Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

November 8, 1999

10 CFR 50.4
10 CFR 50.59(b)(2)
10 CFR 50.109(a)(7)

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 - SUMMARY REPORT FOR JANUARY 1, 1998, THROUGH MAY 31, 1999

This letter provides the BFN Summary Report (SR) for January 1, 1998 through May 31, 1999. Enclosure 1 contains summaries which include brief descriptions of any changes, tests, and experiments with safety evaluations prepared for revisions to the Updated Final Safety Analysis Report, plant procedures (new and revised), special operating conditions, special tests, temporary alterations, and plant modifications that were field completed during this reporting period as required by 10 CFR 50.59(b)(2).

Enclosure 2 contains the BFN Fire Protection Report, Revision 14.

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Enclosure 3 includes copies of changes and additions to the BFN Technical Requirements Manual (TRM) made in accordance with TRM Section 5.1, TRM Control Program. These changes have been incorporated during the period since the Improved TS and TRM were implemented (July 27, 1998) and meet the criteria described within the above control program for changes not requiring prior NRC approval.

Enclosure 4 contains copies of changes to the BFN Technical Specifications (TS) Bases in accordance with BFN TS Section 5.10, TS Bases Control Program. These changes also have been incorporated during the period since the Improved TS and TRM were implemented and also meet the criteria described within the above control programs for changes not requiring prior NRC approval.

Enclosure 5 contains summaries of docketed commitments that TVA has evaluated and revised using administrative controls that incorporate the Nuclear Energy Institute's (NEI) "Guideline For Managing NRC Commitments" in accordance with 10 CFR 50.109(a)(7).

There are no commitments contained in this letter. If you have any questions regarding this report, please contact me at (256) 729-2636.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosures
cc: See page 3

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

10 CFR 50.59(b)(2)
SUMMARIES

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT

1998
SUMMARY OF
SAFETY EVALUATIONS

1998

SUMMARY OF SAFETY EVALUATIONS FOR CORE COMPONENT DESIGN CHANGE REQUESTS AND CORE OPERATING LIMITS REPORTS

UNIT 2 CYCLE 10 CORE OPERATING LIMITS REPORT

This safety evaluation addresses revisions to the BFN Unit 2 Cycle 10 Core Operating Limits Report (COLR) (see TVA BFN 1997 Annual Operating Report). The COLR is being revised to support implementation of Improved Technical Specifications (ITS). Changes to the COLR include: 1) revision of the Tau calculation to change the reference point for scram time measurements, 2) addition of statements that the Average Planar Linear Heat Generation Rate limits contained in the COLR are applicable for Turbine Bypass Out Of Service , 3) addition of Minimum Critical Power Ratio operating limits for Recirculation Pump Trip Out Of Service, 4) addition of operability requirements for the Rod Block Monitor, 5) incorporation of the Shutdown Margin limit, and 6) administrative type changes to Technical Specification reference numbers in order to match the ITS format. The changes are supported by Maximum Extended Load Line Limit and ARTS Improvement Program analyses (NEDC-32433P) and Equipment Out Of Service analyses (NEDC-32774P) performed by General Electric for the BFN plant. These changes are also consistent with the Unit 2 ITS requirements being implemented (TVA-BFN-TS-362).

There is no increase in the probability or consequences of an accident or malfunction of equipment from that previously evaluated in the SAR. The proposed activity does not create the possibility of a different type of accident or malfunction from that previously evaluated in the SAR. The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, it does not involve an unreviewed safety question.

UNIT 3 CYCLE 8 CORE OPERATING LIMITS REPORT

This safety evaluation addresses revisions to the BFN Unit 3 Cycle 8 Core Operating Limits Report (COLR) (see TVA BFN 1997 Annual Operating Report). The COLR is being revised to support implementation of Improved Technical Specifications (ITS). Changes to the COLR include: 1) revision of the Tau calculation to change the reference point for scram time measurements, 2) addition of Minimum Critical Power Ratio (MCPR) operating limits for Turbine Bypass Out Of Service (TBOOS) and indication that existing Average Planar Linear Heat Generation Rate limits are applicable for TBOOS, 3) addition of MCPR operating limits for Recirculation Pump Trip Out Of Service, 4) incorporation of Average Power Range Monitor setpoint changes for operation above 25% power with the Maximum Fraction of Limiting Power Density greater than the Fraction of Rated Power , 5) addition of operability requirements for the Rod Block Monitor, 6) incorporation of the Shutdown Margin limit, and 7) administrative type changes to Technical Specification reference numbers in order to match the ITS format. The changes are supported by Maximum Extended Load Line Limit and ARTS Improvement Program analyses (NEDC-32433P) and Equipment Out Of Service analyses (NEDC-32774P) performed by General Electric for the BFN plant. These changes are also consistent with the Unit 3 ITS requirements being implemented (TVA-BFN-TS-362).

There is no increase in the probability or consequences of an accident or malfunction of equipment from that previously evaluated in the SAR. The proposed activity does not create the possibility of a different type of accident or malfunction from that previously evaluated in the SAR. The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, it does not involve an unreviewed safety question.

UNIT 3 CYCLE 9 CORE OPERATING LIMITS REPORT

This safety evaluation supports the BFN Unit 3 Cycle 9 reload core design and addresses cycle specific revisions to the Unit 3 Core Operating Limits Report (COLR). The reload core design and licensing analyses for Cycle 9 were performed by General Electric (GE) with results documented in the Supplemental Reload Licensing Report. The GE analyses were performed in accordance with NRC approved methodology as described in GE licensing topical report GESTAR II (NEDE-24011-P-A-13). Operating limits for the cycle (i.e., Linear Heat Generation Rate, Minimum Critical Power Ratio, and Average Planar Linear Heat Generation Rate) are being incorporated into the Unit 3 COLR.

The Cycle 9 core is a Modified Control Cell Core design which includes 292 fresh GE13 fuel assemblies and 8 new Marathon control blades. Cycle 9 is the initial 5% power-uprate cycle for a BFN unit. This cycle will also implement the Average Power Range Monitor (APRM) Rod Block Monitor (RBM) Technical Specification (TS) / Maximum Extended Load Line Limit Analysis (MELLLA) [ARTS/MELLLA] Improvement Program which includes RBM design improvements, APRM/RBM Technical Specification improvements, and expansion of the allowable operating domain to include the MELLLA region of the power/flow map. Other operating flexibility options analyzed include Increased Core Flow, Final Feedwater Temperature Reduction, Feedwater Heaters Out Of Service, EOC Recirculation Pump Trip Out Of Service, and Turbine Bypass Valves Out Of Service. Consistent with NRC commitments, GE also performed a cycle specific Safety Limit MCPR analysis.

There is no increase in the probability or consequences of an accident or malfunction of equipment from that previously evaluated in the SAR. The proposed activity does not create the possibility of a different type of accident or malfunction from that previously evaluated in the SAR. The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, it does not involve an unreviewed safety question.

1998

SUMMARY OF SAFETY EVALUATIONS FOR FIRE PROTECTION REPORT REVISIONS

FIRE PROTECTION REPORT CHANGE NOTICE #98005

This change notice makes various changes primarily to the Appendix R safety shutdown program contained in the Fire Protection Report (FPR). These changes were identified as part of the corrective action for BFPER971873. The changes involve adding additional components to Section V, "Testing and Monitoring," of the Safe Shutdown Program and correcting minor discrepancies. Other changes consist of minor corrections, clarifications, and enhancements to the Fire Protection Plan and Fire Hazards Analysis section of the FPR.

None of the changes made under FPR Change Notice #98005 result in any adverse impact on the Appendix R Safe Shutdown Program, therefore, there is no adverse impact on nuclear safety. No unreviewed safety question is involved.

FIRE PROTECTION REPORT CHANGE NOTICE #98006

This safety evaluation is written in support of Fire Protection Report Change Notice #98006. The Safe Shutdown Program Section of the Fire Protection Report (FPR) Volume I has been revised to implement the Improved Technical Specifications (ITS) requirements. The current listing of Limiting Conditions for Operation (LCO) and Surveillance Instructions (SI) in the Technical Specifications are being changed to corresponding ITS LCOs and Surveillance Requirements (SRs). However, no changes in the content of the LCOs and SRs is being proposed. Some minor requirements (HPCI Speed Controller system flow verification) in Section V (Testing and Monitoring) have been deleted.

The administration of the Appendix R Testing and Monitoring Program under the requirements of the Appendix R Safe Shutdown Program as defined in Volume 1 of the Fire Protection Report meets the requirements of the Technical Specifications and therefore would not reduce the margin of safety as defined in the basis for any of the Technical Specifications. The surveillance requirements as imposed by the new ITS Program has been incorporated into the FPR (Volume I), and thus, the Appendix R Program is in compliance with the new ITS Program

None of the changes being made under FPR-98006 result in any physical changes to the plant which could alter the function or mode of operation of any plant system.

No unreviewed safety question is involved.

FIRE PROTECTION REPORT CHANGE NOTICE #98007

This safety evaluation is written in support of Fire Protection Report (FPR) Change Notice #98007. This FPR change provides corrective action for BFPER98011207 and BFPER98014543. Additionally, the compensatory measures in the Appendix R Safe Shutdown Program (contained in Volume I of the FPR) have been revised to remove reference to the Technical Specifications and clarify that Appendix R compensatory measures require the restoration of the equipment function(s) in seven days or make provisions for equivalent shutdown capability from other means.

BFPER98011207 identified discrepancies associated with the 120-V Instrument and Control (I&C) Bus equipment listing in the Appendix R Safe Shutdown Program. Basically all the equipment associated with any given alignment of the I&C Bus, necessary to ensure power available at the bus, was not listed in the Appendix R Safe Shutdown Program. This could result in a specific piece of equipment being removed from service which would render the I&C Bus unavailable without it being noticed that this specific piece of equipment was required for Appendix R, thus creating the potential to overlook the need to enter into an Appendix R Limiting Condition of Operation (LCO) which would be required if the I&C Bus was unavailable. Therefore, the necessary equipment for the required I&C Bus alignments has been added to the equipment listing in the Appendix R Safe Shutdown Program for Units 2 and 3. BFPER98014543 identified a discrepancy between the Technical Specification LCO (Compensatory Measure A) requirements and the Appendix R LCO (Compensatory Measure B) in that the Technical Specifications allow 14 days for High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) equipment unavailability while Appendix R only allows seven days. Since the Appendix R LCO is more stringent, the Appendix R Program was revised to eliminate the Technical Specification option and require that the Appendix R option be utilized. In addition, the compensatory measures have been changed to clarify Appendix R requirements associated with the unavailability of other Appendix R equipment. The references to Technical Specifications have been deleted to clarify that Appendix R compensatory measures require entering into a seven day Appendix R LCO when required equipment is unavailable to perform its required function. Two additional Safety Evaluation Reports (SERs) have also been added to the References Section in the Fire Protection Plan. These SER's were issued by the Nuclear Regulatory Commission (NRC) to approve the BFN Fire Protection Program.

None of the above proposed changes result in any physical or procedural changes to the plant which could alter the function or mode of operation of any plant system. Nor would any of the proposed changes adversely affect the safe shutdown capability of the plant. Therefore, no unreviewed safety question is involved.

1998

SUMMARY OF MAINTENANCE ACTIVITY SAFETY EVALUATIONS

WORK ORDER 97-011954-000

This maintenance work order is being performed to remove and reinstall the ductwork and supports associated with Standby Diesel Generator (DG) 3C in association with engine maintenance. Portions of the ductwork and the supports may be removed as required to obtain access for the maintenance activities and later reinstalled with the DG operable to minimize the unavailability of the DG during the 12 year preventive maintenance activities.

