted by the DBVP.
4C	Section 3.8.2.6.1	3.8.2-18	For Austenitic stainless steel the Grade should be F304 or 316.	Add 316. Revised to match design criteria as reviewed and accepted by the DBVP.
5C	Section 3.8.2.6.1	3.8.2-19	For Austenitic stainless steel there should be an "or" after WP 316.	Add "or" to text.
6C	Figure 3.8.2-10		"TMD NODAL VLUMES" ID do not match WB-DC-20-3.	Revise to match WB-DC-20-3 as reviewed and accepted by the DBVP.
7C	Section 3.8.3.1.4	3.8.3-3	For the field-splices of seals, a cold bond overlay may also be used. This is reflected in the design criteria.	Revise to reflect acceptability of "cold bond overlay" as stated in the design criteria as reviewed and accepted by the DBVP.
8C	Section 3.8.3.6.3 Section 3.8.3.6.7	3.8.3-38 3.8.3-41	The sub-sections "Seals Between Upper and Lower Compartments" and "Escape Hatches in Elevation 756.63 Floor" should not be located under the section "Construction Techniques"	They should be located under section 3.8.3.7 "Testing and Inservice Surveillance Requirements."
9C	Section 3.8.4	3.8.4-1	South Steam Valve Room omitted	Add South Steam Valve Room for clarification
10C	Section 3.8.4.1	3.8.4-2	"...which are necessary to the two reactor units"	"...which are necessary for the one reactor unit."

SA-SE Number: WBPLCE-98-020-0 ***Implementation Date:*** 01/12/1999

	The following sections are being revised to update the FSAR for Clarification and Consistency.			
	Section	Page	Current FSAR	Changes for Update
11C	Section 3.8.4.1	3.8.4-3	Watertight Equipment Hatch Covers are provided with administratively controlled locking system which prevents removal of the covers during the plant operation.	Delete the administratively controlled locking requirement since the hatches are adequately locked using redundant bolting attachments.
12C	Section 3.8.4.1	3.8.4-5	The railroad access door is not identified as an ABSCE Boundary.	Identify the railroad access door as an ABSCE Boundary.
13C	Section 3.8.4.1	3.8.4-7, -8, -10	Unit 2 doors are still referenced.	Delete reference to Unit 2 doors which do not have any common functions for Unit 1.
14C	Section 3.8.4.1.2	3.8.4-13	Diesel Generator Building Door and Bulkhead Design applicable to SQN.	Revise to reflect WBN design which resists tornado loading by the concrete bulkhead.
15C	Section 3.8.4.1.5	3.8.4-15	The expansion joint between the North Steam Valve Room and Shield Building is incorrectly shown to be 1-inch.	Change the joint to a 2-inch expansion joint which is the actual expansion joint size.
16C	Section 3.8.4.3.1	3.8.4-25	Clarify Masonry walls as reinforced masonry walls	Reinforced masonry walls restricted to a maximum of 20 psf on one face.
17C	Section 3.8.4.3.2	3.8.4-25	Flood water.	Change to hydrostatic pressure.
18C	Section 3.8.4.4.1	3.8.4-33	Waste Packaging Structure design and analysis procedure not current	Revise Waste Packaging Structure design and analysis procedure to make current. The current design and analysis were reviewed and accepted by the DBVP.
19C	Section 3.8.4.4.1	3.8.4-29	100 psf tornado pressure differential on roof of refueling room.	Revise to explain tornado pressure differential was evaluated and remained within allowable stress limits.
20C	Section 3.8.4.4.1	3.8.4-33	"watertight"	Change to "near watertight" to be consistent with the actual case.
21C	Section 3.8.4.4.2	3.8.4-34	Diesel Generator Building Door and Bulkhead Design applicable to SQN.	Revise to reflect WBN design which resists tornado loading by the concrete bulkheads.
22C	Section 3.8.4.4.5	3.8.4-36	NSVR Blow off roof description is out of date.	Update NSVR Blow off roof description to match updated configuration for clarification.
23C	Section 3.8.4.7.2	3.8.4-44	Inspection requirements for equipment hatch locks no longer required.	Delete inspection requirements to be consistent with Section 3.8.4.1.

SA-SE Number: WBPLCE-98-020-0 **Implementation Date:** 01/12/1999

	The following sections are being revised to update the FSAR for Clarification and Consistency.			
	Section	Page	Current FSAR	Changes for Update
24C	Section 3.8.5.5.2	3.8.5-6	The maximum allowable uniform bearing pressure and a maximum allowable pressure within 70% or more of the base in compression for the Waste Packaging Structure does not agree with calculation. The Diesel Generator Building values for the maximum allowable uniform bearing pressure and a maximum allowable pressure with 70% or more of the base in compression does not agree with design criteria.	Revise to match the design criteria.
25C	Table 3.8.1-1	Page 1	Load Combination not consistent with Design Criteria.	Revise to match current design criteria as reviewed and accepted by the DBVP.
26C	Table 3.8.1-1	Page 2	Loading Combination 10 same as Load Case 1.	Eliminate Load Case 10 which is a duplicate load case.
27C	Table 3.8.3-3		Value of 8.7×10^7 rads total for 12 hours is a value for the seal at a higher elevation.	Change to state the following: Normal Operating Radiation is 2×10^7 rads for 40 year life; Accidents with jet and/or missile is 4.8×10^8 rads per hour (gamma) and 2.5×10^7 rads per hour (beta). Presray seals are qualified for a value of 1×10^8 rads total. Presray seal qualification bounds the Environmental conditions and the 8.7×10^7 rads total for 12 hours value.
28C	Table 3.8.4-01	Page 1	Definition for D' unclear	Revise definition to clarify hydrostatic pressure is from groundwater.
29C	Table 3.8.4-02	Page 1	Live Load and Dead Load values need to be updated.	Added updated values for Live Load and Dead Load.
30C	Table 3.8.4-07	Page 1	Table refers to Unit 2 Doors	Delete Unit 2 Doors
31C	Table 3.8.4-07	Page 2	Table refers to Unit 2 Doors	Delete Unit 2 Doors
32C	Table 3.8.4-07	Page 2	Clarify "All Doors"	Reference new Table 3.8.4-7a listing doors.
33C	Table 3.8.4-07	Page 5	Table refers to Unit 2 Doors	Delete Unit 2 Doors.
34C	Table 3.8.4-07	Page 1	Clarify "All Doors"	Reference new Table 3.8.4-7a listing doors.
35C	Table 3.8.4-07a	Page 1	New Table	Add new table listing doors.

SA-SE Number: WBPLCE-98-020-0 **Implementation Date:** 01/12/1999

	The following sections are being revised to update the FSAR for Clarification and Consistency.			
	Section	Page	Current FSAR	Changes for Update
36C	Table 3.8.4-08	Page 1	Clarify scope of load cases 11a and 111.	Add clarification on pump wells being full.
37C	Table 3.8.4-08	Page 1	It's not clear if the "factors of safety" are "required" or "calculated"	Clarify heading to state factors of safety are "calculated."
38C	Table 3.8.4-08	Page 1	There are no calculated Overturning and Sliding values for load case IV	Provide calculated values for Overturning and Sliding, Load Case IV
39C	Table 3.8.4-10	Page 1	Allowable stresses are reduced	Clarify reason for reducing allowable stresses
40C	Table 3.8.4-11	Page 1	Load Case V water level is not correct	Change water level to 736.9 to match latest flood levels.
41C	Table 3.8.4-12	Page 2	Normal stresses reference ACI Code 318-63 only	Revise footnote
42C	Table 3.8.4-13	Page 1	Load Cases II and III for Structural Parts do not specify if for door opened or closed as does the Mechanical Parts	Specify Load Cases II and III for Structural Parts is for the door closed position to be consistent with the Mechanical Parts Case.
43C	Table 3.8.4-16	Page 1	Load Case III needs to be clarified to include "concentrated surcharge where applicable."	Revise Load Case III to include "concentrated surcharge where applicable" to match design criteria as reviewed and accepted by the DBVP.
44C	Table 3.8.4-16	Page 1	Load Case VI needs to be clarified for Tornado Wind loading.	Revise Load Case VI to specify Tornado Load Combinations and Allowable Stresses.
45C	Table 3.8.4-16	Page 2	Load Case III needs to be clarified for Tornado Wind loading.	Revise Load Case III to specify loading combinations for tornado wind.
46C	Table 3.8.4-19	Page 1	Tornado wind needs to be clarified.	Revise to clarify Tornado wind term by adding "wind and missiles and pressure differential as applicable" to match design criteria as reviewed and accepted by the DBVP.
47C	Table 3.8.4-19	Page 1	ERCW Standpipe Structure does not have a load case for normal wind.	Add Load Case IV for factored normal wind + dead load + live load as reviewed and accepted by the DBVP.
48C	Table 3.8.4-20	Page 1	Tornado wind needs to be clarified.	Revise to define tornado wind consisting of wind and missiles.

SA-SE Number: WBPLCE-98-020-0 **Implementation Date:** 01/12/1999

The following sections are being revised to update the FSAR for Clarification and Consistency.				
	Section	Page	Current FSAR	Changes for Update
49C	Table 3.8.4-22	Pages 4 and 5	All load cases not shown.	Add load cases which were not shown for consistency with design criteria as reviewed and accepted by DBVP.
50C	Table 3.8.4-22	Pages 1 and 2	Definition of load terms.	Add load term definitions for load cases that were not shown for consistency with design criteria as reviewed and accepted by the DBVP.
51C	Appendix 3.8B; Table 3.8B-1	Page 1	Load combinations do not match design criteria.	Correct load combinations to reflect design criteria and calculations performed as reviewed and accepted by the DBVP.
52C	Appendix 3.8B; Table 3.8B-1	Page 2	Load combinations do not match design criteria.	Correct load combinations to reflect design criteria and calculations performed as reviewed and accepted by the DBVP.
53C	Appendix 3.B; Table 3.8B-1	Page 3	Load combinations do not match design criteria.	Correct load combinations to reflect design criteria and calculations performed as reviewed and accepted by the DBVP.
54C	Appendix 3.8E; Section 3.8E.4	3.8.3-4	The text refers to "each" Reactor Building in reference to access doors in the Crane Wall.	Revise text to be consistent with a one unit plant and change "each" to "the" in reference to "the Reactor Building."
All of the above changes (1C through 55C were reviewed and do not change any results. These changes clarify and enhance consistency between the UFSAR and existing design criteria, calculations and other sections in the UFSAR.				

The following sections were modified due to editorial changes.			
	Section	FSAR Page	Changes for Update
1E	Section 3.8.1.3	3.8-7	Add "historical information" after construction load
2E	Section 3.8.1.4	3.8-10	Reference to section 3.7 for seismic and deleted unnecessary wording
3E	Section 3.8.1.6.3	3.8-15	Add "historical information" after construction load
4E	Section 3.8.2.3	3.8.2-8	For wind loads revise verb tense to past tense in second paragraph
5E	Section 3.8.2.6.1	3.8.2-19 and -20	Remove added space between words
6E	Section 3.8.2.6.1	3.8.2-20	Clarify first paragraph under "Fittings" the material type to be Presray Type. Second paragraph clarify that the installed seals are to be examined at 18 month interval.

SA-SE Number: WBPLCE-98-020-0 ***Implementation Date:*** 01/12/1999

The following sections were modified due to editorial changes.			
	Section	FSAR Page	Changes for Update
7E	Section 3.8.2.6.3	3.8.2-23	Remove duplicate information in Protective Coatings Section 6.1.4 on Paints and Coatings Inside Containment.
8E	Section 3.8.2.7.1	3.8.2-25	Add "Historical Information"
9E	Section 3.8.2.7.2	3.8.2-25	Add "Historical Information"
10E	Section 3.8.2.7.3	3.8.2-25	Add "Historical Information"
11E	Section 3.8.2.7.4	3.8.2-25	Add "Historical Information"
12E	Section 3.8.2.7.5	3.8.2-26	Add "Historical Information"
13E	Section 3.8.2.7.6	3.8.2-26	Add "Historical Information"
14E	Section 3.8.2.7.7	3.8.2-26	Add "Historical Information"
15E	Section 3.8.2.7.8	3.8.2-27	Add "Historical Information"
16E	Section 3.8.3.1.4	3.8.3-2	Title "Compartment Above Reactor" should be between 3rd and 4th paragraph. The 14-inch Reactor Cavity Bulkhead Wall should be under the section 3.8.2.1.3 Reactor Cavity Wall.
17E	Section 3.8.3.1.6	3.8.3-3	Add words "steel containment vessel" for clarification of SCV.
18E	Section 3.8.3.4.3	3.8.3-19	Add "Historical Information" after Independent Design
19E	Section 3.8.3.4.4	3.8.3-20	Add "Historical Information" after Independent Design
20E	Section 3.8.3.4.6	3.8.3-22	Add "Historical Information" after Independent Design
21E	Section 3.8.3.4.7	3.8.3-26	Add "Historical Information" after Independent Design
22E	Section 3.8.3.4.8	3.8.3-27	Add "Historical Information" after Independent Design
23E	Section 3.8.3.4.9	3.8.3-28	Add "Historical Information" after Independent Design
24E	Section 3.8.3.4.10	3.8.3-29	Add "Historical Information" after Independent Design
25E	Section 3.8.3.4.13	3.8.3-30, -31	Add "Historical Information" after Independent Design
26E	Section 3.8.3.6.2	3.8.3-37	Add "Historical Information" after Quality Control
27E	Section 3.8.3.6.3	3.8.3-38	Add "Historical Information" after Construction Technique
28E	Section 3.8.3.9	3.8.3-43	Remove section "Interface Control," Reg. Guide 1.70 does not require this section. This section explained how Westinghouse controlled it's design process for constructing the plant. Therefore, it is no longer needed.

SA-SE Number: WBPLCE-98-020-0 **Implementation Date:** 01/12/1999

The following sections were modified due to editorial changes.			
	Section	FSAR Page	Changes for Update
29E	Section 3.8.4	3.8.4-1	The "d" in the word "Associated was left off. Item 30 add "." and delete "This door…" for consistency Item 35 through 41 add "." and delete "This door…" for consistency.
30E	Section 3.8.4.1.1	3.8.4-8	Move Table 3.8.4-1 to same line.
31E	Section 3.8.4.1.1	3.8.9-9	Item 35 through 41 add "." and delete "This door…" for consistency.
32E	Section 3.8.4.1.1	3.8.4-11	Move Table 3.8.4-1 to same line.
33E	Section 3.8.4.1.2	3.8.4-14	Make "e" in Elevation lower case. Add reference to load cases in Table 3.8.4-1 to same line.
34E	Section 3.8.4.1.4	3.8.4-15	Add "1E" to modify manholes.
35E	Section 3.8.4.2.1	3.8.4-21	Revise Item 18 to correct NCIG-01 title.
36E	Section 3.8.4.2.1	3.8.4-20	Item 6, revise to put "Standards for…" on the same line.
37E	Section 3.8.4.2.1	3.8.4-21	Revise Item 17 to correct NFPA acronym.
38E	Section 3.8.4.2.1	3.8.4-21	Revise Item 14 to correct report names.
39E	Section 3.8.4.4.5	3.8.4-36	Revise section title deleting last "Pumping Station" in title.
40E	Section 3.8.4.6.2	3.8.4-44	Revise to delete "all" from "materials used for load-carrying members."
41E	Table 3.8.4-08	Page 1 and 2	Revise WSD Normal Concrete by adding reference to ACI.
42E	Table 3.8.4-08	Page 1	Load Case III has no factor of safety values. Add "N.A." for values because it is not a valid stability case per the design criteria as reviewed and accepted by the DBVP
43E	Table 3.8.4-08	Page 2	Add definition for ACI.
44E	Table 3.8.4-09	Page 1	Revise Load Case 1A to put fc in terms of fc'
45E	Table 3.8.4-10	Page 1	Revise WSD Normal Concrete by adding reference to ACI.
46E	Table 3.8.4-22	Pages 1 through 6	Revise page numbers due to added pages.
47E	Appendix 3.8A	Page 3.8A-1	Some words were inadvertently omitted from UFSAR. The following is how the paragraph should read: "The curve labeled shell adjacent to the ice compartment indicates the temperature of the shall adjacent to the ice compartment. **The shell is separated from the ice compartment** with a thick layer of insulation, hence the rather slow response for the temperature of the shell adjacent to the ice compartment. After the ice is all melted the temperature **inside the ice compartment will be the same as the temperature** in the lower compartment; however, the shall temperature adjacent to the ice **will always be less than the temperature in the ice compartment because of insulation. The temperature of the shell adjacent to the ice compartment will** peak at less than 220°F."

SA-SE Number: WBPLCE-98-020-0 ***Implementation Date:*** 01/12/1999

The following sections were modified due to editorial changes.			
	Section	FSAR Page	Changes for Update
48E	Appendix 3.8B Sections 3.2d, 3.3 and 5.0	3.8B-7 & 8	Reword and use past tense.
49E	Appendix 3.8D	3.8D-1	Add statement "Computer Programs used for structural analysis meet the TVA Quality Assurance Program for Computer Software. The following sections are for historical purposes." These programs were used in the structural design of the plant.
50E	Section 3.8.3.2	3.8.3-7	Remove the word "all." Does not change intent of sentence. Add "The following discussion is for historical purpose" before 3rd paragraph.
51E	Section 3.8.3.4.11	3.8.3-29	Add "Historical Information" after Independent Design.
All of the above changes (1E through 51E) were reviewed and do not change any results. These changes correct editorial content for consistency and correctness.			

The changes to the UFSAR section 3.8 can be characterized as minor changes. These changes were corrections of typographical errors, corrections of external references, clarifications of UFSAR test to avoid misinterpretation and/or correct minor misrepresentations, removal of duplicate information, removal and/or rewording of information with respect to Unit 2, corrections of obvious discrepancies between UFSAR sections and with other documents (design criteria, calculations, etc.), and corrections to conversion errors of files from WordPerfect to Microsoft Word (symbols, column alignment, etc.). Based on the NRC review of calculations during the IDI inspections (390/91-201, 390/92-201, 390/93-202) the DBVP inspections (390/93-66 and 390/94-69), and various other inspections, the NRC indicated that in general the Essential Calculation Program included the necessary calculations and had been adequately implemented. This evaluation was made, reviewed and accepted by the NRC prior to licensing. Some changes to the UFSAR constitute and update to those areas where changes were not incorporated in a timely manner but approved through the referenced program. These changes have not (1) increased the probability of occurrence or the consequences of an accident or malfunctions of equipment important to safety previously evaluated in the SAR, (2) created any different type of accident or malfunction previously evaluated, or (3) reduced any margin of safety as defined in the Technical Specification. There is no unreviewed safety question.

SA-SE Number: WBNLCE-98-021-0

Implementation Date: 10/06/1998

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Change Package 1529	Detailed description of the change, test, or experiment, including the design basis accident, and credible failure modes of activity (UFSAR Section 2.5).

Description and Safety Assessment:

This UFSAR change request is provided as part of the FY 98 FSAR review. Regulatory Guide 1.70, Revision 2, "Standard Format and Content of Safety Evaluation Reports for Nuclear Power Plants," was utilized in this verification and content effort. As stated in the Regulatory Guide 1.70, this section of the FSAR provided information regarding the seismic and geological characteristics of the site and the region surrounding the site. It gave the principal seismic and geologic considerations that guided the NRC staff in its evaluation of the acceptability of the site and seismic design bases. Therefore, the entire section is historical information and will be retired in place. Additionally, this change request contains the following typical changes:
Correction of typographical errors.
Correction of external references.
Clarification of FSAR text to avoid misinterpretation and/or correct minor misrepresentations.
Removal of duplicate information.
Removal and/or rewording of information with respect to Unit 2.
Correction of obvious discrepancies between FSAR sections, and with other documents (design criteria, calculations, etc.)
Corrections to conversion errors of files from Word Perfect to Microsoft Word (symbols, column alignment, etc.)

The following are specific UFSAR sections of section 2.5 which are being revised by Change Package 1529. (Note: Page refers to approximate page numbers found in the UFSAR in order that a reviewer may locate change. For various printers, these page numbers may be different.)

Section	Page	Typographical Error	Correct
2.5	2.5-1	a real	areal
2.5	2.5-1	site ex	site-ex
2.5	2.5-2	Logs, Physiographically	New paragraph after Logs
2.5	2.5-2	G's...G's	g's...g's
2.5.1.1.1	2.5-6	Appalachinas	Appalachian
2.5.1.1.4	2.5-18	makes	marks
2.5.1.2.2	2.5-28	soil towards	Sentence to be completed with "the surface of the ground"
2.5.2.1	2.5-42	The number 0	The capital letter O
2.5.4.2.1.1	2.5-58	Is	is
2.5.4.2.1.3	2.5-62	Additional word "a"	Remove the word "a"
2.5.4.2.1.3	2.5-63	Close parenthesis missing	Add close parenthesis after Feature
2.5.4.2.1.3	2.5-73	The word "was" missing	Add word "was" between gravel and classified
2.5.4.2.2.6.4	2.5-82	Double underline used	Use single line under the word feet

SA-SE Number: WBNLCE-98-021-0 ***Implementation Date:*** 10/06/1998

Section	Page	Typographical Error	Correct
2.5.4.8	2.5-106	Too many spaces between elevation and to	Remove all but one space
2.5.5.2.3	2.5-129	Space in the word backfill	Remove space
Table 2.5-7		Elevations incorrect and not in correct order	Correct elevation and arrange in correct order
Table 2.5-8		Incorrect Boring number and elevations	Correct boring number from 25 to 31 and correct elevations
Table 2.5-9		Incorrect Boring number	Correct boring number from 58 to 41

Section	Typographical Error	Correct
Table 2.5-10	Incorrect order for elevations	Correct order
Table 2.5-17	Incorrect symbol and numbers	Correct symbol and numbers
Table 2.5-17A	Incorrect symbol	Correct symbol
Table 2.5-17B	Incorrect symbol	Correct symbol
Table 2.5-17C	Incorrect symbol	Correct symbol
Table 2.5-17D	Incorrect symbol	Correct symbol
Table 2.5-18	Incorrect number	Correct number from "Gravel" to 27
Table 2.5-21	Words omitted	Insert under symbols "Mechanical and hydrometer analysis"
Table 2.5-22	Values improperly placed	Insert values in correct row
Table 2.5-29	Boring and Drill Number incorrect location	Place boring and drill number on correct row
Table 2.5-33	Boring and Drill Number	Replace SS-65A with SS-65B
Table 2.5-34	Incorrect Soil Symbol.	Replace NL with ML
Table 2.5-35	Incorrect symbol	Replace with correct symbol
Table 2.5-36	Incorrect format for math equation	Correct format
Table 2.5-37	Heading misalignment	Correct alignment of headings
Table 2.5-38	Inadequate format for table	Insert line space between (el.. 707.5) and 2 and SS-138 and SS-138a
Table 2.5-41	Inadequate format for table	Insert line space between (el.. 707.5) and 2 and US-77 and US-92
Table 2.5-42	Incorrect order for elevations, values and units	Correct order for elevations, values and units
Table 2.5-43 (Sheet 1)	Incorrect values for Sand % $D_{10,}$ and elevations	Correct values
Table 2.5-43 (Sheet 2)	Value 701-1 incorrect	Correct value 701.1
Table 2.5-54	Misaligned units	Correct alignment of units tsf
Table 2.5-56	Eight line incorrect value	Correct value from 85 to 89
Table 2.5-57	Incorrect value of 96.7	Correct value to 95.7
Table 2.5-58 (Sheet 4)	Incorrect value of 7123	Correct value to 712.3
Table 2.5-18 (Sheet 5)	Incorrect value of 7126	Correct value to 712.6

SA-SE Number: WBNLCE-98-021-0 ***Implementation Date:*** 10/06/1998

Section	Typographical Error	Correct
Table 2.5-58 (Sheet 14)	Incorrect value of 0.54	Correct value to 0.53
Table 2.5-59	Alignment of columns	Correct alignment of columns
Table 2.5-60	Incorrect Symbols	Correct symbols
Table 2.5-61	Incorrect Symbols	Correct symbols
Table 2.5-62(Sheet 1)	Bars missing	Insert bars to tie same sample together
Table 2.5-62(Sheet 2)	Bars missing	Insert bars to tie same sample together
Table 2.5-62(Sheet 4)	Bars missing	Insert bars to tie same sample together
Table 2.5-63(Sheet 1)	Bars missing	Insert bars to tie same sample together
References	Reference 1&2 incorrect spelling of Bollinger	Correct spelling to Bollinger

The following sections did not incorporate the latest FSAR into the UFSAR

Table	Problem	Correction
Table 2.5-6	Values left out	Add values 0.18 and 1.2 to column C_c and P_c
Table 2.5-12	Previous revision removed numbers	Remove values 122 and 121 from γ_s Column
Table 2.5-66	Values incorrect	Correct Values

The following sections were modified due to editorial changes.

System	Description
2.5.1.1.2	The last paragraph was deleted. Same information in section 2.5.1.2.11.
2.5.1.2.4	Delete the last two sentences of the first paragraph. Information provided is no longer needed.
2.5.1.2.9	Last paragraph was reworded for minor editorial changes only that do not change the information provided in the text.
2.5.2.1	Removed word "all" from the last 5 paragraphs. Added date as though when data was tabulated. Moved reference to reference section as Reference 173. This change does not change information provided in the text.
2.5.2.4	Changed distance from Sequoyah and Watts Bar Nuclear Plants from approximately 40 to 50 miles apart to approximately 31 miles apart. This is consistent with Section 2.1.1.1.
2.5.2.7	Add abbreviation "OBE" for Operating Basis Earthquake.
2.5.3.4	Next to last paragraph of that section, add the word "former" to "Tellico Project" since it is no longer a project. Last paragraph changed word from "our" to "the" editorial change.
2.5.4.2.1.2	Last paragraph of section, delete the word "all" from "All laboratory tests." Editorial change, does not change information provided in text.
2.5.4.5.1.2	Page 2.5-98, first paragraph remove the word "all" from the text. This change does not change information provided in text.
2.5.4.5.1.3	Corrected entire section from future verb tense to past verb tense.
2.5.4.5.1.4	Corrected entire section from future verb tense to past verb tense.

SA-SE Number: WBNLCE-98-021-0 ***Implementation Date:*** 10/06/1998

System	Description
2.5.4.5.2	Corrected entire section from future verb tense to past verb tense.
2.5.4.6	Corrected entire section from future verb tense to past verb tense.
2.5.4.8	Page 2.5-113, second paragraph, revised wording to past tense. Page 2.5-113, last paragraph, deleted reference to a report since the remainder of the section describes results.

The changes to the UFSAR Section 2.5 can be characterized as minor changes. These changes were corrections of typographical errors, corrections of external references, clarifications of UFSAR text to avoid misinterpretation and/or correct minor misrepresentations, removal of duplicate information, removal and/or rewording of information with respect to Unit 2, corrections of obvious discrepancies between UFSAR sections and with other documents (design criteria, calculations, etc.), and corrections to conversion errors of files from Word Perfect to Microsoft Word (symbols, column alignment, etc.). These changes have not (1) increased the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR, (2) created any different type of accident or malfunction previously evaluated, or (3) reduced any margin of safety as defined in the Technical Specification. There is no unreviewed safety question.

SA-SE Number: WBPLCE-98-022-0

Implementation Date: 12/31/1998

Document Type:	*Affected Documents:*	*Title:*
FSAR	FSAR Change Package 1530	FSAR Change Package Sections 3.3, 3.5 and 3.6

Description and Safety Assessments:

This UFSAR change request is provided as part of the UFSAR review. Regulatory Guide 1.70, Standard Format and Content of Safety Evaluation Reports for Nuclear Power Plants was utilized in this verification and content effort. This change request contains the following typical changes:

- Correction of typographical errors.
- Correction of external references.
- Clarification of FSAR text to avoid misinterpretation and/or correct minor misrepresentations.
- Removal of duplicate information.
- Removal and/or rewording of information with respect to Unit 2
- Correction of obvious discrepancies between UFSAR sections, and with other documents (design criteria, calculations, etc.).
- Corrections to conversion errors of files from WordPerfect to Microsoft Word (symbols, column alignment, etc.)

The following are specific UFSAR sections 3.3, 3.5, and 3.6 are being revised by Change Package Number 1530. (Note: Page refers to approximate page numbers found in the UFSAR in order that a reviewer may locate change.

The following sections were revised as a result of editorial and clarification type changes, changes that adequately reflect the latest documentation, removal of duplicate information, and removal of reference to Unit 2.

Section	Page	Description
3.3.2.2	3.3-3	Should have been revised to eliminate the reference to the 100 psf tornado pressure differential load on the roof and exterior walls of the spent fuel pool room and cask loading area. Additionally the pressure of 180 psf acting on the roof should be eliminated. It should have been stated that the roof and walls had been evaluated for the effective tornado-generated differential pressure. The roof and walls were found to be acceptable for the differential pressure loading (WCG-1 -166).
3.3.2.2	3.3-3	Correct the phrase quasi-steady to steady state
3.5.1.2.1	3.5-6	Table 3.5-11 contains duplicate information that is contained in Table 3.5-1. Delete Table 3.5-11.
3.5.1.2.1	3.5-7	Table 35-12 contains duplicate information that is contained in Table 3.5-2. Delete Table 3.5-12.
3.5.1.2.1	3.5-8	Table 3.5-13 contains duplicate information that is contained in Table 3.5-3. Delete Table 3.5-13.
3.5.1.2.6	3.5-9	Table 3.5-11 contains duplicate information that is contained in Table 3.5-1. Delete Table 3.5-11.
3.5.1.3.1	3.5-10	Remove wording that refers to Unit 2. Reword.
3.5.1.3.1	3.5-10	Clarification on the word governor needed. Replace "mechanical overspeed

SA-SE Number: WBPLCE-98-022-0 ***Implementation Date:*** 12/31/1998

Section	Page	Description
3,51.3.1	3.5-11	Reference to Section 10.2.4 needed after "low bearing oil pressure". Add reference.
3.5.1.3.1	3.5-11	Clarification on the word "trip' at the end of the first paragraph. Add the word manual" trip
3.5.1.3.1	3.5-12	The wording "l 200 degree" should be "l 200 segment". Make change.
3.5.1.3.2	3.5-20	Replace "two turbine generator sets" with "turbine generator set".
3.5.1.3.3	3.5-20	Replace "two Reactor Buildings" with "Reactor Building".
3.5.1.3.4	3.5-22	Move Pr(H) definition to in front of Pr(H) equation rather than page 3.5-25 which is at the end of all the equations.
3.5.1.3.6	3.5-28	Replace wording "Each reactor" with "The reactor": Eliminates reference to Unit 2.
3.5.3	3.5-31	Add statement that the first paragraph is being left in the UFSAR for historical purposes. Paragraph describes how formula arrived at in computing penetration into concrete. Second paragraph added statement that the following section is used in the design of concrete barriers.
3.5.3	3.5-32	Added clarification as to if pressurizer heater becomes a missile could strike the pressurizer surge line, but the line will not be perforated and will not jeopardize the safe shutdown of the plant.
	TABLE 3.5-11	Delete table as it is duplicate information found in TABLE 3.5-1
	TABLE 3.5-12	Delete table as it is duplicate information found in TABLE 3.5-2
	TABLE 3.5-13	Delete table as it is duplicate information found in TABLE 3.5-3
3.6A	3.6A-1	Replace present tense verbs with past tense. Remove wording about "field routed lines are kept to a minimum" and replace with where "field routing was required" . Remove the phrase "by the pipe rupture team". Evaluations were performed
3.6A	3.6A-2	Item 7 the word "wall" should be the more descriptive word building.
3.6A.1.1.2	3.6A-8	Item 2 the words "shown by analysis" changed to the word "justified". Editorial change that does not change the context of the statement
3.6A.1.3	3.6A-1 1	Removed words each, every, and all This is an editorial change that does not change information provided in the text.
3.6A.2.3,4	3.6A-21	Clarified item B by adding word "continuous" to "Pipe supported". Additionally added definition for L, L1, and L2
3.6 B. 1	3.6B-1	Changed second paragraph to past tense instead of present tense. Deleted second sentence since this has been performed. Deleted fifth paragraph since this item has been performed.
References	3.6B-3	Added title to the sixth reference.

These sections of the UFSAR discuss the wind, tornado, missile, and pipe rupture for the Watts Bar Nuclear Plant . As can be seen above, there are no new accident scenarios created by these changes to the UFSAR nor do the changes affect any of the existing accident scenarios.

The changes to the UFSAR sections 3.3, 3.5, and 3.6 can be characterized as minor changes. These changes were correction of typographical errors, correction of external references, clarification of FSAR text to avoid misinterpretation and/or correct minor misrepresentations, removal of duplicate information, removal and/or rewording of information with respect to Unit 2, correction of obvious discrepancies between FSAR sections, and with other documents (design criteria,

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Implementation Date: 12/31/1998

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Change Package Number 1531	Updated FSAR Review - Section 3.7

Description and Safety Assessments:

calculations, etc.), and corrections to conversion errors of files from WordPerfect to Microsoft Word (symbols, column alignment, etc.). These changes have not (1) increased the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR, (2) created any different type of accident or malfunction previously evaluated, or (3) reduced any margin of safety as defined in the Technical Specification. There is no unreviewed safety question.

This UFSAR change request is provided as part of the UFSAR review. Regulatory Guide 1.70, "Standard Format and Content of Safety Evaluation Reports for Nuclear Power Plants," was utilized in this verification and content effort. This change request contains the following typical changes: Correction of typographical errors; Correction of external references; Clarification of FSAR text to avoid misinterpretation and/or correct minor misrepresentations; Removal of duplicate information; Removal and/or rewording of information with respect to Unit 2; Correction of obvious discrepancies between UFSAR sections and with other documents (design criteria, calculations, etc.); and Corrections to conversion errors of files from WordPerfect to Microsoft Word (symbols, column alignment, etc.)

The following are specific UFSAR sections of section 3.7 which are being revised by Change Package Number 1531.

Section	Page	Description
3,7.2.1.3	3.7-17	There are no 1E Electrical Systems Handholes at WBN. Therefore, handholes needs to be removed from the UFSAR.
3.7.2.4.1	3.7-22	Add comma after the word soils. Editorial.
3.7.2.10.1.1	3.7-28	References a wrong section (3.7.2.1.2). The section referred to is for Set B. Eliminate reference.
3.7.2.10.1.2	3.7-28	References a wrong section (3.7.2.1.1). The section referred to is for Set A Section (3.7.2.1.2). Insert correct reference.
3.7.3.8.1	3.7-37	Code Case 4.b is Welded Attachments on Class 2 or 3 and not for Welded Attachments on Class 1. Oversight, just corrects the title.
3.7.3.12	3.7-48 & 49	Delete "-" at end of sentences. Microsoft conversion error.
3.7.3.17.6	3.7-57	ASTM A446 Grade A galvanized is incorrect. The ASTM should be ASTM A527 galvanized steel sheet with ASTM A446 Grade A (minimum) base metal. Correction of material type with no change for material yield or tensile strength.
3.7.4.2	3.7-58 to 60	Editorial and clarification. First paragraph, clarified the horizontal (0.09 g) and vertical (0.06 g) ground accelerations. For the strong motion triaxial accelerometer the wording for the remote trigger band width should be between 0.5 Hz to 15 Hz nominal. This wording also applies to the triaxial acceleration trigger. This does not change the information for the text. For the active and passive triaxial response

SA-SE Number: WBPLCE-98-023-0 *Implementation Date:* 12/31/1998

Section	Page	Description
3.7.4.2 (continued)	3.7-58 to 60	spectrum recorder, the set of discrete frequencies within the specified bandwidth should be 25.4 Hz rather than 25 Hz.. This matches the vendor information for the recorder. For the passive triaxial response spectrum recorder delete column lines from text and replace with the "Auxiliary Control Room". Since Figure 3.7-40 shows location, this editorial change does not change information that is provided in the text
3.7.4.4.1	3.7-62	Editorial change to second paragraph. The rewording still provides the information that strip chart records will be made from the magnetic tape recording system.
	TABLE 3.7-22	Values for total weight were not updated from new calculation. Additionally, the values for the weight moment of inertia of the N-S and E-W motion were not updated. These values are shown in calculation WCG-1-578 which was prepared as part of the Seismic Analysis CAP. This CAP was reviewed and accepted by the NRC staff.
	TABLE 3.7-23	Values for N-S and E-W Motion for 1/2 SSE and SSE were not updated. These values are shown in calculation WCG-1 -578 which was prepared as part of the Seismic Analysis CAP. This CAP was reviewed and accepted by the NRC staff.

These sections of the UFSAR discuss the seismic analysis for the Watts Bar Nuclear Plant. As can be seen above, there are no new accident scenarios created by these changes to the UFSAR nor do the changes affect any of the existing accident scenarios.

The changes to the UFSAR section 3.7 can be characterized as minor changes. These change were correction of typographical errors, correction of external references, clarification of FSAR text to avoid misinterpretation and/or correct minor misrepresentations, removal of duplicate information, removal and/or rewording of information with respect to Unit 2, correction of obvious discrepancies between FSAR sections, and with other documents (design criteria, calculations, etc.), and corrections to conversion errors of files from WordPerfect to Microsoft Word (symbols, column alignment, etc.).

SA-SE Number: WBPLCE-98-024-0

Implementation Date: 10/23/1998

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Change Package 1533	Updated FSAR Review - Section 3.10.

Description and Safety Assessment:

This UFSAR change request is provided as part of the design basis FSAR review. Regulatory Guide 1.70, Revision 2. Standard Format and Content of Safety Evaluation Reports for Nuclear Power Plants was utilized in this verification and content effort. This change request contains the following typical changes:

1. Correction of typographical errors.
2. Clarification of FSAR text to avoid misinterpretation and/or correct minor misrepresentations.

The following are specific UFSAR sections of section 3.10 which are being revised by Change Package Number 1533.

Page	Typographical Error	Correction
TABLE 3.10-3 (Sheet 2 of 32)	Extra close parenthesis before Hz	Remove close parenthesis
TABLE 3.10-3 (Sheet 7 of 32)	***after word phase should be removed	Remove
TABLE 3.10-3 (Sheet 14 of 32)	The word "shall" should be "shell"	The word "shall" should be "shell"
TABLE 3.10-3 (Sheet 24 of 32)	The word "Call" should be "Cell" and the word "specie\men is misspelled	Correct the word to "Cell" and correct spelling of specimen
TABLE 3.10-3 (Sheet 28 of 32)	The number "1" was inadvertently left out after "No."	Add the number "1"
TABLE 3.10-4 (Sheet 2 of 6)	The word "mounded" should be "mounted"	Correct the word to "mounted"

The following are additional changes that were required to be made in Section 3.10 of the UFSAR:

Section	Page	Description
3.10.1	3.10-3	Remove sentence "The Watts Bar Nuclear Plant does not use the Eagle Signal Timer that is under question by the NRC Staff." This is an editorial deletion. Since Watts Bar does not use this device there is no need for this statement to be in the UFSAR.
3.10.3.2.1	3.10-6	Added words "of bolted parts" for clarification as to what parts when test data is used to establish capacities.
TABLE 3.10-3		In the section "Seismic Test," subsection 2, the last sentence should include "and one SSE" after the words "Five ½-level SSE's."

SA-SE Number: WBPLCE-98-024-0 ***Implementation Date:*** 10/23/1998

The changes to the UFSAR section 3.10 can be characterized as minor changes. These changes were correction of typographical errors and clarification of FSAR text to avoid misinterpretation and/or correct minor misrepresentations. These changes have not (1) increased the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, (2) created any different type of accident or malfunction previously evaluated, or (3) reduced any margin of safety as defined in the Technical Specification. There is no unreviewed safety question.

SA-SE Number: WBPLCE-98-025-0

Implementation Date: 05/25/1999

Document Type:	Affected Documents:	Title:
Design Change FSAR	EDC E-50010-A FSAR Change Package 1550 TRM Change Package 98-020 TRM Revision 17	FSAR Review and Verification of Section 2.4 and Implementation of the New Flood Plan

Description and Safety Assessments:

Watts Bar Nuclear Plant is designed to withstand the effects of the probable maximum flood, a design basis event which assumes an extreme flood resulting in more than 10 feet of water above plant grade for a period of several days. Preparation for operation in the flood mode is a 27 hour process governed by Abnormal Operating Instruction (AOI)-7.01 and several other procedures. Warning of impending floods is provided by TVA Water Management based on pre-established criteria which ensure sufficient advance warning to accomplish flood mode preparations prior to the flood waters exceeding plant grade.

Over a multi-year period, several changes have been made to dams upstream of Watts Bar Nuclear Plant. Fort Loudon was raised 3.25 feet. Embankments were raised at Watts Bar, Boone, Cherokee, Douglas and Watauga. A spillway was added at Tellico Dam. New UFSAR Table 2.4-16 (part of this change package) contains a complete list of dam modifications. These changes were made to prevent failure of those dams during severe flood events. Corrective action document WBPER970841 was issued to document the changes. The net effect of the changes was to lower predicted flood levels at WBN for all postulated flood and combination seismic/flood events. This provides additional margin in the WBN design for these events. In addition, the warning times available to respond to these events was increased. Both of these positive impacts result directly from elimination of the previously postulated dam failures. The detailed description below addresses the following topics:

Rainfall Flood Reanalysis

Seismic Flood Reanalysis

Flood Warning Notification Process

UFSAR Changes

Technical Requirements Manual/Bases Changes

Design Document/Drawing Changes

Affected Procedures

The document changes are summarized in the appropriate paragraphs below. A more detailed description is provided as an attachment to this safety evaluation. The revised documents present descriptions of the analyses performed and the revised results. Where appropriate, the bases for the reanalysis are provided. The previous distinct warning plans for rainfall and seismic floods have been updated and combined into a single notification process. Portions of the original safety analysis describing analytical techniques and past meteorological data were declared historical information. Minor changes which have not been explicitly addressed are considered editorial or typographical.

SA-SE Number: WBPLCE-98-025-0 ***Implementation Date:*** 05/25/1999

Rainfall Flood Reanalysis

The previous analysis showed that two storms produced the worst case flood levels for WBN. These were the 7,980 square mile storm and the 21,400 square mile storm. Both storm profiles were developed from U.S. Weather Bureau Hydrometeorological Reports for the region and were accepted by the staff for use in the original analysis. These storms were reanalyzed using the same analysis techniques accepted for the original analysis. Other rainfall storms which had previously been shown to be non-governing were not reanalyzed. In the new analysis, Chickamauga Dam downstream of the plant was assumed not to fail. This assumption conservatively resulted in slightly higher flood levels than would have been calculated otherwise.

The reanalysis results demonstrated that both storms continue to produce essentially the same flood level at WBN (the 0.2 foot difference was not considered significant). The new probable maximum flood level is 3.2 feet below the old level. Noteworthy in this respect is the fact that the new flood level inside the Auxiliary Building remains below Elevation 737.

Seismic Flood Reanalysis

This was the portion of the flood analysis most greatly affected by the dam safety modifications. The original analysis had shown that floods caused by seismic events were less severe than those resulting from extreme rainfall. The reanalysis showed that, with the elimination of upstream dam failures, none of the seismically induced floods exceed plant grade at WBN. Only a single postulated event involving the multiple seismic failures of Norris, Cherokee and Douglas Dams combined with summer headwater elevations and the 25-year flood could result in a flood approaching plant grade (flood Elevation 727.5 vs. grade Elevation). This event combined with wind and wave runup from a worst case sustained wind could cause water entry into plant structures. The flood peak for this scenario occurs approximately 50 hours after the dam failures.

These reanalysis results eliminated the event which previously resulted in the shortest flood warning time (Fontana Dam failure). As with the rainfall analysis, events which were previously shown to be non-governing were not reanalyzed.

Flood Warning Notification Process

The flood warning notification process was simplified and organizational references were clarified as a result of this reanalysis. River levels are monitored by River System Operations (RSO) within the TVA Water Management Group. When flood conditions are predicted, RSO personnel activate the Knoxville Emergency Operations Center (KEOC). This occurs in advance of conditions equivalent to 1/2 the WBN probable maximum precipitation. As described in the UFSAR, an elaborate rainfall measuring and river level prediction system is in place to perform this function. A benchmarked computerized model of the river system uses inputs from 98 rain gages, 23 streamflow gages, and 18 hydro plants to track river conditions and provide estimates of flood levels at downstream locations such as WBN. The notification process is proceduralized to ensure that WBN and SQN are informed and that contact is subsequently maintained throughout the flood event. A review of the warning process was performed as a result of the reanalysis and it was concluded that the process is adequately designed to provide the required minimum of 27 hours of warning time for a flood event exceeding plant grade.

SA-SE Number: WBPLCE-98-025-0 ***Implementation Date:*** 05/25/1999

UFSAR Changes

The specific changes are summarized below. In addition to typographical corrections and editorial changes, the revisions included:

deletion of events and dam failures which will no longer occur,

corrected organizational references,

added a description of the dam safety program,

added notes to distinguish discussions of reanalyzed events from discussions of other events which were not reanalyzed,

revised flood levels,

added references to the WBN west saddle dike, which is still allowed to fail,

corrected references to tables and figures affected by the revision,

marked various sections, tables, and figures historical,

described the updated flood warning notification process, and

revised/deleted tables and figures to reflect the above changes.

A section by section listing is provided as an attachment to this safety evaluation. These changes accurately reflect the inputs and results of the flood event reanalysis. Since the results indicate increased margin with respect to flood levels used in the design of the unit, these changes are considered acceptable from a safety standpoint. The verification review of UFSAR Section 2.4 was also performed as part of this effort and minor changes which resulted from that review have been included in the Change Package addressed by this SA/SE.

Technical Requirements Manual/Bases Changes

The TRM and TRM Bases changes were developed considering the simplified warning plan and significantly reduced threat from seismic dam failures. The revised warning plan no longer requires notification of the site when reservoir elevations reach summer pool levels. This need was eliminated by eliminating the seismic flood events which could exceed grade under non-flood conditions. The surveillance requirements pertaining to flood levels at the Intake Pumping Station were eliminated, as these readings are not used in predicting the flood level at WBN and have no other value to the plant. The applicability statement was revised to reflect an increased threshold of initial concern resulting from the increased margin provided by the reanalysis. The threshold criteria for the Stage I and Stage II warnings were left unchanged, as these provide the basis for protecting the plant from all flood events. Organizational references were corrected and various editorial changes were made.

SA-SE Number: WBPLCE-98-025-0 ***Implementation Date:*** 05/25/1999

A section by section listing is provided as an attachment to this safety evaluation. These changes accurately reflect the inputs and results of the flood event reanalysis. The revised TRM and Bases provide assurance that the plant will correctly implement the required flood protection provisions.

Design Document/Drawing Changes

Twenty-two design criteria and system descriptions required revision to eliminate or correct references to the previous flood levels. In many cases, this change was editorial, since the system affected was not designed to function during flood mode. In the remaining cases, the margin for acceptable system operation was enhanced by the reduced flood levels.

Several drawings were identified which made reference to the previous flood elevations. These drawings were revised to eliminate or correct the references, as appropriate. Since the maximum flood level has been reduced, this change will not impact the function or qualification of any plant system or component.

A section by section listing is provided as an attachment to this safety evaluation. These changes accurately reflect the inputs and results of the flood event reanalysis.

Affected Procedures

Procedures will be revised prior to closure of EDC E-5001O-A.

This change does not constitute an unreviewed safety question because the only impact on the plant is to increase the margin available for response to the design basis flood event, both by increasing available warning time and by decreasing the predicted worst case flood elevation. This conclusion is also based on the following:

- The reanalysis used the same meteorological inputs as the original analysis.
- All potentially controlling events were reanalyzed.
- The same analytical techniques were used as in the original analysis.
- The same river system model was also used. Only the dam outflows and retention capabilities were adjusted to reflect the elimination of dam failures.
- The WBN flood mode response plan documented in AOI-7.01 remains unchanged.
- All plant system and component designs and functions remain unchanged.
- The design basis for plant structures remains unchanged. No structural calculations were revised to take advantage of the lower water elevation.
- The warning plan has been simplified and verified to ensure that the required 27 hour advance warning will be provided. Interfaces with TVA organizations outside TVAN have been reconfirmed.

SA-SE Number: WBPLEE-97-047-0

Implementation Date: 02/23/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39303-A	Equipment Replacement on the Gas Analyzer Panel and Components on Containment Isolation Valves

Description and Safety Assessments:

DCN M-39303-A replaces the oxygen and hydrogen recorders on the gas analyzer panel (0-L-206) with a multi-channel recorder, and replaces the gas analyzer stream sequencer with a programmable stream sequencer. This DCN also modifies the control circuit for 1-FCV-77-17 and 1-FCV-68-307 to allow normally open operation of these valves, and replaces the solenoid valves associated with the valves. This DCN also replaces the diaphragm of 1-FCV-77-17, which is a Saunders type valve. Additionally, this DCN adds nine switches to the gas analyzer panel to facilitate the collection of gas samples when the sequencer is inoperable.

The gas analyzer determines the quantity of oxygen and hydrogen in the gas space of various tanks in the Waste Disposal System (WDS), the Chemical & Volume Control System (CVCS), and the Reactor Coolant System (RCS). A local and main control room (MCR) alarm on 2% oxygen concentration and 4% concentration is provided. These alarms require operator action in order to prevent the formation of a combustible gas mixture. The gas analyzer is not safety related and is not required to function during or after a design basis event.

Presently, the gas analyzer stream sequencer automatically provides a sample to the gas analyzer from each of the various tanks by opening a selected sample inlet solenoid valve at three minute intervals. The three minute intervals do not take into account sample line lengths, pressures, etc., that can affect the amount of time a stream must be sampled in order to assure a representative sample from each tank. Providing a programmable sequencer allows variable sample intervals for each of the sample streams. Since the existing stream sequencer is integral to the existing oxygen recorder, the recorder must also be replaced. A multi-channel recorder that can also record hydrogen concentrations will replace the oxygen and hydrogen recorders on the gas analyzer panel. In order to allow sampling to occur when the sequencer is inoperable, nine switches and the associated wiring will be added to the panel to operate the sample inlet solenoid valves manually. These switches will have no interface with the containment isolation valve circuits.

As stated above, the stream sequencer presently selects a sample inlet solenoid valve to be opened for three minutes, and continuously cycles through the sample points at three minute intervals for each solenoid inlet valve/sample point. In order to receive samples from the Reactor Coolant Drain Tank (RCDT) and the Pressurizer Relief Tank (PRT), it is necessary to open containment isolation valves 1-FCV-77-17 and 1 -FCV-68-307, respectively. These valves, and their associated solenoid valves, are environmentally qualified for a harsh environment. Their qualification is partially based on a limited duty cycle. Since the valves have been cycled by the stream sequencer at the rate of approximately three times per hour, replacement of the solenoid valves and control circuits for 1-FCV-77-17 and 1 -FCV-68-307 are being modified in order to allow normally open operation of the valves by removing the gas analyzer automatic control of these valves, which will significantly reduce the number of cycles per year these devices are required to perform.

Containment isolation valves 1-FCV-77-17 and 1-FCV-68-307 are safety related and do perform a safety function. Automatic closure of these valves is required in order to maintain containment integrity following a design basis event to minimize release of any radioactive material.

The information presented in the FSAR for the gas analyzer and containment isolation systems are not impacted by this modification.

SA-SE Number: WBORAD-97-001-0 ***Implementation Date:*** 03/27/1998

The changes described above for the gas analyzer do not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. The replacement of the equipment on the gas analyzer panel does not change the method of sampling the gas spaces of the various tanks, nor does it change any alarm setpoints or operator actions required due to the alarms. These changes do not alter the interface of the gas analyzer with systems important to safety, but changes to the containment isolation valve circuits actually reduces the gas analyzer interface with systems important to safety. The changes affect only a non-safety grade system which has no accident mitigation function.

The changes described above for the containment isolation valves do not affect any FSAR accident analysis evaluations or equipment malfunction evaluations. Any increase in the probability of a malfunction of the valves (failure to close upon receipt of phase A containment isolation signal) due to the change from a "normally" closed valve (the valves are closed except when the RCDT or PRT is being sampled) to a normally open valve is offset by the testing and surveillance program for these valves, which is identical to the program for the existing normally open inboard containment isolation valves. The changes do not change functional or performance requirements of the valves, nor do they inhibit the valves from performing their safety function. The changes do not create any new accidents or equipment malfunction failures, and they do not reduce the margin of safety as identified in the Technical Specifications.

Therefore, on the basis of the evaluation of effects, 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-97-167-0

Implementation Date: 03/05/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN W-39787-A	RCS Loop Delta T/Auctioneered Delta T Deviation Alarm Setpoint

Description and Safety Assessments:

DCN W-39787-A revises the Reactor Coolant System (RCS) Loop delta T/Auctioneered delta T deviation alarm setpoint from +2 degrees F to +3 degrees F. This alarm is annunciated on main control room (MCR) window 5A-93A and is actuated when the difference between any loop delta T and the auctioneered high delta T of the four RCS loops exceeds the setpoint. The change will eliminate a nuisance alarm by raising the setpoint above the normal band for deviations between loop delta Ts. The alarm is actuated by bistables located in a non-safety related process control rack.

During normal operating conditions at full power, small differences between the delta T in each RCS loop are expected. In addition, fluctuations in loop temperatures due to streaming can result in deviations from the nominal difference between loop delta Ts. The delta T deviation alarm provides an alert that the plant may be operating outside normal steady state conditions. It may also be indicative of other abnormal conditions such as failure of hot or cold leg instrumentation, steam flow/feed flow mismatch, or reactor coolant pump trip. Additional alarms are provided in the MCR for all of these conditions.

The alarm setpoint is selected to distinguish between normal loop deviations and abnormal operating conditions. The current value of 2 degrees F is typical for initial fuel cycles and proved to be adequate for WBN Cycle 1. Due to modifications such as more aggressive core designs and the increased streaming resulting from such modifications, the normal loop differences for delta T in some plants can approach the 2 degrees F value. Increasing the alarm setpoint to a value of +3 degrees F will eliminate the nuisance alarm condition while preserving the intended function of notifying the operator of plant operation outside steady state conditions.

Section 7.2.2.3.2 of the SAR describes temperature deviation alarms which are actuated if any temperature channel deviates significantly from the auctioneered (highest) value but does not list setpoints. No changes are required to information or descriptions presented in the SAR. However, in paragraph 4.4.3.4 of Supplement 8 of the Watts Bar Safety Evaluation Report, a description of the delta T deviation alarm was provided, including the setpoint of 2 degrees F. This evaluation was based on TVA submittals for the RTD Bypass Elimination project, including a July 9, 1991 response to an NRC request for additional information which specifically requested information concerning delta T and Tavg loop deviation alarms and setpoints.

The loop delta T deviations are not used as input to any protection functions and there are no associated Technical Specifications or Technical Requirements. The proposed change will not affect any reactor protection functions such as the over temperature (OT) delta T and over power (OP) delta T reactor trip functions. The protection system setpoint study assumes that the loop specific delta T values are normalized; this ensures that the nominal delta T values used in the OT delta T and OP delta T reactor trip functions are consistent with the initial conditions used in the analyses which credit these functions. There are no control functions associated with these loop differences and, thus, the alarm setpoint change will have no impact on control systems which use delta T as an input (e.g., rod control system). The change does not involve any new or different type of equipment or hardware modifications and, therefore, no additional or different failure modes will be created. Revising the setpoint provides a benefit in that it will reduce nuisance alarms which can detract operator attention from more important tasks.

The change satisfies the intent of the alarm as specified in the design basis and does not affect the input assumptions to any safety analyses. The safety analyses do not model or take any credit for operator action associated with this alarm and do not explicitly model loop to loop variations in delta T. Thus, the change does not affect any SAR evaluations (accident analysis or equipment malfunction failures) previously performed and no new accidents or equipment malfunction failures are created. Therefore, on the basis of the evaluation of effects, 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-97-168-1

Implementation Date: 01/28/1998

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Change Package 1498 & Supplement Number 1	Deletion of Bus Splitting Relays from the Watts Bar Hydro Plants 161 kV switchyard.

Description and Safety Assessments:

The proposed activity is the deletion of Bus Splitting Relays from the Watts Bar Hydro Plants 161 kV switchyard and the required change to Watts Bar SAR as identified in Change Package Number 1498. These relays were originally installed to provide assurance that the Watts Bar Hydro Plant Generators would remain stable under postulated worst case fault conditions.

TPS no longer considers a three-phase fault and simultaneous stuck breaker as part of their planning criteria. TPS now considers a phase-to-ground fault with a stuck breaker for the worst case scenario.

Transmission Planning Department's most recent study documents that the bus splitting relay scheme is not required to maintain the Hydro Plant generators stable under postulated fault conditions of a phase-to-ground fault with a stuck breaker. This study shows that offsite power supply voltage recovery is significantly improved if the automatic bus split does not occur. Therefore, the Transmission Planning Department is proposing that the automatic bus split relay scheme used at the Watts Bar Hydro Plant 161 kV switchyard be disabled by permanently lifting of wires on these relays and opening trip cutout switches or PK blocks on the associated trip circuits.

Although there is not a clear mechanism for a new failure mode and one is certainly not expected, the worst case failure mode scenario of the proposed activity that could be hypothesized would be the complete loss of offsite power. This scenario is adequately enveloped by FSAR Section 15.2 9 which addresses the accident analysis for a complete loss of all offsite power coincident with the loss of onsite AC power to the station auxiliaries. This is addressed as a condition 11 (Faults of Moderate Frequency) event. The deletion of the bus splitting relays at the Wafts Bar Hydro plant for its 161kV switchyard will have no impact on the accident analysis that has been performed to address the much broader issue of the complete loss of offsite power source coincident with the loss of onsite AC power. The loss of the offsite power supply is also adequately controlled by TS Section 3.8.1 for operating Modes 1-4 and TS Section 3.8.2 for shutdown Modes 5-6 during movement of irradiated fuel assemblies.

This proposed activity does not present an unreviewed safety question as the disabling of the bus splitting relays at the Watts Bar Hydro plant does not diminish the capability or capacity of the 161kV offsite power requirements as imposed by GDC 17. Based on issued transmission system studies, the deletion of these relays will actually be an improvement to the offsite power system in that voltage recovery is significantly improved if the automatic bus split does not occur. The accident analysis in the FSAR addresses the complete loss of offsite power coincident with the loss of onsite AC power. The current FSAR analysis is bounding for the worst possible results that could be postulated from this proposed activity and this activity will not result in any new accidents or malfunctions of a type than what has been previously analyzed. A review of the Technical Specifications Sections 3.8.1 and 3.8.2 which address loss of offsite power as well as the past NRC SERs has not identified any margin of safety or acceptance limits which would be affected by this proposed activity.

SA-SE Number: WBPLEE-97-170-0

Implementation Date: 02/13/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39785-A FSAR Figure 9.4-17	Various System Control Diagrams Revised

Description and Safety Assessments:

DCN S-39785-A revises Sampling & Water Quality system control diagram 1-47W610-43-6 per Drawing Deviation (DD) 97-165 to correct the component identification of manual isolation valve (1 -ISV- 13 -200J), check valve (1-CKV- 13 -200K), manual isolation valve (l-ISV-43-210J), and check valve (l-CKV-43-210K).

DCN S-39785-A revises Turbo-Generator Auxiliaries schematic diagram 1-45W600-47-7 per DD 97-167 to show that 120V power is provided from PT-47-13 to 1-XI-47-13A & -13E on drawing 1-45W600-47-8. It is important that schematics account for all power loads.

DCN S-39785-A also revises Containment Ventilating system control diagram 1-47W610-30-4 per DD 97-167 to remove the time delay for annunciator window 6E-138A which is provided by the Ronan annunciator system. The 64 second time shown was incorrect and due to timing functions not normally being shown on control diagrams, the time was removed from the control diagram. The time delay provided by the Ronan annunciator system is shown on 1-45B655-6E (Annunciator Inputs for window box 6E).

DCN S-39785-A also revises Radiation Monitoring system control diagram 1-47W610-90-5 per DD 97-169 to correct UNID from PLOT-90-402 to PLOT-90-452.

The (a) correction of the component identification for 1 -ISV-43-200J, 1-ISV-43-210J, 1-CKV-43-200K & l-CKV-43-210K, (b) revising 1-45W600-47-7 to show that 120V power is provided from PT-47-13 to 1-XI-47-13A & -13E on drawing 1-45W600-47-8, (c) revising 1-47W610-30-1 to remove the time delay which is provided by the Ronan annunciator system, and (d) revising 1-47W610-90-5 to correct UNID from PLOT-90-402 to PLOT-90-452 does not affect any system operational or functional features of the systems involved. These changes are documentation only and no physical changes are made by DCN S-39785-A.

Based on the previous discussion, the minor change to FSAR Figure 9.4-17 (Drawing 1-47W610-30-4) changed by DCN S-39785-A does not impact the probability of occurrence of consequences of any accident or equipment malfunction currently evaluated in the FSAR. 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 of DCN S-39785-A do not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-98-003-0

Implementation Date: 06/05/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39306-A FSAR Package 1448	Gas Analyzer Replacement

Description and Safety Assessments:

This change, DCN M-39306-A, replaces the existing Gas Analyzer supplied by Comsip Custom Line Corp. with one supplied by Orbisphere Laboratories). The replacement system is designed to provide continuous online monitoring of the gas stream and provide local readout of measurement data. The measurement data is transmitted to existing recorders for plant evaluation and for permanent record documentation. Also, the replacement system provides liquid separation and collection from the gas stream and in-line calibration capability. The analyzer system contains an Orbisphere Gas Analyzer Module which consists of a flow chamber, hydrogen and oxygen sensor, and pressure sensor. The system uses supporting pressure and flow control components and conditioning devices needed for optimum sensor performance and in-line calibration. The gas analyzer assembly is located in the Unit 2 hot sample room, Auxiliary Building, elevation 713.

The system design requires the monitoring of cover gas in selected tanks for the presence of hydrogen and oxygen. Excessive levels of hydrogen and oxygen would create the potential of an explosion which could result in a release of radiation in excess of 10CFR100 limits. The minimization of a potential for an explosion is accomplished by monitoring for hydrogen and oxygen and by maintaining the oxygen concentration less than or equal to 2% by volume when the hydrogen concentration is greater than or equal to 4% by volume. The analyzer shall detect and alarm a condition where oxygen level is less than or equal to 2% and greater than or equal to 4% by volume. (Hydrogen level is conservatively assumed to be >4% and is therefore, not alarmed). The replacement Orbisphere analyzer can accurately measure hydrogen and oxygen (full range) and provide corresponding analog output signal (4-20 ma) and adjustable setpoint alarm output. This system meets all gas monitoring functional requirements. Additionally, the gas analyzer system has no operability requirements during or after a design basis event (DBE). This change meets all established design parameters and is safe from a nuclear safety standpoint.

This change does not significantly change any equipment failure modes. A loss of electrical power or sample stream flow will result in a sample measurement loss which is the same as the existing gas analyzer. (Plant instructions provide alternate gas sample methods due to an inoperable gas analyzer with is described in FSAR Section 11.3). The electrical supply source is not changed. The sample stream (inlet and outlet) paths are not changed. No interface equipment has been added that would cause a different type of sample line flow failure.

The Orbisphere gas analyzer system requires an N2 purge flow to support the H2 detector operation. Two 300 cu ft N2 tanks are furnished with separate regulator/gauges to monitor each tank. A 3-way selector valve is furnished to allow one tank to supply the normal N2 purge flow with the other N2 tank in standby (or while the other N2 tank is being replaced). Under normal operating conditions, a 300 cu ft N2 tank should provide adequate purge flow for approximately one year based on vendor operating experience. This N2 tank arrangement does not add a significant failure mode.

The Orbisphere analyzer processor is a menu-driven microprocessor-based for additional accuracy and operator interface convenience. The operation software is in the form of programmable read-only memory. This type of software is commonly called firmware. The use of firmware maintains the integrity of the operating software since the customer does not have access to the program steps. Microprocessor-based equipment is in widespread use in the nuclear industry. The Orbisphere analyzer is used in most nuclear plant gas analyzer applications. This equipment is reliable as experienced at WBN and other TVA facilities. The Orbisphere gas analyzer processor software uses self-diagnostics routines that ensures all analyzer operations are within prescribed limits. Therefore, the replacement Orbisphere gas analyzer system is an improvement in equipment reliability.

SA-SE Number: WBPLEE-98-003-0 ***Implementation Date:*** 06/05/1998

FSAR Impact
FSAR Section 9.3.2, "Process Sampling System" discusses the general operation of the gas analyzer. The hydrogen and oxygen measurement concentrations are stated to be displayed, recorded, and alarmed at the analyzer when appropriate. Also, general sample line routing requirements and piping code class assignments are discussed. This change complies with these descriptions.

Table 11.3-1, "Gaseous Waste Processing System Component Data"
FSAR Change Package Number 1448 S1 updates the component data description of the sequential automatic gas analyzer related to H2 and 02 measurement type, calibrated range, and the number of sampled points.

This change revises FSAR Figure Nos. 11.3-2 sheet 2 and 11.3-2 sheet 3.
Safety Evaluation Report (SER) June 1982 (including supplements I through 20) are not impacted.

The rupture of a single Waste Gas Decay Tank (WGDT) is a postulated (condition M) design basis accident. The replacement of the gas analyzer does not affect this postulated fault condition. The replacement gas analyzer provides accurate hydrogen and oxygen measurement capability needed to alert operations personnel to take preventative measures required to correct any potentially dangerous gas mixtures. Therefore, this change does not impact the consequences or effects of this faulted condition.

This change, DCN M-39306-A, does not affect any FSAR 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-98-009-0

Implementation Date: 03/09/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39795-A FSAR Figures	Revision of Radiation Sampling System Drawings

Description and Safety Assessments:

DCN S-39795-A corrects several unrelated Drawings Deviations so that the drawings more accurately reflect the as constructed plant configuration and to ensure that all design output documents agree. No hardware or functional changes are being made by this Design Change Notice.

DCN S-39795-A revises Radiation Sampling System electrical drawing 1-47W625-19 per Drawing Deviation (DD) 98-0001 to correct the unique identification number of panels 1-PNL-43-210C1-B, 1-PNL-43-210C2-B, 1-PN-43-200C1-A and l-PN-43-200C2-A. These panel identifications were previously changed by DCN S-34033-A from 1-L-167A, 1-L-167B, 1-L- 168A and 1-L-168B to 1-PNL-43-210C1 -B, 1-PNL 13-210C2-B, l-PNL-43-200C1-A and l-PNL-43-200C2-A, however, drawing 1-47W625-19 was overlooked. These panels are the Hydrogen Analyzer Calibration Gas Panels for the Hydrogen Analyzers.

DCN S-39795-A revises the Master Equipment List (MEL) per DD 98-0001 to correct the description for manual isolation valves l-ISV-43-200J-A and l-ISV-43-210J-B and check valves, l-CKV-43-200K-A and 1-CKV-43-210K-B. For example l-ISV-43-200J-A had the description as "CYL Air Isolation Valve to PNL 1-L-167A." DCN S-39795-A changes this description to "CYL Air Isolation Valve to l-PNL-43-200C1-A."

