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

SEP 3 0 2005

10 CFR 50.59(d)(2)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Gentlemen:

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

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

Pursuant to 10 CFR 50.59(d)(2), this letter provides the Summary Report of the implemented changes, test, and experiments in which evaluations were performed in accordance with 10 CFR 50.59(c). The enclosure provides a summary of the evaluations for the Updated Final Safety Analysis Report Amendment 5 provided under separate cover, and includes other evaluations performed during the period from March 17, 2004 to August 30, 2005.

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. D. Pace
Manager, Site Licensing
and Industry Affairs

Enclosure
cc: See page 2

IE47

SEP 30 2005

cc (Enclosure):

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ENCLOSURE

WATTS BAR NUCLEAR PLANT UNIT 1
10 CFR 50.59 SUMMARY REPORT

SA-SE Number: CM-3.01, Rev. 51

Implementation Date: 03/28/05

Document Type:	*Affected Documents:*	*Title:*
Procedure Change	CM-3.01 R51	Revision of the Chemistry Specification for the Reactor Coolant System (RCS)

Description and Safety Assessments:

Revise the Chemistry Specification for the RCS at temperature pH limit for Mode 1 and 2 from a constant 7.1 pH to a constant 7.2 pH, not to exceed 3.5 ppm Lithium. This change increases the system pH by increasing the amount of Lithium-7 hydroxide for a given boron concentration. By increasing the operating cycle at temperature pH, reduced corrosion product release rates can be achieved. Reducing release rates will lower corrosion product deposition on system piping which will lead to reduced shutdown dose rates.

The design function of the control of coolant chemistry is to protect the materials of construction of the RCS pressure boundary which might otherwise reduce the structural integrity of the boundary during its service lifetime. By providing chemistry controls on the RCS system, mechanisms which could reduce the structural integrity of the system, such as corrosion, are minimized. Lithium-7 hydroxide is used to control the pH of the system in a range which minimizes general corrosion of the system. This chemical is chosen for its compatibility with the RCS material and it is produced by the neutron irradiation of the soluble boron. This change increases the system pH by increasing the amount of Lithium-7 hydroxide for a given boron concentration. By increasing the operating cycle pH, reduced corrosion product release rates can be achieved. This will reduce corrosion product deposition on system piping, thereby reducing shutdown dose rates.

The increase in pH may contribute to an increase in Primary Water Stress Corrosion Cracking (PWSCC). The condition for stress corrosion cracking to occur involves three primary elements; a susceptible material, a tensile stress, and a susceptible environment. Industry experience has shown that Alloy 600 and its associated weld materials Alloys 82 and 182, are susceptible to PWSCC. The most susceptible location is Steam Generator tubing material due to the tube thickness being relatively small. "PWR Primary Water Chemistry Guidelines" states, "chemistry variables are likely to influence, to some extent, the occurrence of PWSCC, even though it is clear that coolant chemistry is, at most, a second-order effect in the cracking process compared with material and stress parameters."

Using an updated Material Reliability Program (MRP)-68 study the Electrical Power Research Institute (EPRI) documents predicts a 10-15 percent increase in susceptibility of PWSCC from the pH change. An assessment of a constant pH 7.2 prepared by Westinghouse for Watts Bar, WAT-D-11318, "Effects of Elevated pH on Operation on Stress Corrosion Cracking of Primary System Materials and Components" statistically projects that an additional 19 steam generator tubes may experience PWSCC crack initiation from the pH change. The EPRI and Westinghouse documents state that the increase in pH has no impact on crack growth rates after initiation. The assessment concludes that, "neither the newly initiated cracks nor those that may already be present to some undetermined depth will experience an increase in the probability of through-wall degradation. That is, the probability of an accident is the same after the chemistry change as it is if the change is not made."

CRDM penetrations and the associated J-grove attachment welds have also demonstrated significant susceptibility for PWSCC. The estimates that due to

SA-SE Number: CM-3.01, Rev. 51

Implementation Date: 03/28/05

Description and Safety Assessments (Continued)

the low head operating temperature "it will require over 30 additional operating years to reach the 8 Effective Damage Years experience that has been established as the level for moving from a low to moderate risk situation.