The removal of portions of the ductwork and supports will not affect the DG's design, functional requirements, or performance characteristics. The purpose of the ductwork is to redirect warm air exhausted from the generator ventilation grills away from electrical panels located near the generator. Ductwork to direct this warm exhaust air away from the panels will remain installed until the DG is removed from service and will be reinstalled prior to the DG being declared operable. As such, the proposed activities and temporary configuration of DG 3C and its associated ductwork do not adversely affect nuclear safety.

Work Order 97-0011954-000 is a maintenance work order controlled and processed in accordance with SSP-6.2, Maintenance Management System. The work order and the removal and installation of the generator ductwork do not differ from any operational characteristics in the SAR. The activity does not conflict or affect any process or procedures outlined, summarized, or described in the SAR. UFSAR Figure 10.12-6, a flow diagram showing DG building air flows, is indirectly affected, therefore a safety evaluation is required.

No unreviewed safety question is involved.

WORK ORDER 97-011954-001

This maintenance work order is being performed to remove and reinstall the ductwork and supports associated with Standby Diesel Generator (DG) 3B in association with engine maintenance. Portions of the ductwork and the supports may be removed as required to obtain access for the maintenance activities and later reinstalled with the DG operable to minimize the unavailability of the DG during the 12 year preventive maintenance activities.

The removal of portions of the ductwork and supports will not affect the DG's design, functional requirements, or performance characteristics. The purpose of the ductwork is to redirect warm air exhausted from the generator ventilation grills away from electrical panels located near the generator. Ductwork to direct this warm exhaust air away from the panels will remain installed until the DG is removed from service and will be reinstalled prior to the DG being declared operable. As such, the proposed activities and temporary configuration of DG 3B and its associated ductwork do not adversely affect nuclear safety.

Work Order 97-0011954-001 is a maintenance work order controlled and processed in accordance with SSP-6.2, Maintenance Management System. The work order and the removal and installation of the generator ductwork do not differ from any operational characteristics in the SAR. The activity does not conflict or affect any process or procedures outlined, summarized, or described in the SAR. UFSAR Figure 10.12-6, a flow diagram showing DG building air flows, is indirectly affected, therefore a safety evaluation is required.

No unreviewed safety question is involved.

WORK ORDER 98-002350-000

This safety evaluation is written in support of Work Order 98-002350-000 which allows for the temporary replacement of a General Electric (GE) Magne-Blast breaker with a Siemens vacuum type breaker in two compartments of 4-KV Unit Board 1B. The purpose of this work is to provide assurance that these two types of breakers are in fact interchangeable. The existing breakers in Compartments 8 and 10 of 4-KV Unit Board 1B will be removed and a single Siemens breaker will be inserted in each of these compartments (one at a time) to inspect for same fit, form and function. The breaker will only be operated in the test position in Compartment 8 (Condensate Booster Pump 1B) and will be operated in the test and connect position in Compartment 10 (RCW Pump 1B) but the load cables to the pump motors will be lifted to prevent any inadvertent operation of the pumps. The breaker will be operated locally (at 4-KV Unit Board 1B) and from the control room to ensure that the breaker functions correctly. This board (4-KV Unit Board 1B) is not a safety-related board nor are there any safety-related loads on this board.

The temporary unavailability of Condensate Booster Pump 1B and Raw Cooling Water Pump 1B or 4-KV Unit Board 1B does not jeopardize the operation of the plant nor place the plant in any unanalyzed condition.

No unreviewed safety question is involved.

WORK ORDER 98-005505-005

This safety evaluation addresses the effect of the configuration on the operability of the High Pressure Coolant Injection (HPCI) system. During scheduled work activities on the HPCI systems, the motor operator for valve 3-FCV-73-35, HPCI Pump Condensate Storage Tank (CST) Test Valve, was damaged. Steps in Word Order (WO) 98-005505-005 will remove the valve motor and operator from 3-FCV-73-35 valve in this condition (i.e., with no valve motor or operator). 3-FCV-73-35 will be restrained in the open position as a personnel safety measure, to ensure that the valve stem does not suddenly move outward if the HPCI system were to initiate. Since this valve will be restrained open and will not be able to close, the 3-FCV-73-36 valve, HPCI Pump CST Block Valve, and the 3-FCV-71-38 valve, RCIC Pump CST valve, will administratively be tagged closed via the Hold Order procedure. Removal of the valve motor and operator from 3-FCV-73-35 affects the operation of the HPCI system as described in Section 6.0 of the Final Safety Analysis Report (FSAR).

The time required for HPCI to achieve rated flow will not be affected by this activity, since the test return line will already be isolated when the system is actuated (i.e., flow to the vessel is not dependent on any closure time for valves in the test return line). No other components in the HPCI system which may affect the time to rated flow are impacted by this activity. Therefore, this activity will not reduce the margin of safety as defined in the basis for any Technical Specification. This change does not involve an unreviewed safety question.

WORK ORDER 98-009871-001

This safety evaluation addresses locking of a reactor recirculation Motor Generator (MG) set scoop tube while the reactor is in power operation. Locking of a scoop tube is occasionally necessary to allow maintenance on an MG set's associated speed control circuitry/equipment without affecting the MG set speed. Under normal conditions, speed demand to the recirculation MG sets is set manually by the control room operator. Automatic speed demands changes to the recirculation MG sets only occur under two plant process conditions, both of which result in MG set speed runback:

1. Whenever total Reactor Feedwater flow decreases to less than 20% the MG set speeds will be runback if greater than and limited to 28% speed.
2. Whenever any RFW pump's flow decreases to less than 20%, and reactor water level decreases to 27" narrow range, the MG set speeds will be runback if greater than and limited to 75% speed.

The full range of reactor recirculation MG set/recirculation pump speed variations, from runaway to maximum speed to MG set trip, have been analyzed previously. The scoop tube lock activity remains within these boundaries and therefore no margin of safety is decreased. This change does not involve an unreviewed safety question.

1998

SUMMARY OF
SAFETY EVALUATIONS
FOR
PLANT MODIFICATIONS

DESIGN CHANGE NOTICE A40675A

The safety evaluation for W17846B (see TVA BFN 1995 Annual Operating Report) was revised due to DCN A40675A which removes the task scoping document testing plan, PMT-261. The safety evaluation now reads "Testing performed after completion of the modifications will be according to General Construction Spec G-37...and verified to meet design requirements." The determination that no unreviewed safety question exists is still valid.

DESIGN CHANGE NOTICE D40257A

DCN D40257A revises drawing 3-47E200-11 (FSAR Figure 10.2-1b) to delete a notation pertaining to the type of cover on the skimmer surge tanks. The north skimmer surge tanks for Units 1, 2, and 3 have covers consisting of an aluminum frame topped with grating. The south tanks have a concrete shield plug for a cover. The aluminum frame with grating replaced the concrete plugs so that operations personnel would be provided visual and personnel access to the surge tank opening while manually adjusting the adjacent shut-off valve.

The Fuel Pool Cooling and Cleanup System is discussed in Sections 3.9.2 and 3.9.3 of the Technical Requirements Manual (TRM). The TRM does not discuss or address the concrete shield plugs for the skimmer surge tanks. The replacement of these concrete shield plugs with the aluminum frame and grating, therefore, would not be discussed. The Technical Specifications are not changed or altered by this proposed change. Therefore, this activity does not reduce the margin of safety as defined in the bases for any Technical Specification. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE D40678A

DCN D40678 was issued to correct discrepancies between Mechanical Control Diagram 0-47E610-90-2 and plant configuration. This change involves the resolution of drawing discrepancies with the Post Treatment Off Gas Radiation Monitoring System. The affected equipment is not safety related or quality related and performs a monitoring function only. The proposed change does not alter the initial design or function of the affected equipment. Consequently, the DCN does not create an unreviewed safety question.

DESIGN CHANGE NOTICE D40696A

DCN D40696A provides an alternate Mark No. for Units 1, 2, and 3 Raw Cooling Water (RCW) System valves and in addition implements a drawing discrepancy change on flow diagram 2-47E844-1 to depict valves 2-ISV-24-834A, B, and C as gate valves instead of globe valves. Gate valves are installed in the plant and are specified on the bill of material. This flow diagram appears in the UFSAR as Figure 10.7-1a Sheet 2. This safety evaluation supports the UFSAR figure change. This is a design document change only; no physical change is made in the plant. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE D40708A

The changes being made under DCN D40708A consist of revising wiring diagrams 0-45E506 (Main Single Line SG 1 151-KV SWYD) and 0-45N520-1 (161-KV Switchyard Connection Diagram). These wiring diagrams are revised to show the current transformer ratios that are specified on the relay information and setting sheets associated with 161-KV Bus Tie Breakers No. 924 and 928. There are no physical changes being made to the facility nor are there any changes which could adversely impact the operation of any plant system. This is a documentation change only and does not affect the availability of the 161-KV system as a qualified offsite power source, therefore, no unreviewed safety question exists.

DESIGN CHANGE NOTICE D40820A

DCN D40820A shows moving 1 and 3-RTV-027-0202A from upstream to downstream of the respective screen wash pump discharge check and shutoff valves. The tap for the root valve in the process line has always been at the current location as shown on 0-37W205-6.

The change to the flow drawings for Unit 1 and Unit 3 are required to make the drawings match the plant. There are no safety-relatedfunctions for the root valves and there are no safety implications in moving the location of the root valves on the drawings. Therefore, this change is safe from a nuclear safety standpoint, and no unreviewed safety question is involved.

DESIGN CHANGE NOTICE D40862A

During implementation of DCN W18298, 2-FE-69-101 was removed. The element was not utilized and, therefore, not reinstalled during the pipe replacement modification. Design Engineering coordination did not occur to ensure affected drawings were updated. BFPER971545 documents this drawing discrepancy. DCN D40862A provides the necessary changes to the affected drawing, which is also a UFSAR figure.

This DCN does not require any physical modification and, therefore, does not affect the Reactor Water Cleanup System's ability to maintain the reactor coolant pressure boundary and the primary containment pressure boundary. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE D40945A

The safety evaluation for DCN D40945A was initiated to correct the drawing discrepancies documented by PER 98-003947-000. This PER documents discrepancies between the 47W455 and 47W456 physical piping drawings series, the 47E812 and 47E813 series flow diagrams, and the installed configuration of the High Pressure Coolant Injection System (HPCI), System 073, and the Reactor Core Isolation Cooling System (RCIC), System 071, steam supply trap discharge piping. This safety evaluation is required because the following flow diagrams which appear in UFSAR must be revised.

HPCI flow Diagram	2-47E812-1	SAR Figure 7.4-1b Sheet 1
HPCI flow Diagram	3-47E812-1	SAR Figure 7.4-1b Sheet 2
HPCI flow Diagram	2-47E813-1	SAR Figure 4.7-1a
HPCI flow Diagram	3-47E813-1	SAR Figure 4.7-1c

The Technical Specification and Technical Specification Bases were reviewed for impact from these drawings changes. The drawing revisions will not affect the automatic actuation or change any controls for the High Pressure Coolant Injection (HPCI) Reactor Core Isolation Cooling or (RCIC) systems. Operation of the HPCI and RCIC systems are not changed and the design requirements of the HPCI and RCIC systems is unchanged. The drawing revisions do not change the description of any safety-relatedequipment. This DCN does not modify the function of any component. Therefore, the margin of safety is not reduced as defined in the basis for any Technical Specification document. The change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE D41004A

The safety evaluation is to support DCN D41004A. The spare shunt reactor for the Union 500-KV transmission line is being used in place of the C-phase. The C-phase shunt reactor is no longer available for service and has been removed. The shunt reactor is used to reduce transmission line voltage during light load conditions. A shunt reactor is required for each phase to provide voltage control. Placing the spare in service, for C-phase, will allow the shunt reactor to continue to be used to reduce transmission line voltage as necessary.