DCN S-39795-A revises Sampling & Water Quality System electrical schematic diagram 1-45W600-43-1 per DD 97-0171 to correctly show solenoid valves 0-FSV-43-40B, 1-FSV-43-42B, 1-FSV-43-45B, 1-FSV-43-44B, 2-FSV-43-45B, 2-FSV-43-45A, l-FSV-43-15A, 0-FSV-43-46, 0-FSV-43-47, 0-FSV-43-40A, 1-FSV-4341, 1-FSV-43-42A, 1-FSV-43-43, and l-FSV-43-44A as three way type valves instead of two way type valves. These solenoid valves are used by the gas analyzer sequencer to take samples from the Spent Resin Storage Tank, Volume Control Tank, Unit 1 & 2 Holdup Tank, Boric Acid Evaporator, Gas Decay Tanks Plant Vent Header, Gas Decay Tanks Gas Sampling Header, Spend Resin Storage Tank, Unit 1 Reactor Coolant Drain Tank, and the Reactor Coolant System Unit 1 Pressure Relief Tank.

DCN S-39795-A revises Sampling & Water Quality System control diagrams 1-17W610-43-5 and 1-47W610-43-5A per DD 97-0171 to correct the way that solenoid valves l-FSV-43-42B, l-FSV-43-44B, l-FSV-43-45B, 0-FSV-43-40B and 2-FSV-43-45B are drawn so that the pilot solenoids normal position will match the actuated valve's normal position. 1-47W610-43-5 and 1-47W610-43-5A are FSAR Figures 11.3-2 Sheet 3 and 11.3-2 Sheet 3A respectively. These solenoid valves are used by the gas analyzer sequencer to take samples from the Spent Resin Storage Tank, Unit 1 Volume Control Tank, Unit 1 & 2 Holdup Tank, and the Unit 1 Boric Acid Evaporator.

DCN S-39795-A revises Turbo-Generator Control System control diagram 1-47W610-47-3 per DD 97-0171 to delete Emergency Response Facility Data System (ERFDS) point designator V9037 and plant process computer point designators Y2001A through Y2015A. These points were previously used for the Turbine Supervisory Instrumentation and were deleted by DCNs W-39759-A and M-39242-A respectively. Log point numbers Y2001A through Y2015A are no longer used in the process plant computer and point number V9037 is spared in ERFDS. 1-47W610-47-3 is FSAR figure 10.2.4.

DCN S-39795-A revises Post Accident Sampling System control diagram 1-47W610-43-8 per DD 97-0170 to correct the location of the demineralized water connection (D1) to Panel l-L-567.

SA-SE Number: WBPLEE-98-009-0 ***Implementation Date:*** 03/09/1998

The (a) correction of the unique identification number for l-PNL-43-210C l-B, l-PNL-43-210C2-B, l-PNL-43-200C1-A and l-PNL-43-200C2-A (b) revising MEL to correct descriptions for l-ISV-43-200J-A, l-ISV-43-210J-B, l-CKV-43-200K-A, l-CKV-43-210K-B (c) revising 1-45W600-43-1 to correctly show solenoid valves as three way type valves instead of two way (d) revising l-47W610-43 -5 and 1-47W610-43 -5A to correctly depict the correct normal position of solenoid valves l-FSV-43-42B, l-FSV-43-44B, l-FSV-43-45B, 0-FSV-43-40B and 2-FSV-43-45B (e) revising 1-47W610-47-3 to delete unused ERFDS point designator V9037 and unused process plant computer point designators Y2001A through Y2015A, and (f) revising 1-47W610 13-8 to correct the location of D 1 demineralized water connection to panel 1-L-567 do not affect any operational or functional features of the systems involved. These changes are documentation only and no physical changes are made by DCN S-39795-A.

Since these are documentation change only and do not represent any functional, operation, or physical change to the plant, the minor changes to FSAR figures 10.2.4, 11.3-2 Sheet 3 and 11.3-2 Sheet 3A by DCN S-39795-A do not impact the probability of occurrence or consequences of any accident or equipment malfunction currently evaluated in the FSAR. 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 of DCN S-39795-A do not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-98-010-0

Implementation Date: 04/03/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change FSAR	DCN W-39608-A FSAR Change Package 1508	Containment Sump Level Transmitter Replacement

Description and Safety Assessments:

This DCN (M-39608-A) replaces the containment sump level transmitters in Unit 1. The existing transmitters are Barton transmitters with a diaphragm seal and capillary tubing. These existing transmitters have a problem with the capillary tubing leaking fill fluid, and maintaining the transmitter within calibration. The new transmitters are Class 1E qualified, do not have capillary tubing, are more accurate, and can be submersed during a LOCA. This change upgrades equipment used to perform a function. Functional performance of the plant is not affected and protective logic is not affected.

The range of the transmitter is changing from 0 to 20 feet (240") to 0 to 200" (16' & 8") which will improve instrument loop accuracy. The existing setpoint for switchover from RWST remains the same. The new transmitters' range is fully adequate to monitor the maximum equilibrium flood level, which is above the PAM requirement of 600,000 gallons.

The sump is in the lower containment, below the refueling cavity. The sump is a water source for long term recirculation for the functions of RHR, emergency core cooling, containment atmosphere cleanup, and containment long term cooling. The transmitters will be located just outside the sump in the raceway. These transmitters are associated with the protective features used to detect and mitigate the effects of Condition III & IV events associated with a LOCA. Four safety-related level transmitters (one per channel) are provided to measure the containment sump level. These transmitters provide input to allow switchover from RWST to containment sump recirculation and also provide input to PAM Category I indicators 1-LI-63-180 and -181. The four containment sump level high trip signals are combined in a 2 out of 4 circuit to produce an output that is combined with the output of the RWST low level switches. When this logic signal is made up the valves from the RWST are closed, and the containment sump becomes the water source for long term recirculation.

Implementation of this DCN requires the mounting of the new transmitters, rerouting instrument sense lines, and cables, and revising the dropping resistor at the Eagle racks. The setpoint will not change, and indication scales are not affected as they currently read in 0-100% scale.

This change upgrades existing plant equipment. The failure modes of the replacement equipment do not differ from the equipment being replaced, and common mode failure has been demonstrated not to be an issue based on experience with these types of transmitters at Sequoyah. The installed loops (equipment and cable/conduit) are separated physically and electrically. Proper separation/isolation of cable routing and equipment is maintained by the DCN and appropriate plant installation procedures. The independence of safety related equipment is not challenged. Civil calculations have been performed to verify that, when installed per the DCN, the equipment will remain able to perform its function following a seismic event. The equipment has been tested and the test report reviewed documenting the new transmitters are not to be susceptible to EMI/RFI and will not cause radiated emissions outside the requirements of the design standard and adversely affect the operation of surrounding equipment. This change will not compromise the ability of plant safety-related equipment to perform its intended function. Westinghouse calculation, WCAP-12096 shows the loop accuracy is within the previous calculated loop accuracy, therefore, with the swapover setpoint unchanged, the safety margin is not affected.

SA-SE Number: WBPLEE-98-010-1

Implementation Date: 05/08/1999

Document Type:	*Affected Documents:*	*Title:*
Design Change FSAR	DCN W-39608-A FSAR Package 1508 FSAR Package 1508S1	Containment Sump Level Transmitter Replacement

Description and Safety Assessments:

This DCN (M-39608-A) replaces the containment sump level transmitters in Unit 1. The existing transmitters are Barton transmitters with a diaphragm seal and capillary tubing. These existing transmitters have a problem with the capillary tubing leaking fill fluid, and maintaining the transmitter within calibration. The new transmitters are Class 1E qualified, do not have capillary tubing, are more accurate, and can be submersed during a LOCA. This change upgrades equipment used to perform a function. Functional performance of the plant is not affected and protective logic is not affected.

The range of the transmitter is changing from 0 to 20 feet (240") to 0 to 200" (16' & 8") which will improve instrument loop accuracy. The existing setpoint for switchover from RWST remains the same. The new transmitters' range is fully adequate to monitor the maximum equilibrium flood level, which is above the PAM requirement of 600,000 gallons.

The sump is in the lower containment, below the refueling cavity. The sump is a water source for long term recirculation for the functions of RHR, emergency core cooling, containment atmosphere cleanup, and containment long term cooling. The transmitters will be located just outside the sump in the raceway. These transmitters are associated with the protective features used to detect and mitigate the effects of Condition III & IV events associated with a LOCA. Four safety-related level transmitters (one per channel) are provided to measure the containment sump level. These transmitters provide input to allow switchover from RWST to containment sump recirculation and also provide input to PAM Category I indicators 1-LI-63-180 and -181. The four containment sump level high trip signals are combined in a 2 out of 4 circuit to produce an output that is combined with the output of the RWST low level switches. When this logic signal is made up the valves from the RWST are closed, and the containment sump becomes the water source for long term recirculation.

Implementation of this DCN requires the mounting of the new transmitters, rerouting instrument sense lines, and cables, and revising the dropping resistor at the Eagle racks. The setpoint will not change, and indication scales are not affected as they currently read in 0-100% scale.

This change upgrades existing plant equipment. The failure modes of the replacement equipment do not differ from the equipment being replaced, and common mode failure has been demonstrated not to be an issue based on experience with these types of transmitters at Sequoyah. The installed loops (equipment and cable/conduit) are separated physically and electrically. Proper separation/isolation of cable routing and equipment is maintained by the DCN and appropriate plant installation procedures. The independence of safety related equipment is not challenged. Civil calculations have been performed to verify that, when installed per the DCN, the equipment will remain able to perform its function following a seismic event. The equipment has been tested and the test report reviewed documenting the new transmitters are not to be susceptible to EMI/RFI and will not cause radiated emissions outside the requirements of the design standard and adversely affect the operation of surrounding equipment. This change will not compromise the ability of plant safety-related equipment to perform its intended function. Westinghouse calculation, WCAP-12096 shows the loop accuracy is within the previous calculated loop accuracy, therefore, with the swapover setpoint unchanged, the safety margin is not affected.

SA-SE Number: WBPLEE-98-013-0

Implementation Date: 08/06/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	W-39459-A FSAR Figure 8.2-1 FSAR Figure 8.2-3	Installation of new surge arresters.

Description and Safety Assessments:

The 500kV transmission line which terminates into the station are currently not equipped with surge arresters. Consequently, the circuit breakers terminating the lines could be exposed to lightning overvoltages (caused by lightning strokes striking the lines) in excess of their insulation capability. This safety assessment addresses the acceptability (from a nuclear safety standpoint) of DCN W-39459-A which adds surge arresters in the 500kV switchyard to line terminations for each power phase of the five 500kV lines (total of 15 new surge arresters). These new surge arresters are being installed to limit incoming surges to levels below the capability of the associated circuit breaker insulation.

This DCN will have no impact on the site's 161kV source of preferred (offsite) power or its connection to WBN's Class 1E distribution system. These 500kV line surge arresters perform no safety related functions and will be installed in an area outside of the "Plant Operations Area" in accordance with SSP 6.52 "Activities of Transmission/Power Supply at Watts Bar Nuclear Plant." UFSAR Section 15.2.7 adequately deals with the plants reaction to a loss of external electrical, load (which while not expected from this activity is the worst accident scenario that could be postulated from this change).

UFSAR Section 8.2.1.2 addresses Transmission Lines, Switchyard, and Transformers. While the UFSAR text does not address surge arresters, UFSAR Figures 8.2-1(1-75W500) and 8.2-3 (75W200) reflect the plant interface with the transmission system and will be revised by this design change to reflect the addition of the surge arresters.

These new surge arresters are being installed based on recommendations from a Watts Bar Switchyard review performed by TVA's Transmission Power Supply Group (TPS). These new surge arresters have been designed, sized, and procured in accordance with TPS's Substation and Switchyard Design Standards.

This DCN was divided into five stages to provide maximum flexibility for switchyard work options. Each stage is related to one of the existing 500kV lines and the switchyard bay in which it terminates

TPS's Security and Services section recommends that only one 500kV line be taken out of service at a time. The reason for this recommendation is because TVA's transfer capabilities with other utilities suffer if too many WBN 500kV lines are out of service due to first contingency overloads of critical 161kV lines if another 500kV line trips. To ensure that WBN's preferred offsite power supply is not affected, each stage documents coordination requirements that must be followed before lines can be taken out of service. This required coordination and approval of TPS's Transmission Maintenance Coordinator, Grid Coordinator, and Dispatch Management will assure that when the line/lines are taken out of service there will be no adverse interactions to TVA's Grid or WBN's source of preferred offsite power. With the exception of this one limitation, the stages may be worked and returned to service at the discretion of Operations and the implementing organization.

This proposed activity does not present an unreviewed safety question as the addition of surge arresters to the 500kV line terminations does not diminish the capability of the 500kV switchyard. The DCN controls the work process such that the required offsite power sources will not be challenged. The addition of these devices will improve the 500kV switchyard system by limiting incoming surges to levels below the capabilities of the associated circuit breaker insulation. Therefore:

SA-SE Number: WBPLEE-98-013-0 ***Implementation Date:*** 08/06/1998

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR was not increased and
2. The possibility of an accident or malfunction of a different type than previously evaluated was not created and
3. There was no reduction in a margin of safety as defined in the bases 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-98-014-0

Implementation Date: 04/03/1998

Document Type:	Affected Documents:	Title:
Design Change	DCN S-39835-A FSAR Figures	6.9KV Diesel Generator Schematic Diagrams

Description and Safety Assessments:

DCN S-39835-A implements drawing deviation (DD 98-0006) so that the drawings accurately reflect the as constructed plant configuration and to ensure that all design output documents agree. No hardware or functional changes are being made by this Design Change Notice.

DCN S-39835-A revises 6900V Standby Diesel Generator schematic diagrams 1-45W760-1,- 1A, -1B, and -1C per DD 98-0006 to correct the voltage regulator model numbers. Vendor Manual WBN-VTD-P318-1070 documents voltage regulator model number 72-06204-100 as a part of the stator exciter voltage regulators as model 72-06200-100. Contrary to this documentation, the drawings of concern show the voltage regulators as model number 72-05000-100. This model is supplied by the vendor to several nuclear utilities, but is not used at Watts Bar Nuclear Plant. Therefore, the vendor supplied model shall be shown on the schematic diagrams to allow agreement with the installed parts. These schematic diagrams are also FSAR Figures 8.3-14B, 8.3-14C, 8.3-14D, and 8.3-14E.

Additionally, DCN S-39835-A corrects a typographical error on 6900V Diesel Generator diagrams 1-45W760-1, -1A, -1B, and -1C in which note 6 shows a sample fuse unique identification as system 32 instead of 82. However, the Master Equipment List correctly lists the fuse identification (for fuses used within circuits covered by the schematics) as system 82.

Since these are documentation changes only and do not represent any functional, operation, or physical change to the plant, the revisions to FSAR figures 8.3-14B, 8.3-14C, 8.3-14D, and 8.3-14E by DCN S-39835-A do not impact the probability of occurrence or consequences of any accident or equipment malfunction currently evaluated in the FSAR. 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. Design basis accidents and credible failure modes as described in the FSAR are not impacted by this change. Therefore, the changes of DCN S-39835-A are acceptable from a nuclear safety standpoint and does not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-98-020-1

Implementation Date: 07/20/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39854-B FSAR Package 1511	Deletion of the real-time particulate monitoring function from the Service Building ventilation radiation monitor.

Description and Safety Assessments:

This change, DCN M-39854-B, deletes the real-time particulate monitoring function from the Service Building ventilation radiation monitor, 0-RE-90-132. This radiation monitor currently provides off-line continuous real-time detection of particulate, iodine, and noble gas radioactivity. This radiation monitor will continue to provide continuous real-time iodine and noble gas monitoring, however, a fixed filter is used for collection of particulates. This change is in response to repetitive failures of the particulate moving-filter detector assembly as documented by WBPER970582. Additionally, the sample stream flow control valve controllers will be functionally removed in order to allow manual (valve) flow balancing between the iodine and noble gas detector flow paths. The removal of the particulate moving filter detector assembly significantly changed the relative resistance of the flow paths.

The Service Building vent monitor (0-RE-090-132) continuously monitors the radioactivity release from the Service Building Vent and performs real-time detection of noble gas radioactivity as required in 10 CFR Appendix A GDC 64, 10 CFR 50 Appendix I, and meets the intent of the guidance in Appendix A of Regulatory Guide 1.21. The monitor also currently provides real-time iodine and particulate channels, although there are no requirements for such monitoring. The monitor also provides filters for the collection of iodine and particulates.

UFSAR Section 11.4.2.2.4, Ventilation Monitors and Containment Atmosphere Monitors, Service Building. Ventilation monitor and Design Criteria WB-DC-40-24 contain text and tables which states that the Service Building ventilation monitor has the capability for continuous particulate real-time monitoring. The UFSAR and Design Criteria also states that particulate real-time monitoring is not required. UFSAR Change Package 1511 has being submitted to delete the description of the particulate real-time monitoring. The DCN revises the table in the Design Criteria, accordingly. This change also revises UFSAR Figure 9.4-12.

The other functions of the Service Building ventilation monitor, real-time noble gas, real-time iodine detection and, collection of particulate and iodine for laboratory analysis, remain unchanged and unaffected.

This change does not change any equipment failure modes and does not create any new creditable failure modes. The particulate detector and particulate radiation analyzer being deleted are also being removed from the Service Building vent monitor enclosure and panel 0-M-12, respectively. The associated power and signal wiring are being removed, or determinated and left in-place. The particulate detector is being replaced by stainless steel tubing in the sample stream from the Service Building vent to the Service Building ventilation monitor enclosure noble gas detector. There are no other impacts to the sample stream (inlet and outlet) paths to the Service Building ventilation monitor. The electrical supply source to the Service Building ventilation monitor is not changed.

Deletion of the particulate real-time monitoring function of the Service Building ventilation monitor and disabling the automatic control function from the flow control valves to facilitate flow balancing simplifies the monitor, and therefore, is an improvement in equipment reliability.

There are no accidents which are evaluated in the UFSAR which take credit for the Service Building vent real-time detection of particulate activity.

SA-SE Number: WBPLEE-98-020-1 ***Implementation Date:*** 07/20/1998

This change, DCN M-39854-B, does not affect any UFSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. Technical Specifications are not affected. This change is in compliance with 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 exists.

SA-SE Number: WBPLEE-98-036-1

Implementation Date: 04/04/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M39265-A FSAR Change Package 1512 TS Bases Amendment 18 TRM Change Package TRM 98-010, TRM Revision 13	Neutron Flux Negative Rate Trip Deletion.

Description and Safety Assessments:

DCN M39265-A deletes the power range high neutron flux negative rate trip (NFRT). The only Chapter 15 accident analysis that previously took credit for the NFRT were the dropped RCCA and dropped RCCA banks events. An evaluation prepared by Westinghouse, entitled "Dropped Rod Methodology for Negative Flux Rate Trip Plants" (WCAP-10297), determined that the negative flux rate trip was only required when a dropped rod (or bank) exceeded a specific threshold value. Any rod or bank dropped which had a worth below the threshold value, would not require a reactor trip to maintain DNB limits.

An additional evaluation was performed by Westinghouse titled "Methodology for the Analysis of the Dropped Rod Event," (WCAP-11394-P-A), which determined that sufficient margin existed for all Westinghouse plant designs and fuel types, without the negative flux rate trip. The NRC has subsequently reviewed and approved the Westinghouse analysis and results and concluded that this was an acceptable analysis procedure for deletion of the negative flux rate trip function. Therefore, the Negative Flux Rate Trip is not required to maintain existing DNB limits and will be deleted at Watts Bar.

This DCN deletes the Nuclear Instrumentation Systems (NIS) negative flux rate trip circuitry by modifying existing Westinghouse hardware for the Solid State Protection System (SSPS) and NIS. This trip function was indicative of a rod cluster control assembly misalignment (dropped rod).

This modification does not install any new equipment or create any new interfaces with existing plant equipment. The previous interfaced functions (high positive flux rate trip) will undergo post modification testing prior to return to service to verify that they will perform their safety function. Therefore, there will be no increased probability of malfunction of equipment or accident previously evaluated in the FSAR. Since DNB margin is maintained even without the NFRT, the consequences of equipment malfunction or the consequences of an accident are not increased. No possibility of a malfunction of equipment of a different type or accident of a different type is created since no functions or equipment are added and no functions or equipment remaining after the change implementation are modified. A setpoint is being deleted with no remaining limits or setpoints affected, therefore, no margins to safety are decreased.

NRC approved this change in Amendment 18 of the Technical Specifications.

SA-SE Number: WBPLEE-98-047-0

Implementation Date: 04/14/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39953-A	Replace condensate system bypass valve.
FSAR Change Package	FSAR Change Package 1545	

Description and Safety Assessments:

This change, DCN M-39953-A, replaces the condensate polishing demineralizer system (CPDS) bypass valve, 1-FCV-14-3, with a remote auto/manual controlled, pneumatically operated, modulating, ball-type valve. The control loop for this valve is redesigned and changes include a new electric auto/manual controller located on the condensate polishing panel O-L-436 and a new flow transmitter located on Panel 1-L-455A.

The existing bypass valve is used to regulate the amount of condensate flow to the condensate demineralizer service vessel (CDSVs) and the amount of condensate flow bypassing the CDSVs. With the control system in auto and differential pressure less than setpoint, 1-FIC-14-3 is used for relatively moderate to high flow conditions through the CDSVs, while 1-HIC-14-3 is used to throttle the valve upon high differential pressure or when 1-HS-14-3 is in THROTTLE. Operations establishes the setpoints of both controllers dependent upon desired flow conditions. 1-BYV-14-550 is a manual valve in parallel with 1-FCV-14-3. Manual positioning of this valve is the control mechanism used for desired low flow conditions through the polishers. The bypass valve and the manual bypass valve are both butterfly-type design, and the bypass valve control loop is all pneumatic; butterfly-type valves with pneumatic control loops typically do not provide precise flow control features. Therefore, the butterfly-type bypass valve is being replaced with a ball-type valve and the control loop from the common inlet flow transmitter to the valve positioner is being changed from pneumatic to electro-pneumatic. The manual bypass valve will not normally be used for control purposes,

The function of the CPDS is to remove dissolved and suspended impurities from the secondary system. The removal of impurities and corrosion products in the secondary system reduces corrosion damage to the secondary system equipment. The CPDS is used to polish the condensate before startup, during restart, and power generation, as required. The CPDS consists of six mixed bed CDSVs, CDSV inlet and outlet valves, the common inlet to outlet header bypass valve, and the manual bypass valve. The number of inservice CDSVs varies with system conditions. Under normal plant operating conditions, condensate of good quality partially bypasses all the condensate polishers when common inlet instrumentation monitors low conductivity, low silica, and low sodium. Accumulated crud on top of the resin bed of the condensate polishers causes a pressure drop across the unit common inlet and outlet headers. Under conditions of good condensate quality or high delta pressure, most of the condensate bypasses the condensate polishers with only sufficient flow to the polishers to maintain compact resin beds.

The condensate system is used to supply sufficient quantity of feedwater to the steam generator secondary side inlet during all normal operating conditions. The subject CDSV bypass valve is installed downstream of the hotwell pumps. The condensate system and the condensate polishing demineralizer system do not serve any safety-related functions. These systems are not required to operate for safe shutdown of the plant following any design basis events.

FSAR Section 10.4.6, Subsection 10.4.6.2 provides a system description of the CPDS. Included in this section is a description of the CPDS service vessel operation in either of three modes as determined by the position of the bypass valve. UFSAR Change Package Number 1545 will revise this section to describe the revised control logic for the bypass valve.

The replacement of a butterfly-type control valve in the condensate polisher system with a remote auto/manual actuated, pneumatically operated ball-type control valve and associated control scheme change from pneumatic to electro-pneumatic does not provide a different failure mode.

SA-SE Number: WBPLEE-98-047-0 ***Implementation Date:*** 04/14/1999

A malfunction of the pneumatic control system or valve stem mechanical failure of the existing bypass valve could result in the bypass valve being in the closed position. Similarly, a malfunction of the electric auto/manual controller or valve positioner or mechanical failure of the valve stem for the replacement bypass valve could cause the bypass valve to be in the close position. The existing bypass valve failure mode design is to fail open on loss of motive power. Since the replacement valve is positioned using both electrical (signal from the transmitter/controller to the positioner and signal to the control solenoids) and pneumatic (air to the valve actuator) power, a loss of either or both motive power sources would result in the valve failing in the desired open position.

The existing manual-operated bypass valve is positioned by a handwheel to develop the desired low flow to the condensate polisher demineralizer system. The failure mode of the manual bypass valve due to a mechanical failure (i.e., stem failure, etc.) is position as is. This mechanical failure mode will be eliminated under the change implemented by DCN M-39953-A.

The CPDS bypass control valve is located in the condensate supply line downstream of the hotwell pumps. This bypass valve is used to regulate condensate flow to the condensate demineralizer service vessels during plant modes when full condensate polishing demineralizer flow is not required. The design basis events 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 open failure state of the subject bypass valve does not cause or affect either of these events. This open failure state only results in a reduction of condensate bypass flow to the CPDS. The excessive heat removal event is caused by a malfunction of one or more feedwater regulating valves. The subject bypass valve does not interact electrically or mechanically with the feedwater regulating valves. However, a failure state that causes the subject bypass valve to close would cause a loss of normal feedwater event. Approximately 60% of the normal feedwater flow is controlled by the subject bypass valve. 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 auxiliary feedwater system is used to remove stored and residual heat needed to prevent reactor coolant system (RCS) 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 auxiliary feedwater system used to mitigate its consequences.

This change, DCN M-39953A, does not affect any FSAR 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. The replacement valve is designed to fail in the desired open position upon a loss of electrical or pneumatic power. A malfunction of any of the replacement valve's control accessories could cause a LONF event, however, this failure is unlikely and this same failure type exists for other control valves located in the condensate/feedwater flow path. This change does not affect any statements in the Technical Specification. Also, 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-98-049-0

Implementation Date: 07/31/1998

Document Type:	***Affected Documents:***	***Title:***
Tech Spec Bases	Tech Spec Bases Change Number 98-013 TSB 3.3.1, Revision 17	Source Range Hi Flux at Shutdown

Description and Safety Assessments:

Technical Specification Bases Change Number 98-013 revises requirements for the monitoring function of the Source Range Neutron Monitoring channels during shutdown as described in B 3.3. 1, Reactor Trip System Instrumentation. Currently, B 3.3.1, Function 5, Source Range Neutron Flux, states that in Modes 3, 4, and 5 with the reactor shut down and with rods not capable of withdrawal, the monitoring function of the SR channels must be operable to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution. Bases Section B 3.9.3, Nuclear Instrumentation, also requires the source range channels to be available for detecting changes in core reactivity during refueling operations and specifies that visual indication and audible alarm are provided for this purpose. The proposed Bases change will require that, in addition to the control board indicators, the alarm functions associated with these channels also be available during shutdown to alert operators of an increase in reactivity. The high flux at shutdown alarm provides both audible and visual indication of an increase in neutron flux levels. Other means are also available for identification of reactivity changes, such as the audible count rate. The revised wording will make B 3.3.1 consistent with B 3.9.3.

There are no design changes associated with this Bases change which could affect the reactor trip function of the source range channels as described in the accident analyses. The discussion of the boron dilution event in FSAR section 15.2.4, Uncontrolled Boron Dilution, indicates that the high flux at shutdown alarm, as well as audible and visual count rate indications, are available for detection of this event in both shutdown and refueling modes. The plant operating instructions for shutdown and refueling operations already require this alarm to be in operation anytime the reactor is shutdown with fuel in the vessel. Thus, this Bases change is consistent with the FSAR and does not require any changes in plant operation. The time available for operator action in response to a boron dilution event is not altered. This Bases change does not involve any physical modifications to the plant and, therefore, will not result in the creation of any additional or different credible failure modes. It is also consistent with the conclusions of Safety Evaluation Report with respect to the boron dilution event.

The change does not alter any SAR evaluations (accident analysis or equipment malfunction failures) previously performed, and no new accidents or equipment malfunction failures are created. The change is consistent with the licensing basis for the source range channels. 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-98-053-0

Implementation Date: 10/14/1998

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Change Package Number 1526	Clarification of Safety Question to FSAR Section 8.1.

Description and Safety Assessments:

FSAR Section 8.1 is revised to reflect changes resulting from a complete review to the section.

1. UFSAR pages 8.1-1, 2, 3, 4, 9, 13, and 17 - Various references to the facility as a two unit plant have been changed to indicate that the facility is a one unit plant. This is an editorial change because there is no credit taken for Unit 2 portion that was not in the Unit 1 scope. Included in this category is the removal of Regulatory Guide 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit nuclear Power Plants" from the list of Regulatory Guides for which the facility meets the intent of its requirements. This regulatory guide is not applicable because its requirements address the sharing of equipment between two or more nuclear units.

2. Page 8.1-16 - This is a clarification that when diesel generator protective trips are alarmed in the main control room by groups, only the group (several contacts paralleled and sent to one window) may be compared with other groups to determine which alarmed or operated first, not the individual devices within the group. This is determined to be an editorial change that does not change the intent in text because position C1.7.2 of Regulatory Guide 1.9 R3 states that "...the surveillance system should indicate which of the emergency diesel generator protective trips has been activated first," and the FSAR states that WBN does not comply. Were this change not made, it may appear that WBN does comply.

3. Page 8. 1-2 - This is a clarification that some non-safety related loads are also supplied from the 120V ac vital distribution system.

4. Page 8.1-4 - Clarifies that the vital batteries have adequate capacity not only for the Appendix R event, but also for the longer station blackout condition with load shedding. The FSAR (Section 8.1.4) that was clarified is part of a broad general overview of the onsite power system . The FSAR section that discusses station blackout is Section 8.3.2.1.1 and correctly identifies that the vital batteries have capacity to supply required loads for a minimum of four hours with a loss of all ac power. As this does not represent an actual change to the FSAR, only a clarification to a generic section, there are no design basis accident analyses or credible failure modes that are applicable.

5. Page 8.1-6 - Clarifies which specific ANSI C57 standards are met by the design of transformers, regulators, and reactors. As this identifies the specific standards from a family of approximately 55 C57 standards, there are no design basis accident analyses or credible failure modes that are applicable.

6. Page 8.1-7 - Typographical error or minor editorial change. IPCEA changed to ICEA and NRC IE Circular subject was corrected.

7. Page 8.1-7 - Minor editorial change that reflects that NRC IE Circulars do not contain requirements.

8. Pages 8.1-7 and -8 - Clarifies that only the torque switch for the direction of travel for which there is a safety function, is bypassed.

9. Page 8.1-22 - Typographical error or minor editorial change. 24VDC was changed to 24VAC in Table 8.1-2.

SA-SE Number: WBPLEE-98-053-0 **Implementation Date:** 10/14/1998

10. Pages 8.1-10 and -18 - Clarifies that while the requirements of IEEE Std 450-1980 are met, specific sections in IEEE Std 450-1995 are used. These specific sections are extensions of the 1980 version. As the 1995 version allowed the use of a modified performance test for the vital batteries, and established an acceptance criteria for connection resistance measurements which may be established by the manufacturer's recommended limit, there are no design basis accidents or failure modes applicable.

11. Page 8.1-13 - Clarifies that the requirements of Regulatory Guide 1.63 are met as stated in Position C.2 which involves penetration qualification tests. X/R ratios in excess of 8.0 for low voltage penetrations and 15 for medium voltage penetrations were used although the FSAR wording only stated a ratio of 15 was used. The requirements of the regulatory guide were met. There are no design basis accidents or failure modes applicable to this item.

12. Page 8.1-13 - Typographical error. Regulatory Guide 1.63, Position C.4 requires that an impulse withstand test be made by applying a 1.2 x 50 micro-second test. Actual testing was 1.2 x 50 micro-second. Although the FSAR specifically stated that these requirements were met, the numerical value stated was 2 x 50 micro-seconds. There are no design basis accidents or failure modes applicable to this item.

13. Page 8.1-18 - Corrects the RIMS Accession number for the reference.

14. Pages 8.1-20 and -21 - Removes items from Table 8.1-1, Safety Loads and Functions, which have no safety function. Table 8.1-1 is addressing the safety related power system. While the CRDM Cooling Fans and the Emergency Lighting Cabinets are fed from safety power systems, they have no primary safety functions. The CRDM Cooling Fans are addressed in FSAR Section 9.4.7.2.2 and in Section 9.4.7.3, Safety Evaluation they are specifically identified as not engineered safety features. The emergency lighting cabinets are addressed in FSAR Section 9.5.3, Lighting Systems and Section 9.5.3.4, Safety Related Functions of the Lighting Systems, specifically states that "Lighting systems are classified as non-safety related." As the FSAR sections specifically addressing these loads identify them as non-safety, this change to Table 8.1-1 is considered an editorial change that does not change the intent in text.

15. Page 8.1-22 - The purpose of Table 8.1-2 is to demonstrate that for each penetration assemble, the tested short circuit symmetrical amperes is greater than the manufacturers rated short circuit symmetrical amperes, and that the rated I^2t is greater that a conservatively calculated maximum I^2t that any circuit could deliver for that penetration. The rated I^2t is the amount of energy the penetration can withstand due to an electrical low impedance fault without damage. The tested short circuit symmetrical amperes shown in Table 8.1-2 for these penetrations were changed to reflect the value in test report IPS-752, Design Qualification Report for Electrical Penetration Assemblies for Watts Bat Nuclear Plant Units 1 and 2. The changed value is more conservative (increases). The calculated short circuit symmetrical amperes for these penetrations shown in the table could not be verified. These values were replaced by the breaker ratings for conservatism. These new larger values of current were then used to calculate the new value of I^2t displayed in the last column of the table. The result is still an extremely large margin when compared to the rated I^2t. A potential design basis accident which may be impacted by this change is the double-ended guillotine severance of a reactor coolant pipe at the reactor coolant pump suction. This accident is identified in Section 6.2.1.1.1 and causes the highest blowdown rate into the containment and will result in the maximum containment pressure rise. It should be noted that the potential involvement is from the standpoint of primary containment integrity only. The change could not cause the event. The applicable credible failure mode is mechanical failure of the penetration assembly, thus breaching of primary containment. Failure of the conductor to maintain continuity is of no importance because for I^2t to come into play the conductor is faulted which would already disable the conductor.

SA-SE Number: WBPLEE-98-053-0 ***Implementation Date:*** 10/14/1998

16. Page 8.1-22 - The calculated short circuit symmetrical amperes (1200) for this penetration shown in Table 8.1-2 could not be verified. The calculation of record indicates a more conservative value of 5634 amperes. The resulting I^2t is very conservative compared to the rated I^2t of the penetration. A potential design basis accident which may be impacted by this change is the double-ended guillotine severance of a reactor coolant pipe at the reactor coolant pump suction. This accident is identified in section 6.2.1.1.1 and causes the highest blowdown rate into the containment and will result in the maximum containment pressure rise. It should be noted that the potential involvement is from the standpoint of primary containment integrity only. The change could not cause the event. The applicable credible failure mode is mechanical failure of the penetration assembly, thus breaching of primary containment. Failure to conduct is of no importance because for I^2t to come into play is when the conductor is faulted which would disable the conductor anyway.

17. Page 8.1 -11 - This information was removed because it is not needed and is irrelevant to describing how the facility meets the requirements of Regulatory Guide 1.63. There are no physical separation requirements levied for redundant protective devices. As such, there are no design basis accidents or failure modes applicable to this item.

18. Pages 8.1-19, -20, and -21 - These are minor editorial changes to the stated safety function or safety load names in Table 8.1-1 or to the stated voltage for the load.

19. Page 8.1-13 - There are no procedures that require Operations to deenergize electrical equipment inside containment which is not required when the unit is shutdown. This statement is, therefore, removed from the SAR. There is no requirement in Regulatory Guide 1.63 R2, IEEE Std 317-1976, nor IEEE Std 279-1971 for this action. As such, there are no design basis accidents or failure modes applicable to this item equipment have been met.

FSAR Section 8.1 is revised to reflect changes resulting from a complete review of the section Although these items involve equipment that is described in the FSAR (vital batteries, transformers, electrical primary containment penetrations, electrical equipment inside containment, and etc.) none of these changes degrades the 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.

SA-SE Number: WBPLEE-98-054-0

Implementation Date: 01/08/1999

Document Type:	*Affected Documents:*	*Title:*
Technical Requirements Manual	Technical Requirements Bases B.3.3.6, Revision 11	Optional channel calibration method for Loose Parts Monitoring System.

Description and Safety Assessments:

This change to the Technical Requirements Manual B 3.3.6 describes an optional channel calibration method for the Loose Parts Monitoring System (LPMS). This option uses a computer-based analytical system to provide spectral data on channel performance which can be used to demonstrate proper channel performance in lieu of simulating a loose part impact near a sensor (accelerometer). Watts Bar Design Criteria Number WB-DC-30-31 Revision 1 and UFSAR Section 7.5.7 were revised under DCN M-31241-B and UFSAR Change Package Number 1547 to allow the use of the above described option. Currently, plant procedures require the use of a mechanical impact device to simulate a mechanical impact. The mechanical impact must be made within a specified distance from the sensor. The use of this device causes maintenance personnel to be unnecessarily exposed to high radiation fields. This change provides the basis for the use of this option in plant procedures.

The LPMS provides the capability to detect acoustic disturbances indicative of loose parts within the reactor coolant system pressure boundary. The LPMS uses two sensors (accelerometers) located at each of the six natural primary system collection regions; the top and bottom plenums of the reactor vessel and the primary coolant inlet plenum to each steam generator. The system actuates a local and main control room alarm and starts a frequency modulated tape recorder upon detection of a loose part impact. An audio monitor is provided to listen to the output signal of a selected channel. The LPMS does not perform a safety related function.

Credible failure modes of proposed activity

This change does not create or impact any credible failure modes. This change provides an alternate method for verifying channel calibration. This alternate method does not interact with any safety system and does not increase the probability of equipment failure.

Accidents Evaluated as the Design Basis
The Loose Parts Monitoring System is a non-safety related system and does not affect directly or indirectly any systems relied on to detect or mitigate any design basis events.

Summary of the basis for the SE conclusions
This change to the Technical Requirements Manual Section B 3.3.6 does not affect any FSAR 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 impacted. This change is in compliance with system operational and 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-98-055-0

Implementation Date: 08/18/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39973-A FSAR Figures 9.4-17, 9.4-31 FSAR Figures 10.2-1, 10.2-3	Correction of drawing discrepancies in accordance with Drawing Deviation numbers 98-0041, 98-0040, 98-0039, and 98-0036

Description and Safety Assessments:

Design Change Notice (DCN) S-39973-A resolves several unrelated Drawing Deviations so that the drawings more accurately reflect the as-constructed plant configuration and to ensure that design documents are consistent. No hardware or functional changes are being made by this DCN.

This DCN corrects drawing discrepancies in accordance with Drawing Deviation numbers 98-0041, 98-0040, 99-0039, and 99-0036.

Specifically, this DCN makes the following changes:

Revise electrical control diagrams 1-47W610-90-4 And 1-47W610-30-4 (FSAR Figure 9.4-17) to remove the flow signal from 0-EM-90-300/C to ERFDS and the P2500 computer as shown on 1-47W610-30-4. The flow signal output to ERFDS and the P2500 computer is already shown on 1-47W610-90-4, and 0-EM-90-300/C does not provide two outputs to these computer based system. Additionally, electrical control diagram 1-47W610-90-4 is revised to change P2500 computer input point from 2704A to F2704A.

Revise electrical control diagram 1-47W610-30-2 (FSAR Figure 9.4-31) to indicate that TE-30-210Q through TE-30-210AH input to the P2500 computer instead of a recorder.

Revise electrical connection diagram 45N1678-1 to show the "to" designation for cable 1C1182 as 1-CMPT-261-R158 instead of 1-CMPT-264-R158.

Revise the Cable and Conduit Routing System (CCRS) for cable 1C1181 to indicate the system as 261, the "from ID" as 1-CMPT-261-R158, and the "to" drawing as 45N1678-5.

Revise electrical schematic 1-45W600-46-1 to correct the fuse identification for auxiliary relay rack 1-R-72, and to provide a table to identify fuses used in the stop valve circuits for both MFPT 1A and MFPT 1B.

Revise electrical control diagram 1-47W610-47-2 (FSAR Figure 10.2-3) to show the correct electrical overspeed trip setpoint of 111% of rated speed as described in FSAR Section 10.2.2. Additionally, revise the "Gen Bkr Open" trip for Turbine Trip Bus "A" and "B" to indicate that there is no dependence on time.

Revise electrical schematic 1-45W600-47-2 (FSAR Figure 10-2- 1) and System Description Document, N3-47-4002, to correctly identify the function of 1-LS-47-105 as sensing lube oil tank level.

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 FSAR figures by this DCN do not impact the probability of occurrence or consequences of any accident or equipment malfunction currently evaluated in the FSAR. 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 DCN S-39973-A do not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-98-058-0

Implementation Date: 01/12/1999

Document Type:	***Affected Documents:***	***Title:***
FSAR	SAR Change Package Number 1534	Revision of Section 8.2.

Description and Safety Assessment:

UFSAR Section 8.2 is revised to reflect changes resulting from a complete review of Section 8.2. The following listing identifies the specific changes by item number. These item numbers will be used through out the SA/SE to identify the specific change such that the specific change will not have to be duplicated multiple times within the SA/SE. Item Numbers 2, 3, 7, 9, 11, 13, 14, 14a, 15, 16, 21, 22, 23, 25, 26, 27, 28, 29, 30, 31, 32, 33, 34, 35, 39, 40, 41, 42, 45, 47, 49, 53, and 54 are corrections of non-numerical typographical errors or editorial changes that do not change intent in text. Also, Item Numbers 1, 37, 38, 44, 46, 48, 50 and 51 identify the Balance of Plant Load Shedding feature as "Historical Data." This load shedding feature is not required for Unit 1 operation and these changes do not change intent in text. As such, the above items meet the definition of minor SAR changes/corrections as stated in SPP-9.4, Section 5.0 and do not require an SA, Screening Review, or SE. These will not be addressed further.

There are no Chapter 15 Design Basis Accidents or credible failure modes associated with any of the changes being made.

1. Load shedding feature has been identified as "Historical Data." This feature is not required for Unit 1 operation. (Section 8.2.1, page 8.2-1, paragraph 5, last sentence.)

2. "transmission arrangement" was changed to "development single line" to more correctly identify the figure title. (Section 8.2.1.1, page 8.2-2, paragraph 2, second sentence.)

3. Changed the figure number from 8.2-2 to 8.2-1A which is a typographical correction. (Section 8.2.1.1, page 8.2-2, paragraph 2, second sentence.)

4. Deleted the statement identifying the Athens transmission line length "is approximately 21.78 miles long and." The exact length of this transmission line is not a significant parameter. (Section 8.2.1.1, page 8.2-2, paragraph 3, third sentence.)

5. Deleted the statement identifying the Sequoyah transmission line length "is approximately 36.41 miles long and." The exact length of this transmission line is not a significant parameter. (Section 8.2.1.1, page 8.2-2, paragraph 3, fourth sentence.)

6. Deleted the statement identifying the Watts Bar-Great Falls transmission line length "is approximately 53.12 miles long. This line is." The exact length of this line is not a significant parameter. Also, changed "terminated" to "terminates" which is a grammatical correction necessitated by the previous change. (Section 8.2.1.1, page 8.2-2, paragraph 4, first sentence.)

7. Added the word "approximately" in front of 2.87 miles. This change was made for clarification. Also, added the Figure Number "Figure 8.2-2." This figure depicts the line crossing. (Section 8.2.1.1, page 8.2-2, paragraph 4, last sentence.)

SA-SE Number: WBPLEE-98-058-0 ***Implementation Date:*** 01/12/1999

8. Deleted the statement identifying the Watts Bar-Spring City transmission line length "is approximately 7.38 miles long. It is." The exact length of this line is not a significant parameter. Also, changed "terminated" to "terminates" which is a grammatical correction necessitated by the previous change. (Section 8.2.1.1, page 8.2-3, paragraph 1, first sentence.)

9. Added the figure number "(Figure 8.2-2)." This figure number depicts the transmission system. (Section 8.2.1.1, page 8.2-3, paragraph 1, last sentence.)

10. Deleted the statement identifying the Rockwood transmission line length "is approximately 23.67 miles long and is." The exact length of this line is not a significant parameter. Also, changed "terminated" to "terminates" which is a grammatical correction necessitated by the previous change. (Section 8.2.1.1, page 8.2-3, paragraph 2, second sentence.)

11. Same as number 9 above. (Section 8.2. 1. 1, page 8.2-3. paragraph 2, fourth sentence.)

12. Deleted the statement identifying the Watts Bar-Winchester transmission line length "is approximately 76.2 miles long and." The exact length of this line is not a significant parameter. (Section 8.2.1.1, page 8.2-3, paragraph 2, fifth sentence.)

13. Same as number 9 above. (Section 8.2.1.1, page 8.2-3, paragraph 2, sixth sentence.)

14. Same as number 9 above. (Section 8.2.1.1, page 8.2-3, paragraph 3, second sentence.)

14a. Added "meet or." This change was made to clarify that in some cases the design may meet the requirements and not exceed them. (Section 8.2.1.1, page 8.2-3, paragraph 4, first sentence.)

15. Deleted the words "galloping conductors." The sentence containing the statement is addressing conductor vibration and galloping conductors are discussed in the following sentence and was not needed. (Section 8.2.1.1, page 8.2-3, paragraph 4, third sentence.)

16. Deleted the following "and the offsite transmission line routing in the vicinity of the Hydro Plant switchyard is shown on Figure 8.2-4." This figure was deleted in the initial updated FSAR. Section 8.2.1.2, page 8.2-4, Paragraph 1, second sentence.

17. "The transmission lines for CSSTs A and D and CSSTs B and C are routed to the east and west of the transformer yard respectively." was changed to "These transmission lines provide power to the nuclear plants CSSTs A and D and CSSTs B and C and are routed to the east and north of the nuclear plant transformer yard respectively." This a minor change. (Section 8.2.1.2, page 8.2-4, paragraph 1, fourth sentence.)

18. Deleted the following sentence: "Physical separation is 61 feet centerline to centerline and 32 feet 9 inches between closets parts." The physical dimensions can be determined from the referenced figure and are not needed. (Section 8.2.1.2, page 8.2-4, paragraph 2, second sentence.)

19. Deleted the following sentence: "Physical separation is 70 feet centerline to centerline and 40 feet 9 inches between closest parts." The physical dimensions can be determined from the referenced figure and does not need to be duplicated here. (Section 8.2.1.1, page 8.2-4, Paragraph 3, second sentence.)

SA-SE Number: WBPLEE-98-058-0 ***Implementation Date:*** 01/12/1999

20. Deleted the following: "The 6.9-kV start buses enter the Turbine Building spaced 8 feet, 6 inches center to centerline and continue on this spacing across the building." The physical dimensions can be determined from the referenced figure and does not need to be duplicated here. (Section 8.2.1.3, page 8.2-5, paragraph 2, third sentence.)

21. Changed the word "identical " to "similar." This is an editorial change. (Section 8.2.1.3, page 8.2-6, paragraph 4, first sentence.)

22. Added the following: "via Watts Bar Hydro Plant." This is a editorial change needed for clarification. (Section 8.2.1.5, page 8.2-6, paragraph 1, first sentence.)

23. Added the following: "at the hydro plant." This is an editorial change needed for clarification. (Section 8.2.1.5, page 8.2-6, paragraph 2, first sentence.)

24. Added the following sentence: "The two 161 kV offsite power lines to the nuclear plant are protected by two-zone step distance phase relays, breaker failure, and backup ground relays." This change is needed for clarification. (Section 8.2.1.5, page 8.2-7, paragraph 3, second sentence.)

25. Added the words "other," "connected to the Hydro plant," and "breaker failure". (Section 8.2.1.5, page 8.2-7, paragraph 3, third sentence.) These are editorial changes needed for clarification.

26. Added the words "at the hydro plant." (Section 8.2.1.5, page 8.2-7, paragraph 4, first and second sentence and also added "nuclear" to the first sentence.) These are editorial changes needed for clarification.

27. Deleted the letter "b" in the word "Bboard." (Section 8.2.1.6 under heading <u>6.9kV Common Station Switchgear C and D Control</u>, page 8.2-11, paragraph 1, fifth sentence.) This is a typographical correction.

28. Same as Item 26 above. (Section 8.2.1.5, page 8.2-7. paragraph 6, first sentence.)

29. Added the following. "at the nuclear plant," "hydro plant," "at the nuclear plant," and "at the nuclear plant." This is an editorial change needed for clarification. (Section 8.2.1.5, page 8.2-7, paragraph 8, first-second-second-second sentences respectively.)

30. Added the following "at the hydro plant" and rearranged the order of the following two sentences. "There is no appreciable disturbance on the two feeders to the common station service transformers. However, a trip after this will lock out the breaker isolating the faulted line." These are editorial changes needed for clarification. (Section 8.2.1.5, page 8.2-8, paragraph 10 Item Number 1, first sentence and new sentence Number 4 and 5.)

31. Added the following "at the hydro plant." This is an editorial change needed for clarification. (Section 8.2.1.5, page 8.2-8, paragraph 11 Item Number 2, first sentence [subject line].)

32. Added the following "in Watts Bar Hydro plant Switchyard." This change is needed for clarification. (Section 8.2.1.5, page 8.2-8, paragraph 12 Item Number 3, first sentence [subject line].)

33. Added the following "Common Station Service," "faults at the nuclear plant," (hydro switchyard)," and "(nuclear plant)." These changes are needed for clarification. (Section 8.2.1.5, page 8.2-8, paragraph 13 Item Number 4, first sentence (subject line) first two changes, second sentence last two changes.)

SA-SE Number: WBPLEE-98-058-0 ***Implementation Date:*** 01/12/1999

34. Added the following "station service," "faults at the nuclear plant," and "at the hydro plant." These changes are needed for clarification. (Section 8.2.1.5, page 8.2-8, paragraph 14 Item Number 5, first sentence (subject line) both changes.)

35. Deleted "and 8.1-2b." This reference depicts the 250Vdc Battery boards and does not show the secondaries of 6.9kV Common Switchgear C and D. Therefore, its an inappropriate reference. This is an editorial change. (Section 8.2.1.6, page 8.2-10, first sentence after sub-heading 6.9kV Common Switchgear C and D.)

36. Deleted "2. Bus Failure or Undervoltage." Reviewed the Design Criteria and the ARI and this annunciation is not required or provided. (Section 8.2.1.6, page 8.2-11, last line before the beginning of Section 8.2.1.7.)

37. Identified the following as Historical Data. "3. Load shedding Initiated." This feature is not required for Unit 1 operation. (Section 8.2.1.7, page 8.2-12, last line item before the beginning of paragraph 4.)

38. Identified the following as Historical Data. "Annunciation Number 3 is composed of a loss of voltage on either 6.9kV start bus A or B or 161 kV transmission system contingency (load shedding trip circuits are manually enabled) and both Units 1 and 2 tripped." This feature is not needed for unit 1 operation. (Section 8.2.1.7, page 8.2-12, last sentence before the beginning of Section 8.2.1.8.)

39. Changed "permitted" to "acceptable." This was an editorial change needed to ensure exact quote of Criterion 17. Section 8.2.1.8, page 8.2-12, Item Number 2 under Criterion 17, last sentence.)

40. Changed "sources" to "supplies." This is an editorial change needed to ensure exact quote of Criterion 17. (Section 8.2.1.8, page 8.2-13, Item Number 5 under Criterion 17.)

41. Changed "circuits" to "systems." This is an editorial change needed to ensure exact quote of Criterion 18. (Section 8.2.1.8, page 8.2-13, Item Number 2 under Criterion 18.)

42. Deleted "General Design." This is an editorial change needed to ensure exact quote of Regulatory Guide 1.32. (Section 8.2.1.8, page 8.2-13, paragraph 1 under Regulatory Guide 1.32, third sentence.)

43. Changed "48" to "approximately 86." This review corrected this dimension.

44. The following has been identified as Historical Data. "A load-shedding scheme is provided to reduce the BOP loads under certain conditions, but no credit is taken for load shedding in the TSS." The load shedding scheme is not required for Unit 1 operation. (Section 8.2.1.8, page 8.2-16, paragraph 4 under Functional Measures, last sentence.)

45. Changed the words "both units are" to "the unit is." WBN is a single licensed unit. This a minor change. (Section 8.2.1.8, page 8.2-17, fourth paragraph, next to last sentence.)

46. The following paragraphs are being identified as Historical Data for the same reasons as Item Number 44 above. (Section 8.2.1.8, page 8.2-17, paragraphs 5 and 6, complete paragraphs.)

47. Added the word "approximately" in front of "93%." This is an editorial change. (Section 8.2.1.8, page 8.2-18, Paragraph 10, first sentence.)

48. The following is being identified as Historical Data for the same reasons as Item Number 44 above "(except when blocked by the load shedding scheme described above)." (Section 8.2.1.8, page 8.2-18, paragraph 11, last sentence.)

SA-SE Number: WBPLEE-98-058-0 ***Implementation Date:*** 01/12/1999

49. Deleted the following: "The two preferred source circuits are, however, shared between the two nuclear units." This is not a licensed two unit plant. (Section 8.2.1.8, page 8.2-19, paragraph 15, last sentence.)

50. Identified the following as Historical Data "Provisions exist for individual testing of the BOP load-shedding circuits while maintaining the load-shedding capability of the circuit not being tested for any 161 kV grid contingency." This feature is not required for Unit 1 operation. (Section 8.2.1.8, page 8.2-20, paragraph 23, last sentence.)

51. Identified the following as Historical Data for the same reason as Item Number 44 above. "Each of the offsite preferred power sources is monitored by an undervoltage relay. In the event of a loss of voltage on either 6.9-kV start bus A or B with both units tripped, or a 161kV transmission system contingency (load shedding trip circuits are manually enabled) and both Unit 1 and Unit 2 tripped, the load-shedding scheme will be initiated. This load-shedding scheme will trip off part of the BOP loads. The alternate supply breakers on 6.9kV unit boards 1 C, 1 B, 2C, and 2B; 6.9kV RCP boards 1 C, 1 D, 2C and 2D; and 6.9kV common board A, panel 16 will be tripped and locked out. Two redundant trip and lockout circuits are provided for each circuit breaker being load-shed. These redundant circuits have coincident logic features to minimize the probability of failure to operate and spurious trips. Functional test capability is built into each load-shedding circuit. The test features allow independent testing of each circuit while maintaining the load shedding feature of the circuit not undergoing testing. The redundant load-shedding circuits will be tested periodically." (Section 8.2.2, page 8.2.22,. last paragraph.)

52. Deleted the following "This shielding has been effective for an area isokeraunic level of 55 and is reflected in the average operating record of only 3.86 flash over interruptions annually per 100 miles of line." This is weather related phenomena and would require annual update to keep current. Deleted "these" and added "due to lighting" to the sentence following the one that was deleted. (Section 8.2.1.1, page 8.2-3, last paragraph, second and third sentence.)

53. Changed "both units are" to "the unit is." WBN is a single licensed unit. This is a minor change. (Section 8.2.1, page 8.2-1, first paragraph.)

54. Added "s" to the word "line." (Section 8.2. 1. 1, page 8.2-2, fourth paragraph, second sentence.) This an editorial change needed for clarification.

FSAR Section 8.2 is revised to reflect changes resulting from a complete review of the section. The following changes that did not meet the definition of minor SAR change are as follows:

Deletion of the transmission line lengths of transmission lines that are considered part of the system and not part of the preferred offsite circuits from WB Hydro to WB Nuclear. These line lengths can still be determined from design drawings. No physical changes were actually made.

Deletion of physical dimensions from UFSAR text and correction of a physical dimension that is more conservative. The dimensions can still be determined from design drawings.

This correction better describes the routing of the preferred offsite power circuits from the Hydro Plant. No physical changes were actually made and the routing can still be determined from design drawings.

SA-SE Number: WBPLEE-98-058-0 ***Implementation Date:*** 01/12/1999

A clarification of the transmission line protection was made to enhance the description and prevent any confusion.

The deletion of the description of annunciation that does not exist. This annunciation is not required and was never provided.

The deletion of specific weather related phenomena data is not a concern because there is no unusually high thunderstorm occurrence rates involved.

Although these items involve equipment that is described in the FSAR (transmission lines, physical dimensions, transmission line electrical protection, annunciation, etc.), none of these changes degrades the 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.

SA-SE Number: WBPLEE-98-060-1

Implementation Date: 02/10/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number M-39931-A FSAR Package 1541	Emergency Response Facilities Data System (ERFDS) Upgrade

Description and Safety Assessment:

This DCN (M-39931-A) will replace the Emergency Response Facilities Data System (ERFDS) workstations in the Unit 1 Main Control Room (MCR) as well as the ERFDS workstations in the Technical Support Center (TSC). A new workstation will also be added in the TSC. The workstations in the MCR consist of a new Personal Computer (PC), which includes a central processing unit (CPU) and a monitor. The workstations in the TSC will consist of a new PC, including a CPU, monitor, and a mouse, The existing keyboards will remain since they are specially designed to support the existing ERFDS software which will be reinstalled on the new workstations. These workstations will be upgraded to the fastest CPUs currently available with new larger touch screen monitors (Liquid Crystal Displays LCDs) in the MCR and larger non-touch screen monitors (Cathode-Ray Tubes CRTs) in the TSC. The communications hub for the TSC workstations will. be replaced with dual hubs by this modification and a new fiber optic (F/0) jumper will be added in the computer room to supply the signal to the second hub added in the TSC. All UNIDs for the ERFDS monitors will also be changed from "CRT" to "MON". The "MON" description is more general and will allow any type monitor to be installed (CRT, LCD, etc..) without the need for UNID changes as technical advancements occur. The existing ERFDS line printer in the TSC will also be replaced with a new laser printer as part of this modification.

<u>ERFDS</u>
EMS acquires, processes, and displays all data to Support the assessment capabilities of the MCR, Technical Support Center (TSC) and the Emergency Operation Facility (EOF).The ERFDS also provides the safety parameter display system (SPDS) and the bypassed and inoperable status indications (BISI) system for WBN.

ERFDS is not defined as being primary safety-related and it is not required to meet the single failure criterion or be qualified to IEEE criteria for Class 1E equipment.

<u>SPDS</u>
The principal purpose and function of the Safety Parameter Display System (SPDS) is to aid control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing if abnormal conditions require corrective action by the operators to avoid a degraded core. During emergencies the SPDS serves as an aid to evaluating the current safety status of the plant executing function-oriented emergency procedures, and monitoring the impact of engineered safeguards or mitigation activities. The SPDS also operates during normal operations, continuously displaying information from which the plant safety status can be readily and reliably accessed. The SPDS is not class 1E qualified and is not powered from a class 1E power source. As such, the SPDS is electrically isolated from equipment and sensors used in safety systems.

The SPDS equipment must be installed so that it does not degrade existing safety systems. The SPDS is not a safety system but may result in an improvement to safety. Operators must be trained to respond to accidents both with and without the SPDS available. The SPDS shall be designed to provide reliable indication during all modes of plant operation, although it is not required to withstand a design basis event.

<u>BISI</u>
The BISI system is a computer based system that provides automatic indication and annunciation of the abnormal status of each ESFAS actuated component of each redundant portion of a system that performs a safety-related function. The determination of the bypassed or inoperable status of a system is left up to the reactor operator. The BISI system does not perform functions essential to safety. No operator action is required based solely on the abnormal status indication. The BISI system has no effect on plant safety systems.

SA-SE Number: WBPLEE-98-060-1 ***Implementation Date:*** 02/10/1999

This modification does not change the function of any of the systems described above. It only enhances the speed of the CPUs and size and type (CRT to LCD) of the monitors for the ERFDS workstations located in the MCR and TSC. The above systems are non-safety related and are properly isolated and separated from safety related equipment They not required to meet the single failure criterion or to be qualified to IEEE criteria for Class 1E equipment. There are no now single failures or equipment failure modes introduced by this modification.

There are no analyzed design basis accidents (DBA's) directly associated with the ERFDS workstations. However, ERFDS is designed to provide a complete data set to permit accurate assessment of the event without interfering with emergency operations in the MCR. Upgrading the ERFDS CPUs and monitors will not adversely impact any previously completed analysis of DBA's. The replacement of the ERFDS state-of-the-art workstations does not create any new accidents of any type that would represent an unreviewed safety question.

SA-SE Number: WBPLEE-98-062-0

Implementation Date: 01/26/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number W-39932-A FSAR Fig. 8.3-22	Spare breaker for 480 Volt shutdown board.

Description and Safety Assessments:

48O Volt Shutdown Board safety related breakers 1-BKR-212-Al/4D-A, 1-BKR-212-A2/2D-A, 1-BKR-212-B1/2D-B, 1-BKR-212-B2/1D-B, 2-BKR-212-Al/4D-A, 2-BKR-212-A2/2D-A. 2-BKR-2l2-Bl/2D-B and 2-BKR-212-B2/1D-B are the supply breakers for the Diesel Auxiliary Boards. DCN W-39832-A revises design documents in order to provide a spare breaker for substitution of any one of the above 480V Diesel Auxiliary Board feeder breakers. The spare breaker shall be stored in compartment 1D of 480V Shutdown Board 1B1-B, in order to incorporate it into the breaker maintenance program and to make it available for use. As a result the breaker is designated as 1- BKR-212-B1/1D-B, and it's settings, which are the same as that for all the above breakers are specified. However, since settings for spare breakers are not required to be maintained, a requirement for breaker setting and or verification of the settings is required by a note, and must be performed prior to the breaker substitution. The single-line drawing, to which the spare breaker is added, is an FSAR figure. However, this change to the FSAR figure is minor and this change is determined to be acceptable from a Nuclear Safety Standpoint.

Figure 8.3-22 is changed to reflect a spare breaker in a previously identified future compartment 1D and adding a reference note number to compartment 2D of 480V Shutdown Board 1B1-B. The note number refers the drawing user to a note concerning setting requirements that are to be made or verified prior to spare breaker substitution for the breaker in compartment 2D. Similarly the note number is added to the remaining figures listed above. The plant design basis, electrical system, board alignments, equipment function, and system operation remains unchanged. The text information presented in the FSAR was reviewed for direct and indirect effects. No change to the FSAR text is needed. As a result, the change is considered a non-significant FSAR change. The change is not a unreviewed safety question because the function of the electrical system and involved boards is unchanged, and the installation and substitution process is in accordance with previously established procedures governing installation or replacement of installed 480V breakers.

SA-SE Number: WBPLEE-98-069-0

Implementation Date: 12/16/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN-D-50003-A FSAR Change Package 1562	Recalibration of the setpoints for the level alarms in the PRT

Description and Safety Assessments:

The pressurizer relief tank (PRT) condenses and cools the discharge from the pressurizer safety and relief valves. Discharge from specific relief valves located inside and outside the containment is also piped to the PRT. The expected leakage from various reactor coolant pressure boundary components also goes to the PRT.

The high temperature alarm is intended to warn the operator of RCS leakage into the PRT. During hot weather and minor temperature excursions in containment, the ambient temperature in the vicinity of the PRT approaches and sometimes exceeds the setpoint of the high temperature alarm. Thus, the alarm comes into the control room even though the liquid temperature of the PRT has not increased due to RCS leakage. This causes a nuisance alarm in the main control room and masks any increase in temperature in the PRT that is due to the relieving of the RCS to the PRT. Increasing the temperature setpoint would prevent this from happening.

This design change, DCN-D-50003-A, revises the high temperature setpoint and normal operating range for level in the PRT. The current high temperature setpoint is 112.5 degrees F and will be recalibrated to 120 degrees F. The current operating range for level is 55.5 (low level) to 80 (high level) IN H20. This range will also be recalibrated to 87 to 80 IN H_2O. When the temperature or level in the PRT exceeds the setpoint, an alarm in the main control room is actuated and the operator takes the appropriate corrective action. The existing alarm setpoints for PRT temperature and level are also being revised in ERFDS and the P2500 Computer.

If there is an accidental depressurization of the reactor coolant system (UFSAR 15.2.12) due to an inadvertent opening of a pressurizer safety or relief valve, the discharge from the valve will go to the pressurizer relief tank. However, the PRT is not a component that is important to safety and is not required to mitigate this design basis accident and, therefore, the failure of the PRT is inconsequential to nuclear safety.

This change is revising the setpoints for the level alarms because the increase in temperature of the water reduces the cooling capacity of the PRT from that at the lower temperature alarm setpoint (112.5 degrees F). Westinghouse letter, WAT-D-10558, establishes the new level alarm setpoints; for the PRT. With the new level alarm setpoints, the PRT will have the same cooling capacity as it did prior to the change. There is no change in the function, operation, testing, maintenance or surveillance of the affected components. The system will operate as before. Therefore, there is no increase in probability or consequences of evaluated accidents and malfunctions, no possibility of 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-98-072-1

Implementation Date: 04/16/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN W-40008-B FSAR Figure 10.4-13 FSAR Figure 10.4-34 FSAR Figure 10.4-35	Auxiliary oil pump run logic.

Description and Safety Assessments:

This change, DCN 40008-B, affects two non-safety related systems: Condensate, and Heater Drains & Vents (HD&V). This change involves the logic for the auxiliary and shaft driven oil pumps servicing the three Condensate Booster Pumps (CBP), three Number 3 and two Number 7 HD&V Pumps (main pumps).

The present logic requires the main pump start command to start the auxiliary oil pump which causes the lube oil pressure to reach a predetermined value. This in turn starts the main pump which has a shaft driven oil pump. As both oil pumps are 100% rated, the auxiliary oil pump stops as the main shaft driven oil pump builds pressure. If oil pressure drops below a setpoint (an indication that the shaft driven pump is malfunctioning), the auxiliary oil pump starts again and both pumps run. During shutdown, neither oil pump runs, and the oil drains from the system, i.e., pump bearings.

This DCN changes the auxiliary pump logic to have the auxiliary oil pump run when the main pump is not running. Upon starting the main pump, the auxiliary pump stop but restart after 20 seconds if normal operating pressure is not maintained. The auxiliary pump normally is not running when the main pump runs.

FSAR Chapter 15 accidents were reviewed and none may be affected by this change. Condition II fault, Loss of Normal Feedwater, was specifically considered because loss of some of the main pumps associated with this change could precipitate or promote this accident. However, this change enhances the reliability of these pumps by providing additional lubricating oil to the main pumps bearings. Also, FSAR Sections 10.4.10.3, Heater Drains and Vents Safety Evaluation, and 10.4.7.3 Condensate System Safety Evaluation were reviewed and there are no accidents or challenges to the Rector Coolant System which may be affected by this change.

Credible failure modes of the change for the auxiliary oil pumps are the same as before the change i.e. fail to run and provide lubricating oil pressure. Credible failure modes of the main pumps also remain the same which is fail to run. The function and operation of the main pumps do not change and the relationship of the main pump to the auxiliary oil pump remain the same while the main pump is running, which is auxiliary oil pump not running.

Revision 1 to this 50.59 evaluation is to support a revision of DCN W-40008-A to the B level which divides the DCN into eight stages. Each stage consists of a complete modification of one of the auxiliary oil pumps and associated main pump logic. The revision also corrects a wiring error made to the schematic diagrams and connection diagrams when implementing the circuit logic diagrams. The circuit operating philosophy will be changed.

A unresolved safety question does not exist because the function and operation of the main pumps with respect to other plant systems remain unchanged. The reliability of the main pumps is not diminished by this change. Therefore, there is no impact to equipment important to safety.

SA-SE Number: WBPLEE-98-074-0

Implementation Date: 04/04/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change FSAR	DCN M-39767-A FSAR Change Package 1561	Electronic Governor Control Systems Obsolete Equipment Replacement.

Description and Safety Assessments:

Woodward Governor Company issued Service Alert Number 059701 in May of 1997 which advised all customers that Bodine motors used on many Woodward actuators and Motor Operated Potentiometers (MOP's) are no longer available from the manufacturer and recommended that WBN replace the MOP and the electronic governor control unit (Model 2301) for each of the diesel generator sets.