Based on the above discussion, the proposed change in operating chemistry will have an insignificant impact on that estimate."

SA-SE Number: DCN 51661-A

Implementation Date: 03/25/2005

Document Type:	*Affected Documents:*	*Title:*
Design Change	DCN 51661-A UFSAR Change Package 1847, Change Package 04-16 TRM	Re-gearing Specific Valves in the Generic Letter 89-10 Program

Description and Safety Assessments:

This Design Change Notice (DCN) is being issued to provide for re-gearing the specific valves (1-FCV-63-0022B, 1-FCV-63-0156A, 1-FCV-63-0157B, 1-FCV-72-0002B) in the population of the Generic Letter (GL) 89-10 program. Calculations, System Description, drawings, design criteria, Technical Requirements Manual, and Final Safety Analysis Report (FSAR) sections or tables are also being revised, as required, to reflect the stroke time change requirements associated 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 by the design standard. A Problem Evaluation Report (PER) has been generated for which this DCN will partially implement the corrective action.

The re-work of these valves ensures adequate margin and provides for minimal impact from future changes to the motor operated valve (MOV) program. These changes decrease the test frequency requirements for these valves associated with the GL 96-05 aspect of the MOV program.

In order to gain additional thrust for valve operation the motor-operators were re-geared which increases the stroke times. The additional thrust is a change in the conservative direction and the increased stroke times is an adverse change to an operational parameter.

Valves are normally closed. The valves have an active safety function to open during changeover to hot leg recirculation and passive function to remain closed during cold leg recirculation and safety injection phase. (Stroke time increased from 10 to 17 seconds.)

Valve is normally open. It has the function to close during changeover to hot leg recirculation and the passive function to remain open during the cold leg recirculation and safety injection phase. (Stroke time increased from 10 to 17 seconds.)

Valve is normally closed. It has the function to open upon receipt of a 2 out of 4 channel high high containment pressure signal (2.8 psid) to provide flow path for cool borated water to be sprayed into the containment atmosphere so that the containment design pressure is not exceeded. (Stroke time increased from 15 to 28 seconds.)

Therefore, it is concluded that regearing of the operator is an acceptable change with a minor adverse affect.

Based upon the results of the evaluation, the activity can be implemented per plant procedures. The safety evaluation results show that implementing this modification is acceptable from a nuclear safety standpoint and does not require NRC approval.

SA-SE Number: DCN D-51716-F

Implementation Date: 04/15/2005

Document Type:	*Affected Documents:*	*Title:*
Design Change	DCN D-51716-F UFSAR Change Package 1868	Implements Site Security Upgrades through the Installation of a Vehicle Barrier System (VBS).

Description and Safety Assessments:

As part of a security upgrade at WBN, a Vehicle Barrier System (VBS) consisting of vehicle barriers and channeling devices was placed in the owner-controlled area to protect the plant from vehicle intrusions. A new sally port with larger vehicle inspection area and active popup barriers has been constructed just west of the existing sally port, west of the Plant Office Building. Additional lighting has been provided. Gates have been installed to allow access through the barriers as required. Power has been supplied to the new security equipment and lighting from non-safety related 480V Office Building Vent Board 1.

The VBS is located outside of the protected area fence. These barriers support compliance with NRC's Design Basis Threat (DBT) Security Order.

The VBS has been evaluated and designed to minimize adverse effects upon site drainage characteristics. The Updated Final Safety Analysis Report (UFSAR) discusses 4 watershed areas and the respective water elevation of each resulting from the local storm probable maximum precipitation (PMP) event. The VBS resulted in a slightly higher water elevation in Watershed Area 2 drainage channel, which is the natural channel flowing from west of Service, Auxiliary, Reactor, and Diesel Generator Buildings.

This represents an insignificant change with respect to the design basis presented in the UFSAR and flooding of safety-related structures to occur. Because the maximum water level quoted in the UFSAR is changing, a UFSAR Change Request is included in the DCN.

