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

APR 2 0 2004

10 CFR 50.59(d)(2)

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

Gentlemen:

In the Matter of)
Tennessee Valley Authority)

Docket No.50-390

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

Pursuant to 10 CFR 50.59(d)(2), this letter provides the Summary Report of the implemented changes, test, and experiments in which evaluations were performed in accordance with 10 CFR 50.59(c). The enclosure provides a summary of the evaluations for the Updated Final Safety Analysis Report Amendment 4 provided under separate cover dated April 20, 2004, and includes other evaluations performed during the period from August 10, 2002 to March 16, 2004.

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

Sincerely,

P. L. Pace
Manager, Site Licensing
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Enclosure
cc: See page 2

IE47

APR 20 2004

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ENCLOSURE

WATTS BAR NUCLEAR PLANT UNIT 1
10 CFR 50.59 SUMMARY REPORT
ABBREVIATIONS

%	Percent
AHU	Air Handling Units
ALARA	As Low As Reasonably Achievable
AOI	Abnormal Operating Instruction
BIT	Boron Injection Tank
CLA	Cold Leg Accumulator
COLR	Core Operating Limits Report
CRDM	Control Rod Drive Mechanism
CSS	Containment Spray System
CVE	Condenser Vacuum Exhaust
DCN	Design Change Notice
ECCS	Emergency Core Cooling system
EQ	Equipment Qualification
FCV	Flow Control Valve
FHA	Fuel Handling Accident
GDC	General Design Criteria
HO	Hold Order
HZP	Hot Zero Power
ISV	Isolation Valve
LBLOCA	Large Break Loss-of-Coolant Accident
LCC	Lower Compartment Coolers
LOCA	Loss-of-Coolant Accident
LCO	Limiting Condition for Operation
MOV	Motor-Operated-Valves
MSLB	Main Steam Line Break
ODCM	Offsite Dose Calculation Manual
PASF	Post Accident Sampling Facility
PMWS	Primary Makeup Water System
PWST	Primary Water Storage Tank
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFO	Refueling Outage
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SCADA	Supervisory Control, Annunciation, and Data Acquisition
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIP	Safety Injection Pump
SIS	Safety Injection System
SOC	Systems Operations Center
SRP	Standard Review Plan
SSC	Structure, System, or Component
TACF	Temporary Alteration Control Form
TCO	Trip Cutout
TPBARS	Tritium Producing Burnable Absorber Rods

UFM	Ultrasonic Flow Meter
UFSAR	Updated Final Safety Analysis Report
WO	Work Order

Safety Evaluation Number: WBPLMN-00-034-1

Implementation Date: December 23, 2003

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50528-B	Operation of 480V Board Room Cooling System During Tornado Warning

Description and Safety Assessments:

The DCN will allow the 480V board room air conditioning systems (both trains) to remain operable during a tornado warning, and to lock open and abandon in place tornado dampers. These dampers are associated with the supply air to the Train A air-cooled condensing units. Presently if a tornado warning is issued the dampers are required to be closed, by manual action from the main control room, per AOI and the system description. The AOI also requires that each of the 480V board room AHUs be shut off once a tornado warning has been issued. However, in performing these two actions, cooling is lost to the 480V Board and 125V Battery Rooms. Depending upon outside temperature conditions, internal room cooling loads, and the length of time the air conditioning systems are deenergized, area temperature limits specified in Table 3.7.5.1 of the Technical Requirements Manual could be exceeded. This could result in declaring the equipment served inoperable if the abnormal temperature limits are exceeded for the specific time periods.

This DCN will also lock open and abandon in place isolation dampers located in the exhaust of air-cooled condensing units. These dampers perform no technical functions, and locking them open will not only eliminate their potential failure modes, but also eliminate the required periodic maintenance performed on these dampers. These dampers were installed originally to prevent the possibility of any back flow of air through their associated air-cooled condensing unit if the unit is off and the pressurizing fan is running since both take suction from the same mechanical equipment room. However, the room is open to the outside atmosphere via a large louvered opening which will provide the path of least resistance for make-up air to the pressurizing fans. Therefore, back flow through the air-cooled condensing units and subsequent rotation of the fan in the reverse direction while the unit is not running will not occur.

None of the dampers are credited as pressure mitigating devices during a design basis tornado. The differential pressures determined by the calculation have been evaluated, and the results indicate there are no adverse consequences or effects to the structural adequacy of the Auxiliary Building concrete walls/slabs and roof. These dampers will no longer be active components, and their failure modes are eliminated. No new potential single failures of existing components will occur as a result of these changes. The proposed changes will not cause this system or any other system important to safety to fail to perform its functional requirements. The changes will not affect any equipment required for safe operation or shutdown. The margin of safety identified in the applicable technical specifications will not decrease as a result of the proposed changes. Based on the results of the safety evaluation, it is concluded that the proposed changes do not involve an unreviewed safety question and are acceptable from a nuclear safety standpoint.

Safety Evaluation Number: WBPLEE-00-056-1

Implementation Date: May 7, 2003

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50595-A FSAR Package 1668	Deletion of Automatic Flow to Radiation Monitoring System

Description and Safety Assessments:

DCN is being issued to delete calibration requirements for Barksdale flow controllers in the radiation monitoring system and change FSAR to delete discussion of automatic flow controls since the flow controls will be operated in the manual mode. This changes the method of establishing flow through the radiation monitors from automatic control to manual positioning of control valves.

Barksdale model flow controllers (flow switches) are used in the Radiation Monitoring system to provide automatic flow control in the sample stream for particulate radiation monitors. These flow switches have performed poorly and have proven to be unreliable. The automatic flow control function is being deleted from these monitors and the flow switches will be spared. Flow control will be accomplished by manual setting of the valve position during calibration.