DCN M-39767-A replaces obsolete equipment related to the electric governor control scheme on all four emergency diesel generators (EDG) in the Standby Diesel Generator System. The Woodward model 2301 electric load sharing and speed control (ELSSC) unit is replaced with an updated Woodward model 2301A and the speed reference setting Woodward Motor Operated (MOP) Is replaced with an updated Woodward Digital Reference Unit (DRU). The new ELSSC requires the addition of one Magnetic Pickup and cable to the engine control panel per EDG set. The panel wiring and harness wiring for the circuits associated with the electric governor actuator transducers are also rewired to provide a continuous shield from the 2301A to the transducers.

This DCN also removes equipment that is no longer needed due to the improved design of the new 2301A and the Digital Reference Unit. This consists of a dropping resistor module that was needed to reduce the operating voltage to 24 volts for the old 2301 module and a resistor that was used to drive the generator to rated speed in emergency situations. The new 2301A operates at a higher voltage and does not require this dropping resistor module. The new 2301A also senses voltage input rather than resistance changes and does not require the resistor (FRR) to accomplish the same function. The elimination of these components provides for a better design as it simplifies the design and eliminates potential failure points.

To improve the consistency of diesel control schemes between Watts Bar Nuclear Plant and Sequoyah Nuclear Plant the following changes were made under this DCN (M-39767-A) to provide consistency between the two plants:

- A new time delay relay is added to the control circuits so that the EDG stays at rated speed for two seconds before returning to idle speed for a ten minute cool down for all normal stops.

- The local emergency stop hand switch is rewired so it is functional in all modes of operation and not just in local mode operation.

- The DRU is bypassed until the engine reaches 550 RPM to allow the engine to run at a predetermined and fixed set point for idle speed operation.

Although there is not a clear mechanism for a new failure mode and one is certainly not expected, the worst case scenario of the proposed activity that could be hypothesized would be the loss of the Emergency Diesel Generators during implementation of this modification. This scenario is adequately enveloped by FSAR accident analysis section 15.2.9 which addresses the coincident loss of onsite and external (offsite) AC power to the station. An inoperable diesel generator is also adequately covered by existing Technical Specification Section 3.8.1 for operating Modes 1-4 and Section 3.8.2 for shutdown Modes 5-6.

SA-SE Number: WBPLEE-98-074-0 **Implementation Date:** 04/04/1999

There are no new failure modes introduced to the equipment or the EDG system by the replacement of the obsolete Woodward model 2301 ELSSC and Motor Operated Potentiometer with the updated Woodward Model 2301A (ELSSC) and Digital Reference Unit. The failure of these components (both old and new) would cause the engines to increase speed until they reached the setpoint of the mechanical governor. At that point, the mechanical governor would control the engine. The failure mode of the new magnetic pickup unit is the same as the Electronic Load Sharing and Speed Control Unit in that the engine will increase speed until the mechanical governor takes control. The current FSAR analysis is bounding for the worst possible results 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

This proposed activity does not present an unreviewed safety question as the replacement of these obsolete components will enhance the EDG capabilities in the performance of its safety function of supplying emergency onsite power to all required Engineering Safety Feature Loads. These modifications will improve the noise rejection qualities of the control system to EMI/RFI interference. These modifications will allow the EDG to operate on the electronic governor control unit with the mechanical (centrifugal) governor acting as a backup. This feature will give greater assurance of compliance to technical specification diesel loading times. These modifications will increase the reliability and availability of the EDG by having current design components which are reliable and readily available as spare parts.

As discussed above, the current FSAR accident analysis in section 15.2.9 is bounding for any postulated accidents or malfunctions that could possibly be associated with this activity. There is no reduction of margin of safety as evaluated in the Technical Specification from these design improvements.

SA-SE Number: WBPLEE-98-075-0

Implementation Date: 06/14/1999

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50023-A	FSAR Review and Verification Program - Section 7.8
FSAR	FSAR Change Package 1547	

Description and Safety Assessments:

FSAR Section 7.8 is revised to reflect changes resulting from a complete review to the section. A copy of the entire section with reference number identified in included as Attachment A. Reference numbers are added to each change which correspond to the description numbers shown below. Items 1, 2, 3, 4, 5, 8, 7, 9, 10, 11, 13, 14, 15, 18, and 17 are considered editorial/clarification changes that do not change the intent of the text. The remaining items, 8 and 12, are not considered minor changes and are discussed further. Item 12 requires the issuance of Engineering Document Change (EDC) Number E-50023-A to correct Design Basis Document (DBD) Number WB-DC-30-31.

1. (FSAR Pages 7.8-1 & 7.6-2). - The RHR isolation valves interface with the RCS system is discussed in Section 7.6.2. An RCS high pressure interlock is used to prohibit opening the RHR isolation valves to prevent the overpressurization of the RHR system. The specific interlock value and associated MCR RCS high pressure alarm setpoint value is removed from this section and replaced with a functional description (i.e., RHR System design pressure limit) of these setpoint values. These specific setpoint values represent information that is considered too detailed and does not contribute to the understanding of the operation of the subject RHR isolation valves. This change does not affect any functional or operational features of the subject RHR isolation valves. The RHR isolation valve logic is shown an UFSAR Figure Number 7.6-6 Sheet 3.

2. (Page 7.6-1). - The term 'RCS' is added to the first and second paragraphs for clarification purposes. This change does not affect the content of the discussion of the RHR isolation valves as given in Section 7.6.2.

3. (Page 7.6-1). - Corrected the Figure Number associated with logic drawing 1-47W611-74-1 as referenced in the fifth paragraph. The correct Figure Number is 7.6-6, Sheet 3, Figure Nos. 7.6-7 Sheets 1 & 2, are deleted. The logic sketches depicted are a simplified version of the logic information shown on Figure Number 7.6-6. Therefore, Figures Nos. 7.6-7, Sheets 1 & 2 are deleted to avoid a possible misinterpretation of the logic information shown.

4. (Page 7.6-1). - The last sentence in the fifth paragraph discusses the RHR isolation bypass valves. The term "letdown" is deleted for text consistency. This term is not used elsewhere in this Section 7.6.2 or in other references associated with the RHR isolation bypass valves.

5. (Page 7.6-2). - In the last sentence of Section 7.6.2, a reference is made to UFSAR Section 3.11 related to environmental qualification of the RHR isolation valves, This sentence implies that the environmental qualification of the subject valves are specifically discussed. Section 3.11 discusses the environmental qualification program of which the subject valves are a part. This change makes this clarification. This change does not affect any technical issues associated with the environmental qualification of the subject valves.

SA-SE Number: WBPLEE-98-075-0 ***Implementation Date:*** 06/14/1999

6. (Page 7.6-3). - The interlock/permissive logic for the SIS accumulator isolation valves is discussed. The valves receive an automatic open signal when the RCS pressure reaches the P-11 permissive setpoint. This permissive signal is currently described as the "safety injection unblock pressure" which is not consistent with other sections of the UFSAR (i.e., Table 7.3-3). This change corrects this consistency problem. This change does not affect any functionally or operational features of the SIS accumulation valves.

7. (Page 7.6-4). - Section 7.6.6 discusses the potential spurious operation of various control valves. To preclude spurious valve operation, open and/or closed contacts were placed before and after the opening/closing coil, as required. The term *'redundant' is* used to describe the use of the above described contacts. The term redundant is used in the nuclear industry to describe two or more totally independent features. This term is changed to *'separate'* to more clearly describe this design feature. This change is considered to be a clarification and does not affect any functional or operational feature of the subject motor operated valves.

8. (Page 7.6-4). - Section 7.6.6, third paragraph, discusses the use of protective covers installed over MCR handswitch as for specified motor operated valves. This paragraph did not identity the handswitches associated with valves 1-FCV-62-98 and -99 as an exception to the use of these protective covers. This change corrects this incomplete statement.

9. (Page 7.6-4). - Section 7.6.8, third paragraph, second sentence, discusses the use of protective covers. This sentence was revised to change the verb from future to present tense. This is an editorial change and does not affect the discussion of the protective hand switch covers used in the main control room.

10. (Page 7.6-4). - Section 7.6.6, fourth paragraph, discusses the removal of motive power from specific motor operated valves during normal operation. The term *"normal operation" is* changed to "*specific modes of plant operation.*" This change is considered a clarification since the term normal operation may imply full power operation. The power removal may occur during startup activities as directed by Technical Specifications. This change does not affect any functional or operational feature of the subject motor-operated valves.

11. (Page 7.6-5). - Section 7.6.7 describes the loose parts monitoring system (LPMS). The first paragraph discusses the sensor locations and physical separation installation. Three sentences provide detailed information of sensor location such as; sensors are stud mounted on the vessel head lifting lugs, etc. This information is too detailed and does not significantly contribute to the understanding of the LPMS installed at Watts Bar. These three sentences are deleted. This change does not affect LPMS performance as described in this UFSAR section. Also, the general sensor location described in this paragraph provides adequate information related to sensor location.

SA-SE Number: WBPLEE-98-075-0 ***Implementation Date:*** 06/14/1999

12. (Page 7.6-5). - Section 7.6.7, second paragraph, discusses LPMS functional features related to Regulatory Guide (RG) 1.133 requirements. Specifically, the computer-based analytical system is described to use statistical and spectral analysis in order to demonstrate channel performance. The statistical analysis feature of this system have not provided any meaningful results. The software program that performs this statistical analysis was found to be difficult to administer without consultation with the (vendor supplied) software program's originator. Also, TVA engineering staff determined the program to be of limited benefit. Therefore, this feature is deleted from this section. The spectral analysis feature has proven to be a valuable analysis tool. This feature provides a frequency versus amplitude plot of any selected channel and provides a spectral signature of acoustic energy generated by primary loop equipment during normal plant operation. Normal plant operating noise is dynamic which provides an excellent basis for sensor operability determination. This spectral data is compared to previous data taken for each sensor in order to trend sensor operating characteristics for possible sensor degradation (i.e., changes to bandwidth/amplitude data) or complete sensor failure. Channels which exhibit repeatable spectral data is considered to be operational and capable of accurately converting acoustical energy to electrical signals for processing by the LPMS signal conditioning circuitry. The recording and evaluation of this spectral data is considered sufficient to meet commitments discussed in this section of the UFSAR related to channel performance and channel calibration requirements. Since this change reflects a deviation to commitments made in this section of the UFSAR and the SER (Supplement Number 16), this condition is documented in the PER.

13. (Page 7.6 -5). - Section 7.6.7, third paragraph, last sentence, describes the LPMS's background noise averaging feature. This feature measures the background noise signal and adjusts the impact alarm monitoring circuitry to detect acoustic energy that occurs above this background noise. The term "maximum" is used to describe this automatic sensitivity adjustment The term is replace by *'high'* to more precisely describe this feature. This change does not affect the function or operation of the LPMS as described in the section.

14. (Page 7.6-6). - Section 7.6.7, second paragraph, discusses sensor location for the secondary side monitoring. The term *"trunnion"* is deleted associated with the Steam Generator sensor installation. This information is too detailed and does not significantly contribute to the understanding of the LPMS installed at Watts Bar. The general sensor location described in this paragraph provides adequate information related to sensor location. This change does not affect LPMS performance as described in this UFSAR section.

15. (Page 7.6-6). - Section 7.6.7, fifth paragraph, identified two references used to ensure ALARA issues related to the LPMS are implemented. Reference Number 7 is removed. This change is considered to be a clarification and does not affect the function or operation of the LPMS as described in the section.

16. (Pages 7.6-6, 7.6-7, and 7.6-8). - Section 7.6.8 provides a general functional description of the RCS Cold Overpressure Mitigation System (COMS). Several sentences were changed for clarification and/or editorial purposes. The current discussion adequately describes this mitigation system, however, the changed version is more concise. This change does not affect the mitigation logic as shown of UFSAR Figure Number 7.6-5.

SA-SE Number: WBPLEE-98-075-0 ***Implementation Date:*** 06/14/1999

17. (Pages 7.6-9, 7.6-10, and 7.6-11). - Section 7.6.9 discusses the instrumentation used for switchover from injection to recirculation after a loss-of-coolant accident (LOCA). This entire section was re-written to provide a more concise description of the instrumentation and controls. The current discussion adequately describes this feature, however, this discussion is unnecessarily wordy and several topics are duplicated in Section 7.3. This change provides a more concise description of the switchover from injection to recirculation logic and references other applicable Chapter 7 sections.

There are no failure modes associated with this change. Item 8 is associated with the use of protective covers placed over specified MCR hand switches. The MCR hand switches for control valves, 1 -FCV-82-98 and -99, do not require protective covers since power is removed from the valve's power source. Thus, the protective covers are not needed for these hand switches since inadvertent actuation would not cause valve movement. Item 12 is associated with data gathering activities related to the LPMS. Specifically, a computer-based analytical system is described to use statistical and spectral analysis in order to demonstrate proper channel performance. The statistical analysis feature of this system have not provided any meaningful results. Therefore, this feature is deleted from this section. The spectral analysis feature has proven to be a valuable analysis tool. Sensors which exhibit repeatable spectral data is considered to be operational and capable of accurately converting acoustical energy to electrical signals for processing by the LPMS signal conditioning circuitry. The recording and evaluation of this spectral data is considered sufficient to meet commitments discussed in this section of the UFSAR related to channel performance and channel calibration requirements.

Therefore, the above described changes do not affect proper equipment/system operation and there are no credible failures associated with these changes.

These changes do not impact any accidents evaluated in the UFSAR. These changes do not affect the operation of any safety related equipment/system and no credible failure modes a created or changed. Therefore, these changes do not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-98-077-0

Implementation Date: 05/25/1999

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E50044-A	Review and Verification of FSAR Section 9.5.2.

Description and Safety Assessments:

FSAR Section 9.5-2 is revised to reflect changes resulting from a complete review of the section. The following changes that did not meet the definition of minor FSAR change are as follows:

- (See FSAR page 9.5-1) This is a clarification that the "Code" portion of the Codes, Alarms, and Paging system (CAPS) is not functional. Various references are made to the Codes, Alarms, and Paging system (CAPS). The Codes portion of the CAPS system is not functional. The Codes was a code call system which used specific audible tones to locate persons. The plant telephones, radios, intercoms, pocket pagers, and speaker paging are used today to locate persons. The "alarms" portion of the CAPS system is the accountability/evacuation and fire/medical alarms. Due to the many documents referring to the CAPS system, the plant documents will still refer to the CAPS system, even though the code call portion is not used. To clarify the FSAR, the word "code" will be retained and an explanation added that WBN no longer uses codes, however WBN still calls the WBN system the CAPS system. The word "alarms" will be supplemented with alarms (accountability/evacuation and fire/medical).

- (See FSAR Page 9.5-2) This is a clarification that describes the dual ac voltages which supply the Node 2 chargers. Presently only one of the sources is described.

- (See FSAR Page 9.5-3) Clarifies that the two chargers are redundant and are not normally connected in parallel except when switching from one to the other.

- (See FSAR Pages 9.5-3 and 9.5-6) Clarifies that the by pass controls are isolation devices to isolate the two operating control stations and the isolation occurs in the Communications room.

- (See FSAR Page 9.5-6) Clarifies that only two power sources exist for the three tone generator consoles.

- (See FSAR Page 9.5-7) Removes statement regarding availability and stocking of spares.

- (See FSAR Page 9.5-3) Clarifies that paging can be advanced in priority beyond the second level.

- (See FSAR Page 9.5-3) Clarifies where the alarms associated with the Codes, Alarm, and Paging system are controlled from. The alarms referred to are the assembly and accountability alarm, fire and medical alarm, all clear alarm, and paging. All of these are controlled from the MCR and auxiliary control room except the fire and medical alarm is only controlled from the MCR.

SA-SE Number: WBPLEE-98-077-0 ***Implementation Date:*** 05/25/1999

- (See FSAR Figure 9.5-19) This figure identified the CCTV equipment availability during any of the described postulated conditions. The CCTV system is a portable system, which is installed as required, and is not applicable to the postulated conditions.

- (See FSAR Page 9.5-5) Clarifies that the fiber optics equipment is similar to the microwave equipment in regard to redundancy. The fiber optics equipment is not safety related, therefore this change does not affect plant safety and is safe.

Although these items involve communications equipment that is described in the FSAR (Codes, Alarm, Paging, Chargers), none of these changes degrade 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, these changes do not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-98-080-0

Implementation Date: 06/10/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50045 FSAR Change Package 1553	Updated FSAR Review - Section 7.2

Description and Safety Assessments:

FSAR change package 1553 documents changes resulting from a review of UFSAR Section 7.2. Items 1-15 and 17-38 are corrections of non-numerical typographical errors, administrative changes, or minor editorial changes that do not change the intent. As such, they are considered to be non-significant changes as defined in Nuclear Assurance Department Procedure (NADP) 7, "FSAR Management," Section 5.0, and do not require an SA, SR or SE. Therefore, these non-significant changes will not be further addressed. Item 1 includes those non-significant changes for which no explanation is needed. The remaining items (16 and 39) are addressed in the Screening Review and Safety Evaluation. Design basis document changes are being made by EDC E-60045 to ensure consistency with existing design and licensing bases, including changes associated with the UFSAR changes described below.

16 (page 7.2-14, 34, 35): The update to Section 7.2.1.1.5 is taken from text in Section 7.2.2.3.4 with clarifications and editorial changes. The relocated discussion of the pressurizer water level instrumentation is more appropriately included in this section than Section 7.2.2.3.4, which deals with control and protection system interaction. The changes to 7.2.1.1.5 are based on a general description of the Westinghouse pressurizer level design, channel independence, and actual installation attributes found on TVA physical drawings. Also, the hydrogen gas entrainment issue documented in NRC Information Bulletin Number 92-54, Level Instrumentation Inaccuracies Caused by Rapid Depressurization, is retained and clarified. Similar clarification is made to Reactor Protection System Description N3-99-4003 Section 3.1.1.2(d). The original text in 7.2.2.3.4 provides some information that is too detailed and is not pertinent to the subject of discussion. It also includes a statement that the error effect on the level measurement during a blowdown accident would be about one inch. The basis for this value is not known; however, the worst case reference leg loss of fill error due to a rapid RCS depressurization event is no more than 12 inches elevation head. This value is based on the relative elevation difference between the condensing chamber and the reference leg sensor bellows. The channel error value discrepancy is documented in a WBN PER. The remaining text in 7.2.2.3.4 is revised to clarify the control and protection system interaction discussion.

39 (Figure 7.2-1 Sheet 3): This drawing, 1-47W611-99-6, shows time delays of 0.5 and 0.1 seconds, respectively, for Reactor Coolant Pump undervoltage (UV) and underfrequency (UF) reactor trip signals. Setpoint and Scaling Documents specify settings of 23 cycles (0,383 sec) for the UV and 5 cycles (0.087 sec) for the UF as determined by calculations WBPE0689009007 and WBPE0689009008. This discrepancy is documented in WBPER980417. The drawing will be revised by DCN E-50045.

In addition to the UFSAR changes described above, Reactor Protection System Description N3-99-4003 is revised by EDC E-50045 to clarify the design basis and functions, correct minor errors, and make editorial changes to maintain consistency between the design basis and the UFSAR. Also reference 7.5.26 of N3-99-4003 is changed from WCAP-14419 to WCAP-14738, "Revised Thermal Design Procedure Instrument Uncertainty Methodology," to reflect the current design and licensing basis for the RCS flow and reactor power calorimetrics instrument uncertainty, which became effective at cycle 2 startup. WCAP-14419 was superseded by WCAP-14738. These documents are not listed in the UFSAR or TS.

These changes do not involve any physical modifications to the plant or modify the safety function of any equipment. The changes do not alter any design basis accident or operational transient analyses previously performed, and no new accidents or equipment failure modes are created. The changes do not affect setpoints or safety limits and, thus, do 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-98-080-0 ***Implementation Date:*** 06/10/1999

FSAR change 1553 documents changes resulting from a review of Section 7.2; design change EDC E-50045 makes related documentation changes identified during the SAR review. Corrections of non-numerical typographical errors, administrative changes, or other minor editorial changes that do not change the intent are considered to be non-significant changes as defined in procedure NADP-7, and as such do not require a safety evaluation. The remaining items are as follows:

- Part of the discussion of the pressurizer water level instrumentation in Section 7.2.2.3.4 is moved to Section 7.2.1.1.5 since it is not specifically related to control and protection system interaction. The changes to this relocated discussion are based on a general description of the Westinghouse pressurizer level design, channel independence, and actual installation attributes found on TVA physical drawings. Also, the hydrogen gas entrainment issue documented in NRC Information Bulletin Number 92-54, Level Instrumentation Inaccuracies Caused by Rapid Depressurization, is retained and clarified. The original text in 7.2.2.3.4 provides some information that is too detailed and is not pertinent to the subject of discussion. It also includes a statement that the error effect on the level measurement during a blowdown accident would be about one inch. The basis for this value is not known; however, the worst case reference leg loss of fill error due to a rapid RCS depressurization event is no more than 12 inches elevation head. The value of 12 inches is based on the relative elevation difference between the condensing chamber and the reference leg sensor bellows. Instrument uncertainty calculations were based on the attributes found on the installation drawings and, therefore, the change to the UFSAR has no impact on the design basis or Technical Specifications and no protection system parameters such as setpoints or scaling are affected by the discrepancy. The remaining text in 7.2.2.3.4 is revised to clarify the control and protection system interaction discussion. The change does not require any plant modifications and is consistent with the existing design and licensing bases.

- Drawing shows time delays of 0.5 and 0.1 seconds, respectively, for Reactor Coolant Pump undervoltage (UV) and underfrequency (UF) reactor trip signals. The Setpoint and Scaling Documents (SSDs) specify settings of 23 cycles (0.383 sec) for the UV and 5 cycles (0.087 sec) for the UF as determined by calculations. The values on the drawings were not used in establishing the settings and the actual settings are based on the SSDs. No plant modifications are required. The drawing will be revised by a DCN.

These changes are documentation only and 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 design basis accident or operational transient analyses previously performed, and no new safety limits and, therefore, do 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-98-082-0

Implementation Date: 04/28/1999

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50024 FSAR Change Package 1509	Review of FSAR Section 7.5.

Description and Safety Assessments:

FSAR Section 7.5 is revised to reflect changes resulting from a complete review of the section.

Engineering Document Change (EDC) 50024 is revising design criteria documents WB-DC-30-7, "Post Accident Monitoring instrumentation", and WB-DC-30-8, "Emergency Response Facilities Data System (ERFDS)," in order to support the changes being made to the FSAR. The changes to these documents are included in the items below. Additional minor changes to WB-DC-30-7 that are not listed below are: updating reference DS-El 8.1.20 to DS-El 8.1.24, which incorporates the former design standard; correcting a typographical error for variable number 79 to indicate that it is a Type/Category 02 instead of 01; and correcting a typographical error for variable number 100 by changing "Ioding" to "Iodine."

1. (Page 7.5-9) References to the facility as a two unit plant have been changed to indicate that the facility is a one unit plant, This is an editorial change because there is no credit taken for the unit 2 portion that was not in the unit 1 scope.

2. (Page 7.5-10) This is a clarification of the operator use of the page keys and acknowledgment of alarms on the Safety Parameter Display System (SPDS) keyboard. The page keys are available to use, but the operator may not choose to use them for paging up, down, left, or right. Operators do not use keystrokes to acknowledge alarms.

3. (Page 7.5-11) This change corrects the sources of power to the SPDS. The "normal" source of power to the SPDS is derived from the 480V AC Unit Board instead of from the 480V AC Common Board, and the "maintenance" source is derived from the 480V AC Common Board instead of the 480V AC Station Unit Board. This configuration is supported by WBN configuration control drawings, The "normal", "alternate", and "maintenance" sources of power are all non-1E power sources. The Emergency Response Facility Data system (ERFDS), and subsequently the SPDS, are also non-1E systems and not primary safety related. Design criteria WB-DC-30-8 is being revised to indicate that the maintenance source of power is from the 480V AC Common Board in order to support the change to the FSAR. There are no design basis accidents associated with these systems. Credible failure modes, such as loss of power or degraded conditions in its non-redundant circuits and instrumentation, is not altered by this change.

4. (Page 7.5-14) This change corrects the statement concerning the communication between the plant computer and the ERFDS. The ERFDS can receive and display plant computer data, and ERFDS data can be received by the plant computer. The flow of data information is not only from the plant computer to the ERFDS as currently implied in the FSAR. The ERFDS and plant computer are both non-lE systems and are not primary safety related. There are no design basis accidents associated with these systems. This change does alter the configuration of either system or the interface between them or other plant systems. Therefore, there is no change in credible failures for these systems due to this change.

SA-SE Number: WBPLEE-98-082-0 ***Implementation Date:*** 04/28/1999

5. (Table 7.5-2, Sheets 1 through 18) This change is an editorial change that revises the number of sheets in the table due to the inclusion of the Regulatory Guide 1.97 R2 Deviations and Justification for Deviations.

6. (Table 7.5-2, Sheet 1) This change is an editorial change that refers to the inclusion of the R. G. 1.97 R2 Deviations in Table 7.5-2. The deviations and deviation justifications are the same as those currently found in the references of FSAR Section 7.5.

7. (Table 7.5-2, Sheet 3) This change is an editorial change that reverses the upper and lower minimum range limits for the Refueling Water Storage Tank Level PAM variable. The minimum range is not changed. Instead of reading the range as "from 100 to 0 percent", the range will be read as "from 0 to 100 percent." This change is consistent with the manner in which other ranges are listed in the table.

8. (Table 7.5-2, Sheet 6) This change is a clarification of the variable name for variable number 37. The CCS Surge Tank Level variable has a minimum range of 0 - 100 percent, which encompasses normal as well as abnormal levels. Deletion of the word "Abnormal" from the variable name provides a more appropriate name for the function of the variable. This same change is being made to design criteria WB-DC-30-7 for variable number 37 in order to support the change to the FSAR.

9. (Table 7.5-2, Sheets 19 through 38) This change is an editorial change that is listing R. G. 1.97 R2 Deviations and Deviation Justifications in Table 7.5-2. This information has been previously provided to the NRC through licensing submittals, which are currently referenced in FSAR Section 7.5. The deviation numbers, wording, and justifications are exact duplicates of information found in the FSAR Section 7.5 references. The purpose of this listing is to provide TVA with a means of revising the deviations and/or deviation justifications if required. In order to support this change to the FSAR, design criteria WB-DC-30-7 is being revised to add Deviation number 37 to Table C of Appendix C. This deviation was unintentionally omitted when the other 36 deviations were added to the design criteria.

Each of the above items are editorial with the exceptions of Items 3 and 4. Items 3 and 4 do not alter the interface between the ERFDS, SPDS, or the plant computer and safety systems and systems important to safety, nor do these items create any new interfaces with safety systems or systems important to safety. The function and operation of the ERFDS, SPDS, and plant computer with respect to plant systems remain unchanged. There are no design basis events associated with these changes, and there are no new credible failure modes created for the involved systems. Therefore, there is no impact to equipment important to safety, and no unreviewed safety question exists.

SA-SE Number: WBPLEE-98-083-0

Implementation Date: 11/24/1998

Document Type:	***Affected Documents:***	***Title:***
FSAR Change Package	FSAR Change Package Number 1564	Updated FSAR Review - Section 7.7

Description and Safety Assessments:

The review of UFSAR Section 7.7 identified several editorial changes and the three items discussed below. This evaluation addresses the revision of UFSAR Section 7.7 to reflect the required changes. The three items discussed below were documented as Problem Evaluation Report (PER) WBPER980417:

1. The first paragraph of UFSAR Page 7.7-12, specifies the Low and Low-Low insertion limit alarm equations and defines constants K_4 and K_5. The Low-Low insertion limit alarm is configured to actuate when the control bank reaches the calculated insertion limit (i.e., Z_{LL}), thus, causing constant K_5 to be set to zero. This change specifies this K_5 value and deletes a sentence related to how the K_5 value would be chosen. Also, the wording in this section is incorrect and conflicts with the wording in the first paragraph of page 7.7-11.

2. Section 7.7.2.1, discusses separation and isolation features between protective channels and control channels. This information is addressed in UFSAR Section 7.2 (Subsections 7.2.2.2 and 7.2.2.3) and will not be duplicated in this section. Therefore, the contents in this subsection will be replaced with a statement that refers to UFSAR Section 7.2.2.2 and 7.2.2.3. The sentences that identify the isolation device test voltages and cable voltage level classification are not accurate. These isolation device fault voltages are apparently taken from information contained in WCAP-7506-P-A and is too detailed. Also, it is not needed in any discussion of the application of protective channel isolation features. The test voltage values do not reflect the test reports of other protective channel isolation devices. The cable tray voltage level designations are identified in Design Criteria (DC) WB-DC-305 and UFSAR Section 8.3.1.4. The subject UFSAR statement is not consistent with the DC or Section 8.3.1.4. Therefore, this statement is removed from Section 7.7.2.1 of the UFSAR. Also, UFSAR Section 8.3.1.4 discusses related subjects such as cable routing and separation criteria, potential damage sources, Class 1E to non-Class 1E isolation features, and the degree of compliance to R. G. 1.75. Therefore, no cable voltage level routing information is required in UFSAR Section 7.7.

3. The first sentence in the third paragraph of Section 7.7.2.2 discusses the number or groups and mechanisms in each Control Bank and Shutdown Bank. This sentence is being revised to clarify that Shutdown Banks C and D each have one group as specified in System Description N3-85-003.

The changes discussed in this evaluation do not affect any plant equipment or plant operating instructions. However, the changes are needed to more accurately describe the existing plant system/component design. Two of the above items are associated with the Control Rod Drive System. The Control Rod Drive System safety function is to maintain sufficient shutdown margin (SDM) and to fully insert when required to maintain the reactor core in a subcritical condition. Items 1 and 3 above do not affect this safety function. The SDM is maintained by comparing the Control Bank Rod Position with the insertion limits specified in the COLR. The Low Insertion Limit alarm is used to identify to the operator of an approach to the insertion limit (margin of 10 steps). The Low Insertion Limit alarm is used to demonstrate compliance with Technical Specification Section 3.1.7. Also, Shutdown Bank C and D group assignment is based on the original Westinghouse design of the Control Rod Drive System. The portion of the second item is associated with isolation/separation issues between protective channels and control channels is addressed in Section 7.2 and therefore is being removed. The information in this item related to isolation device test voltages is considered too detailed and is also being removed. Also, cable voltage level classifications are addressed in UFSAR Section 8.3 and is deleted from this section to eliminate any duplication. Therefore, the above described changes do not affect proper equipment/system operation and there are no credible failures associated with these changes.

SA-SE Number: WBPLEE-98-088-0

Implementation Date: 5/03/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50059	Review and Verification of FSAR
FSAR	FSAR Change Package 1552	Section 7.1

Description and Safety Assessments:

FSAR Change Package 1552 documents changes resulting from a review of Section 7.1. Items 1-3, 5-8, 10-14, and 16-18 below are corrections of non-numerical typographical errors, administrative changes, or minor editorial changes that do not change the intent. As such, they are considered to be non-significant changes, and do not require safety evaluation. Therefore, these non-significant changes will not be further addressed. Item 1 includes those non-significant changes for which no explanation is needed. The remaining items (4, 9, 15) are addressed in this safety evaluation. Items 19 and 20 describe design basis document changes which are necessary to ensure consistency with existing design and licensing bases. These additional changes are being made by EDC E-50049 and are also addressed in this safety evaluation.

1. (Pages 7.1-2 through 7.1-7, 12, 13, 16-19; and Table 7.1-1 Sheets 1-6): These page contain minor changes which do not need explanation and, therefore, will not be addressed individually. Examples of these are addition of acronyms, correction of reference or figure numbers, addition of cross-references to other sections which address related topics, and editorial changes such as verb tense, word choice, grammatical corrections.

2. (Page 7.1-1): Change the identification of the ANS event classifications associated with normal, transient, and faulted conditions to be consistent with the definitions given in Chapter 15.

3. (Page 7.1-3): Revise the definition of hot shutdown to be consistent with the Technical Specification definition.

4. (Page 7.1-6 and Table 7.1 -1 Sheet 1): Delete Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment." This document describes acceptable methods of complying with GDC 55 and 56. There is no reference to RG 1.11 in UFSAR Section 6.2.4 which describes compliance with GDC 55 and 56. Previous revisions to the FSAR, incorporated in Amendments 52 and 69, deleted from Section 6.2.4 statements indicating that the WBN design met the requirements of the RG. Therefore, this change to Section 7.1 is consistent with previous FSAR changes and will result in consistency between UFSAR sections. There are no commitments to RG 1.11 in WBN design criteria WB-DC-30-16 and WB-DC-40-34; therefore, no design basis changes are required. This issue was documented in WBPER980417.

5. (Page 7.1-10): Delete redundant information - the design bases for the Vital Control Power System are given in Section 8.3.

6. (Page 7.1-12): Delete information which is provided in another section. The parameters which initiate safety injection are listed in Table 7.3-1.

SA-SE Number: WBPLEE-98-088-0 ***Implementation Date:*** 05/03/1999

7. (Page 7.1-12, 19): Add ISA-DS-67.04 1982, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants," as Reference 4. The methodology of both this document and the Westinghouse Setpoint Methodology for Protection Systems, WCAP-12096, which is Reference 6 of Section 7.1, were used to determine protection system setpoints. This change is consistent with Section 7.2.1.2.4, which references both of these documents as a basis for establishing setpoints.

8. (Page 7.1-13, 14): Delete redundant information - the design bases for the separation of cables and raceways of redundant circuits are given in Section 8.3.

9. (Page 7.1-17): Add the nameplate color code requirements for Post-Accident Monitoring equipment located inside the main control room (MCR). The listed color codes apply only to components located outside the MCR. This addition is consistent with the existing design basis and will not require any changes in the identification of PAM equipment in the plant.

10. (Page 7.1-19; Table 7.1-1 Sheets 3 and 5): Delete Reference 4 (WCAP 10271) and Reference 7 (WCAP 7486) since these documents are not referenced in the text of Section 7.1 and, therefore, should not be listed to conform to WBN FSAR convention. Both of these documents are, however, referenced in Notes 1 and 3 of Table 7.1-1. These notes are modified to fully identify the reference documents within the notes since there is not a list of references in the table.

11. (Table 7.1-1 Sheet 1): Delete Regulatory Guide 1.40, "Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants," and Regulatory Guide 1.73, "Qualification Tests for Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants." These documents are not applicable to the plant instrumentation and are also addressed in Section 8.1.5.3, which indicates full compliance with both of these documents.

12. (Table 7.1-1 Sheet 1): The table indicates full compliance with Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." It also refers to Note 7 of the table which then refers to Section 5.2.7 for discussion of compliance. This change eliminates the full compliance notation from the table so that compliance is discussed in only one place. Section 5.2.7 states that the leakage detection systems comply with applicable parts of GDC 30 and RG 1.45.

13. (Table 7.1-1 Sheet 2): The table indicates full compliance with IEEE Standard 308-1971, "Class 1E Power Systems for Nuclear Power Generating Stations." Compliance with this document for the electrical systems which provide power to the safety related plant instrumentation is discussed in Chapter 8. Section 8.1.5.3 indicates that the WBN electric power system design meets the intent of the standard. This change eliminates the full compliance notation from the table and adds a reference to Chapter 8 so that compliance is discussed in only one place.

14. (Table 7.1-1 Sheet 2): Add IEEE Std. 323-1971, "IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and IEEE Std. 379-1972, "IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems," and reference appropriate notes. Conformance to these standards is already provided in the table notes. IEEE 323-1971 is referenced in the discussion of compliance with Regulatory Guide 1.89 in Note 4 of the table. Compliance with IEEE 379-1972 is discussed in Note 3 of the table.

SA-SE Number: WBPLEE-98-088-0 **Implementation Date:** 05/03/1999

15. (Table 7. 1 -1 Sheet 3): Note 1 of the table discusses conformance to the periodic testing requirements of IEEE Std. 338-1971. Item 2 of the note discusses development of reliability goals and adequacy of test frequencies but does not relate the two. Although specific goals were not developed for protection system reliability, the evaluation of test intervals in WCAP-10271 Supplement 1 and WCAP-1 0271 -P-A Supplement 2, "Westinghouse Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrument System," established that the test frequencies are adequate to confirm acceptable protection system reliability consistent with risk assessment results.

16. (Table 7.1-1 Sheet 4): Note 2 of the table lists some of the equipment and functions which are not tested during operation because of the risk of plant upset and describes the basis for not testing such equipment at power. The intent of this list was to identify examples of plant equipment which is not tested at power; it was not intended that this be a complete list. This change provides clarification.

17. (Table 7.1-1 Sheet 5): This change clarifies that the test circuitry being discussed is part of the SSPS.

18. (Figure 7.1-2): Sheet 1 of the figure is drawing 45W1640. This drawing was originally one sheet but was expanded to two sheets to include additional design information. This change adds 1-45W1640-1 as sheet 2 of the figure and adds sheet 1 to the existing figure number. The drawing will added to the UFSAR per NADP-7.

In addition to the UFSAR changes described above, design basis documents (DBD) are revised by EDC E-50049 to clarify the design basis and functions, correct minor errors, and make editorial changes to maintain consistency between the DBD and the UFSAR. Specifically, System Descriptions N3-38-4002, Auxiliary Feedwater, and N3-99-4003, Reactor Protection, are revised as follows and there are no associated UFSAR changes:

19. UFSAR Section 7.1.2.2 states that exceptions to instrument sense line independence requirements will be documented in design basis documents. System Description N3-36-4002 does not document an exception for the use of common sense lines for Auxiliary Feedwater flow transmitters for steam generator loops 2 and 3 (1 -FT-3-155A and B, 1 -FT-3-147A and B). This problem is identified in WBPER980417 and is resolved by addition of an exception to N3-3B-4002. The basis for this exception was previously documented in N3E-934 by DCN P-03131 -A. This change is documentation only and no UFSAR changes are required.

20. Section 2.2.8 of N3-99-4003 is revised by EDC E-50049 to clarify the requirements for separation of redundant protection system channels (1, 11, 111, and IV) and separation of the four protection set channels from the two logic trains (A and B). The change also clarifies that the requirements apply to the Essential Safety Features Actuation System (ESFAS) as well as to the Reactor Trip System. The changes are consistent with UFSAR Section 7.1.2.2.2.

These changes do not involve any physical modifications to the plant or modify the safety function of any equipment. The changes do not alter any design basis accident 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, thus, do 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.

SA-SE Number: WBPLEE-98-089-0

Implementation Date: 05/03/1999

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50048 FSAR Change Package 1554	Review of FSAR Section 7.3.

Description and Safety Assessments:

FSAR change package 1554 documents changes resulting from a review of Section 7.3. Non-significant changes items 1-25 and 27 will not be further addressed. Item 1 includes those non-significant changes for which no explanation is needed. The remaining item (26) is addressed in this safety evaluation. Item 28 describes design basis document changes which are necessary to ensure consistency with existing design and licensing bases. This additional change is being made by EDC E-50048 and is also addressed in this safety evaluation.

1. (Pages 7.3-1-8, 11, 16, 17, 20; and Table 7.3-3 Sheet 1): These pages contain minor changes which do not need explanation and, therefore, will not be addressed individually. Examples of these are addition of acronyms, correction of reference or figure numbers, addition of cross-references to other sections which address related topics, and editorial changes such as verb tense, word choice, grammatical corrections.

2. (Page 7.3-2, 5): In the list of functions which are initiated by the Engineered Safety Features Actuation System (ESFAS) (Section 7.3.1.1.1), combine and simplify items 2 and 3, both of which describe Emergency Core Cooling System (ECCS) functions. Remaining items are renumbered. Similarly in Section 7.3.1.1.4, change safety injection to ECCS. ECCS is a more broadly descriptive term which includes safety injection. System Description N3-99-4003 is similarly revised.

3. (Page 7.3-3, 5): Also in the list of ESFAS-initiated functions in Section 7.3.1.1.1, revise and simplify Item 4, Auxiliary Feedwater (AFW), and include AFW valves since these values are also actuated by ESFAS. Similarly, add the AFW valve actuators to the list in Section 7.3.1.1.4.

4. (Page 7.3-4): Delete orifice plates from the list of device types used in the measurement of protection system variables. Orifice plates are a subset of flow elements, which are also listed. In addition, orifice plates are not used as sensors for protection system variables.

5. (Page 7.3-4): Revise Item 3 of Section 7.3.1.1.2 to simplify the discussion of valve position information available during the post-LOCA recovery period.

6. (Page 7.3-5): Clarify that, in addition to the safety injection lines, the containment spray lines also are not isolated by a Phase B containment isolation signal. This change is consistent with Section 7.3.1.1.1.

7. (Page 7.3-5): Add the Auxiliary Building Gas Treatment System, Emergency Gas Treatment System, and Motor-Operated Valve Thermal Overload Bypass to the list of equipment actuated by the ESFAS. This change is consistent with the discussions of these features in the referenced chapters of the FSAR.

SA-SE Number: WBPLEE-98-089-0 **Implementation Date:** 05/03/1999

8. (Page 7.3-6): Revise the section summarizing the generating station conditions which require protective action. The list is not intended to be a complete list of the design basis events which the protection system is designed to mitigate. The change simplifies the summary, adds feedwater line break, and adds a reference to Chapter 15 for identification of the conditions requiring protective action. System Description N3-99-4003 is similarly revised.

9. (Page 7.3-7 and Table 7.3-2, Item 3): Revise the summary of the generating station variables which are required for initiation of protective action by the ESFAS. The change simplifies the summary, eliminates repetition, and adds steam generator level and reactor coolant temperature (T_{avg}) as monitored variables. Low-low Steam Generator (SG) level starts AFW. High-high SG level initiates feedwater isolation. Low T_{avg} coincident with a reactor trip also initiates feedwater isolation. Low T_{avg}, with a note to identify the interlock with Permissive P-4 (reactor trip), is also added to Table 7.3-2, item 3, which lists the conditions that initiate Feedwater Isolation. Addition of these variables is consistent with discussions of the Main and Auxiliary Feedwater Systems in Sections 10.4.7, 10.4.9, various Chapter 15 events (e.g., Sections 15.2.10, 15.3.1, 15.4.2), and Technical Specification Bases 3.3.2 for the P-4 interlock. System Description N3-99-4003 is similarly revised to add SG level and reactor coolant temperature.

10. (Page 7.3-8): This change is a clarification, replacing loss of coolant and steamline break with a more general term, design basis events, which also includes feedwater line breaks. A reference to Chapter 15 is also added for identification of the postulated events for which the ESFAS is required to actuate.

11. (Page 7.3-8): Revise the list of typical ranges of the instrumentation required for initiation of protective action by the ESFAS. The change simplifies the summary; eliminates repetition; replaces the terms loss of coolant and steamline break with a more general term, design basis events; and adds a reference to Chapter 15. SG level and T_{avg} are added to the list since these variables actuate ESFAS as described in Item 9 above. System Description N3-99-4003 is similarly revised.

12. (Page 7.3-9): Editorial change to more accurately describe the drawings which reflect the design of the protection systems.

13. (Page 7.3-9): Revise the discussion of the failure mode and effects analysis performed for the ESFAS. This change simplifies the discussion and eliminates unnecessary detail. The reference provided in the section describes the analysis in detail.

14. (Pages 7.3-10, 12, 13): As with reactor trip channels, most ESFAS channels are designed so that loss of instrument power results in trip of the ESFAS channel, i.e., the protection system comparator output is normally energized and de-energizes to actuate. Containment spray is identified as an exception to the typical design in order to avoid spurious actuations. In addition to the containment spray function, the switchover from injection to recirculation following a safety injection is also designed so that the comparator output energizes to actuate. This change adds the switchover function as an exception consistent with the switchover discussion in Section 7.6.9.5. System Description N3-99-4003 is similarly revised.

SA-SE Number: WBPLEE-98-089-0 ***Implementation Date:*** 05/03/1999

15. (Pages 7.3-10): This change provides clarification of the manual controls provided for containment spray actuation. The discussion notes that there are two sets of switches with one set/train and two switches/set. The change is to clarify that simultaneous operation of both switches in either set will actuate containment spray in both trains (i.e., the sets are not aligned with a specific train). This is shown on Figure 9.4-30 and is described in Technical Specification Bases 3.3.2 for containment spray.

16. (Page 7.3-13): The discussion of online testing of the ESFAS and ESF actuators notes that there are exceptions to the normal test procedure which results in operation of the ESF device. This change clarifies that the exceptions are for devices which cannot be operated at power without causing plant upset. This is consistent with the more detailed discussion of an exception for such equipment in Table 7.1-1, which is referenced in this section.

17. (Page 7.3-15): Clarify that the ECCS and containment spray system tests are performed as described in the applicable sections of Chapter 6 and in accordance with the Technical Specifications. This is consistent with discussions of testing in Sections 6.2 and 6.3.

18. (Page 7.3-16): Delete momentary from the description of the main steam isolation valve control switches. The switches are rotary type with spring return, i.e., momentary, from the OPEN position; the CLOSE position is maintained. This is consistent with Figure 10.3-5 (drawing 1-47W611-1-1).

19. (Page 7.3-17): Change preferred operating position to preferred failure position in the description of the position pneumatically operated valves assume upon loss of control air. Failure mode or position is a more commonly used term when describing the response of components to loss of motive power.

20. (Page 7.3-17, 18 and Table 7.3-1): The initiating signals for AFW are moved from Section 7.3.2.3 to Table 7.3-1, which lists ESF instrumentation. A reference to the Table is added. This change also clarifies that the AFW pumps are started by trip of both Turbine-Driven Main Feedwater (MFW) pumps rather than all MFW pumps as currently stated since trip of the Standby MFW pump does not initiate AFW. This is consistent with the description of the AFW System in Section 10.4.9. This change also deletes ATWS Mitigation System Actuation Circuitry (AMSAC) from the list of AFW start signals. As described in Section 7.7.1.12, the AMSAC system is non-safety and provides a diverse means of initiating AFW and turbine trip under conditions indicative of an ATWS event. AMSAC was not designed as an Engineered Safety Feature and is not included in the ESFAS Technical Specification 3.3.2 for AFW start. Therefore, it does not belong in the Table which identifies ESF instrumentation. The change does not after the AMSAC functions of AFW start and turbine trip. The switchover from injection to recirculation and the switchover initiating signals are also added to Table 7.3-1 since they are considered to be part of the ESFAS. The listing of switchover instrumentation is consistent with the description of the switchover function in Section 7.6.9. Also numbered the notes at the bottom of the table.

21. (Page 7.3-19): This section specifies that out of service channels are placed in the trip mode except containment spray channels, which are placed in the bypass mode. Other channels are also placed in bypass when out of service, e.g., RWST and containment sump level channels, which initiate switchover from injection to recirculation. Since the Technical Specifications dictate the required mode for an out of service channel, the exception for containment spray is replaced with a reference to the Technical Specifications.

SA-SE Number: WBPLEE-98-089-0 ***Implementation Date:*** 05/03/1999

22. (Page 7.3-20): Delete the response time for generation of the protection system signal for steamline break protection since it is given earlier in the same section. Also delete the closing time of the main steam isolation valves. This information is provided in Section 10.3 and is therefore redundant.

23. (Table 7.3-2, Item 1): In the list of containment isolation (CI) initiators, delete "automatic" from "automatic safety injection" since manual SI also actuates CI Phase A as shown on Figure 7.3-3 Sheet 4.

24. (Table 7.3-2, Item 2): Clarify that the high steamline pressure rate which initiates steamline isolation is a negative rate, consistent with Table 7.3-3 (P-1 1) and Sections 15.2.13 and 15.4.2.

25. (Table 7.3-2, Item 4b): Clarify that the containment gas monitor which initiates containment vent isolation (CVI) monitors the containment purge air exhaust. Also clarify that there are a total of two channels (one per train); only one is required for actuation. These changes are consistent with Sections 9.4.6 and 11.4.2.2.6, the CVI logic shown on Figure 7.3-3 Sheet 4, design basis document N3-30RB-4002, and Technical Specification 3.3.6.

26. (Table 7.3-2, Items 4c and 4d): Auxiliary Building gas and air particulate monitor high radioactivity do not initiate CVI as indicated in this table and, therefore, are deleted from the table. The CVI signal is provided to isolate the containment purge lines on detection of high radiation in the purge exhaust lines or in the event isolation of containment is otherwise required. This is accomplished by initiating CVI on high radiation from the purge exhaust monitors or on safety injection. The Auxiliary Building gas and air particulate monitors do not initiate CVI on high radioactivity as indicated in UFSAR Table 7.3-2. Deletion of these functions from the Table is consistent with UFSAR Sections 9.4.6 and 11.4.2.2.6, the CVI logic shown on UFSAR Figure 7.3-3 Sheet 4, design basis documents, and Technical Specification 3.3.6. The change does not require any plant modifications and is consistent with the existing design and licensing bases for containment isolation.

27. (Table 7.3-3): Clarify that when Permissive P-4 is present (reactor tripped), automatic reactuation of SI can be manually blocked (i.e., after SI has been initiated and the reactor tripped) . The present wording implies that with P-4 present, SI can be manually blocked/reset before automatic actuation of SI has occurred. This is not the case as shown on Figure 7.3-3, Sheet 3. Similarly, reactuation of SI (after SI reset) cannot be blocked when the reactor is not tripped.

    In addition to the UFSAR changes described above, Reactor Protection System Description N3-99-4003 is revised by EDC E-50048 to clarify the design basis and functions, correct minor errors, and make editorial changes to maintain consistency between the design basis and the UFSAR. Also the following problem is resolved by a revision to the Auxiliary Feedwater System Description N3-3B-4002:

28. The AFW system is required to start when both Turbine-Driven Main Feedwater Pumps (MFWP) trip. System Description N3-313-4002, Section 2.2.6.1 requires this feature to meet single failure criteria and be implemented with Class 1E circuits using redundant, coincident logic. This signal is derived from a single non-safety-grade switch on each pump and, therefore, does not satisfy the design basis requirement. N3-3B-4002 is revised to identify the MFWP trip signal as an exception to these requirements. This is acceptable based on the following: The loss of feedwater indicated by the MFWPs trip would be followed by a drop in SG levels which, on reaching the low-low level setpoint, would initiate reactor trip and AFW actuation. Thus, this signal provides earlier AFW start (and decay heat removal) than would otherwise be the

SA-SE Number: WBPLEE-98-089-0 ***Implementation Date:*** 05/03/1999

case. Since AFW start by this signal is not credited in any safety analysis (i.e., not a required accident mitigation function), it is not required that protection system criteria be applied to the components which initiate the signal. The interface of this signal with the Class 1E AFW circuits is accomplished in accordance with the applicable requirements of the WBN separation criteria. The Technical Specification for AFW start (3.3.2) requires only two channels and the Bases for the function note that each MFWP is provided with one pressure switch.

These changes do not involve any physical modifications to the plant or modify the safety function of any equipment. The changes do not alter any design basis accident or transient analyses previously performed, and no new accidents or equipment failure modes are created. The changes do not affect setpoints or safety limits and, thus, do 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-98-093-1

Implementation Date: 04/03/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change FSAR and TS Bases	DCN M-39911-A FSAR Package 1560 and TS Bases 98-023 TS Bases Revision 28	P2500 Computer Replacement with new Integrated Computer System

Description and Safety Assessments:

DCN M-39911-A, replaces the obsolete Unit 1 Westinghouse P2500 Plant Process Computer and consolidates the Emergency Response Facilities Data System (ERFDS) into a new Plant Integrated Computer System (ICS). The ICS provides an operator friendly, state of the art, real time process computer system for the WBN plant operators and Emergency Operation Facility (EOF) personnel. The new ICS computer is operational and performing all functions of the old P2500 computer and most of the functions of the ERFDS, including Safety Parameter Display System (SPDS), Bypass and Inoperable Status Indication (BISI), Balance of Plant (BOP), NSSS, Communications Data Links, and RHR Mid-Loop Operation Monitoring Functions.

Essentially all "at power" design basis accidents are associated with the ICS because accident analysis assumes reactor conditions are within Technical Specification conditions. Several of these Technical Specification parameters are monitored with ICS computer software including thermal power, axial flux difference (AFD), Quadrant Power Tilt Ratio (QPTR), Rod Supervision, heatup/cooldown, and RCS inventory. The power range nuclear instrument gains are adjusted based on calorimetric calculations completed by the ICS. This ICS calorimetric calculation software will be designed, developed, and tested in accordance with TVA procedures. Test case results of the calorimetric calculation and the other calculations identified above from the old P2500 are compared to similar calculations made by the ICS as part of the validation testing. The procedure requirements include formal test cases for ICS software as well as informal supplemental testing to further demonstrate software features and challenge calculation algorithms. Therefore, there is a high degree of confidence that the ICS Technical Specification compliance calculations are correct.

The ICS is not safety related and is property isolated and separated from safety related equipment. It is designed to seismic Category 1(L)B criteria inside seismic Category 1 areas. In the event of an accident, Main Control Room (MCR), and Emergency Response Facility (ERF) personnel can use the SPDS and other aspects of the ICS as an aid to restore the plant to a safe condition. However, Operators must be trained to respond to accidents both with and without the SPDS available. The ICS was not designed to safety system criteria, and it is not used to perform functions essential to the health and safety of the public. Although the ICS indirectly provides support to safety related systems by alerting operators that an abnormal condition may exist, operators cannot procedurally take inappropriate safety-related action based solely on ICS information. There are other safety grade equipment that is provided for mitigating the events of design basis accidents.

SA-SE Number: WBPLEE-98-093-1 ***Implementation Date:*** 04/03/1999

Therefore, the new ICS will not adversely impact any previously completed analysis of DBA's. The replacement of the P2500 and limited ERFDS equipment with state-of-the-art ICS equipment does not create any new accidents of any type that would represent an unreviewed safety question.

The credible failure modes associated with the existing P2500 computer and ERFDS have not changed as a result of this modification. The new ICS is susceptible to the same failure modes as the P2500.

1. Total Loss of ICS
2. Display of Incorrect Information
3. Loss of ICS Satellite Display Stations (SDS)
4. Loss of one or more data sources

The ICS is not defined as being safety-related and it is not required to meet the single failure criterion or be qualified to IEEE criteria for Class IE equipment. The ICS is not to be used to perform functions essential to the health and safety of the public. Although the ICS indirectly provides support to safety- related systems by alerting operators that an abnormal condition may exist. Operators cannot procedurally take inappropriate safety-related action based solely on ICS information. Since it is designed to seismic Category 1(L)B criteria inside seismic Category 1 areas, it will not fail during a design basis seismic event in a manner which will adversely affect safety-related structures, system or components. Therefore, this modification does not create any new credible failure modes of any type that would represent an unreviewed safety question.

Class B and C software on the ICS was designed, developed, tested, and verified in accordance with the requirements of TVA procedure. Consequently no margin of safety as defined in the basis for Technical Specifications is reduced.

Multiple FSAR drawings are modified to show new and revised computer point IDs. Various FSAR text sections and tables are modified in a minor fashion to reflect the new ICS arrangement, expanded functions, and communications scheme.

Technical Specification Bases Section 3.3.3 "Post Accident Monitoring (PAM) Indication," Table 3.3.3-1 function 15 & 16 "Steam Generator Water Level (Wide and Narrow Range)" must be revised. This revision is minor in nature and only removes a reference to the Emergency Response Facility Data System (ERFDS). In this reference a signal was input to both the Plant Computer and the ERFDS. With the new ICS redundant inputs are no longer required and therefore, this reference was removed from this Technical Specification bases. Therefore, there is no reduction, in the margin of safety as defined in the Basis for any Technical Specifications.

Therefore, per the previous discussion DCN M-39911-A does not result in an unreviewed safety question.

SA-SE Number: WBPLEE-98-096-0

Implementation Date: 01/12/1999

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Change Package 1516	Change package submittal - complete review of Section 8.3.

Description and Safety Assessment:

FSAR Section 8.1 is revised to reflect changes resulting from a complete review to the section. Items 1, 2, 3, 4, 5, 6, 7, 9, 10, 11, 12, 13, 14, 15, 16, 17, 19, 21, 22, 23, 24, 25, 26, 32, 33, 34, 35, 36, 37, 38, 39, 40, 41, 42, 43, 45, 46, 47, 48, 49, 50, 52, 55, 57, 58, 59, 60, 61, 62, and 64 are corrections of non-numerical typographical errors or editorial changes that do not change the intent in text. As such, they meet the definition of minor FSAR changes/corrections as stated in Standard Programs and Processes (SPP) 9.4, Section 5.0 and do not require an Safety Assessment, Screening Review, or Safety Evaluation. These will not be addressed further.

1 (Pages 8.3-1, 6, 14, 27, 30, 31, 69, 70, 74, 75 and Table 8.3-2)
Various references to the facility as a two unit plant have been changed to indicate that the facility is a one unit plant. This is an editorial change because there is no credit taken for unit 2 portion that was not in the unit 1 scope. Included in this category is the removal of Regulatory Guide 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit nuclear Power Plants" from the list of Regulatory Guides for which the facility meets the intent of its requirements. This regulatory guide is not applicable because its requirements address the sharing of equipment between two or more nuclear units.

2 (Page 8.2-2 and Figure 8.3-58)
Added "(ADGU and supporting auxiliaries are not required for Unit 1 operation)". This is as an editorial clarification that no credit is taken for the Additional Diesel Generator (ADGU) and it's supporting auxiliary systems for Unit 1 operation.

3 (Pages 8.3-4, 5)
Minor editorial change to correct typographical errors. Corrected the 480 Volt Shutdown Board identifier from "lA2-Al" to 1A2-A". Changed the identifier of the 480V Pressurizer Heater Transformer from "ID" to "1D". Also added the close parenthesis to the title "480V Diesel Auxiliary Supply Board (C- S) on page 8.3-5.

4 (Page 8.3-9)
Deleted the "s" from the word "Figures" and deleted references to Figure 8.1-2b. This is an editorial change. Figure 8.1-2b depicts the tap settings and voltage limits of the Electric Power Distribution Transformers and does not show the loads connected to the secondary side of the 6.9kV Common Station Service Transformers C & D.

5 (Page 8.3-10)
Change "Table 8.3-2" to "Table 8.3-4." This is a minor change.

6 (Page 8.3-13)
Deleted "Internal combustion engines operate most reliably at the rating for which they are designed. At extended light load operation, lube oil can be expected to accumulate in the exhaust system." This is a non-significant editorial change.

7 (Page 8.3-14)
Deleted "(there is no loss of field relay in the ADGU protection scheme)." The Additional Diesel Generator (ADGU) is not required for unit 1 operation. Deleted the last sentence of the first paragraph "Also, the additional diesel generator is available to be substituted." This sentence does not provide any relevance to the subject of this paragraph.

8 (Page 8.3-14)
Deleted "(450 rpm)." This is the diesel generator idle speed which is not a critical parameter.

SA-SE Number: WBPLEE-98-096-0 ***Implementation Date:*** 01/12/1999

9 (Pages 8.3-15, 20)
Deleted the word "five." This is an editorial change. The Additional Diesel Generator is not required for Unit 1 operation. Also on page 8.3-20, deleted the word "five" in the paragraph on General Design Criteria 5.

10 (Page 8.3-16)
Added "A complete description of the diesel fuel oil system is given in Section 9.5.4" and deleted the complete original paragraph. This is an editorial change made to eliminate duplicated information presented in Section 9.5.4 of the UFSAR.

11 (Pages 8.3-16, 17)
Deleted the complete paragraph. This is an editorial change made to eliminate duplicated information presented in Section 9.5.5 of the UFSAR.

12 (Page 8.3-17)
Added "A complete description of the diesel generator air starting system is given in Section 9.5.6 and deleted everything in the original paragraph. This is an editorial change made to eliminate duplicated information presented in Section 9.5.6 of the UFSAR.

13 (Page 8.3-17)
Deleted the complete paragraph. This is an editorial change made to eliminate duplicated information presented in Section 9.5.7 of the UFSAR.

14 (Page 8.3-18)
Minor editorial change to revise the first two sentences from "There are five diesel generator battery systems, one per diesel generator. Each system is comprised of a battery, battery charger, distribution center, cabling, and cable ways." to "There is a diesel generator battery system for each diesel generator. Each system is comprised of a battery, battery charger, distribution center, and cabling."

15 (Page 8.3-18)
Minor editorial changes to clarify the purpose of the battery charger.

16 (Page 8.3-18)
Minor editorial change to revise the last sentence of the first paragraph from "They are ungrounded" to "The diesel generator control power systems are ungrounded".

17 (Page 8.3-18)
Minor editorial change to add the word "approximately" in front of 135 volt equalizing voltage, and in front of 140% of rated output.

18 (Page 8.3-18)
Revised the following sentences from "The charger is a solid-state type which converts a 3-phase 480V ac input to a nominal 125V dc output having a rated capacity of 20 amperes. Over this output current range the dc, output voltage" to "The charger is a solid-state type which converts a 3-phase 480V ac input to a nominal 125V dc output. The dc output voltage..."

19 (Page 8.3-19)
Minor editorial changes to revise the first sentence from "The diesel generator 125V dc control power system is comprised of five physically and electrically independent battery systems (see Figure 8.3-1)." To "Each diesel generator 125V dc control power system is comprised of a physically and electrically independent battery system (see Figure 8.3-1)."

SA-SE Number: WBPLEE-98-096-0 ***Implementation Date:*** 01/12/1999

20 (Page 8.3-20)
Clarifies that while the requirements of IEEE Std. 450-1990 are met, specific sections for IEEE Std. 450-1995 are used. These specific sections are extensions of the 1980 version.

21 (Page 8.3-21)
Minor editorial change to revise wording from "above paragraphs" to "previous paragraphs".

22 (Page 8.3-21)
Identifies the "Cold Dead Load Pickup and Hot Dead Load Pickup ratings," as historical data. This information was established in the initial design and testing by the vendor and will not change in the future.

23 (Pages 8.3-22, 23)
Identified all the information following the heading "Fuel Consumption Tests" as historical data. This information was established in the initial design and testing by the vendor and will not change in the future.

24 (Page 8.3-25)
Minor editorial change to add the word "maximum" in front of harmonic distortion.

25 (Page 8.3-26)
Minor editorial changes to correct a typographical error of misspelled word from "Lose" to "Loss".

26 (Page 8.3-26)
Minor editorial change to add the words "or tripped" to "open breaker" to clarify when the control room operators are alerted when molded-case breakers with alarm contacts annunciate.

27 (Page 8.3-26)
Revised the third sentence in the paragraph on Tests and Inspections from "Panel-mounted instruments monitoring the inverter will be calibrated." to "Panel-mounted instruments monitoring the inverter will provide nominal indication, compliance instruments will be calibrated." And revise the fifth sentence from "During plant power operations the vital 120V ac control power system will be periodically tested and inspected to ensure its continued capability to perform its operation." to "The vital 120V ac control power system will be periodically tested and inspected to ensure its continued capability to perform its operation."

28 (Page 8.3-27)
Added the word "normally" to the first sentence. Motors rated at or above 400 horsepower are typically supplied at 6900 volts. However, the raw cooling water pumps are rated at 450 horsepower and are supplied from the 480V Intake Pumping Station Boards A and B. As shown in electrical calculation, there is adequate starting and running voltage for these motors.

29 (Page 8.3-28)
Added the exceptions for the alternate feeders for the power system to be a coordinated selective trip system. These exceptions are for the alternate feeders from the 480V load centers to motor control centers where current limiting fuses are required to limit downstream fault currents to within equipment ratings. And for some non-safety-related breakers using Westinghouse type LS amptectors.

30 (Page 8.3-28)
Added the instantaneous pickup value for the Reactor Coolant Pumps for motor protection. Relay settings for fault contact instantaneous pickup is typically set at 3 times normal locked rotor currents. The RCP is set at 2.1 times normal locked rotor current.

SA-SE Number: WBPLEE-98-096-0 **Implementation Date:** 01/12/1999

31 (Page 8.3-28)
Added static-type overcurrent relays with long and short time settings as also providing overcurrent protection for the incoming supply breakers on 480V switchgear boards.

32 (Page 8.3-29)
Minor editorial change to replace the words "start bus" with "boards."

33 (Page 8.3-29) Minor editorial change to delete the word "therefore."

34 (Page 8.3-34)
Minor editorial change to correct a typographical error to change the letter "O" to the number zero.

35 (Page 8.3-34)
Added General Design Criteria numbers 1-5, 17, and 18 to Section 8.3.1.2.2 to clarify which specific criteria the 120V ac Class 1E Electrical Systems will meet.

36 (Page 8.3-34)
Deleted the sentence "The total design load for each board is listed in Table 8.3-11."

37 (Page 8.3-37)
Added the words "or less" to clarify the valve closure time used for containment isolation.

38 (Page 8.3-38)
Minor editorial change to remove the hyphen in the word penetration.

39 (Page 8.3-39)
Deleted the words "outboard end of each." These words are not needed to identify where the penetration header plate weld rings are to be field welded to the containment nozzles.

40 (Page 8.3-39)
Minor editorial change to correct a typographical error of the word "scaled." This word should be "sealed." Also deleted the length of the penetration feed throughs.

41 (Page 8.3-39)
Minor editorial change to delete the words "spade type".

42 (Page 8.3-40)
Minor editorial change to correct the referenced section of the FSAR that provides a description of the manholes and duct runs.

43 (Page 8.3-41)
Minor editorial change to delete a duplicate punctuation point.

44 (Page 8.3-42)
Revised the description of how the mimic buses and switch modules on the control boards in the main control room are color coded and identified as either safety-related or not. These changes are made to clarify how the requirements of design standards are implemented.

45 (Page 8.3-43)
Minor editorial change to clarify the section providing additional information for cable qualification type tests.

SA-SE Number: WBPLEE-98-096-0 **Implementation Date:** 01/12/1999

46 (Page 8.3-43)
This is an editorial change that deletes the statement "Circuit breakers are used for high speed clearing of faults to prevent damage to the 3-phase power cables." Section 8.3.1.4.1 describes cable derating and raceway fill and does not discuss cable or circuit protection. Similar statements are discussed elsewhere in Section 8.3 of the UFSAR. Therefore, the sentence is irrelevant to Section 8.3.1.4.1 and is not needed.

47 (Page 8.3-47)
The information was removed because the handling and storage requirements of combustible materials is discussed in the Watts Bar Fire Protection Report and is not needed in this section addressing separation requirements in the auxiliary instrument room. This is an editorial change to eliminate duplicated information.

48 (Pages 8.3-51, 52)
Delete reference to Section 9.5.1. This section was removed and referenced to the fire protection report.

49 (Page 8.3-51)
Added reference to figure number 8.3-3. This figure depicts locations of 480V boards.

50 (Page 8.3-52)
Deleted the statement identifying the separation distance between the vital inverters "by a distance of 60 feet." The physical dimensions can be determined from the referenced figure and is not needed.

51 (Page 8.3-53)
Corrected the tolerance dimension to -1/2 inch for vertically stacked trays, and -1 inch tolerance for trays installed side by side.

52 (Page 8.3-53)
Minor editorial change to correct the maximum width of the trays used at WBN from 24 inches to 30 inches.

53 (Page 8.3-53)
Added the sentence: "or the bottom tray has a top cover." This change clarifies separation requirements between vertical tray stacks to be in agreement with Watts Bar Design Criteria WB-DC-30-4.

54 (Page 8.3-60)
Replaced the description of the steps for substitution of the fifth vital battery for a primary battery with the following: "The process for substituting the fifth vital battery for a primary battery is administratively controlled through plant operating procedures." The actual procedural process for substituting the fifth vital battery is an approximate 21 step process.

55 (Page 8.3-63)
Minor editorial change to add the word "battery" to clarify the room location.