Vehicle intrusion protection specifics are not addressed in the UFSAR, but rather in the UFSAR-referenced site Physical Security Plan. The Physical Security Plan has been revised to comply with the NRC's DBT Security Order and the physical changes being made by the security upgrade project.

This evaluation addresses the required WBN UFSAR change and the portion of this DCN which impacts the predicted maximum water level within the site Watershed Area 2 drainage channel, i.e., placement of the VBS bollards across the drainage channel on the northwest corner of the site. As the predicted maximum water level remains below the critical flood elevation, it concludes that there is no negative impact on safety-related structures, systems, or components.

SA-SE Number: EDC E-51860-A

Implementation Date: 06/23/2005

Document Type:	*Affected Documents:*	*Title:*
Engineering Document Change	EDC E-51860-A	Revise the Primary Makeup Water System (PMWS) Description.

Description and Safety Assessments:

The proposed change revises the PMWS description and drawings to clarify that the Primary Water Storage Tank (PWST) may be bypassed during times when the tank is required to be taken out of service. The bypassed flow path will be from the Demineralized Water Storage Tank, through PMWS Pumps (PMWSPs) and on to the PMWS users. To facilitate this flow path, the PMWSPs will be deenergized; i.e., free wheeling and not be running. In addition, valves will be closed to isolate PWST, and other instrument panel valves will be open or closed to isolate pressure instrumentation. Both Demineralized Water Booster Pumps shall be operated continuously to achieve the maximum flow and pressure to both PMWS and Condensate System. This will allow the Demineralized Water and Distribution System being supplied to Condensate System to remain in automatic makeup. Annunciation system alarms currently used to monitor PMWS normal operation will remain, but response to the alarms may be procedurally or administratively controlled by Operations to allow interim operation when PWST is bypassed. This is acceptable because the alarms are on the PWST level indication, and the pressure differential across the PMWSPs will not be required during this configuration.

A review of the function of the PWST and the PMWSPs indicates they are 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 Storage and Distribution System and 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 both systems are radiologically clean. The water quality from both systems meets or exceeds the water specifications. Since the system is not safety grade, the addition of the need for both booster pumps to operate is not different from the loss of the PMW Pumps or failure of a principal flow component in that system.

No design basis limits or values are challenged as a result of bypassing the PWST when the tank is required to be taken out of service. No fission products barriers are unduly challenged or affected due to this change. Therefore, this change cannot affect these barriers.

SA-SE Number: EDC 51859-A

Implementation Date: 08/25/05

Document Type:	Affected Documents:	Title:
Engineering Document Change	EDC 51859-A SAR Change Package 1876	This EDC will revise the system descriptions and Final Safety Analysis Report (FSAR) to permit liquid radioactive releases at greater concentrations than described above without the 20,000 gpm CTB dilution flow.

Description and Safety Assessments:

The Steam Generator Blowdown (SGB), Condensate Polishing Demineralizer (CPD) System Descriptions, and Final Safety Analysis (FSAR) section 11.2.4 currently allow radioactive fluid releases with less than 20,000 gpm cooling tower blowdown dilution flow if the gross gamma activity is less than or equal to 5E-7 uCi/cc and tritium less than or equal to 1E-5 uCi/cc. This Engineering Design Change (EDC) revises the system descriptions and the associated calculation to allow release of liquids from the SGB and CPD systems with the Cooling Tower Blowdown (CTB) dilutions flow is less than 20,000 gpm provided the activity for tritium is less than or equal to 5.322E-3 uCi/cc and the other isotopes in the release mix are at or below the LLD by overriding the interlock with 1-FCV-15-44 and/or 0-FCV-14-451. If it is desired to utilize gross gamma radioactivity instead of individual lower limit of detection (LLDs) for isotopes other than tritium for establishing the acceptability of release, then a total concentration criterion of 5E-7 uCi/cc gross gamma may be used as the lower limit in conjunction with the tritium lower limit of 5.322E-3 uCi/cc with no cooling tower blowdown. This is more restrictive than utilizing the individual isotopes since 5E-7 uCi/cc is less than or equal to any single isotope's effluent concentration limit (ECL). The above release criterion is a subset of the following general release criteria.