The Union 500-KV transmission line is one of seven 500-KV lines which provide offsite power to BFN. The shunt reactor is not safety related nor does it provide a safety-related function. The proposed change is associated with only one circuit of the offsite power system. The failure of the Union shunt reactor or the transmission line does not result in a complete loss of offsite power. The loss of the Union transmission line does not reduce the margin of safety as defined in the basis of any Technical Specification. Therefore no unreviewed safety question is involved.

DESIGN CHANGE NOTICE D41036A

This safety evaluation for DCN D41036 addresses Problem Evaluation Report (PER) 98-008628-000 that identified a drawing discrepancy associated with Mechanical Control Diagram 0-47E610-65-1. The PER states, "Mechanical Control Diagram 0-47E610-65-1 (CCD), indicated references to note 4, that FS-65-20 and FS-65-42 are 'no-flow' switches. However, the TRM (TR3.3.2 and Basis), UFSAR (Section 5.3.3.7), NESSD's for the FS's Calculation ED-Q0065-890227, O-SI-4.2.A-13 (A), Design Criteria BFN-50-7065 [Section 3.5.2(3)], 0-SIMI-65B, and 1-ARP-9-3B indicate that the switches have setpoints for 'low flow' not 'no flow'."

This activity revises Note 4 of Mechanical Control Diagram 0-47E610-65-1, to comply with issued design input and other design bases documents identified above, **FROM** "FS-65-20A or FS-65-20B turn the R-H control heater on when air is flowing and off *when air is not flowing.* FS-65-20 alarms on *no* flow. *TIS-55-12 is* a high-temperature protective *device* that turn the R-H heater control off when temperatures exceed 180° F. *XA-65-12* alarms on loss of heater power supply." **TO,** "FS-65-20 alarms on *decreasing* flow. *TS-65-12A and TS-65-13B are* high-temperature protective *devices* that turns the R-H heater control off when temperatures exceed 180° F. *XA-65-12B* alarms on loss of heater power supply." The revised note corrects a typographical error in sentence three.

Note 6 will also be revised to state, "*Component* abandoned in place." Since Note 6 was added to the subject drawing, the limit switches mentioned in the note have been abandoned in place by DCN W17317.

The subject flow switches are described in the Technical Specifications as providing a low flow control instead of a no flow control. Therefore, the proposed activity will not reduce the margin of safety as defined in the basis for any Technical Specification. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE D41099A

The safety evaluation for DCN D41099A provides the corrective action for Problem Evaluation Reports (PERs) 98-011127-000 and 98-012701-000. This activity changes six UFSAR figures. These changes are made to correct the drawing discrepancies and to make drawings agree with other UFSAR figures. There are no physical changes being made to the facility, nor are there any changes which could adversely impact the operation of any plant system. The drawings changes are to correct minor drawing errors or bring drawings into agreements with previous design changes.

The safety-related operation, function and response to events for components and systems affected by this change will not degraded. Therefore, based on these facts, this change will not involve an unreviewed safety question.

DESIGN CHANGE NOTICE D41124A

The safety evaluation for DCN 41124A revises drawings 2,3-47E803-1 to show sodium injection and sample valves 2,3-SMV-3-634, -635, -636 and –637 as normally open gate valves. A cautionary note is added to the drawings warning that these valves should not be closed while the ¼" probe line runs through the valves. To provide appropriate isolation of these sample/injection lines, the associated globe valves, 2,3-SMV-3-675, and 2,3-SHV-3-701 and -702, will be shown as normally closed. The respective unit's valve alignment checklist in the associated Operating Instructions will be revised to show these valve alignment changes. A UFSAR Change Request is submitted with this DCN to revise Figures 11.8-1 Sheets 1 and 3 in the UFSAR associated with these design basis drawings.

The affected feedwater sodium injection and sample lines are not used to maintain reactor coolant chemistry. The lines are outside the primary containment boundary, thus, their leakage does not affect the results of the valve Local Leak Rate Testing or containment leakage. Therefore, this change does not affect any margin of safety as defined in the Technical Specification or its bases. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE S38764C

This safety evaluation addresses DCN S38764C which provides additional mounting details for thermocouples which are located at the reactor vessel feedwater nozzles of each reactor vessel. These thermocouples (4 attached at each of the 6 nozzles) monitor the feedwater nozzle temperature to detect feedwater nozzle bypass leakage. The affected thermocouples are TE-3-238xn (where x is the specific feedwater nozzle designation A through F, and n is the thermocouple number, 1 through 4, on a particular nozzle). This temperature monitoring system was originally supplied by NUTEC under contract 829498. The system was installed in response to NUREG 0619 and uses the temperature readings at each nozzle to calculate nozzle bypass leakage. This DCN will not affect the location or function of these thermocouples as currently shown on TVA Drawing 0-47E600-91.

Revision C of this DCN removes the alternate mounting detail for the feedwater nozzle thermocouples for SAR Figure 7.8-3. The detail is maintained on a new drawing, but is not required to be presented on the SAR figure which is for reactor vessel thermocouple locations. Minor dimensional changes allow the mounting block to match the thermocouple. These thermocouples provide fatigue evaluation monitoring points and do not affect the operation of the feedwater system. The mounting details are basically unchanged and this change does not affect any possible failure modes. This change does not affect the operation of any equipment required for the safe shutdown of any unit and thus does not constitute an unreviewed safety question.

DESIGN CHANGE NOTICE S40606A

DCN S40606A revises the Fuel Pool Cooling and Cleanup System's (FPCS) piping design pressure starting at the FPCS pumps' suction, through the FPCS heat exchangers, filter demineralizers, and out to the fuel pool discharge diffusers on all three units from 150 psig to 200 psig. This change is being made as corrective action for BFPER970946. This problem evaluation report identified drawing errors on the flow diagrams which depicted design pressure for the aforementioned piping as 150 psig which is the normal operating pressure. Notes on the flow diagrams properly identified the design pressure as 200 psig which agrees with the original General Electric Company design specification.

The change in piping design pressure implemented by this DCN corrects output documents to reflect original piping design specifications. This change is a design document change only and does not affect the ability of the FPCS piping to provide primary or secondary containment during any accidents or transients. No field work is required for this change.

No unreviewed safety question is involved.

DESIGN CHANGE NOTICE S40742A

This safety evaluation is written to support revisions being made to four design drawings that appear in the UFSAR, the creation of one new drawing that will be substituted for an existing drawing that appears in the UFSAR, and revision of Table 1.3-1, "List of UFSAR Engineering Drawings", Table 1.3-2, "Engineering Drawing Cross-Reference List", and the List of Illustrations in Section 2.0. Drawings 0-10E201-01, 0-10E300, 0-10E400, and 0-31E400-1 are being revised so that the drawings will accurately reflect existing plant features. Features which may have previously existed on the plant site but have been removed from the plant in years past and are still shown on the drawings will be removed also. Drawing 0-10E201-01, "Location of Structures", will be removed from the UFSAR, and will be replaced by drawing 0-10E201-07, "Location of Principal Plant Structures." This is due to the excessive amount of detail found on drawing 0-10E201-01, and the difficulty in keeping the drawing current. Drawing 0-10E201-01 has been prepared in accordance with guidelines found in USNRC Regulatory Guide 1.70. Drawing 0-10E201-01 will not be voided or removed from the TVA drawing system. It will continue to exist and will be revised as required to keep it current with future changes made to the plant site. It will no longer, however, be part of the UFSAR. The changes being made to the drawings, the creation of a new drawing, and changes being made to the UFSAR figures, tables, and lists are documentation only in nature and do not require any modifications or field work. They are, therefore, being made per DCN S40742A.

These UFSAR changes do not constitute an unreviewed safety question because the changes do not affect the safety function of any system, structure, or component.

DESIGN CHANGE NOTICE S40792A

The safety evaluation for DCN S40792A was written to evaluate revisions to the environmental data drawings to reflect changes to the Unit 2 plant environmental conditions. Most of the changes to the environmental data drawings are due to the 5% uprate of Unit 2, an increase in reactor thermal power from 3293 MWt to 3458 MWt. The 5% uprate in core thermal power results in the following changes to plant environmental conditions:
- 5% increase in the normal radiation doses for rooms 0,00,10,11,16, 17A&B, 19 and 20.
- 8% increase in the post-Loss of Coolant Accident (LOCA) radiation dose in Room 22 (Standby Gas Treatment System building).
- Revised containment pressure and temperature responses for the design basis LOCA (recirculation line break) and main steam line breaks inside primary containment.
- Revised pressure, temperature, and humidity responses for high energy line breaks outside of primary containment.

The NRC has approved the license amendment request for operation at a core thermal power of 3458 MWt as documented in the Power Uprate Safety Evaluation Report dated September 8, 1998 (L44 980917 001).

DCN S40792A also revises the environmental data drawings to reflect the following changes, which are not related to the 5% uprate in core thermal power:
- The flood level is increased in Room 10 (Reactor Water Cleanup [RWCU] Pump Room) based on the flooding evaluation performed for the RWCU piping reroute (DCN W17811).
- New flood levels are specified for Rooms 19 and 20 since the doors to these rooms are not required to be closed during plant operation.
- A post-LOCA temperature is specified for Room 7 (Main Steam Valve Vault) since certain cables/equipment in third room is required to operate long term for post accident monitoring.
- Normal/abnormal temperatures for the RWCU pipe trench in Room 8 are specified separate from the general floor area on the 565' elevation since temperature measurements during plant operation indicate the temperatures in the pipe trench are higher than the general floor area temperature.

The proposed drawing revisions reflect changes to plant environmental parameters, which do not affect the bases for the Technical Specifications. Therefore, the proposed drawing revisions will not reduce the margin of safety as defined in the basis for any Technical Specification. This change does not involved an unreviewed safety question.

DESIGN CHANGE NOTICE S40793A

This safety evaluation for DCN S40793A addresses revisions to the environmental data drawings to reflect changes to the Unit 3 plant environmental conditions. Most of the changes to the environmental data drawings are due to the 5% uprate of Unit 3, an increase in reactor thermal power from 3293 MWt to 3458 MWt. The 5% uprate in core thermal power results in the following changes to plant environmental conditions:

- 5% increase in the normal radiation doses for rooms 0,00,10,11,16, 17A&B, 19 and 20.
- 8% increase in the post-Loss of Coolant Accident (LOCA) radiation dose in Room 22 (Standby Gas Treatment building).
- Revised containment pressure and temperature responses for the design basis LOCA (recirculation line break) and main steam line breaks inside primary containment.
- Revised pressure, temperature, and humidity responses for high energy line breaks outside of primary containment.

The NRC has approved the license amendment request for operation at a core thermal power of 3458 MWt as documented in the Power Uprate Safety Evaluation Report dated September 8, 1998 (L44 980917 001).

DCN S40793A also revises the environmental data drawings to reflect the following changes, which are not related to the 5% uprate in core thermal power:

- The flood level is increased in Room 10 (Reactor Water Cleanup [RWCU] Pump Room) based on the flooding evaluation performed for the RWCU piping reroute (DCN W17811).
- New flood levels are specified for Rooms 19 and 20 since the doors to these rooms are not required to be closed during plant operation.
- A post-LOCA temperature is specified for Room 7 (Main Steam Valve Vault) since certain cables/equipment in third room is required to operate long term for post accident monitoring.
- Normal/abnormal temperatures for the RWCU pipe trench in Room 8 are specified separate from the general floor area on the 565' elevation since temperature measurements during plant operation indicate the temperatures in the pipe trench are higher than the general floor area temperature.

The proposed drawing revisions reflect changes to plant environmental parameters, which do not affect the bases for the Technical Specifications. Therefore, the proposed drawing revisions will not reduce the margin of safety as defined in the basis for any Technical Specification. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE S40868A

DCN S40868A provided changes to BFN System and General Design Criteria Documents. The changes revised existing criteria to make them applicable both for the pre-uprated BFN unit(s) (rated reactor thermal power of 3293 MWt) and for the uprated BFN unit(s) (rated reactor thermal power of 3458 MWt). The Design Criteria Document (DCD) revisions did not cause changes to be implemented in plant hardware or in plant procedures. The DCD changes set forth design input information which was to be used by other activities which supported implementation of power uprate on BFN Units 2 or 3.