There are three off-line particulate, iodine and noble gas monitors affected by this DCN change. These monitors are the Auxiliary Building Vent Monitor, which is required for compliance with Regulatory Guide 1.97 and the ODCM, Containment Lower Compartment Monitor, and Containment Upper Compartment Monitor, which are required for compliance with Regulatory Guide 1.45 and the Technical Specification to provide notification of a small leak in the Reactor Coolant System Pressure Boundary. The sample line pumping system for each monitor consists of two 100% capacity pumps in parallel. Each pump has a flow capacity of 10 ft^3/minute; the flow for each pump under the current design is controlled by an automatic Barksdale flow controller which operates a motor-driven valve on the suction side of the pump. The parallel motor-driven valve/pump configuration is down stream of a ball valve flow control manifold in the monitor sample stream. After this DCN change, each motor-operated valve may be manually positioned by hand switch operation during calibration of the monitor. Monitor sample stream flow of 10 ft^3/minute will be established by manually adjusting the motor-driven valve for the operating pump and / or manually adjusting the ball valves in the flow control manifold. Sufficient flow is provided to ensure compliance with Regulatory Guides 1.45 and 1.97, the Technical Specifications and the ODCM. Sample line flow is indicated by a flowmeter and pressure indicator at the monitor enclosure. If the flow drops below the value required for compliance, an alarm annunciates in the main control room. The setpoint for this alarm is not revised by this change document.

There are also four continuous air (airborne particulate) monitors affected by this DCN change. These monitors are the Spent Fuel Pool Monitor , Shipping Bay Monitor, Holdup Valve Gallery Monitor, and SI Pump Area Monitor. They are used for personnel protection and to assist the operator in the diagnosis of transients.

The change being implemented by the DCN and FSAR Change Package does not increase the probability of malfunction of equipment or accident as previously analyzed, does not increase the consequences of an accident or equipment malfunction as previously analyzed, does not create the possibility for different accidents or malfunctions of a different type than previously analyzed, and safety margins are not affected nor reduced. Deletion of the automatic control signal for positioning of the flow control valves on the suction side of the monitor sample pumps does not prevent the monitors from performing their intended functions. Proper monitor sample flow is established by manual positioning of the control valves during monitor flow instrumentation calibration. The DCN change does not affect how the affected monitor's low and high flow conditions are alarmed in the Main Control Room.

Consequently, this DCN and FSAR change does not involve an unreviewed safety question.

Safety Evaluation Number: WBPLEE-01-060-0

Implementation Date: ***December 19, 2002***

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-50965-A	Setpoints for Radiation Monitor 1-RE-90-112

Description and Safety Assessments:

Radiation monitors 1-RE-90-106 and -112 (when aligned to the lower compartment) monitor the containment atmosphere for particulate and gaseous radiation, and, in conjunction with the containment pocket sump level monitor, provide early warning of reactor coolant system pressure boundary leakage, as required by General Design Criteria (GDC) 30 of Appendix A to 10 CFR 50 and described by Regulatory Guide 1.45. The leakage detection systems must have the capability to detect reactor coolant pressure boundary degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus an early indication is necessary to permit proper evaluation of all unidentified leakage. While the radiation monitors are utilized to detect leakage of the RCS, the monitors do not perform a primary safety function and are not required for the mitigation of a design basis accident.

Because of differing backgrounds between the upper and lower compartments of containment, of both the particulate channels and the gas channels, it has been difficult to establish alarm setpoints for 1-RE-90-112 that will maintain the design basis when it is aligned to the lower compartment and still have meaningful setpoints without causing nuisance alarms when it is in its normal alignment, to the upper compartment.

This revision to the alarm setpoints for 1-RE-90-112 enables the monitor to comply with the design basis requirements while monitoring the lower compartment and still be able to adequately monitor the upper compartment. Therefore, the change in setpoints does not affect the ability of the monitor to perform its UFSAR described in design function.

The requirement to recalibrate the alarm setpoints of 1-RE-90-112 if it is aligned to the lower compartment while purge is running could lead to another malfunction if the recalibration if not done when necessary. In the event that the monitor is aligned to the lower compartment with purge running, and the setpoints are not adjusted, the alarm would not annunciate at the proper level of radiation. However, this malfunction or failure mode is enveloped by the already existing malfunction of the ratemeter's alarm bistable failure, which would prevent the high radiation signal annunciating in the main control room. There are no new failure modes that could arise from this change. Therefore there is no increased risk to nuclear safety and this change is considered acceptable.

Safety Evaluation Number: WBPLMN-03-042-0

Implementation Date: January 27, 2004

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN D-51416-A FSAR Package 1771 TRM-03-017	Primary Water Flow Setpoint

Description and Safety Assessments:

DCN D-51416-A revises the boron requirements for the RWST and CLAs as described in EDC E-50629-A. The boron concentrations supplied in EDC E-50629-A were for a full complement of TPBARs in the core. The new approach provides the RWST and CLAs boron concentration requirements corresponding to the number of TPBARs in the core. The number of TPBARs is limited to no more than 240. The number of TPBARs in the reactor core is contained in the COLR for each operating cycle. The appropriate design and licensing documents, including the FSAR are revised to reflect the new "single step" boron approach.

The proposed boron concentration values ensure that the post-LOCA accident sump boron concentration is sufficient to maintain subcriticality, dependent upon the number of TPBARs in the core. This change will minimize cost and reduce operational burden associated with the addition of large amounts of boron into these systems until the TPBAR loading requires the boron to support the accident mitigation functions. The boron in the primary water is normally operated automatically. This change will allow system flow to be operated in manual mode.

DCN D-51416-A revises the design as required for recalibration of instrumentation impacted by the changes in boron concentration. Boric acid concentration will increase by a single step as a part of modifications to implement the Tritium Production Program.

This change does not alter the methods of detecting or mitigating an Uncontrolled Dilution Event. Therefore, it is concluded that allowing the operator to control the primary water flow setting during automatic operation of the RCS makeup subsystem would have a minimal to no increase on the likelihood or frequency of occurrence of a malfunction or accident in the FSAR. This change does not affect equipment/components needed to detect or mitigate radiological consequences. Therefore, this change does not provide an increase in risk to nuclear safety and is considered to be safe.

Safety Evaluation Number: WBPLMN-03-047-0

Implementation Date: October 7, 2003

Document Type:	***Affected Documents:***	***Title:***
Design Change	DCN No. D-51488-A FSAR Package 1787	Generic Letter (GL) 89-10 Motor Operator Valves (MOV) Long Term Degradation

Description and Safety Assessments:

GL 89-10 MOV calculations for 1-FCV-063-0008 and 0011 show that adequate margin is not available when considering long term degradation. The operators on these MOVs was modified to regain needed margin.