In general, this Engineering Document Change (EDC) change will allow release of liquids from the SGB or CPD systems with the CTB dilution flow is less than 20,000 gpm provided the sum of the release concentration/ECL for all isotopes released is less than or equal to 10 as required by the Technical Specifications and Off Site Dose Calculation Manual (ODCM), the maximum SGB flow is 220 gpm, and the releases are controlled and limited such that the 10CFR50, Appendix I limits are not exceeded.

The EDC changes require that administrative controls be implemented by Operations while using the valve "OPEN" positions to assure that radiation greater than expected is not released from the plant. The radiation monitors will be set in accordance with the applicable setpoint and scaling documents to ensure the isolation requirements are satisfied. If valve(s) do "CLOSE" while processing to the CTB, Operations should notify Site Chemistry as soon as possible.

This change will make the affected design output system descriptions and the FSAR consistent with the current Technical Specification and ODCM requirements for liquid radioactive releases. All design functions will be maintained with this change. In addition, the 10CFR20 and 10CFR50, Appendix I release limitations will continue to be met. The change does not result in new accidents or malfunctions evaluated in the Updated Final Safety Analysis (UFSAR). In addition, no fission product barriers or evaluation methods described in the FSAR are affected by this change.

SA-SE Number: SOI 15.01, Rev. 41, UIC-1

Implementation Date: 8/13/04

Document Type:	*Affected Documents:*	*Title:*
Procedure Change	Procedure SOI 15.01 Rev. 41, 1-ODI-90-2	Steam Generator Blowdown Increased Flow

Description and Safety Assessments:

Steam Generator Blowdown (SGBD) System currently establishes the total maximum SGBD flow at 350 gpm through the heat exchanger flow path and a maximum of 262 gpm when blowdown is process to the Cooling Tower Blowdown. System Operating Instruction (SOI)-15-01, "Steam Generator Blowdown System" is revised to allow an increase in SGBD flow from 262 gpm to 329 gpm. This change is necessary to accelerate removal of sulfates in the secondary system water, which resulted from inadvertent intrusion of resin beads. Procedure 1-ODI-90-2 "Steam Generator Blowdown Releases" is revised to insure the setpoint for SGBD radiation monitors, 1-RE-90-120, and 121, do not exceed 2115 counts per minute (cpm) for consistency with the monitor flow rate which supports a response time of 41.5 seconds. If the monitor setpoint must be increased above 2115 cpm, the Offsite Dose Instruction (ODI) will require that the maximum blowdown flow be returned to 262 gpm. The above procedure changes will be applicable until the outage scheduled for Spring of 2005. After the outage, the maximum allowable SGBD flow will return to 262 gpm, or a permanent change will be issued to change the FSAR and other documentation. The Steam Generator Tube Rupture accident is involved with this change. There are no credible failure modes created by this change.

The overall design bases flow for the SGBD system is 350 gpm. Consequently, the flow increase from 262 gpm to 329 gpm instituted by this change is within the design bases relative to affect on the steam generators, the Main Steam System, and portions of the SGBD upstream of the line to the cooling tower blowdown. The reduced radiation monitor response time and limitation on the setpoint will insure the SGBD is isolated in the event of high radioactivity in the flow stream in accordance with the intended design function. The STGR piping and components downstream of the blowdown isolation valve are adequate for the increased flow conditions. All design functions will be maintained with this change. The change, therefore, does not result in new accidents or malfunctions, and do not result in increased frequency or consequences of accidents or malfunctions evaluated in the Updated Safety Analysis Report (UFSAR). In addition, no fission product barriers are challenged by this change.