At the time this DCN was issued, the proposed change was determined to involve an unreviewed safety question (USQ) as defined in 10 CFR50.59. Basis for determination that a USQ existed was that some of the revised design criteria documents reflected the increased reactor thermal power level and that the calculated consequences of certain design basis accidents increase slightly at the uprated power. A description of the proposed changes which resulted in the determination of a USQ condition was included in a license amendment request which had been submitted to the NRC for review. Subsequently, the NRC issued Unit 3 Technical Specification Amendment 214. Documentation that this USQ was reviewed and approved by the NRC was included in the parent DCN package (DCN S40868A) after NRC issuance of the Power Uprate Safety Evaluation Report.

DESIGN CHANGE NOTICE S40876A

DCN S40876A revised thirteen drawings to reflect details that are affected by the 5% uprate of BFN Unit 3, an increase in reactor thermal power from 3293 MWt to 3458 MWt. The DCN required the changed drawings to be put on Administrative Hold and issued prior to Unit 3 restart as part of the implementation of power uprate (U3C9).

At the time this DCN was issued, the proposed change was determined to involve an unreviewed safety question (USQ) as defined in 10 CFR50.59. Basis for determination that a USQ existed was that some of the revised drawings included the reactor thermal power level and that the calculated consequences of certain design basis accidents increase slightly at the uprated power. A description of the proposed changes which resulted in the determination of a USQ condition had been submitted to the NRC for review. Subsequently, the NRC issued Unit 3Technical Specificationamendment 214. Documentation that this USQ was reviewed and approved by the NRC was included in the parent DCN package (DCN S40876A) after NRC issuance of the Power Uprate Safety Evaluation Report.

DESIGN CHANGE NOTICE S40894A

This safety evaluation is written in support of DCN S40894A. This DCN documents the service test duty cycles for the 250-V DC Unit Batteries 1, 2, and 3, the 250-V DC Shutdown Batteries (control power supplies) SB-A, SB-B, SB-C, SB-D, and SB-3EB, and the 125-V DC Diesel Generator Batteries, A,B,C,D, 3A, 3B, 3C, and 3D. These are all the safety-related batteries. The service tests are specified in the Improved Technical Specifications (ITS) battery surveillance requirement SR 3.8.4.3. The service test duty cycles are added to the battery single line drawings by this DCN. Additionally, changes are made to the Final Safety Analysis Report (FSAR) and design criteria to make corrections and provide consistency with the ITS.

The duty cycles for the battery service test are constructed based on calculations which include all required loads which my operate for the 30 minute duty cycle duration. The battery service test duty cycles were constructed in accordance with Institute of Electrical and Electronic Engineers (IEEE) Standards 450-1995 and 485-1983. The margins added for temperature ranges, load growth and aging will provide adequate battery capacity to meet the battery duty cycle throughout its service life.

Rated battery capacity is based on 210-V for the 250-V DC batteries and 105-V for the 125-V batteries. Battery performance tests which determine capacity as a percentage of rated capacity are performed to these minimum voltage limits. The minimum acceptable voltages for the service tests are based on voltages determined for the actual duty cycle. Thus the proposed activities will not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, no unreviewed safety question is involved.

DESIGN CHANGE NOTICE S40910A

This safety evaluation is written for DCN S40910A which revises drawings 1/2/3-47E610-90-1 by adding a note to reflect that use of the alpha subtraction channels is optional for the System 90 (Radiation Monitoring) Air Particulate Monitors. The procedures that govern the use of the monitors were revised as a result of an April 5, 1995 Safety Assessment/Safety Evaluation #90-9504-003, that concluded that the alpha was at such a low level that it was not needed by the monitors and the nonuse of the alpha subtraction channels had no impact on the safety of the plant.

The Air Particulate Monitors are a diverse subsystem of the Area Radiation Monitoring System (known as Additional Area Radiation Monitoring System) and is used to assist in the detection of abnormal radiation levels at various locations throughout the plant. They are described in the FSAR, but not in the Technical Specifications/Improved Technical Specifications. They do not initiate any automatic actions, they are not classified as safety related and the use of the alpha subtraction channel was not discussed in the safety analysis of the plant. Therefore, an unreviewed safety question does not exist.

DESIGN CHANGE NOTICE S40979A

The safety evaluation for DCN S40979A was written to revise ten drawings to reflect details that are affected by the proposed 5% power uprate of BFN Unit 3, an increase in reactor thermal power from 3293 MWt to 3458 MWt. The DCN requires the revised drawings to be place on Administrative hold and used prior to Unit 3 restart as part of the implementation of power uprate (U3C9). The proposed drawing revisions change certain Category 1 drawings to reflect details affected by the power uprate of BFN Unit 3. For the affected mechanical drawings, companion Appendix J and In Service Inspection (ISI) boundary drawings are also changed. The drawing revisions reflect changes which are required to support or which result from the proposed 5% thermal power uprate of BFN Unit 3. A requirement for issuance of these drawings is receipt of an approved Safety Evaluation Report of the proposed power uprate on BFN Unit 3.

Applicable Technical Specification (TS) changes, including changes to Bases Section, have been proposed and referenced. All the proposed Technical Specification changes were reviewed and none of the proposed DCN activities require a revision to the description of safety margins as defined in the Technical Specification Bases. Therefore, this change does not reduce the margin of safety and no unreviewed safety question is involved.

DESIGN CHANGE NOTICE S40985A

DCN S40985A revised or created nine drawings to reflect details that are affected by the 5% uprate of BFN Unit 3, an increase in reactor thermal power from 3293 MWt to 3458 MWt. The DCN required the changed drawings to be put on Administrative Hold and issued prior to Unit 3 restart as part of the implementation of Power Uprate (U3C9). At the time of issue for this DCN, the NRC had approved the uprate of BFN Unit 3. The activity was compared against the criteria of 10 CFR 50.59 and was determined not to involve an unreviewed safety question.

DESIGN CHANGE NOTICE S41013A

This safety evaluation is written in support of DCN S41013A which revises the design temperature of the Containment Atmosphere Dilution (CAD) System piping located in the drywell from 231°F to 281°F. The correct design temperature of 281° F is taken from the Design Criteria BFN-50-7064A and Design Calculation MD-Q0084-880104. This change is being incorporated for all three units' respective drawings at Browns Ferry. These drawings (1,2 and 3-47W862-1) appear in the SAR as Figures 5.2-7 Sheet 1 for Unit 1, 5.2-7 and Sheet 2 for Unit 2, and 5.2-7 Sheet 3 for Unit 3. The CAD system is required, in the event of a design basis Loss of Coolant Accident, to inject nitrogen into the drywell in order to maintain the oxygen concentration in the drywell at a level which would mitigate any potentially volatile mixtures of hydrogen/oxygen. This administrative change will have no effect on the capabilities of the CAD system, the response of the CAD system to any documented accident scenario, nor will it introduce any new credible failure modes for the CAD system.

The requirements of the CAD system as noted in the Technical Specifications will be unaffected by the drawing change. The CAD systems operational parameters will also be unaffected, therefore there is not a reduction in the margin of safety as defined in the basis of any Technical Specification. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE S41120A

The safety evaluation for DCN S41120A provides the design documentation to restore the design pressure/temperature line mark number designation, at 3-FCV-73-44, on flow diagram 3-47E812-1 and associated UFSAR Figure 7.4-1b Sheet 2. This change meets the original component design specifications and is consistent with line mark number designation on the Unit 2 flow diagram, 2-47E812-1. Also, UFSAR text Section 6.4.1, page 6.4-6 is revised to clarify the description of the High Pressure Coolant Injection (HPCI) support system function and its association with the HPCI flow controller. The intent of the affected UFSAR sections is not changed by the DCN.

HPCI is an Emergency Core Cooling System provided to assure that the reactor is adequately cooled to limit fuel cladding temperature in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel.

The DCN does not affect the function of the HPCI system or any other system to mitigate their applicable design basis events. The changes do not require a physical modification to equipment or to the plant configuration. This evaluation supports a clarification change to a UFSAR paragraph located on page 6.4-6 of UFSAR Section 6.4.1. Statements concerning the HPCI support system function for operation of the turbine stop and control valves are edited for clarity. Also, ambiguous statements associated with achieving HPCI flow were removed. The change does not alter the operation of HPCI components, but provides a clear description of HPCI flow control logic that maintains HPCI at a constant flow rate.

No other safety-related equipment is affected by this change. This change does not modify the function or operation of any components of the HPCI system. Therefore, the Technical Specifications are not affected, nor is the margin of safety as defined in these documents reduced. This change does not involved an unreviewed safety question.

DESIGN CHANGE NOTICE T28024A

DCN T28024A replaces three obsolete SMA-2 triaxial strong-motion accelerographs in the reactor building and diesel building with SSA-1 digital triaxial strong motion accelerographs. Both perform the same function of detecting and recording ground motion during earthquakes and sending an alarm signal to the control room. This modification does not alter the existing configuration of the seismic monitoring system at BFN. The replacement is recommended per TVA Corporate Engineering in the Strong Motion Seismic Instrumentation Report. The replacement provides capability in the future to take the advantage of the new, less stringent plant shutdown criteria made available through Electric Power Research Institute (EPRI).

The margin of safety is not reduced because this activity affects only the Seismic Monitoring System which is not a safety-related system and which has no impact on the margin of safety for any system specified in the Technical Specifications. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE T36529B

The safety evaluation for T36529B (see TVA BFN 1997 Annual Operating Report) was revised in accordance with BFPER980368 to address the effect on electrical loads due to horsepower changes of the motor being installed on valve 2-FCV-75-25. The safety evaluation also referred to the change as a non-significant change to the UFSAR. Non-significant changes are no longer defined in the safety assessment/safety evaluation process. This was noted during review by the 10CFR50.59 Safety Assessment/Safety Evaluation Committee. The conclusion that no unreviewed safety question exists is not affected by this revision.

DESIGN CHANGE NOTICE T39698A

DCN T39698A modifies the air supply to the actuators of the Unit 3 Reactor Feedwater Pump (RFP) discharge check valves, FCVs 3-92, 3-93, and 3-94 to allow testing of the actuator and use of the actuator to "bump" the valve disc in the closed direction on demand. This safety evaluation addresses a change to UFSAR Figure 7.8-1, Sheet 3, due to the recent addition of TVA drawing 3-47E610-3-1 to the UFSAR. This drawing was not in the UFSAR at the time this DCN was originally issued; therefore, the safety assessment was revised and a safety evaluation was created.

There are no changes which could adversely impact nuclear safety. The changes will make operation of the non-return valves more reliable so that they close when required by the operating conditions. The physical changes do not involve any safety-related equipment and cannot cause any component to function in a different manner than it already functions. No unreviewed safety question exists.

DESIGN CHANGE NOTICE T39737A

DCN T39737A deleted pressure switches 3-PS-2-5B and 3-PS-2-5D and removed pressure indicating switches 3-PIS-66-21A and 3-PIS-66-21B and temperature indicating switches 3-TIS-66-22A and 3-TIS-66-22B from the electrical trip interlock logic of the steam jet air ejectors.

The deletion of pressure switches 3-PS-2-5B and 3-PS-2-5D was performed by physically removing the components and by directly connecting the control circuits on both sides of their contacts to each other. The removal of pressure indicating switches 3-PIS-66-21A and 3-PIS-66-21B and temperature indicating switches 3-TIS-66-22A and 3-TIS-66-22B from the electrical trip interlock logic of the steam jet air ejectors was performed by disconnecting the contacts of control relays 3-RLY-66-R1, 3-RLY-66-R2, 3-RLY-66-R3, and 3-RLY-66-R4 from the interlock logic. The contacts of the control relays used for alarms will remain intact.