This DCN is being issued to revise documentation to allow the re-gearing of specific valves in the population of the Generic Letter 89-10 Program. Calculations, system description and FSAR sections or tables are also being revised as required to reflect the stroke time change requirements conjunctional with the replacement gears.

These changes are required as a result of negative thrust margin when evaluating the minimum requirements considering long term degradation of the operator capabilities as required. A Problem Evaluation Report has also been generated in which this DCN will partially implement the corrective action.

The evaluation of the thrust margin of each valve in the GL 89-10 valve population and the risk associated with failure to operate determined the scope of the valves to be re-geared by this DCN. In order to gain additional thrust for valve operation, the stroke time was increased. The additional thrust is a change in the conservative direction and the increased stroke time is an adverse change to an operational parameter. The change to the valve stroke time is a change to an input parameter which does not directly or indirectly impact the frequency of the likelihood of occurrence of an accident or malfunction or consequences of an accident.

In conclusion, the evaluation shows that implementing this modification is acceptable from a nuclear safety standpoint and does not require NRC approval.

Safety Evaluation Number: WBPLMN-02-068-0

Implementation Date: March 5, 2003

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-51071-A FSAR Package 1725 WBN-TS-03-01	SGTR, FHA and Effluent Releases Updated

Description and Safety Assessments:

This evaluation was prepared for the evaluation of EDC 51071-A, FSAR Change Package 1725, and Technical Specification Bases Change TS-03-01. FSAR Change Package 1725 is being issued to revise FSAR sections and tables associated with the Steam Generator Tube Rupture Accident, Fuel Handling Accident, and Liquid and Gaseous Effluent Releases. EDC 51071-A is being issued to revise applicable system descriptions, design criteria, setpoint and scaling documents and design output calculations associated with changes to the Fuel Handling Accident calculation and Effluent Release calculations."

As a result of the revisions to the calculation for, "Control Room Operator and Offsite Does Due to an SGTR," revisions to assumptions, methodologies, and radiation dose results are necessary for the FSAR SGTR. The changes associated with the SGTR have previously been approved by the NRC as documented in Watts Bar Nuclear Plant Unit 1 - Technical Specification Change No. WBN-TS-01-012, RCS Specific Activity.

As a result of the revisions to the calculation for "Main Control Room Operator and Offsite Radiation Doses from a FHA," revisions to assumptions, methodologies, and radiation dose results are necessary for the FSAR FHA. The changes associated with the FHA have been previously approved by the NRC as documented in Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment to Irradiate Up to 2304 Tritium-Producing Burnable Absorber Rods in the Reactor Core.

As a result of changes associated with design radioactive liquid and gaseous releases as presented in the calculation for, "Design Releases to Show Compliance with 10CFR20," revisions are necessary to liquid and gaseous release tables. Changes associated with liquid effluents with the exception of the power level are evaluated under this 10 CFR 50.59 evaluation as a result of increases to the liquid release concentrations and design value basis change. Liquid and gaseous effluents are typically applied under 10 CFR 20. However, these changes are being conservatively evaluated under 10 CFR 50.59.

The increases in total liquid release concentrations do not affect any Chapter 6 or Chapter 15 accidents and thus do not increase the frequency or consequences of any accident malfunction. The concentration increases do not create a possibility for an accident of a different type or malfunction of equipment with a different result. Changing the UFSAR values for the design RCS concentrations to values is considered to be essentially the same and thus does not result in a departure from a method of evaluation used in establishing the design bases. Therefore this change does not provide any undue risk to nuclear safety.

Safety Evaluation Number: WBPLMN-02-044-0

Implementation Date: June 9, 2003

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-51278-A FSAR Package 1735	Condenser Vacuum Exhaust Portable Radiation Monitor

Description and Safety Assessments:

The UFSAR and supporting design documentation currently requires that a portable radiation monitor be placed in the CVE flow stream if the steam generator blow down and CVE radiation monitors are inoperable. The CVE monitors are inoperable prior to achieving a full vacuum in the main condenser due to excess moisture in the CVE. The design function of the CVE radiation monitors is to detect a steam generator primary to secondary system tube leak (one gpm in one hour), and to monitor normal and accident effluents. The design function of the portable monitor is to detect a steam generator primary to secondary system tube leak (one gpm in one hour). The current licensing basis per the ODCM requires manual sampling of the CVE when the CVE radiation monitor is inoperable during times after a full vacuum is established in the condenser. This change revises the UFSAR and design documentation to delete the requirement for the portable monitor and extends the ODCM CVE sampling requirements to the period of time prior to establishing a vacuum in the condenser. The portable monitor was initially included in the design because the CVE radiation monitor was considered operable during times prior to establishing a full vacuum in the condenser. The monitor was not operated, due to possible damage to the monitor from moisture in the CVE which exists prior to establishing a vacuum in the condenser. The CVE monitor is considered inoperable prior to achieving a vacuum in the condenser. Consequently sampling of the CVE under the ODCM methodology is acceptable and the compensatory portable monitor is no longer required. The ODCM, however, is changed to indicate sampling of the CVE is required prior to and after establishing a vacuum in the condenser.

The only accident that could be affected by this change is the SGTR. This change impacts the method of detecting a primary-to-secondary leak to provide operators an early indication of a SGTR. However, since this function is indication only and does not perform any isolation of mitigation functions, and does not affect the failure modes and effects analysis in the UFSAR, it does not increase the likelihood of occurrence of an accident or malfunction nor does it result in an increase in the consequences of an accident or malfunction. This change also does not create a possibility for an accident of a different type or a malfunction of equipment with a different result. Fission product barriers are not impacted and this change does not result in a change to the evaluation methodologies described in the UFSAR. In addition, in accordance with the approved licensing basis ODCM, manual sampling of the CVE is an acceptable method of assessing the CVE when the normal CVE monitors are inoperable after a vacuum is established in the main condenser. Extending the ODCM sampling methodology to the period of time prior to establishing a vacuum in the condenser is consistent with the current approved methodology.

Safety Evaluation Number: WBPLMN-02-062-0

Implementation Date: ***January 6, 2003***

Document Type:	***Affected Documents:***	***Title:***
Engineering Document Change	EDC E-51334-A, FSAR Package 1742, TS Bases Change 02-12	Containment Analyses Incorporates 10% Containment Spray Heat Exchanger Tube Plugging

Description and Safety Assessments:

The LOCA containment integrity analysis was revised to incorporate a 10% containment spray heat exchanger tube plugging margin. The mass and energy release rates with respect to the LOCA form the basis to evaluate the criteria presented in the SRP Section 6.2.1.3. The relevant requirements of GDC 50 and 10 CFR Part 50, Appendix K, are included.