56 (Page 8.3-64)
Added statement "following a 30 minute alternating current power outage." This was added to clarify the requirements in Watts Bar Design Criteria, WB-DC-30-27, Section 6.1 and Regulatory Guide 1.32.

57 (Page 8.3-65)
Revised paragraph to clarify ratings of the vital battery chargers I, II, III, IV, VI, and VII, and added the ratings of Charger V for the fifth vital battery.

SA-SE Number: WBPLEE-98-096-0 **Implementation Date:** 01/12/1999

58 (Page 8.3-65)
Changed the charger output voltage adjustable range from "129 - 140 volts" to "125 - 140 volts" to agree with actual equipment ratings.

59 (Page 8.3-68)
Minor editorial change to the revision level of applicable regulatory guides.

60 (Page 8.3-68)
Minor editorial change to clarify the number of cells for each vital battery.

61 (Page 8.3-71)
Changed the figure numbers from "8.3-37 through 8.3-40" to "8.3-47 through 8.3-50" which is a typographical error.

62 (Table 8.3-2)
Minor editorial change to delete the word "Containment" and add the word "Reactor."

63 (Tables 8.3-4, -5, -6, -7, -8)
Deleted the columns showing the connected loads and only show the board ratings.

64 (Page 8.3-67)
Revised the words "will be" to "have been" in the second sentence of the third paragraph for clarification. Fuses were verified during the pre-operational test program and are under configuration control. No periodic inspections to verify the size and types specified on the single-line diagram are required to be performed per the Technical Specifications.

Accidents Evaluated as the Design Basis

The proposed changes being made to Items 8, 18, 20, 27, 28, 29, 31, 44, 51, 53, 54, 56, and 63 do not result in any changes in the design, material and construction standards of equipment important to safety. These changes do not involve any physical modification to the plant or modify the safety function of any equipment. These changes do not alter any design basis accident or operational transient analysis previously performed, and no new accidents or equipment malfunction failures are created. The consequences and probability of accidents previously evaluated are not affected.

Item 30 is associated with the motor protection for fault conditions of the Reactor Coolant pumps. The accidents associated with the Reactor Coolant Pumps involve: Operation with permissible deviations (Condition I events), Partial loss of Forced Reactor Coolant Flow (Condition II events), Complete loss of Forced Reactor Coolant Flow (Condition III events), and Single Reactor Coolant Pump Locked Rotor (Condition IV events). The changes being made do not impact any of the accidents previously evaluated in the UFSAR.

FSAR Section 8.3 is revised to reflect changes resulting from a complete review of the section. The following changes that did not meet the definition of minor SAR change are as follows:

- Removed the value listed for the idle speed of the diesel generator. This value is not a critical parameter.
- Removed the vital battery chargers rated output value from Section 8.3.1.1. This information is duplicated in Section 8.3.2.1.1
- Clarification that while the requirements of IEEE Std 450-1980 are met, specific sections in IEEE Std 450-1995 are used.
- Added clarification as to what voltage level normally supplies motive power to motors 400 Horsepower and above.
- Added clarification identifying those exceptions for the power system to be a coordinated selective trip system as defined in the design basis document.
- Added clarification for the fault contact pickup values for motor protection.
- Clarified the relay types that provide overcurrent protection of 480V Switchgear Boards.
- Revised the description of how mimic buses and switch modules on the control boards in the main control room are color coded to identify if they are safety related or not.

SA-SE Number: WBPLEE-98-096-0 ***Implementation Date:*** 01/12/1999

- Corrected typographical errors on separation requirements.
- Clarified the process for substitution of the fifth vital battery for a primary vital battery.
- Added clarification of the normal battery charging from the design discharge condition.
- Removed the connected loads values presented in UFSAR Tables 8.3-4, -5, -6, -7, and -8.
- These tables now reflect the board ratings. The actual loads are reflected in the appropriate issued electrical calculation.

These changes are documentation only and 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 design basis accident 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, do 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-98-101-0

Implementation Date: 05/07/1999

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50065-A FSAR Change Package 1551	Review of FSAR Section 5.6.

Description and Safety Assessments:

FSAR Section 5.6 is revised to reflect changes resulting from a complete review to the section. Items A, B, C, D, E, F, G, H, I, J, K, L, M, N, Q, R, S, T, and U are editorial changes that do not change the intent in the text. Therefore, they meet the definition of minor FSAR changes/corrections and do not require a safety evaluation. These items will not be addressed further.

Engineering Document Change (EDC) E-50065-A is revising system description N3-68-4001, "Reactor Coolant System," in order to support the changes being made to the FSAR. The changes to this document is included in Item P below.

A (See FSAR page 5.6-1) This is a editorial change to state that FSAR Figure 5.1 -1 includes the instrumentation and control diagram for the Reactor Coolant System (RCS). This is also a editorial change that FSAR Figure 5-5-4 includes the instrumentation and control diagram for the Residual Heat Removal System (RHRS).

B (FSAR page 5.6-1) This is an editorial change to remove redundant information.

C (FSAR page 5.6-1) This is an editorial change in word tenses.

D (FSAR page 5.6-1) This is a editorial change which adds the appropriate reference for each item.

E (FSAR page 5.6-2) This is an editorial change which removed the thermal time constant of the thermowell and RTD. The time constant is not in the Safety Evaluation Report (SER) and is not required as part of the licensing basis. The actual response time for this function is included within the SER. In addition, an editorial change to the paragraph structure was made to group similar subjects.

F (FSAR page 5.6-2) This is an editorial change which added information that the ΔT and T_{avg} for each loop are displayed on the main control board, presently, it is stated that only T_{avg} is displayed.

G (FSAR page 5.6-2) This is an editorial change which modified the information to include indicators along with recorders as receiving signals from temperature detectors. This is also an editorial change to remove the sentence which stated the indicators are provided on the control board. In the same paragraph it states that recorders and indicators are used by the operator, therefore, it is understood that the indicators are on the control boards.

H (FSAR page 5.6-4) This is an editorial change to change the intended word in the sentence from “type" to "temperature."

SA-SE Number: WBPLEE-98-101-0 **Implementation Date:** 05/07/1999

I (FSAR page 5.6-5) This is an editorial change which restructured the paragraph to include the fact that one of the functions of the pressurizer pressure transmitters was to initiate safety injection on low pressure.

J (FSAR page 5.6-5) This is an editorial change to change a word to lower case.

K (FSAR page 5.6-5) This is an editorial change that changed the reference for low pressurizer pressure safety injection.

L (FSAR page 5.6-5) This is an editorial change to remove redundant information.

M (FSAR page 5.6-5) This is an editorial change to remove the location for the dead weight tester. This equipment is non-permanent and is only installed during calibrations. The indication from the dead weight testing is generally not used in favor of other methods for calibration.

N (FSAR page 5.6-5) This is an editorial change that clarified that five wide-range pressure transmitters are provided to monitor the RCS hot leg pressure instead of loop pressure. Clarified how the PAM transmitter sense lines are shared with the RVLIS system.

0 (FSAR page 5.6-6) This is an editorial change to changed word from "these" to "the" for the PAM transmitters.

P (FSAR page 5.6-6) Clarified that the purpose of the pressurizer relief tank pressure transmitter provided a signal to isolate the tank from the waste processing system vent header when the pressure relief tank pressure exceeded a certain setpoint to prevent over pressurizing the vent header. The transmitter is 1-PT-68-301 and is quality related due to it being mounted in a Seismic Category 1 LB manner. The transmitter provides a signal to 1-PS-68-301 to isolate (close) non-safety related valve 1-PCV-68-301. There is no adverse impact on nuclear safety. There are no design basis accidents associated with these components. This change does not alter the configuration of the RCS system or the interface between the RCS and other plant systems. Therefore, there is no change in credible failures for these systems due to this change.

Q (FSAR page 5.6-6) This is an editorial change that moved the description of the RCP oil reservoir liquid level measurement from the pressure section to the liquid level section.

R (FSAR page 5.6-6) This is an editorial change that clarified that the function of the oil lift pressure switch was to prevent the starting of the RCP motor until the oil lift pump developed the required pressure.

S (FSAR page 5.6-7) This is an editorial change that revised the sentence structure and added reference sections for pressurizer water level.

T (FSAR page 5.6-7) This is an editorial change that revised the description of the functions for two of the three pressurizer water level signals.

SA-SE Number: WBPLEE-98-101-0 ***Implementation Date:*** 05/07/1999

U (FSAR page 5.6-8) This is an editorial change that clarified details related to the RHR system instrumentation. These clarifications include: (a) Identified that there are multiple RHR pump bypass lines; (b) Identified that there is a RHR heat exchanger outlet valve being controlled; (c) Identified that 'S' means safety injection.

Item P does not change any design basis or accident analysis, nor is it associated with a specific accident.

This change updates UFSAR Section 5.6 based on a general review for clarity, accuracy, and completeness. Identified changes are considered editorial which meets the definition of minor FSAR changes/corrections. This UFSAR change does not affect any FSAR 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. The WBN Technical Specification is not impacted by this change. 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-98-114-0

Implementation Date: 12/17/1998

Document Type:	***Affected Documents:***	***Title:***
Technical Requirements Manual	TRM Change Package Number 98-025 Revision 10	Ice Condenser Inlet Door Position Monitoring System

Description and Safety Assessments:

This change, Technical Requirements Manual (TRM) Change Package Number 98-025, makes a revision to the TRM Bases Section Number B3.6.2, Technical Surveillance Requirements (TSR) 3.6.2.1. This section provides the technical surveillance requirement for the performance of a channel check for the Ice Condenser Inlet Door Position Monitoring System. Also, this change updates TRM Bases 3.6.2, Background to specifically identify the Inlet Door Position Monitoring System as the open/shut indication display panel. This is a clarification and does not warrant any further discussion.

The Ice Condenser Inlet Door Position Monitoring System involves six (6) sets of status indication lights (green and red) located in the main control room (MCR) on panel 1-M-10. These status lights provide inlet door position information. Each green/red status light set, monitors a zone. There are six zones (A through F) with eight doors in each zone for a total of 48 doors. Inlet door mounted zone switches provide the necessary switching action (i.e., contact action - open/closed) to complete the circuit and illuminate the appropriate status light The zone switch contacts are configured for each status light set to illuminate a green light if all 8 doors are closed (the 8 "closed" contacts are wired in a series configuration) and to illuminate a red light if any of the 8 doors are open (the 8 "open" contacts are wired in a parallel configuration).

The MCR Inlet Door Annunciation Circuit consists of a parallel contact arrangement of all 48 inlet door zone switches. With all inlet doors closed, the zone switch contacts are open, thus, creating an equivalent open circuit. If one or more inlet door zone switch(s) detect an open door, the associated contact(s) close, thus providing a closed circuit (electrical current flow) and a corresponding annunciation in the MCR (window number 144A). This Inlet Door Annunciation Circuit and the above described Inlet Door Position Monitoring System uses separate inlet door zone switches and interrogation supply power.

This change allows an alternate method for performance of the channel check requirement for the Inlet Door Position Monitoring System described above. Currently, the Inlet Door Annunciation Circuit is used as a diverse measurement of the inlet door position. However, if the subject Inlet Door Annunciation Circuit is inoperable, an alternate channel check method is needed. This alternate channel check method consists of a Inlet Door Annunciation Circuit continuity measurement to ascertain the position of the inlet door zone switch contacts. Specifically, the alarm circuit cables will be temporarily determinated to electrically isolate the annunciator (Ronan) interrogation voltage. A 24V DC (nominal) power supply source and an incandescent light bulb will be connected to the two conductors of the common field cable at MCR panel 1-M-21 and used to determine if continuity exists in the zone switch contact circuit (48 contacts connected in parallel). If one or more zone switch contacts are closed [which indicates open door(s)], the total equivalent resistance will be sufficiently low to allow current flow to cause the incandescent light bulb to illuminate. The bulb should be rated at between .20 amps and .30 amps. This rating allows, the bulb to illuminate at 25% of its nominal value if the equivalent cable resistance is as high as 40 ohms. The calculated worst case cable resistance is <20 ohms based on the design routing of approximately 3600 ft at 5.5 ohms per 1000 ft. If all contacts are open (which indicates all doors are closed) the total equivalent resistance should be greater than 50K ohms. In this situation the total current flow would be I=V / R or 24V / 50K ohms = .00048 amps. This value is .00048 amps/.25 amps =.2% of rated bulb current. This value would not result in a detectable bulb illumination. This method provides a reasonable assurance of inlet door position and may be used as a channel check to the Inlet Door Position Monitoring System.

SA-SE Number: WBPLEE-98-114-0 ***Implementation Date:*** 12/17/1998

The Ice Condenser Inlet Doors (48 total) form the barrier to minimize air flow between the reactor lower compartment and the ice beds. Periodic monitoring of the inlet door position during operational Modes 1, 2, 3, and 4 is important to prevent long term degradation of the ice bed inventory due to ice melting or sublimation. In the event of a design basis accident, (DBA), the inlet doors open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting steam condensation within the ice condensers limits the pressure and temperature buildup in the primary containment. The Inlet Door Position Monitoring System and the Inlet Door Annunciation Circuit are not required for proper operation of the inlet doors nor are they required to be "operable" as an initial condition for a DBA.

This TRM change allows the above described alternative method for a channel check for the Inlet Door Position Monitoring System. This change does not cause any permanent field modifications and does not interact with any safety related plant features. The UFSAR and Technical Specifications are not affected by this change. Therefore, based on compliance with established design basis requirements, this change is safe and does not involve an unreviewed safety question

SA-SE Number: WBPLEE 98-118-0

Implementation Date: 12/23/1998

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration Procedure Change	TACF Number 1-98-18-61, Revision 1, ARI-138-144, Revision 5	Ice Condenser Inlet Door Position Monitoring System

Description and Safety Assessments:

The Ice Condenser Inlet Door Position Monitoring System involves six (6) sets of status indication lights (green and red) located in the main control room (MCR) on panel 1-M-10. These status lights provide inlet door position information. Each green/red status light set monitors a zone. There are six zones (A through F) with eight doors In each zone for a total of 48 doors. Inlet door mounted zone switches provide the necessary switching action (i.e., contact action - open/closed) to complete the circuit and illuminate the appropriate status light. The zone switch contacts are configured for each status light set to illuminate a green light if all 8 doors are closed (the 8 "closed" contacts are wired in a series configuration) and to illuminate a red light if any of the 8 doors are open (the 8 "open" contacts are wired in a parallel configuration).

The MCR Inlet Door Annunciation Circuit consists of a parallel contact arrangement of all 48 inlet door zone switches. With all inlet doors closed , the zone switch contacts are open, thus, creating an equivalent open circuit. If one or more inlet door zone switch(s) detect an open door, the associated contact(s) close, thus providing a closed circuit (electrical current flow) and a corresponding annunciation in the MCR (window number 144A). This Inlet Door Annunciation Circuit and the above described Inlet Door Position Monitoring System uses separate inlet door zone switches and interrogation supply power.

This change allows an alternate method for performance of the channel check requirement for the Inlet Door Position Monitoring System described above. Currently, the Inlet Door Annunciation Circuit is used as a diverse measurement of the inlet door position. However, if the subject Inlet Door Annunciation Circuit is inoperable, an alternate channel check method is needed. This alternate channel check method consists of a Inlet Door Annunciation Circuit continuity measurement to ascertain the position of the inlet door zone switch contacts. Specifically, the alarm circuit cables will be temporarily determinated to electrically isolate the annunciator (Ronan) interrogation voltage. A 24V DC (nominal) power supply source and an incandescent light bulb will be connected to the two conductors of the common field cable at MCR panel 1-M-21 and used to determine if continuity exists in the zone switch contact circuit (48 contacts connected in parallel). A switch is used to test the light bulb. See Attachment 1 for a sketch of the continuity check circuit. If all contacts are open (which indicates all doors are closed) the total equivalent resistance should have been greater than 50K ohms (Reference 8). However, the cable is experiencing degradation in that its conductor to conductor insulation resistance is breaking down allowing increasing leakage current to flow. The bulb chosen will not be illuminated, with all contacts open, as long as this leakage current remains relatively low (a few milliamperes). However, if leakage current becomes large enough to cause a perceptible illumination of the bulb, the status of the door monitoring limit switches will become indeterminate. This method provides a reasonable assurance of inlet door position and may be used as a channel check to the Inlet Door Position Monitoring System.

The Ice Condenser Inlet Doors (48 total) form the barrier to minimize air flow between the reactor lower compartment and the ice bed. Periodic monitoring of the inlet door position during operational modes 1, 2, 3, and 4 is important to prevent long term degradation of the ice bed inventory due to ice malting or sublimation. In the event of a DBA, the inlet doors open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting steam condensation within the ice condenser limits the pressure and temperature buildup in the primary containment. The Inlet Door Position Monitoring System and the Inlet Door Annunciation Circuit are not required for proper operation of the inlet doors nor are they required to be 'operable' as an initial condition for a DBA.

SA-SE Number: WBPLEE 98-118-0 ***Implementation Date:*** 12/23/1998

There are no accidents that have been evaluated in the USFAR which may be affected by the proposed alternate method for performance of the channel check requirement for the Inlet Door Position Monitoring System. A credible failure mode for the proposed aftermath method for performance of the channel check is a power supply failure, however a power failure relay will provide an alarm. An additional failure mode would be if the test switch contact failed to open following the test of the light bulb. This would be conservative. It is unlikely that an ice condenser inlet door limit switch contact closure would occur at the same time a failure of the test switch contact to open, therefore this condition would be identified and resolved. Although this change does impact the functioning of the alarm described in FSAR Section 6.7.15.2, the intent of the alarm is met using the continuity check. The Ice Condenser System serves a passive safety function by providing a heat sink for DBAs. The Ice Condenser and inlet

This TACF change allows the above described alternative method for a channel check for the Inlet Door Position Monitoring System. This change does not cause any permanent field modifications and does not interact with any safety related plant features. The UFSAR and Technical Specifications are not affected by this change. Therefore, based on compliance with established design basis requirements, this change is safe and does not involve an unreviewed safety question.

SA-SE Number: WBPLEE-99-036-1

Implementation Date: 04/01/1999

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Figure 7.1-1 SH 1 FSAR Figure 10.3-8 FSAR Figure 7.3-3 SH 3	Correction of drawing references.

Description and Safety Assessments:

This is an administrative change to drawings 1-47W610-90-05, 1-47W611-63-1, -68-1, and 99-1 to only correct drawing references.

- 1-47W610-90-5-The reference to FE-90-400D should be 1-47W610-65-1 (not 1-47W610-68-1).
- 1-47W611-63-1-The reference the annunciator drawings should be 1-45B655-SERIES (not 47B601-55-0 through 65)
- 1-47W611-68-1-The reference the annunciator drawings should be 1-45B655-SERIES (not 47B601-55-0 through 65)
- 1-47W611-99-1-The reference the annunciator drawings should be 1-45B655-SERIES (not 47B601-55-0 through 65)

TVA drawing 1-47W611-99-1 is UFSAR Figure 7.2-1, Sheet 1 and TVA drawing 1-47W611-63-1 is UFSAR Figures 10.3-8 and 7.3-3, Sheet 3. 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 FSAR figures by this administrative change do not impact the probability of occurrence or consequences of any accident or equipment malfunction currently evaluated in the FSAR. 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 this administrative change does not constitute a unreviewed safety question.

SA-SE Number: WBPLEE-99-039-0

Implementation Date: 04/07/1999

Document Type:	***Affected Documents:***	***Title:***
Technical Requirements Manual	TRM Revision 14 WBN-TRM-99-001	Submerge components list.

Description and Safety Assessments:

The proposed change will add a note to TSR 3.8.4.2 and to the bases identifying that some components identified in Table 3.8.4-1 (Submerged Components With Automatic De-energization Under Accident Conditions) must be secured prior to resetting the accident signal. It will also identify in Table 3.8.4-1 an "*" by each component that must be secured prior to resetting the accident signal. This "*" will identify a note stating that "Are secured prior to resetting accident signal." TSR 3.8.4.2 bases will also have the following two sentences added: "The note clarifies that some Table 3.8.4-1 components require securing to prevent component energization due to plant process conditions which may exist concurrent with accident signal reset. These actions are contained in the applicable emergency procedures. The identified additional compensatory measures are a preset action and is not a required action for accident mitigation.

Emergency Operating Instruction E-0 will also be changed to reflect the need to secure the required devices. This change adds clarification that the subject components should be secured in such a way that will not allow them to come back on once the accident signal is reset. This would preclude their potential adverse impact (e.g. from being submerged) on the power supplies of other safety related equipment.

The identified additional compensatory measures are a preset action for restoration function (reset of an accident signal) and is not a required action for accident mitigation. The addition of the proposed clarification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR nor does create the possibility of an accident or malfunction of a different type than evaluated previously in the FSAR. Also, it does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, the proposed change does not constitute an unreviewed safety question.

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

Implementation Date: 04/07/1999

Document Type:	***Affected Documents:***	***Title:***
Technical Requirements Manual	TRM Revision 15 WBN-TRM-99-002	Submerge Components List Revision.

Description and Safety Assessments:

The proposed change will add a note to TSR 3.8.4.2 and to the bases identifying that some components identified in Table 3.8.4-1 (Submerged Components With Automatic De-energization Under Accident Conditions) must be secured prior to resetting the accident signal. It will also identify in Table 3.8.4-1 an "*" by each component that must be secured prior to resetting the accident signal. This "*" will identify a note stating that "Are secured prior to resetting accident signal." TSR 3.8.4.2 bases will also have the following two sentences added: "The note clarifies that some Table 3.8.4-1 components require securing to prevent component energization due to plant process conditions which may exist concurrent with accident signal reset. These actions are contained in the applicable emergency procedures (except for the Pressurizer Heater Backup Group 1C which is controlled by the clearance procedures). The identified additional compensatory measures are a preset action and is not a required action for accident mitigation.

Emergency Operating Instruction E-0 will also be changed to reflect the need to secure the required devices (except for the Pressurizer Heater Backup Group 1C which is controlled by the clearance procedures). This change adds clarification that the subject components should be secured in such a way that will not allow them to come back on once the accident signal is reset. This would preclude their potential adverse impact (e.g. from being submerged) on the power supplies of other safety related equipment.

The identified additional compensatory measures are a preset action for restoration function (reset of an accident signal) and is not a required action for accident mitigation. The addition of the proposed clarification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR nor does create the possibility of an accident or malfunction of a different type than evaluated previously in the FSAR. Also, it does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, the proposed change does not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-99-039-2

Implementation Date: 04/13/1999

Document Type:	***Affected Documents:***	***Title:***
Technical Requirements Manual	TRM Revision 16 WBN-TRM-99-003	Submerged Components List Revision

Description and Safety Assessments:

The proposed change will add a note to TSR 3.8.4.2 and to the bases identifying that some components identified in Table 3.8.4-1 (Submerged Components With Automatic De-energization Under Accident Conditions) must be secured prior to resetting the accident signal. It will also identify in Table 3.8.4-1 an "*" by each component that must be secured prior to resetting the accident signal. This "*" will identify a note stating that "Are secured prior to resetting accident signal." TSR 3.8.4.2 bases will also have the following two sentences added: "The note clarifies that some Table 3.8.4-1 components require securing to prevent component energization due to plant process conditions which may exist concurrent with accident signal reset. These actions are contained in the applicable emergency procedures and addressed in Reference 6. The identified additional compensatory measures are a preset action and is not a required action for accident mitigation. A double asterisk will be added to Table 3.8.4-1 for Pressurizer Heater Backup Group 1C identifying that it may not remain de-energized following an accident signal reset. Therefore, compensatory measures to prevent energization after reset of a accident signal is not required for Pressurizer Heater Backup Group 1C. This is acceptable because it has been shown by revision to calculation WBN EEB-MS-T108-0009 (R9) that there will be no adverse impact (e.g. from being submerged) on the power supplies of other safety related equipment.

Emergency Operating Instruction E-O will also be changed to reflect the need to secure the required devices (except for the Pressurizer Heater Backup Group 1C). This change adds clarification that the subject components should be secured in such a way that will not allow them to come back on once the accident signal is reset. This would preclude their potential adverse impact (e.g. from being submerged) on the power supplies of other safety related equipment.

The identified additional compensatory measures are a preset action for restoration function (reset of an accident signal) and is not a required action for accident mitigation. The addition of the proposed clarification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR nor does create the possibility of an accident or malfunction of a different type than evaluated previously in the FSAR. Also, it does not reduce the margin of safety as defined in the basis for any Technical specification. Therefore, the proposed change does not constitute an unreviewed safety question.

SA-SE Number: WBPLEE-99-041-0

Implementation Date: 04/30/1999

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50245	Drawing Discrepancies.

Description and Safety Assessments:

This EDC corrects drawing discrepancies in accordance with DD 99-0014, DD 99-0015, DD 99-0016, and DD 99-0018.

Specifically, this EDC makes the following changes:

EDC E-50245-A (DD 99-0018) revises design output Wiring Diagram 45W1618-5 to change the color of the insulation on the three conductor cable 1M1727. The cable is for connecting 1-TE-61-132F to Junction Box (JB) 954. The drawing showed the cable's conductor colors as black (BK), white (W) & red (R) and the colors in the field are (W), (R), & (R). In addition, the initiator of the 00 identified by phone that the splice shown on the drawing does not exist. Based on the Cable & Conduit Routing System (CCRS), cable 1 M 1727 is a type MFR 24 and is routed from the Resistance Temperature Device (RTD) to the junction box and confirms that the RTD leads were likely long enough to be routed to the junction box and a splice does not exist. The splice will be removed from the drawing. In addition, note 2 on the drawing supports the color code for a typical RTD as being (W), (R), & (R).

EDC E-50245-A (DD 99-0015) revises design output Conduit & Grounding Details 45W883-4 to add 1-FSV-77-17-A to note G which provides requirements for heat shrink material (i.e., Raychem) on solenoids with conduit seals which are continuously energized. WBPER980859 contains documentation (WO 97-007347-003) that shows 1-FSV-77-17-A satisfies note G.

EDC E-50245-A (DD 99-0016) revises design output Annunciator System Key Diagram 1-45W600-55-14 to correct the inputs to annunciator window 108-A. DCN W-39199-A did not remove cable 1A3681 and add cable 1A6800. The instrument number for one of the two inputs to annunciator window 108-A was corrected. 1-MUX-55-40 was added the title of the terminal area of the drawing.

EDC E-50245-A (DD 99-0014) revises design output Control Diagram 1-47W610-35-2 to correct the multiplexer channel for annunciator window 1A-1 B.

EDC E-50245-A (DD 99-0014) revises design output Wiring Diagram Turbo-Generator Auxiliaries Schematic Diagrams 1-45W600-47-2 (UFSAR figure 10.2-1) to remove note 8 and add the appropriate drawing reference for annunciator windows. Note 8 had a vendor drawing identified which has been replaced with a new TVA annunciator input/output drawing 1-45B655-XX, where XX represents the annunciator window box number

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 figure 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-50245-A do not constitute a unreviewed safety question.

SA-SE Number: WBPLEE-99-045-0

Implementation Date: 04/16/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50285-A	Diesel generator 2A-A room exhaust fan 1A motor replacement.
FSAR Figure	FSAR Figure 8.3-30	

Description and Safety Assessments:

DCN D-50285-A, replaces the motor for the Diesel Generator 2A-A Room Exhaust Fan 1A (2-MTR-30-448-A) which is defective and obsolete. The replacement motor had been installed for a similar application in the unused 5th diesel generator building (0-MTR-30-332-S). The fan is required to maintain area temperature as specified in the Technical Requirements Manual, Section TR 3.7.5. System Description Document N3-30DB-4002 gives the actual fan configuration required to maintain temperatures and thus maintain operability of the diesel generator.

The replacement motor is the same horsepower (15) and power factor (1.15) as the obsolete motor. However, the full load current changed from 19.5 amperes to 18 amperes and the locked rotor current changed from 118 amperes to 105 amperes. As a result of these reductions in currents, the breakers trip setting that feeds the motor changed from 270 amperes to 235 amperes and the protective overload device changed from a G30T49 to a G30T48.

This fan is one of two 50% capacity fans whose safety function is to provide ventilation cooling to the diesel generator room for generator 2A-A. Each fan has independent controls such that one of the fans will operate when the diesel is running and the exhaust temperature is 60°F or greater. Both fans operate if the room exhaust temperature is 80°F or greater (see System Description N3-30DB-4002, Section 3.3.2). There are no chapter 15 design basis accidents which may be affected by the proposed activity.

The credible failure modes of this fan is to not operate when needed.

A unresolved 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-99-048-0

Implementation Date: 04/27/1999

Document Type:	*Affected Documents:*	*Title:*
Design Change	DCN D-50286-A	Wiring Change for Turbine Thrust Bearing Thermocouples.

Description and Safety Assessments:

Dual-element thermocouples 1-TE-47-232A and 232B, 1-TE-47-233A and 233B, 1-TE-47-234A and 234B, and 1-TE-47-235A and 235B monitor the turbine thrust bearings front and rear face temperature and provide inputs to the Integrated Computer System (ICS) (LOG T2021A, T2022A, T2023A, & T2024A) and 1-TR-47-1 (Pens 12, 13, 14, & 15) in the MCR. DCN D-50286-A installed jumpers so that the TCs in each pair are connected in parallel. This configuration will maintain the capability to monitor bearing temperature at both the ICS and recorder when one TC of a pair fails.

These components do not perform any safety functions and there are no associated Technical Specifications.

This change affects UFSAR Figure 10.2.4 (1-47W610-47-3).

The paralleling of the T/Cs in each pair will maintain the capability to monitor bearing temperature at both the ICS and recorder when one TC has failed. The change meets all functional requirements specified in the applicable design basis documents. The modification testing requirements will ensure continued proper operation of the affected components. The UFSAR text and Technical Specifications are not affected by this change, although an UFSAR Figure is impacted. Therefore, based on compliance with the established design and licensing basis, this change is safe and is acceptable from a nuclear safety standpoint.

SA-SE Number: WBPLEE-99-069-0

Implementation Date: 07/20/1999

Document Type:	***Affected Documents:***	***Title:***
DCN	EDC Number E-50339-A	Drawing Discrepancies

Description and Safety Assessments:

EDC E-50339-A is a documentation only change to provide disposition of DDs 99-0046, 99-0047, 99-0049, 99-0050, and 99-0051. A safety evaluation is required for this change because it will impact FSAR Figure 8.1-3 (TVA drawing 1-45W700-1). Therefore, this safety evaluation will evaluate the impact of changing the drawing. The symbols representing "safety wired in normal closed position" will be deleted, and the correct transfer switch symbols for break-before-make (BBM) contacts will be added on the drawing for the non-safety related Main Control Room (MCR) Panels 1 and 2-M-7 (System 237 - 120V AC Instrument Power A Rack Transfer Switch (Mark Number 21AP) and 120V AC Instrument Power B Rack Transfer Switch (Mark Number 21AP)). Companion drawing will also have the same changes for non-safety related MCR Panels 1 and 2-M-7 (System 238 - 120V AC Preferred Power Transfer Switch (Mark Number 21AO). The associated Bill of Material drawings describe the transfer switches, Mark Number's "21AO" and "21AP" as "break-before-make contacts, safety wire seal to lock switch in the normal position". These drawings will be changed (change paper against drawings) to clarify that the transfer switches do not have safety wire seals.

Additionally, a similar condition for "safety wired in normal closed position" symbols as described above exists for System 235 transfer switches 1 -XSW-235-0001 -D, 1 -XSW-235-0002-E, 1 -XSW235-0003-F, I-XSW-235-0004-G, 2-XSW-235-0001-D, 2-XSW-235-0002-E, 2-XSW-235-0003-F, and 2-XSW-235-0004-G (120V AC Vital Instrument Power Boards 1-1, 1-11, 1-111, 1-IV, 2-1, 2-11, 2-111, and 2-IV, respectively) on drawing (FSAR Figure 8.1-3) and will be corrected by this EDC.

The proposed change has been evaluated against (1) Loss of Offsite Power and (2) Loss of External Electrical Load design basis accidents that have been previously evaluated in Chapter 15 of the FSAR, and the change was determined to have no adverse effect on the analyzed accidents . System 235 (120V AC Vital Power) is a safety related system, System 237 (120V AC Instrument Power) and System 238 (120V AC Preferred Power) are non-safety related systems and are not used in the mitigation of any accident. There will be no design bases accidents introduced by this EDC.

The deletion of the "safety wired in normal closed position" symbols from FSAR Figure 8.1-3 and companion drawing, and the addition of the correct transfer switch symbols for break-before-make (BBM) contacts will not inhibit the operation or adversely affect the functional requirements of the transfer switches. There will be no new credible failure modes introduced by this EDC. The failure modes for the manually operated non-safety related Panels M-7 120V AC Power Rack transfer switches are failure of the contacts to break-before-make, failure in the normally open position, or failure in the normally closed position. The failure modes for the manually operated safety related Units 1 and 2 120V AC Vital Instrument Power Board transfer switches are failure of the contacts to make-before-break, failure in the normally open position, or failure in the normally closed position.

SA-SE Number: WBPLEE-99-069-0 ***Implementation Date:*** 7/20/1999

EDC E-50339-A is a documentation only change to provide disposition of DDs 99-0046, 99-0047, 990049, 99-0050, and 99-0051. A safety evaluation is required for this change because the disposition of DD 99-0051 will impact FSAR Figure 8.1-3 as follows: (1) The symbols representing "safety wired in normal closed position" will be deleted for the Panels M-7 120V AC Power Rack transfer switches (DD 99-0051), and the Units 1 and 2 120V AC Vital Instrument Power Board transfer switches. A field inspection during the DD 99-0051 evaluation identified that the Units 1 and 2 120V AC Vital Instrument Power Board transfer switches do not have safety wire seals. (2) The correct transfer switch symbols to identify that the non-safety related Panels M-7 120V AC Power Rack transfer switches have break-before-make (BBM) contacts will be added. The proposed design change does not increase the probability of an accident or occurrence of a malfunction of equipment important to safety since the Design Basis requirements of the systems have not been changed by this EDC. According to design criterion WB-DC-30-27 Revision 21 *(AC and DC Control Power Systems),* the 125 - VDC Vital Battery System transfer devices, including 480 - VAC Vital transfer devices, are the only transfer devices that are required to be "safety-wired in the normal position". The consequences of an accident or a malfunction of equipment will not be increased. No new accidents or malfunctions of a different type than evaluated in the FSAR are created since the transfer switches will continue to function as specified by the design basis documents. The proposed design change does not affect any technical specifications: therefore, no margins of safety are reduced. Based on the above, the proposed design change does not involve an unreviewed safety question (USQ).

SA-SE Number: WBPLEE-99-083-0

Implementation Date: 09/02/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC Number E-50364-A	Drawing Deviation (DD) - Spare Breakers

Description and Safety Assessments:

DESCRIPTION:
EDC E-50364-A is a documentation only change to provide disposition of DDs 99-0060 and 99-0064. A safety evaluation is required for this change because DD 99-0064 will impact FSAR Figures 8.2-12 and 8.2-13. Therefore, this safety evaluation will evaluate the impact of changing these drawings. Wiring diagram symbols associated with "spare" breakers 1-BKR-239-1/215 and 2-BKR-239-2/215 will be added on the 250V Battery Boards 1 and 2 (front views), Panels 2, of drawings, respectively, in order to be consistent with the associated circuit schedule and other 'spare' breakers shown on the drawings.

DESIGN BASES ACCIDENT:
The proposed change has been evaluated against (1) Loss of Offsite Power and (2) Loss of External Electrical Load design basis accidents that have been previously evaluated in Chapter 15 of the FSAR, and the change was determined to have no adverse effect on the analyzed accidents. System 239 (250V DC Battery System) is a non-safety related system, and the "spare" breakers are not used in the mitigation of any accident. There will be no design bases accidents introduced by this EDC.

CREDIBLE FAILURE MODES:
The addition of wiring diagram symbols associated with "spare" breakers 1-BKR-239-1/215 and 2-BKR-239-2/215 on the 250V Battery Boards 1 and 2 (front views), Panels 2, of FSAR Figures 8.2-12 and 8.2-13, respectively, in order to be consistent with the associated circuit schedule and other "spare" breakers shown on these FSAR Figures will not introduce any new credible failure modes. Since the breakers are "spares" presently not being utilized, their failure modes are not relevant for this EDC.

The consequences of an accident or a malfunction of equipment will not be increased. No new accidents or malfunctions of a different type than evaluated in the FSAR are created since the breakers will continue to function as "spares" presently not being utilized. The proposed change does not affect any technical specifications; therefore, no margins of safety are reduced. Based on the above, the proposed change does not involve an unreviewed safety question.

SA-SE Number: WBPLMN-95-067-1

Implementation Date: 08/10/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-37414-A	Documentation changes associated with the ERCW system.

Description and Safety Assessments:

This Safety Evaluation evaluates several documentation changes associated with the ERCW system and addressed under DCN S-37414-A. In general, these changes include the evaluation and acceptance of the degraded condition of all eight ERCW pumps which was identified during pre-operational pump performance testing. The DCN also evaluates and provides instructions for aligning the ERCW Train A and Train B supply headers during the ASME Section XI pump and valve performance tests and during Modes 5 and 6. There is no field work associated with the subject DCN nor are there any changes to the ERCW system design or operating requirements.

Specifically, DCN S-37414-A evaluates the ERCW pump performance requirements to determine the minimum performance levels at which the pumps can still meet the ERCW system design requirements for Unit 1-only operation. As documented in the corrective action program, WBSCA950011, the pre-operational test data for the ERCW pumps indicated the pumps are performing significantly below the specified pump ratings indicated on the original pump curves supplied by the pump manufacturer. An analysis determined that the ERCW pumps could operate as low as 72% of the specified rating and still satisfy the ERCW system design requirements for Unit 1-only operation.

DCN S-37414-A also evaluates the alignment of the Train A and Train B ERCW supply headers to facilitate ERCW pump and valve testing under the ASME Section XI testing program. The ERCW pumps are trained but not unitized. In normal operating configuration, the four Train A ERCW pumps discharge into a common Train A manifold; likewise, the four Train B ERCW pumps discharge into a common Train B manifold. The two separate manifolds, in turn, are each connected to a pair of trained and unitized supply headers (1A/2A and 1B/2B, respectively) which feed trained and unitized equipment. Due to the system flow balance, pump discharge flow is normally split unevenly between the two headers on the same train. In order to achieve the required instrument accuracy during testing of individual ERCW pumps, however, it is necessary to maximize the supply header flowrate. This is accomplished by isolating the low-flow supply header at the pump manifold and directing the total discharge flow from the ERCW pump under test through the remaining supply header. For the duration that the low-flow header is isolated, the header cross-tie valves (which are normally closed to maintain train separation) must be opened to maintain ERCW supply to Reactor Building HVAC equipment normally served by the isolated header. This equipment is not required for accident mitigation, but is required during normal operation to maintain Reactor Building temperatures within Technical Specification limits. The ERCW system configuration in the test alignment is as follows:

Train A ERCW pump test alignment (with Train B pumps normally aligned): Valve 2-FCV-67-0022-A open and valve 1-FCV-67-0022-A closed (this isolates supply header 1A and directs Train A ERCW pump discharge flow through supply header 2A only); cross-tie valve 1-FCV-67-0147-A open (this allows equipment normally served by supply header 1A to receive cooling water from supply header 2B).

Train B ERCW pump test alignment (with Train A pumps normally aligned): Valve 2-FCV-67-0024-B open and valve 1-FCV-67-0024-B closed (this isolates supply header 1B and directs Train B ERCW pump discharge flow through supply header 2B only); cross-tie valve 1-FCV-67-0458-A open (this allows equipment normally served by supply header 1B to receive cooling water from supply header 2A).

Prior to the evaluation performed in support of this change, opening the ERCW Train A and B cross-tie isolation valves during Modes 1, 2, 3, or 4 would have placed the plant in an unanalyzed condition. An analysis has been performed which concluded that the ERCW system can adequately cool the required equipment when aligned as described above to support

SA-SE Number: WBPLMN-95-067-1 ***Implementation Date:*** 08/10/1998

ASME testing. Whenever the ERCW system is aligned in this configuration during Modes 1, 2, 3, or 4, the plant will enter the applicable limiting conditions for operation (LCOs) of the Technical Specifications (TS). LCO 3.7.8 for the ERCW System states "Two ERCW trains shall be OPERABLE." If one train is inoperable, the required action is to restore that ERCW train to operable status within 72 hours. During the performance of the ASME tests, some equipment normally served from the ERCW supply train under test will be isolated from that train. ERCW flow to the affected equipment will be supplied from the remaining train via the cross-ties as described above. All actions will be performed within TS limitations. Compensatory measures will be implemented to ensure the cross-tie valves can be returned to their normal positions if a design basis event (DBE) occurs concurrent with performance of a test.

DCN S-37414-A also evaluated another ERCW supply header alignment for use during Modes 5 and 6 only as requested by Operations (OPS) personnel. In the requested alignment the ERCW pumps on both trains would discharge into a pair of common loop headers, with supply header 1A cross-tied to supply header 2B and supply header 1B cross-tied to supply header 2A. Both ERCW trains would remain in service, but there would be no train separation since both normally-closed cross-tie isolation valves would now be open. In Modes 5 and 6, the TS state only that the operability requirements of the ERCW system are determined by the systems it supports. An analysis has been performed to confirm the adequacy of the ERCW system to satisfy the TS requirement for Modes 5 and 6 with the system in the "common loop" configuration. During Modes 1, 2, 3, and 4, however, LCO 3.7.8 requires that two ERCW trains be operable. Therefore, if the ERCW supply headers are cross-tied in Mode 5, the cross-tie isolation valves must be closed to establish train separation before ascending to Mode 4.

Accidents Evaluated as the Design Basis (Accident/Analysis - FSAR Section 15):

Condition I -Normal Operation and Operational Transients
Condition II - Faults of Moderate Frequency
Condition III -Infrequent Faults
Condition IV - Limiting Faults:

The ERCW system is a mitigating system for all Condition II, III, and IV accidents, principally because it is the safety-related water source for the AFW pumps suction and supplies cooling water to various safety-related equipment required for accident mitigation. The subject documentation changes do not adversely affect the ability of the ERCW system to perform its intended safety functions. The cross-tied system configuration and valve alignment required to support the ASME Section XI ERCW pump tests have been evaluated to determine the adequacy of the ERCW system performance and was found to be acceptable. The unit enters the applicable TS LCOs upon commencement of the performance phase of a test and exits upon completion of that phase; therefore, consideration of single failure criteria is not required. The ERCW pump test Surveillance Instructions incorporate administrative controls which specify the Operator actions required as compensatory measures in the event an accident occurs during the performance of a test implementation of these actions ensures the ability of the ERCW system to perform its intended safety functions is not compromised. Appendix R impacts are limited to the temporary closing of certain breakers required to restore power to valves which are normally de-energized for Appendix R concerns. This condition is addressed by invoking the appropriate Fire Operating Requirements (FOR). It was concluded that no design basis accident or anticipated operational transient evaluations in the FSAR are impacted by the subject documentation changes.

There are no credible failure modes introduced by the subject documentation changes which have not been previously accounted for in the ERCW system design or which would prevent the system from performing its intended safety function. The subject documentation changes do not directly or indirectly impact any safety analysis that forms the basis for any TS. Based on this review, it is concluded that no TS changes will be required due to implementation of DCN S-37414-A. Therefore, the subject documentation changes will not reduce the margin of safety as defined in the basis for any Technical Specifications nor does it constitute and unreviewed safety question.

SA-SE Number: WBPLMN-96-073-1

Implementation Date: 04/15/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-38974-A	Main steam condenser tube replacement.
FSAR	FSAR Package 1513	

Description and Safety Assessments:

Main Steam Condenser Retubing

The primary design purpose of the main steam condenser is to provide a heat rejection path for the plant secondary cycle by condensing the steam exhausts from the low pressure turbine and transferring the heat removed from the steam to the condenser circulating water system (CCW). During cold startup, the main condenser must also deaerate the initial inventory of water contained within the condensate feedwater systems.

The main steam condenser also serves as the discharge point for the steam dump system. The steam dump system in the condenser permits the plant to accept sudden large load decreases, remove stored energy and residual heat from the reactor following a turbine/plant trip, maintain the plant in a hot standby condition, or permit a manual controlled RCS cooldown of the plant to the point where the RHR system can be placed in service. The performance of the condenser after the new ferritic stainless steel tubes are installed will be essentially the same as with the 90-10 copper nickel tubing assuming the same amount of tube plugging; i.e. condenser back pressures will be approximately 0.04 inches of mercury less. In actuality, the condenser performance will improve due to the decrease in the amount of tube plugging, since the condenser currently has 792 tubes plugged out of a total 27,410 tubes installed in the condenser.

Replacing the existing tubes within the main steam surface condenser will not affect any design basis accidents or anticipated operational transients. The main condenser does not perform any reactor safety related function.

A condenser tube failure would result in contamination of the condensate feedwater system. Depending on the severity of the leak and the ability of the demineralizers to handle the contamination it might be necessary to lower unit operating levels and take one side of the condenser out of service to perform tube inspections and maintenance, or to ultimately shut the reactor down. These are conditions which have occurred in the past. The new ferritic stainless steel tubes and tube staking will significantly reduce the likelihood of tube failures occurring.

Removing the 90-10 copper nickel tubes from the condenser and replacing them with ferritic stainless steel tubes reduces the likelihood of steam generator tube ruptures. Copper contributes to degradation of steam generator tubes. Removing the copper tubes in the condenser improves the service life of the steam generator tubes by reducing long term tube degradation and therefore reducing the probability of steam generator tube ruptures.

The main steam condenser is part of the Condensate System and is supplied cooling water by the Condenser Circulating Water System. Both of these systems are balance of plant systems and do not support the operation of any nuclear safety related systems. This modification is replacing existing steam surface condenser tubes with tubes made of a superior material, and is installing anti-vibration tube stakes to eliminate concerns about tube damage resulting from tube vibration.

Failures of these systems do not contribute to or initiate any of the accident scenarios in the FSAR; therefore, this modification will not increase the probability of occurrence or the consequences of an accident or malfunction previously evaluated in the FSAR or create the possibility of an accident or a malfunction of a different type from those previously evaluated in the FSAR. This modification to these systems does not reduce the margin of safety for any basis for any Technical Specification.

SA-SE Number: WBPLMN-96-073-1 ***Implementation Date:*** 04/15/1999

Replacement of the Number 5, 6 and 7 Extraction Steam Expansion joints

Replacing the existing number 5, 6, and 7 extraction steam expansion joints within the main steam surface condenser will not affect any design basis accidents or anticipated operational transients. The extraction steam system does not perform any safety related function. The normal functions of the Number 5, 6, and 7 extraction steam lines are to transport extraction steam from the low pressure turbines to the feedwater heaters for regenerative feedwater heating and to transport condensate, which is removed from the low pressure turbines via moisture removal stages, to the feedwater heaters. This moisture removal improves turbine efficiency and prolongs the life of the turbine blading.

A failure of the expansion joint would result in the admission of steam into the condenser. This would impact the efficiency of the unit by removing heating steam that would normally go to the low pressure condensate heaters and discharging it directly into the condenser. The failure of the expansion joint would not affect feedwater flow to the steam generators, since the steam would be condensed in the condenser, fall to the hotwell and be returned to the condensate/feedwater cycle.

Depending on the severity of the failure, parts of the expansion joint could fall on the condenser tube bundle and damage the tubes (see discussion on tube failures above) or fall past the tube bundles to the bottom of the condenser and into the hotwell. The design and materials being used in the new expansion joints incorporate lessons learned from Sequoyah Nuclear Plant (SQN) in upgrading the equivalent expansion joints at SQN. Therefore, the likelihood of these failures occurring at WBN will be reduced with the installation of the new expansion joints.

The Number 5, 6, and 7 extraction steam expansion joints are installed in the extraction steam piping connecting the low pressure turbine extraction steam connections to the number 5, 6, and 7 heaters installed in the condenser. The piping is internal to the condenser. The new expansion joints are replacing existing expansion joints, and have an improved design and better materials of fabrication. This is a result of lessons learned from similar expansion joints in the SQN Number 5, 6, and 7 extraction steam lines to the respective heaters.

Failures of these components do not contribute to or initiate any of the accident scenarios in the SAR; therefore, this modification will not increase the probability of occurrence or the consequences of an accident or malfunction previously evaluated in the FSAR or create the possibility of an accident or a malfunction of a different type from those previously evaluated in the FSAR. This modification to these systems does not reduce the margin of safety for any basis for any Technical Specification.

SA-SE Number: WBPLMN-96-103-0

Implementation Date: 03/09/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39034-A Fire Protection Report Figure	Aqueous Film Forming Foam (AFFF) System

Description and Safety Assessments:

a. This is an administrative change to allow the Aqueous Film Forming Foam (AFFF) storage tank to be drained and its isolation valves (0-26-1624 and 1632) to be closed when the 5th Diesel Generator is not in service. The change will be accomplished by adding a note to drawing 1-47W850-10-which is also Figure II-8 in the Fire Protection Report. The system description N3-26-4002 will also be revised to note this option.

b. The Interim Office Buildings A & B are being removed which also removes the fire protection for the buildings. This results in valve 0-26-646 becoming a normally closed valve because it is the isolation valve in the branch line that isolates the main fire protection loop header from the Interim Office Buildings (IOB) A & B.

c. Valve 0-26-1376 was shown as a normally closed valve on the flow diagram and it should have been shown as normally open. DD 96-0064 identified this discrepancy that is being corrected by this DCN.

d. Valve 0-26-9371 was shown as 9731 on drawing 1-47W850-5. DD 96-0064 identified this discrepancy that is being corrected by this DCN.

e. Rooms 692.0-A17 and 18 have been renamed the Hot M&TE Tool Room and the Hot Tool Room. Problem Evaluation Report, WBPER960426, identified that these room name changes had not been incorporated on 47W240-1. This DCN incorporates the changes.

The AFFF system is not a safety related system nor does it interface with a safety related system. It is not covered by a Technical Specification and the system is not required for compliance with Appendix R requirements. The system is provided to address a specific hazard (diesel fuel oil) in the 5th Diesel Generator Building. This diesel generator and building are not required for Unit 1 operation and the diesel generator is not in service. The AFFF system is only needed when the 5th Diesel Generator is in service. This change allows the option to take the AFFF system for the 5th Diesel Generator Building out of service when the 5th Diesel Generator is out of service. The installed suppression system becomes a normal preaction sprinkler system and provides adequate fire protection to the building.

The Interim Office Building fire protection is not safety related and the closing of the valve that isolated the main plant fire protection loop header from the Interim Office Building does not degrade the fire protection capability to the safety related structures or systems.

Valve 0-26-1376 should be a normally open valve and is configured as such in the plant. The flow diagram shows the valve to be normally closed and this discrepancy was identified on DD 96-0064. This change corrects that drawing discrepancy and makes the drawing reflect correct plant configuration. Valve 0-26-9371 is incorrectly identified as 0-26-9731 on drawing 1-47W850-5. This discrepancy was identified on DD 96-0064 and is being corrected by this DCN to make the drawing reflect correct plant configuration. A corrective action to WBPER960426 is to revise drawing 47W240-1 to show the correct name for rooms 692.0-A17 and A18. This DCN revises that drawing to reflect correct plant configuration.

SA-SE Number: WBPLMN-97-036-0

Implementation Date: 06/22/1998

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF-1-97-014-062 Procedure TP-62-029	Relocation of U1/U2 Interface Boundary to allow transfer of spent resin from Chemical Volume Control System, Mixed Bed Demineralizer 1A to Solid Radwaste Disposal, Spent Resin Storage Tank.

Description and Safety Assessments:

A transfer of spent resin from chemical volume control system (CVCS), mixed bed demineralizer 1A (1-DEMN-062-0001/IA) to solid radwaste disposal, spent resin storage tank (SRST) (O-TANK-077-0007) was being performed per SOI-77.03 DN-1. High radiation levels were identified, during this transfer, upstream of isolation valves 1-ISV-062-0995 and 2-ISV-062-0995. These closed valves are Unit 1/Unit 2 (UI/U2) interface boundary valves that isolate the CVCS evaporator condensate demineralizers (1-DEMN-062-0126 and 2-DEMN-062-0127) from the Unit 1 spent resin transfer piping. The Unit l/Unit 2 interface boundaries were established as those physical points of interface between the licensed and unlicensed unit necessary to control the interactions between portions of nonsafety-related systems that are to be utilized by Unit 1 operations from portions of the system not used by Unit 1 operations. The Demineralizers are not required for Unit 1 operation and the resin apparently leaked through the closed boundary valves.

Temporary Alteration Control Form (TACF) Number 1-97-014-062 is relocating the UI/U2 interface boundary to allow flushing of residual resin out of the CVCS Evaporator Condensate Demineralizers. The new boundaries include the Demineralizers and upstream isolation valves, to allow flushing of the Demineralizers with primary makeup water. The flush will be accomplished using Temporary Procedure TP-62-029.

The piping to be included in the temporary boundary contains vent, drain, and resin discharge valves, as well as the PMW supply valve.

The solid radwaste disposal and its associated components, piping, and valves are located in the Auxiliary Building. This equipment does not perform a primary safety function, is installed in a Seismic Category I 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 waste gas decay tank (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 a WGDT rupture. This TACF does not change the logic or function of any system that is important to safety.

FSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunctions in the liquid radwaste processing system such as pump or valve failures or evaporator failures. This TACF 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 modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This TACF 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 relocating the UI/U2 interface boundary to allow flushing of residual spent resin out of the demineralizers. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The solid radwaste disposal and 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 design bases event (DBE), all safety related equipment is expected to operate as designed to limit the consequences of the DBE.

SA-SE Number: WBPLMN-97-036-0 ***Implementation Date:*** 06/22/1998

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 disposal system are not revised or challenged by these changes. A review of the detailed changes leads to the safety evaluation conclusions that this change is safe and does not constitute an unreviewed safety question."

SA-SE Number: WBPLMN-97-040-2

Implementation Date: 04/19/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change FSAR	DCN M-39308-B FSAR Change Package 1525	Raw Cooling Water discharge routed to Unit 2 Cooling Tower Flume during plant outages.

Description and Safety Assessments:

This modification installs a permanent connection to the CCW Unit 2 cooling tower discharge flume to allow the ability to dewater the CCW Unit 1 cooling tower discharge flume and CCW system piping while still maintaining normal operating conditions for the RCW system.

Current configuration requires plant personnel to install 3 temporary pipe spools to the RCW discharge. This is a personnel safety concern, time consuming, and costly. This modification will eliminate outage time and save money by making a permanent modification to the piping system as noted above.

Rerouting piping for RCW discharge line will not affect the normal operating path of the RCW system.

In addition, a document change only to drawing 1-47W831 -1 was made to show bypass valve, 0-BYV-027-0500, as normally closed since, during normal plant operation, the only water discharged into the Unit 2 flume is ERCW Train B, and this effluent is then discharged to Unit 2 Cooling Tower Basin and eventually discharged to 48" Blowdown line.

Modifying the existing RCW discharge to allow for a permanent discharge connection with the Unit 2 CCW cooling tower discharge flume will not change the function of the RCW as it is described in the FSAR. There are no design basis accidents or operational transients associated with the proposed modifications in Chapter 15 of the FSAR. There are no Appendix R components or equipment or any nuclear safety-related systems or portions of systems affected by the proposed modifications. The RCW system does not perform any reactor safety-related function, nor will the RCW compromise the ability of safety-related systems to perform their intended functions. Therefore, this modification will not affect any design basis accidents or anticipated operational transients.

The credible failure modes associated with the implementation of this DCN are:
1. Failure of either of the valves to change positions when being manually operated, or
2. Catastrophic valve failure, such as stem or disc breakage which would allow the disc to become entrained in the system now.

These failure modes are prevented or minimized by the selection of valve materials for use in the intended service.

SA-SE Number: WBPLMN-97-058-1

Implementation Date: 03/10/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN W-39410-A Fire Protection Report Figure	Design Change to Cap the Fire Protection Lines.

Description and Safety Assessments:

This modification removes the fire hydrants and associated piping that were for the low level radwaste storage area which was abandoned. A section of pipe downstream of valve 0-26-0237 was left in place and a plug was installed in the pipe. The other part of the modification placed a plug in the section of pipe that was left downstream of valve 0-ISV-26-0646. The piping, valves, etc. that were downstream of these were for Interim Office Buildings A and B fire protection system. These buildings were removed and all the fire protection piping was removed. Placing a plug in the end of the piping downstream of these two valves ensures that if the valve is inadvertently opened, there would not be a loss of fire protection water. Neither of these valves and their downstream components are located in a portion of the fire protection piping nor does this modification change the hydraulic performance of a fire protection system that is required for any safety related structure, system, or component. Therefore, this modification has no impact on fuel cladding, reactor coolant system, or containment integrity and it does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-97-070-0

Implementation Date: 03/12/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN 39482 FSAR Figure 9.3-9	Waste Gas Analyzer Sample Drain Lines

Description and Safety Assessments:

While draining the Waste Gas Analyzer sample lines, the Auxiliary Building had to be evacuated due to actuation of local radiation monitor alarms. The drain piping for these sample lines are part of the Unit 2 closed equipment drain header on Elevation 713 and some common equipment uses this header. Because most of the equipment connected to this header is Unit 2, and not required for Unit 1 operation, not all of it has been installed or in some cases has been removed for use in Unit 1. The drain lines for some of these uninstalled or removed pieces of equipment have been left open to the atmosphere. When draining into this header, as in the case of the Waste Gas Analyzer, these open drain lines are potential airborne release paths into the Auxiliary Building. Additionally having these open lines to Unit 2 equipment presents the potential to contaminate unnecessarily that equipment.

DCN W-39482-A evaluates all of the drain connections to this closed header and plugs all those to equipment not required for Unit 1 operation. Additionally a note will be added to the floor and equipment drain flow diagram (1-47W852-2) and piping drawings (47W479-6) to help to ensure that in the future work to remove all equipment from Unit 2 will also cap the associated drain line.

Section 9.3.3, "Equipment and Floor Drainage System," and Section 11.2, and 11.3, "Liquid Waste Systems, and "Gaseous Waste Systems," cover the system which could be affected by these drains. There is no specific reference in those FSAR sections to any of the drains being modified by this DCN. Because the drains are on Unit 2 equipment not required for Unit 1 operation, there is no affect on those systems as described in the FSAR. No changes to any FSAR text was identified and Figure 9.3-9 is the only figure change required.

FSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunctions in the Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. This DCN 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 this modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN 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 plugging these drain lines. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The closed drain header, its associated connections, 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.

The drain header and its associated drain connections being modified by this DCN are located in the Auxiliary Building in various areas on Elevation 713. These drains do not perform any safety function, are installed in a seismic structure, and are seismically supported, and is not used during any accident. 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 safety evaluation conclusions that this change is safe and does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-97-084-2

Implementation Date: 09/23/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number M-39302-A, F-39882-A, F39909 SOI-14.03 FSAR Change Package 1491	Replacement of Nonreclaimable Waste Pumps

Description and Safety Assessments:

The existing Nonreclaimable Waste Pumps (NRWPs) are obsolete and, due to their limited flow capability, greatly extend the time required for recirculation of the Nonreclalmable Waste Tank (NRWT) prior to discharge. The pumps will be replaced with higher head and flow capacity pumps. The replacement pumps have a design point of 115 gpm at 300 ft of head. The taper bore stuffing box design of the new pump provides internal fluid circulation in the area of the mechanical seal and; thereby, allows elimination of the seal flush water that is currently supplied to the pump. Minor piping modifications are required in the area of the suction and discharge connections to facilitate installation of the new pumps. Pressure reduction orifices are installed in the recirculation flow path to the NRWT and also in the flow path to the cooling tower blowdown/Turbine Building sump. These pressure reduction orifices are necessary to prevent the pumps from running out beyond their maximum flow limit. Due to the higher flow capacity of the replacement pumps, 0-FT-14-192 is also rescaled within the scope of this DCN. 0-FT-14-192 is currently scaled for 0-60 gpm. It will be rescaled for its maximum range of approximately 130 gpm. In addition, the 60 gpm low range flow loop 0-FI-014-0456A will be eliminated because it is no longer needed with the 115 gpm flow capacity pumps. The new piping material being installed per this DCN is Carpenter 20 alloy. This material has a better resistance to caustic/acid systems than the stainless steel currently installed. Additional isolation valves are being added to the X pump suction and the 1 1/2" drain line to the demineralizer sump near the NRWT discharge per DCN F-39882-A. This will ensure tank isolation if system piping downstream of new valves is breached.

The replacement pumps will allow revisions to operating procedure SOI-14-03 to enhance recirculation and release of a NRWT. The SOI-14-03 revision can include any changes associated with the pump replacement. The revision can include, but will not be limited to the following changes: The recirculation time can be reduced due to the higher capacity pumps, references to seal water supply to the existing NRWPs can be removed due to the elimination of seal water requirements to the replacement pumps, and steps associated with the adjustment of the valve in the NRWT recirculation path to provide increased pressure for sampling can be eliminated.

FSAR Section 10.4.6 has been reviewed and the design or functional requirements of the condensate demineralizer system are not changed. DCN M-39302 and F-39882-A revise flow diagram 1-47W838-3, which is FSAR Figure 10.4-36C, to reflect the changes necessary for installation of the higher head vs. flow capacity NRWPs, and the additional valves and material change from stainless steel to Carpenter 20 alloy. FSAR Section 11.2.3.1, page 11.2-9 is revised to delete the flow capacity specified for the NRWPs. FSAR Table 11.2-3, Sheet 7, is revised to specify the correct flowrate, design pressure, design temperature, material, and head for the replacement NRWPs.

This change also affects the Offsite Dose Calculation Manual (ODCM). Tables 1. 1 -1 and 2.1 -1 must be revised to delete reference to 0-FI-14-456A. This flow indicator is no longer required with the increased NRWP flow capacity and is being deleted within the scope of DCN M-39302. Page 73 and Figure 6.3 must be revised to change the discharge flowrate from the NRWT.

The Watts Bar Safety Evaluation Report (SER) and Supplements 1-20 have been evaluated for impacts. The changes of DCN M-39302 do not impact the NRC's understanding of the design and operation of Watts Bar Nuclear Plant as described in the FSAR.

SA-SE Number: WBPLMN-97-084-2 ***Implementation Date:*** 09/23/1998

The changes of DCN's M-39302-A, F-39882-A, F-39909-A and FSAR Change Package 1491 do not affect any FSAR Chapter 15 fault or operational transient evaluations. The changes to replace the NRWPs comply with design and licensing basis requirements for the condensate polishing demineralizer system. The NRWPs do not perform any safety-related functions and are not required for the orderly shutdown of the reactor. Therefore, the probability of occurrence or consequences resulting from a previously evaluated accident or equipment malfunction is not increased. In addition, since the NRWPs do not have the potential to adversely impact any other safety related equipment, the possibility for an accident or equipment malfunction of a type different than any other evaluated previously in the FSAR is not increased. Since the Tech Specs do not address the condensate polishing demineralizer system and the changes of DCN 39302-A do not have the potential to indirectly affect Tech Spec systems, the safety margins defined in the Tech Specs Bases are unaffected.

The design basis accidents and anticipated operational transients of FSAR Chapter 15 have been reviewed with respect to the changes performed by DCN M-39302 and F-39882-A. The specific design basis accident and operational transient considered are a steam generator tube rupture and a steam generator tube leak. No credit is taken in FSAR Chapter 15 or in Design Basis Events Design Criteria WB-DC-40-64 for operation of the NRWPs to mitigate a steam generator tube rupture event. The condensate demineralizer system, of which the NRWPs are a part, is designed to function during normal plant operation with a steam generator tube leak. Change out of the NRWPs has no affect on operation of the condensate demineralizer system in support of the plant with a steam generator tube leak.

The only credible failure mode associated with DCN M-39302, F-39882-A and F-39909-A is a failure of the replacement NRWPs to operate. Complete failure of the replacement pumps to operate is no different than for the existing pumps and is of no consequence since they perform no safety related function. A failure of the pumps in which they were to deliver flow in excess of their acceptable operating range is also considered with respect to the ODCM release rates. 0-FI-14-456B is used for ODCM release rate calculations and has a 200 gpm calibrated range. The replacement pumps are flow restricted with pressure reduction orifices to approximately 130 gpm within the scope of the DCN which is well within the calibrated range of 0-FI-14-456B.

Therefore, on the basis of the evaluation of effects, it is concluded that DCN's M-39302-A, F-39882-A, F-39909-A and associated FSAR Change Package 1491 are acceptable from a nuclear safety standpoint and no unreviewed safety question exists

SA-SE Number: WBPLMN-97-120-0

Implementation Date: 01/28/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39378-A	Waste Gas Compressors

Description and Safety Assessments:

The Waste Gas Compressors (WGCs) (0-OMP-77-091 and 0-OMP-77-105) were originally supplied by Westinghouse which were procured from The Nash Engineering Company as ASME Section III, Class 3 compressors. Nash Engineering no longer maintains their "N" stamp capability. Nash still manufactures these compressors as Non-ASME. Design Change Notice (DCN) Number 39378-A replaces the existing ASME Section III, Class 3 WGCs with Non-ASME compressors. The licensee, in accordance with NRC Generic Letter 89-09, can replace a component with a dedicated Non-ASME component where "replacements are no longer available in full compliance with the stamping and documentation requirements of Section III of the Code." Since Nash Engineering no longer maintains "N" stamp but still manufactures originally supplied compressor as Non-ASME quality, a note will be added to TVA flow diagram 1-47W830-4 to allow a Non-ASME compressor to be used in lieu of ASME Section III, Class 3 compressors following the guidelines contained within NRC Generic Letter 89-09. An equivalent (fit, form, and function) "N" stamp compressor from another manufacturer is not available.

FSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunctions in the Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. This DCN replaces the WGCs with a Non-ASME item that is a like for like component. 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 FSAR. 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 design basis accident (DBA), all safety related equipment is expected to operate as designed to limit the consequences of the DBA. This DCN replaces the WGCs with a Non-ASME item. The previously evaluated malfunctions of Radwaste components were reviewed and there is no increase of the consequences or these malfunctions. This change does not result in a radioactive release in excess of those established by 10 CFR 20 and 10 CFR 100 since this DCN does not create a new radioactive gaseous effluent release pathway as defined in ODCM. No new potential single failures of existing components will occur as a result of replacement of the WGCs with a Non-ASME item. 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 except for containment isolation valves which have not been affected. The accidents and/or malfunctions associated with the Radwaste system is a failure of Waste Gas Decay Tank (FSAR Section 15.3.5) or associated piping and failure of Radwaste components. Although this change does affect Radwaste components, this equipment is not used in the mitigation of these accident/malfunctions and does not change the radiological consequences of an accident previously evaluated in the FSAR.

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 Gaseous Waste Disposal System are not revised or challenged by these changes.

SA-SE Number: WBPLMN-97-120-0 ***Implementation Date:*** 01/28/1998

This system's associated components, piping, and valves are located in the Auxiliary Building on Elevation 713. This equipment does not perform a primary safety function (except for containment isolation valves 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 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 change and equipment, although apart of the Gaseous Waste Disposal System, is not associated with the accident described above, does not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis. 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 Safety Evaluation conclusions that this change is safe and does not constitute an unreviewed safety question."

SA-SE Number: WBPLMN-97-127-0

Implementation Date: 12/29/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number W-39723-A	DCN W-39723-A resolves WBPER961211 by adding an alarm vent trim package to alarm check valves.