Neither the steam jet air ejectors nor the Off Gas System are classified as safety related. Deletion of the pressure switches 3-PS-2-5B and 3-PS-2-5D is recommended as part of the SCRAM reduction program. Removal of the trip logic associated with pressure indicating switches 3-PIS-66-21A and 3-PIS-66-21B and temperature indicating switches 3-TIS-66-22A and 3-TIS-66-22B are acceptable based on a letter dated May 31, 1996, to H. L. Williams, Browns Ferry Nuclear Plant Engineering and Materials Manager, from Dale E. Porter, General Electric Engineering Manager, which says the isolation function is no longer necessary.

The affected components of the proposed activity does not affect the sampling requirements described in the bases of Section 3.6.B/4.6.B of the Technical Specifications.

No unreviewed safety question is involved.

DESIGN CHANGE NOTICE T39980A

DCN T39980A installs a lightning strike prevention system to protect the BFN Off Gas Stack (BFN-0-STRU-303-STACK) and associated equipment and removes the existing air terminals. The strike prevention system consists of a dissipation array structure (DAS) mounted to the top of the stack and spline ball ionizers installed on the stack balconies. The DAS is designed to prevent lightning strikes to the stack and associated damage to site security system electronics located in the area of the stack.

The DAS and spline ball ionizers are a non-safety-related lightning prevention system which is mounted on a safety-related Category I structure, the Off Gas Stack. The installation of the DAS and spline ball ionizers do not affect the performance of the Off Gas Stack, the Standby Gas Treatment System or any other safety-related system, structure or component. The DAS and spline ball ionizers are designed for a 250 mph tornado wind which is above the wind speed at which the top part of the stack fails. Therefore, the lightning prevention system will fall with the top part of the stack and not become a separate missile which is a previously analyzed condition.

This change is to be performed with the reactor and drywell heads and shield plugs in place, affording the maximum protection of the reactor in the event the helicopter or DAS were to drop on the reactor building. However, the helicopter striking the reactor building is not considered a credible event due to the restricted flight plan. This change is safe from a nuclear safety standpoint and no unreviewed safety question is involved.

DESIGN CHANGE NOTICE T40075A

This safety evaluation is to support DCN T40075A. The DCN replaces forebay level loops 0-L-27-121B and -122B. These loops provide input to annunciation for forebay level and differential level between the forebay and warm water channels. The Level Element (LE) and Level Indication Transmitter (LIT) combination, for each loop, will be replaced with a single unit and new power supplies will be installed on Panel 2-9-20. The power supplies will receive power from the Unit 2, 120-V AC Unit Preferred Bus. This modification does not change the function of these loops. This modification will make the forebay loops identical to the warm water channel loops, 0-L-27-212A and -122A, which were revised by DCN T38873A. These instruments do not initiate or mitigate any accident as described in UFSAR Chapter 14.

The forebay level does not provide a safety-related function and forebay level is not a Technical Specification requirement. The Technical Specification requirements are associated with the ultimate heat sink temperature and Wheeler Reservoir level (LS-23-75A &B). The instrumentation power source has been moved from the 120-V AC I&C Bus, which is safety related, to the 120-V AC Unit Preferred Bus, which is a highly reliable non-safety-related source which is the same as other forebay instrumentation. Changing the forebay level instrumentation does not reduce the margin of safety as defined in the basis for any Technical Specification. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40211A

This safety evaluation addresses the replacement of existing BFN Unit 3 Equipment Core Cooling System (ECCS) suction strainers with larger, higher debris capacity strainers. The new strainers employ a passive-type design that does not require any operator action to ensure an uninterrupted suction flow to the ECCS Systems. The new strainers have an open flow area approximately 25 times larger than the existing strainers and, therefore, accommodate more debris loading. A larger debris loading capacity allows the new strainers to maintain required ECCS pump net positive suction head (NPSH) requirements per current design basis analysis. The margin of safety as described in the Technical Specification is not impacted by this modification. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE T40231A

This safety evaluation was included in the 1997 Annual Operating Report. However, failure modes for the 24 VDC power supplies to the Analog Trip Units (ATUs) identified in BFPER960378 were not considered in the design of the Main Steam Relief Valve (MSRV) auto actuation logic (Stage 1 of DCN T40231A). The failure modes for the 24 VDC power supplies for the ATUs have been analyzed and design revised to eliminate inadvertent opening potentials for the MSRVs by DCN F40453A. This is accomplished by eliminating logic contacts contained solely in Panel 2-9-81 for the 1125 psig setpoint MSRVs. With the new logic, the need for two of the slave trip units and associated relays in Panel 2-9-81 was eliminated and have been deleted from the design. The determination that no unreviewed safety question exists is still valid.

DESIGN CHANGE NOTICE T40232A

Stage 1 of this DCN (T40232A) involves the Main Steam Relief Valves (MSRVs). The MSRVs are experiencing an industry wide phenomena related to mechanical actuation setpoint drift. The setpoints are drifting high by more than the Technical Specifications (TS) allowed ±1% tolerance due to growth of spinel oxide corrosion of the pilot seat/seat interface.

The recommended fix from the Owners group is to use safety grade pressure sensors (in a non-safety-related function) to actuate the MSRVs during inservice pressure transient events in the relief mode. This method of automatic opening of the MSRVs permits application of the full main steamline pressure to brake the corrosion bonds that may have developed between the pilot seat/disc interface. When the Relief Mode is actuated the setpoint spring preload is removed from the pilot disc, and a rapidly applied full differential pressure (force) is seen across the pilot disc. This alternate actuation is capable of opening the MSRVs which have pilot seat/disc bonding with a high degree of confidence.

The key advantages to this approach are as follows:

- It offers a reliable backup means to MSRV setpoint spring actuation.
- It uses a safety grade transmitters arranged in a similar manner to that currently approved for application in the BWR/6 plants.
- It provides an externally sensed/power automatic actuation of the MSRVs.
- It improves the response of the relief valves by assisting actuation performance thus lessening the effect of corrosion induced bonding between the pilot disc and seat.
- It does not impact the ASME Code or licensing basis while reducing Licensee Event Reports due to inservice results.

This modification will provide defense in depth for the opening of the MSRVs, but no credit will be taken for the logic from a safety-related functional standpoint and it is not relied upon to meet Technical Specification functional requirements for the MSRVs.

The requested change will be implemented as follows:

- Existing Anticipated Transient Without SCRAM (ATWS) loops 3-P-3-204A, -204B, -204C and -204D will be used to derive the pressure input to the new MSRV Actuation Logic.
- Existing Master Trip Units (MTU) 3-PIS-3-204A, -204B, -204C and -204D will have Slave Trip Units (STU) attached (6 total). This will allow the setpoint logic to be established. Of the six additional STUs four will be set at 1135 psig, one at 1145 psig, and one at 1155 psig. These STUs will drive additional relays that will actuate at the same setpoints. The output of these will form a two-of-two times one logic that will energize 13 new relays that will have direct contact to the MSRV trip solenoids.
- The addition of these STUs and associated relays will be in the Aux Instrument Room Panels 3-9-81 and 3-9-82 (ATU Cabinets). The addition of the 13 MSRV relays will be in the Aux Instrument Room Panel 3-9-30. Cabling will be required between Panels 3-9-81, 3-9-82 and 3-9-30 which are located in Unit 3 Aux Instrument Room. For MSRV relays associated with MSRVs 3-PCV-1-31, -179 and -180, cabling will be required from Panel 3-9-30 to Panel 3-9-3 in the Main Control Room (MCR). Cables will be routed either through exiting cable trays or installation of new conduits which will be supported in accordance with 0-48B805 typical drawing series.
- The electrical setpoints will be adjusted to correspond to the new MSRV mechanical actuation setpoints which will be increased as part of the Unit 3 power uprate. The new MSRV electrical setpoints will be (1135, 1145 and 1155 psig) without regards to instrument loop accuracies. The current instrument loops have a ±23.81 psig inaccuracy. This is above the ±1% Technical Specification tolerance, but there is a

Technical Specification change submitted to the NRC to change the tolerance to ±3%. This Technical Specification change must be approved prior to implementation of this DCN. With the approval of the Technical Specification change, loop inaccuracies will fall within the ± 3% Technical Specification limit and therefore are acceptable. No Credit is taken for the new logic to meet any Technical Specification requirements.

- An inhibit handswitch (3-XS-1-202) will be added to Panel 3-9-3 in the MCR to allow operations to inhibit the logic from actuating should the need arise. An annunciator window will be added to 3-XA-55-3D to annunciate when the inhibit switch is in the inhibit position. To accomplish the inhibit logic, two additional relays will be required. One in each Aux Inst Room Panel 3-9-81 and 3-9-82. Cabling will be required from Panel 3-9-3 to Panels 3-9-81 and 3-9-82. Cables will be routed either through exiting cable trays or installation of new conduits which will be supported in accordance with 0-48B805 typical drawing series.

Stage 2 of this DCN involves the Automatic Depressurization System (ADS). Changes to the annunciator logic for the ADS Logic Bus A and B Inhibit Switch annunciation are required for corrective action for Problem Evaluation Reports BFNPER961230 and BFPER931764. This modification changes annunciator windows 18 and 31, and adds window 1 on 3-XA-55-3C, Panel 3-9-3. The "RCIC RELAY LOGIC POWER FAILURE" annunciator currently located in Window 31 will be moved to spare Window 1. This change will eliminate a unit difference which currently exists between Units 2 and 3. This move will also allow the ADS system annunciators to be placed in Window 18 and 31, on 3-XA-55-3C, which are the same locations as Unit 2. This arrangement will ensure annunciation is received separately for the ADS Logic Bus A or B inhibit handswitch being placed in the inhibit position, while also eliminating unit differences.

UFSAR Section 4.4.5 (Description of Nuclear System Pressure Relief Valves) will be revised to describe the new MSRV Auto Actuation Logic method of operation.

This change is acceptable from a nuclear safety standpoint. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE T40354A

DCN T40354A replaces the Reactor Feedwater (RFW) minimum flow valves with new valves which have similar operating characteristics, but which provide better shutoff and are designed to withstand the severe service conditions of high differential pressure and temperature. The valves will receive the same opening and closing signals, are controlled by air and will fail to the same (open) position on loss of air or power.

There are no design basis accidents for which the minimum flow valve is required to operate. Neither replacement of the valve nor operation with the new valve can cause an accident or malfunction of equipment important to safety. Installation of a larger minimum flow valve with slightly increased flow capacity does not change the function of the valve. It receives the same opening and closing signals and provides the same control room position indication. The new valve fails open on loss of air or power exactly as the existing valve. This change does not affect any nuclear safety-related components and is acceptable. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE T40424A

This DCN includes the turnkey design, procurement, construction, testing, and turnover by the contractor of one 16-cell mechanical draft cooling tower similar to the four existing cooling towers and will be a replacement for cooling tower #3 which was destroyed by fire on May 23, 1996. The work scope includes widening the existing concrete pad foundation, construction of one cooling tower, piping connections to existing water supply piping stubs, electrical connections to existing power supplies, grounding grid and control circuits for new fan motors, lighting, ground, and lightning arrestors. Also included is a fire protection design for the new tower consisting of fusible wire installed around each fan and motor, connected such that each circuit encompasses a pair of fans. A fire in any zone shuts down all the fans and alarms through a new local panel. The DCN scope also includes installation of a new temperature indicator on the tower cold water basin output, tower testing, and operational turnover.

Operation of the cooling tower has no affect on Technical Specifications required systems and thus cannot reduce the margin of safety defined in the bases of the Technical Specifications. This change is safe from a nuclear safety standpoint and no unreviewed safety question is involved.