SA-SE Number: SOI-15.01, Rev. 44

Implementation Date: 03/25/05

Document Type:	***Affected Documents:***	***Title:***
Procedure Change	Procedure SOI-15.01, Rev. 44	Allow Steam Generator Blowdown to Remain Inservice with Auxiliary Feedwater (AFW) Pump Running

Description and Safety Assessments:

This Safety Evaluation covers a proposed procedure change to System Operating Instruction (SOI)-15.01 to allow Steam Generator Blowdown (SGBD) to remain in service while an Auxiliary Feedwater (AFW) Pump is running up to and reactor power is less than or equal to 10 percent full power. This change can only be used for start-up from the Spring 2005 Unit 1 Cycle 6 (U1C6) Refueling Outage (RFO).

A problem at Sequoyah Nuclear Plant (SQN) identified conditions where the SGBD isolation valves will not close when an AFW pump start signal has been generated. A Problem Evaluation Report (PER) was subsequently generated for Watts Bar Nuclear Plant (WBN). This condition exists when a AFW pump is started and the SGBD system is desired to be placed in service after the AFW pump is in service. The only condition for automatically closing the SGBD valves with an AFW pump in service is containment isolation Phase A signal. Other AFW pump start signals will not close the valves unless the pump in service is secured to reset the bypass relay prior to the automatic pump start signal.

WBN has the ability to start-up using the electric Standby Main Feedwater Pump without using the AFW system. However, due to the secondary side water inventory not meeting chemistry specifications it is desired to have the ability to assist start-up from U1C6 RFO using the AFW system. Using the AFW system will allow clean water from the Condensate Storage Tank to be used directly to maintain Steam Generator (SG) level. Also, the SGBD system is needed to be in service to help clean up the secondary side water inventory.

Westinghouse has evaluated this proposed change with respect to impacts to loss-of-coolant accident (LOCA) and non-LOCA accident analysis. The results show that it is acceptable to maintain a SGBD flow up to 120 gpm while AFW is in service during start-up from a RFO up to 10 percent full power. The evaluations provide justification that adequate AFW decay removal flow will still be available with the 120 gpm being discharged to the blowdown system.

This proposed change is a temporary interim change applicable for start-up from the Unit 1 Cycle 6 Refueling Outage (U1C6 RFO). Therefore, the Final Safety Analysis Report (FSAR), and System Descriptions will not be updated at this time.

The design functions will be maintained with this change. The change therefore does not result in any new accidents or malfunctions, and does not result in increased frequency or consequences of accidents or malfunctions evaluated in the Updated Final Safety Analysis Report (UFSAR). In addition, no fission product barriers are challenged by this change.

SA-SE Number: TACF 1-03-011-030, R0

Implementation Date: *06/22/04*

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-03-011-030, Rev. 0	Control Rod Drive Mechanism (CRDM) Alteration

Description and Safety Assessments:

This Temporary Alteration Change Form (TACF) allows Control Rod Drive Mechanism (CRDM) Cooler 1A-A Motor 2 to be disabled while CRDM Cooler 1 A-A Motor 1 will remain operational. CRDM Cooler 1 A-A Motor 2 will be disabled by racking out breaker, and removing its control power fuses to prevent CRDM Cooler 1 A-A Fan 2 from running while allowing CRDM Cooler 1 A-A to remain available for service, but only in BYPASS mode. This TACF will maintain CRDM Cooler 1 A-A in the BYPASS. The equipment affected is not safety related. Any malfunctions of the CRDM coolers or associated dampers or duct would not be important to safety.

Sensitivity analyses were performed for combinations of two Lower Compartment Coolers (LCCs) and one or two CRDM coolers operating at reduced air flow conditions. This was assumed to account for wet LCC coils due to high humidity conditions and failed CRDM motors. Comparison of each area's calculated temperature to its equipment qualification profile shows that with three exceptions, all areas remain bounded by their respective equipment qualification profiles. The three exceptions are the lower reactor cavity, the upper reactor cavity, and the upper containment. For the upper compartment area, the short duration of these calculated excursions are minor and constitute only one percent of the total Appendix R event duration of 72 hours. The demonstrated test temperature profile for the equipment located in both the upper and lower reactor vessel cavity areas bounds the temperature profile associated with the sensitivity cases. Therefore, the conclusions for the base cases are also valid for the sensitivity cases and thereby, the proposed TACF may be implemented without increased risk to nuclear safety.