This revised analysis results in a slight increase in the maximum peak containment pressure from 10.46 psig to 10.64 psig. The results of the LOCA integrity analysis using the 10% containment spray heat exchanger tube plugging margin show that the heat removal capability of the containment is sufficient to absorb the energy releases and keep the maximum calculated pressure below design limits. The WBN containment pressure acceptance criterion of 13.5 psig and the maximum allowable internal containment pressure of 15 psig, are maintained.

Incorporating a 10% tube plugging margin in the LOCA containment integrity analysis is not an initiator of an accident or a new malfunction, and no new failure modes are introduced. Therefore, the revised LOCA containment analysis does not result in an increase in the frequency of an accident or a malfunction previously evaluated in the UFSAR. The results of the LOCA containment analysis incorporating a 10% tube plugging margin do not create the possibility of an accident of a different type or a malfunction with a different result than previously evaluated in the UFSAR.

The LOCA containment integrity analysis incorporates an input parameter change only; the evaluation methodologies are not impacted. Incorporating a 10% containment spray heat exchanger tube plugging margin is the only input parameter change associated with the revised LOCA containment integrity analysis. There are no changes to the other assumptions. The mass and energy releases calculated for the LOCA containment integrity analysis are based on the assumption that offsite power is lost. The input radiation energy source term, determined from the core energy and decay heat, is not changed in the revised LOCA containment integrity analysis incorporating a 10% containment spray heat exchanger tube plugging margin. The results of this analysis have no impact on 1) the radiological dose consequences or 2) the fission product barrier. The increase in peak containment pressure is less than 10% of the margin between the existing peak containment pressure and the WBN containment pressure acceptance criterion of 13.5 psig. There is no increase in the peak containment temperature. There is no increase in the consequences of an accident or the consequences of a malfunction of equipment important to safety.

Safety Evaluation Number: WBPLMN-03-046-0

Implementation Date: September 25, 2003

Document Type:	***Affected Documents:***	***Title:***
Design Change	EDC E-51485-A FSAR Package 1786	Safety Analyses for Loss of Load/Turbine Trip Event and Main Steam Line Break

Description and Safety Assessments:

Westinghouse Nuclear Safety Advisory Letter, NSAL 03-1, describes the potential application of a non-conservative initial condition assumption in the safety analysis of the Loss of Load/Turbine Trip event. Westinghouse issued a letter to indicate that this issue affected Watts Bar. The use of a lower reactor vessel coolant average temperature at hot full power initial conditions results in a higher peak RCS pressure during the transient.

The results of the reanalysis of the limiting RCS pressure case for the Loss of Load/Turbine Trip event performed by Westinghouse shows an increase in the peak RCS pressure of 3 psi, resulting in a new peak RCS pressure of 2655.3 psia. However, this peak RCS pressure remains below the applicable maximum allowable RCS pressure limit of 2748.5 psia. Since, the applicable acceptance criterion for maximum allowable peak RCS pressure continues to be met, this issue does not result in a condition adverse to safety.

Westinghouse Nuclear Safety Advisory Letter, NSAL 02-14, advises Licensees of Westinghouse-designed plants to review cooldown procedures to ensure that the cold shutdown boration requirements prior to manually blocking the automatic SI signals (P-11) are met. Westinghouse issued a letter to WBN to communicate the clarified Westinghouse MSLB analysis assumptions.

The licensing basis MSLB event is conservatively analyzed at HZP, Mode 2, to bound all lower modes of operation. The Westinghouse MSLB analysis basis assumes that the SI system is available to mitigate the consequences of the MSLB event. Further, if the automatic SI signals are blocked, it is assumed the RCS is borated to ensure that the current licensing basis MSLB analysis is bounding. The design bases is being revised to ensure the criticality from a MSLB event could not result if the SI system were not available to mitigate the event by strengthening the requirement that the RCS be borated to cold shutdown conditions prior to blocking automatic safety injection signal. This activity does not result in any new or modified operator actions since procedure revisions will ensure boration requirements are met to preclude any operator actions as a result of a MSLB.

The current Loss of Load/Turbine Trip analysis has a significant margin to the RCS pressure safety analysis limit, 96.8 psi. A decrease of 3 psi in the margin to the safety analysis limit does not challenge the analysis limit nor invalidate the conclusions of the event. In addition, the penalty does not significantly impact the governing characteristics of the pressure transient. Therefore, the current assumed initial value for reactor vessel coolant average temperature in the analysis is acceptable and does not result in a design basis limit for a fission product barrier being exceeded.

Safety Evaluation Number WBPLMN-03-046-0 ***Implementation Date:*** 9/25/03

This change clarifies the requirement that the RCS be borated to cold shutdown conditions prior to blocking automatic SI signals. The current MSLB analysis assumes that automatic SI actuation is available through Mode 3, until the RCS is borated and the SI is blocked. Procedure revisions will ensure that the RCS is borated to cold shutdown conditions prior to blocking P-11. Therefore this change agrees with the current design functions that ensure maintenance of fission product barriers and does not result in an increase in the frequency or consequences of any accident or malfunctions of SSCs important to safety.

Safety Evaluation Number: WBPLEE-03-034-0

Implementation Date: March 16, 2004

Document Type:	***Affected Documents:***	***Title:***
FSAR	FSAR Package 1795	Watts Bar Hydro Automation Program.

Description and Safety Assessments:

The Watts Bar Hydro Plant Switchyard provides offsite (preferred) power from TVA's 161kV Transmission grid with two separate power circuits connecting to four (2 per line) 161-6.9kV Common Station Transformers at Watts Bar Nuclear Plant. Section 8.2 of the WBN FSAR describes how Hydro Plant switchyard control is performed by operators continuously on duty in the Hydro Plant Control Room. FSAR Package 1795 revises Section 8.2 to change operator presences from Watts Bar Hydro to Systems Operations Center (SOC) in Chattanooga, TN as part of the Watts Bar Hydro Plant Automation Program. The SOC will continuously monitor and control or coordinate control by dispatching responders to reestablish switchyard functions, i.e., reset lockout relays and other switchyard connections.