Description and Safety Assessments:

DCN W-39723-A resolves WBPER961211 by adding an alarm vent trim package to alarm fire protection system check valves 0-ACKV-26-334, -1978, -1985, -2010 and -3134 located in the Modifications Building (MDB), Training Center, Engineering and Quality Building (EQB) and Power Stores Renovation. The current configuration of the trim packages associated with alarm check valves without water motor alarms cannot compensate for pressure surges on the system and as a result the pressure switches in the trim packages are being actuated which in turn causes an unwanted alarm and unwanted automatic starting of the electric driven fire pumps. The manufacturers of the alarm check valves recommend that an alarm vent trim package be used with the alarm check valves when a water motor alarm is not used. This is the case with the four valves identified above. The DCN also removes the automatic start of the electric driven fire pumps from pressure switch PS-26-3126 which is associated with alarm check valve ACKV-26-3134. None of these valves are located in or provide fire suppression for a safety related structure. The modification to these valves does not affect the function or response of fire protection equipment that provides fire protection for safety related structures, systems, or components. The valves are not covered by a Technical Specification (TS) nor can they impact a component covered by a TS. The fire protection of the buildings associated with the valves does not affect the plant's ability to fight a fire in a safety related structure. The only-design basis event that credits the use of the electric driven fire pumps is the Flood Event. The electric driven fire pumps can be used as a source to supply water for auxiliary feedwater. Per 10CFR50.48, Appendix R a fire is not postulated to occur concurrent with a design basis event. This modification has no impact on fuel cladding, reactor coolant systems, or containment integrity; therefore, it does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-97-147-0

Implementation Date: 07/28/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39789-A FSAR Package 1501	Testing Frequency of the Main Turbine Throttle Valves.

Description and Safety Assessments:

FSAR Change Package 1501 and DCN S-39789-A revises FSAR Section 10.2.3.6.2 and System Description N3-47-4002 (Part 2), Section 6. 1, respectively, to change the testing frequency of the Main Turbine throttle valves, governor valves, reheat stop valves, and reheat intercept valves from one month to three months based upon the Westinghouse approval letter TG-97012 and NRC recommendation via Generic Letter 93-05.

There are no WBN design basis events for which the Turbine Generator Control and Protection System (TGCPS) is required to operate. The High Pressure (HP) Turbine and associated throttle, governor, reheat stop, and reheat intercept valves are not safety related and are not required to perform a primary or secondary nuclear safety function. The major plant safety concern for the TGCPS is the prevention of generation of turbine missiles due to turbine overspeed condition (uncontrolled run away of the turbine) which could impact safety related equipment. There is no increase in the probability of the generation of internal missiles as a result of this DCN change. Based on the Westinghouse study TM-95125 and reported in letter TG-97012, the requirement for total overall probability of a turbine missile damaging a safety related system must remain below 10E-7 (Note that 2.79 x 10E-7 for two units operating is the licensing basis for Watts Bar). Based on all the variables that factor into the equation for calculating the overall probability of a turbine missile damaging a safety related system, the study determines that a 10E-5 allowable probability is left for a missile ejection from the turbine. The study calculates that the probability for ejection of a missile from the turbine is less than 10E-5, concluding that the effect of the extended valve testing interval is negligible. Also, Design Criteria WB-DC-40-65, "Missiles" concludes the potential for turbine generated missiles has been determined to be credible but not significant.

The implementation of the subject FSAR and DCN change does not introduce different failure modes from the existing turbine valving configuration and the valve testing frequency change affects only a non-safety grade system that has no accident mitigation function. The change in valve testing frequency does not affect the ability of the overspeed protection to close the turbine valving. The only credible failure mode would be the failure of a throttle, governor, reheat stop, or reheat intercept valve to close and remain unchanged.

The specific design basis accident evaluated by this safety evaluation is the Condition II fault, "Loss of External Electrical Load and/or Turbine Trip" (FSAR Section 15.2.7). This accident does not specifically address failure of one or more throttle, governor, reheat stop, or reheat intercept valves to close. However, Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) trips the turbine by closing the throttle, governor, reheat stop, and reheat intercept valves. The change in valve testing frequency does not affect the ability for the AMSAC system to function. Compliance with other applicable design basis requirements is not affected by the changes and nuclear safety is not degraded. Overspeed protection can also be achieved by manual closure of the main steam isolation valves (MSIVs). Additionally, each of the subject valves will be tested quarterly. Also, each throttle and governor valve will be disassembled and inspected every 39 operating months (60 months for the reheat stop and reheat intercept valves). Therefore, the faults and operational transients of FSAR Chapter 15 have been evaluated and have been determined not to be affected by this documentation change and an unreviewed safety question does not exist.

SA-SE Number: WBPLMN-98-013-0

Implementation Date: 02/12/1998

Document Type:	***Affected Documents:***	***Title:***
Tech Spec Bases	TS Bases Change Package 98-001 TS B 3.6.9.3 TS Bases Revision 15	Swapping Logic for the Pressure Control Isolation Valves

Description and Safety Assessments:

This safety evaluation addresses the scope of Technical Specification (TS) Bases Change Package 98-001. The Bases for SR 3.6.9.3, Page B3.6-59 are revised to clarify that the swapping logic for the pressure control isolation valves is tested as part of the Emergency Gas Treatment System (EGTS) actuation/response to the Phase A containment isolation signal. Problem Evaluation Report (PER) WBPER971174 documented a condition that the EGTS logic for swapping the isolation valves was not being tested. The description of condition for WBPER971174 explains that the swapping logic was not tested as part of Surveillance Requirement 3.6.9.3 because the TS did not specifically require testing of this function. This TS Bases change implements corrective action step sequence 04 for WBPER971174. The PER also contains a corrective action step to issue a new or revise an existing surveillance instruction to implement the required testing. Work Order WO 97-014018-000 was issued to perform the first EGTS test of the swapping logic and was completed 10-10-97.

The design basis accidents evaluated in WB-DC-40-64 "Design Basis Events Design Criteria" and in FSAR Chapter 15 "Accident Analysis" have been reviewed and the Condition IV, limiting faults, event of a Major Reactor Coolant System Pipe Rupture is applicable as well as Chapter 15.5.3 which presents the environmental consequences of a postulated loss of coolant accident (LOCA). However, this TS Bases change does not impact the results and conclusions of these analyses. Plant radioactive releases are unchanged and the design basis function of the EGTS to keep LOCA generated activity releases at or below the limits specified in 10 CFR 100 is not affected.

The annulus vacuum control subsystem (AVCS) is utilized during normal operation to maintain a negative pressure in the annulus relative to the Auxiliary Building. After a LOCA, the annulus pressure is controlled at a less negative pressure by the air cleanup unit (ACU) subsystem. Each ACU train contains pressure control isolation valves and modulating pressure control dampers. The pressure control dampers in the EGTS exhaust ducts modulate to maintain annulus differential pressure at the design value. One train's isolation valves located in the EGTS exhaust ducts are normally kept in the A-AUTO position and open upon receipt of a Phase A containment isolation signal. The other train's isolation valves located in the EGTS exhaust ducts are normally kept in the A-AUTO STANDBY position and remain closed. A switchover logic exists to close the "A-AUTO" train discharge isolation valves and open the "A-AUTO STANDBY" train isolation valves if the annulus pressure is not controlled to within limits after a 45 minute time delay. Change Package 98-001 clarifies that this swapping logic will be tested as part of EGTS actuation/response to the Phase A containment isolation signal. There are no credible failure modes associated with this change. The TS Bases clarification simply helps to assure system performance for mitigation of the consequences of a LOCA.

The subject TS Bases change does not constitute an unreviewed safety question because the design basis function of the EGTS to keep LOCA generated radiological releases at or below the limits specified in 10 CFR 100 is unchanged. The probability of a LOCA, of which the radiological consequences are mitigated by the EGTS, is not increased and the probability of an EGTS train malfunction while in the LOCA mitigation mode is not increased. The radiological consequences of a LOCA as presented in FSAR chapter 15.5.3 are unchanged because the input assumptions used in these analyses are not changed. The radiological consequences of a malfunction of a train of RHR are not changed because the TS Bases change enhances, rather than degrades, EGTS availability for accident mitigation. The EGTS system actuation/response testing, including the swapping logic, can be performed during a plant outage when the EGTS is not required to be operable such that the possibility for an accident or equipment malfunction of a different type than evaluated previously in the FSAR is not created. TS Bases Change 98-001 deals with a clarification of the scope of required EGTS system testing. The change does not affect any system performance requirements, such that the possibility to reduce the margin of safety as defined in the basis for any TS does not exist.

SA-SE Number: WBPLMN-98-020-0

Implementation Date: 03/20/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN W-39814-A	Roughing Cooler Skid Added to Provide Cooling Water to Conductivity Analyzers

Description and Safety Assessments:

The Condensate Polishing Demineralizer System (CPDS) is sampled for conductivity. This sample temperature has increased to approximately 120 degrees F with the CPDS operating temperature of 125 - 135 degrees F. The conductivity analyzers, even though they are suppose to condensate and correct for temperatures greater than 77 degrees F, are not providing the desired result when compared to a grab sample that has a roughing cooler.

DCN 39814-A adds a roughing cooler skid to provide cooling water to each of the conductivity analyzers. This will reduce the sample temperature and provide better results at high temperatures. The cooling water, just like the samples to the conductivity analyzers, is routed to CPDS for further processing. The CPDS is sampled and analyzed prior to release (if required), and the release has a radiation monitor (0-RE-90-225) that automatically isolates the discharge flow.

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, this change ensures that the potential radioactive fluid is processed to the CPDS where the activity is verified with sampling and analysis prior to release. This DCN does not change the logic or function of any system that is important to safety. These changes are within the existing design basis limitations of the ODCM and, therefore, do not represent change to radioactive release criteria or result in higher discharge concentrations (non-radioactive).

FSAR does not identify any equipment faults which could occur as a result of changes. Also, changes are 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 FSAR. This revision to the FSAR does not change or affect the design basis for any system that is important to safety.

This change does not alter the system design from an operational perspective. The equipment involved in the modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety. Additional components have been added by this change. These components, if a malfunction occurs, would not cause radioactive releases in excess of the limits established by 10 CFR 20 and 10 CFR 100 since both the sample and cooling water are routed to the CPDS and a release from the CPDS is permitted only when the activity is below the limit as defined in ODCM. No new potential single failures of existing components will occur as a result of the new operational philosophy. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The CPDS, its associated components, and piping do not perform any accident mitigation function. This change does 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 CPDS are not revised or challenged by these changes.

SA-SE Number: WBPLMN-98-020-0 ***Implementation Date:*** 03/20/1998

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. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. These affected system's associated components, and piping do not perform any accident mitigation. 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 FSAR.

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. Therefore, these changes are safe and does not constitute and unreviewed safety question.

SA-SE Number: WBPLMN-98-021-0

Implementation Date: 10/08/1998

Document Type:	*Affected Documents:*	*Title:*
Design Change	DCN M39817A	Changing of the material type from Stainless Steel to Alloy 20/Carpenter 20. in the Condensate Demineralizer system

Description and Safety Assessments:

The changing of the material type from Stainless Steel to Alloy 20/Carpenter 20 in the Condensate Demineralizer system is acceptable from a nuclear safety stand point. The Alloy 20 material is very resistant to all concentrations of sulfuric acid. The corrosion rate is excellent at room temperature, approximately 0.002 mils per year, and satisfactory, less than 0.050 mils per year, up to 150 degrees, while stainless steel is unsatisfactory with a corrosion rate of greater than 0.50 mils per year for concentrations less than 80%. Both materials are equally resistant to caustic. In accordance with ASME/ANSI B31.1, the allowable stresses are better for Alloy 20 than for stainless steel pipe, therefore, this material may be utilized without redesign of the system, supports etc. Since the Alloy 20 has better resistance to acid corrosion the probability of a leak or spill has been reduced.

The addition of the isolation valves creating double isolation will provide a more positive means of isolating the tanks for maintenance activities and future repairs

The deletion of the root valves is acceptable from a nuclear safety stand point. The current system design and operation requires the high crud tank (HCT) contents to be considered waste and is discharged to the cooling tower blowdown or the Turbine Building sump (if radioactivity levels are within discharge specifications) or, otherwise, is discharged to the radwaste facility for treatment. Thus, the conductivity measurement of the (HCT) contents is of no value and the associated instrument loops are to be removed under (pending) DCN 39771. As a result, this DCN will remove conductivity elements, O-CE-14-177 and - 179, from the process line. Their associated root valves, O-RTV-14-474A and -475A, will not be reinstalled. The removal of these conductivity elements does not impact radiation measurements of any associated effluent releases to the cooling tower blowdown path. This will also reduce the probability of leakage by deleting a potential leak path and reduce the maintenance cost by not having, to maintain the valves.

In addition the condensate demineralizer system is not safety related and is not discussed in the Technical Specifications. The condensate demineralizer system is not required to perform any function as required for safe shut down of the reactor.

The new material which is more corrosion resistant and has a higher allowable stresses, will have no impact on the function of the system. The addition of the isolation valves provides a more positive means of isolating the tanks for maintenance activities and repairs.

SA-SE Number: WBPLMN-98-027-0

Implementation Date: 03/06/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39825-A	Discrepancy Between Configuration Control Diagram and System Operating Instruction for the Secondary Chemical Feed System.

Description and Safety Assessments:

A drawing deviation (DD) 98-0002 was initiated to address a discrepancy between Configuration Control Diagram (CCD) 1-47W854-1 and System Operating Instruction (SOI-36.01). The SOI requires the valve 0-ISV-036-0663 to be in the normally closed position during normal plant operation. The CCD requires the valve to be in the normally open position during normal plant operation. Since the valve is equipped with an open ended pipe stub with a quick disconnect fitting for attaching a rubber hose and is used as an alternate demineralized water supply to System 36 (Secondary Chemical Feed System), the valve must be in the normally closed position. Therefore, Design Change Notice (DCN) 39825-A revises the CCD to reflect the actual plant condition and agree with the SOI-36.01. This CCD is FSAR Figure 10.3-9.

DD 98-0003 Item Number 1 addresses discrepancy between CCD 1-47W856-1 and Control Diagram 1-47W610-43-8 and Radiation Sampling drawing 1-47W625-15. The discrepancy identifies System 59 pressure indicator PI-59-352 as a System 43 indicator on drawing 1-47W610-43-8 and on drawing 1-47W625-15. The correct identification is as shown on the CCD 1-47W856-1 (PI 59-352). DCN revised drawings 1-W610-43-8 and 1-47W625-15 to show the valves in the closed position. The CCD is FSAR Figure 9.2-28. Drawings 1-47W610-43-8 and 1-47W625-15 are not FSAR figures.

DD 98-0003 Item Number 2 addresses discrepancy between drawing 1-47W625-15 and actual plant configuration. Isolation valves for the spare sample coolers on panel L-579 are for spare sample lines on the inlet of the spare coolers and are for future use. The lines attached to the valves are capped and, therefore, the drawing is being revised to agree with the actual plant configuration. DCN revised drawing 1-47W625-15 to show the valves in the normal closed position. This drawing 1-47W610-43-8 is not a FSAR figure.

This system's associated components, piping, and valves are located in the Auxiliary and Turbine Buildings. This equipment does not perform a primary safety function, are installed in a Seismic Category I and non-Seismic structures, and are not used during any accident. The Chapter 15 accident analysis does not identify any failure that is associated with revising the CCDs. 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.

FSAR does not identify any equipment faults which could occur as a result of this change. This documentation change only DCN revises the drawings to change the valves' alignment to normally close and to revise the component identifiers (CIDs) to system 59. Also, this change is not associated with 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 FSAR. This DCN does not change or affect the design basis for any system that is important to safety.

SA-SE Number: WBPLMN-98-027-0 ***Implementation Date:*** 03/06/1998

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 FSAR.

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. Therefore, these changes are safe and do not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-98-029-1

Implementation Date: 04/01/1999

Document Type:	*Affected Documents:*	*Title:*
Design Change	DCN W-39815-A DCN F-39894-A	Replacement of Main Feedwater Pump Turbine (MFPT) Drain Tank Discharge Line

Description and Safety Assessments:

This safety evaluation addresses the scope of DCNs W-39815-A and F-39894-A. This modification replaces the 6 inch diameter main feedwater pump turbine (MFPT) drain tank discharge line with a 10 inch diameter drain and adds an anti-vortex device at the MFPT condenser drain tank outlet. In addition to this modification, two new nitrogen injection ports are being added at points closer to condenser hotwell and downstream of valves 1-LCV-006-2026 and 1-ISV-006-2028. DCN F-39894-A revises a note for the, existing nitrogen injection port locations to ensure these locations are not used to inject nitrogen with more than one Turbine Driven Main Feedwater Pump (TDMFP) in service, thus preventing two phase flow across the control valves. The modifications to the MFPT condenser drain tank discharge and new nitrogen injection ports will eliminate the potential for vortexing in the drain tank 6 inch discharge line and reduce the potential for two-phase flow through the tank level control valve when the plant is operating at full power.

The design basis accidents evaluated in WB-DC-40-64, "Design Basis Events Design Criteria" and in UFSAR Chapter 15, "Accident Analyses," have been reviewed and this modification will have no impact on any existing accident evaluations or create a new accident scenario. Plant radioactive releases due to this modification are unchanged. There are no credible failure modes associated with this change for which the consequences have not been previously analyzed in the UFSAR. The change to UFSAR Figure (Drawing 1-47W805-2, Revision 24) is minor in that only the pipe size changes from 6 inch to 10 inch and two new one half inch nitrogen injection ports are added to the drawing. In addition DCN F-39894-A revises the note for the existing nitrogen injection locations from "Optional Nitrogen Injection Ports" to "This port is not to be used for Nitrogen Injection when more than one TDMFP is in service."

The subject DCN changes do not constitute an unreviewed safety question because operation of the condenser drain system is unchanged. The frequency in which this mode of operation must be entered is not increased and the probability of an condenser drain malfunction while in any mode is not increased. Also, the changes are 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 revision to the UFSAR does not change or affect the design basis for any system that is important to safety.

SA-SE Number: WBPLMN-98-036-0

Implementation Date: 10/01/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number S-39878-A FSAR Figure 9.3-15, Sheet 3	Resolution of drawing deviations - Primary Water Makeup, CVCS, and High Pressure Fire Protection Systems

Description and Safety Assessments:

This safety evaluation addresses the scope of DCN S-39878-A which resolves drawing deviations 98-0014, 98-0018 and 98-0019. Drawing deviation 98-0014 identifies a discrepancy between a flow diagram drawing and a control diagram drawing where the flow diagram depicts a level indicator (LI) as LI-81-1A and the control diagram shows same level indicator as LI-81-1B. The DCN revises the flow diagram to show the level indicator as LI-81 -1B to be consistent with the control diagram. Drawing deviation 98-0018 identifies a discrepancy where the flow diagram shows 1-ISV-62-945 normally closed and the system lineup checklist from S01-62.06 shows this valve in the open position. The flow path from valve 1-62-945 goes to either Holdup Tank A or Holdup Tank B. Leaving this valve in the normally open position allows flow from Monitor Tank Pumps when the Monitor Tank Pump valve alignment is to the Holdup Tanks. The DCN revises the flow diagram to show this valve in the normally open position. Drawing deviation 98-0019 identifies a discrepancy on a flow diagram where valve 0-ISV-26-3049 is shown as normally open although immediately downstream of this valve is a cap effectively dead ending flow path. The DCN revises flow diagram to show this valve as normally closed.

The design basis accidents evaluated in WB-DC-40-64 "Design Basis Events Design Criteria" and in UFSAR Chapter 15 "Accident Analysis" have been reviewed and this documentation only change does not impact the results and conclusions of these analyses. Plant radioactive releases are unchanged and continue to remain well below the limits of 10 CFR 100.

There are no credible failure modes associated with this change. The only change to the UFSAR is a revision to Figure 9.3-15, Sheet 3, which will show valve 1-62-945 in the normally open position. Fire Protection Report Figure 11-9 is also affected by DCN by changing valve 0-26-3049 from normally open to normally closed. These changes have no impact on any credible failure modes.

The subject revisions to flow diagrams do not constitute an unreviewed safety question because operation of the Fire Protection, Chemical and Volume Control and Primary Makeup Water is unchanged. The frequency in which this mode of operation must be entered is not increased and the probability of a malfunction while in any mode is not increased. The radiological consequences of performing the refueling operation itself are unchanged. The radiological consequences of a malfunction of any kind associated with this DCN are not changed because operation of the systems is not changed. System operation is in accordance with Technical Specifications and is unchanged such that the possibility for an accident or equipment malfunction of a different type than evaluated previously in the UFSAR is not created. The subject drawing changes do not require operation of the Systems affected by this DCN to be in conflict with the Technical Specifications and no physical changes are performed such that the margins of safety as defined in Tech Spec bases are not reduced.

SA-SE Number: WBPLMN-98-042-0

Implementation Date: 08/20/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	M-39827-A	Deletion of mechanical controls for Flow Control Valves and Installation of Electrical and Air Controls in the Fire Protection System.

Description and Safety Assessments:

This safety evaluation addresses the scope of DCN 39827-A which deletes the mechanical controls for Flow Control Valves FCV-26-3145 and FCV-26-3146 and installs electrical and air controls. Each valve will have an electrical interlock which will shut down the System 31 Chillers upon high pressure from the System 026 fire pumps start. A pressure switch will be installed at the existing up-stream root valve for each flow control valve. These switches will monitor header pressure to actuate a solenoid and relieve air pressure that will close the flow control valves. The valves will close upon loss of electrical power or loss of air (i.e., air to open spring to close).

1. The design basis accidents evaluated in design criteria WB-DC-40-64 "Design Basis Events Design Criteria" and in UFSAR Chapter 15 "Accident Analyses" have been reviewed and this modification will have no impact on any existing or create any new accident evaluations. Plant radioactive releases due to this modification are unchanged and continue to remain well below the limits of 10 CFR 100.

2. This modification adds additional failure modes from loss of power or air (i.e., air to open spring to close), however these failures will close the valve insuring water is available for Fire Protection system. This modification will ensure adequate water pressure and flow are available upon demand by the Fire Protection System.

3. The only design basis event that credits the use of the electric driven fire pumps is the Flood Event. The electric driven fire pumps can be used as a source to supply water for auxiliary feedwater. This modification will insure water is available for the fire protection system. Per 10 CFR 50.48, Appendix R, a fire is not postulated to occur concurrent with a design basis event. This modification has no impact on fuel cladding, reactor coolant systems, or containment integrity; therefore, it does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-98-044-0

Implementation Date: 07/15/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number S-39947-A and FSAR Change Package Number 1520	Updated FSAR review/verification programs - Sections 3.8, 3.11, 6.2, 6.5, 6.8, and 9.4

Description and Safety Assessments:

This change package addresses document changes identified as part of the Updated FSAR Review/Verification Program. 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 and documents revised as appropriate. Design Change Notice revises the UFSAR, system description documents, design criteria, calculations, and drawings for documentation consistency. These changes are for "documentation" only with no impact on WBN's design bases or operational configuration. All document changes have been evaluated for plant operability during the review process and found not to affect the physical plant.

The design criteria WB-DC-40-64, which analyzes the operational transients and the design basis accidents to demonstrate that the plant can be operated without undue risk to the health and safety of the public, has been reviewed and concluded that the results and conclusions therein have not been impacted by these "documentation only" changes.

This safety evaluation addresses FSAR changes submitted under the FSAR Change Package Number 1520, and design document changes effected by DCN. These changes were identified during the UFSAR re-review project to provide consistency among the FSAR, 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. 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 FSAR Change Package Number 1520 and/or the DCN implement documentation-only changes, not affecting the design bases of systems/equipment, or their functional/operational characteristics, these changes do not constitute a USQ.

SA-SE Number: WBPLMN-98-045-0

Implementation Date: 06/17/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39842A FSAR Figure 5.1-1	Safety classification for loop seal drain line.

Description and Safety Assessments:

DCN S-39842-A revises the safety classification for the 3/4 inch loop seal drain line from pressurizer relief valves 1-RFV-068-0563, -0564, -0565 up to and including 3/4 inch diameter instrument sense line to I-LT-068-0320 from Safety Class 2 to Safety Class 1. In addition to this change a note has been added to flow diagram 1-47W813-1 to clarify the Safety Class 1 exemption for instrument sense lines to pressure transmitters 1-PT-068-322/323,-334 and -340. Design Criteria WB-DC-40-36 (Section 3.5) is also being revised to add a clarification note which states that even though 3/8 inch flow restrictors are installed on the loop seal drain lines they are not exempted from safety class 1 requirements since the basis for this exemption is not valid for this particular line, this is discussed in detail below.

The original bases for classifying the above lines as TVA Class B piping can be found in UFSAR Section 3.2, System description N3-68-4001 Section 3.2.7, Design Criteria WB-DC-40-36 Section 3.5 and Note 17 on flow diagram 1-47W813-1. The basis for this piping exemption was derived from the August 1970 draft issue of the ANS Document, "Nuclear Safety Criteria for Stationary Pressurized Water Reactor Plants (later adopted as ANSI/ANS-51.1 - 1983) which Westinghouse references in their Systems Standard 2.5.8. The above referenced documents allow portions of the Reactor Coolant System (RCS) with 0.375 inches orifices installed and the 0.83 inches inside diameter instrument lines from the steam space portion of the pressurizer to be exempted from Safety Class 1 and be classified as Safety Class 2. This exemption can be taken since RCS liquid level can be maintained by normal charging system arrangement, however the pressurizer heaters cannot generate enough steam mass to compensate for the instrument line break steam loss. The result would be a decrease in pressurizer pressure which would eventually result in boiling occurring at the surface of the pressurizer liquid, at this point flashing would occur and steam mass loss would be made up. This design methodology assumes that flashing would occur only in the pressurizer and that sufficient subcooling would exist in the rest of the RCS. Furthermore, a reactor trip and possibly safety injection could be activated on low pressurizer pressure. If safety injection is activated pressurizer level will be maintained by the safety injection until Emergency Instructions are activated at which point the safety injection system would continue to be used to perform an "orderly shutdown and cooldown" as defined by ANSI/ANS-51.1-1983 as "A shutdown and cooldown in which the fuel and reactor coolant pressure boundary conditions are within technical specification operational limits. Automatic actuation of an engineered safety feature may be required." This is documented by Westinghouse calculation RFS-DAP-1365 dated 8-12-71, titled, "Sizing criteria for RCS Flow Restrictors." The instrument lines that are sensing pressure in the steam space of the pressurizer have an inside diameter of 0.83 inches or less and therefore, per above mentioned Westinghouse calculation the amount of Reactor Coolant lost due to pipe rupture for this line would be less than normal charging pump could makeup. Specifically the maximum steam leakage from the pressurizer at 2250 psia to the containment through a 0.83 inches inside diameter instrumentation nozzle was calculated to be approximately 16.3 lb/sec., which is equivalent to approximately 118 gpm at 2250 psia. Maximum charging pump makeup is 20.5 lb/sec., or 149 gpm which will compensate for mass lost though a postulated instrument line break. This meets the criteria for establishing a Safety Class 1 to Safety Class 2 piping classification break, therefore, pressurizer steam space instrument lines do not require Safety Class 1 designation (TVA Class A) and will remain classified as Safety Class 2 (TVA Class B).

The flow diagram 1-47W813-3 will be revised to remove the Class B designation from the loop seal drain lines mentioned above since charging pump makeup is not adequate to compensate for total mass loss that would occur if a break in this line is postulated. This line has the potential to lose approximately 177 gpm which is greater than the 149 gpm charging pump can supply. Documentation has been reviewed for this portion of pipe, fittings and valves and Class A equivalency has been established for this portion of the RCS, per DCN S-39842-A.

SA-SE Number: **Implementation Date:**

The design basis accidents evaluated is WB-DC-40-64 "Design Basis Events Design Criteria" and in UFSAR Chapter 15 "Accident Analyses" have been reviewed and this design change will have no impact on any existing or create any new accident evaluations. Plant radioactive releases due to this design change are unchanged and continue to remain well below the limits of 10 CFR 100.

There are no credible failure modes associated with this change. The change to UFSAR Figure 5.1-1 Sheet 1 (TVA Drawing 1-47W813-1) is a documentation only change and any potential failure mode that exists for the subject instrument sense lines/loop seal drain lines has previously been analyzed.

UFSAR does not identify any equipment faults which could occur as a result of changes. Changes made by DCN S-39842-A, which are documentation only, do not impact any protective features used to detect and mitigate the effects of any events. This revision to the UFSAR does not change or affect the design basis for any system that is important to safety.

The subject DCN changes do not constitute an unreviewed safety question (USQ) because operation of the RCS is unchanged. The frequency in which this mode of operation must be entered is not increased and the probability of an RCS malfunction while in any mode is not increased. The radiological consequences are-unchanged. The radiological consequences of a malfunction of the RCS system are not changed because operation of the RCS system is not changed.

SA-SE Number: WBPLMN-98-048-0

Implementation Date: 06/09/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39916-A FSAR Figure 9.2-29C	Documentation of the actual plant configuration for the Service Building Potable Water System.

Description and Safety Assessments:

This safety assessment addresses the scope of DCN S-39916-A which documents the actual plant configuration for the Service Building potable water system. Engineering Change Notice (ECN) 5748 revised as designed drawing 1-47W835-3 to show modifications to be made to the Service Building potable water system to add offices in the Turbine Building. The revisions were carried forward with back circles labeled 0-OSB-E-5748 to show the modifications have not been completed up through the current revision of the drawing. The ECN was voided on 10/06/1995 after the as-constructed drawing was issued but prior to any field work being performed and the information was not removed from the drawing. A field walkdown and review of the history drawings before the ECN was incorporated verified the correct configuration of the piping. Drawing Deviation 98-0028 was generated to resolve this discrepancy and DCN S-39916-A revises the flow diagram and piping drawings to reflect the as-built configuration. Implementation of this design change enhances legibility and understanding of the drawing. Potable water is non-essential for normal plant operation and safe shutdown of the nuclear reactor. This change has no affect on the potable water supply to other areas of the plant. The potable water system is a non-safety and non-seismic system. The Service Building (the area affected by the subject changes) is a non-safety related, non-seismic, non-plant process structure.

The design basis accidents evaluated in WB-DC-40-64 “Design Basis Events Design Criteria” and in UFSAR Chapter 15 'Accident Analyses' have been reviewed and this modification will have no impact on any existing or create new accident evaluations. Plant radioactive releases due to this modification are unchanged and continue to remain well below the limits of 10 CFR 100.

There are no credible failure modes associated with this change. The change to UFSAR Figure (Drawing 1-47W835-3, Revision 14) is minor in that only water closets, lavatories, and a kitchen unit are affected.

The subject DCN changes do not constitute an unreviewed safety question because operation of the Potable Water system does not affect any safety systems nor have any affect on plant operations. Potable Water is not cross-connected to any radioactive system nor does it have any impact on radiological releases or on any system or components which control or mitigate the affects of any releases.

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

SA-SE Number: WBPLMN-98-051-0

Implementation Date: 08/21/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-39823-A FSAR Package. Number 1518	Accumulator Room and Instrument Room maximum normal temperature.

Description and Safety Assessments:

WBPER970832 identified that temperatures in the Reactor Building Accumulator Room Number 2 and Reactor Building Instrument Room have exceeded the temperature limitations specified on Environmental Data Drawings (EDDs) 47E23542 and -45. DCN S- 39823-A revises these drawings to reflect new temperature limits for these areas as established by revision to Harsh Environment Calculation WBNAPS4-008, R 16. The Accumulator Room Number 2 maximum normal temperature is revised from 120 degrees F for 99% of plant life to 130 degrees F for 95% of plant life and maximum abnormal temperature from 130 degrees F for 1 % of plant life to 140 degrees F for 5% of plant life. The Instrument Room maximum normal temperature is revised from 75 degrees F to 100 degrees F while the maximum abnormal temperature (120 degrees F) is not changed. In addition, this DCN deletes the requirement for Reactor Building spaces that individual abnormal temperature excursions be limited to 12 hours. The 75 degrees F Instrument Room maximum normal temperature is quoted in UFSAR Section 9.4.7.1. In addition, FSAR Section 3.11.2.1 states the abnormal temperatures can exist for up to 1% of plant life, with individual excursions limited to 12 hours. Correction of the UFSAR to reflect changes by DCN S-39823 will be addressed in UFSAR Change Package 1518. In addition, this DCN revises the Environmental Qualification (EQ) binders to reflect the temperature changes and the revised qualified lives of affected equipment as determined by the associated qualified life calculations. This condition results in a reduction of the qualified life of components located in the affected rooms. The reduced qualified life, however, is addressed in the EQ binders, and will have no adverse affect on equipment operation during normal and accident conditions. The Civil piping analysis for affected piping in Reactor Building Accumulator Room Number 2 and Reactor Building Instrument Room have been evaluated with the new temperature limitations and durations and these changes will have no adverse affect on piping performance.

The change in temperature limits as described above will result in a reduction in the EQ qualified life for safety related equipment located in the Reactor Building Accumulator Room Number 2 and the Reactor Building Instrument Room. As part of the DCN, the EQ Binders will be revised to address the reduction in qualified life. The reduced qualified life values will be tracked under the EQ program and equipment will be replaced, if necessary, before the end of the qualified life. Deletion of the 12 hour limit for individual abnormal temperature excursions does not affect equipment qualification since the qualified life of equipment is determined by the overall percent of plant life at which the equipment experiences the maximum normal and the maximum abnormal temperatures. Reevaluation of piping analysis for the rooms considering the revised temperature limitations has confirmed the piping will perform as designed. Consequently, the equipment will remain functional during normal and accident conditions. The changes do not result in an increase in the probability or consequences of accidents currently evaluated in Chapter 15 of the UFSAR, does not result in different accidents or malfunctions than evaluated in the UFSAR, and does not reduce the margin of safety as defined in the Technical Specification Bases.

SA-SE Number: WBPLMN-98-053-0

Implementation Date: 06/17/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number S-39861-A SOI 62.02 & AOI-34 FSAR Figure 9.3-15-2 FSAR Figure 9.3-15-9	Revision to flow diagram and electrical drawings to show valve as normally open.

Description and Safety Assessments:

This safety assessment addresses the scope of DCN S-39861-A which revises flow diagram 1-47W809-2 and electrical drawings 1-47W610-62-3 and 1-47W611-62-2 to show valve 1-FCV-062-0140A as normally open. Flow Control Valve 1-FCV-062-0140A controls the flow of Boric Acid to the Boric Acid Blender (1-BLDR-062-0123) and based on operational experience this valve is normally in the open position. Since this valve is a fail open valve the conservative position for it to be depicted on the above noted drawings is in the open position. Since adding boron to the system enhances reactivity in the core, it is acceptable to be shown in the normally open position. This valve is controlled such that it delivers an amount of Boric Acid to the Blender or Charging pump suction to maintain a preset concentration to match the RCS (Chemistry Boron concentration).

The design basis accidents evaluated in WB-DC-40-64 "Design Basis Events Design Criteria" and in UFSAR Chapter 15 'Accident Analyses' have been reviewed and this design change will have no impact an any existing or create any new accident evaluations. Plant radioactive releases due to this design change are unchanged and continue to remain well below the limits of 10 CFR 100.

There are no credible failure modes associated with this change. The change to UFSAR Figures 9.3-15-2 and 9.3-15-9 (TVA Drawings 1-47W809-2 and 1-47W610-62-3, respectively) is a documentation only change and any potential failure mode that exists for the subject boron supply line has previously been analyzed.

UFSAR does not identify any equipment faults which could occur as a result of this change. Changes made by DCN S-39861-A, which are documentation only, do not impact any protective features used to detect and mitigate the effects of any events. This revision to the UFSAR does not change or affect the design basis for any system that is important to safety.

The subject DCN changes do not constitute an unreviewed safety question (USQ) because operation of the chemical and volume control system (CVCS) is unchanged. The frequency in which this mode of operation must be entered is not increased and the probability of an CVCS malfunction while in any mode is not increased. The radiological consequences are unchanged. The radiological consequences of a malfunction of the CVCS system are not changed because operation of the CVCS system is not changed.

SA-SE Number: WBPLMN-98-054-0

Implementation Date: 08/04/1998

Document Type:	*Affected Documents:*	*Title:*
Design Change	DCN W-39898-A Fire Protection Report Figure	Removal of Fire Protection System valves and associated piping.

Description and Safety Assessments:

This modification removes the isolation valves 0-ISV-26-237 and -646 and associated piping and installs a plug at the tees. Valve 0-ISV-26-237 was the isolation valve for the hydrants in the Low Level Radwaste Storage area. These hydrants have been removed and this valve is no longer required. Valve 0-ISV-26-646 was the isolation valve for Interim Office Buildings A and B fire protection system. These buildings were removed and all the fire protection piping was removed. Neither of these valves are located in a portion of the fire protection piping that is required for safety related structures or for Appendix R compliance, nor does this modification change the hydraulic performance of a fire protection system that is required for any safety related structure, system or component. Therefore this modification has no impact on fuel cladding, reactor coolant systems, or containment integrity and it does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-98-059-0

Implementation Date: 06/12/1998

Document Type:	***Affected Documents:***	***Title:***
Procedure	SOI-70.01, Change Notice A	Realign component cooling system due to leaking temperature well 1-TW-70-154B on Residual Heat Removal System Heat Exchanger 1B-B.

Description and Safety Assessments:

Temperature well 1-TW-70-154B on component cooling system (CCS) to residual heat removal (RHR) system heat exchanger 1B-B (1-HTX-74-31) outlet is leaking CCS water to an area drain. System Operating Instructions (SOI) SOI-70.01 is being changed to realign the CCS to maximize service to the affected components during the replacement of the temperature well. The CCS flow to the RHR heat exchanger will be diverted and rerouted to the other components and then back to the inlet of the CCS pumps. The RHR heat exchanger will be taken out of service in order to replace the temperature well. This Safety Evaluation addresses only the rerouting of the CCS flow. The RHR heat exchanger is being taken out of service under Limiting Condition of Operation (LCO) 3.5.2 as required by the appropriate Technical Specification (TS). SOI-70.01 is being revised to realign CCS and establish this alternate flow path. The normal flow path for CCS B heat exchanger (2-HTX-70-185) is to the waste gas Compressor B heat exchanger and spent fuel pool cooling (SFPC) B heat exchanger, and then this flow is returned to the suction of the CCS pumps. The alternate flow path will isolate the CCS B heat exchanger and reestablish this flow path with the CCS C heat exchanger including associate users for CCS C heat exchanger except for RHR heat exchanger 1B-B. This path provides sufficient flow for the CCS pumps to prevent pump damage as a result of removing the major RHR heat exchanger 1B-B flow branch from service.

FSAR does not identify any equipment faults which could occur as a result of this change. This SOI revision does not change or affect the design basis for any system that is important to safety. These changes do not affect any equipment relied upon for safe operation or shutdown. In the event of a DBA, all safety related equipment which is being relied upon 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 temporary procedure. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. This S0I revision does not change the radiological consequences of an accident previously evaluated in the FSAR.

These changes do not reduce the margin of safety identified in the applicable TS. These changes do not prevent any component from performing its function as described in the TS. RHR injection is lost as recognized by entry into LCO 3.5.2 and loss of RHR cooling is addressed through entry into LCO 3.7.7.

This system's associated components, piping, and valves are located in the Auxiliary and Reactor Buildings. This equipment does perform a primary safety function, is installed in a Seismic Category I structure, and is used during any accident. The Chapter 15 accident analyses does not identify any failure that is associated with revising the procedure. This change and equipment are not associated with increasing the consequences of an accident previously evaluated, and is bounded by the existing analyses. This SOI revision 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.

SA-SE Number: WBPLMN-98-062-0

Implementation Date: 01/04/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN W-39961-A	Unit 1/Unit 2 Interface Boundary

Description and Safety Assessments:

When the Reactor Coolant System (RCS) temperature is within limits, the RCS letdown flows to the Chemical Volume Control System (CVCS) demineralizers. The letdown passes through one of two CVCS mixed bed demineralizers. The mixed bed demineralizer cleans the coolant by removing ionic impurities and corrosion products, certain fission products and acts as a filter. One demineralizer is in continuous service and the second mixed bed demineralizer serves as a standby unit for use if the operating demineralizer becomes exhausted. Highly radioactive spent resins (changed for reasons of pressure drop or activity levels) are initially fluidized by backflushing with primary water and are then flushed to the solid waste disposal system spent resin storage tank (SRST) via the spent resin transfer piping.

The transfer piping contains valves that isolate this piping from equipment not required for Unit 1 operation. The Grinnell diaphragm valves 1-ISV-62-995 and 2-ISV-62-995 which are normally closed are apart of the Unit 1 and 2 (UI/U2) interface program and isolate the CVCS Evaporator Condensate Demineralizer A and B from the transfer piping. These demineralizers are not required for Unit 1 operation and these valves leak through in the closed position. This was identified during transfer of the CVCS Mixed Bed A demineralizer to the SRST.

DCN 39961-A adds a blank plate between the transfer piping and the UI/U2 interface valves used to isolate the resin header. The plate will become the new UI/U2 interface and the valves will be placed in the Unit 2 boundary (not required for Unit 1 operation). This will provide an additional measure of protection to ensure that when the highly radioactive spent resins are being transferred to the SRST, cross contamination from Unit 1 to Unit 2 equipment can not occur.

These systems' associated components, piping, and valves are located in the Auxiliary Building. The equipment associated with this change does not perform a primary safety function, is installed in a Seismic Category I 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 Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decay Tank or associated piping. This modification and equipment are not associated with the accident described above, do not increase the consequences of an accident previously evaluated, and are bounded by the existing analysis. This DCN does not change the logic or function of any system that is important to safety. 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). A review of the detailed changes leads to the conclusions that this change is safe and does not constitute an unreviewed safety question.

FSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunctions in the liquid radwaste processing system such as pump or valve failures or evaporator failures. This change does not involve any of this type of equipment. This change and equipment do not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis. 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 modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety.

This change does not alter the system design from an operational perspective. The equipment involved in the modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety. Additional components have been added by this change. These components, if a malfunction occurs, would not cause radioactive releases in excess of the limits established by 10 CFR 20 and 10 CFR 100. 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

SA-SE Number: WBPLMN-998-062-0 ***Implementation Date:*** 01/04/1999

to safety to fail to fulfill its functional requirements. This change does 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 CPDS are not revised or challenged by these changes.

These changes do not affect any equipment required for safe operation or shutdown. In the event of a design basis accident, 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 change. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. These affected system's associated components, and piping for this modification 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 FSAR.

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.

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Implementation Date: 07/30/1998

Document Type:	Affected Documents:	Title:
Design Change	DCN S-39964-A FSAR Figures 10.4.2 and 10.4-5	Drawing Deviations on U1/U2 Boundary.

Description and Safety Assessments:

This safety evaluation addresses the scope of DCN S-39964-A which revises physical drawings 17W303-1 and 47W435-3. Drawing deviation 98-0035 identified a discrepancy between flow diagram 1-47W831-1 and physical piping drawing 17W303-1 with regards to where Unit 1/2 boundary is maintained. The flow diagram is the upper tier document and as such the physical piping drawing was revised to match the flow diagram. Drawing deviation 98-0037 identified a portion of the piping drawing (47W435-3) which did not have a mark number identifying the type of material used. DCN S-39964-A adds a mark number to this portion of piping consistent with the type of material originally installed. In process of revising physical piping drawing 17W303-1 it was noted that further discrepancies existed on flow diagram 1-47W831-1 and control-diagram 1-47W610-27-2. These changes, which added 2-LS-27-93B and 2-LS-27-93D to Unit 2 cross-hatching from flow diagram 1-47W831-1 and removed basin drain sluice gate (UNID 2-ISV-027-0522) for the Unit 2 Cooling Tower from the Unit 2 cross-hatching, will make the flow and control diagram consistent with each other. System 27 "Condenser circulating Water" is a non-safety related system and, therefore, not required to mitigate any postulated design basis event. Safety injection system (SIS) is a safety related system and DCN adds a mark number to the pumps (1-PMP-063-0010A-A and 1-PMP-063-0015B-B) which are required during a small break LOCA. Adding this mark number has no effect on pumps/systems functionality and will not impact systems ability to perform it's safety function.

The design basis accidents evaluated in WB-DC-40-64 "Design Basis Events Design Criteria" and in FSAR Chapter 15 "Accident Analysis" have been reviewed and this documentation only change does not impact the results and conclusions of these analyses. Plant radioactive releases are unchanged and continue to remain well below the limits of 10 CFR 100.

There are no credible failure modes associated with this change. The change to FSAR Figures 10.4-2 and 10.4-5 are considered minor in nature and do not affect the functionality of either system. The FSAR figures will be updated as part of routine UFSAR update while the flow and control diagrams (1-47W831-1 and 1-47W610-27-2, respectively) will be updated upon closure of DCN S-39964-A to reflect the actual configuration of the plant.

The subject FSAR figure changes and associated flow/control drawing and physical piping drawing changes do not constitute an unreviewed safety question because operation of the SIS and condenser circulating system is unchanged. The frequency of any mode of operation is not increased and the probability of an SIS or condenser circulating system malfunction while in any mode is not increased. The radiological consequences of performing these drawing only changes is unchanged. SIS and the condenser circulating system operations are in accordance with Technical Specifications and are unchanged such that the possibility for an accident or equipment malfunction of a different type than evaluated previously in the UFSAR is not created. The subject DCN changes do not require operation of the SIS or condenser circulating system to be in conflict with Technical Specifications and no physical changes are performed such that the margins of safety as defined in Technical Specification bases are not reduced. Therefore this change does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-98-074-0

Implementation Date: 04/19/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change FSAR	DCN M-39923-A FSAR Change Package 1527	Replacement and relocation of twelve primary containment thermal expansion protection check valves.

Description and Safety Assessments:

Some of the process fluid supply and return lines which penetrate the Watts Bar primary containment have thermal expansion protection bypass check valves. These check valves maintain containment integrity, and prevent piping overpressurization caused by the thermal expansion of the process fluid which may be trapped between the inboard and outboard containment isolation valves in the main process line after containment isolation. The check valves are located in bypass lines around the inboard containment isolation valves. The check valves relieve into the process piping on the inboard side of the isolation valve via the bypass lines when the trapped process fluid expands. These check valves and bypass lines are a portion of the (TVA Class B) containment boundary.

In the current configuration of the twelve check valves, the check valves are located in low points of the piping. This results in particulate settling in the check valves, therefore, preventing the check valves from adequately closing. This results in creating a leak path from containment should the outboard isolation valve fail to close. Excessive failure of the check valves reduces the reliability of the primary containment capability to prevent radioactive releases in excess of the leakage rate limits in the event of a design basis accident (DBA).

DCN M-39923-A authorizes the relocation of the eight Essential Raw Cooling Water (ERCW) and four Ice Condenser (Glycol) containment isolation thermal expansion protection check valves. The check valves are moved from low points in the bypass lines, to positions that are the same elevation as the process lines, or at least above the existing low point. 1-CKV-061-0533 is the only one of the check valves that is not located at the same elevation as its process line, however, it is above the low point (1'-6" above the low point). Relocating the check valves at these higher elevations will reduce the likelihood of particulate settling in the check valves.

The check valves will be replaced with stainless steel check valves. The ERCW piping will be replaced with stainless steel piping as is currently installed. The Glycol piping will be replaced with carbon steel piping as is currently installed, and the valve and piping being replaced will be increased from 3/8"diameter components to 1/2" diameter components.

Essential Raw Cooling Water and Ice Condenser Systems are required to mitigate LOCAs (Loss-of-Coolant Accidents) and HELBs (High Energy Line Breaks) inside containment (FSAR 3.11-1). However, the portions inside the containment isolation boundary are not required to mitigate a DBA. These portions of the ERCW and Glycol Systems are isolated during these events.

A LOCA is a hypothetical accident that would result in the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system to maintain reactor inventory. A pipe rupture in the reactor coolant system is considered a LOCA when the flowrate is greater than the equivalent from a 3/8-inch diameter hole. A line break in a fluid system that during normal plant conditions is either in operation or maintained pressurized under the conditions where maximum operating temperature exceeds 200°F is conservatively classified as an HELB.

The credible failure modes of the check valves are 1) failure to isolate primary containment, or 2) failure to relieve thermally expanding fluid that may be trapped between the inboard and outboard containment isolation valves after the containment has been isolated.

These thermal expansion overpressure protection check valves do not receive a containment isolation signal from any design basis event. However, these active valves have to perform their safety functions of maintaining containment integrity, and overpressure protection relief.

SA-SE Number: WBPLMN-98-074-0 ***Implementation Date:*** 04/19/1999

The replacement valves meet the design specifications of the existing valves; all non-metallic materials used in the check valves are suitable for Harsh Environmental conditions. The resilient sent in each new check valve has a service life of 40 years. Relocating the bypass lines and replacing the valves, spring-loaded at a higher differential pressure than the existing check valves, increases the reliability of the check valves. The relocation of the check valves above the low points in the bypass loops reduces the likelihood of particulate settlement in the valves.

None of the existing design or functional requirements of the thermal expansion protection valves (i.e.. opening to prevent piping overpressurization and closing to maintain containment integrity) or the bypass piping have been shored by this modification. The modifications will be performed under conditions which comply with Technical Specification operability requirements. Therefore, this activity is not an unreviewed safety question.

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

Implementation Date: 09/01/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-40002-A 1-TRI-47-1 PAI-6.02 FSAR Change Package 1536	Mechanical overspeed trip, low vacuum trip, low bearing oil pressure trip and thrust bearing oil trip mechanisms testing frequency.

Description and Safety Assessments:

FSAR Change Package 1536 and DCN S40002-A revises FSAR Section 10.2.3.6.2 and System Description N3-47-4002 (Part 2), Section 6.1, respectively, to change the testing frequency of the mechanical overspeed trip, low vacuum trip, low bearing oil pressure trip and thrust bearing oil trip mechanisms from one month to quarterly (plus 25% industry standard margin) based upon the Westinghouse approval letter number ESS-98-0101. The quarterly testing frequency is consistent with the testing of the Main Turbine valving. It is noted that the Technical Requirements Manual (TRM) TR 3.3.5, "Turbine Overspeed Protection," requires trip testing, but does not specifically identify testing frequency and references procedure PAI-6.02 as the document defining test frequency. Therefore, the TRM is not required to be revised.

There are no WBN design basis events for which the Turbine Overspeed Protection System (TOPS) is required to operate to mitigate an accident. The high pressure (HP) Turbine and associated valving and the trip mechanisms are not safety related and are not required to perform a primary or secondary nuclear safety function. The major plant safety concern (economic) for the TOPS is the prevention of generation of turbine missiles due to turbine overspeed condition (uncontrolled run away of the turbine). There is no significant increase in the probability of the generation of turbine missiles as a result of this DCN change. Based on the Westinghouse evaluation SAE/RRA-073 (98) and reported in letter ESS-98-0101 the total overall probability of a turbine generated missile remains below 1 x 10E-5 per year when the valves and trip mechanisms are tested quarterly. This evaluation determines that the effect of the extended testing interval is negligible and remains bounded by existing analyses for the FSAR The 10E-5 probability for a missile ejection from the turbine is bounded by the Value used in the design basis calculation. TI-521 "Probabilities for Turbine Missile Strike Damage at Watts Bar Nuclear Plant", which concludes that damaging a safety related system remains below 10E-7. This information is also presented in FSAR Section 3.5.1.3, "Turbine Missiles." Design Criteria WB-DC-40-65, "Missiles" concludes the potential for turbine generated missiles has been determined to be credible but not significant. This DCN does not impact nuclear safety because the simulated turbine overspeed and trip block tests have proven the turbine trip mechanisms to be highly reliable at SQN Units 1 and 2 and WBN Unit 1. Since the frequency of current performance exceeds current requirements the plant is undergoing unnecessary testing. This challenges the human element to perform the test more often than required. It is therefore safer to plant operation to reduce the frequency of testing these highly reliable components. Since an inadvertent turbine trip is a challenge to plant safety systems, specifically initiating Auxiliary Feedwater injection, transients to the plant are reduced which in turn improves the plant's performance from a nuclear safety perspective.

The specific design basis accident evaluated is the Condition II fault, "Loss of External Electrical Load and/or Turbine Trip" (FSAR Section 15.2.7). This accident does not specifically address failure of the trip mechanisms to trip the Main Turbine. However, anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) trips the turbine by closing the throttle, governor, reheat stop, and reheat intercept valves. The change in trip mechanism testing frequency does not affect the ability for the AMSAC system to function. Compliance with other applicable design basis requirements is not affected by the changes and nuclear safety is not degraded. Additionally, each of the subject tripping mechanisms (excluding the redundant electrical trip) and main turbine valving will be tested quarterly. Also, each throttle and governor valve will be disassembled and inspected every 39 operating months (60 months for the reheat stop and reheat intercept valves). Therefore, the faults and operational transients of FSAR Chapter 15 have been evaluated and are not affected by this documentation change and an unreviewed safety question does not exist.

SA-SE Number: WBPLMN-98-081-0 ***Implementation Date:*** 09/01/1998

The implementation of the subject FSAR and DCN change does not introduce different failure modes from the existing turbine mechanism configuration and the tripping mechanisms testing frequency change affects a non-safety grade system that has no accident mitigation function. The change in testing frequency does not affect the ability of the overspeed protection to close the turbine valving and shutdown the main turbine. The only credible failure modes would be the failure of a throttle, governor, reheat stop, or reheat intercept valve to close or one of the tripping mechanisms failing to function and these failure modes remain unchanged.

SA-SE Number: WBPLMN-98-085-1

Implementation Date: 01/05/1999

Document Type:	***Affected Documents:***	***Title:***
Technical Specification Bases	Technical Specification Bases Change Package 98009, Revision 23	Residual Heat Removal (RHR) Loop Operability

Description and Safety Assessments:

This safety evaluation addresses the scope of DCN S-40007-A, Technical Specification Bases Change Package and FSAR Change Package 1538. The changes are documentation only and are made for the purpose of more accurately describing RHR operation in support of filling the refueling cavity for refueling operations and for performance of injection testing, such as inservice check valve/flow testing. The proposed change to the WBN Bases is consistent with a change to the Standard MERITS Technical Specifications, NUREG- 1431 R1. This generic change (Technical Specification Traveler Form (TSTF) 2 1, Revision 1). was made to the BASES of LCO 3.9.6 to allow alignment of RHR suction to the RWST for the purpose of filling the cavity or testing and was approved by NRC September 18, 1996, provided the change conforms to the licensing basis.

The Technical Specification Bases for specifications 3.9.5 is revised to clarify that both RHR pumps may be aligned to the Refueling Water Storage Tank (RWST) for continued filling of the refueling cavity or for performing RHR hot leg or cold leg injection testing. This Bases change further requires that during these modes of operation, the RCS temperature is monitored using the RCS wide range temperature indicators. The flow path for filling the refueling cavity and for cold leg injection testing is from the RWST to the RCS cold legs. The RHR loops remain operable during this condition since adequate decay heat removal capacity is maintained, adequate boron mixing is maintained, and the RCS temperature is monitored. During hot leg injection testing with only one pump operating, the RHR loop being tested is operable (since the loop can be realigned to inject into the cold legs) but considered out of service since significant decay heat is not removed by the RHR during this mode of operation. Consequently, this testing must be performed under the existing requirements of the Technical Specifications (i.e., 23 feet of water is required above the reactor vessel flange for backup decay heat removal and the required train of RHR can be out of service for up to one hour per eight hour period).

The Technical Specification Bases for specification 3.9.6 is revised to clarify that both RHR pumps may be aligned to the Refueling Water Storage Tank for filling of the refueling cavity or for performing RHR hot leg or cold leg injection testing. This Bases change further requires that during these modes of operation, the RCS temperature is monitored using the RCS wide range temperature indicators. The flow path for filling the refueling cavity and for cold leg injection testing is from the RWST to the RCS cold legs. The Technical Specification Bases is also revised to state that RHR hot leg injection testing may be done provided the other RHR train is injecting into the RCS cold legs. The RHR loops remain operable during the above conditions since adequate decay heat removal capacity is maintained, adequate boron mixing is maintained, and the RCS temperature is monitored.

This change revises the Technical Specification Bases for specifications 3.9.5 to clarify that both RHR pumps may be aligned to the Refueling Water Storage Tank (RWST) for continued filling of the refueling cavity or for performing RHR hot leg or cold leg injection testing. This Bases change further requires that during these modes of operation, the RCS temperature is monitored using the RCS wide range temperature indicators. The flow path for filling the refueling cavity and for cold leg injection testing is from the RWST to the RCS cold legs. The RHR loops remain operable during this condition since adequate decay heat removal capacity is maintained, adequate boron mixing is maintained, and the RCS temperature is monitored. During hot leg injection testing with only one pump operating, the RHR loop being tested is operable (since the loop can be realigned to inject into the cold legs) but considered out of service since significant decay heat is not removed by the RHR during this mode of operation. Consequently, this testing must be performed under the existing requirements of the Technical Specifications (i.e., 23 feet of water is required above the reactor vessel flange for backup decay heat removal and the required train of RHR can be out of service for up to one hour per eight hour period).

SA-SE Number: WBPLMN-98-085-1 ***Implementation Date:*** 01/05/1999

This change also revises the Technical Specification Bases for specification 3.9.6 to clarify that both RHR pumps may be aligned to the Refueling Water Storage Tank for filling of the refueling cavity or for performing RHR hot leg or cold leg injection testing. This Bases change further requires that during these modes of operation, the RCS temperature is monitored using the RCS wide range temperature indicators. The flow path for filling the refueling cavity and for cold leg injection testing is from the RWST to the RCS cold legs. The Technical Specification Bases is also revised to state that RHR hot leg injection testing may be done provided the other RHR train is injecting into the RCS cold legs. The RHR loops remain operable during the above conditions since adequate decay heat removal capacity is maintained, adequate boron mixing is maintained, and the RCS temperature is monitored.

The subject Technical Specification Bases changes do not constitute an unreviewed safety question because the ability of the RHR system to perform the required functions of decay heat removal and boron mixing during refueling operations is unchanged, and RCS temperature indication is provided by the wide range RCS temperature indicators. Since the frequency in which the refueling mode of operation must be entered is not increased and no specific RHR design bases accidents are postulated to occur while operating in the refueling mode, the probability or consequences of an accident previously evaluated in the FSAR is not increased. The radiological consequences of performing the refueling operation itself are unchanged. The probability or radiological consequences of a malfunction of a train of RHR are not changed because the operability requirements of the RHR trains are unchanged with the exception of the origin of the suction source while filling the refueling cavity. No new equipment is introduced by the proposed changes. Further, the RHR system will continue to meet the operability and single failure requirements of the current Technical Specifications. Consequently, the proposed activity does not create a possibility for an accident of a different type than any evaluated previously in the FSAR. The proposed changes do not introduce new equipment or operational requirements that results in use of existing equipment that is not evaluated for malfunctions in the UFSAR. Consequently, the proposed activity does not create a possibility for a malfunction of a different type than any evaluated previously in the FSAR During both the refueling cavity filling and injection testing modes of operations addressed by the proposed changes, the RHR system can maintain the RCS water within the temperature limits and accomplish boron mixing as required by the Technical Specifications. The RHR system will continue to meet the operability and single failure requirements of the current Technical Specifications. Consequently, the proposed activity will not reduce the margin of safety as defined in the bases of the Technical Specifications.

SA-SE Number: WBPLMN-98-086-0

Implementation Date: 09/23/1998

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-40001-A FSAR Figure 9.3-15, Sheet 1 FSAR Figure 5.1-1, Sheet 1	Drawing deviation for RCS valve type.

Description and Safety Assessments:

This safety evaluation addresses the scope of DCN S-40001-A which revises flow diagrams 1-47W809- 1, 1-47W813-1, and physical drawing 47W406-8. Drawing Deviation 98-0044 identified valves 1-RTV-068-1103 and -1106 that are needle type valves but are shown as globe valves an the flow diagram, 1-47W813-1. Valves 1-RTV-068-1103 and -1106 are 1500-lb. Kerotest Y-type needle globe valves, per the Kerotest drawing TVD-D-9954N-(2), Contract 74C35-83015, Mark Number 47W465-17, (Reference ASME NPV-1 form). The NPV-1 form describes these valves as globe type. Therefore, these valves are best shown on the flow diagram as globe valves. No change is required.

Additionally, during the review for this drawing deviation, valves 1 BYV-068-0552 and -0555 are globe valves but are shown as needle valves on flow diagram 1-47W813-1. Valves 1-BYV-068-0552 and -0555 are 1500-lb. Kerotest Y-type globe valves, Mark Number 47W465-20. These valves are true globe valves and should be shown as such on the flow diagram. The flow diagram, 1-47W813-1 was revised to show these valves as globe type.

Revised flow diagram 1-47W809-1 to change the valve symbol for valve 1-THV-062-0620 from a needle valve to a globe valve. Revised piping drawing 47W406-8 to change the mark number for the valves from 47W406-93 to 47W406-65.

The corresponding valves on Loops 2, 3, and 4 (1 -THV-062-0621, -0622, and -0623, respectively) are also incorrectly shown on the flow diagram as needle valves. Therefore, the symbols for these valves were revised to a globe valve. These valves are correctly shown on the piping drawing as mark number 47W406-93.

No modifications, re-tagging, or other field work is required for any of the above changes. Revising the subject drawings has no effect on the systems functionality and will not impact systems ability to perform its safety function.

The design basis accidents evaluated in WB-DC-40-64 "Design Basis Events Design Criteria" and in FSAR Chapter 15 "Accident Analysis," have been reviewed and this documentation only change does not impact the results/conclusions of these analyses. Plant radioactive releases are unchanged and are well below the limits of 10 CFR 100.

There are no credible failure modes associated with this change. The change to FSAR Figures 9.3-15, Sheet 1, and 5.1-1, Sheet 1, are considered minor in nature and do not affect the functionality of either system. The figures will be updated as part of routine UFSAR update while the flow and control diagrams (1-47W809-1 and 1-47W813-1, respectively) will be updated upon closure of DCN S-40001-A to reflect the actual configuration of the plant

The subject FSAR figure changes and associated flow drawings and physical piping drawing changes do not constitute an unreviewed safety question because operation of the reactor coolant system (RCS) and chemical and volume control system (CVCS) is unchanged. The frequency of any mode of operation is not increased and the probability of an RCS or CVCS malfunction while in any mode is not increased. RCS and the CVCS operations are in accordance with Technical Specifications (TS) and are unchanged such that the possibility for an accident or equipment malfunction of a different type than evaluated previously in the UFSAR is not created. The subject DCN changes do not require operation of the RCS or CVCS to be in conflict with TS and no physical changes are performed such that the margins of safety as defined in TS bases are not reduced. DCN S-40001-A is a documentation only DCN. No field work required. The scope of DCN S-40001-A is to revise flow diagrams 1-47W809-1 and 1-47W813-1 to show the correct valve type symbols. There are no unreviewed safety questions.

SA-SE Number: WBPLMN-98-088-0

Implementation Date: 08/03/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN No. M-39991-A	Incorporation of revised net heat input (NHI) value into design basis.

Description and Safety Assessments:

DCN M-39991-A, Stage 1, incorporates NHI value analysis based on a WBN specific evaluation of heat losses/additions into the design basis. Technical specifications require the plant to perform a power calorimetric to confirm that the reactor thermal power is within acceptable limits and the nuclear instrumentation system (NIS) output is consistent with the calculated power. The power calorimetric is performed in accordance with plant procedures and the plant computer to calculate the reactor thermal power by subtracting the reactor coolant system (RCS) net heat input from the total power generated from the steam generators (which represents the total NSSS power).

The current NHI value is 14 MWt and is based on a generic calculation by Westinghouse for a representative 4 loop plant design. A new value of 16.0 MWt has been calculated based on specific WBN heat inputs and heat losses using the latest, most accurate system and component design and operating data. TVA's letter, dated 5/12/97 to Westinghouse provided WBN data required for calculating net heat input value. The NHI value was calculated by Westinghouse by summing all the RCS heat inputs and subtracting off all the heat losses. Additional details on the specifics of the RCS heat balance are contained within Westinghouse letter, dated 9/22/98

The new value will replace the existing 14 MWt value used in the plant computer and in plant procedures to calculate power calorimetric. The practical impact of this change is that the plant will be able to improve its thermal output by 2.0 MWt (16-14 MIA) While still maintaining core power within the applicable licensed power of 3411 MWt.

This increase in NHI value will be reflected in the FSAR (FSAR change package 1535) as the value of 14 MWt is presented directly and indirectly in some Sections and Tables of the FSAR. There are no Appendix R components or equipment, or any nuclear safety-related systems or portions of systems adversely affected by the proposed change in NHI. Therefore, this modification will not affect any design basis accidents or anticipated operational transients.

This change in NHI value does not make any physical changes to the plant; therefore there are no credible failure modes affected by this change and there are no new types of failures/malfunctions created by this change.

This change in the NHI value is based on a WBN specific evaluation of heat losses/additions to the system as oppose to the previous evaluation for a generic 4-loop plant. Various plant analyses were evaluated to confirm either the existing analyses were bounding or there was an insignificant effect on the existing analyses results. The NHI value is used in the calculation of the RCS thermal hydraulic design parameters which are used in many of the analysis; however, the revised net heat input value did not require a change to the RCS temperature, pressure and flow values as it has an insignificant effect on the design basis steam (<O. 1 % increase), temperature (<0.5°F decrease) and pressure (<1 psi decrease) that are used in the analyses. WBN licensed core power of 3411 MWt remains unchanged by this increase NHI value. Based on these evaluations, the change in NHI value will not increase the probability of occurrence or the consequences of an accident or malfunction previously evaluated in the FSAR or create the possibility of an accident or a malfunction of a different type from those previously evaluated in the FSAR. This change does not reduce the margin of safety for any bases presented in the Technical Specification. Therefore, this change does not constitute an unreviewed safety question.

SA-SE Number: WBPLMN-98-091-0

Implementation Date: 04/07/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39943-A	Flow Accelerated Corrosion Program.
FSAR	FSAR Change Package 1549	

Description and Safety Assessments:

DCN M-39943-A implements the requirements of the Flow Accelerated Corrosion (FAC) program for portions of the Feedwater System. The pipe and fittings being replaced are:

The 6" Sch 80 elbow between 1 FCV-3-236 and 1-CKV-3-652

The 6" Sch 80 elbow and 7' of 6" Sch 80 pipe downstream of 1-FCV-3-239.

The 6" Sch 80 elbow and 4' of 6" Sch 80 pipe downstream of 1-FCV-3-242..

The 6" Sch 80 pipe and fittings between 1-FCV-3-245 and 1-CKV-3-638.

The FAC grids will be re-established on the new piping and will be similar to the existing grids. A new wall thickness baseline will be established during construction in accordance with the FAC program.

The portion of these lines being replace were identified by UT during RFO-1, and predictions made in accordance with the FAC Program determined these sections need to be replaced prior to RFO-3. The replacement will be done as part of the Feedwater System. The affected piping is TVA Class B (ASME Section III Class 2). The configuration of the piping is to remain the same as is now installed and shall utilize existing pipe supports.

These lines are being replace using 2 ¼% Chrome, 1% Molybdenum pipe (SA 335 Gr P22) and fittings (SA 234 Gr WP22 Class 3) material using the same size, schedule, and configuration that was originally installed. 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. These substitutions are documented in revisions to Piping Analysis Calculations 0600200-02-05, 060200-05-02, 0600200-05-01, and 0600200-02-08. The existing carbon steel valves adjacent to the replaced piping will be reused if in acceptable condition. However, if any valve <u>must</u> be replaced due to degradation, it will be replaced with a like-for-like valve during the implementation of this DCN. Future replacements will be made with the same type of valve made of appropriate material during a subsequent outage under another DCN.