SA-SE Number: TACF 1-04-003-027, R1

Implementation Date: 09/07/04

Document Type:	*Affected Documents:*	*Title:*
Temporary Alteration	TACF 1-04-003-027 R1	Add an Alternate Cooling Tower Blowdown (CTBD) Flow Signal to Steam Generator Blowdown (SGBD) isolation valve.

Description and Safety Assessments:

The purpose of the evaluated temporary alteration control form (TACF) is to provide an interlock between Cooling Tower Basin (CTB) Level and SGBD valve in order to provide an alternate permissive for verification of adequate CTBD flow. The valve controls the flow from the SGBD system to the CTBD. This SGBD isolation valve is designed to remain open if the following plant conditions (i.e., permissives) are satisfied: 1) SGBD radiation is less than the setpoint and 2) CTBD flow is greater than setpoint.

Frequent down scale failures due to flow element fouling are causing repeated closures of the valve which is complicating efforts to manage Steam Generator water chemistry. Since the CTB gravity feeds to the CTBD line through a weir, there is a direct correlation between CTB Level and CTBD flow. Therefore, this TACF provides for a CTB level switch function to provide an additional permissive which verifies adequate CTBD flow and allows the valve to receive an alternative CTBD flow verification signal. Thus, the failure of the loop will not inadvertently close the SGBD to CTBD flow path.

The SGBD isolation valve permissives will be as follows: 1) SGBD radiation is less than setpoint and either 2.a) CTBD flow is greater than setpoint or 2.b) CTB weir level is above setpoint.

This TACF does not affect any Design Basis Events. The SGBD isolation valve will automatically close upon detection of a high radiation signal from radiation monitors. This control feature is not impacted by this TACF. Also, SGBD flow is terminated for events that require containment isolation or Auxiliary Feedwater operation. These safety features are not affected by this TACF. Therefore, this change does not adversely impact any Design Basis Events.

Based on the above description, it is concluded that the temporary plant configuration allowed by the subject TACF is safe to implement. Reasonable assurance has been provided that if the river flow were to drop below the required flow to maintain CTBD flow to the river, no radiation, in excess of the 10 CFR 20 standards would be discharged to the Yard Holding Pond.

SA-SE Number: TACF 1-04-005-062 R0

Implementation Date: 10/26/04

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-04-005-062 R0 WO 04-824686-000	Eliminate Leakage to Pressurizer Relief Tank (PRT)

Description and Safety Assessments:

Work Order (WO) 04-824686-000 will gag relief valve closed under the control of TACF 1-04-005-062 R0. An increasing level trend exists for the pressurizer relief tank (PRT). Ultrasonic testing (UT) indicates that the most likely source of the additional water input to the PRT is leakage through the relief valve. Gagging the relief valve closed is expected to reduce or eliminate leakage through the relief valve; and therefore, stop the increasing level trend for the PRT. The primary focus of this evaluation is the over-pressure protection design of the Chemical and Volume Control System (CVCS) with the relief valve gagged closed.

There are no design basis accidents during which the operating pressure in the CVCS boundary protected by the relief valve will reach the set pressure of the relief valve. Normal shutdown operations with the Residual Heat Removal (RHR) System in operation is the plant operating condition in which over-pressurization of the subject CVCS system piping must be evaluated. This operating condition is described as a Condition 1 (Normal Operation and Operational Transients) event in Updated Final Safety Analysis Report (UFSAR) Section 15.1. The RHR is connected to the CVCS piping, but is isolated by the closed valve. The relief valve protects the common suction piping for the centrifugal charging pumps from over-pressurization due to in-leakage through the closed valve. The failure mode that must be evaluated is the possibility for over-pressurization of the CVCS system piping normally protected by the relief valve while TACF 1-04-005-062 R0 is in effect.