The automation of the Watts Bar Hydro Plant will not introduce the possibility of a change in the frequency of an accident because the operator actions, whether performed locally at the Hydro Plant or remotely, is not an initiator of any accident or any new malfunctions. The design basis event associated with the Watts Bar Hydro Plant operator involves coincident Loss of Onsite and External (Offsite) AC power to the station auxiliaries. Loss of Offsite Power is analyzed in Chapter 15 of the FSAR. Location of the operator will not alter the design function of the Hydro Switchyard including protective relays and circuit breakers. Reliability between the Watts Bar Hydro Plant Switchyard and the remote dispatchers is ensured through the use of two physically separate and dedicated SCADA channels operating through TVA controlled equipment.

This change will not inhibit the operation or adversely affect the functional requirements of the two Watts Bar Nuclear Plant offsite power circuits. The failure modes of the offsite power circuits remain unchanged, with local operator changed to remote operator. Dispatched responders are used in either scenario to respond to a breaker failure or transformer differential. Credible failure modes introduced by this change are possible delay in response time to restore loss of offsite power and SCADA reliability. SCADA is used throughout the Tennessee Valley region for remote control and monitoring of TVA's substations and generating plants. SCADA is a more reliable communication method between dispatchers and responders than the present telephone interface. Furthermore, the design basis does not take credit in the design function of the Offsite AC Power System in regards to response time to reestablish connections. Therefore, this condition does not provide undue risk to nuclear safety.

Safety Evaluation Number: WBPLMN-03-069-0

Implementation Date: ***October 6, 2003***

Document Type:	***Affected Documents:***	***Title:***
FSAR Change Pkg.	FSAR Package 1803	Secondary Neutron Sources

Description and Safety Assessments:

During Cycle 5 refueling outage core reload one of the secondary neutron sources was damaged beyond repair. As a result of this damage, the plant will operate with only one operable secondary source assembly. The damaged secondary will be replaced by a dually compatible thimble plugging device. The neutron sources are incorporated in the core design to provide enough neutrons in the Source Range neutron flux monitors when the reactor is shutdown to assure their operability. This evaluation supports startup of Watts Bar Unit 1 with a single secondary source assembly, as opposed to the two secondary source assemblies.

The design basis accidents potentially affected by this change are ones that result in reactivity events such as an Uncontrolled Boron Dilution. This event can cause changes to the Inverse Count Rate Ratio as a function of measured boron concentration. However, an uncontrolled boron dilution accident cannot occur during refueling due to administrative controls which isolate the RCS from the potential source of unborated water. An uncontrolled boron dilution in Modes 3, 4, and 5 is addressed by operating procedure AOI-3. However, it is estimated that with the entire core reloaded there will be sufficient counts on the source range neutron flux monitor to ensure it remains operable. This is based on sufficient counts seen during the Cycle 6 reload with a fraction of the core reloaded and one secondary source removed. Therefore, the source range neutron flux monitors will remain operable during Modes 3, 4, and 5 and will not adversely affect the uncontrolled boron dilution event analyzed in Section 15.2.4 of the UFSAR.

Other reactivity events are not impacted since the use of a sufficient number of neutron source assemblies provide an adequate neutron activity for the source range neutron flux monitor(s), for which the neutron source assembly is nearest, during core loading, refueling and approach to criticality.

There is no credible failure mode for losing the neutron source from the remaining secondary source assembly. The secondary source assembly uses six double-encapsulated source rods. The double encapsulated secondary source rod design provides additional margin against source material leakage. In the highly unlikely event of a breach of the secondary source cladding, any leaching of source material into the coolant would be on a very slow time scale relative to the duration of core loading and startup activities.

The design function of the neutron source assembly is to provide base neutron level to ensure that the source range neutron flux monitors are operational and responding to core multiplication neutrons and that a positive neutron count is provided during loading, refueling, shutdown and approach to criticality. An additional design function of the neutron source assembly is to limit the amount of bypass flow through the guide tubes during core operation.

This change does not result in an increase in the frequency of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR. Nor does this change result in an increase in the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the UFSAR. This change does not create a possibility for an accident of a different type or a malfunction of equipment important to safety with a different result than any previously evaluated in the UFSAR. This change does not impact any design basis limit for a fission product barrier as described in the UFSAR. Therefore, this change is acceptable from a nuclear safety standpoint as described in this evaluation.

Safety Evaluation Number: WBPLMN-03-071-0

Implementation Date: February 3, 2004

Document Type:	***Affected Documents:***	***Title:***
Hold Order	HO 0916R5043	Leakage from RCS Hot Leg Sample Line

Description and Safety Assessments:

Minor leakage from a fitting in the RCS Hot Leg sample line to the PASF downstream of manual isolation valve 1-SMV-68-578 (RCS Loops No. 3 Hot Leg Sample valve) was observed during startup from the Unit 1 Cycle 5 (U1C5) refueling outage. Since the leaking fitting is in the annulus, access for repair work is not practical during startup or normal plant operation. Repair of the fitting will be accomplished during the next entry into Mode 5 or later depending on priority work. Until then, Hold Order 0916R5043 will close 1-SMV-68-578 to isolate the fitting leak. The design configuration of the valve is normally open to allow sampling of the RCS from Hot Leg Loop No. 3.

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

Isolation of RCS Hot Leg Loop No. 3 sample point does not affect any accidents or malfunctions of equipment important to safety previously evaluated in the UFSAR or create a new accident or malfunction because sampling of the RCS is not an initiator of any new accidents or malfunctions and does not introduce any failure modes. This change does not impact any design basis limit for a fission product barrier as described in the UFSAR. This change does not affect the consequences of an accident or malfunction because closure of this valve does not affect the dose results of the accident analyses reported in the UFSAR. Therefore, this change is acceptable from a nuclear safety standpoint as described in this evaluation.