The replacement material is a low-alloy pressure retaining material which requires compliance with Regulatory Guide 1.50 - Control of Preheat Temperature for Welding of Low-alloy Steel. FSAR Section 10.3.6.2 says there are no low-alloy pressure retaining materials used in the feedwater system, therefore, this change deviates from information presented in the FSAR, and a revision to the FSAR is required.

SA-SE Number: WBPLMN-98-091-0 ***Implementation Date:*** 04/07/1999

Section 15.3.2 of the FSAR gives the accident analysis for the failure mode of a Minor Secondary System Pipe Break, and Section 15.4.2.2 provides the accident analysis for the failure mode of a Major Rupture of a Main Feedwater Pipe. The failure modes are the same for either carbon steel or chrome-moly material, and this change has no affect on either analysis.

The material replacement is an enhancement that will provide a more reliable system and decrease the probability of an equipment failure. The consequences of an accident or malfunction of equipment important to safety are not affected by this material replacement. The margin of safety is not reduced because the material replacement enhances the ability of the piping system to maintain its pressure retention properties because of the increased resistance to corrosion of the replacement material.

SA-SE Number: WBPLMN-98-092-1

Implementation Date: 07/19/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number S-40016 FSAR Change Package 1546	Updated FSAR Review - Section 6

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification Program [Reference Problem Evaluation Report (PER) WBPER980417]. Specifically addressed are UFSAR Section 6.7 and the associated system descriptions. 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 and documents revised as appropriate by DCN S-40016-A which partially implements the corrective action for WBPER980417. These changes are for "documentation" only with no impact on WBN's design bases or operational configuration. All document changes have been evaluated for plant operability during the review process and found not to affect the physical plant

The following is a discussion on the changes in the UFSAR:

1. Design condition h. in Section 6.7.1.1 states that the "Foam" Concrete Density is 40 lbs/ft^3. This appears to be a typo because the "Foam" Concrete Density is actually 35 lbs/ft^3 as stated in Section 6.7.1.2, Page 6.7-5, in System Description Document (SDD) N3-61-4001 Section 3.2.16, and in E-Spec 952214 EP-6.

2. The statement "The gate is held in a closed position by virtue of its design as an almost vertical flapper with a hinge at the top. The 100 angle from vertical holds the flap in place by gravity." In Section 6.7.1.3 is incorrect. The valve is actually installed in the vertical position and the 100 was given as a tolerance for installation. The gate is held in a closed position by virtue of its design as a vertical flapper with an offset hinge at the top. The gate is verified to be closed by periodic inspection which measures the distance from the gate to the seat.

3. In the last paragraph of Section 6.7.6.3 it states "at less than 11,000 Btu/hr to the ice bed." This value is actually 10,000 Btu/hr as stated on Page 6.7-30 and in WCAP7611-C Page B.2.

4. The last sentence on Page 6.7-68 is not tied to any of the other discussion. This same statement was removed from Sequoyah's UFSAR recently because it was tied to previous discussion in the UFSAR which was deleted in 1975. Water addition is not used at Watts Bar as mentioned in this sentence. Sequoyah Electric Generation Plant evaluated the affect of ice compaction vs. flake ice on the performance of the Ice Condenser System and concluded that the affects of solid ice vs. flake ice had a negligible impact on the efficiency of the Ice Condenser System. Section 6.7.14.3 of the Watts Bar UFSAR discusses the possibility of compaction of ice in the Ice condenser System and notes that the compaction of the Ice would be limited to 4 inches in every 6-foot section. Adding a statement to this section noting that the affects of compacted ice has a negligible affect on the Ice Condenser efficiency is a clarification.

5. Two references are made to forty-eight resistance temperature detectors (RTD) in Section 6.7.15.5. Table 6.7-24 references TE 164 which is noted as a spare. Figure 6.7-39 refers to 48 RTD's and as noted above there are only 47 at Watts Bar. One of the RTD's was meant to be a spare and was never installed at Watts Bar, therefore there are only 47 RTD's at Watts Bar. As noted one RTD was a spare so 47 RTD's is adequate coverage.

SA-SE Number: WBPLMN-98-092-1 **Implementation Date:** 07/19/1999

6. Section 6.7.18.2 states that "Galvanizing is done in accordance with ASTM, Al 23, or A386." Galvanizing is done at Watts Bar per ASTM, A123. ASTM, A386 was incorporated into ASTM, Al 23 and all design documents refer to ASTM, Al 23.

7. The UFSAR Section 6.7.18.2 gives a 50° F - 150° F temperature range in which galvanized material is not expected to fail due to corrosion during a 40 year exposure. 150° F is the average temperature, not the maximum expected temperature. This statement does not add value as there is sufficient justification given in the rest of the text. Additionally it could not be readily verified where this statement came from and will be deleted.

8. The materials in Table 6.7-4 do not match the materials on design drawing (1191E57). The materials changes made in Table 6.7-4 were reviewed and evaluated by Westinghouse in WAT-D-10583 and found acceptable. Westinghouse evaluated the yield stresses for the revised materials and found that they were all equal to or greater than the minimum required yield stresses used in their calculations for the design of those components.

The changes to the system description were all deletions of text which was already available in other sections of the system description or other design documents or were made to agree with the correct information given in the UFSAR. None of these changes have any affect on the UFSAR which is not discussed above.

This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification Program (Reference WBPER980417). Specifically addressed are UFSAR Section 6.7 (Ice Condenser System) and the associated system description 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 and documents revised as appropriate by DCN S-40016-A which partially implements the corrective action for WBPER980417. These changes are for "documentation" only with no impact on WBN's design bases or operational configuration. All document changes have been evaluated for plant operability during the review process and found not to affect the physical plant. The proposed "documentation only" changes to the UFSAR and the design documents will not increase the likelihood of the design basis accidents occurring. These changes to the UFSAR and the design documents do not affect physical changes to the plant, nor do they involve any plant procedures.

These documentation-only changes will not increase the dose to the public analyzed in the UFSAR Chapter 15.5. No change will occur to the radiological consequences of accidents analyzed as a result of these changes. The proposed changes do not affect 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 SAR and concluded they do not introduce a failure pathway different from those identified and evaluated in the SAR 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. The bases of the Technical Specifications have been reviewed for determining if any margins of safety are affected by these documentation changes. No margin of safety is identified in the bases section of the Technical Specifications which could be reduced by these changes.

SA-SE Number: WBPLMN-98-093-0

Implementation Date: 07/30/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number S-40022-A FSAR Change Package Number 1528	Documentation changes identified as part of the FSAR review. (Section 9.2)

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification Program. Specifically addressed are UFSAR sections 9.2.1 through 9.2.5, 9.2.7 and 9.2.8 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 and documents revised as appropriate by DCN S-40022-A which partially implements the corrective action. These changes are- for "documentation" only with no impact on WBN's design bases or operational configuration. All document changes have been evaluated for plant operability during the review process and found not to affect the physical plant. The changes are summarized below:

These documentation-only changes will not increase the dose to the public analyzed in the UFSAR Chapter 15.5. 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 FSAR and concluded they do not introduce a failure pathway different from those identified and evaluated in the FSAR 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. The bases of the Technical Specifications have been reviewed for determining if any margins of safety are affected by these documentation changes. No margin of safety is identified in the bases section of the Technical Specifications which could be reduced by these changes.

SA-SE Number: WBPLMN-98-094-1

Implementation Date: 05/05/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39992-A	Gland Steam Condenser Retubing
FSAR	FSAR Change Package 1558	

Description and Safety Assessments:

WBN has implemented a program to remove copper materials from the condensate/steam cycle systems in an effort to extend the life of the steam generators. Nuclear industry experience has determined that high copper concentration contributes to tube denting and failure in the steam generators.

As part of this program, DCN M-39992-A was initiated to replace the copper tubes currently installed in the gland steam condenser (GSC). The GSC for the Unit 1 turbine has 17 BWG 90-10 copper nickel tubing which contributes to high copper concentrations in the condensate and feedwater systems. DCN M-39992-A replaces the existing tubes in the gland steam condenser with stainless steel tubes. These copper alloy tubes are the last significant source of copper remaining on the secondary side of the plant.

The GSC is a heat exchanger designed to condense steam pulled off of the turbine shaft seals, turbine valve leakoff, and various other sources. The GSC is TVA Class H and is located on Elevation 729.0 near column lines T2 and H in the Turbine Building. The GSC uses condensate from the hotwell pump discharge in the tubes as the cooling medium. The steam on the shell side is condensed and drained to the 8 inch vent line from the atmospheric drain tank, while non-condensable gases are discharged to atmosphere by fans.

The existing copper-nickel tubes are being replaced with stainless steel tubes to help prevent tube failures and to remove a source of copper from the secondary side of the plant.

DCN M-39992-A also installs two sets of pipe flanges, one in the inlet piping and the other in the outlet piping, to facilitate inspection and maintenance activities of the gland steam condenser.

The replacement of 17 BWG 90-10 copper-nickel tubes in the GSC with 20 BWG stainless steel tubes under DCN M-39992-A has been determined to be acceptable. The wall thickness of the 20 BWG stainless steel tubes is acceptable for the existing design parameters. The GSC will continue to perform its function of condensing steam which is received from the high pressure turbine seals, the low pressure turbine seals, the main steam throttle valve leak-offs, and the main feed pump turbine valve leak-offs and condensers acceptably. There is sufficient margin in the performance of the GSC to permit tube plugging if required. The change in pressure drop through the GSC will slightly increase the condensate flow through the GSC. This results in a slight reduction in the flow to the steam generator blowdown 2nd stage heat exchanger; however, the steam generator blowdown 2nd stage heat exchanger will still have acceptable flow provided to it to perform its function of cooling the steam generator blowdown.

SA-SE Number: WBPLMN-98-094-1 ***Implementation Date:*** 05/05/1999

The GSC and its associated piping are not included in the evaluation of any accident in the FSAR. The design and operational requirements of the GSC have not been changed, other than having a different tube material which will be less susceptible to tube failure. Pipe flanges are being installed, but they have the same design and installation requirements as other pipe flanges already installed in the condensate system. No new equipment is being added to the system.

Therefore:

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR will not increase as a result of the activities in DCN M-39992-A.
- A possibility of an accident or malfunction of a different type than those previously evaluated in the FSAR will not be created as a result of the activities in DCN M-39992-A.
- A margin of safety as defined in the basis for any Technical Specification will not be reduced as a result of the activities in DCN M-39992-A.

SA-SE Number: WBPLMN-98-097-0

Implementation Date: 05/14/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN S-40024-A	Reactor Coolant System (RCS) Pressure and Temperature Limits Report.

Description and Safety Assessments:

This revision to the Pressure, Temperature Limits Report (PTLR) is required to incorporate applicable data from WCAP-15048, "Analysis of Capsule U from the Tennessee Valley Authority Watts Bar Unit 1 Reactor Vessel Radiation Surveillance Program." The changes primarily involve Table 4.0-1, "Surveillance Capsule Removal Schedule," and Table 5.1, "Comparison of the Watts Bar Unit 1 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decrease with Regulatory Guide 1.99, Revision 2 Predictions."

The changes in Table 4.0-1 involve pulling back the times for the withdrawal of Capsules W, X and Z, revising the lead factor for all six capsules, and revising the fluence values for the four primary capsules based on the measured fluence for Capsule U and the predicted fluence for the other capsules. Note (b) for this table is being revised to agree with Section 7.6.3.4 of ASTM E185 - 82 which indicates that the capsule should be removed at the outage closest to the time shown in the table rather than prior to the time shown.

The changes to Table 5.1 involve the addition of the Capsule U results associated with the Reactor Vessel controlling materials. This is the first data to be incorporated in this table which provides a comparison between predicted values calculated using Regulatory Guide 1.99, Revision 2 methodology and measured values for materials taken from Capsule U. The values included in the table are the value for measured fluence, the predicted and measured shift in 30 ft-lb transition temperature, and the predicted and measured decrease in upper shelf energy. It should be noted that the value for upper shelf energy for Forging 05 (axial) actually increased from 62 ft-lb to 72 ft-lb; therefore, the value recorded for decrease was 0.

These are the only components (figures or tables) of the PTLR that can be revised based on data from one capsule. Revisions to other components of the PTLR require the data from two or more capsules based on Regulatory Guide 1.99, Revision 2. There are no design basis accidents or credible failure modes associated with this activity.

These changes are documentation only and do not impact analysis in the FSAR for an accident or malfunction which was previously analyzed nor does it create the potential for an accident or malfunction different than those that have been previously analyzed. This change has no impact on the margin of safety defined in the basis of any Technical Specification since there are no changes in operating procedures nor are there equipment modifications associated with it. This change, therefore, does not result in an unreviewed safety question.

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

Implementation Date: 04/05/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-39977-A	Limitorque Actuators Regearing to restore thrust margin.

Description and Safety Assessments:

The Limitorque actuators on specific MOVs within the WBN Generic Letter (GL) 89-10 population will be modified under DCN M-39977-A to increase the thrust margins. The proposed modifications and the associated documentation changes will incorporate the revised torque and thrust requirements of TVA Design Standard DS-MI8.2.21, Revision 9, and the revised sizing methodology of Limitorque's Technical Update 98-01 (as referenced in NRC Information Notice 96-48). The modifications involve replacement of the existing motor and worm shaft pinion gears with similar parts having a numerically higher gear ratio, thereby increasing the available torque. However, this will also increase the calculated GL 89-10 stroke times values from those previously calculated and, in some cases, will also affect the design basis stroke times listed in the system descriptions. The new stroke times are tabulated below:

TABLE 1: Stroke Times				
	89-10 Calculated Value		System Description Value	
MOV Number	Old (sec)	New (sec)	Old (sec)	New (sec)
1-FCV-001-0015-A	9.28	15.20	20	No Change
1-FCV-001-0016-A	9.28	15.20	20	No Change
1-FCV-001-0017-A	9.91	13.36	16	No Change
1-FCV-001-0018-A	9.28	13.36	16	No Change
1 -FCV-063-0005-B	8.91	11.62	10	14
1 -FCV-063-0008-B	13.21	18.13	15	21
1-FCV-063-0011-B	13.21	18.13	15	21
1-FCV-068-0332-B	8.82	12.74	10	17
1-FCV-068-0333-B	8.27	12.74	10	17
1-FCV-070-0087-B	15.88	24.00	66	No Change
1-FCV-070-0090-A	15.88	24.00	66	No Change

The revised design incorporates provisions for future actuator degradation consistent with current nuclear industry practice and will optimize the GL 96-05 testing frequency for the affected MOVs once such testing is implemented at WBN. The proposed modifications may also minimize the impact of future changes in the GL 89-10 program.

The subject modifications are considered enhancements which will not adversely affect the ability of the affected MOVs or any associated -safety system from satisfactorily performing their intended safety functions. Other than the increase in valve stroke times associated with the change in final drive ratios caused by the installation of the new gears, the "as left" and "as-found" configuration of these MOVs is essentially the same. The increased stroke times for the Systems 001 and 070 MOVs are still within the existing design requirements. The increased stroke times for the Systems 063 and 068 MOVs have been evaluated against the applicable design bases and Technical Specification requirements and determined not to degrade the response times 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. Failure of any single MOV modified under DCN M-39977-A will not in itself initiate any accident evaluated in the SAR or compromise the safety function of any associated safety-related system. The subject modifications 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 DCN M-39977-A.

It has been determined that implementation of the subject modifications:

- will not create the possibility of a new type of accident or equipment malfunction not previously analyzed in the FSAR 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 FSAR,
- 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 unresolved safety question and are acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-98-102-0

Implementation Date: 07/16/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	ECN Number E-50038-A; FSAR Change Package 1569	Updated FSAR revision/verification program - Chapters 1, 5, 9, and 10.

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification Program (Reference WBPER980417). Specifically addressed are UFSAR sections 1.3.2, 5.5.2, 5.5.4, 5.5.5, 5.5.9, 9.2.6. 10.1 through 10.4 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 and documents revised as appropriate by EDC E-50038-A, which partially implements the corrective action for WBPER980417. EDC E-50038-A also implements the corrective action of WBPER980870. These changes are for "documentation" only with no impact on WBN's design bases or operational configuration. All document changes have been evaluated for plant operability during the review process and found not to affect the physical plant. The proposed "documentation only" changes to the UFSAR and the design documents will not increase the likelihood of the design basis accidents occurring. These changes to the UFSAR and the design documents do not effect physical changes to the plant, nor do they involve any plant procedures. These documentation-only changes will not increase the dose to the public analyzed in the UFSAR Chapter 15.5. 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 FSAR and concluded they do not introduce a failure pathway different from those identified and evaluated in the FSAR 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 to affect the physical plant configuration or change the operating parameters of the affected systems.

A PER was written as a result of the programmatic review of nuclear industry licensing issues (LERs) distributed by the Corporate Licensing Manager. A Catawba LER indicated a lack of knowledge of specific water temperatures at all times in the Condensate Storage Tanks at Catawba which is the primary water supply for the Auxiliary Feedwater System and at Watts Bar Nuclear Plant must be maintained between 40°F and 120°F. Review of supporting WBN design input and output documents indicated inconsistencies in these documents and Westinghouse letter TG-98004 which approved operating the Main Condenser backpressure at 6.2 in HgA above 90% power. All issues were resolved and incorporated into EDC E-50038-A and FSAR Change Package Number 1569.

Changes, which are being implemented, typically fit into four categories, as follows:

I. Administrative in nature (e.g., typographical errors, misplaced/incorrect reference numbers, grammatical errors, duplicate information, excessive verbiage, text requiring clarifications, information that is no longer valid, inadvertent exclusion of text, historical information, and minor Figure changes, etc.).
II. Markings or deleting of text for features specifically identified in the UFSAR as not required for WBN Unit 1 operation or having to do with Unit 2 or 2 unit operation (e.g., the Gas stripper and boric acid evaporator package, etc.).
III. Deletions of unnecessary or non-contributory details (e.g., such as Materials used, pipe sizes, etc.).
IV. Corrections, technical in nature, i.e., revisions necessary to provide consistency among the UFSAR, System Descriptions, Design Criteria, supporting calculations, and drawings, etc.

SA-SE Number: WBPLMN-98-102-0 ***Implementation Date:*** 07/16/1999

In summation, the design document changes:

- Clarify WBN's design bases 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;
- Have been evaluated for plant operability during the review process and do not to affect the physical plant configuration or change the operational parameters of the affected systems;
- Are not expected to adversely affect NRC's understanding of the design, configuration, or operation of WBN;
- Will not alter the frequency class of any accident or event in the SAR to a higher frequency class;
- Will not adversely affect the ability of the affected systems or equipment 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 SAR, 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 or equipment as a result of an accident or malfunction of equipment;
- Will not impede access to vital areas of the plant, hamper actions required to mitigate an accident, or cause an increase in onsite or offsite dose as the result of an accident or malfunction of equipment;
- Will not adversely affect 10 CFR 20 or 10 CFR 100 compliance;
- Have been evaluated against the applicable accidents identified in the SAR with respect to the affected systems and equipment and determined not to introduce any new accident scenarios or failure pathways;
- Do not increase the probability of any analyzed accident;
- Do not involve any new single failures; and
- Have been reviewed to determine if any margins of safety specified in the bases section of the Technical Specifications might be reduced and none was identified.

Therefore; based on the above evaluation, implementation of the changes listed in Table 1 and the associated design document changes:

- Will not create the possibility of a new type of accident or equipment malfunction not previously analyzed in the FSAR;
- Will neither increase the probability nor the consequences of an accident or equipment malfunction already evaluated in the FSAR;
- 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: WBOLMN-98-103-0

Implementation Date: 02/24/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	47W760-67-4, Revision 11 47W1767-1, Revision H 47W1769-1, Revision K 47W2767-1, Revision D 47W2769-1, Revision G	Control of the Closure of ERCW Isolation Valves

Description and Safety Assessments:

This evaluation addresses the scope of DCN W-40004-A which revises applicable schematic diagrams to allow valves 1-FCV-067-0022-A, 2-FCV-067-0022-A, 1-FCV-67-0024-B, and 2-FCV-67-0024-B to be controlled by closed limit switches in lieu of existing closed torque switches. The Essential Raw Cooling Water (ERCW) strainer 1A-A inlet isolation valve, 1-FCV-067-0022-A, will not fully close due to high differential pressure across the valve. The torque switch is stopping the valve before full closure. Therefore, a red/green light combination exists when the valve completes its closure travel. Work Request C369843 was written to request that a test be run with the close torque switch bypassed in order to demonstrate that the valve will property close and seat with limit switches only. A jumper was installed across the closed torque switch and the valve went to the full close position (verified by handwheel operation) and no flow was heard through the valve. After removing the jumper, the valve was stroked from closed to open to closed and back to open. The valve was then left in the open position. With the jumper removed, the valve would not close fully and gave dual indication in the closed position. This change, DCN W-40004-A, bypasses the closed torque switch from the valve closure circuit.

Valves 1-FCV-067-0022-A, 2-FCV-067-0022-A, 1-FCV-67-0024-B, and 2-FCV-67-0024-B are administratively locked in the open position (with breakers open) due to potential Appendix R interactions. These valves are closed during the performance of Surveillance Instructions (SIs) 0-SI-67-901-A, 0-SI-67-901-B, 0-SI-67-902-A, 0-SI-67-902-B, 0-SI-67-903-A, 0-SI-67-903-B, 0-SI-67-904-A, and 0-SI-67-904-B. These SIs verify the operational readiness of the ERCW pumps.

The design basis accidents evaluated in WB-DC-40-64, "Design Basis Events Design Criteria," and in FSAR Chapter 15 "Accident Analysis," have been reviewed and this modification does not impact the results and conclusions of these analyses. Plant radioactive release criteria was reviewed and remains unchanged.

There are no credible failure modes associated with this change. A change required for FSAR Figure 8.3-35 is considered minor in nature and does not affect the functionality of System 67. The revision of the FSAR figure and the schematic diagrams do not constitute an unreviewed safety question because operation of the ERCW system has not been changed. This modification does increase or decrease mode requirements for operation of the ERCW system. Also the change does not increase the probability of an ERCW malfunction while in any mode. Operation of the ERCW system in accordance with Technical Specifications remains unchanged. Therefore, the possibility for an accident or equipment malfunction of a type different from that previously evaluated in the FSAR is not increased.

SA-SE Number: WBPLMN-98-105-0

Implementation Date: 07/14/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50046-A FSAR Change Package 1573	Updated FSAR Review - Section 3.9.3

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification Program [Reference Problem Evaluation Report (PER) WBPER980417]. Specifically addressed are UFSAR section 3.9.3 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 and documents revised as appropriate by EDC E-50046-A which partially implements the corrective action for WBPER980417. These changes are for "documentation only" with no impact on WBN's design bases or operational configuration. All document changes have been evaluated for plant operability during the review process and found not to affect the physical plant. The proposed 'documentation only' changes to the UFSAR and the design documents will not increase the likelihood of the design basis accidents occurring. These changes to the UFSAR and the design documents do not effect physical changes to the plant. These "documentation only" changes will not increase the dose to the public analyzed in the UFSAR Chapter 15.5. 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 or the physical plant. The credible failure modes for the systems affected by these changes, have been evaluated against the accidents identified in the FSAR and concluded they do not introduce a failure pathway different from those identified and evaluated in the FSAR 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. The bases of the Technical Specifications have been reviewed for determining if any margins of safety are affected by these documentation changes. No margin of safety is identified in the bases section of the Technical Specifications which could be reduced by these changes.

SA-SE Number: WBPLMN-98-108-1

Implementation Date: 05/13/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change FSAR	EDC E-50047-A FSAR Change Package 1581 & 1581S1	FSAR Review and Verification of Section 3.8.6, 9.3.1, 9.5.4, 9.5.5, 9.5.6, 9.5.7, and 9.5.8.

Description and Safety Assessments:

This safety evaluation addresses design document changes identified as part of the Updated FSAR Review/Verification Program. 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. The following FSAR sections (and their associated tables and figures, if affected) are being revised to incorporate changes that resulted from this review: Section 3.8.6, Category I(L) Cranes: Section 9.3.1, Compressed Air System; Section 9.5.4. Diesel Generator Fuel Oil Storage and Transfer System: Section 9.5.5, Diesel Generator Cooling Water System; Section 9.5.6, Diesel Generator Starting System; Section 9.5.7, Diesel Engine Lubrication System, and Section 9.5.8, Diesel Generator Combustion Air Intake and Exhaust System.

Discrepancies between the UFSAR and other design documents (such as system descriptions, design criteria, calculations, and drawings) were investigated and the affected documents revised as required to achieve documentation consistency under Engineering Document Change (EDC) E-50047-A, which partially implements the corrective action. These "documentation only" changes clarify WBN's design bases. Document changes have been evaluated for plant operability during the review process and do not to affect the physical plant configuration or change the operating parameters of the affected systems.

The UFSAR changes being implemented fit into one of the four following categories (listed in order of increasing significance):

I. Administrative in nature (i.e., non-intent changes such as corrections involving typographical errors, misplaced/in-correct reference numbers, grammatical errors, duplicate information, excessive verbiage, text requiring clarifications, information that is no longer valid, inadvertent exclusion of text, historical information, and minor figure changes).

II. Marking or deleting of text for features specifically identified in the UFSAR as not required for WBN Unit 1 operation (e.g., the Additional Diesel Generating System, etc.). These are also non-intent changes.

III. Deletions of unnecessary or non-contributory details (e.g., data not specifically called for by Reg Guide 1.70, such as airflow rates, numbers and sizes of HVAC components, etc.).

IV. Corrections, technical in nature, i.e., revisions necessary to provide consistency among the UFSAR, System Descriptions, Design Criteria, supporting calculations, drawings, or other design documents.

SA-SE Number: WBPLMN-98-108-1 ***Implementation Date:*** 05/13/1999

Where the appropriate change category was uncertain, the change was assigned to the category of greater significance to ensure it would be adequately evaluated.

Table 1: **GENERAL OVERVIEW OF CATEGORY III AND IV CHANGES**

UFSAR SECTION	DESCRIPTION OF CHANGE
Section 3.3.6 Category I(L) Cranes Figures 3.8.6-1 through 3.8.6-11	• Deleted figures showing crane arrangements and details, and the associated FSAR text references to them, because they were not required and did not contribute value or understanding to the FSAR. • Deleted a note from Tables 3.8.6-1 and 3.8.6 concerning loading direction which had been deleted from the corresponding design criteria documents by DCN S-37734-A. • Corrected the applicable edition of the National Electrical Code.
Table 9.3-7 Failure Mode and Effects Analysis, Auxiliary Air Supply Equipment	• Revised entry for valves 0-FCV-32-71 & -95 to include additional "Effects on System" not previously identified. • Added FMEA for redundant control air dryer purge valves 0-FCV-32-73 & -97 which had not been previously addressed. • Deleted entries for all Auxiliary Control Air Compressor cooling water valves formerly identified as system 32 components but which have now been redesignated as system 67 components and are addressed in the system FMEA.
Table 9.3-8 Equipment Supplied with Auxiliary Control System Air	• Revised table to designate which valves are isolated from the Compressed Air System by the Unit 1/2 interface • Added the Op Mode/Failure Mode for each valve where applicable.

SA-SE Number: WBPLMN-98-108-1 **Implementation Date:** 05/13/1999

Table 1: **GENERAL OVERVIEW OF CATEGORY III AND IV CHANGES**	
UFSAR SECTIONS	DESCRIPTION OF CHANGE
Section 9.5.4 Diesel Generator Fuel Oil Storage and Transfer System Section 9.5.5 Diesel Generator Cooling Water System Section 9.5.6 Diesel Generator Starting System Section 9.5.7 Diesel Engine Lubrication System Section 9.5.8 Diesel Generator Combustion Air Intake and Exhaust System	• Deleted figures and associated text references to the additional diesel generator, • Deleted redundant information. • Clarified/enhanced discussions of the 7-day and day tanks. • Added references to NFPA Code for 7-day tanks and 480V power sources for fuel oil pumps and valves. • Added information to fully describe the design bases for the cooling water and starting systems.

Other design documents associated with the FSAR review and revised under EDC E-50047-A include the system descriptions for the Compressed Air System and the Standby Diesel Generator System, failure modes and effects analysis (FMEA) EPM-JPJ-100892 for the Auxiliary Control Air System (part of the Compressed Air System), control diagram for the additional diesel generator (ADG) starting air system, heavy equipment drawings for the 175-ton Polar Crane, and heavy equipment drawings for the 125-ton Auxiliary Building Crane. There is no system description document for the Category I(L) cranes and no design criteria documents were affected by this activity.

The affected system descriptions were revised as required to reflect and/or complement the associated FSAR changes. The changes are of the same Category 1, 2, 3, and 4 types as described above for the FSAR. The existing plant components added to the System 32 FMEA had been inadvertently omitted in previous revisions of the analysis; however, no new failure modes or effects were introduced since these components are identical to redundant components which had been previously analyzed. Components deleted from the subject FMEA are now addressed in the System 67 (ERCW) FMEA. No other calculations were revised under EDC E-50047-A. The affected drawings for the ADG, Polar Crane, and Auxiliary Building Crane were revised to delete the FSAR figure reference where the corresponding FSAR figure had been deleted.

EDC E-50047-A addresses documentation changes to Control Air (System 32), Service Air (System 33),Standby Diesel Generator (System 32),and Containment & Auxiliary Building Cranes and Miscellaneous Heavy Equipment (System 271). These systems may be required to mitigate the consequences of various Condition 11, 111, and/or IV accident evaluated in chapters 6 & 15 of the FSAR or other events for which WBN is designed to cope (e.g., LOOP, ATWS, flooding, etc.). However: the subject changes do not have any impact on any design basis accidents or operational transients previously evaluated.

SA-SE Number: WBPLMN-98-108-1 ***Implementation Date:*** 05/13/1999

The failure modes and effects analysis (FMEA) for the Control Air system was impacted by the FSAR review (several components added/deleted). However, there were no credible failure modes introduced by the proposed documentation changes which would prevent the affected components or systems from performing their intended safety functions or which would cause or intensify any design basis accident or event.

In summation, the changes listed in Table 1 and the associated design document changes:

- Clarify WBN's design bases 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;
- Have been evaluated for plant operability during the review process and do not to affect the physical plant configuration or change the operational parameters of the affected systems,
- Are not expected to adversely affect NRC's understanding of the design, configuration, or operation of WBN;
- Will not alter the frequency class of any accident or event in the FSAR to a higher frequency class;
- Will not adversely affect the ability of the affected systems or equipment 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 FSAR, 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 or equipment as a result of an accident or malfunction of equipment;
- Will not impede access to vital areas of the plant, hamper actions required to mitigate an accident, or cause an increase in onsite or offsite dose as the result of an accident or malfunction of equipment;
- Will not adversely affect 10 CFR 20 or 10 CFR 100 compliance;
- Have been evaluated against the applicable accidents identified in the SAR with respect to the
- affected systems and equipment and determined not to introduce any new accident scenarios or
- failure pathways;
- Do not increase the probability of any analyzed accident;
- Do not involve any new single failures; and
- Have been reviewed to determine if any margins of safety specified in the bases section of the Technical Specifications might be reduced and none was identified.

SA-SE Number: WBPLMN-98-108-1 ***Implementation Date:*** 05/13/1999

Therefore; based on the above evaluation, implementation of the changes listed in Table 1 and the associated design document changes:

- will not create the possibility of a new type of accident or equipment malfunction not previously analyzed in the FSAR;
- will neither increase the probability nor the consequences of an accident or equipment malfunction already evaluated in the FSAR;
- 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-98-109-0

Implementation Date: 05/18/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change FSAR	EDC E-50064-A FSAR Change Package 1580	FSAR Review and Verification of Sections 9.1.1, 9.1.3, 9.1.4, 9.3.6, and 9.3.8.

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the Updated FSAR review and verification effort. Specifically addressed are changes associated with engineering design change EDC E-50064-A. These include revisions to UFSAR sections 9.1.1, "New Fuel Storage", 9.1.3 "Spent Fuel Pool Cooling and Cleanup System (SFPCCS)", 9.1.4 "Fuel Handling System", 9.3.6 "Auxiliary Charging System", 9.3.8 "Heat Tracing", System Description Documents (SDDs) N3-78-4001 "Spent Fuel Pool Cooling and Cleaning System", N3-79-4001 "Fuel Handling and Storage System", and N3-84-4001 "Flood Mode Boration." Regulatory Guide 1.70, November 1978, "Standard Format and Content of Safety Evaluation Reports for Nuclear Power Plants" was utilized in this verification effort. Any discrepancies between the UFSAR and the design documents were investigated and documents were revised as appropriate by EDC E-50064-A which partially implements the corrective actions. Changes discussed are "documentation only" in nature and have no impact on the design basis of the plant or its operational configuration. Document changes have been evaluated for plant operability during the review process and found to not affect the physical plant. Items 1 through 6, 9, 10, 14, 15, 17, 18, 19, 22, 23, 26, 27, 28, 30 through 44 (excluding 33a), 46, 48, 50, 52a, and 53 through 56 are all considered as minor changes and do not require a safety assessment and safety evaluation. The remaining items 7, 8, 11, 12, 13, 16, 20, 21. 24, 25, 29, 33a, 45, 47, 49, 51, 52, and 57 through 62 are significant enough to be considered as technical in nature and necessary to provide consistency among the UFSAR, system descriptions, design criteria, supporting design calculations, drawings, etc. No design calculations are impacted, or revised as part of this review.

Design document changes which are being implemented are summarized as follows:

System Description Document (SDD) revisions: SDD changes consist of deleting reference to Unit 2 operation, correcting references, deletion of duplicate or non-contributory information, correcting typographical errors, clarification of design requirements, establishing consistency with design calculations and operating procedures, etc.

UFSAR revisions: UFSAR changes consist of deletion of reference to Unit 2 operation, deletion of duplicate information, omissions, correction of typographical errors, clarification of information by rewording or addition of text, establishing consistency with SDDs and operating procedures, deletion of excessive detail not required by Regulatory Guide 1.70, etc.

There are no failure modes associated with this change.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

These changes do not impact any accidents evaluated in the UFSAR. As described in section A of this safety evaluation, these changes do not affect the operation of any safety related equipment/systems and no credible failure modes are created or changed (some of the failure modes were re-addressed in this change and determined to be acceptable).

These changes will not increase the dose to the public analyzed in UFSAR chapter 15.1. No change will occur to the radiological consequences of accidents analyzed as a result of these changes. The proposed changes do not result in any changes to the plant design basis or the physical plant. The credible failure modes for the systems affected by these changes have been evaluated against the accidents identified in the FSAR. It is concluded that they do not introduce a failure pathway different from those identified and evaluated in the FSAR 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. 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 below.

Item 1, Section 9.1.1.2	Added a period to the last sentence of second paragraph.
Item 2. Section 9.1.3	Changed "...when one or both reactor vessels are open "to" when a reactor vessel is open ... " in the third sentence to reflect one unit operation.
Item 3, Section 9.1.3.1.1	Changed "...assemblies stored in the pool following a full core ..."to " ... assemblies stored in the pool and maintain acceptable pool temperatures following a full core ... " in the first sentence of the first paragraph. Also, revised the second sentence by referring to Table 9.1 -1 for temperatures associated with the various full core off-load scenarios rather than repeating the information in the text. The temperature of 129.3°F specified in the text for a full core discharge following a normal refueling is unchanged and appears in Table 9.1-1. These are considered as minor editorial changes.
Item 4, Section 9.1.3.1.1	Changed "The system design incorporates..." to "The SFPCCS incorporates ... " in the second paragraph for consistency with previous sections. This is considered a minor editorial change.
Item 5, Section 9.1.3.3.3	The temperature associated with the various discharge scenarios and number of Spent Fuel Pool (SFP) cooling trains operating are already presented in Table 91 -1. Therefore, the fourth sentence was revised to refer to Table 9.1-1 for each off-load scenario. This is considered as a minor editorial change since the temperatures have not changed.
Item 6, Section 9.1.3,3.3	Revised the last sentence in paragraph five to refer to Table 9.1-1 for the SFP heat-up rates rather than duplicate the information in the text of this section. This is considered as a minor editorial change since the SFP heat-up rates have not changed.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

Item 7, Section 9. 1 3.3.4	Deleted the last two sentences of the first paragraph. The Westinghouse system description WAT/WBT-292/4 indicates the refueling water purification pumps are sized to clean the water in the refueling cavity in one day with an effectiveness of 85%. In addition, Section 2.0, paragraph three indicates the purification filter will be used for spent fuel pit water cleanup by removal of the gate between the canal and the spent fuel pit. There is no discussion to indicate that the capacity of the system is "large," however, if cleanup times are on the order of a few days it may be reasonable to conclude that the system is large. It is reasonable to assume that as additional load is placed on the demineralizer, the resin change out frequency would be reduced. Since this level of detail is not required by Regulatory Guide 1.70 and is not discussed in the WBN SER, they were conservatively deleted. This revision is not associated with a physical change to the system, or change in operating procedure. Therefore, this change does not affect nuclear safety.
Item 8, Section 9.1.3.3.4	Revised the second paragraph to credit only the use of high purity water in preventing any significant crud buildup on the SFP walls since a temperature below which crud buildup would be reduced or eliminated could not be verified in any design input, or output documentation. Water purification is provided by the SFP mixed bed demineralizer which removes dissolved ionic impurities and the SFP filter removes particulates from the pool water as described in system description N3-78-4001. This revision does not result from any physical or operational changes in the system and will not affect nuclear safety
Item 9, Section 9.1.4.2	Deleted "...a fuel assembly, core component and..." from the list of Fuel Handling System (FHS) equipment in the second sentence of the first paragraph and replaced with "... the ..." The fuel assembly and core component are items manipulated by and not components of the FHS. This is considered a minor clarification of terminology.
Item 10, Section 9.1.4.2 (first paragraph)	Changed "...handling equipment..." to "...fuel handling equipment..." changed "and a FTS" to "and the FTS" in the second sentence. This is considered a minor editorial change.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

Item 11. Section 9.1.4.2 (second paragraph)	Deleted paragraphs two and three in their entirety and replaced with a new paragraph which accurately describes the process of moving new fuel assemblies from a shipping container into the two possible storage locations prior to being moved into the reactor core. This change brings the UFSAR description up to date with current Fuel Handling Instructions (FHIs) for fuel handling beyond the initial fuel load and is consistent with UFSAR Section 1.2.2.4 as revised by EDC E-50062-A and Section 3.1.3 of system description N3-79-4001 as revised by EDC E-50064-A.
Item 12, Section 9.1.4.2 (fourth paragraph)	Changed "...refueling canal wall for transferring..." to "...refueling wall and may be used for transferring..."to the second sentence. Also, added the following statement: "...the Rod Cluster Control Assembly (RCCA) change tool is used from the spent fuel pool bridge crane to transfer control elements from one assembly to another in the spent fuel pool." Since full core off-loads are now normally performed, the transfer of control elements from one fuel assembly to another is accomplished in the spent fuel pool and not in the reactor building where the rod cluster control changing fixture is located. FHIs used by Operations support the use of either the rod cluster control changing fixture, or the RCCA and is the basis for adding the words "...and may be used..." since the transfer of control elements within the reactor building can be performed if required. The secondary safety function of the fuel handling equipment is not challenged by this change, rather this change simply reflects the current fuel handling procedures. Therefore, this revision does not affect nuclear safety.
Item 13, Table 9.1-1	The number of fuel assemblies remaining in the SFP for the full core off-load following a normal refueling storage case was changed from "...1600 assemblies stored plus one additional 64 assembly discharge ..." to "...1600 assemblies stored plus one additional 80 assembly discharge ..." This change reflects the assumptions used in the analysis of record calculation WBNOSG4-239, Revision 1. Revision 0 of the analysis unfortunately considered only 64 assemblies instead of 80. However, the burnup for the reload batch during an unplanned discharge was conservatively assumed to be 48,000 MWD/MTU. The unplanned discharge case was rerun by changing the first batch size to 80 assemblies, correcting the burnup of the 80 reload batch (in the second full-core off-load) to 1,400 MWD/MTU. Revised Cases 2A' and 2B' were added as Appendix E in Revision 1. The conclusion

reached in the analysis is that the results are bounded by the original analysis results as documented in the analysis. This was reflected in Chapter 5 of the licensing report associated with WBN's amendment application to re-rack the WBN Pool. Since the original results were not impacted by the revision, the decay heat production, SFP water temperature, and average heat-up rate values are still correct. Therefore, this change does not affect nuclear safety.

Item 14, Section 9.1.4.2 (Item e)

Changed "...to the vertical position and is unloaded by the spent..." to "...to the vertical position by the lifting arm. The fuel assembly is lifted and moved by the spent..." This change more accurately describes the process of rotating and moving the spent fuel assemblies. This change is not associated with a physical change to the system, nor does it reflect a change in the way it is operated. This is considered a minor change.

Item 15, Section 9.1.4.2.2 (fifth paragraph)

Changed "... trolley and winch can be operated ..." to "... trolley and hoist can be operated..." to be consistent with terminology used in the previous paragraph and that used in system description N3-79-4001.

Item 16, Section 9.1.4.2.2 (sixth paragraph)

Deleted the paragraph in its entirety since portable cameras are used in lieu of a permanent television system described. Replace with the following: "Portable, underwater cameras are used, as required, during refueling operations and can permit viewing of all fuel assembly positions." The closed circuit television cameras used are of various types and designs tailored to the task being performed. They are portable rather than being permanently mounted as was previously described in this UFSAR section. The cameras are suspended underwater by cables as needed. Some of the cameras are telephoto and are equipped with pan and tilt features. Some are high radiation tolerant and ran be lowered to an assembly while others are not radiation tolerant. The closed-circuit television systems used at WBN are described in UFSAR Section 9.5.2.2. This change does not affect nuclear safety since the portable camera system used does not impair the fuel handling equipment's ability to perform it secondary safety function and permits viewing of all required refueling activities.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

Item 17, Section 9.1.4.3.1 (Industrial codes and standards discussion - Item 3)	Changed "MGI" to MG 1. (typographical error).
Item 18, Section 9.1.4.3.1 (Item 3)	Deleted "ICS" and "installation and manufacturing" from Item 3 under the heading of industrial codes and standards used in the design of the fuel handling equipment. ICS, (Industrial Control Systems) is not invoked in either Westinghouse equipment specifications E 677055, or E 677063. The listing of industrial codes and standards was never intended to identify codes and standards applicable to "Installation and manufacturing processes", only those applicable to the design. Therefore, the list was corrected to reflect this and to agree with section 2.2.15 of system description N3-79-4001, "Codes and Standards Requirements." This is considered a minor change. It is detail not required by Regulatory Guide 1.70, nor is it discussed in the WBN SER.
Item 19, Section 9.1.4.3.1 (Item 5)	Deleted reference to ANSI-N18.2 since this document is not invoked in either Westinghouse equipment specifications E 677055, or E 677063. Neither is this standard listed in Section 2.2.15 of system description N3-79-4001. Also, this level of detail is not required by Regulatory Guide 1.70 and is not discussed in the WBN SER.
Item 20, Section 9.1.4.3.1 (1. Electrical Interlocks)	Deleted statements from Items a through e which imply the fuel handling equipment (FHE) interlocks can withstand a single failure. Westinghouse, drawing 22407-50, shows some redundant interlocks, however, they eventually come to a single relay which could fail in either direction. Therefore, the single failure criteria as defined in design criteria, WB-DC-40-64, is not met. Appendix 8, Section B.2, "Single Failure Criteria" requires that safety related systems be designed such that a single failure of any active component (assuming passive components function properly), or a single passive failure of any passive component (assuming active components function properly) will not result in the loss of the capability of the system to perform its safety functions. This change is acceptable since only the fuel transfer tube with its associated blind flange and the fuel storage racks portions of the fuel handling area (FHA) are classified as primary safety related and therefore required to withstand the effects of an OBE or SSE and remain functional. The fuel handling equipment performs a secondary safety function and is designed such that a structural failure will not produce an unacceptable influence on the performance of Category I plant safety features having a primary safety function. This revision is not associated with a physical change to the FHE system, or its operation and is considered to not affect nuclear safety since the secondary safety function requirements are met.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

Item 21, Section 9.1.4.3.1 (5. Fuel Assembly Support System)	Deleted the last four sentences which provided detail concerning the wire rope maximum static load capacity, the capacity in relation to the hoist limit and in relation to the 5500 pound emergency pullout load applied to the handwheel. Replaced with the statement, "Each wire rope has a load rating 5 times the design load." Section 6.3.4 of Westinghouse Equipment Specification Number 677055 states that the rope load rating shall be sufficient to support five times the design load of Section 6.3.3 when reeved as specified in Section 6.3.1. This level of detail is not required by the Reg. Guide 1.70, revision 1, nor is it discussed in the SER or any of its supplements. This revision is not associated with any physical change to the system and the ability of the FHE to perform its intended secondary safety function is also not changed. Therefore, the change does not affect nuclear safety.
Item 22, Section 9.1.4.3.1 (second paragraph)	Deleted the statement "The working load of fuel assembly plus gripper is approximately 2500 pounds" since it is a repeat of information contained in earlier Section 9.1.4.3.1. This is a minor editorial change.
Item 23, Section 9.1.4.3.1 (third paragraph)	Deleted the statement "The gripper itself has four fingers gripping the fuel, any two of which will support the fuel assembly weight. " Section 6.6 of Westinghouse equipment specification 677055 contains details concerning how the gripper works; however, there is no documentation supporting the number of fingers and that two of the four are sufficient to support a fuel assembly. This is considered as unnecessary detail not required by Regulatory Guide 1.70 and not discussed in the WBN SER and was deleted for these reasons.
Item 24, Section 9.1.4.3.1 (Fuel transfer system)	Deleted reference to single failure and redundancy in Items 1 through 6. The description indicated that the interlocks associated with the transfer car permissive switch, lifting arm - transfer car position, transfer car - valve open, transfer car - lifting arm, lifting arm - refueling machine, and lifting arm - spent fuel pit bridge can withstand a single failure. Westinghouse Drawing 22407-50 shows some redundant electrical interlocks, however, they eventually come to a single relay which could fail in either direction. Furthermore, this equipment is not powered from a Class 1E source. Therefore, the single failure criteria as defined in WB-DC-40-64 is not met. Appendix B, Section 8.2, "Single Failure Criteria" requires that safety related systems be designed such that a single failure of any active component (assuming passive components function properly), or a single passive failure of any passive component (assuming active components function properly) will not result in the loss of the capability of the system to perform its safety functions. This change is acceptable since only the fuel transfer tube with

SA-SE Number: WBPLMN-98-109-0 **Implementation Date:** 05/18/1999

	its associated blind flange and the fuel storage rack portions of the fuel handling and storage system are classified as primary safety related, are required to mitigate the consequences of Design Basis Events (DBEs), and must meet single failure criteria specified in design criteria WB-DC-40-64. The fuel handling equipment performs a secondary safety function such that its structural failure will not produce an unacceptable influence on the performance of Category I plant safety features having a primary safety function. This revision is not associated with a physical change to the fuel handling equipment system, or its operation and is considered to not affect nuclear safety since the secondary safety function requirements are met.
Item 25. Section 9.1.4.3.1 (Spent Fuel Pit Bridge)	Item 4 in this section discusses the design load of the hoist. This is considered unnecessary detail beyond that required by Regulatory Guide 1.70 and was deleted. Furthermore, design loads for the hoist are not discussed in the WBN SER. Item 5 was renumbered as 4 due to this deletion. This revision is not associated with a physical change in either the fuel handling system, or its operation and is considered not to impact nuclear safety.
Item 26, Section 9.3.6,1 (second paragraph, item 1)	Changed the description of the number of full capacity auxiliary charging pumps to reflect one unit operation. "4 full-capacity" was changed to "2 full-capacity ... " and "(2 per unit)" was deleted.
Item 27, Section 9.3.6.1 (second paragraph)	Changed Item 2 "1 auxiliary makeup tank" to "1 auxiliary boration makeup tank" tank to agree with the descriptive name used in design output such as system description N3-84-4001 and flow diagram 1-47W809-7. This is considered a minor clarification change.
Item 28, Section 9.3.6.1 (third paragraph)	Changed "...the maximum leakage loss..." to "...the maximum postulated leakage loss ..." in the second sentence. Changed "Leakage loss is based..." to "Postulated total recoverable leakage is based ..." and deleted "... the total recoverable leakage of ..." in the third sentence for clarification. The revised statement now agrees with Westinghouse letter WAT-13-144 dated February 7, 1972. This is considered a minor clarification and is not associated with any physical change to the system.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

Item 29, Section 9.3.6.1	Changed the leakage loss value from "(580 gpd)" to "(≈576 gpd)" (third paragraph) in the third sentence based on the RCS seal leakage at RCS pressure less than 350 psig. Section 2.2.13 of system description N3-84-4001 references WAT-D-9078, "Natural Circulation Evaluation at Reduced RCS Pressure" and specifies the RCP seal leakage as 0.1 gpm per RCP. Therefore, 0.1 gpm/RCP x 1440 min/day x 4 RCPs (for one unit operation) = 576 gpd. This is a slightly more accurate estimate of the total leakage. No physical changes to the system are associated with this revised leakage flow rate and the available capacity of the ABMT is much greater than required to support one unit operation. Therefore, this change does not result in any reduction in nuclear safety.
Item 30, Section 9.3.6.1 (third paragraph)	Changed "...coolant pump of both units plus the total recoverable leakage of 225 gpd ..." to "coolant pump plus 225 gpd..." to reflect one unit operation and as part of the change discussed in Item 28.
Item 31, Section 9.3.6.1 (third paragraph)	Changed "...generators will provide adequate..." to "...generators provides adequate ..." in the fourth sentence. This is considered as a minor editorial change.
Item 32, Section 9.3.6.1 (third paragraph)	The last sentence was revised by changing "...will be considerably less than during..." with " ... will be insignificant since the operating pressure during flood mode is considerably less than during..." This clarification agrees with the fourth paragraph of Section 2.2.13 of system description N3-84-4001. This is a minor change not associated with any physical changes to the system and provides consistency with design output documents.
Item 33, Section 9.3.6.1 (fourth paragraph)	Changed "The auxiliary makeup tank..." to "The auxiliary boration makeup tank (ABMT)..." to agree with the descriptive name used in system description N3-84-4001 and other design output such as flow diagram 1-47W809-7, Revision 14. Also, changed "each" to "one" to reflect one unit operation. These are considered as minor changes which provide consistency with design output documents and are not associated with a physical change to the system.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

Item 33a, Section 9.3.6.1 (fourth paragraph)	Changed "... (801 gallons)..." to "... (≈ 400 gallons)...". The original value of 801 gallons was based on the quantity required to support two unit operation and is documented in calculation EPM-WUC-091889. Since only unit one is operating, approximately one half of the original required volume is required. This change is not associated with a physical change to the system or size of the ABMT. Rather, it is as a result of one unit operation. Since more makeup volume is available, there is no decrease in nuclear safety.
Item 34, Section 9.3.6.1 (fifth paragraph)	Changed "The filters and demineralizers are provided..." to "The demineralizer is provided..." and added "and the filters prevent the demineralizer resins from leaving the FMBMS." to the first sentence. This change more accurately reflects the function provided by the Flood Mode Boration Makeup System (FMBMS) filters and demineralizers as described in Section 3.2.3 of system description N3-84-4001. This change in the description of the functions provided by these components is not associated with any physical changes in the system and are simple clarifications to the existing text. Therefore, this is considered as a minor change.
Item 35, Section 9.3.6.1 (fifth paragraph)	Since the flow rate values of 10 gpm for the filters and 27 gpm for the demineralizer are maximum design capacity values, changed "...a flow..." to "... a maximum flow ..." in the second sentence (two places). This more accurately reflects the description in Section 3.2.3 of SDD N3-84-4001 and is not associated with any physical changes to the system. This is considered as a minor editorial change.
Item 36, Section 9.3.6.1 (fifth paragraph)	Changed "All auxiliary charging..." to "Auxiliary charging..." and changed ".. on the 757.0 elevation of the..." to "...on Elevation 757.0 of the..." in the third sentence. These are considered as minor and editorial in nature.
Item 37, Section 9.3.6.2 (first paragraph)	Changed "...auxiliary makeup..." to "auxiliary boration makeup tank..." in the second and third sentences to make the statement more precise. Also, changed "tanks" plural to "tank" singular since only one demineralized water tank exists at WBN. These changes are considered as minor clarifications and do not reflect any physical changes in the plant design.
Item 38, Section 9.3.6.2 (first paragraph)	Changed "...(1) accumulator tanks..." to "cold leg accumulator tanks..." to more accurately describe the accumulator tanks in item "(1)" of the last sentence. This is considered to be a minor clarification and is not associated with any physical changes to the system.

SA-SE Number: WBPLMN-98-109-0 **Implementation Date:** 05/18/1999

Item 39, Section 9.3.6.2 (second paragraph)	Changed "...source is connected..." to "...requires manual addition,..." in the last sentence to more accurately describe how the backup supply of water is added if the preferred sources are unavailable and is consistent with Section 4.4 of system description N3-84-4001. This is considered to be a minor clarification and is not associated with any physical change to the system.
Item 40, Section 9.3.6.2 (third paragraph)	Deleted the second sentence since it partially duplicates information presented in the next sentence. Changed "The boric acid,..." to "Boric acid .... " in the next sentence and changed "...auxiliary makeup tank..." to "...auxiliary boration makeup tank..." This is considered a minor editorial change.
Item 41, Section 9.3.6.2 (fourth paragraph)	Deleted the first sentence in its entirety and replaced with a description that is consistent with Section 2.2.10 of system description N3-84-4001. This change is not associated with any changes in the plant design, rather it more accurately states that water quality analysis can be performed. Since water quality analysis can include much more than simply determining boron concentrations, that portion of the statement was deleted.
Item 42, Section 9.3.6.2 (fifth paragraph)	Changed "...auxiliary makeup tank..." to "...auxiliary boration makeup tank..." Also, replaced "...as demanded by pressurizer level..." with "...as required to maintain pressurizer level ... " to more accurately describe why the makeup water is pumped from the auxiliary makeup tank to the primary system. No change in system operation is associated with this revised wording. These are considered as minor and editorial in nature.
Item 43, Section 9.3.6.2 (fifth paragraph)	Revised second sentence to reflect one unit operation by deleting "per plant" and "per unit". Also, changed "...required makeup;..." to "required one unit makeup;... " and changed "...and four ... " to "... and two ..."
Item 44, Section 9.3.6.2 (sixth paragraph)	Changed "Spool pieces are used..." to "A spool piece is used..." in the first sentence to reflect one unit operation. Changed "...charging liners." to "charging line." In the same sentence to correct the typographical error and to also make it one unit specific. The second sentence was also revised to reflect one unit operation by changing "These spool pieces are..." to "This spool piece is...". These changes are considered minor and editorial in nature.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

Item 45, Section 9.3.6.3 (first paragraph)	Changed "The auxiliary..." to "See Table 3.2-2a for classification of..." Rather than simply stating that the auxiliary charging components are commercial grade, the first sentence in this section was revised to refer to FSAR Table 3.2-2a for the classification of components. Table 3.3-2a clearly defines which portions of the auxiliary charging system are TVA class B, C, or H. Type H is typically commercial grade. As described in Table 3.2-2a, those portions of the system necessary for containment isolation and piping essential for makeup and boration in the event of a flood above plant grade are TVA class B and C respectively and are not commercial grade. This revised wording does not change the classification of any component within the auxiliary charging system. Rather, it establishes consistency between this section and Table 3.2-1a. Consequently, this change does not affect nuclear safety.
Item 46, Section 9.3.6.3 (second and third paragraphs)	Changed "...jeopardize the operation..." to "...jeopardize system operation..." in the first sentence as a clarification. Changed "All components... " to "The components..." in second sentence. Changed reference to UFSAR Section 2.4.14.9 to 2.4.14.10. Section 2.4.14.9 which was referred to previously addresses the basis for the flood protection plan in rainfall floods and did not address seismic events which is the subject of Section 9.3.6.3, paragraph three. These changes are considered minor clarifications.
Item 48, Section 9.3.6.4	Changed "All components of..." to "Components of..." in the first sentence. Revised second sentence to clarify that the system has been preop tested by changing "... system will be tested ..." to "... system was tested..." Also clarified that the system is tested periodically by adding "and is tested periodically to ensure that degradation has not occurred" to the second sentence. This is considered a minor editorial change since the intent of the original statements is not changed and additional information is added concerning periodic testing which was addressed previously. This level of detail is not required by Regulatory Guide 1.70. Furthermore, the auxiliary charging system is not discussed in the WBN SER.
Item 49, Section 9.3.6.5 (second paragraph)	Changed "...has both "Hi" and..." to "has a..." in the first sentence and changed "The "Hi" level..." to "The "Hi-Hi" level..." in the second sentence to clearly describe that only a "Hi-Hi" alarm exists. No separate "Hi" alarm exists in the current design. This change correctly describes the operation of the reactor coolant drain tank and pumps relative to the high level alarm. This section of the FSAR currently states there are two high level alarms (Hi and Hi-Hi). As indicated in UFSAR Section 5.2.7.1 and logic diagram

1-47W611-77-1, there is one low level (Lo-Lo) and one high level (Hi-Hi) alarm for the RCDT. The RCDT pumps are started by the '1-161-11' level alarm. This change is not a physical change in the RCDT equipment and does not affect operation of the system. The change is to make Section 9.3.6.4 of the UFSAR consistent with Section 5.2.7.1 of the UFSAR and with the design basis. As indicated in UFSAR Table 3.2-2, the RCDT and associated pumps do not perform a primary safety function (i.e., equipment is TVA class G) and thus do not serve to prevent or mitigate the consequences of a design basis event. Since the RCDT equipment does not perform a primary safety function and since no physical or operational changes are made to the equipment, this revision of the UFSAR has no affect on nuclear safety.

Item 47, Section 9.3.6.3 (third paragraph)

This revision is to clarify the seismic requirements of the auxiliary charging system and make this section of the UFSAR consistent with UFSAR Table 3.2-2a. Section 9.3.6.3 currently includes a general statement that essential features of the auxiliary charging system are qualified to limited seismic requirements. This general statement is deleted and a statement is included that the essential auxiliary charging piping is seismic category 1, and the balance of the system is seismic category I(L) in accordance with UFSAR Table 3.2-2a. Seismic events located sufficiently near the plant to cause damage to equipment have been shown by the flood analysis not to result in a flood above plant grade at WBN. This is supported by Civil calculations WC-1-552, -563, and -565 and is consistent with Section 2.2.1 of System Description N3-84-4001 as modified by EDC 50064-A (Also see Item number 58). The seismic qualifications or seismic requirements of the auxiliary charging system equipment is not changed by this revision. Rather, this change merely updates Section 9.3.6.3 to reflect the requirements in UFSAR Table 3.2-2a. Consequently, this change does not affect nuclear safety.

Item 50, Section 9.3.6.5 (second paragraph)

Changed "...auxiliary makeup tank (AMT)..." to "...auxiliary boration makeup tank (ABMT)..." in the third sentence. Changed "Levels in the AMT can..." to "Levels in the ABMT can ..." in the fourth sentence. This is considered a minor editorial change.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

Item 51, Section 9.3.6.5 (second paragraph)	Changed "...tank has a 1/2 day supply..." to "...tank can provide a one day supply..." based on one unit operation and the capacity of the ABMT. Previously, the tank capacity was adequate to supply for 12 hours based on two unit operation as documented in calculation EPM-WUC-091889. This calculation determined that the required makeup volume of the ABMT to provide an initial 12 hour supply is 801 gallons for two unit operation. Therefore, one half of that value (400.5 gallons) is required to support single unit operation. As stated in the calculation, the capacity of the tank is 868.34 gallons. Therefore, sufficient capacity exists to provide 24 hours of makeup in support of one unit operation. This change does not make a physical change to the plant, but describes the additional makeup capacity available due to one unit operation. Nuclear safety is not reduced since the makeup volume available is increased.
Item 52, Section 9.3.8.5 (second paragraph)	Changed "...redundant pressure loops..." to "...redundant pressure and pressurizer level loops..." This revision adds the statement that redundant RCS pressurizes level instrumentation serves as indications of low pressure necessary for activation of the auxiliary charging pumps. The section currently states that redundant RCS pressure instrumentation is used for activation of the auxiliary charging pumps. This revision is verified by Table 4.1-4 in design criteria WB-DC-40-29 which lists redundant pressurizer level instrumentation as instruments required for flood operation. This revision does not make physical changes to equipment, but describes additional equipment which can be used under the design basis document for flood mods operations. Consequently, this revision does not affect nuclear safety.
Item 52a, Figure 9.3-18	Revised figure to agree with current revision level of associated CCD 1-47W809-7, revision 14. This change reflects the unit 1/unit 2 interface points.
Item 53, Section 9.3.8 (Heat Tracing)	Deleted the current description and replaced with the following: "Electric heat tracing is used to supply heat to some of the insulated mechanical piping systems to prevent freezing of the fluid in the pipe to provide process temperature control to maintain the media within its specified temperature range and on some instrument sense lines." There is no change to the location, function, operation, or procedure of any heat trace system equipment. The heat trace system is not a safety grade system nor is it required to mitigate any design basis accident. It is not a system that requires discussion by Regulatory Guide 1.70, or the Standard Review Plan (NUREG-0800) and is not mentioned in the WBN SER or supplements.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

The NRC has stated in SECY-98-087, "The most recent staff position on the content of the FSAR (and the updated FSAR) is Regulatory Guide 1.70, Revision 3. Licensees are not required to comply with the guidance in the regulatory guide; however, the guide may be used as a reference for an appropriate amount of information to be provided on specific issues."

Therefore, the change deletes information that is not necessary and provides a level of detail that is sufficient for a system that is not required to be documented in the FSAR. This change is considered to be a minor editorial change that does not require additional discussion or evaluation.

Item 54, N3-78-4001, Sections 1.0, 2.1.2, 2.2.2, 2.2.13, 2.2.15, 3.2.8, Page 16, Sections 3.5, 5. 1. 5.3, 7.1.5, Figure 1.0-2, Table 2.2-2, Table 3.2-1.

Numerous changes were made to reflect one unit operation and to correct typographical errors throughout the document. Other Minor changes include the following: Added reference to tables rather than other sections for design parameters. Corrected refueling water purification filter size to agree with Table 3.2-1. Deleted page 18 since all sections are duplicated on the next page. Added words to complete the last sentence of section 3.5 based on the duplicate page 16 deleted. Changed page 17, page number to 16. Changed page 17A page number to 17. Updated calculation reference revision level. Added skimmer pump and water purification pump motor horsepower values to Table 3.2-1 to be consistent with Sections 3.2.2 and 3.2.3. Revised Figure 1.0-2 to reflect one unit operation.

Item 55, N3-79-4001, List of Abbreviations, Sections 1.0, 2:1.1, 2.1.2, 2.2, 2.2.1, 2.2.13, 2.2.15, 2.2.16, 5.0. 10.0

Numerous changes were made to reflect one unit operation and correct typographical errors throughout the document. Other minor changes include the following: Added the fuel transfer tube to the list of items comprising the FHSS in section 1.0. Changed "New Fuel Storage Racks" to "NFSRs". Changed "Spent Fuel Storage Racks" to "SFSRs". Changed "Spent Fuel Pool" to "SFP" Changed "Fuel Handling and Storage System" to "FHSS". Changed "and blind flanges perform" to " and associated blind flanges perform" to reflect one unit operation. Changed "MG-1. 1970 edition" to "MG 1-1970 edition." Changed "absorbers" to "adsorbers". Changed "1/2 SSE" to "OBE." 1/2 SSE was the original design requirement prior to development of the OBE accelerations. WBN's racks are analyzed to both the full SSE and OBE accelerations. Deleted source notes listed on page 53 since they are duplicated on page 52. Added "DWS, Demineralized Water System" and "LRPS, Liquid Radwaste Processing System" to list of abbreviations and acronyms.

SA-SE Number: WBPLMN-98-109-0 **Implementation Date:** 05/18/1999

Item 56, N3-84-4001, List of Abbreviations and Acronyms, Table of Contents, Sections 1.0,2.1.2, 2.2.10, 3.1, 3.2.1, 3.2.2, 3.2.4, 3.2.5, 3.3.3, 4.5, Table 8.1, Table 8.2, Figure 9.1	Numerous changes were made to reflect one unit operation (w.r.t. the number of spool pieces and pumps used) and to correct typographical errors throughout the document. Other minor changes include the following: Added "DWS, Demineralized Water System" and "LRPS, Liquid Radwaste Processing System" to list of abbreviations and acronyms. Page number corrections in table of contents. Added a statement to section 2.2.10 which refers to Table 8.1 for Flood Mode Boration Makeup System (FMBMS) interface with the Unit 2 FMBMS Auxiliary Charging Pumps (ACPs) and Unit 2 LRPS. Replaced the statement addressing two Auxiliary Charging Booster Pumps (ACBPs) as being needed to provide the required system makeup for two units with a statement Deleted the 480V auxiliary building vent boards listed in Section 3.2.4 associated with Unit 2 ACP motors. Deleted "gate" from the valve descriptions in Section 3.2.5 and replaced with "isolation" to reflect the type of valves used. Added isolation valve numbers "1-84-501 and -502" to more accurately describe the location where the discharge lines come together to form a one-inch line which is connected to the CVCS charging line by a spool piece. Changed valve "2-84-530" to "the blind flange upstream of valve 2-84-528" in the description of segment 1 (Section 3.2.5). This change more accurately describes this the limits of this segment since the blind flange forms the Unit 1, Unit 2 interface point. Added a statement to Section 4.5 which describes the blind flange that isolates the Unit 2 LRPS from the Unit 1 FMBMS. The original statement only mentioned the locked closed valves and referred to the system flow diagram for interface points. Added the Unit 2 FMBMS ACPs and LRPS to the list of interfacing systems of Table 8.1 and specified the interfacing requirements for these Unit 2 systems. Added "CVCS" and "LRPS" to Table 8.2, Items 2 and 7 respectively which more accurately describes the system connected to the FMBMS and reduced the number of spool pieces to one in each case to reflect one unit operation. Revised Figure 9.1 to depict the Unit 1 and Unit 2 interface points and to show all valves as globe rather than gate type. This figure now agrees with the system flow CCD 1-47W809-1.