Since malfunction of the over-pressure protection design of the centrifugal charging pump suction piping will not occur, implementation of TACF-1-04-005-062 R0 does not increase the frequency of an accident, including normal shutdown operations with the RHR in operation. The ASME Section III Boiler and Pressure Vessel Code requirements regarding over-pressure protection are met, with TACF 1-04-005-062 R0 in place. Therefore, the UFSAR described design function of over-pressure protection for the CVCS is not adversely affected and the likelihood of a malfunction of a Structure, System, or Component (SSC) important to safety (over-pressurization of the centrifugal charging pump common suction piping) is not increased. The proposed activity does not result in more than a minimal increase in the consequences of an accident or malfunction previously evaluated in the UFSAR because the over-pressure protection provided for the common centrifugal charging pump suction piping still meets the ASME Code requirements while TACF 1-04-005-062 R0 is in effect. Continued ASME Code compliance for over-pressure protection also assures that the possibility for an accident of a different type or a malfunction with a different result does not exist. The centrifugal charging pump suction piping is protected from over-pressure, so that it will function in support of the Emergency Core Cooling System (ECCS) to maintain design temperature limits for the fuel cladding (fission product barrier). Based on the preceding summary, the temporary change of TACF 1-04-005-062 R0 can be implemented.

SA-SE Number: TACF 1-05-002-079 R0

Implementation Date: 03/13/2005

Document Type:	***Affected Documents:***	***Title:***
Temporary Alteration	TACF 1-05-002-079 R0	Fuel Transfer Cart Interlock

Description and Safety Assessments:

This Temporary Alteration Change Form (TACF) affects an input to the end-of-travel interlock on the reactor side operator actuated lifting arm (upender) of the Fuel Transfer System (FTS). The Interlock is to ensure the fuel transfer cart has reached its end-of-travel position before the lifting arm can be raised to the vertical position. The TACF allows the interlock to be satisfied by permitting a second input from the system encoder to be placed in parallel with proximity/limit switches. Specifically, the permissive input of the proximity/limit switches will be "OR" gated with an input from the system encoder that will identify the fuel transfer cart travel position as being end-of-travel, prior to the upender being allowed to raise to the vertical position (i.e. starting the automatic sequence of upending a fuel assembly to vertical). Thus, either the proximity/limit switches <u>or</u> the system encoder identification of end-of-travel along with the existing 1) encoder travel position indication, and 2) cable tension indication will serve to make the required logic interlock that will allow manual and automatic operation of the upender. The TACF will be implemented by a change to the logic software governing the programmable logic controller (PLC).

Two redundant inputs to one of the interlocks are provided by a transfer cart encoder position indication of approximately 560 to 570 inches and limit sensing or 150 lbs tension on the cart cable. A second interlock is provided by the proximity/limit switches which, with this change, will be "OR" gated with transfer cart encoder option of 1 inch maximum length in the approximate area of the proximity/limit switch.

This TACF changes the fuel transfer cart upender logic from three redundant interlock permissives for upender operation of a transfer cart to two interlocks with one being the original transfer cart encoder position indication of approximately 560 to 570 inches plus limit sensing of 150 lbs tension on the cart cable, and the second interlock being an "OR" gate including the original proximity/limit switch. When the logic for the interlock is satisfied the reactor side upender is allowed to raise the fuel assembly to the vertical position either manually or in automatic transfer operation. The failure mode of the Traverse Encoder used in the interlock and it falls open which results in incorrect traverse position reading to PLC and readout." The failure mode would allow the transfer art to continue movement to the end stop. The FTS is set up to trip if an overload is detected as would occur if the end stop was reached. Also, as a backup to that, the motor drive controlling the carriage has a current limiting feature to limit the drive from producing enough torque to damage the carriage or the end stop. The limit sensing/cable tension inputs to the interlock would continue to provide a diverse means to prevent the upender arm from being raised if a fuel assembly was not inserted fully in the upender. Poor reliability and periodic unavailability of the proximity switch has necessitated this temporary change.

This activity does not increase the frequency of an accident or the likelihood of a malfunction. Any consequences resulting from this activity are bounded by the UFSAR Chapter 15 Fuel Handling Accident. This change does not affect the ability of the refueling system to perform its UFSAR described design function. This change does not impact the fission product barriers and is not a departure from a method of evaluation described in the UFSAR. The Technical Specifications are not affected by this change.