Safety Evaluation Number: TACF 1-03-019-081 Rev. 1

Implementation Date: October 20, 2003

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-03-019-081 R1	Alternate Source of Primary Makeup Water System

Description and Safety Assessments:

Watts Bar Nuclear Plant has experienced problems with abbreviated reactor coolant pump seal life. A potential contributor to this problem is iron oxides in the seal water supply which can deposit on the internal seal surfaces. Deposition of contaminants on the seal can upset the hydraulic balance of the film seal and increase seal leakoff flowrates. Increased seal leakage was the primary symptom of seal problems in Cycle 5. The PMWS supplies makeup to the Volume Control Tank and charging system and is one potential source for soluble iron. This system has a history of having high iron content. During RFO 5, the PWST was opened and inspected. The PWST is a 187,000 gal. Seismic 1L(B), and is located inside the protected area. It is sized to provide primary water storage for one operating unit and is non-safety related. The tank provides a passive source of water for the PMWS pumps. Significant quantities of rust were found deposited on the floor of the tank. The tank itself is carbon steel with a coated interior and flexible bladder to control dissolved oxygen. Previous outages had identified coating degradation which was repaired. Investigation of the tank and its attachments during this outage revealed a number of the carbon steel attachments to the tank, including instrument sensing ports, inlet and outlet piping, and the overflow line, were not coated on their interiors and were a possible source of iron contamination in the system. Only minor coating damage was noted. For this reason, TACF 01-03-019-081, Revision 1 has been prepared to provide the PMWS with a source of water that does not come from the PWST. The PWST will be isolated from the system and replaced with a direct feed from the demineralized water system. This source includes a demineralized water tank for passive storage and redundant active pumps to facilitate moving the water through aluminum and stainless piping to the primary water system. This TACF will remain in place until a permanent resolution of the iron contamination of the PWST has been implemented.

The PWST is not required to mitigate any design basis events. Replacement of this tank as a passive surge volume of water for the PMWS with water from the demineralized water system does not create the potential for changes to probability of an accident or malfunction. It does not contribute to accident radiological consequences since the demineralized water is radiologically clean. Since the water quality from this system meets or exceeds the PWST water specification and eliminates the source of soluble iron, the likelihood of reactor coolant pump seal problems is expected to be reduced. The need for a booster pump to transfer the water to the PMWS is accommodated by existing redundant pumps and the fact that primary water use does not need to be continuous for any safety functions.

Safety Evaluation Number: TACF 1-03-6-062, Rev. 2

Implementation Date: April 24, 2003

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-03-6-062 R2 Procedure SOI-68-02 Procedure ARI-95-01 Procedure AOI-24	Leakoff Limits for Reactor Coolant Pump 4

Description and Safety Assessments:

In accordance with TACF 1-03-6-062, a UFM was installed on the No. 1 seal leakoff piping of RCP 4. The UFM will allow the operation of the RCP up to 7 gpm and monitoring of leakoff flow up to 8 gpm, which is above the upper limit of 6 gpm of the installed instrumentation. The normal range of leakoff is to 5 gpm, with a maximum of 6 gpm as long as other pump parameters are monitored and seal injection flow is greater than No.1 seal leakoff flow

The UFM does not violate pressure boundary integrity of the seal leakoff piping since the transducers are non-intrusive, strap-on type. The temporary UFM does not interact with any Class 1E circuitry. It was evaluated for adverse impacts to other programs such as Seismic Category 1(L) and Appendix R.

Revision 1 to the TACF allows for an intermediate alarm (below the HI-HI alarm of 6.6 gpm) to be activated in ICS. The setpoint will be variable between 5.0 and 6.5 gpm at Operations discretion. This variable setpoint alarm does not affect the Evaluation.

Revision 2 to the TACF clarifies that transients may cause short duration leakoff excursions above 7 gpm. The required plant shutdown is not required if leakoff returns to below 7 gpm under the conditions specified in the TACF.

The frequency of occurrence of seal failure or likelihood of malfunction is not increased, therefore, this evaluation concludes that this TACF is safe to implement with no undue challenge to plant operation. The TACF does not adversely affect any safety-related systems, structures, or components.

Safety Evaluation Number: TACF 1-03-8-062 Rev. 1

Implementation Date: July 26, 2003

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF No. 1-03-8-062 R1 Procedure SOI-68.02, Procedure ARI-95-101 Procedure AOI-24	Leakoff Limits for Reactor Coolant Pump 2

Description and Safety Assessments:

Reactor Coolant Pump (RCP) 2 is exhibiting an end-of-cycle leakoff trend as it has in the last two fuel cycles. An UFM was installed on the No. 1 seal leakoff piping of RCP 2. The UFM will allow operation of the RCP up to 7 gpm seal leakoff flow and monitoring of leakoff flow up to 8 gpm, which is above the upper limit of 6 gpm of the installed instrumentation. The RCP vendor manual gives a normal range of leakoff of 1 to 5 gpm, with a maximum of 6 gpm as long as other pump parameters are monitored and seal injection flow is greater than seal leakoff flow. A Westinghouse Technical Bulletin provided further guidance for operation and shutdown of RCPs with leakoff outside the previously described limits.

The UFM does not violate pressure boundary integrity of the seal leakoff piping since the transducers are non-intrusive, strap-on type. The temporary UFM does not interact with any Class 1E circuitry. It was evaluated for adverse impacts to other programs such as Seismic Category 1(L) and Appendix R.

The revision to the temporary alteration clarifies that transients may cause short duration leakoff excursions above 7 gpm. Plant shutdown is not required if leakoff returns to below 7 gpm under the conditions specified in the temporary alteration.

The catastrophic failure of the RCP seal is a small break LOCA. The frequency of occurrence of a seal failure is not increased by this temporary alteration to monitor the seal leakage. This 50.59 evaluation for TACF 1-03-8-062 R1 concludes that the TACF is safe to implement with no undue challenge to plant operation. The TACF does not adversely affect any safety-related systems, structures, or components.

Safety Evaluation Number: TACF 1-03-013-246 Rev. 0

Implementation Date: August 26, 2003

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-03-013-246 RO	The Trip Cutout for the Sudden Pressure Device on the Main Bank Transformers.

Description and Safety Assessments:

TACF 1-03-013-246 affects non-safety related System, Main Relay Boards, by defeating the Unit 1 Main Bank Transformer Sudden Pressure Device Circuitry associated with the generator/turbine trip. This is accomplished by opening the TCO block to defeat the Unit trip function of the Sudden Pressure Device on Unit 1 Main Bank Transformers.