SA-SE Number: WBPLMN-98-109-0 **Implementation Date:** 05/18/1999

Item 57, N3-79-4001, Section 2.2.1	Revised Section 2.2.1 to add a statement concerning the primary safety function performed by the fuel storage racks. The present wording implies that only the fuel transfer tube and associated blind flange perform a primary safety function. The fuel storage racks also perform a primary safety function of preventing accidental damage to the fuel assemblies and maintaining new and spent fuel assemblies in a subcritical array during all storage conditions. This function assures the capability to prevent or mitigate the consequences of DBEs (including OBE, SSE, and fuel handling accidents) which could result in potential offsite exposures to a significant fraction of the limits given in 10CFR100. The storage racks meet this requirement since they are seismic Category I. They are also designed to withstand the impact of a dropped spent fuel assembly from the maximum lift height of the spent fuel pit bridge hoist. Such a postulated fuel handling accident will not result in criticality. This revision does not represent a change in the original design requirements for the storage racks, or physical change to this portion of the FHSS. It is simply a clarification of the requirements within Section 2.2.1 of the system description. Therefore. This change does not affect nuclear safety.
Item 58, N3-79-4001, Sections 2.2.5 and 3.4.2	Revised the minimum water shield above the active fuel region of a spent fuel assembly as it is moved from the reactor vessel to the spent fuel pool from 10 ft to 9.9 ft. Acceptability of this change is based on the results of calculation WBNTSR-059 which determined a maximum does rate at the refueling bridge of less than 2.5 mrem/hr at a water depth of 9.90 ft above the active fuel. This revision does not affect nuclear safety and does not result in the operator located on the refueling bridge receiving a dose rate greater than the acceptance criteria of 2.5 mrem/hr.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

Item 59, N3-79-4001, Section 3.1.3	Replaced the fifth paragraph with more a accurate description concerning the process of moving new fuel from a shipping container into the two possible storage locations prior to being moved into the reactor core. The sentence concerning the movement of assemblies for initial fuel loads is no longer applicable to WBN. Rather than only storing new fuel in the new fuel storage racks, the Fuel Handling Instruction (FHI) series of procedures allow for storage in either the new vault (for dry storage), or in the spent fuel storage racks as a staging area for the next refueling. If storage in the spent fuel pool is desired, assemblies are placed into the new fuel elevator and lowered into the transfer canal where normal spent fuel handling equipment is used to complete the movement into its storage location. Current FHIs allow for transfer of new fuel directly from the shipping container or from the new fuel vault into the reactor core or spent fuel pool via the new fuel elevator and normal spent fuel handling equipment. This change simply brings the system description up to date with current FHIs. The revision is not associated with a physical change to the fuel handling and storage system. Therefore, this change does not affect nuclear safety.
Item 60, N3-79-4001, Section 3.2.1	Revised the sixth paragraph to describe the portable underwater camera system currently used during refueling operations. A permanent television system mounted to the refueling machine is not used at WBN. The closed circuit television cameras used are of various types and designs tailored to the task being performed. They are portable rather than being permanently mounted to the refueling machine as was previously described. The cameras are suspended underwater by cables as needed. Some of the cameras are telephoto and are equipped with pan and tilt features. Some are high radiation tolerant and can be lowered to an assembly while others are not radiant tolerant. This change does not affect nuclear safety since the portable camera system used does not impair the fuel handling equipment's ability to perform its secondary safety function and permits viewing of all required refueling activities.

SA-SE Number: WBPLMN-98-109-0 ***Implementation Date:*** 05/18/1999

Item 61, N3-84-4001, Section 2.2.1	System description N3-84-4001, Section 2.2.1 and design criteria WB-DC-40-64, Section 4.4.5.5 currently states that essential equipment must tolerate the resultant ground acceleration at the site which causes a seismic induced flood without loss of function. These documents are revised to delete the above statement and to state that any seismic event located sufficiently near the plant to cause damage to equipment will not result in an upstream dam failure. This revision is supported by civil calculations WC-1-551, -563, and - 565. This change does not affect nuclear safety since the equipment necessary to mitigate the consequences of a seismically induced flood will not be affected by the seismic event and can therefore perform its intended safety functions subsequent to the event.
Item 62, N3-84-4001, Sections 3.1, 3.2.1, and 3.6	Changed the length of time associated with a filled ABMT from 12 hours to 24 hours based on one unit operation. Calculation EPM-WUC-091 889, determined that the required makeup volume of the ABMT to provide an initial 12 hour supply is 801 gallons for two unit operation. Therefore, one half of that value (400.5 gallons) is required to support single unit operation. As at stated in the calculation, the capacity of the tank is 868.34 gallons. Therefore, sufficient capacity exists to provide 24 hour's of makeup in support of one unit operation. As discussed in section 3.2.1, the submergence depth required to prevent vortex formation and air entrainment is 0.051 ft which resulted in a margin of 60 gallons above the required initial 12 hour makeup reserve volume for two unit operation. Since the system is only required to support one unit, a margin of approximately 868.34 - 400.5 =≈460 gallons now exists above the required initial 12 hour makeup reserve volume. Due to the excess capacity available in the ABMT when filled to capacity, a clarifying note was added to the second sentence of section 3.6 stating that 24 hours of makeup is available.

SA-SE Number: WBPLMN-98-113-0

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Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E50060-A	FSAR Review and Verification of
FSAR	FSAR Change Package, 1577	Section 6.3

Description and Safety Assessments:

FSAR Section 6.3 and the associated tables and system description N3-63-4001 are revised to incorporate changes that resulted from a review of the FSAR. All items except those described below are considered editorial/clarification changes that do not change the intent of the text. As such, they meet the definition of minor SAR changes/corrections and do not require a Safety Assessment, Screening Review, or Safety Evaluation. The items that require additional review are as follows:

1. Section 6.3.2.2, page 6.3-6, last sentence: The sentence currently reads: “Globe valves of the “T” and “Y” styles are full-ported with outside screw and yoke construction.” Revise the sentence to read: “Globe valves of the “T” and “Y” styles are full-ported.” This is a clarification in that not all of these valves are provided outside screw and yoke construction

2. Section 6.3.2.2, page 6.3-7, second paragraph: Delete the last sentence. There is no carbon steel manual valves in the system. This change deletes information that is not necessary and does not change the intent of the text.

3. Section 6.3.2.2, page 6.3-7, last paragraph: Revise the first sentence to read: “The check valves are tested for leakage as soon when the RCS is being pressurized during the normal plant heatup operation.” This removes unnecessary detail, but does not change the pertinent information that is contained later in the paragraph; therefore, it is considered a minor editorial change.

4. Section 6.3.2.2, page 6.3-8, second paragraph under Relief Valves: Delete the sentences that reads: “The valve stem and spring adjustment assembly are isolated from the system fluids by a bellows seal between the valve disc and spindle. The closed bonnet provides an additional barrier for enclosure of the relief valves.” This is determined to be information that is of too much detail for inclusion in the FSAR and not all relief valves have back pressure compensating bellows. The text intent is not change, therefore, this is considered a minor editorial change.

5. Section 6.3.2.2, page 6.3-12: Delete the paragraph that begins: “Remotely operated valves for the injection mode…” This information is a duplicate of the information presented in 6.3.2.16; therefore, this is considered to be an editorial correction to minimize duplication of information.

6. Section 6.3.2.2, page 6.3-14: Delete the third paragraph. This is superfluous information that does not belong in this section and its deletion does not change the intent of the text. Therefore, this is considered to be a, minor editorial correction and does not require further evaluation.

7. Section 6.3.2.2, page 6.3-15: Delete the last sentence in the middle paragraph that reads “The effective flow area of the 1/4-inch screen attached to the outer trash rack is 25.8 times the combined total pipe flow area; the 1/4-inch sump suction pit screen area is 6.6 times the pipe flow area.” The first part of the paragraph provides an adequate level of information to satisfy Regulatory Guide 1.70 and the SRP guidelines. The SER does not contain or reference this level of detail. Therefore, this deletion of unnecessary detail is considered to be a minor editorial change

SA-SE Number: WBPLMN-98-113-0 ***Implementation Date:*** 05/03/1999

8. Section 6.3.2.2, page 6.2-17, Under External Recirculation Loop, the third and fourth paragraphs: Revise the first sentence and delete the rest of the two paragraphs to read "An analysis has been performed to evaluate the radiological effects of recirculation loop leakage as discussed in Section 15.5.3." The information being deleted is a duplication of information presented in 15.5.3; therefore, it is considered an editorial change and requires no further discussion.

9. Section 6.3.2.11.2, page 6.3-21, last paragraph: Revise the last sentence to read "This procedure requires that low head safety injection flow to the core be terminated under single train operating conditions prior to initiating RHR spray flow." This is a clarification that does not change the operation of the system and is therefore considered as meeting the definition of a minor editorial change that does not require further discussion.

10. Section 6.3.2.11.3, page 6.3-23, second full paragraph: The second half of the paragraph is deleted because the ECCS leakage is not routed to the tritiated equipment drain sump or tank. This change is considered a minor editorial change that does not change the intent of the text because the destination of ECCS pump compartment leakage has been adequately addressed in the first portion of this paragraph. The discussion on the tritiated equipment drain sump and pumps is not pertinent to the discussion on ECCS leakage. A discussion of these components is currently provided in FSAR section 11.2.3.1. This item is therefore considered a minor editorial change that minimizes duplication of information and does not require further evaluation.

11. Section 6.3.2.11.3, page 6.3-24, second paragraph is deleted because the detail contained is not required by Reg Guide 1.70 nor by the Standard Review Plan. The information required to describe system operation is presented in the preceding paragraphs; therefore, this deletion is considered a minor editorial correction to delete unnecessary detail and does not require further discussion.

12. Section 6.3.2.19, page 6.3-26: Revise the second sentence of the paragraph "Coatings specified for use on the ECCS components (mainly the cold leg accumulators) are listed in Section 6.1.4." This change is considered a minor editorial change that minimizes duplication of information and does not require additional review.

13. Section 6.3.3.1, page 6.3-26: Revise first paragraph to read "The analyses reflected in Section 15.4 were performed to ensure that the limits on core behavior following various pipe ruptures, etc are met by the ECCS operating with minimum design equipment."

    Revise the first sentence of the last paragraph to read: "The performance characteristics utilized in the accident analyses includes a 3-5 percent decrease in the design head for margin."

    This change deletes duplication of information found in Section 15.4 and is to make the FSAR consistent with WAT-D-10107 and does not change the intent of the text since the maximum decrease in design head for margin does not change. Therefore, it is considered to be a minor editorial change that does not require further evaluation.

SA-SE Number: WBPLMN-98-113-0 ***Implementation Date:*** 05/03/1999

14. Section 6.3.3.2, page 6.3-27: Revise the first sentence to read "The large pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe ruptures, as indicated in Section 15.4.1, up to the double-ended rupture of the largest pipe in the RCS." This correction makes this section agree with Section 15.4 of the FSAR and the system descriptions; therefore, this is considered to be a minor editorial correction to fix a typographical error that does not require further evaluation.

15. Section 6.3.3.3, page 6.3-27, first paragraph: Revise the sentence to read "The small pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe ruptures from 3/8-inch up to and including ruptures as defined in Section 15.3." This makes this section consistent with the discussion in Section 15.3 and the system descriptions; therefore, this is considered a minor editorial change and does not require further evaluation.

16. Section 6.3.3.3, page 6.3-27, second paragraph under Main Steam System Single Active Failure: Delete the first sentence of the paragraph. This information is presented in Chapter 15 (where it is more appropriate) and is an unnecessary duplication in Chapter 6.3; therefore, this is considered a minor editorial change to minimize duplication of information and will not be discussed further.

17. Section 6.3.3,12, page 6.3-31, section titled, Limiting Conditions for Maintenance During Operations: Delete all information and replace with the following: "See the Technical Specification 3.0 for the details concerning the limiting conditions for maintenance during operations." This considered an editorial change that eliminates duplication of information contained in another more appropriate licensing document and therefore does not require any additional discussion.

18. Section 6.3.3.15, page 6.3-32: Delete second paragraph. This is a duplication of the information in the paragraph immediately above it; therefore, this is considered as a minor editorial change to minimize duplicate information and does not need to be discussed further.

19. Section 6.3.3.16, page 6.3-32, first paragraph: Delete the first sentence. This change is considered to be a minor editorial change that does not change the intent of the text. As already discussed in the text that follows the deleted sentence and also in the discussion provided in Section 6.2.2.2, initiation of RHR spray is dependent on the time after LOCA initiation and not the number of containment spray trains available. Therefore, the change is performed to establish consistency with the existing FSAR text. The change also achieves agreement with FSAR Section 6.2.2.2 with the deletion of the sentence. Therefore, this change is considered to be a minor editorial change that does not require further evaluation.

20. Section 6.3.3.16, page 6.3-32: first paragraph: Revise the fourth sentence to change 105 lbm/sec to 93 lbm/sec. In addition, the first full sentence on page 6.3-33 states that the coolant entering the RCS piping is stated to be roughly 1.5 times (used to be 2 times) the decay head mass boil-off. The revised value of 93 lbm/sec injected coolant flow is based on the latest results of ECCS performance analyses as reflected in design criteria WB-DC-40-70. The 93 lbm/sec is still 1.5 times greater than the conservative calculated value of 61.5 lbm/sec decay heat mass boil-off. Therefore decay heat removal capability is adequate and does not impact radioactive releases. This change reflects actual values that the system delivers.

SA-SE Number: WBPLMN-98-113-0 ***Implementation Date:*** 05/03/1999

21. Section 6.3.4.3, page 6.3-34, first paragraph: Delete the fourth sentence. This is a level of information that is too detailed and not required for the FSAR and is documented in the test program procedure. This is therefore considered a minor editorial change and will not be addressed further.

22. Section 6.3.5, page 6.3-34: Delete the last sentence. This sentence does not present any meaningful information and each alarm discussed in the succeeding section is identified as to its location. Therefore this deletion is considered a minor editorial change and does not require additional discussion.

23. Section 6.3.3.5, page 6.3-37: Delete second paragraph under Accumulator Isolation. This is a duplication of information already presented in 6.3.2.15 and is being deleted to minimize duplication of information. This change is considered a minor editorial change to eliminate duplication of information and therefore does not require further discussion.

24. Section 6.3.5.5, page 6 1-37, Refueling Water Storage Tank Isolation Valve: Delete the description and replace with the following description: "The RWST isolation valve is provided with red (open) and green (closed) position indication lights located on the main control room handswitch. These lights are powered by valve control power and actuated by valve motor operator limit switch." This is not a change in the information, but only an aid to the reader so he does not have to look back up in the section to see what indicators are provided. Therefore, this change is considered to be a minor editorial change that does not require additional evaluation.

25. Table 6.3-2, Sheet 2: Delete component identified as "Motor-Operated Valves Containing Non-Radioactive, Boron-Free Fluids: Body, Bonnet, and Flange" made of "Carbon Steel" and "Stems" made of "Corrosion Resistant Steel." Delete "Carbon Steel Bodies" and material "Carbon Steel." These types of components do not exist in the system. This is an editorial change to delete information about components that are not in the system. Since this change does not change the system, it is being considered a minor editorial change to delete superfluous information and does not require further discussion.

26. Table 6.3-3, Sheet 3: Revise the realignment of the ECCS from cold leg mode to hot leg recirculation mode. The change to the sequence when realigning the ECCS from cold leg mode does not impact safety injection core cooling, radioactive release pathways and does not change system design parameters. This change actually reduces the time the safety injection pumps are down and thereby reduces the duration of in interrupted flow to RCS.

27. Table 6.3-4, Sheet 1: Change Nominal Cold Leg Accumulator Water Volume, ft^3 1040 to Minimum Cold Leg Accumulator Water Volume, ft^3 1019.6. Nominal value is meaningless in the context of the table. The Minimum value is the value used in the WAT-D-9213 letter and the system description; therefore, this is considered to be a minor editorial change to use the most acceptable value and does not require additional discussion.

SA-SE Number: WBPLMN-98-113-0 ***Implementation Date:*** 05/03/1999

28. Table 6.3-8, Sheet 39, Items 61 and 62 under column "Remarks:" Add the following after Normally open valve: "and breaker locked open to prevent spurious operation". This provides additional clarifying information but does not change the system configuration or description. Therefore, this is considered a minor editorial change and does not require additional discussion.

29. Table 6.3-8, Sheet 40, Item 63 under column "Remarks:" See number 28 above.

30. Table 6.3-8, Sheet 41, Item 64 under column "Remarks:" See number 29 above.

31. Table 6.3-8, Sheet 41, Item 65 under column "Remarks:" Add the following information: "Compensating provision/action will occur/can be performed for Train A valves required to change position following a LOCA to ensure the required function is not disabled due to Train A power failure." This provides additional clarifying information but does not change the system configuration or description. Therefore, this is considered a minor editorial change and does not require additional discussion.

32. Table 6.3-8, Sheet 42, Item 66 under column "Remarks:" See number 31above.

33. Table 6.3-8, Sheet 44: Add Item 71, Relief Valves to the FMEA table. The addition of the ECCS relief valve is not a change in the system design or function. This addition does not impact the margin of safety, the reliability or the radioactive release paths.

34. System Description N3-63-4001, paragraph 3.3.2.2.15: The FSAR and the schematics were both correct and this change to the system description brings it into agreement with the FSAR and schematics. This change is therefore considered a minor editorial change since it only provides additional clarification in the system description that is already provided in the FSAR. Since this is a minor editorial clarification, it does not require additional discussion.

The changes to the FSAR above do not impact any design basis accident parameters or evaluations as discussed in Chapter 15 of the FSAR. Neither do they create a new, or change any existing, credible failure mode of any components. Therefore, these changes are acceptable from a nuclear safety standpoint.

The justifications provided above for the changes are adequate to demonstrate that they are for the most part administrative in nature and do not change the location, function, or operation of equipment nor do they change the intent of the text. An adequate margin of safety is maintained. equipment reliability is unaffected, and previously evaluated accident analysis are not impacted. These changes do not involve new or special tests or experiments nor do they involve an unreviewed safety question. Therefore, since the changes do not involve a unreviewed safety question and do not reduce nuclear safety, then they are acceptable.

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Implementation Date: 06/11/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC-50058-A	FSAR Review and Verification Program
FSAR	FSAR Change Package 1574	Sections 5.1. through 5.5

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification Program. Specifically addressed are UFSAR Sections 5.1, 5.1.3, 5.2, 5.2.1, 5.2.2, 5.2.3, 5.2.4, 5.2.5, 5.2.7, 5.2.8, 5.3.1, 5.3.2, 5.4.3, 5.5.1, 5.5.3, 5.5.4, 5.5.6, 5.5.7, 5.5.10, 5.5.12 & 5.5.13 and Tables 5.1-1, 5.2-1, 5.2-8, 5.2-9, 5.2-12, 5.2-16, 5.2-21, 5.5-1, 5.5-6, 5.5-7, 5.5-10, 5.5-13, 5.5-15A, 5.5-15B & 5.5-16. and the associated calculations and system descriptions. 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 and documents revised as appropriate by FSAR Change Package Number 1574 and EDC 50058-A.

1. UFSAR Section 5.2.1.1 was revised to show that during plant cooldown, the boron concentration in the RCS is increased to the cold shutdown concentration. This change in reactivity control was made when Watts Bar previously reduced the boron concentration from 12% to 4%. Current operating cooldown processes, using boron injection, helps maintain a better core geometry and prevents any abnormal or unacceptable reactivity changes. This results in a more stable core with less potential for core damage and increased reactor safety.

2. UFSAR Section 5.2.1.2 notes the RCS piping with design temperature above 650°F. The pressurizer spray line is not included in this list. The initial Westinghouse design was for this line to be 650°F, however TVA qualified this line at 680°F because it is part of the pressurizer which is designed for 680°F. Qualifying this line for the higher temperature is more conservative and would thus not impact the ability of this line to perform its safety function.

3. UFSAR Section 5.2.1.5 in the Emergency Conditions section under "Small Loss of Coolant Accident" it is stated "For design purposes the small loss of coolant accident is defined as a break equivalent to the severance of a 3/8-inch inside diameter branch connection." This statement is confusing since a small break loss of coolant is considered from 6 inches down to 3/8 inch. Breaks 3/8 inch or less are not considered a small break since the high head injection pumps are capable of maintaining the RCS pressure. This sentence will be reworded to clarify that small loss of coolant accidents is greater than 3/8 inch but less than 6 inches. This is defined in several places in the FSAR (including Section 6.3.2.2) so this does not change the NRC's understanding of what constitutes a small break LOCA for Watts Bar.

SA-SE Number: WBPLMN-98-114-0 ***Implementation Date:*** 06/11/1999

4. In UFSAR Section 5.2.7.7.2, reference is made to the Reactor Building Floors and Equipment Drains (RBF&ED) pocket sump alarms on low level. No alarms are installed to perform this function nor are any required. No useful information would come from having a low level alarm on these sumps and thus no safety function is affected by removing this statement.

5. In UFSAR Sections 5.2.8.3 and 5.2.8.7, the references to the Code cases is being replaced by ASME Section XI subarticle IWA-2440. Code Cases are required to be listed in the ISI program which is available for the NRC review. This list is updated periodically and not appropriate for listing in the FSAR. These changes will have no impact on the integrity of the Reactor Coolant Pressure Boundary which will still be maintained in accordance with the applicable code cases in ASME Section XI.

6. UFSAR Section 5.5.1.3.4 is being revised from "low oil levels in the lube oil sumps signal an alarm in the control room and require shutting down the pump," to "low oil levels in the lube oil sumps signal an alarm in the control room and upon verification, could require shutting down the pump." Additionally in this section it is noted that high bearing temperature would require shutting down the pump. This statement is also being reworded to say it may require shutting down the pump. These pumps are non-safety related and bearing failure would not affect RCS integrity. Shutting the pumps down without attempting to correct the problem would however create a challenge to the safety system which is potentially avoidable and might not be required. Because an investigation could prevent avoidable safety system challenges without compromising safety, because the pumps are not shutdown until an investigation determines that it is necessary, and the UFSAR is being revised to agree with that.

7. In UFSAR Section 5.5.6.3.1 and in System Description Document (SDD) Section 3.3.3.2, it is noted that because there are two normally closed, de-energized valves in series in each flow path of the reactor head vent system, power lockout to the valves at the control board is not considered necessary. 10CFR50, Appendix R, added power lockouts to these valves so this statement will be deleted. The power lockouts are a additional safety measure and give additional assurance that the valves will be in their safe position.

8. In UFSAR Section 5.5.7.3.6 it is implied that all RHR can be remotely controlled from the control room. All operations of RHR which would be needed to mitigate the consequences of a accident can be controlled from the control room, but there are some non-essential functions which are not remotely controlled. This statement will be clarified to show that most RHR operations are controlled from the control room. The intent of this statement is to show that there is no radiation hazard associated with the operation of the RHR system and this wording does not change that intent.

9. In UFSAR Section 5.5.12.4 and 5.5.13.4, the FSAR notes that there are no full penetration welds within the valve body walls on RCS valves. This is a true statement at this time but the code does not prohibit valves of welded construction or the repair of valve bodies. It is foreseeable that a valve repair or new valve could have or require full penetration welds. ASME section XI requires the identification, inspection, and testing of these type welds and is part of TVA's program. These comments are being deleted to allow for possible future use without being in violation of the FSAR.

SA-SE Number: WBPLMN-98-114-0 ***Implementation Date:*** 06/11/1999

10. In accordance with vendor bill of materials or Code Data Reports alternate materials were referenced and used which were not shown in the UFSAR Tables 5.2-9 and 5.2-12 and SDD Tables 8, 10, and 22. This updates the FSAR and SDD to material types which has already been designed, fabricated and installed in the plant. This is not a change to the equipment and has no impact on the function of the equipment. The material being added is either the same chemical properties only in a different form or is equivalent to what is already specified. These materials were considered in the initial equipment design and inadvertently left out of the tables.

11. UFSAR Table 5.5-15A in the "Method of Detection" column in several places references Status monitor lights. These lights were replaced with annunciators around the time this table was made. The annunciators perform the same intent as the lights and attract the operators' attention more readily than the lights would.

12. Changes made and discussed for UFSAR Tables 5.5-15A and -15B were also made in TVA calculation EPM-SNM-062992. Since the discussion for the changes to the FSAR have already been addressed they will not be discussed again for the calculation.

13. In UFSAR Table 5.5-16 shows that the Reactor Vessel Head Vent System Solenoid Isolation valves are Safety Class 1. These valves are actually Class 2 as noted in UFSAR Section 5.5.6.1 and Figure 5.1-1 Sheet 1. These valves are downstream of a Class 1 to Class 2 transition piece flow restrictor and as such do not need to meet Class 1 requirements. Because of the transition piece a failure of these valves can be compensated for by the makeup pumps without a loss RCS pressure.

14. UFSAR Table 5.2-16 is being revised to show members 33-70 and their upset stresses in percent of allowable (9.3). This information is being added per updated information from Westinghouse in WCAP-9149. The stresses on these members is very small as a percent of allowable and poses no challenge to the RC pump supports to fulfill their safety function.

15. UFSAR Table 5.2-21 (Sheet 1 of 1), values in the "Combination of SSE and LOCA (in-kip)" column, are erroneously shown as "120, In-kip". The value for "Longest CRDM" will be changed to "278.6 in-kip" and the value for "Shortest CRDM" will be changed to "124.2 in-kip" per WCAP-13754. WCAP-13754 is a calculation which is titled: "Watts Bar Units 1 and 2 qualification of the reactor internals, Control Rod Drive Mechanisms (CRDMs), and CRDM supports for revised seismic spectra and the addition of a permanently attached head shield supports structure". This 1993 calculation resulted in modifications to the CRDM supports, however, these changes to the FSAR are only made to update the FSAR to make it consistent with the existing calculation. This FSAR change does not affect any plant equipment, testing or plant operation. Although these values are higher than the original table values, the values are still within the stress limits for the materials and thus create not challenges to the ability of this equipment to perform its safety function.

SA-SE Number: WBPLMN-98-114-0 ***Implementation Date:*** 06/11/1999

16. In SDD Section 3.2.3.3, it is stated that air passing though the RCP coolers is cooled to below 120°F. In the UFSAR this value is stated as 123°F. Westinghouse "E" Spec Number 678814, Section 4.4, specifies that the air discharged from the coolers does not exceed 122°F. The SDD will be revised to be consistent with the UFSAR and "E" spec. The SDD value appears to be based on containment limits of 120°F however the containment coolers will insure this value is not exceeded and the small difference in heat input from the pump coolers will have no impact on containment temperature.

17. SDD Section 3.2.4.1, 3.2.4.2, and 3.2.7.1 indicate that the surge line and spray line from the RCS to the pressurizer have thermal sleeves at each end and the normal and alternate charging connections, and cold leg accumulator injection connections to the RCS loop have thermal sleeves. The sleeves at the RCS end of these lines has been removed. The SDD will be revised to agree with the UFSAR and current plant configuration and design. In the referenced documents it was concluded that the nozzles are qualified to withstand all applicable design transients and will maintain their structural integrity without thermal sleeves and that they meet the ASME Code requirements, therefore, there is no impact on Nuclear Safety.

These changes are for "documentation" only with no impact on WBN's design bases or operational configuration. Document changes have been evaluated for plant operability during the review process and found not to affect the physical plant. Most changes were classified as minor. The changes which were determined to be technical in nature were evaluated for their impact on Nuclear Safety and plant operation.

The proposed changes do not affect any physical changes to the plant nor do they involve any plant procedures. No degradation to system performance is caused by these documentation only changes. The design perimeters and operating requirements are either improved or more conservative, therefore, no margins of safety are affected.

SA-SE Number: WBPLMN-98-116-0

Implementation Date: 06/12/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-50029-A	FSAR Review and Verification Program
FSAR	FSAR Change Package 1579	Chemical and Volume Control System

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification Program. Discrepancies between the UFSAR and design documents such as system descriptions, design criteria, calculations, and drawings were investigated and revisions recommended as appropriate. FSAR Change Package 1579 revises the UFSAR and Design Change Notice E-50029 revises System Description N3-62-4001 for consistency. These changes are documentation only with no impact on WBN's design basis or operational configuration. All document changes have been evaluated for plant operability during the review process and found not to affect the physical plant, design basis, or operating procedures. The proposed changes will not increase the likelihood of the design basis accidents occurring. They will not increase the dose to the public analyzed in Chapter 15 of the FSAR. No change will occur to the radiological consequences of accidents analyzed as a result of these changes. These changes do not introduce a failure pathway different from those identified and evaluated in the FSAR accidents. No new malfunction pathway will be introduced which have not previously been evaluated and identified. Also the margin of safety is not reduced by these changes.

The major changes are discussed below.

N3-62-4001 Section 3.2.9.5.6 is revising the set pressure of 1-RFV-062-518 to ≤ the design pressure of the pump discharge. The ASME Code requires pressure relief valves be set so that the pressure in the system or component not exceed 110% of design. Changing the set pressure equal to or less than design meets the ASME Code requirement and brings the system description in agreement with the FSAR. Table 15 of the system description already specified the discharge pressure for this valve which is the design pressure for the pump discharge.

The discussion associated with the Boric Acid recycle System such as Boric Acid Evaporator and its associated equipment is being deleted from both the FSAR and N3-62-4001 since it is not used for Unit 1 operation. The purpose of the evaporator was to reclaim boric acid from the RCS letdown. This is no longer done. The RCS letdown which is not returned to the RCS is processed by the Waste Disposal Mobile Demineralizer. The flow diagram for System 62 as well as the figures in the FSAR have already been previously revised to identify/remove those pieces of equipment (Unit 1 and Unit 2 interface) not being used for Unit 1 operation.

N3-62-4001, Section 3.1.3 is being revised to change the 100 additional minutes to 200 additional minutes.

SA-SE Number: WBPLMN-98-116-0 ***Implementation Date:*** 06/12/1999

N3-62-4001, Table 11 is being revised to change the design pressure of the centrifugal pump mini flow orifice from 2598 to 2800 psig. This orifice was supplied with the pump per the vendor's orifice drawing. Since the design pressure of the pump is 2800 psig this value will change for the orifice also. The Code data report for the orifice states that the orifice was hydro tested at 4950 psig. The Code typically requires hydro tests to be performed at 1.25 or 1.5 times the design pressure depending on the type of component. This equates to a design of 3300 psig therefore the change is well within these parameters. This change is primarily a clarification.

N3-62-4001, Paragraph 4.1.6 is deleted. This discussed charging flow rate which was initially specified for testing of flow control valve during hot functional testing. Westinghouse WAT-D-10532 states this applies only to hot functional testing and is not part of the design basis. However, WAT-D-10532 does state that 15 gpm is part of the design basis. This is flow rate discussed in Section 3.4.1.2, Item 8 and is being added to Paragraph 4.1.6 for clarification.

FSAR 9.3.4.2 C 2 Reworded from "hydrazine is employed" to "may be employed as an oxygen scavenging agent." Delete sentence which states "Hydrazine is not employed at any time other than at start up from cold shutdown." This statement implies that hydrazine has to be used. However, the water chemistry may be acceptable without using it. The FSAR also states that hydrogen is used as a oxygen scavenger due to radiolysis of the water in the core. The system description does not mandate the use of hydrazine during start up conditions but allows it if needed. This change only clarifies the use of hydrazine and makes the FSAR and system description consistent.

FSAR 9.3.4.2 C 5 discusses the primary water pumps and states that one starts on demand from reactor controller. This was revised to state that one pump runs continuously and provides flow to the blender as required. The logic diagram does not indicate these pumps start on demand. The system description states that one runs continuously and the flow controllers blend makeup accordingly. This change is a clarification and is now consistent with the system description. Because one pump is running continuously there is no need for a start signal.

FSAR 9.3.4.2.14 A 1 discusses leak off piping on the CCPs. The reference to leak off piping was deleted. Neither the system description, vendor drawings for the pumps, nor flow diagram show leak off piping. The CCPs have mechanical seals and do not require leak off lines. This is a clarification and makes the FSAR consistent with these documents.

FSAR 9.3.4.2.14 G 1 discusses the letdown orifices and their alignment. It had stated "...when the RCS pressure is less than normal or greater letdown flow during maximum purification or heat up." This was very confusing. The statement has been revised to "RCS pressure is less than the maximum allowable during normal RHR operating conditions. Maximum purification letdown flow is limited to 120 gpm when RCS exceeds allowable RHR operating conditions." This change is a clarification and is consistent with the system description, the Westinghouse process flow diagram, and System Operating Instructions.

FSAR 9.3.4.2.1 1 3 a was labeled relief valve for VCT however, the discussion in this paragraph was for the tank instead of the relief valve. This was revised to delete the discussion of the tank which is discussed in another paragraph for the tank and to discuss the relief valve. This is a clarification and brings the FSAR in agreement with the system description.

SA-SE Number: WBPLMN-98-116-0 ***Implementation Date:*** 06/12/1999

FSAR 9.3.4.2.2 A 1 e discusses the letdown orifices isolation valves and stated they were closed. However, the system description states the valves are open and the valves have to be open in order for pressure control valve PCV-62-81 to maintain RCS pressure. In addition, the flow diagram shows at least one valve open for normal operation. This change is primarily a clarification.

FSAR 9.3.4.2.2 A 3 adds discussion on RCS vacuum refill. This has already been evaluated in Safety Evaluation WBPLMN-97-117 for TI-68.012 and will not be discussed here.

FSAR 9.3.4.2.2 A 4 reworded the discussion on the RCPs being used in conjunction with the pressurizer heaters to heat up RCS. Also deleted the discussion of the RHR pumps being stopped after the RCPs are started and the pressure is maintained by the RHR system and low pressure letdown. This was revised to state the pressurizer heaters are used to form the steam bubble. The changes made are consistent with both System 62 and 68 system descriptions and actual operation as to when the RCPs are started and how the pressure and temperature is maintained. This is primarily a clarification.

FSAR 9.3.4.2.2 B 2 discusses the boration vs. rod position which was originally wordy. "The most important intelligence available to the plant operator, enabling him to determine whether dilution of the RCS is necessary, is the position of the control rods. For example, if the control rods are below their desired position, the operator must borate the reactor coolant to bring the rods outward. If, on the other hand, the control rods are above their position, the operator must dilute the reactor coolant to bring the rods inward." This was changed to " Control rod position provides the operator with an indication of whether dilution or boration of the RC is necessary. If rod position is out of the desired range proper manipulation of boron concentration will return the rods to the desired range." The revised discussion still has the same meaning but is stated in a more concise manner. It does not change the meaning of the paragraph.

FSAR 9.3.4.2.2 B 3 discusses xenon and its affect on the degree of shutdown. The paragraph was reworded as shown below. It still has the same meaning as the original test, but it is more concise. "Following shutdown, xenon buildup occurs and increases the degree of shutdown ($\Delta k/k$). The effect of xenon buildup is to increase the degree of shutdown ($\Delta k/k$) to a maximum at about eight hours following shutdown from equilibrium full power conditions. If hot shutdown is maintained past this point, xenon decay results in a decrease in degree of shutdown. Since the $\Delta k/k$ value of the initial xenon concentration is high (assuming that an equilibrium concentration had been reached during operation), boration of the reactor coolant is necessary to counteract the xenon decay and maintain shutdown."

SA-SE Number: WBPLMN-98-116-0 ***Implementation Date:*** 06/12/1999

FSAR 9.3.4.3 E 2 discusses valve packing and leak off. This was revised to add the use of enhanced live load packing. This was evaluated by design change process and Amendment 69 and is already discussed in SER Supplement 11. Therefore, there will be no further discussion.

FSAR 9.3.7 discussed the boron recycle/recovery system. This entire section was revised with the items no longer being used for unit one operation being deleted and the others either transferred to 9.3.4 or to other sections as appropriate. The discussion associated with the boric acid recycle system such as boric acid evaporator and its associated equipment is being deleted from both the FSAR and system description since it is not used for Unit 1 operation. The purpose of the recycle system was to reclaim boric acid from the RCS letdown. This is no longer done. The RCS letdown which is not returned to the RCS is processed by the waste disposal mobile demineralizer. The flow diagram for System 62 has already been previously revised to identify/remove those pieces of equipment not being used for Unit 1 operation.

A new section is being added to Section 6.1.2 of the FSAR to include the temporary catch basins and the lead blankets. A large percentage of the reactor coolant system is stainless steel. The catch basins used inside the containment are made from polyethylene which is normally compatible with stainless steel unless it is fire retardant which then is halogenated. The Material Safety Data Sheet (MSDS) for the catch basins does not indicate that it is halogenated. The catch basins are temporary and are used only between outages to direct leaks to appropriate drains. The covers for the lead blankets per the MSDS contains Hypalon (chlorosulfonated polyethylene), Vinyl, or methylpolsiloxanes which is not normally compatible with stainless steel. These blankets are normally used during outages for ALARA protection when the temperature is ambient and the conditions are dry. Therefore, it is not likely that the covers will contaminate the stainless steel RCS. During operations they are stored in designated areas outside the crane wall away from the stainless steel RCS. The cover is susceptible to break down due to radiation and will evolve hydrogen chloride (HCl) gas. However, the 1200 lbs. of cover which is being allowed inside the containment now in conjunction with the 40 year dose for the raceway from calculation WBNAPS4-008 will yield less than 1/4 lb. of gas. In addition the blankets are stacked which leaves only the top and sides of the stack which are exposed to radiation thus generating even less gas. Assuming the gas emitted is all chlorine, the amount will produce a concentration in the containment equivalent to approx. 1 ppm which is acceptable for demineralized water.

Paragraph 6.1.2.1 was revised to correct the amount of cable insulation inside the containment. The organic materials are required by Regulatory Guide 1.70 to be identified and quantified. This is due to the material emitting hydrogen due to the radiation in accident conditions. The hydrogen emitted by the cable insulation , lead blanket covers, and catch basins were evaluated in calculation WBNSSG4-002 and determined to be inconsequential (less than 0.1 %) of total containment volume and well within the allowable tolerance in the calculation. Therefore, organic materials will no longer be tracked.

The FSAR is being revised to add Epoxy coating, to delete the reference to the thickness of the coatings, and add a statement to allow the use of other qualified coating systems. The thickness of the coating both under coat and top coat are specified in General Engineering Specification, G-55 and are as specified by the manufacturer and the design basis accident (DBA) test results. There is no restriction as to the type of coating used as long as it meets the DBA testing requirements in ANSI 101.2. G-55 requires coatings used inside containment to meet DBA testing requirements prior to use.

SA-SE Number: WBPLMN-98-117-0

Implementation Date: 04/30/1999

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50059-A FSAR Change Package 1584	Review of FSAR Sections 4.2, 6.2, and 7.6.6.

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification. Specifically addressed are UFSAR Sections 4.2.2, 4.2.3. 4.2.4, 6.2.1, 6.2.2, 6.2.4, 6.2.6 and 7.6.6 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 and documents revised as appropriate by EDC E-50059-A. These changes are for "documentation" only with no impact on WBN's design bases or operational configuration. The document changes have been evaluated for plant operability during the review process and found not to affect the physical plant. The proposed "documentation only" changes to the UFSAR and the design documents will not increase the likelihood of the design basis accidents occurring. These changes to the UFSAR and the design documents do not effect physical changes to the plant. These documentation-only changes will not increase the dose to the public analyzed in the UFSAR Chapter 15.5. No change will occur to the radiological consequences of accidents analyzed as a result of these changes. The proposed changes do not adversely affect the plant design bases nor do they cause any changes to the physical plant. The credible failure modes for the systems affected by these changes, have been evaluated against the accidents identified in the FSAR and concluded they do not introduce a failure pathway different from those identified and evaluated in the FSAR 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.

The bases of the Technical Specifications have been reviewed for determining if any margins of safety are affected by these documentation changes. No margin of safety is identified in the bases section of the Technical Specifications which could be reduced by these changes. The change to FSAR section 7.6.6 to describe the removal of power to FCV-63-8 and -11 during residual heat removal (RHR) cooldown does not involve an unreviewed safety question. Removal of power in itself does not have the potential to increase the probability of any accidents previously evaluated in the FSAR. The change does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated. Power removal to these valves is only applicable during Mode 4. Power can be restored and the valves opened in the event that they are needed for long term containment sump recirculation in the event of a Mode 4 LOCA. The consequences of an accident previously evaluated is not increased. In the event of a Mode 4 LOCA, FCV-63-8 and -11 are opened to transition from the refueling water storage tank (RWST) injection mode to the containment sump recirculation mode. Previous analyses of required flow during a Mode 4 LOCA have determined that the flow of one centrifugal charging pump is adequate to maintain core cooling. The duration of the injection mode for a Mode 4 LOCA, with only a centrifugal charging pump (CCP) drawing suction from the RWST, is approximately 10 hours. The relatively low flowrate out of the RWST would allow time for Operations to evaluate restraints and restore power to the valves to accomplish the RWST to containment sump swapover.

SA-SE Number: WBPLMN-98-117-0 ***Implementation Date:*** 04/30/1999

Therefore, safety injection/core cooling capability is not impacted by the change to FSAR section 7.6.6. The consequences of a malfunction of equipment important to safety is not increased. The design redundancy that exists by having both 1-FCV-63-8 and -11 for the recirculation mode is not impacted. Removal of power to FCV-63-8 and -11 during RHR cooldown does not create the possibility for an accident of a different type and does not create the possibility for a malfunction of a different type. Power removal during Mode 4 actually helps to prevent the overpressurization of the safety injection pump and CCP suction piping. Technical Specifications 3.5 for the ECCS system have been reviewed and the change to FSAR section 7.7.6 to describe removal of power to FCV-63-8 and -11 during RHR cooldown does not have the potential to reduce any margins of safety defined in the bases for these Technical Specifications.

Therefore, based on the above justifications, the proposed changes do not involve a unreviewed safety question and are acceptable from a nuclear safety standpoint.

SA-SE Number: WBPLMN-98-117-1

Implementation Date: 04/30/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change FSAR	EDC E-50059-A FSAR Change Package 1584 FSAR Change Package 1584-51	Review of FSAR Sections 4.2, 6.2, and 7.6.6.

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification. Specifically addressed are UFSAR Sections 4.2.2, 4.2.3. 4.2.4, 6.2.1, 6.2.2, 6.2.4, 6.2.6 and 7.6.6 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 and documents revised as appropriate by EDC E-50059-A. These changes are for "documentation" only with no impact on WBN's design bases or operational configuration. The document changes have been evaluated for plant operability during the review process and found not to affect the physical plant. The proposed "documentation only" changes to the UFSAR and the design documents will not increase the likelihood of the design basis accidents occurring. These changes to the UFSAR and the design documents do not effect physical changes to the plant. These documentation-only changes will not increase the dose to the public analyzed in the UFSAR Chapter 15.5. No change will occur to the radiological consequences of accidents analyzed as a result of these changes. The proposed changes do not adversely affect the plant design bases nor do they cause any changes to the physical plant. The credible failure modes for the systems affected by these changes, have been evaluated against the accidents identified in the FSAR and concluded they do not introduce a failure pathway different from those identified and evaluated in the FSAR 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.

The bases of the Technical Specifications have been reviewed for determining if any margins of safety are affected by these documentation changes. No margin of safety is identified in the bases section of the Technical Specifications which could be reduced by these changes. The change to FSAR Section 7.6.6 to describe the removal of power to FCV-63-8 and -11 during residual heat removal (RHR) cooldown does not involve an unreviewed safety question. Removal of power in itself does not have the potential to increase the probability of any accidents previously evaluated in the FSAR. The change does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated. Power removal to these valves is only applicable during Mode 4. Power can be restored and the valves opened in the event that they are needed for long term containment sump recirculation in the event of a Mode 4 LOCA. The consequences of an accident previously evaluated is not increased. In the event of a Mode 4 LOCA, FCV-63-8 and -11 are opened to transition from the refueling water storage tank (RWST) injection mode to the containment sump recirculation mode. Previous analyses of required flow during a Mode 4 LOCA have determined that the flow of one centrifugal charging pump is adequate to maintain core cooling. The duration of the injection mode for a Mode 4 LOCA, with only a centrifugal charging pump (CCP) drawing suction from the RWST, is approximately 10 hours. The relatively low flowrate out of the RWST would allow time for Operations to evaluate restraints and restore power to the valves to accomplish the RWST to containment sump swapover.

SA-SE Number: WBPLMN-98-117-1 ***Implementation Date:*** 04/30/1999

Therefore, safety injection/core cooling capability is not impacted by the change to FSAR section 7.6.6. The consequences of a malfunction of equipment important to safety is not increased. The design redundancy that exists by having both 1-FCV-63-8 and -11 for the recirculation mode is not impacted. Removal of power to FCV-63-8 and -11 during RHR cooldown does not create the possibility for an accident of a different type and does not create the possibility for a malfunction of a different type. Power removal during Mode 4 actually helps to prevent the overpressurization of the safety injection pump and CCP suction piping. Technical Specifications 3.5 for the ECCS system have been reviewed and the change to FSAR section 7.7.6 to describe removal of power to FCV-63-8 and -11 during RHR cooldown does not have the potential to reduce any margins of safety defined in the bases for these Technical Specifications.

Therefore, based on the above justifications, the proposed changes below do not involve a unreviewed safety question and are acceptable from a nuclear safety standpoint.

SA-SE Number: WBPLMN-98-118-0

Implementation Date: 12/02/1998

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-50063-A	Drawing Discrepancies - Turbogenerator Control

Description and Safety Assessments:

Drawing Deviation (DD) 98-0069 documented a discrepancy on the high pressure steam seal supply line to the turbine glands with respect to the inlet and discharge pipe sizes at relief valve 1-RFV-047-0701. The subject relief valve and attached piping are part of the Westinghouse-designed portion of the Turbogenerator Control, and is located near the northwest corner of low pressure Turbine 1A on Turbine Building Elevation 755.0. Specifically, the flow diagram incorrectly depicts the inlet piping to the relief valve as 6-inch diameter and does not indicate the discharge pipe size. Westinghouse drawings 4601DOI; 880C748, Sheets 2 and 3; and 721J854, Sheets 2 and 3 as well as TVA physical piping drawing 47W430-7 indicate the inlet pipe size is 4-inch diameter and the discharge pipe size is 6-inch diameter, which agrees with the as-built configuration.

DD 98-0069 has been incorporated into Engineering Document Change (EDC) E-50063-A. The EDC will revise the affected flow diagram to correctly reflect the as-built inlet and discharge pipe sizes at the relief valve. Implementation of this "documentation only" change will resolve the subject discrepancy; no field work will be required or implemented under EDC E-50063-A.

The following design basis accidents (Reference FSAR Chapters 6 and 15) have been evaluated for impact:

a. Condition I - Normal Operation and Operational Transients
b. Condition II - Faults of Moderate Frequency, 15.2.7, Loss of External Electrical Load and/or Turbine Trip

The proposed activity implements a minor "documentation only" drawing change with respect to passive components (i.e., pipe and fittings). The only credible failure mode for the affected components would be pipe rupture. However, the proposed activity will neither increase nor decrease the likelihood of such a failure nor will it introduce any new failure modes.

In summation, the "documentation only" change to be implemented under EDC E-50063-A:

- Is not expected to adversely affect NRC's understanding of the design, configuration, or operation of WBN;
- Will not escalate the frequency class of any accident or event in the FSAR to a higher frequency class;
- Will not adversely affect the ability of any safety-related system or equipment from performing their intended safety functions;
- Will 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 bases;
- Will not cause any undesirable interactions with other systems important to safety;
- Has 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 FSAR, nor will it introduce any new malfunction pathways;

SA-SE Number: WBPLMN-98-118-0 **Implementation Date:** 12/02/1998

- Will not increase the likelihood of a radiological release or have any adverse radiological impact on other affected systems or equipment as a result of an accident or malfunction of equipment.
- Will not impede access to vital areas of the plant, hamper actions required to mitigate an accident, or cause an increase in onsite or offsite dose as the result of an accident or malfunction of equipment.
- Will not adversely affect 10 CFR 20 or 10 CFR 100 compliance.
- Has been evaluated against the applicable accidents identified in the FSAR with respect to the affected system and determined not to introduce any new accident scenarios or failure pathways;
- Does not increase the probability of any analyzed accident;
- Do not involve any new single failures; and
- Has been reviewed to determine if any margins of safety specified in the bases section of the Technical Specifications might be reduced and none was identified.

Therefore, based on the preceding evaluation, it has been determined that implementation of the subject documentation change:

- Will not create the possibility of a new type of accident or equipment malfunction not previously analyzed in the FSAR 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 FSAR;
- 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 change does not involve an unresolved safety question and is acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-98-120-0

Implementation Date: 06/23/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC Number E-50061 FSAR Change Package 1582	FSAR Documentation Changes - Chapters 2, 3, 11, and 12

Description and Safety Assessments:

FSAR Sections 2.2.3, 3.1, 3.11 11.1, 12.1, 12.2, and 12.3 are revised to reflect changes resulting from a complete review of these sections. The majority of the items are considered minor as defined in Section 6.0 of SPP-9.4 and do not require a Safety Assessment and Safety Evaluation in accordance with NADP-7 Section 3.5.5. Safety Assessment items 10, 30, 33, 35, 59, 65, 69, 70, and 77 (as tabulated below) are changes which could be considered to be a technical change to the UFSAR. These items are addressed in the Safety Evaluation. These changes are documented in Problem Evaluation Report (PER) WBPER980417 and are being resolved under EDC E-50061 -A.

10. (Page 3.1-40) In the compliance statement for Criterion 63, revised the first sentence. The previous wording indicated that the failure of the spent fuel pool cooling would produce a local and control room alarm. The change reflects the fact that the loss of cooling would be determined by local temperature indication and that loss of spent fuel pool level would annunciate in the control room. (Refer to item 10 of SA Section A)

30. (Page 12.2-1) Revised the Axial Peaking Factor listed in Section 12.2.1.1.1, for the six foot planes, to correct a typographical error and reflect the correct value as provided in Westinghouse WCAP7664, R1. The change revises the historical information in this section to reflect the actual values used in the design of the plant (Refer to item 30 of SA Section A.)

33. (See Table 1.2-4) Revised the value for Iodine 132 to reflect the correct historical value documented in SQN calculation TI-654. (Refer to item 33 of SA Section A.)

35. (See Table 12.2-7) Revised the Table to reflect the correct historical values documented in SQN calculation TI-656. (Refer to item 35 of SA Section A-)

59. (Page 12.3-11) In the first sentence of the third paragraph changed the minimum water shielding from 10' 6" to 9.9'. This is an editorial change to be consistent with the first sentence of the previous paragraph. The value of 9.9 feet was determined by reanalysis as an acceptable value and the FSAR previously revised to reflect this information. This change was overlooked in a previous FSAR corrections.

65. (Page 12.3-13) Revised the list of extra-control room missions required following an accident. The list was changed to accurately reflect the current post accident action requirements. The supporting calculations were reviewed to determine active missions. This change makes the FSAR information consistent with the information reviewed by the NRC in resolution of IFI 390/93-81-03. The change also deletes the requirement for special protective equipment when performing actions in the shutdown board room. Based on reanalysis the protective equipment is no longer required. (Refer to item 65 of SA Section A).

SA-SE Number: WBPLMN-98-120-0 ***Implementation Date:*** 06/23/1999

69. (Page 12.3-20) The next to the last sentence in the first paragraph of Section 12.3.4.2.2 had specified that there were 11 channels for monitoring airborne particulate activity. Table 12.3-5 shows only 8 channels. This editorial change makes the text compatible with the Table which uses the correct number and is consistent with Radiation Monitoring System Design Criteria WB-DC-40-24. (Refer to item 69 of SA Section A.)

70. (Page 12.3-20) The third sentence of the second paragraph of Section 12.3.4.2.2 was revised to clarify the operation of the portable CAMs. The CAM does not automatically regulate sample flow through the filter. The flow is set during calibration. This is not considered to be a major change since the monitoring function of the CAMs as described in the FSAR is not affected. (Refer to item 70 of SA Section A).

77. (See Table 12.3-6, Sheet I of 8) Corrected a typographical error for the wall thickness at location A4 on elevation 676 to be consistent with the as built drawings. (Refer to item 77 of SA Section A.)

UFSAR Change Package 1582 documents the changes to the UFSAR that are addressed by this evaluation.

The following design basis accidents (Reference FSAR Chapters 6 and 15) have been evaluated for impact

a. Condition I - Normal Operation and Operational Transients
b. Condition 11 - Faults of Moderate Frequency
c. Condition III - Infrequent Faults
d. Condition IV - Limiting Faults:

The changes addressed by this evaluation do not impact any accidents evaluated in the UFSAR. Also, these changes do not affect the operation of any safety related equipment/system. There are no failure modes associated with these changes. These are changes to the UFSAR to replace incorrect information with the correct information as specified in System Descriptions, Design Criteria, Calculations, and Vendor Letters. The changes do not affect proper equipment/system operation and there are no credible failures associated with these changes.

This change updates UFSAR Sections 2.2.3, 3.1, 3.11, 11.1, 12.1, 12.2, and 12.3. Based on a review for clarity, accuracy, and completeness. The majority of the identified changes are considered either minor, editorial, or administrative which do not change the technical information already provided in the UFSAR. Other changes were made which could be considered functional changes. These changes fall Into the following categories:

1. Changes to historical information. These changes provide the correct values used in the initial design of the plant and do not affect current plant design or analyses.
2. Changes made to revise outdated information overlooked in previous amendments to the UFSAR. The changes are made to make the UFSAR material consistent by changing the outdated information to reflect the correct values specified elsewhere in the UFSAR.
3. Changes to tabular information. The change is made to show actual values of the constructed plant as designed and shown in plant documents.
4. Clarification of monitoring system capabilities.
5. Clarification of personnel radiation protection .

SA-SE Number: WBPLMN-98-120-0 ***Implementation Date:*** 06/23/1999

Additional information on each of the proposed changes addressed in this Safety Evaluation is provided below:

10. (Page 3.140) In the compliance statement for Criterion 63, revised the first sentence for consistency with FSAR Section 2.4.14.2.2. The previous wording indicated that the failure of the spent fuel pool cooling would produce a local alarm. The change reflects the fact that the loss of cooling would be annunciated in the control room and indicated locally. This change has no affect on event mitigation since actions are taken based on control room instrumentation and not local instrumentation.

30. (Page 12.2-1) Revised the Axial Peaking Factor listed in Section 12.2.1.1.1, for the six foot planes, to correct a typographical error and reflect the correct value as provided in Westinghouse WCAP7664, Rl. The change revises the historical information in this section to reflect the actual values used in the design of the plant.

33. (Table 12.2-4) Revised the value for CVCS Holdup Tank source term value for Iodine 132 to reflect the correct historical value documented in TVA calculations.

35. (Table 12.2-7) Revised the Table to reflect the correct historical CVCS Evaporator source term values documented in TVA calculations.

59. (Page 12.3-11) In the first sentence of the third paragraph changed the minimum water shielding from 10' 6" to 9.9'. This is an editorial change to be consistent with the first sentence of the previous paragraph. The value of 9.9 feet was determined by reanalysis as an acceptable value and the FSAR previously revised to reflect this information. This change was overlooked in a previous FSAR amendment.

65. (Page 12.3-13) Revised the list of extra-control room missions required following an accident. The list was changed to accurately reflect the current post accident action requirements. The supporting calculations were reviewed to determine active missions. This change makes the FSAR information consistent with the information reviewed by the NRC in resolution of IFI 390/93-81-03. The change also deletes the requirement for special protective equipment when performing actions in the shutdown board room. Based on reanalysis the protective equipment is no longer required,

69. (Page 12.3-20) The next to the last sentence in the first paragraph of Section 12.3.4.2.2 had specified that there were 11 channels for monitoring airborne particulate activity. Table 12.3-5 shows only 8 channels. This editorial change makes the text compatible with the Table which uses the correct number and with the Radiation Monitoring System Design Criteria.

70. (Page 12.3-20) The third sentence of the second paragraph of Section 12.3.4.2.2 was revised to clarify the operation of the portable Continuous Air Monitors (CAMs). The CAM does not automatically regulate sample flow through the filter. The flow is set during calibration. This is not considered to be a major change since the monitoring function of the CAMs as described in the FSAR is not affected.

77. (See Table 12.3-6, Sheet 1 of 8) Corrected a typographical error for the wall thickness at location A4 to be consistent with the as built drawings.

SA-SE Number: WBPLMN-98-120-0 ***Implementation Date:*** 06/23/1999

None of the changes, including the numerical changes, affect existing design or analyses. The changes do not involve modifications to any system, structure, or component including the radwaste system and do not involve special tests or experiments. The consequences and probability of accidents previously performed and malfunctions of equipment important to safety are not affected. The changes do not create any new failure modes and the Technical Specifications are not impacted. 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-98-121-0

Implementation Date: 05/03/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC Number E-50037-A	Review of UFSAR Section 6.2.5
UFSAR	FSAR Change Package 1571	

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification Program. Specifically addressed are UFSAR Section 6.2.5 and the associated system description document and calculation changes which were identified as a result of that review. Regulatory Guide 1.70, "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 and documents revised as appropriate by Engineering Document Change (EDC) E-50037-A, which partially implements the corrective action for WBPER980417. These changes are for "documentation" only with no impact on WBN's design bases or operational configuration. All document changes have been evaluated for plant operability during the review process and found not to affect the physical plant. The proposed "documentation-only" changes to the UFSAR and the design documents will not increase the likelihood of the design basis accidents occurring. These changes to the UFSAR and the design documents do not effect physical changes to the plant, nor do they involve any plant procedures. These "documentation-only" changes will not increase the dose to the public analyzed in the UFSAR Chapter 15.5. 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. 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.

Changes, which are being implemented, typically fit into four categories, as follows:

I. Administrative in nature (e.g., typographical errors, misplaced/incorrect reference numbers, grammatical errors, duplicate information, excessive verbiage, text requiring clarifications, information that is no longer valid, inadvertent exclusion of text, historical information, and minor Figure changes, etc.).
II. Markings or deleting of text for features specifically identified in the UFSAR as not required for WBN Unit 1 operation or having to do with Unit 2 or 2 unit operation (e.g., the Gas stripper and boric acid evaporator package, etc.).
III. Deletions of unnecessary or non-contributory details (e.g., such as Materials used, pipe sizes, etc.).
IV. Corrections, technical in nature, i.e., revisions necessary to provide consistency among the UFSAR, System Descriptions, Design Criteria, supporting calculations, and drawings, etc.

SA-SE Number: WBPLMN-98-121-0 ***Implementation Date:*** 05/03/1999

All changes are shown on marked-up copies of the UFSAR in FSAR Change Package Number 1571, though not necessarily identified by category number. Category I, II, and III changes are not included in Table 1 below since they are considered editorial/clarification changes that do not change the intent of the UFSAR text. Category IV are technical changes and are condensed in Table 1 below and will require further evaluation herein.

Table 1: GENERAL OVERVIEW OF CATEGORY IV CHANGES

UFSAR SECTION, ITEM DESCRIPTION, CALCULATION	DESCRIPTION of REVISION	JUSTIFICATION OF REVISION
UFSAR 6.2.5.2	Revised the H_2 Recombiners design range from 0-4 % H_2 to 0 - 6 % H_2	The original design of the Recombiners was 0 - 4 % H_2 based on test results. WCAP-7820, Supplement 6, retested the Recombiners for IEEE 323-1974 qualification. This new test qualified the Recombiners to 0 - 6 % H_2. The Recombiners will not be energized if the H_2 concentration in containment is ≥ 5.0 % H_2 (includes inaccuracy of instruments). In addition, the H_2 generation analysis shows that 5 % H_2 is not reached for approximate 6 days with no Recombiners in operation. Therefore, this "documentation-only" change does not present any safety issue since (1) the Recombiners are required to be energized within 24 hours of a DBE, (2) the H_2 generation analysis show that H_2 concentration is below 4 % flammable limit, and (3) this change is consistent with the safety analysis presented in this section.
UFSAR 6.2.5.2, System Description N3-83-4001 EPM-ED-120392	The EOP calculation (WBN-OSG4-188), which is design output, has that H_2 Recombiners are not energized if the H_2 concentration is ≤ 0.6% or if the H_2 concentration is ≥ 5.0%. The UFSAR, SDD, and EPM-ED-120392 are being revised to clarify these set points.	The low setpoint is used to prevent superfluous actuation of the Recombiners. The high setpoint is used to prevent the Recombiners from being energized outside the design bases by providing a 1% margin as compared to the design. The H_2 generation analysis shows that 5% H_2 is not reached for approximate 6 days with no Recombiners in operation. Therefore, this "documentation-only" change does not present any safety issue since (1) the Recombiners would not be energized outside the design range, (2) if the H_2 concentration in containment was at detonable limits, the high setpoint would prevent an explosion, and (3) this change is consistent with the safety analysis presented in this section.

SA-SE Number: WBPLMN-98-121-0 ***Implementation Date:*** 05/03/1999

In summation, the changes listed in Table 1 and the associated design document changes:

- Clarify WBN's design bases 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;
- 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 NRC's understanding of the design, configuration, or operation of WBN;
- Will not alter the frequency class of any accident or event in the UFSAR to a higher frequency class;
- Will not adversely affect the ability of the affected systems or equipment 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 or equipment as a result of an accident or malfunction of equipment;
- Will not impede access to vital areas of the plant, hamper actions required to mitigate an accident, or cause an increase in onsite or offsite dose as the result of an accident or malfunction of equipment;
- Will not adversely affect 10 CFR 20 or 10 CFR 100 compliance;
- Have been evaluated against the applicable accidents identified in the SAR with respect to the affected systems and equipment and determined not to introduce any new accident scenarios or failure pathways;
- Do not increase the probability of any analyzed accident:
- Do not involve any new single failures; and
- Have been reviewed to determine if any margins of safety specified in the bases section of the Technical Specifications might be reduced and none was identified.

Therefore; based on the above evaluation, implementation of the changes listed in Table 1 and the associated design document changes:

- 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;
- 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-98-123-1

Implementation Date: 01/27/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC Number E-50079A Drawing Deviations (DD) 98-0075 and 98-0082	Documentation changes only - Drawing Deviations

Description and Safety Assessments:

This Safety Evaluation for Design Change EDC E-50079-A resolves the discrepancies identified in Drawing Deviations (DDs) 98-0075 and 98-0082. DD 98-0075 identifies equipment that should be removed from TVA drawing and the Master Equipment List (MEL) since the equipment is contained within the vendor-controlled equipment boundary of the Spent Resin Packaging System (SRPS). Design Change EDC E-50079-A will revise the TVA flow diagram to (1) add resin sampler (0-SMPL-77-1817), (2) delete cask, 0-CASK-77-1818, and pump, 0-PMP-77-1819, which are contained within the boundaries of the vendor-controlled equipment of the SRPS, (3) and correct the vent flow path. A drawing was also revised to show two hoses (0-HOSE-77-1816 and 0-HOSE-77-1817).

DD-98-0082 identifies the following valves 1-ISV-059-522, -683, -684, -685, -686, -687, -688, -689, 691, -692, -693, -695, -696, and -697 which should be labeled as diaphragm valves. This design change revises the flow diagram to show the valves as diaphragm valves which eliminates the discrepancy between the drawing and Bill of Materials (BMs). In addition, DD 98-0082 identifies TVA Drawings which will be revised to show valve 1-62-949, the manual isolation valve for the Reactor Coolant Drain Tank, to be in the normally open position. This change will eliminate the discrepancies between procedure SOI-62.06 and the drawings.

This change is technically acceptable since valve 1-62-949 only provides manual isolation of the Waste Disposal System (WDS) RCDT discharge flow into Chemical and Volume Control System (CVCS) holdup tank A. Normal alignment is to either the CVCS holdup tank or Tritiated Drain Collector Tank (TDCT). The current operating instruction (SOI-62.06) uses the CVCS holdup tank as the normal alignment. Flow to the CVCS holdup tank is provided by Reactor Coolant Drain Tank (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 Main Control Room 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, 1-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 changes documented within DDs 98-0075 and 98-0082 are documentation only changes and do not require field work. The components that are impacted by the documentation only change and it associated components are located in the Auxiliary Building. No components have 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.

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 FSAR, will not increase the consequences of an accident or malfunction previously evaluated in the FSAR, does not create the possibility for a new accident or malfunction evaluated in the FSAR nor reduce the margin of safety as defined in the FSAR.

SA-SE Number: WBPLMN-98-125-1

Implementation Date: 04/14/1999

Document Type:	*Affected Documents:*	*Title:*
Design Change	DCN M-39972-B	Temporary Outage Cooling System
FSAR Change Package	FSAR Change Package 1537	

Description and Safety Assessments:

This change adds permanent connections and isolation valves in the ERCW "A" Train supply and return headers to the Unit 1 Reactor Building. These connections and valves meet ASME Section III Class 3 (TVA Class C) requirements for fabrication, installation and materials. ERCW system pipe supports will be modified as needed to accommodate the load increases from the permanent or temporary configurations. Permanent penetrations are also added in the manway access to the ERCW Pipe Tunnel in the yard near the primary water storage tank and in the Al column line wall into the El 692.0 Penetration Room in the Auxiliary Building. These penetrations will be used to connect the portable chiller in the yard to the lower compartment coolers (LCCs) using the newly installed connections in the ERCW "A" Train headers. During normal plant operations (Modes 1 Through 4) the new isolation valves will be closed with blind flanges installed on all connections including the penetrations in the ERCW Pipe Tunnel. In Modes 5 and 6 the blind flanges will be removed from the valves and penetrations and flexible hoses attached to connect the system to the chiller. The portion of the ERCW system that serves the Unit 1 Reactor Building will be isolated from the rest of the ERCW system by closing the existing isolation valves in the supply and return headers. The temporary chilled water system will be initially filled with water from the ERCW system and vented of air and pressurized. A pressure test will be conducted of the chilled water system after it is completed to assure its leaktightness. The potential for flooding during operation with chilled water (Modes 5 and 6) is enveloped by the flooding potential during normal operation. The Unit 1 Reactor containment will then be cooled by the external chilled water system.

While making the analysis revision for the lower compartment cooling changes described above, a number of deficiencies were found in the existing analysis documentation for ERCW problem N3-6709A. This is documented by WBPER981116. To achieve compliance of the N3-67-09A piping, equipment, and supports with the applicable civil engineering design criteria (WB-DC-40-31.7, WB-DC-40-31.9, and WB-DC-40-31.2), it is necessary to change 11 pipe supports (delete 5, modify 5, and add 1). It is also necessary to grind one pipe weld flush so that a lower stress intensification factor is justified. The design changes due to WBPER981116 are shown on DCAs in the DCN M-39972 Revision B package. Supporting civil engineering calculations are also referenced in the package.

In addition, PER 98-013305-000 documents missing stiffeners on the embedded plate attachment for pipe support 67-1 ERCW-R1 45, which is located within the boundaries of ERCW piping analysis problem N3-67-23A. A DCA to correct this deficiency is also in the DCN M-39972 Revision B package. Supporting civil engineering calculations are referenced in the package.

This DCN provides design information related to the following independent activities:

- Installation of a 480V substation as a source of temporary power for a portable chiller
- Installation of isolation valves and flanges into the existing ERCW piping for future use during outages
- Modification of existing piping and pipe supports to re-establish design basis configuration
- Installation of permanent penetrations and loop seal piping for use in routing temporary flex hose during future outages
- Sketches for future installation (by others) of temporary flex hose and equipment to provide chilled cooling water to the lower compartment coolers during outages

SA-SE Number: WBPLMN-98-125-1 ***Implementation Date:*** 04/14/1999

The 480V substation will be installed as Stage 1 of DCN M-39972-6. The pipe connections in the ERCW headers, the additional penetrations, and the changes for WBPER9811 16 and PER 98-013305-000 will be done as Stage 2. It is not necessary to define separate stages for the PER related design changes since they do not require welding to the pressure boundary. These modifications to existing piping and supports can be done prior to or during the outage. Sequencing and temporary support requirements in DCN M-39972-B ensure that compliance with the applicable operability design criteria (WB-DC-20-33) is maintained during installation. Operability of the existing configuration prior to installation of the design changes has been verified for the PER conditions. Pipe support work on permanent piping is performed in accordance with existing instructions in MAI-4.2A. Sketches and requirements for future installation of the temporary flex hose, its supports, and temporary equipment are provided for use by others in the future.

Pressure boundary integrity of the permanent and temporary equipment is required to prevent building flooding from normal equipment usage and to prevent building flooding during flood mode. The permanent connections to the ERCW headers have blind flanges installed for normal operation. The connections will be pressure tested to the same requirements as the original piping. The flexible hose is specified and pressure tested at 150% of operating pressure or 225 psi. The pressure test will ensure, with an acceptable factor of safety, that the rubber hose will not burst and that the end connections will not separate. The hose and other pipe fittings used will also have a similar pressure rating. The test will ensure that the temporary equipment will be capable of withstanding the pressure that could be experienced by permanent equipment in this type service. A leak check test is required at start up of the system and at periodic intervals during service to ensure that there is no leakage with the temporary equipment installed.

The steps needed to achieve this permanent condition and to achieve temporary cooling are outlined below.