DESIGN CHANGE NOTICE T40503A

This safety evaluation is written to assess a change to the Residual Heat Removal (RHR) System and the High Pressure Coolant Injection (HPCI) System. A ¾" test connection (consisting of piping, isolation valves, and supports) will be installed on the downstream side of valve 3-FCV-74-07 and on both the upstream and downstream sides of valve 3-FCV-73-16. The RHR valve, 3-FCV-74-07, is located on the pump minimum flow line at elevation 543' in the southwest quadrant of the Unit 3 reactor building. This section of the RHR System is an extension of the primary containment pressure boundary. The HPCI valve, 3-FCV-73-16, is located on the HPCI turbine steam supply line at elevation 527' in the Unit 3 HPCI room. Both the RHR and HPCI Systems are safety related and seismic Class I. The test connections will be used for differential pressure testing across valves 3-FCV-73-16 and 3-FCV-74-07 as part of the long term Motor Operated Valve (MOV) program associated with NRC Generic Letter 96-05.

The UFSAR currently includes RHR and HPCI System flow diagrams for Unit 2, but not for Unit 3. Since DCN T40503A adds test connections to the RHR and HPCI Systems in Unit 3 only, implementation of DCN T40503A will result in differences between the Unit 2 and Unit 3 system configurations. As a result, a Sheet 2 will be added to UFSAR Figure 7.4-1b and a Sheet 3 will be added to Figure 7.4-6a to include the Unit 3 flow diagrams for the HPCI and RHR Systems, respectively. Table 1.3-1, which identifies all the engineering drawings in the UFSAR, is also being revised to reflect the new sheets added to the existing figures. Adding this information to the UFSAR and adding the actual test connections to the Unit 3 RHR and HPCI Systems has no affect on the SAR with respect to the described safety functions of these systems, or to the ability to perform these functions. As such, these changes do not constitute an unreviewed safety question.

DESIGN CHANGE NOTICE T40604A

DCN T40604A will add a moisture trap down stream of drain valve in the control air dryer prefilter to remove excess water from the system. The drain piping will be routed to an existing equipment drain. The drain valves are UNIDs 2-DRV-32-1685 and 3-DRV-32-1685. These valves are non-quality related. Since installation of new control air compressor "G", excess water has been accumulating in the prefilters 2 and 3-FLT-32-1 requiring frequent draining. The moisture trap will allow the water to be removed without requiring operator attention. The prefilter is performing its design function to remove excess water upstream of the dryer. Additionally, a drain line may be installed upstream of the prefilter in Unit 2 as a low point drain.

DCN F40687A added new scope to DCN T40604A. This additional scope installs vent valves with an orifice on the moisture traps for the service air receiver tank, three control air receiver tanks, and the Unit 2 and 3 control air dryers.

Neither the Control Air System nor the Service Air System piping components in the turbine building are required to mitigate any accident postulated in the BFN UFSAR. The addition of the vent valves will allow operations to service the system and will provide a reliable, functional system which will perform as required. This will ensure air quality to the users which are supplied during normal operation. Therefore, an unreviewed safety question does not exist.

DESIGN CHANGE NOTICE T40605A

This change will install a debris filter and associated equipment and its corresponding backwash line in the Condenser Circulating Water (CCW) inlet pipe to the 3A1 condenser inlet waterbox. The debris filter is being installed to eliminate the accumulation of debris in the condenser 3A1 waterbox. This modification will provide a means of collecting and removing the debris before it enters the condenser waterbox, therefore improving the reliability and operability of the 3A condenser.

The CCW System, Demineralized Water System, Compressed Air Station Service System, and associated electrical power supplies are not nuclear safety-related systems. In addition, this modification does not directly affect the operation of any nuclear safety function or impact a balance of plant system in such a manner that it would affect a nuclear safety-related system. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE T40664A

DCN T40664A revises the mechanical setpoint for the Main Steam Relief Valves (MSRVs) to the following values and provides the pressure and flow capacities for revision to the MSRV Code Data Plates to accommodate the power uprate program. The new pressure relief values are as follows:

- 4 valves originally @ 1105 psig will be set @ 1135
- 4 valves originally @ 1115 psig will be set @ 1145
- 5 valves originally @ 1125 psig will be set @ 1155

The new MSRV pressure setpoints addressed in Calculation ND-Q0999-980003 and the new flow capacities determined in Calculation MD-Q0001-870133 are as follows:

Relief Valve Setpoints

- 4-CAP @ 1135 psig - 905,477 LB/HR
- 4-CAP @ 1145 psig - 913,356 LB/HR
- 5-CAP @ 1155 psig - 321,234 LB/HR

The safety-related feature of the MSRVs to prevent reactor pressure from exceeding the Technical Specification and ASME Code limit has not been changed by this modification. Therefore, the margin of safety as defined in the Technical Specifications is not reduced as a result of this modification. This change is safe from a nuclear safety standpoint. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE T40665A

The safety evaluation for DCN T40665A modifies the feedwater and feedwater control systems and reviews the feedwater heater system to support the requirements for Unit 3 power uprate (3458 MWt). This DCN reviews the Unit 3 Setpoint and Scaling calculations associated with the instrumentation loops (3-F-1-013, 3-F-1-025, 3-F-1-035, 3-F-1-036, 3-F-1-050, 3-F-1-050, and 3-P-1-081). Analytical limits such as reactor power and pressure as well as environmental conditions have changed resulting from the power uprate core operating process. The feedwater instrumentation loop accuracies associated with normal and abnormal environmental conditions resulting from the power uprate program have been evaluated. Using revised analytical limits and changed environmental conditions, the instrument setpoint calculations are revised as required while maintaining plant operability. The feedwater instruments will continue to function in the same manner as before this activity. This DCN also revises the feedwater control system computer software to support the changes as analyzed by the Setpoint and Scaling calculations.

The proposed changes will not affect any parameters described in the basis for any Technical Specification. This DCN will not cause the exceeding of any acceptance limit for any accident analysis, nor does it reduce the margin between the designed failure points or system limitations and the acceptance limits as defined in the basis for any Technical Specification. Therefore, the margin of safety is not reduced. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40673A

Stage 1 of this DCN changes out an obsolete recorder for the Wide Range Gaseous Effluent Radiation Monitoring System (WRGERMS) (0-RR-90-360) in Panel 2-9-10. The recorder provides a written record of the gaseous effluent from the main stack. The new recorder will be linked by a new cable to the total stack flow so that the radiation release rate as well as the concentration can be recorded. The recorder is not safety related, but is a post accident monitoring Category 3 device.

The old recorder is unique to the WRGERMS system and is made by Gulton, while the new one is made by Yokogawa which makes numerous recorders used throughout TVA. The Yokogawa is easier to use for both operations and maintenance, has better displays and is more versatile. The Yokogawa will be set to plot the concentration in μ Curies per cubic centimeter. It can also be set to plot the release rate in μ Curies per second or the stack flow rate. It can also be set to periodically print the numerical inputs of concentration and flow as well as the calculated value of release rate.

Stage 2 of this DCN will add a resistor to the WRGERMS audible alarm to reduce the sound level to a more comfortable yet easily recognizable sound level. The current sound level is too high and is not in compliance with the HFE criteria.

FSAR figures are affected by the change, therefore, a safety evaluation is required.

No unreviewed safety question is involved.

DESIGN CHANGE NOTICE T40709A

This safety evaluation is written in support of DCN T40709A. UFSAR Figure 5.2-7 Sheet 3 is being revised to show the increased setpoint values for the pressure regulating valves 3-PREG-84-52 and 3-PREG-84-54, and relief valves 3-RFV-84-615 and 3-RFV-84-676. The regulator valves regulate the pressure at which nitrogen is admitted through the Containment Atmosphere Dilution (CAD) System as a backup to the Drywell Control Air (DCA). The relief valves serve as over-pressure protection in the section of line downstream of the regulator valves.

The Automatic Depressurization System (ADS) operation of the Main Steam Relief Valve (MSRVs) is described in the Technical Specifications in Section 3.5.1 ECCS - Operating. The increased setpoint of the pressure regulator meets this surveillance requirement. The CAD system is described in the Technical Specification in Section 3.6.3.1. This description in the Technical Specification is not affected by this change. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40726A

The safety evaluation for DCN T40726A changes the Off Gas System bypass isolation dampers 0-DMP-066-953A and 0-066-953B from a throttled position to a normally closed position. The dampers are presently in a throttled position in an effort to prevent water from condensing and accumulation on the top of backdraft dampers 0-DMP-066-976A and 0-DMP-066-976B. The purpose of throttling dampers 0-DMP-066-929A, 0-DMP-066-953B, DMP-066-953A, and 0-DMP-066-929B was to force flow through both the normal backdraft dampers and the bypass backdraft dampers at the same time, in an effort to prevent the condensation. However, the steam packing exhaust fans did not provide enough flow to keep both the normal and the bypass backdraft dampers open. Therefore, the steam packing exhaust fan bypass dampers 0-DMP-066-953A and B will be returned to the closed position.

The basis for radiological effluent monitoring in the Technical Specifications is established by federal regulatory requirements, which specify environmental radiological release limits to protect the health and safety of the public. The Off Gas System functions to control and limit the amount of radioactivity released from the plant's main stack in order to help prevent exceeding these established limits. This damper position change has no relationship to establishing or maintaining any margin of safety as defined by regulations (i.e., the basis) which govern control of radiological releases to the environment. Therefore, this activity does not affect any margin of safety defined in the Improved Technical Specifications. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40761A

This safety evaluation is written in support of DCN T40761A. UFSAR Figure 8.7-4b, Sheet 1 (which shows the utilization of 120-V AC unit non-preferred and plant preferred circuit breakers) is affected by the change. The change provides automatic redundant 120-V AC power for the Unit 3 Off Gas Condenser Cooler inlet and outlet valves. An under voltage relay is installed in Panel 3-9-53 to provide automatic transfer to the alternate source if the normal source fails. The existing normal power source is from the 120-V AC unit non-preferred bus Panel 3-9-9 and the new alternate power source is from the plant preferred bus Panel 3-9-9. A new cable is provided for the alternate source of power. Additionally, all Off Gas circuits presently supplied from unit non-preferred bus in Panel 3-9-53 are also connected to the redundant 120-V AC power supply. This provides a more reliable power supply for these Off Gas System valves.

Failure of the 120-V AC power supply will result in loss of the Off Gas flow and may after a period of time result in a unit trip. The purpose of this change is to improve the reliability of the 120-V AC power supply and thus reduce the likelihood of a unit trip. The failure modes for loss of power for the equipment supplied with the new redundant 120-V AC power are not affected. The existing unit non-preferred normal power source is non-safety related, and the new plant preferred alternate power source is also non-safety related. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE T40762A

DCN T40762A replaces the existing Diesel Generator (DG) D battery exhaust fan/housing/motor with one that will perform the same function. The replacement is necessary because the existing unit is deteriorating. The replacement unit has a larger motor and different configuration which will necessitate changing the corresponding Thermal Overload Heater (TOL) and modifications to the ductwork to allow installation. The DG battery exhaust fans provide ventilation to prevent the buildup of hydrogen gas while the batteries are being charged. Design Criteria BFN-50-7030B, Section 7.2, indicates that the hydrogen removal function is a non-safety-related functional and process requirement. Based on this classification, the unit is not necessary to mitigate the consequences of any design basis accident. The failure modes of the replacement fan/motor unit are the same as the existing unit which are failure of function (no ventilation) and failure during a seismic event causing damage to a safety-related feature. The replacement unit is of equivalent construction to the existing unit, therefore, the likelihood that it will fail to perform its intended function is not increased. The replacement unit has been evaluated and found to be seismically adequate, therefore, the probability it will damage safety-related equipment during a seismic event is not increased. It is concluded that replacing the existing DG D battery exhaust fan/housing/motor with one that will perform the same function does not create an unreviewed safety question.