The Main Bank Transformer sudden pressure function is described in the UFSAR. The Main Bank Transformer sudden pressure trip function is one of seventeen non-safety, non-seismic automatic turbine trips due to electrical faults in the generator, transformers, 500KV bus or breakers. These trips are designed for equipment protection. If the plant load is greater than 50%, a reactor trip is also generated. This reactor trip function is not affected and the design accident analysis for loss of load and/or turbine trip is not affected.

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

Safety Evaluation Number: WBOTSS-03-005-0

Implementation Date: August 28, 2003

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-03-015-062, R0	Installation of Nitrogen-Supplied Freeze Seal.

Description and Safety Assessments:

This 50.59 evaluates cooling of seal injection flow to RCP 2 and 4 by use of nitrogen-supplied freeze-seal equipment. The cooling of the seal injection supplied to RCP 2 and 4 is intended to reduce the No. 1 seal leakoff from those pumps to an acceptable level.

The design basis accidents to be evaluated in this 50.59 Evaluation are a Station Blackout and Appendix R event. Following a Station Blackout, seal injection flow would be stopped, since it is supplied by the operating centrifugal charging pump. Reactor coolant makeup control is required for Appendix R. This may be achieved by isolating normal make-up and BIT injection path, isolation of the normal and excess letdown paths, and operations of the charging portion through the RCP seal injection path. The reactor coolant seal injection flow path including the flow transmitters and flow indicators are required for Appendix R.

The failure mode that must be considered relative to this evaluation is the failure to close the nitrogen supply to the freeze-seal equipment, resulting in actually establishing a freeze seal in the seal injection piping to RCP 2 and 4.

The action (including required completion time) is reflected in plant procedures and operator training programs. The licensee has demonstrated that the action can be completed in the time required considering the aggregate affects, such as workload or environmental conditions, expected to exist when the action is required. The evaluation of the change considers the ability to recover from credible errors in performance of manual actions and the expected time required to make such a recovery. The evaluation considers the effect of the change on plant.

This evaluation concludes that cooling of the seal injection water to RCP 2 and 4 by use of freeze-sealing equipment is similar to a common maintenance activity. It does not have the potential to increase the frequency of occurrence of any accident previously evaluated in the UFSAR and does not have the potential to create the possibility for an accident of a different type than any previously evaluated in the UFSAR. Installation of the equipment does not increase the likelihood of occurrence of a malfunction of equipment important to safety, nor does it increase the consequences of such a malfunction. The possibility of a new malfunction - the inability to isolate nitrogen to the freeze-seal equipment - and the consequences of failure to isolate the nitrogen, were evaluated and found to be acceptable.

Safety Evaluation Number: WBOTSS-03-006-0

Implementation Date: December 12, 2003

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-03-011-030, R0	Disable CRDM Cooler Motor 2 with Cooler Motor 1 Operational

Description and Safety Assessments:

A Work Order was initiated to troubleshoot the cause for CRDM Cooler 1A-A tripping offline. CRDM Cooler 1A-A Motor 2's power circuit was identified as having a ground in the load side circuit. During plant operation in Mode 1 or 2, this motor is not accessible for maintenance.

The temporary alteration allows CRDM Cooler 1A-A Motor 2 to be disabled while CRDM Cooler 1A-A Motor 1 remains operational. CRDM Cooler 1A-A Motor 2 will be disabled by racking out and the breaker and removing its control power fuses to prevent CRDM Cooler 1A-A Fan 2 from running while allowing CRDM Cooler 1A-A Fan 1 to remain available for service, but only in BYPASS mode. This TACF will maintain CRDM Cooler 1A-A aligned in the BYPASS, i.e. supplemental cooling mode to lower containment which is the required alignment for the Appendix R event. This also allows Cooler 1A-A to supplement the LCCs in maintaining the normal temperature limits if required.

There are no WBN design basis UFSAR Chapter 15 events for which the CRDM cooling system is required to operate. The CRDM coolers and associated duct/dampers are not safety-related and are not required to perform a primary nuclear safety function. However, the CRDM Coolers, in combination with the LCCs, are required for safe shutdown per 10 CFR 50, Appendix R, (i.e. the Watts Bar Nuclear Plant Fire Protection Report) to keep containment temperatures from exceeding operability (environmental qualification) limits on safe shutdown equipment inside containment. Sensitivity analyses were performed for combinations of two LCCs and one or two CRDM coolers operating at reduced air flow conditions. The demonstrated test temperature profile for the EQ equipment in these areas, bounds the temperatures associated with the sensitivity cases. Therefore, there is no increased risk to nuclear safety for this temporary change due to the disablement of CRDM cooler 1A-A, Motor 2.

Safety Evaluation Number: Technical Specification Bases Package 03-19

Implementation Date: October 15, 2003

Document Type:	***Affected Documents:***	***Title:***
Technical Specification Bases	TSB Package 03-19	Venting of Inaccessible Portions of the ECCS.

Description and Safety Assessments:

This evaluates a revision to the Bases for Technical Specification Surveillance Requirement 3.5.2.3 to clarify venting of inaccessible portions of the ECCS.

Certain portions of the ECCS may not be accessible for venting during normal power operations due to industrial safety or radiological considerations. Numerous vents are being added and other currently inaccessible vents are being routed to lower more accessible locations during the Unit 1 Cycle 5 refueling outage. Performance of venting at these new ECCS high points and re-routed high points provides the necessary level of confidence that the piping is full of water. The vents that are in areas which are accessible based on radiological dose and temperature will be vented in accordance with the requirements of the Technical Specification and provide assurance the ECCS will function when required. The purpose of the Bases change is to help maintain dose to operations personnel ALARA and to minimize the industrial safety risk to operations personnel, when necessary, while also fulfilling the intent of the surveillance requirement. The change provides an exception to physically venting portions of the system that may become inaccessible due to plant conditions.

The subject change does not have the potential to increase the frequency of occurrence of any accident previously evaluated in the UFSAR because the extent to which the ECCS piping is filled with water has no effect on the potential for initiation of a transient or accident evaluated in the UFSAR. ECCS high points are either physically vented, or ultrasonically tested to confirm that no adverse affect to system operation will occur. Previous analyses performed for corrective action program documents indicate that even if small amounts of gas accumulate in inaccessible areas prior to the next venting, the pockets would not cause significant fluid transients, damage the ECCS pumps, or inject quantities of gas into the RCS significant enough to impact core cooling or event mitigation. Therefore, this activity does not have the potential to increase the likelihood of occurrence of a malfunction of an SSC important to safety that has been previously evaluated in the UFSAR. The ECCS venting program assures that the system will function to mitigate the required design basis accidents as per the design and licensing basis analysis. As a result the consequences of an accident previously evaluated in the UFSAR will not be increased. The subject change does not increase the consequences of any malfunctions because the ECCS will still be capable of performing all required accident mitigation functions. The change does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR because the extent to which the ECCS piping is filled with water does not have the potential to create a different type of accident.

The possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR is not created because water hammer transients within the ECCS system that could lead to failure will not occur. No ECCS flow reduction is postulated to occur as a result of the change. Therefore, the fission product barriers are not challenged or affected due to this change.

Based on these conclusions, the revision to the Bases for Technical Specification Surveillance Requirement 3.5.2.3 to clarify venting of inaccessible portions of the ECCS is within the design and licensing basis and the activity can be implemented without a License Amendment.

Safety Evaluation Number: WBPLEE-02-081-0

Implementation Date: October 21, 2002

Document Type:	***Affected Documents:***	***Title:***
Work Order	WO 02-013479-000	Temporary Disabling of Vital Battery Board II Ground Alarm.

Description and Safety Assessments:

Work Order 02-013479-000 requires the temporary disabling of safety related 125V DC Vital Battery Board II ground alarm input for Main Control Room annunciator 18-B located in window box 1-XA-55-1C (Control Power and Fire). The non-safety annunciator system provides the operator with alarms for off-normal plant components/conditions. The 125V DC Vital Battery Board II serves a normal function of providing sufficient control power for switchgear, dc motors, valves, emergency lighting, and other electrical devices and components. The safety function of 125V DC Vital Battery Board II is to provide sufficient power for engineered safety features equipment, emergency lighting, vital inverters, reactor protection system, and other safety-related dc powered equipment to permit safe shutdown and isolation of the reactor for the loss of all ac power conditions.

The installed ground indicating meter has a failed internal component that is causing the annunciator to be in a constant alarm condition even though there is not ground on the bus. Since annunciator 18-B has no reflash function, this condition is masking other alarm functions (blown fuse, feeder overload, tripped breaker on Vital Battery Board II, and breaker 36 on 120V AC Vital Instrument Power Board 1-II open) that are electrically in parallel with each other. The disabling of the alarm input will be accomplished without affecting the other alarm inputs or the local ground indicating function at Vital Battery Board II.

The change does not increase the probability of an accident or occurrence of a malfunction of equipment important to safety since the design basis requirements of the systems have not been adversely changed. The potential of a board bus or feeder(s) becoming grounded will not increase in frequency as a result of this change. The consequences of an accident or a malfunction of equipment will not be increased. No new accidents or malfunctions of a different type than evaluated in the UFSAR are created since 125 DC Vital Battery Board II will continue to perform its safety function as specified by the design basis documents. The change does not affect any technical specifications; therefore, no margins of safety are reduced.

Safety Evaluation Number: WBPLMN-03-055-0

Implementation Date: September 4, 2003

Document Type:	***Affected Documents:***	***Title:***
Work Order	WO 03-014882-000	Venting of Non-Condensable Gases

Description and Safety Assessments:

This 50.59 evaluates venting of non-condensable gases in the vertical piping below valve 1-FCV-063-0011-B in accordance with the instructions provided in WO-03-14882-000.

A review of recent work orders has identified the possibility for non-condensable gas to exist in the vertical piping below SIS flow control valve. As a result, the piping has been ultrasonically tested and gas has been found to exist at this location. The condition is documented in a PER. There is a potential for cavitation of the SIP upon initiation of post-LOCA cold recirculation if excessive gas is present in the piping. The FCV is located in the Auxiliary Building. The FCV is installed in an 8-inch vertical pipe.

In order to remove the gas, the FCV will be opened and the gas will be vented through an existing bonnet overpressure isolation valve. This isolation valve is installed on the flow control valve for RHR Spray Header B. Since this isolation valve is normally closed and part of the CSS and SIS pressure boundary, operator action will be relied upon to close a valve in the event of design basis accidents. Redundant operations personnel will be stationed in the area in contact with the control room to assure this action is completed in the unlikely event of a Design Basis Event.

Technical Specification LCO will be entered for SIP 1B-B, hold orders will be established, and the RWST static head pressure will be used to vent the valves. Both RHR pump trains will remain in service during the venting operation so that if an accident were to occur, a portion of the ECCS flow from one of the RHR pumps could be directed out the open vent valve. Therefore, the quantity of water that could be lost through 1-ISV-072-0041 must be considered in the event that a LOCA occurs while the venting operation is being performed.

The design basis accident that is evaluated the LBLOCA in the RCS. Any design basis accident that would require operation of the ECCS is also evaluated. However, the LBLOCA is the accident that requires the greatest flow to be delivered to the core and is; therefore, bounding relative to the considerations that must be addressed within this evaluation.

The failure mode that must be considered relative to this 50.59 evaluation is the failure to re-close isolation valve in the event that a design basis accident occurs while the venting operation is being performed.

Performance of the additional venting below 1-FCV-63-11-B addressed by this evaluation is a maintenance activity with a required manual action to eliminate a potential operability condition and does not have the potential to increase the frequency of occurrence of any accident previously evaluated in the UFSAR and does not have the potential to create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

Since the activity is the venting operation required to remove the gas that exists below valve 1-FCV-63-11-B, the potential to increase the likelihood of occurrence of a malfunction of an SSC important to safety that has been previously evaluated in the UFSAR and the consequences of these SSC malfunctions, will not increase.

Performance of the venting operation does not change the mitigation of the LOCA event such that the consequences of an accident previously evaluated in the UFSAR would be increased. Since the provisions exist to assure valve 1-ISV-72-41 will be closed during the injection phase, but before the sump recirculation phase, performance of the venting procedure does not create a possibility for a malfunction of an SCC important to safety with a different result than any previously evaluated in the UFSAR. Since the peak fuel clad temperature could increase only slightly if a LOCA were to occur with the isolation valve open, the fission product barrier of the fuel rod cladding is not unduly challenged or affected due to this change. Based on these conclusions, the change to implement the venting procedure of WO-02-14882-000 is within the design and licensing basis and the activity can be implemented without a License Amendment.