Valves 1-FCV-67-83, -88, -91, -96, and 1-67-523A, -530A and -577A are closed to isolate the 10-inch supply and 8-inch return "A" Train ERCW headers. These pipe segments can then be drained and the tees and "spectacle" flanges installed. The spectacle flanges are installed so that the open end is in the flow path, allowing ERCW to flow freely. All welds will be made in accordance with proper codes. The pipe segments are then refilled, pressure tested and a leak check is performed. The ERCW design flow rate will then be verified after reopening the supply and return headers.

1. Design Basis Accidents (DBA)

   Performance of the ERCW system is evaluated through analysis of the design basis accidents and other events in FSAR Chapters 6, 9, and 15 (LOCA or other HELBs inside containment). For all other design basis events there is no interaction with the ERCW system (i.e., steam generator tube rupture, single reactor coolant pump locked rotor, fuel handling accident, RCC assembly ejection). ERCW supply to other systems will not be affected. The new configuration of the ERCW system will operate the same as the old configuration (EPM-PTC-120594 R5, ERCW System Pressure Drop Calculations).

2. Credible Failure Modes of Proposed Activity

   The credible failure modes associated with this modification is a loss of pressure boundary integrity of Train A of the ERCW system. This activity will not affect ERCW pressure boundary due to a hose break since only the portion of the ERCW system that normally serves the Unit 1 Reactor Building has non-seismically qualified piping (flexible chilled water hoses). The rest of the system is isolated and the temporary connections have been evaluated to assure the integrity of the ERCW system.

SA-SE Number: WBPLMN-98-125-1 **Implementation Date:** 04/14/1999

The portion of the ERCW piping system within the boundaries of piping analysis problems N13-67-09A and N3-67-23A (shown on piping isometric drawings 47W450-209, 47W450-248, and associated DCAs) has been evaluated for the conditions identified in WBPER981116 and PER 98-013305-000, respectively. The configuration in existence prior to implementation of DCN M-39972-B has been evaluated, and stresses remain within operability criteria limits defined in WB-DC-20-33. During installation of the WBPER981116 and PER 98-013305-000 design changes, the stresses will be maintained within that operability criteria through sequencing of the support modifications per DCN M-39972-8 implementation requirements. After installation of those modifications, compliance with the design basis criteria (e.g., WB-DC-40-31.7, WB-DC-40-31.9, and WB-DC-40-31.2) will be assured.

3. Other Events

The replacement of permanent ERCW piping with tees and associated flanges and other equipment, which are passive components, has been evaluated and no adverse effect on any other events was found.

SA-SE Number: WBPLMN-98-127-1

Implementation Date: 4/29/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50088-A	Outage Related Changes

Description and Safety Assessments:

This safety evaluation addresses changes to systems 02 (Condensate System), 40 (Station Drainage) and 67 (ERCW). These changes are intended to enhance outage related activities such as: 1) performance of a hydrostatic test for the Condenser after the Condenser Tubes are replaced (Reference DCN M-38974-A, Appendix H), this modification while not directly performing the hydro allows for the option of rapid fill/draining of the condenser, thus allowing plant personnel to expeditiously perform the hydro required by DCN M-38974-A. 2) Installation of Tee and Isolation valve for the discharge of coolant (System 67) for Station Air Compressors will allow the Station Air Compressors to remain in service while valves are being replaced further downstream of valve 1-ISV-067-0638 (Reference DCN W-39951-A) by routing compressors coolant discharge through new tee and valve 0-ISV-067-0638 and attaching temporary piping/hose to allow coolant to drain to existing Turbine Building floor drains/trenches where coolant will be processed. Administrative controls are in place to ensure that dedicated CST volume (≈200,000 gallons) are in place to supply the AFW as required. No changes are being made to any operating conditions. Flooding by means of a line break for the condensate system is not increased since the modifications performed use the same type of installation (materials and supports) as original installation and an isolation valve (normally closed) in conjunction with a blind flange arrangement will ensure minimum leakage wherever permanent plant features are installed. Furthermore, this system when temporarily aligned to either fill or drain the condenser will be monitored as an integral part of the post condenser tube replacement hydrostatic test and any abnormal leakage will be isolated. When the Station Air Compressors coolant (ERCW) is temporarily aligned to Turbine Building floor drains/trenches, to allow operation of the Station Air Compressors, this flow which is normally routed to RCW discharge will temporarily be routed to the Turbine Building sump station where flow (≈ 160 gpm) will be processed. This modification to the ERCW discharge will allow non-safety related Station Air to remain available for plant use, if Station Air is lost due to this modification no system, relied upon to perform a safety-related function during an event, will be adversely impacted since Station Air performs no Safety-Related function either directly or indirectly.

The proposed changes as described above will not increase the likelihood of the design basis accidents occurring. These changes will not increase the dose to the public analyzed in the UFSAR Chapter 15.5. No change will occur to the radiological consequences of accidents analyzed in the UFSAR Chapter 15.5. 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. The credible failure modes for the systems affected by these changes, have been evaluated against the accidents identified in the FSAR and concluded they do not introduce a failure pathway different from those identified and evaluated in the FSAR accidents. The applicable accidents and the equipment served by the malfunction pathways will be introduced which have not previously been evaluated and identified. The bases of the Technical Specifications have been reviewed for determining if any margins of safety are affected by these documentation changes. No margin of safety is identified in the bases section of the Technical Specifications which could be reduced by these changes.

SA-SE Number: WBPLMN-99-001-0

Implementation Date: 07/28/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number S-39962-A FSAR Change Package Number 1572	Updated FSAR Review - Sections 11.2, 11.3, and 11.5

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the Updated FSAR Review/Verification Program. Specifically addressed are UFSAR Chapter 11.2, 11.3 and 11.5 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 and documents revised as appropriate by DCN number, which partially implements the corrective action. Most changes to the UFSAR are minor in nature 'documentation-only' 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. However, a few changes do after the operational characteristics of the systems.

The changes included:

1. Revised the section to clarify that the High Crud Tanks (HCTs) which normally contain High Crud, Low Conductivity (HCLC) waste, may be processed to the Waste Disposal System (WDS) if the HCLC waste exceeds the ODCM limits and requires treatment. The Low Crud, High Conductivity (LCHC) waste from the Neutralization and Non-Reclaimable Waste Tanks can not be processed to WDS if the waste exceeds the ODCM limits and requires treatment. This waste should be processed to a vendor.

2. Revised the section to clarify that the HCTs which normally contain HCLC waste, may also contain LCHC waste if additional capacity is required. The HCTs would then be processed by a vendor if the tank was above the ODCM limits.

3. Revise the design pressure from 150 to 375 psig, and temperature from 1200° F to 1400° F.

4. Deleted the hydrogen (H_2) portion of the Waste Gas Analyzer.

5. Clarified that one (1) compressor is in automatic mode, not continuously operating.

5. Clarified that the drains from laboratory are routed to the Floor Drain Collector Tank (FDCT).

6. Deleted that the Cask Decontamination Collector Tank (CDCT) receives liquids from FDCT or Tritiated Drain Collection Tank (TDCT).

All document changes have been evaluated for plant operability during the review process and found not to affect the physical plant. The proposed "documentation-only' changes to the UFSAR and the design documents will not increase the likelihood of the design basis accidents occurring. These changes to the UFSAR and the design documents do not effect physical changes to the plant, nor do they involve any plant procedures. These "documentation-only" changes will not increase the dose to the public analyzed in the UFSAR Chapter 15.5. 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

SA-SE Number: WBPLMN-99-001-0 ***Implementation Date:*** 07/28/1999

by these changes, have been evaluated against the accidents identified in the FSAR and concluded they do not introduce a failure pathway different from those identified and evaluated in the FSAR 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.

SA-SE Number: WBPLMN-99-003-0

Implementation Date: 03/25/1999

Document Type:	***Affected Documents:***	***Title:***
Work Order	WR/WO 98-014847-000	Nitrogen gas bubbling into the Steam Generator tubes during RCS draindown to midloop.

Description and Safety Assessments:

During the RCS draindown for approach to midloop, the water in the steam generator tubes does not drain due to the U-tube configuration and the siphon effect. When the vacuum at the top of the tubes is broken - when the pressurizer level reaches slightly below the top of the hot leg - the outsurge of water causes instrument fluctuations and momentary instrument instability. This is undesirable as the unstable instrument readings are similar to those of an abnormal perturbation. This work order allows the orderly introduction of N_2 into the steam generator lower head area and the N_2 will then bubble into the steam generator tubes and allow for an orderly draindown. It is the intent to displace the water in the tubes in such a manner as to match the displaced water with the pressurizer draindown. This means the pressurizer level will be maintained at an elevation range of 35% to 20% throughout the N_2 injection. Operations will establish the draindown flow rate at 60 GPM and El. 745.2 (25%), to match any imbalances with the N_2 injection and maintain the pressurizer water level at approximately an elevation of 35% to 20%. It is expected that matching a compressible fluid (N_2) volume with an incompressible fluid will cause a mismatch resulting in an indication of lowering fluid (liquid) level in the pressurizer. Calculation MDQ 1068-990013 supports the volume and pressures of the N_2 required to achieve a near match with the pressurizer water level and the steam generator water level until the water elevation is in the lower elevation of the steam generator tubes.

The injection of molecular Nitrogen into the reactor coolant at no power conditions does not constitute any permanent plant configuration changes. The nitrogen injection enhances the approach to midloop but does not occur at a very reduced coolant level in the pressurizer. Injection of Nitrogen into the RCS does not increase the probability or consequences of accidents or equipment malfunctions, does not create the possibilities of new accidents or malfunctions, and does not affect the margin of safety as defined in the Technical Specifications. Additionally, as no permanent plant changes are involved, no mass or thermal inputs of consequence are involved, nitrogen does not constitute a radioactive source, and there is not a ASME Code Section XI concern, it is concluded that no unreviewed safety question is involved.

1. Termination of N_2 injection at any time can be accomplished by an AUO stationed at the rotameter station.
2. Termination of N_2 injection at any time by the main control room by closing valve 1-FCV-63-64.
3. An abnormal mass of input of N_2 could be detected by the main control room as a rise in pressurizer level.
4. The time limitations imposed, together with the self-limiting effect on the N_2 injection system (orifice effect) acts as a flow totalizer, which prevents a uncontrolled mass input.
5. The reactor vessel head and the pressurizer are open to the PRT atmosphere which prevents fluid suppression in the reactor vessel.
6. The NPSH on the RHR pumps is always in the positive PSIG range which greatly reduces the possibility of nitrogen coming out of solution.
7. The elbow taps on the cross over legs have two class B isolation valve that could be used to shutoff a RCS leak should the alignment become such that one could occur.

Dual pressure regulators serve to limit the results of a single failure and small overpressurize excursions.

SA-SE Number: WBPLMN-99-004-1

Implementation Date: 02/12/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	Drawing Deviation (DD) 98-001 FSAR Figure 10.4-8 FSAR Figure 10.4-28	Drawing Deviation

Description and Safety Assessments:

This safety evaluation addressed the discrepancies given in Drawing Deviation (DD) 98-0081, which eliminate the differences in pressure and temperature given on the different lines, the 8" lines from the main feedwater pumps, A and B, low load bypass to the condenser, and the 4" line stepped up to an 8" from the standby pump low load bypass to the condenser.

The requested change is to determine the correct temperature and pressure data and identify it properly on the flow diagrams. These documentation only changes as described do not impact any of the design basis accidents or credible failure modes evaluated in the FSAR nor do the changes impact any events previously evaluated.

The proposed "documentation only changes" do not interfere with any safety related equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This change does not alter the design basis for any, component important to safety; therefore, the change does not increase the probability of occurrence of an accident as described in the FSAR.

SA-SE Number: WBPLMN-99-012-0

Implementation Date: 03/30/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN M-40023-A	Diesel Generator Air Tank Relief Valve Setpoint

Description and Safety Assessments:

DCN M-40023-A revises the system description N3-82-4002 and drawings 17W586-1 and 1-47W839-1 to change the maximum allowable working pressure for the Diesel Start Air Tanks from 250 psig to 260 psig, which is the setpoint for the tank relief valve. The relief valve was set above the tank design pressure because the air compressor was overshooting due to the loadless start devise original pressure switch settings. The compressor was set to start at 200 psig and to cut off at 250 psig. When the relief valve was set at 250 psig, it would spuriously pop open, causing concern for valve and system degradation. To compensate for this condition, and due to drift on the upper and lower ends of range, the loadless start pressure switch setting was changed to start at 210 psig and cut off at 240 psig, and the relief valve was set at 260 psig. Tied with the loadless start devise pressure switch setting problem was the sequence of operation of the air compressor and air dryer. The manufacturer designed the loadless starter pressure switch to start the compressor which automatically starts the air dryer timer. Normally the air dryer cycle is completed in 5 minutes which is before the compressor stops. However, occasionally when the air dryer cycle is not complete, the circuitry is designed such that with the dryer contact in, the loadless start pressure switch is bypassed and will hold the compressor on and the pressure in the air tank will slightly exceed 250 psig, enough to cause spurious actuation of the relief valve if it is set around 255 psig.

The use of the Section VIII Code equations utilizes a design pressure of 260 psig and does not consider the UG-125 (c) Code requirement which states that pressure-relieving devices shall limit pressure from rising more than 10% above the maximum allowable working pressure (this includes the set pressure tolerances (see UG-133 (f) of ± 3%, 3% accumulation, and ANSI/ASME OM-1-1981 ± 3% set pressure drift). This is because the governing code for this portion of the system, ANSI B31.1, paragraph 102.2.4.13, allows 20% overpressure for 1% of the operating time. As explained before, the only way this piping can be pressurized above 250 psig, for sustained periods of time, is if the air tank becomes inadvertently isolated while the compressor remains on. Even so, using a pressure of 286 psi (260 X 1.10) in the above Section VIII equations shows a minimum required shell & head plate thickness of 0.248 inch and remains conservative compared to the minimum actual wall thickness measured of 0.332 inch with 0.084 inch allowed for erosion/corrosion allowance. Additionally, the air tank manufacturers U-1A Data Sheet states the vessel was hydroed at 375 psig.

The standby diesel generator system serves as the plant emergency standby ac power source. It is designed, installed, and tested to requirements necessary to assure its availability. Its sole purpose is for emergency power in case of a loss of offsite power (LOOP). Redundancy for single failure is provided by maintaining four DGs in ready condition for automatic start. This system performs a primary safety function which is to provide power (upon loss of the Preferred Power Source) to plant components required to assure fuel design limits and reactor coolant pressure boundary design conditions are not exceeded and to assure core cooling and vital functions are maintained for postulated DBEs. The following design basis accidents have been evaluated for impact: Condition II - Faults of Moderate Frequency, - Loss of Normal Feedwater, Condition III - Infrequent Faults, - Small Break Loss of Coolant Accident, Complete Loss of Forced Reactor Coolant Flow, and Single Rod Cluster Control Assembly Withdrawal at Full Power; Condition IV - Limiting Faults: Large Break Loss of Coolant Accident, Major Rupture of a Main Steam System Pipe, Major Rupture of a Main Feedwater Pipe, and Steam Generator Tube Rupture. Review of the above accident analyses indicates rerating of the Diesel Start Air Tanks from 250 psig to 260 psig is a detail not specifically addressed in the Chapter 15 accident analyses, and therefore does not affect any UFSAR Chapter 15 fault or operational transient evaluations.

SA-SE Number: WBPLMN-99-012-0 ***Implementation Date:*** 03/30/1999

Rerating the Diesel Start Air Tanks has no effect on the probability of an accident beyond that previously identified as part of the failure modes analysis, i.e., if either one of two sets of cranking systems fails to crank the engines, then the duplicate air start system on the other engine in the diesel generator set is capable of providing 100% cranking power for both engines in the diesel generator set. The subject documentation change will not change the frequency class of any accident evaluated in the FSAR to a higher frequency class and will not adversely affect the ability of the affected systems or equipment from performing their intended safety function. Therefore, the proposed activity will not increase the consequences of an accident previously evaluated in the FSAR.

Rerating the Diesel Start Air Tanks will not increase the likelihood of a radiological release or have any adverse radiological impact on the affected systems or equipment as a result of any accident. The subject changes will not impede access to vital areas of the plant, hamper actions required to mitigate any accident, or cause an increase in onsite or offsite dose which might adversely affect compliance with the requirements of 10 CFR 20 or 10 CFR 100. Therefore, the proposed activity will not increase the consequences of an accident previously evaluated in the FSAR.

The Technical Specifications Bases, UFSAR, and NRC SERs have been reviewed to determine if the subject documentation change reduces any margins of safety and no margin of safety was identified that might be reduced by these changes. However, based on ASME Section VIII calculations, the design stresses in the vessel at 260 psig remain below the Code allowable stresses and the proposed rerating of the Start Air Tanks to 260 psig does not reduce the tank manufacturers original margin below the Code allowable stress because his calculations are based on an nominal SA-515, Grade 70 plate thickness of 5/16 inch (0.03125 inch) but the rerated calculations are based on actual plate thicknesses which ranges from 0.332 inch to 0.398 inch. Therefore, the proposed activity does not change any operating points, any analysis assumptions or results, or any setpoints in excess of any identified acceptance limits.

Therefore, based on the preceding evaluation, it has been determined that implementation of the subject documentation change: will not create the possibility of a new type of accident or equipment malfunction not previously analyzed in the FSAR 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 FSAR, 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 change does not involve an unresolved safety question and is acceptable from a nuclear safety perspective.

SA-SE Number: WBPLMN-99-013-0

Implementation Date: 04/05/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50165-A	Containment Venting
FSAR	FSAR Change Package 1592	

Description and Safety Assessments:

The primary containment pressure, during normal operation, increases gradually because of instrument air bleeding from pneumatic actuators and leaking from compressed air lines, thus requiring periodic venting of the containment atmosphere into the annulus to maintain the containment-to-annulus pressure differential within the Technical Specification 3.6.4 limits. PER 99-00946-000 documented that the method of venting the containment; i.e., into the annulus, from where the air is exhausted by the annulus vacuum control (AVC) fan to the Auxiliary Building (AB) exhaust stack with no filtration, is contrary to the FSAR Section 11.3 statements.

DCN D-50165-A facilitates the installation of two 100% redundant filter trains, containing HEPA and charcoal filters, in the existing 8-inch containment pressure relief line, downstream of the outboard containment isolation valve FCV-30-37, for continuous venting of the containment atmosphere into the annulus through the newly added Containment Vent Air Cleanup Units (CVACUs). In order to vent the containment air into the Annulus through this filtered path, the 8-inch line, previously connected to the 24-inch containment purge exhaust duct, is disconnected from the 24-inch line. The connection at the 24-inch line is capped, and the CVCAUs are installed in the 8-inch line after the line size is reduced to 4-inches. The discharge ports of the CVACUs are open to the annulus, from where the AVC fans discharge the Annulus air into the suction-side ductwork of the Fuel Handling Area (FHA) exhaust fans. Since the discharge side of the FHA exhaust fans are connected to the AB exhaust stack, this air is then released to the outdoors while being monitored by the AB exhaust monitors. The CVACUs and the 4-inch piping, connecting them to the 8-inch line, are located in the annulus between the El. 716.0 and El. 724.0. The total length of 4-inch piping involved is approximately 60 feet.

This change allows the continuous filtered venting (100 cfm or less) of containment air into the annulus, during normal operation. Calculations determined that the annual routine gaseous releases will result in routine offsite dose impacts that are within the limits of 10 CFR 50 Appendix I criteria. Additional calculations have been generated, and/or revised, to determine the effect of continuous venting on the equipment EQ evaluations. Minor impacts, such as reduction of qualified life values for the valves FSV-30-40 and -37, and EQ Binder revisions to document them, are identified. Any minor inspection requirements for compliance with the current EQ binder requirements, on the these two normally-energized solenoid valves are included in the DCN. No impacts on other safety-related equipment are identified. The only safety-related system, the Containment Isolation (CI) system, which is directly involved with this change has been reviewed, and no adverse effects identified to the failure modes of the CI valves FCV-30-40 and -37 since the CI logic is unchanged. Therefore, it is acceptable for these two CI valves to be normally open, and the containment to be continuously vented, during Modes 1-5.

In addition, the DCN changes the laboratory testing frequency of the containment purge air cleanup Units (ACU)s charcoal adsorbers from 720 hours (per Reg. Guide 1.52) to 18 months (per Reg. Guide 1.140). This change will simply allow the purge air ACU charcoal adsorbent, to be tested to the requirements of RG 1.140, during normal operation, without affecting the ACUs RG 1.52 compliance for Mode 6 during which the filter train serves its only safety function. Since the purge air system is not required to perform any safety-related function during normal operation; the adsorber is justified

SA-SE Number: WBPLMN-99-013-0 **Implementation Date:** 04/05/1999

to be tested to the requirements of RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-water-cooled Nuclear Power Plants" during modes 1 -5. The ACUs are required to be operable during the first 5 seconds of a fuel handling accident, until the containment isolation valves close. Therefore, they will continue to be tested to the requirements of RG 1.52, "Design, Testing, and Maintenance Criteria for Post-Accident Engineered Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" in order to serve this safety function. In-place testing of the purge air ACUs adsorbers to the RG 1.140 requirements, during Modes 1-5, and meeting the applicable requirements of RG 1.52 for Mode 6 will not have any adverse impact on the purge air system safety functions.

There are no failure modes associated with these changes.

Although the continuous containment vent requires that the CI valves FCV-30-40 and -37 be normally open during Modes 1-5, this is acceptable. The Chapter 15 accident analysis were reviewed for three open purge lines (one 24-inch 50 -open supply, one 24-inch 500-open exhaust, and one 8-inch wide-open vent), during a LOCA. As documented by the calculation, this configuration is bounded by two 24-inch wide-open purge lines. Therefore, as previously evaluated, the new configuration has no impact on offsite dose and ECCS performance. Since the use of the 8-inch vent line is prohibited, during Mode 6, the consequences of a fuel handling accident, as previously analyzed, are unaffected.

Testing the purge air ACUs adsorbers in accordance with RG 1.140, during Modes 1-5, does not degrade the capability of the ACUs to perform their intended safety function in a fuel handling accident, since the dose calculations take no credit for the ACUs in a LOCA, and the ACUs will be tested/qualified to the applicable requirements of the RG 1.52, before entering Mode 6. This change is not associated with the protective features used to detect and mitigate the effects of any other event. The equipment associated with this DCN do not interface with any other 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 of, 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. 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 gaseous effluent release pathway as defined in ODCM. No new potential single failures of existing components will occur as a result of this DCN. 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 FSAR.

SA-SE Number: WBPLMN-99-013-0 ***Implementation Date:*** 04/05/1999

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. Minor revisions to the ODCM limits for releases from the Gaseous Waste System have been made in accordance with the infinitesimal increases predicted in the revised dose calculations, as a result of these changes.

This EDC does not affect the design basis for any system that is important to safety. No new potential single failure of existing components has been anticipated to occur. This change will not 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 3.6.3. 3.6.4, 3.9.8, or 5.7.2.14.

There is no potential impact on the SER.

Therefore; based on the above evaluation, implementation of this change:

- will not create the possibility of a new type of accident or equipment malfunction not previously analyzed in the FSAR;
- will neither increase the probability nor the consequences of an accident or equipment malfunction already evaluated in the FSAR;
- 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 changes do not involve an unresolved safety question and are acceptable from a nuclear safety perspective

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

Implementation Date: 04/16/1999

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration Procedure Change	TACF Number 1-99-8-41 SOI-41.05 FSAR Figure 10.4-37	Steam Generator Wet Layup System

Description and Safety Assessments:

TACF 1-99-8-41 will revise flow diagram 1-47W862-2 to add a blank plate in the 4-inch TVA Class G piping downstream of the double isolation valves 1-ISV-41-584 and 1-ISV-41-583. Currently these valves are leaking by and presenting a personnel hazard. The valves are located in steam generator loop Number 4. Valve alignment will be reversed from that shown on the flow diagram for 1-ISV-41-583 and 584, which shows the position for normal layup operation, but is not the alignment for power operations. By this TACF, all three valves will be closed with the blanking plate installed until the next outage when the valves leakage will be repaired and the system and drawings returned to normal configuration. Since these valves are normally closed during power operations (1-TTV-41-585 is normally open during power operations), it is acceptable to block this flow path. No design basis accidents are affected by this modification. The blanking plate being installed is in a class G line which is already isolated by two closed isolation valves, and is an additional means of isolation. The blanking plate is a passive means of isolation and it's unlikely failure could result in damage to the wet layup pump and/or associated valves, but would have no impact on nuclear safety.

The Steam Generator Wet Layup System is non-safety related and is not used to mitigate the consequences of an accident. This system is used for the wet layup of the steam generators during an extended outage, such as a refueling outage, and the design configuration of double isolation valves off the Main Feedwater piping helps insure the integrity of the steam generator secondary side piping. Since the temporary installation of the stainless steel blanking plate downstream of the double isolating valves with a telltale drain valve and the combination closure of all three valves will be restored before it is needed for wet layup, it will not impact the system design function. However, revision of UFSAR Figure 10.4-37 necessitates revision of plant procedure SOI-41.05 regarding the change in valve alignment shown on the revised Figure 10.4-37. This temporary modification will help protect the non-safety related wet layup equipment during power operation. The Steam Generator Wet Layup System is not required to operate during or after an accident so the changes will not increase the consequences of an accident. No increase in the probability of an accident or malfunction of equipment important to safety is created as a result of this modification. This modification will not reduce the margin of safety as defined in the Technical Specification or create an accident of a different type than any previously evaluated in the FSAR. Because the system is non-safety related and the modification will have no impact on the system function or method of performing it's function there is no unreviewed safety question.

SA-SE Number: WBPLMN-99-035-0

Implementation Date: 06/09/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change FSAR	EDC E-50201-A FSAR Change Package 1594	Moderate Energy Line Break Documentation Deficiencies in the Turbine Building

Description and Safety Assessments:

DESCRIPTION:

EDC E-50201-A is required to implement some of the Corrective Actions and Recurrence Controls and was written to document deficiencies in regards to the Moderate Energy Line Break (MELB) evaluation of piping located in the Turbine Building. These deficiencies occurred or were discovered when preparing and issuing DCN M-39816-A.

The actions being performed by EDC E-50201-A to implement some of the PER Corrective Actions and Recurrence Controls involve:

1.0 Revising FSAR Section 10.4.5.3 to state that the seals located in the penetrations between the Turbine Building and Service Buildings and the Auxiliary and Control Buildings are sealed for flooding up to the 711.0 elevation because of the Moderate Energy Line Break flood in the Turbine Building, which results from a rupture of the CCW piping.

2.0 Revising the Environmental Drawings to resolve the discrepancy between note FF on 47E235-22 Revision 4 and note NN added by DCN S-37263-A. Note FF stated "MELBs are not addressed ..."; however, Note NN was added by DCN S-37263-A to address MELBs resulting from a Condenser Circulating Water (CCW) system line break. Note FF is being, revised by EDC E-50201-A to delete the statement about MELBs not being addressed on that drawing

3.0 Revising Design Criteria to acknowledge and address MELB type concerns associated with a CCW line break in the Turbine Building. This information is already contained in other design documents. It is being added to state that the CCW is the bounding system for MELBs in the Turbine Building in the system design criteria. This is being done to facilitate future design efforts involving the CCW as part of the recurrence controls.

4.0 Revising calculation, "Turbine Building Flooding Due To A Break In The Condenser Circulating Water System" to clarify the MELB flood level requirements in the Turbine Building.

There are no nuclear safety accident scenarios or new failure modes involved with the design_changes.

The CCW system is not included in the evaluation of any accident in the FSAR. The design and operational requirements of the CCW have not been changed. Various documents have been revised to better clarify the design of the facility in relationship to a Moderate Energy Line Break involving the Turbine Building portion of the CCW.

The mechanical and electrical penetration seal design requirements specified and the Turbine Building flood level specified in FSAR subsection change the Nuclear Regulatory Commission (NRC) Safety Evaluation Report (SER) evaluation in Subsection 10.4.5 of the SER for line breaks in the Turbine Building portion of the Condenser Circulating Water System. However, the changes which were described and evaluated above do not adversely affect the safety of the plant. EDC E-50201-A is not actually making the changes to the design requirements for MELB flooding in the Turbine Building, it is only revising documentation to more clearly specify those requirements.

SA-SE Number: EDC E-50201-A ***Implementation Date:*** 06/09/1999

Therefore:

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR will not increase as a result of the activities in EDC E-50201-A.
- A possibility of an accident or malfunction of a different type than those previously evaluated in the FSAR will not be created as a result of the activities in EDC E-50201 -A.
- A margin of safety as defined in the basis for any Technical Specification will not be reduced as a result of the activities in EDC E-50201 -A.

SA-SE Number: WBPLMN-99-037-0

Implementation Date: 05/08/1999

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF-1-99-12-61	Removal of power to glycol isolation valves.

Description and Safety Assessments:

The refrigeration portion of the Ice Condenser System contains a glycol expansion tank connected to the glycol header. The expansion tank allows expansion and contraction of the glycol when the fluid is heated or cooled. The expansion tank has two separate glycol flow paths containing isolation valves (1-FSV-61-109 and -118) to isolate the tank in the event of a leak in the glycol piping. The isolation valves automatically close upon receipt of a low-low level signal from expansion tank level instrumentation. Each isolation valve has a check valve bypass to permit flow into the expansion tank in the event the isolation valves are closed. The expansion tank and associated piping and valves are located inside containment and are isolated by separate containment isolation valves subsequent to an accident. Both isolation valves have experienced several internal coil shorts which cause a constant alarm in the Main Control Room (MCR). Currently, no replacement coils are available. To eliminate the MCR alarm, the power to both valves is to be removed by pulling the fuses which causes the valves to fail closed. Consequently, if the system glycol temperature increases, the glycol will expand through the isolation valve bypass check valves into the expansion tank. If the system glycol is then cooled and contracted, the glycol in the expansion tank cannot flow back into the system piping due to the closed isolation valves. This TACF removes the internals from valve 1-FSV-61-118 to permit makeup of glycol to the system (flow out of the tank) in the event of cooling and contraction of the glycol. If a leak occurs in the glycol piping after implementation of the TACF, the expansion tank cannot be isolated automatically and thus the glycol in the tank (340 gallon) could leak into containment. Westinghouse has determined the loss of glycol (2000 gallon) into containment subsequent to an accident is not a safety concern. Consequently, removal of the isolation valve internals by this TACF is acceptable from a nuclear safety standpoint.

The glycol expansion tank isolation valves do not perform a primary safety function. In addition, as stated above, Westinghouse has evaluated leakage of approximately 2000 gallons of glycol into the containment sump and Emergency Core Cooling Systems (ECCS) subsequent to a Loss of coolant Accident (LOCA) and Main Steam Line Break (MSLB) and determined that this transient is not safety significant. Inability to isolate the expansion tank subsequent to a glycol piping leak and an accident could result in the 340 gallon of glycol in the tank leaking to the containment sump and ECCS. This small amount of glycol leakage is bounded by the Westinghouse evaluation and is therefore, not safety significant. Removal of the isolation valve internals will reduce the valve mass, reducing possible seismic loads in the event of a seismic event. Consequently, the Category IL(B) position retention seismic qualification of the valve will be maintained. Thus, the valve will not fall on and damage other safety related equipment as a result of this change. The changes due to this TACF do not result in an increase in the probability or consequences of accidents currently evaluated in Chapter 15 of the UFSAR, does not result in different accidents or malfunctions than evaluated in the UFSAR, and does not reduce the margin of safety as defined in the Technical Specification Bases.

SA-SE Number: WBPLMN-99-040-0

Implementation Date: 06/18/1999

Document Type:	***Affected Documents:***	***Title:***
Procedure Change	Procedure Number MI-0.045, Revision 4	Transporting the Spent Fuel Pool Cask Loading Pit Gate.

Description and Safety Assessments:

This document evaluates Revision 4 to MI-0.045 with respect to the handling activities to support transporting the Spent Fuel Pool (SFP) cask loading pit gate to the refueling floor (south of the fuel pool) to facilitate installation of the gate seal. The SFP cask loading pit gate will be installed as necessary to separate the cask loading pit from the spent fuel pit. Current safe load paths (SLP) contained on 44W411-5 do not include a path to the SFP cask loading area or to the gate location, and these lifts are not currently provided for in MI-0.045.

The new SLP for the SFP cask loading pit gate is due west through the gate slot to the centerline of the cask loading area, then due south to existing SLP's for the refueling floor.

There is no direct effect on any plant system or equipment required to mitigate any design basis accident due to handling the cask loading pit gate. Since this change has no direct effect on any system or component required to mitigate design basis accidents, there is no credible failure mode associated with this activity.

This activity has no direct effect on any system or component important to safety or that is associated with any Technical Specification. Additionally, potential indirect effects have been evaluated as not adversely effecting any system or component. Accidents and malfunctions of equipment have been evaluated with no adverse effects identified. Therefore this activity does not involve an unreviewed safety question

SA-SE Number: WBPLMN-99-040-1

Implementation Date: 06/25/1999

Document Type:	***Affected Documents:***	***Title:***
Procedure Change	Procedure Number MI-0.045, Revision 5	Spent Fuel Pool Gate/Cask Handling

Description and Safety Assessments:

This document evaluates Revision 5 to MI-0.045 with respect to the handling activities for the Legal Weight Truck (LWT) shipping cask and Spent Fuel Pool (SFP) cask loading pit gate. The revision is required to support handling of the LWT cask (~52,000 lb.) between the Auxiliary Building Railroad bay, the cask decontamination facility, and the Cask Loading Pit. It is also being revised to support transporting SFP cask loading pit gate (~4,000 lb.) to the refueling floor (south of the fuel pool) to facilitate installation of the gate seal. The SFP cask loading pit gate will be installed as necessary to separate the cask loading pit from the spent fuel pit. Current safe load paths (SLP) contained on 44W411-5 do not include a path to the SFP cask loading area or to the gate location, and these lifts are not currently provided for in MI-0.045.

The new SLP for the cask is at the centerline of the cask loading pit (west of the fuel pool) and runs directly north-south between the cask loading area and the existing SLP north of the fuel pit.

The new SLP for the Cask Loading Pit gate is due west through the gate slot to the centerline of the cask loading area, then due south to existing SLP's for the refueling floor.

There is no direct effect on any plant system or equipment required to mitigate any design basis accident due to handling the LWT cask or the Cask Loading Pit gate. Since this change has no direct effect on any system or component required to mitigate design basis accidents, there is no credible failure mode associated with this activity.

This activity has no direct effect on any system or component important to safety or that is associated with any Technical Specification. Additionally, potential indirect effects have been evaluated as not adversely effecting any system or component. Accidents and malfunctions of equipment have been evaluated with no adverse effects identified. Therefore this activity does not involve an unreviewed safety question.

SA-SE Number: WBPLMN-99-041-0

Implementation Date: 08/03/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC Number E-50345-A FSAR Change Package Number 1598 TRM Change Package Number 99-006 TRM Revision 18	Documentation changes required to put the containment Control Rod Drive Mechanism (CRDM), Lower Compartment (LC) and Upper Compartment (UC) coolers' handswitches in "A-Auto."

Description and Safety Assessments:

Problem Evaluation Report (PER) 99-004873-000 documents that conflicting requirements exist for the CRDM coolers (CRDMCs) and the Lower Compartment Coolers (LCCs). The PER cites that LCO 3.8.1 requires these coolers to restart following a blackout. The Technical Requirement, TR 3.8.4, requires that these coolers trip when a Phase B signal occurs and remain de-energized when Phase B is reset. The System Operating Instruction (SOI) standby alignment for these coolers requires the control switch to be placed in PULL A-P AUTO. With the control switch in PULL A-P AUTO, the coolers will restart, when the DG re-energizes the Shutdown Boards. With the control switch in PULL A-P AUTO, the coolers will trip, when a Phase B signal occurs, but these coolers will re-energize when the Phase B is reset, thus violating the Technical Requirement.

As part of the corrective actions for PER 99-004873-000 and due to concerns raised while analyzing the PER for the effect of equipment that is potentially submerged, or sprayed by containment spray (as a result of various DBEs, which could result in a Phase B containment isolation signal inside primary containment, and subsequently re-energized after a containment isolation Phase B, signal is reset), the decision was made to place the CRDMC, LCC, and Upper Compartment Coolers (UCC) control switches in "A-AUTO", during Modes 1 through 6 (i.e., after starting a cooler, the switch reverts to A-AUTO) and the redundant coolers' switches are also left in A-AUTO.

EDC E-50345-A revises System Description Document N3-30RB-4002 to require that the handswitches for the standby CRDMCs, LCCs, and UCCs be placed in the A-AUTO position, after starting or while in standby, during Modes 1-6. With the control switches in the A-AUTO position, and if a containment isolation Phase B signal, or a Loss of Offsite Power (LOOP) occurs, CRDMCs, LCCs, and UCCs will trip, and each will require operator action to restart. In addition, when in A-AUTO, each standby cooler will lose its ability to auto-start upon failure of an operating cooler (e.g., loss of airflow, loss of fan differential pressure, etc.); thus, requiring operator action to start it. This change will enable the containment coolers to remain de-energized when a Phase B containment isolation signal is reset; or upon restoration of onsite electrical power after a LOOP or other non-LOCA DBEs; thus, effectively precluding automatic re-sequencing of the CRDMCs and the LCCs onto the Diesel Generators (DGs).

As stated above, the changes being made to System Description no longer require the CRDMCs and LCCs to be automatically loaded onto the Diesel Generators. In support of this change, the System Description and Design Criteria are being revised to indicate that these loads are no longer part of the automatic DG loading sequence. The "Demonstrated Accuracy Calculation for DG Sequencing Relays" is being revised to remove the time delay relays, and the associated "Setpoint and Scaling Documents" are being voided because calibration of these relays is no longer required. The "DG Loading Analysis" is also being revised to remove the loads from the automatic loading sequence.

SA-SE Number: WBPLMN-99-041-0 **Implementation Date:** 08/03/1999

The submergence calculation is being revised to remove a statement that indicated the CRDMCs, LCCs, and UCCs had to be secured before a safety injection signal is reset. TR section 3.8.4 is also being revised to indicate these loads are no longer required to be secured before a safety injection signal is reset.

The EDC is adding a note to the Control Logic and Schematics for the CRDMCs, LCCs, and UCCs to indicate that the control switches are required to be in the A-AUTO position during Modes 1 through 6.

The change requires operator action to, start/restart the coolers, as needed. The change also clarifies that all four LCC fans are started (only two required to be operational), within 1.5 to 4.0 hours following a MSLB inside primary containment, to assure adequate mixing of air between lower containment and all dead-ended compartments, to prevent hot spots from developing. Starting all four LCC fans, which is currently specified in the UFSAR, will preclude the consequences of a single failure. Other related UFSAR Sections and tables, and other affected design basis documents, are being revised to clarify that all four LCCs are started within 1.5 to 4.0 hours following a MSLB.

A part of UFSAR Change Package Number 1598 deletes the description of the acceptance criteria for testing of potentially submerged equipment from Section 8.3.1.2.3. The acceptance criteria has not changed and still exist in the Technical Specifications and the Technical Requirements Manual. This information in the UFSAR was considered excessive detail and therefore, was deleted.

This change does not introduce any new failure modes for the LCCs which are the only safety-related containment air coolers, nor does it affect the previously evaluated failure modes. The failure modes and effects analysis for the LCC system is being revised to reflect the effect of this change.

Consequently, the change will not introduce any new accidents or malfunctions previously evaluated, nor will it increase the frequency or the consequences of accidents and malfunctions currently evaluated. Therefore, this change does not result in a unreviewed safety question.

SA-SE Number: WBPLMN-99-043-0

Implementation Date: 08/03/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC Number E-50249 FSAR Change Package 1597	Documentation changes and changes in total number of fuel assemblies.

Description and Safety Assessments:

This safety evaluation addresses document changes identified as part of the corrective actions and addresses the change in the total number of fuel assemblies considered in the SFP for each of the off-load scenarios. Specifically addressed are the document changes associated with Engineering Design Change (EDC) E-50249-A. These include UFSAR section 9.1.3.3.1 "Availability and Reliability", Table 9.1-1 "Spent Fuel Pool Cooling and Cleanup System Design Parameters" and System Description "Spent Fuel Pool Cooling and Cleaning System." These changes are significant enough to be considered as technical in nature and necessary to provide consistency between the UFSAR, system description document, and supporting calculation WBNOSG4-239. All changes are marked-up in either FSAR change package number 1597, or on copies of affected pages of the system description included in EDC 500249-A. Design calculation WBNOSG4-239 "Thermal Hydraulic and Sparger Replacement Analysis for the Watts Bar Spent Fuel Pool" was revised to include Holtec International report HI-961474, revision 5 which provided the design input reflected in both the system description and UFSAR.

A PER was initiated in response to Holtec International's letter dated November 3, 1998 which notified TVA of errors in their "TBOIL" computer program used to perform time-to-boil calculations for Watts Bar during the Spent Fuel Pool (SFP) re-rack effort. The program error resulted from a code logic error that causes the decay heat load in the SFP (from stored spent nuclear fuel) to be non-conservatively under-predicted during a full core off-load following a normal refueling concurrent with a loss of SFP cooling scenario. Calculated values previously reported for a normal full core off-load case were unaffected. Under-prediction of the decay heat load resulted in a corresponding non-conservative over-prediction of the time-to-boil and time for the fuel pool water level to reach an elevation 10 ft above the stored fuel (if no make-up water is assumed available), and an under-prediction of the maximum boil-off (vaporization) rate and average heat-up rate. A Holtec letter dated December 9, 1998 determined that after consultation with representatives from all of the affected utilities and based on Holtec's own internal reviews, the error was not reportable under the provisions of 10 CFR 21. Subsequently, Holtec submitted revision 5 to report HI-961474 containing the correct values. The report was incorporated into revision 2 of WBN calculation WBN-OSG4-239

The maximum vaporization rate due to boiling is reported in UFSAR section 9.1.3.3.1 and the average heat-up rate for a full core off-load following a normal refueling is reported in UFSAR Table 9.1-1. These values were changed from 64 gpm and 9.20°F/hr to 70.2 gpm and 10.20°F/hr respectively. The maximum vaporization rate and time for the fuel pool water level to reach an elevation 10 ft above the stored spent nuclear fuel are reported in section 5.1 of system description N3-78-4001, while the average heat-up rate is reported Table 2.2-2 of the system description. These values were also updated to agree with revised calculation WBNOSG4-239. The system description discussion of the time for the fuel pool water level to reach an elevation 10 ft above the top of the stored spent nuclear fuel was clarified to note that this assumes no make-up water is added. The calculation analyzes both cases with and without make-up water available.

SA-SE Number: WBPLMN-99-043-0 ***Implementation Date:*** 08/03/1999

The total number of fuel assemblies considered in the SFP for each of the off-load scenarios, as reported in Table 2.2-2 of system description N3-78-4001 and in Table 9.1 -1 of the UFSAR was changed from 1835 to 1873. This value agrees with the current revision of the thermal hydraulic analysis (WBN-OSG4-239) and conservatively bounds the maximum licensed storage capacity of 1610 assemblies. Using this value results in a conservative calculation of the total decay heat load in the SFP.

There are no failure modes associated with this change.

These changes do not impact any accidents evaluated in the UFSAR. As described in this section of the safety evaluation, these changes do not affect the operation of any safety related equipment/systems and no credible failure modes are created or changed.

This safety evaluation addresses document changes identified as part of the corrective actions and addresses the change in the total number of fuel assemblies considered in the SFP for each of the off-load scenarios.

Specifically addressed are changes associated with engineering design change EDC E-50249-A. These include revision to UFSAR section 9.1.3.3.1 "Availability and Reliability", UFSAR Table 9.1-1 "Spent Fuel Pool Cooling and Cleanup System Design Parameters" and System Description N3-78-4001 "Spent Fuel Pool Cooling and Cleaning System". All changes discussed within this safety evaluation are "documentation only" in nature and have no impact on the design basis of the plant or its operational configuration. These changes have been evaluated for plant operability during the review process and found to not affect the physical plant or existing procedures. These changes will not increase the dose to the public analyzed in UFSAR chapter 15. 1. No change will occur to the radiological consequences of accidents analyzed as a result of these changes. The proposed changes do not result in any changes to the plant design basis or the physical plant. The credible failure modes for the SFPCCS is not affected by these changes and these changes have been evaluated against the accidents identified in the SAR. It is concluded that they do not introduce a failure pathway different from those identified and evaluated in the SAR accidents. The applicable accidents and the equipment served by the SFPCCS have been reviewed against these documentation changes and no new malfunction pathways will be introduced which have not previously been evaluated and identified. 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: WBPLMN-99-045-0

Implementation Date: 08/17/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	ECN Number E-50220-A; FSAR Change Package 1601	Documentation Changes - FSAR Section 6.2 (WBPER980808)

Description and Safety Assessments:

Engineering Document Change (EDC) E-50220-A is required to implement the changes addressed as a part of WBPER980808. The SA/SR/SE will address the documentation only changes to show that the WBN 1 Essential Raw Cooling Water (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 loss of coolant accident (LOCA) conditions for a period of 30 days.

The FSAR (Section 6.2.3.1 and Table 6.2.4-3) and System Description (N3-67-4002) will be changed to document the results of the WBN Calculation, WBN-MEB-MDQ1067-980012, System 067 ERCW Containment Isolation Type E-Leakage. 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 will be added to the system description. The limiting conditions established by the calculation will put the bounding leakage rates on the ERCW piping penetration primary containment as follows:

- Total leakage rate per train for an inoperable train A or B is 150 scfh.
- Total leakage rate for an operable Upper Compartment Vent Cooler 1A return line is .4 scfh.
- Total leakage rate for an operable Upper Compartment Vent Cooler 1C return line is .4 scfh.
- Total leakage rate for an operable Lower Compartment Vent Cooler 1A return line is 2 scfh.
- Total leakage rate for an operable Lower Compartment Vent Cooler 1C return line is 35 scfh.

There are no nuclear safety accident scenarios or new failure modes involved with the design changes in EDC E-50220-A.

EDC E-50220-A is required to implement the changes addressed as a part of WBPER980808. The safety evaluation addresses the documentation only changes to show that the WBN 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 loss of coolant accident (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 n the SAR will not increase as a result of the documentation only changes proposed in EDC E-50220-A.

SA-SE Number: WBPLMN-99-045-0 ***Implementation Date:*** 08/17/1999

- The probability of an accident or malfunction of a different type previously evaluated n the FSAR 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.

Considering this, the proposed change does not involve an unreviewed safety question.

SA-SE Number: WBPLMN-99-047-0

***Implementation Date*:** 07/10/1999

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF Number 0-99-2-26 R1	Temporary Alteration to ensure that the diesel driven fire pump will have a flooded casing when called upon to start.

Description and Safety Assessments:

This Safety Evaluation is provided for TACF 0-99-2-26, Revision 1, which specifies the installation of a check valve in the suction supply line for the Diesel Engine driven fire pump. The Diesel Engine driven fire pump draws its supply from the cooling tower basin. The low level setpoint for the cooling tower basin water level is slightly below the elevation of the top of the pump casing. During low cooling tower basin water levels, the pump casing is not completely flooded as required for proper pump operation and performance. The valve is installed to prevent flow from the pump casing to the cooling tower basin during times when the cooling tower basin water level is low and the pump is not running. This will ensure that the Diesel engine driven fire pump will have a flooded pump casing when called upon to start.

This change will have no affect on any accident already evaluated. The previously evaluated accident is an Appendix R fire and the ability of the Plant to extinguish the fire and safely shutdown. This change is to ensure the availability of the Diesel Engine driven fire pump, and therefore could not reasonably cause a failure of the fire protection system or its components in performing its design function.

The addition of the check valve in the suction supply for the Diesel engine driven fire pump does not constitute a unreviewed safety question. The addition of the check valve ensures that the pump casing remains flooded during minimum cooling tower basin water levels, and that the pump is available to start if required. There is no effect on the operation or response of any FSAR or FPR described systems or components.

SA-SE Number: WBPLMN-99-048-0

Implementation Date: 09/16/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number D-50309-A	Diesel Driven Fire Pump Check Valve

Description and Safety Assessments:

This Safety Evaluation is provided for DCN D-50309-A, which specifies the installation of a check valve in the suction supply line for the Diesel engine driven fire pump. The diesel engine driven fire pump draws it's supply from the cooling tower basin. The low level setpoint for the cooling tower basin water level is slightly below the elevation of the top of the pump casing. During low cooling tower basin water levels, the pump casing is not completely flooded as required for proper pump operation and performance. The valve is installed to prevent flow from the pump casing to the cooling tower basin during times when the cooling tower basin water level is low and the pump is not running. This will ensure that the diesel engine driven fire pump will have a flooded pump casing when called upon to start. A priming line from the Raw Service Water system is also provided to fill the pump casing if necessary. The DCN will install a valve vault in the yard to provide access to the suction line check valve for future maintenance. In addition, the DCN will provide an alternate material for the suction screens that is more corrosion resistant than the currently installed galvanized screen. The DCN also provides for the installation of a strainer in the sense line to the control panel to prevent clogging of the 3/32" pressurization orifices in check valves 0-CKV-26-3165 and 3166.

This change will have no affect on any accident already evaluated. The previously evaluated accident is an Appendix R fire and the ability of the Plant to extinguish the fire and safely shutdown. This change is to ensure the availability of the diesel engine driven fire pump, and therefore could not reasonably cause a failure of the fire protection system or it's components in performing it's design function.

The modification described in DCN D-50309-A for the diesel engine driven fire pump ensures that the pump casing remains flooded during minimum cooling tower basin water levels, and that the pump is available to start if required. There is no effect on the operation or response of any systems or components described in the FSAR or Fire Protection Report. The proposed change to the diesel engine driven fire pump will not increase the likelihood of any design basis event occurring. The ability of the pump to supply water in case of a fire will be improved as a result of this change. This change will not prevent the plant from containing or extinguishing an Appendix R fire or achieving safe shutdown in the event of an Appendix R fire. Since the ability of detecting and controlling a fire is unchanged, radiological releases resulting from a fire are unchanged. The modification does not create the possibility of a different type of accident than what has been previously evaluated, nor does it introduce any new initiator or failure. The modification will not cause the system to be operated in a manner different than originally designed. This change does not involve or impact Tech Spec components, therefore, the margin of safety is unaffected. The modification described in DCN D-50309-A for the diesel engine driven fire pump does not constitute a unreviewed safety question.

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

Implementation Date: 08/05/1999

Document Type:	***Affected Documents:***	***Title:***
Problem Evaluation Report (PER)	PER 99-001697-000	Identification of inconsistencies regarding the hydrostatic test pressures specified on a mechanical flow diagram.

Description and Safety Assessments:

Problem Evaluation Report (PER) 99-001697-000 identified that the safety injection system flow diagram 1-47W811-1 contained inconsistencies regarding the hydrostatic test pressures specified for certain pipe segments. The PER determined that under a documentation-only DCN, S-21239-A, the original hydrostatic pressures were incorrectly revised. Upon further investigation, the PER identified that hydrostatic test pressure information was previously added to other flow diagrams to assist the Construction group with testing of completed piping systems during construction of the plant. Now that the plant is operating, the test pressures or description of test pressures on the flow diagrams may not accurately reflect hydrostatic test pressure requirements for an operating plant. Because piping codes and procedures are in place to govern the pressures at which piping is hydrostatically tested for the operating plant, the pressures specified on the flow diagrams are considered historical data. As a part of the corrective action plan of PER 99-001697-000, form NEDP 3-3, "Request for Administrative Change to Drawings", is generated to clarify that the hydrostatic test pressures and designations on mechanical flow diagrams (1-47W800-series) are historical information and no longer maintained as design output.

The change to the flow diagrams does not affect the design basis of the plant, nor does it physically affect the plant or its operation. Some of the requirements changed once the plant was licensed and became operational, and the information shown on the 47W800-series drawings may no longer be applicable. The changes to the flow diagrams are documentation-only to clarify how the hydrostatic test pressures are used.

Due to the nature of the administrative change to the drawings, no design basis accidents or credible failure modes apply.

The safety evaluation does not evaluate the inconsistencies documented in the PER. It only evaluates the administrative changes being made to the 1-47W800-series drawings.

As a part of the corrective action plan for PER 99-001697-000, form NEDP 3-3, "Request for Administrative Change to Drawings", is generated to clarify that the hydrostatic test pressures and designations on mechanical flow diagrams (1-47W800-series) are historical data and no longer maintained as design output. The PER identified inconsistencies regarding the hydrostatic test pressures specified for certain pipe segments on safety injection system flow diagram 1-47W81 1-1. Documentation-only DCN S-21239-A incorrectly revised hydrostatic test-pressures. The evaluation of the inconsistencies was performed under the scope of the PER and is not addressed in this SA/SR/SE. Further investigation of the PER revealed that other flow diagrams contained hydrostatic pressures. Some of the requirements for pressure testing changed once the plant was licensed and became operational, and the information shown on the 47W800-series drawings may no longer be applicable. This could present future problems since piping codes and TVAN procedures are now in place to govern the pressures at which piping is hydrostatically tested and may be different from the information shown on the drawings. This administrative change will reduce the likelihood that incorrect test pressures are used to hydrostatically test the piping.

The administrative change to the mechanical flow diagrams (1-47W800 series) also includes flow diagrams that are figures in the FSAR. The change to the flow diagrams does not modify the design of the plant, modify the plant physically, or affect how it is operated. The changes to the flow diagrams are documentation-only changes to clarify how the hydrostatic test pressures are used. Actual hydrostatic testing of the piping systems will not be affected because there are established piping codes and TVAN specifications and procedures that govern such testing and what test pressures are used.

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

SA-SE Number: WBPLMN-99-051-0

Implementation Date: 08/12/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC Number E-50358A	Documentation Change to Resolve Drawing Deviation

Description and Safety Assessments:

Drawing Deviation(DD) 99-0055 identified a discrepancy between drawings in reference to the TVA safety classification of the instrument tubing attached to HVAC duct in the Diesel Generator Building. Drawing 17W91 0-1 classifies the tubing attached to the HVAC duct as TVA class G and references construction specification N3M-868 as the basis, whereas 47W600-168 classifies the tubing as class C. However, Construction Specification (now changed to Engineering Specification) N3M-868, "Field Fabrication, Assembly, Examination, and Test for Piping Systems", clearly states that its intent is not to "...apply to vent and duct systems". The specification defers to other specifications and drawings for the piping and component classifications. Design Criteria WB-DC-40-36.1 classifies the duct in the diesel rooms as class Q or S, Seismic Category 1. Upon further investigation, WB-DC-40-36, "The Classification of Piping, Pumps, Valves, and Vessels', classifies instrument tubing lines attached to Seismic Category 1, Class Q and S, HVAC boundaries as class C. Drawing 47W600-168, Revision R, 'Electrical Instruments and Controls' is the design output document that shows the design and installation of the tubing and is the more appropriate location for noting the classification of the tubing. This classification is more in line with the classification of the instruments that the tubing is connected to. Furthermore, the classification noted in 47W600-168 is more stringent than that of 17W910-1. The instrument tubing was installed per the requirements of 47W600-168. As a result, Engineering Document Change (EDC), E-50358-A, is issued to delete note 3 from TVA drawing 17W910-1, Revision K.

The deletion of the note is a documentation only change, and no field work is required by this change. Because this is a documentation only change to delete erroneous information, no design basis accidents or credible failure modes are identified.

DD 99-0055 identified a discrepancy between drawing 17W910-1, Revision K and 47600-168, Revision R regarding the TVA classification of instrument tubing. Drawing 17W910-1 classifies the tubing attached to the HVAC duct as TVA class G and referenced construction specification N3M-868 as the basis, whereas, drawing 47W600-168 classifies the tubing as TVA class C. After further inspection, it was discovered that 1) construction specification (now Engineering Specification) N3M-868 does not classify components: 2) design criteria WB-DC-40-36 and WB-DC-40-36.1 classify the tubing and duct. respectively, which was in conflict with the note, and 3) drawing 47W600-168 is the appropriate drawing to contain such information instead of the HVAC drawing since it details design, installation, and configuration of tubing and instruments.

EDC E-50358-A is issued to delete the instrument tubing classification information from 17W910-1, Revision K that is specified in other design output documents and on other design output drawings. The actual classification of the tubing is more stringent than the one incorrectly specified on 17W910-1. The change to the drawing does not change the design of the plant, physically modify any equipment in the plant, or affect how the plant is operated. This change prevents any classification errors associated with the erroneous note. Therefore, the change does not constitute an unreviewed question.

SA-SE Number: WBPLMN-99-064-0

Implementation Date: 10/08/1999

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Change Package 1609	Updated FSAR Review - Containment Penetration Configuration

Description and Safety Assessments:

The details of the steel containment vessel (SCV) penetration and the shield building penetration, which are shown in FSAR Figure 6.2.4-16, "Type XVIII, Ice Blowing Line" were found to be incorrect. These penetrations are used for the ice blowing line in the Ice Condenser System during Mode 6 operation.

FSAR Figure 6.2.4-16 shows the SCV penetration with a blind flange on the inboard and outboard side of the penetration with a single O-ring for each flange. The penetration actually has a single blind flange on the outboard side of the penetration with double O-rings (Reference drawing 1-48W406 Revision 5, FSAR Figure 6.2.4-23, and FSAR Table 6.2.4-1). The use of a double O-ring flange on the outboard side of the penetration meets the design basis requirements and ensures containment integrity.

FSAR Figure 6.2.4-16 shows the Shield Building penetration with a plate bolted to the Auxiliary Building side of the Shield Building. In actuality, the configuration consisted of a pipe sleeve penetrating the wall with a blind flange bolted to each end (Reference drawings 47W462-7 Revision H, 47W470-2 Revision K and 47W470-3 Revision E). The use of a blind flange connection on both the inside and outside of the Shield Building penetration meets design basis requirements for Shield Building integrity. Figure 6.2.4-16 is referenced in the FSAR text on page 6.2.4-10. The description on pages 6.2.4-10 was also incorrect since it matches Figure 6.2.4-16. Pages 6.2.4-15 and -16 also required revision since the Shield Building Penetration is listed as a Class A Leakage Path (Auxiliary Building to Annulus). The penetration is actually a Class C Leakage Path (Outdoors to Annulus).

FSAR Change Package 1609 was generated to correct the discrepancies in the FSAR.

This evaluation concluded that there are no accidents, which have been evaluated in the FSAR, that may be affected by the proposed activity. The required changes are "documentation changes" which resolve the FSAR discrepancies concerning the configuration of the Ice Blowing line penetrations through the Steel Containment Vessel and the Shield Building. Figure 6.2.4-23 correctly shows the Ice Blowing line through the Steel Containment Vessel. TVA drawings 47W462-7, 47W470-2, 47W470-3, and 1-48W406 show the "As-Constructed" configuration of the penetrations. Considering this, there are no new credible failure modes added or changed as a result of the changes required to correct the FSAR.

SA-SE Number: WBPLMN-99-124-0

Implementation Date: 05/04/1999

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN Number D-50217-A	Processing Condensate Polishing Demineralizer System Low Crud and High Conductivity Waste

Description and Safety Assessments:

In the Condensate Polishing Demineralizer System (CPDS), in the unlikely event of a significant primary to secondary leak, the low crud and high conductivity (LCHC) waste may become highly radioactive (i.e., above the ODCM discharge limits) such that it could not be released to the environs. The waste could have been processed, with the current plant configuration, to vendor supplied equipment in the Auxiliary Building (AB) railroad bay. This is not the best approach because there is a potential for creating radioactive crud traps in several hundred feet of pipe from the Turbine Building (TB) to the AB railroad bay. Therefore, the CPDS must have the ability to process this waste to vendor supplied equipment (portable evaporator or shipped off site for processing and disposal) in the TB railroad bay.

DCN 50217-A adds a new connection to process the CPDS low crud and high conductivity (LCHC) waste for processing by a vendor. The vendor equipment will be supplied as necessary to process this waste and the equipment will be located in the TB railroad bay. This DCN revises the configuration control diagram (CCD) 1-47W838-3, physicals, and bills of material to provide this connection.

There are no failure modes associated with this change.

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 significant primary to secondary leak. The primary to secondary leak would prevent the LCHC waste from being discharged to the Cooling Tower Blowdown (CTB) line. This DCN adds a new connection to provide the capability of processing this waste in vendor supplied equipment, and does not change any accident analysis previously evaluated in the UFSAR. 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. This DCN does not affect the design basis for any system that is important to safety. This new connection does not add any different type of equipment failure modes that would prevent the existing equipment to operate as designed. 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.

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: WBPSPE-99-001-0

Implementation Date: 02/22/1999

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Change Package 1589	Editorial changes - FSAR Sections 11A and 15 through 15.5

Description and Safety Assessments:

This FSAR revision is the result of a general review of Chapters 11A, 15, and 15A. The changes are principally editorial in nature with only a few minor technical changes. The review was performed by comparison of the chapter content against existing WBN analyses, the Accident Analysis Parameter Checklist, and NSSS vendor documentation. The majority of Chapter 15 was updated prior to Cycle 2 to reflect the latest Westinghouse analyses supporting Condition I-IV events. These analyses covered small LOCA, large LOCA, and non-LOCA events to use consistent input assumptions and to reflect experience from Cycle 1 operation. Changes to these FSAR sections were therefore found to be minimal. Modifications to the text have been made to assure clarity with the analyses of record. Chapter 15.5 documenting WBN radiation analyses contained numerous typographical errors, which were corrected.

Three sections are subjects of Conditions Adverse to Quality. These include section 15.2.14.2 on Inadvertent Safety Injection (the current analysis is under revision by Westinghouse) where the Positive Displacement Pump is tagged out of service to lower the ECCS flowrate following this event, section 15.4 on design basis hydrogen production where Zn and Al inventory tracking is being confirmed bounding, and section 15.4 on containment purging (LOCA backpressure impact) where purging limits are being clarified in the design basis. The revisions to the these sections are consistent with the current design basis:

<u>11.A.4</u>
Editorial - eliminate duplicative and semantically incorrect wording.

<u>15.0</u>
No changes.

<u>15.1</u>
Editorial - Section 15.1.2.3, change "Technical Specification" to "Core Operating Limits Report." This information was relocated during the Merits Technical Specification project.

Editorial - Section 15.1.2.3, delete references to Figures 4.4-1 and 4.4-2 for consistency with Cycle 2 revisions which eliminated these figures from Chapter 4.

Editorial - update Reference 22 revision level to Revision 7 for consistency with Cycle 2 analysis. Note that this reference is not dated by Westinghouse but was entered into the TVA document retrieval system in April 1997 and received in the preceding month.

Editorial - correct typographical errors in least significant digits for two isotopes presented in Table 15.1-5.

SA-SE Number: WBPSPE-99-001-0 ***Implementation Date:*** 02/22/1999

15.2
Editorial - Section 15.2.3.1, changed reference to the use of moveable incore neutron instrumentation. This is method in "plant" but not "operating" procedures.

Editorial - Section 15.2.5. 1, changed permissive P8 to approximately 48% from 50% to more accurately match the design and remainder of the FSAR.

Editorial - Section 15.2.6, clarify that the analysis is normally reserved for N-1 loop operation.

Editorial - correct exponent errors on axes for Figures 15.2-8, 9, 10.

Editorial - correct Figures 15.2-42 a, c which were revised in Cycle 2 and transmitted in the associated FSAR change package (Change Number 1473) but missed in the editorial update from that change package.

Editorial - change Reference 1 date of publication.

Technical - clarify that the latest inadvertent safety injection analysis does not assume an operating PD pump for the pressurizer overfill case. This analysis was revised prior to Cycle 2 as a result of a Westinghouse Nuclear Safety Advisory Letter (NSAL). The PD pump was tagged out-of-service and a corrective action document was written following final information from Westinghouse. This was done to maximize operator action time to respond to the worst case event. This analysis has since been impacted by another NSAL which is currently being evaluated. Any changes from that NSAL, which deals with pressurizer heater assumptions, will be transmitted in a future change package.

15.3
Editorial - Section 15.3.4.1, eliminated reference back to Chapter 7.

15.4
Editorial - minor correction to References 22 and 23.

Technical - Table 15.4-3 correction to Zn and Al inventory to match WBN analysis. This table is for the inputs to the TVA analysis not the Westinghouse analysis. Both analyses may require revision following the verification of current ZN and Al inventories in a corrective action document. The TVA analysis results already presented in this section are consistent with the revised inputs.

Editorial - clarification to Section 15.4.1.1.5 on containment purging impact on large break LOCA. The clarification is that the bounding assumptions by Westinghouse encompass more than the purge lines currently discussed.

Editorial - delete Table 15.4-13 which is not used or referenced elsewhere in the FSAR. Textual information was removed in a prior amendment and the table was inadvertently retained.

SA-SE Number: WBPSPE-99-001-0 ***Implementation Date:*** 02/22/1999

15.5

Editorial - Section 15.5.1, delete reference to iodine partition in condenser. This credit is not discussed in the analysis of record.

Editorial - Section 15.5.1, correct 0-8 hour dilution factor should be 2-8 hour dilution factor (0-2 hr already described). Clarify both realistic and conservative analysis described in Table 15.5-2.

Editorial - Section 15.5.2 and 15.5.3, correct numerous typographical errors in equations and definitions. These changes match the analyses of record. They are generally self evident but due to the complexity of the equations they may not be immediately determined.

Editorial - Section 15.5.4 and Table 15.5-1, delete reference to the iodine partition in condenser. This credit is not discussed in the analysis of record.

Editorial - Table 15.5-4, correct exponent for Xe-135m to match the analysis of record. 480 Curies should be 48 Curies as verified by the analysis. A transcription error in moving the analysis table to the FSAR evidently occurred.

Editorial - Table 15.5-6, correct descriptive information associated with Auxiliary Building holdup time. The holdup time is assumed to be effective following the initial drawdown of the Auxiliary Building. This occurs after 4 minutes rather than the 10 minutes currently listed.

Editorial - Table 15.5-20, correction to the elemental iodine filter efficiencies for the Reactor Building purge from 99% to 90% which was used in the analysis.

Editorial - Delete Figures 15.5-3 and 15.5-4, these figures are no longer referenced or discussed in the FSAR. Associated text was removed in a prior amendment and the figures were retained in error.

Of the changes discussed above, only three had technical substance requiring further review and evaluation. These included the elimination of a reference to the "operating" procedure for use of movable neutron instrumentation to aid the diagnosis of mispositioned control rods. The operating procedures reference a plant surveillance instruction for the use of movable detection. This change does not effect the ability of the operators to respond to the event and does not result in a disconnect with the existing analyses The tagout of the PD pump and the reanalysis of the pressurizer overfill during an inadvertent S1 maintained compliance with the FSAR analysis criteria, maximized operator action time, is documented in a condition adverse to quality, has a corrective action assigned, and is therefore acceptable. It is noted that subsequent analyses are underway as a result of a subsequent Westinghouse Nuclear Safety Advisory Letter (NSAL). Any resulting changes will be documented in a future FSAR change package. The final change was to document the inconsistency of the galvanized Zn weight shown in the FSAR table documenting the design basis hydrogen analysis with the actual TVA analysis. This was a typographical error but results in a much larger Zn inventory that previously reported in the FSAR. This change is acceptable since the corresponding TVA analysis was performed using this value, demonstrated acceptable hydrogen recombiner performance, and the new values are consistent with the results of the TVA analyses reported in the FSAR. This analysis is also the subject of a separate condition adverse to quality related to the tracking of the total quantities with respect to design changes and may result in future changes to the TVA or Westinghouse hydrogen analyses.

All of these changes are therefore consistent with regulatory requirements and do not constitute an unreviewed safety question.