SA-SE Number: WBN Unit 1 Cycle 7, COLR and Cycle Operation

Implementation Date: 04/01/05

Document Type:	*Affected Documents:*	*Title:*
Other	Core Operating Limits Report (COLR)	WBN Unit 1 Cycle 7, COLR, Cycle Operation

Description and Safety Assessments:

This evaluation was performed to evaluate the maximum lead rod average burnup for cycle 7. As documented in the Westinghouse Reload Safety Evaluation (RSE), the maximum lead rod average burnup for cycle 7 exceeds the currently licensed limit of 60,000 MWD/MTU for the Westinghouse fuel evaluation methodologies. The cycle 7 maximum lead rod average burnup is a change to the fuel evaluation methodology in the Updated Final Safety Analysis Report by reference (WCAP-10125-P-A and WCAP-1 2610-P-A).

Justification of the additional fuel rod average exposure of 1591 MWD/MTU beyond the currently licensed limit is based on the NRC-approved Fuel Criteria Evaluation Process (FCEP) (WCAP-12488-A). As required by FCEP, the Westinghouse notification to the NRC of the use of the FCEP process to justify additional fuel rod average exposures of up to 2000 MWD/MTU beyond the NRC-licensed 60,000 MWD/MTU limit was provided in a letter from N. J. Liparulo (Westinghouse) to R. J. Jones (NRC), "Westinghouse Interpretation of Staff's Position on Extended Burnup (Proprietary)," NTD-NRC-94-4275, August 29, 1994. Consistent with this letter, this burnup extension to the NRC-licensed 60,000 MWD/MTU limit is acceptable provided that all the fuel design criteria stated in the FCEP are satisfied for cycle 7.

For cycle 7, the RSE concludes that all the fuel design criteria, including those specified in the FCEP, are satisfied up to the cycle 7 lead rod burnup of 61,591 MWDIMTU. The lead rod burnup extension has no impact on the radiological consequences of accidents that are TVA scope as long as the core average burnup does not exceed 1,000 EFPD and the lead assembly burnup does not exceed 1,500 EFPD. Both of these burnup criteria are met for cycle 7.

By meeting the fuel design criteria specified in the FCEP, the extension of the burnup range to 61,591 MWD/MTU for cycle 7 by means of the FCEP methodology is applicable. Therefore, the lead rod burnup extension for the cycle 7 reload is licensable under 10 CFR 50.59 and requires no prior NRC approval.

SA-SE Number: Work Order 04-810450-000

Implementation Date: 3/8/2005

Document Type:	*Affected Documents:*	*Title:*
Work Order	WO 04-810450-000	Welding of Coupons Inside the Reactor Building Annulus during Unit 1 Cycle 6 Refueling Outage.

Description and Safety Assessments:

The activities detailed in Work Order (WO) 04-810450-000 determined the potential impact, if any, on sensitive plant equipment and instrumentation as a result of Electromagnetic Interference (EMI) that might be experienced while performing welding activities in the Reactor Building annulus during cold shutdown (Mode 5). In Mode 5, the Engineered Safety Feature (ESF) input circuitry was energized as well as critical instrumentation and control circuitry. These welding activities were in accordance with the Bechtel Special Processes Manual (SPM). The welding incorporated the use of coupons that will be removed after the welding occurs and no permanent plant modifications result from these activities. These welding activities were performed in accordance with Work Plan and Inspection Record (WP&IR) E-EMI-001 and were intended to verify the acceptability to perform welding in the annulus during operation (Modes 1-4).

The performance of EMI testing does not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction, or create a new type of accident or malfunction. No fission product barriers will be affected, so no design basis fission product barrier limits are altered or exceeded. No analytical methodologies used in demonstrating conformance to regulatory requirements are affected, so a departure from a method of evaluation described in the Updated Final Safety Analysis Report (UFSAR) used in establishing the design bases or in the safety analyses will not occur. Therefore, based upon the results of the evaluation, the activity is safe to implement.