DESIGN CHANGE NOTICE T40784A

The safety evaluation for DCN T40784A reviews selected instrumentation loops found in the table below for power uprate impact. Instrumentation loop accuracies associated with normal and accident environmental conditions resulting from the power uprate program have been evaluated. Using revised analytical limits and/or changed environmental conditions, the instrument setpoint calculations are revised as required while maintaining plant operability or "No Impact for Power Uprate" letters are generated. The new Technical Specifications sections are referenced in the associated Nuclear Engineering Setpoint and Scaling Documents (NESSDs). No electrical hardware changes occur. The instruments will continue to function in the same manner as before this activity. This DCN is staged by instrument loop and is issued to support the Cycle 8 outage activities for Unit 3.

Stage	*Loop*	*Function*	*Calculation*
1 2 3 4	3-R-90-0140 3-R-90-0143 3-R-90-0142 3-R-90-0141	Monitor the reactivity for the reactor building Refueling zone. Isolate refuel floor and initiates standby gas treatment system. Close primary Containment purge and vent valves.	ED-Q0090-920073
5 6 7 8	3-P-1-86 3-P-1-82 3-P-1-76 3-P-1-72	Provide input to the control logic for Main Steam Line A, B, C and D containment isolation on low steam line pressure. The pressure transmitters sense pressure down stream of the associated outboard main steam line isolation valves.	ED-N3001-920423
9	0-TI-27-144	Provides indication and alarm functions for the forebay temperature.	ED-N0027-920265

A long-term containment sensitivity study was performed to identify the maximum acceptable core thermal power as a function of Residual Heat Removal Service Water temperature in order to maintain the peak suppression pool temperature. This is achieved by lowering the operating limit of the forebay water temperature. This new requirement ensures that current requirements for peak suppression pool temperature continue to be met; therefore, the margin of safety is not reduced. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE T40785A

The safety evaluation for DCN T407857A addresses the instrumentation loops for the power uprate impact for Unit 3. Analytical Limits such as reactor power and pressure as well as environmental conditions have changed resulting from core operating process. All instrumentation loop accuracies associated with normal and accident environmental conditions resulting from the power uprate program have been evaluated. Using revised analytical limits and changed environmental conditions, the instrument setpoint calculations are revised as required while maintaining plant operability. The new Technical Specifications sections are referenced in the associated Nuclear Engineering Setpoint and Scaling Documents (NESSDs). No hardware changes will be performed and the instruments will continue to function in the same manner as before this activity. This DCN is staged by instrument loop and is issued in support of the Cycle 8 outage activities for Unit 3. The scope of this DCN is not all inclusive. This DCN will have to be revised or subsequent DCNs performed to capture the remaining Unit 3 setpoint and scaling calculations required to accommodate the power uprate.

The setpoint for loops 3-D-1-13(A-D), - 25(A-D), -36(A-D), -50(A-D) are increased as a result of power uprated conditions. These loops provide signals to the control logic of each Main Steam Isolation Valve (MSIV) and MSDIV upon detection of main steam line high flow. The analytical limit remains at 140% of the uprated steam flow. The instrumentation loops will be recalibrated for the higher steam flow condition. The main-steam-line high-flow trip setting was selected high enough to permit the isolation of one main steam line for test at rated power without causing an automatic isolation of the rest of the steam lines, yet low enough to permit early detection of a steam line break.

Main steam line high flow could indicate a break in a main steam line. The automatic closure of various Group A valves prevents the excessive loss of reactor coolant and the release of significant amounts of radioactive material from the nuclear system process barrier. Upon detection of main steam line high flow, the following pipelines are isolated: All four Main Steam Lines and Main Steam Line Drain.

These loops, used in combination with other systems and design features, limit the nuclear system process barrier from being damaged.

The Power Uprate Safety Analysis contained in TS-384 demonstrates the accident evaluations will not result in exceeding the NRC-approved acceptance limits. In addition, it also contains a review of the increase in Loss of Accident Coolant (LOCA) radiological consequences as a direct result of power uprate, and the expected results are within the guidelines of 10CFR100 for plants applying for a 105% steam flow power uprate.

Increasing the setpoint of these loops does not introduce any new credible failure modes that would prevent the instruments to function. No hardware changes occurred and the instruments will continue to function in the same manner as before this activity. The setpoints for loops 3-D-1-13(A-D), -25(A-D), -36(A-D), -50(A-D) are increased for the higher steam flow at uprated power. This ensures that sufficient difference to the trip setpoint exists to allow for normal plant testing of the MSIVs and turbine stop and control valves. Increasing the setpoints for these loops does not create any new accidents of any type that would present an unreviewed safety question.

DESIGN CHANGE NOTICE T40786A

This DCN reviews setpoint calculations for Improved Technical Specification instruments to determine the impact of the 105% power uprate program. As a result, instrument loops 3-P-3-22AA, BB, C, D and 3-P-3-204A-3 are the only loops that require a setpoint change in this DCN. The setpoints increase to maintain adequate differences between plant parameters and trip setpoints and to maintain satisfactory safety performance. The approved calculations supporting these setpoint changes demonstrate that the loops are sufficiently accurate to perform their intended function for the conditions for which they are required.

Increasing the setpoint of these loops does not introduce any new credible failure modes that would prevent the instruments to functions. No hardware changes occurred and the instruments will continue to function in the same manner as before this activity. The setpoints for loops 3-P-3-204(A-D) and 3-P-3-22AA, BB, C, D are increased to avoid spurious scrams at uprated power and yet provide adequate difference to the maximum allowable pressure. Increasing the setpoints for these loops does not create any new accidents of any type that would represent an unreviewed safety question.

DESIGN CHANGE NOTICE T40853A

DCN T40853A is issued to provide a bill of material for a replacement valve for 0-SHV-24-1160, the Raw Cooling Water (RCW) supply shutoff valve to the Auxiliary Boiler blowdown tank "B". The replacement valve will be installed using threaded fittings and a threaded union with appropriate soldered adapter fittings in the 1" copper, RCW piping. Additionally, Flow Diagram 1-47E844-1 will be revised to show the valve and piping as 1" nominal size instead of 1 ½" to match the actual configuration installed in the plant. The flow diagram is a UFSAR figure.

The non-safety-related RCW System piping and Auxiliary Boiler System are not discussed in the Technical Specifications or its bases. Since these systems provide no safety function or support to any safety-related equipment, this change will have no affect on the margin of safety for any Technical Specification related items. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE T40909A

This safety evaluation addresses DCN T40909A which involves raising the setpoint values for Temperature indicating Switches 2-TIS-66-95 and 3-TIS-66-95, which monitor the gas discharge temperature of the Off Gas Condenser (OGC). The Off Gas System (System 66) is part of the Gaseous Radwaste System, which collects and processes gaseous radioactive wastes (condensable and non-condensable gases) from the main condenser air ejectors, the startup vacuum pumps, and the gland seal condensers. The Off Gas System processes and controls the release of these gases to the atmosphere through the plant stack so that the total radiation exposure to persons outside the controlled area is as low as reasonably achievable and does not exceed applicable regulations. These switches provide an alarm to the control room upon an increase in temperature at or above the setpoint. During the summer, river water can reach or exceed 90° and this warm cooling water causes condensate from the condenser to be much warmer. Since condensate is the cooling medium for the OGC, discharge off gas becomes warmer. Consequently, the warmer than expected OGC discharge has caused 2-TIS-66-95 and 3-TIS-66-92 to initiate alarms. These switches are not described in the Technical Specifications or in the Improved Standard Technical Specifications. Therefore, this DCN will not impact any of the safety margins in the Technical Specifications.

Increasing the setpoint values involves the removal of nuisance alarms associated with power generation systems and does not constitute an unreviewed safety question.

DESIGN CHANGE NOTICE T40997A

DCN T40997A was written in an effort to reduce the unidentified inleakage to radwaste by cutting and capping the vents and drains off of the regenerative and non-regenerative heat exchangers associated with the Reactor Water Cleanup (RWCU) System. These vents and drains are currently routed through a closed drain path to radwaste which makes it impossible to quantify the amount of inleakage passing through these lines. The vent and drains lines will have a section of line removed to facilitate the installation of caps. Operation of the RWCU System will not be affected as the vents and drains are normally closed while the system is in service. Venting and draining capabilities will be maintained by the use of a temporary drain/vent hose to be installed, via the threaded connections, as needed.

These modifications to the RWCU heat exchangers vents and drains do not alter the function of the RWCU System. The vents and drains on the regenerative and non-regenerative RWCU heat exchangers are not mentioned in the Technical Specifications nor their basis, therefore there is no reduction to the margin of safety as defined in the basis for any Technical Specification. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE W17792A

The safety evaluation for W17792A (see TVA BFN 1995 Annual Operating Report) was revised to address the change implemented by DCN F40681A. This DCN changes the proper position of valves 1-SHV-067-5003 and 5005 from THROTTLED to OPEN as shown on UFSAR Section 10.10, Figure 10.10-1a, Emergency Equipment Cooling Water Flow Diagram. This change has no effect on nuclear safety, and the conclusion that no unreviewed safety question exists is not affected by this revision.

DESIGN CHANGE NOTICE W26520A

This modification will upgrade the Unit 3 Reactor Feedwater Control System (RFWCS) with a microprocessor based distributed control system and modify the reactor feedpump turbine and main turbine Level 8 trip logic. DCN W26520A will install the RFWCS hardware. This design will affect various UFSAR figures and text. These changes do not affect the design basis function of the RFWCS. The primary purpose of this modification is to improve the reliability of the RFWCS. This modification has maintained adequate separation and isolation between safety-related and non safety-related components, therefore, this modification will not affect equipment important to safety. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE W36676A

DCN W36676A installs a Zinc Injection (GEZIP) System at BFN Unit 3. The GEZIP System consists of a simple recirculation loop off of the feedwater pumps. Specifically, the zinc solution is obtained by passing a stream of feedwater (4-100 gpm) from the feedwater pumps discharge header, through a dissolution vessel containing pelletized DZO, and back to the feedwater pumps suction header.

Addition of the GEZIP System is non-safety related and it interfaces with only the non-safety-related portion of the Feedwater and Condensate Systems. The GEZIP System is designed in accordance with the same requirements as the portions of the Feedwater and Condensate System it ties into. The Feedwater and Condensate Systems are not addressed in any basis in the Technical Specifications. The reactor water chemistry limits discussed in Technical Specification Section 3.6/4.6 will be maintained. Therefore, the modification does not reduce the margin of safety as described in the Technical Specifications. No unreviewed safety question is involved.

DESIGN CHANGE NOTICE W39479A

DCN W39479A replaced all existing power range neutron monitoring (PRNM) equipment in the Unit 3 main control room with a digital Nuclear Measurement Analysis and Control (NUMAC) PRNM retrofit system. The Average Power Range Monitors (APRMs), the Rod Block Monitors (RBMs) and their associated electronic components were replaced. The number of APRMs was changed from six, configured in a one-out-of-two-taken-twice scram logic, to a total of four, configured in a two-out-of-four scram logic. New 2-out-of-4 voters were added to interface the APRMs with the Reactor Protection System. The new APRMs include an oscillation power range monitor (OPRM) function which implements the BWR Owner's Group Option III stability trip function; however, the OPRM trip function was not enabled by this DCN. The change also implemented instrument setpoint and hardware changes necessary to support Unit 3 APRM and RBM Technical Specification (ARTS) improvements and operation in the maximum extended load line limit (MELLL) region of the power/flow domain.

The new PRNM equipment is a digital system with firmware control. It has central processing points and software controlled digital processing where the previous system had analog and discrete component processing. Specific failures of hardware and potentially common cause software failures are different from the previous system. This change was determined to constitute an unreviewed safety question on the basis that it created the possibility of a malfunction of a different type than any previously evaluated in the Browns Ferry SAR. Descriptions of the PRNM design were submitted to the NRC for review in Technical Specification submittal TS-353, which was approved prior to implementation of this change.

DESIGN CHANGE NOTICE W40627A

This safety evaluation is written to support DCN W40627A. The evaluation addresses the software installation and functional aspects of the Reactor Recirculation Control System Upgrade. This upgrade replaces the existing analog speed control components and jet pump instrumentation components with a Foxboro I/A fault-tolerant digital control system. The modified system will perform all major functions of the existing speed control and jet pump flow components, with additional operational enhancements. The existing functions performed include total core flow calculation, motor generator (MG) set speed regulation, 75% and 28% limiter enforcement, start up signal generator, limiter annunciation, MG out of service annunciation, etc. The existing functions not implemented or modified are the loss of signal annunciation, which is replaced with a common control system trouble annunciation. New functions added are operator initiated manual runbacks, two based on steam flow (for approximation of reactor power), and the one based on total core flow, master manual control with bias capability, limited speed feedback control, positioner lock indication, and positioner null indication and initiation.

The jet pump instrumentation portion of the system and the MG limiter enforcing and MG out of service annunciation will be functionally identical to the existing system. Also, the 28% and 75% limiters will function similar to the existing system.

The system is used to control the recirculation pumps speed and provide for jet pump flow indication, and it is not a safety-related system, but it is a quality related system (for Seismic II considerations). The failure modes of the new system is bounded by the failure modes of the existing system and do not affect any margins of safety as defined in the Technical Specifications. Therefore this activity does not reduce the margin of safety as defined in the basis for any Technical Specification. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE W40657A

This safety evaluation justifies the extension of the stroke time and clarifies Safety Analysis Report (SAR) wording related to valve seating method for the High Pressure Coolant Injection (HPCI) steam admission valve 3-FCV-73-16. The extended stroke time is required as a result of DCN W40657A which installs a new valve having a longer stem travel. The stroke time specified in the UFSAR for the HPCI steam admission valve will be changed from 20 seconds to 30 seconds. In order to maximize the opening capability of this valve, the valve will be regeared to its maximum potential within the 30 second stroke time. This regearing for increased thrust will provide approximately 20% more capability at a calculated stroke time of 23.6 seconds. The SAR description for the HPCI System provides a general description of the motor operated valves including a statement that each valve operator is equipped with a torque switch to shut off motor once closure is achieved. Based on the double disc design, the industry practice for this type valve is to "soft seat" the valve. This involves determinating the torque switch and using only the limit switch to remove power to the motor at a preset percentage of closure allowing the motor to coast the disc softly into the seat whereby the design of the double disc provides sealing without the need for excessive torque. The affected SAR text will be revised to state closure by torque switch or limit switch.

A surveillance test will be performed prior to return to service after this modification. The testing will be performed to document the actual performance of the system in regards to the valve opening capabilities. This will provide evidence that the HPCI System will perform its intended function of injecting to the core within the current 30 second time limit which is bounded by the 50 seconds response time used for HPCI in the SAFER/GESTR-LOCA analysis for Unit 2.

The proposed valve stroke/system initiate times are conservatively bounded by the current analyzed 50 second initiation limits. Therefore, any detectable pressure drop would not have an affect on the HPCI System function and initiation would be well bounded by the flow and time analysis currently given in the UFSAR. The increase in the full opening stroke time criteria for the 73-16 valve from 20 to 30 seconds has no affect on the HPCI System to perform its intended function. No unreviewed safety question is involved.

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SUMMARY OF SAFETY EVALUATIONS FOR PROCEDURE REVISIONS

MSI-0-082-DUC001

Maintenance Instruction MSI-0-082-DUC001 is being created to remove and reinstall the ductwork and supports associated with Standby Diesel Generators (DG) in association with engine maintenance. Portions of the ductwork and the supports may be removed as required to obtain access for the maintenance activities and later reinstalled with the DG operable to minimize the unavailability of the DG during the 12 year preventive maintenance activities.

The removal of portions of the ductwork and supports will not affect the DG's design, functional requirements, or performance characteristics. The purpose of the ductwork is to redirect warm air exhausted from the generator ventilation grills away from electrical panels located near the generators. Ductwork to direct this warm exhaust air away from the panels will remain installed until the DGs are removed from service and will be reinstalled prior to each DG being declared operable. As such, the proposed activities and temporary configuration of DGs and its associated ductwork do not adversely affect nuclear safety.

The DG ductwork is not physically described in SAR text, tables, graphs or figures. The ductwork airflows are shown schematically in Figure 10.12-6, but the technical content is unaffected by removal of a portion of the ductwork. The DG's ductwork does not physically connect to the DG building ventilation system, but exhausts into the DG room in a location remote from the electrical panels. Ductwork to direct the generator blower exhaust air away from the panels will remain installed until the DGs are removed from service and will be reinstalled prior to return to service. As a figure in the SAR is indirectly affected, a safety evaluation is required.

No unreviewed safety question is involved.

STANDARD PROGRAMS AND PROCESSES (SPP) 9.5

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

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

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

1998

SUMMARY OF SAFETY EVALUATIONS FOR TEMPORARY ALTERATIONS

TACF 0-97-04-90-1

This safety evaluation addresses the temporary replacement of the existing Main Stack Wide Range Gaseous Effluent Radiation (WRGERMS) Recorder in Panel 2-9-10 with a Yokogawa type recorder. The current recorder is a Gulton type recorder which has been removed and returned to the vendor for repair. The Data from this recorder is from 0-RM-90-306 (WRGERMS). The data recorder is Noble Gas concentration over a range of 10E-7 to 10E+5 uC/cc used to calculate post accident release rate. The recorder is identified as a Post Accident Monitor Category 3 instrument in Design Criteria BFN-50-7307 Table 1.

The Main Stack Radiation Monitoring System and its associated recorder, 0-RR-90-360, monitor the required range as indicated in Technical Specification Table 3.2.F. The replacement recorder provides the same range capability as the existing. Therefore, this activity does not reduce the margin of safety as defined in the bases for any Technical Specification. No unreviewed safety question exists.

TACF 2-98-003-003-0

This safety evaluation is written to support TACF 2-98-003-003-0 which documents Integrated Computer System (ICS) connections previously installed for monitoring purposes by Work Order 98-000539-000.

Reactor water level information from instrument loops 2-L-3-208(A-D) has been added to ICS. Connection was made from the indication circuits to spare ICS analog inputs. These instruments can be used to provide additional reactor water level indications for evaluation. Software changes on ICS were required to add these points.

These ICS connections support troubleshooting of indicated vessel level signal ringing observed on the 2-LT-3-58B and 2-LT-3-53 signals.

This activity affects the indication circuitry of the 2-L-3-208(A-D) instrument loops. As these circuits are electrically isolated from the safety-related portions of these instrument loops, this activity has no effect on the Technical Specification functions of these instrument loops. The indication circuitry of the 2-L-3-208(A-D) instrument loops are not contained in the Technical Specifications. Therefore, no margin of safety is reduced and no unreviewed safety question is involved.

1998

SUMMARY OF SAFETY EVALUATIONS FOR UPDATED FINAL SAFETY ANALYSIS REPORT REVISIONS

GENERAL REVISION OF UFSAR FIGURES

This UFSAR change is adding those Unit 1 drawings for systems required for Unit 2/Unit 3 operation and those Unit 3 drawings which correspond to the Unit 2 drawings currently in the UFSAR. This change is also revising drawing numbers in the text of the UFSAR to agree with the list of drawings contained in Table 1.3-1 (List of Engineering Drawings) and Table 1.3-2 (Engineering Drawing Cross Reference List). Duplicate drawings are also being deleted so that a drawing only appears once in the UFSAR. No physical changes are made to the plant as a result of this change. This is a documentation activity to add the necessary Unit 1 drawings and the equivalent Unit 3 drawings. No unreviewed safety question is involved.

UFSAR SECTION 2.3 AND 2.6

The safety evaluation address the proposed changes in UFSAR Sections 2.3 and 2.5 as indicated below.

- Deleted the excessive details of the description of the meteorological tower grounding in UFSAR Section 2.3.7.1.
- Changed the description of the temperature sensors to reflect current configuration in Section 2.3.7.1
- Deleted descriptive details of the data acquisition system, computer, and location of spare equipment from Sections 2.3.7.2, 2.3.7.3, 2.3.7.4 and 2.3.7.5.
- Deleted a table of meteorological equipment system error specifications and requirements and referenced the Regulatory Guide 1.23 which provides the details on this equipment.
- Made several minor corrections for tense, direction convention, misspelled words and deleted a reference.

The proposed UFSAR revisions do not conflict with the Technical Specification references to radiological control programs, Offsite Dose Calculation Manual, and the radiological environmental monitoring program. This change does not involve an unreviewed safety question.

UFSAR SECTION 2.4

Section 2.4.4.3 and Table 2.4-4 of the UFSAR are being revised to more accurately reflect the location of Browns Ferry Nuclear Plant Condenser Circulating Water (CCW) diffusers with respect to the Tennessee River. The distances between the CCW diffusers and the major users of river water are being changed since the diffuser location is the reference point. Two incorrect Tennessee River Mile (TRM) locations are being changed as well.

The change in Table 2.4-4 which revises the distance between the BFN diffuser and the downstream water users requires a modification to the concentration gradient from an accidental plant source release as shown in Section 2.4.4.3. The concentration gradient of 3.3VE-10 is being changed to 3.2VE-10 to account for the revised distance between the diffuser and the closest water user downstream, West Morgan-East Lawrence Water Authority. (V represents the spilled volume in cubic feet.)

The UFSAR change to the TRM location of the CCW diffusers does not involve a change to plant equipment. This change revises the locations on the Tennessee River in terms of mile markers. The change to the dilution term in UFSAR Section 2.4.4.3 does not alter any Technical Specification liquid effluent limit contained in Section 6.8.4.1.

No unreviewed safety question is involved.

UFSAR SECTIONS 3.3.5.1, 3.3.5.2, AND 3.3.5.3

This safety evaluation activity involved a change to the plant as described in Sections 3.3.5.1, 3.3.5.2, and 3.3.5.3 of the FSAR. These sections provide descriptions and evaluations of the effects of reactor internal pressure differences on reactor internals and core support components. The UFSAR revision was needed both to incorporate a description of current methodology and to include the effects of the Increased Core Flow (ICF) mode of operation. The ICF mode of operation was approved by the NRC in 1983 and 1984. This was a documentation change only; no physical change to the plant was made. The activity was compared against the criteria of 10 CFR 50.59 and was determined not to involve an unreviewed safety question.

UFSAR SECTION 4.3

This safety evaluation addresses changes to SAR Section 4.3 in the response to reviews performed for BFPER9800530. The subject change is editorial in nature and assures proper information relative to the Reactor Recirculation System is contained in the SAR. Specifically, this change involves:

- Wording clarification for reactor recirculation pump seals.
- Added minimum to "Each seal is designed for a minimum life of 18 months based on 90% probability factor".

Since the subject changes to SAR Section 4.3 are editorial in nature, they does not impact operation of the Reactor Recirculation System as is presently described in the SAR; hence no unreviewed safety question exists.

UFSAR SECTION 4.8

This safety evaluation addresses the change to the BFN UFSAR Section 4.8, "Residual Heal Removal System (RHRS)". The changes addressed in this safety evaluation are:

- Throughout UFSAR Section 4.8, the RHR System pumps are referred to as the "RHRS main steam pumps". This description is being simplified in most cases to read "RHRS pumps".
- The UFSAR lists four modes of operation for RHR. A fifth mode of operation (supplemental fuel pool cooling) has been added for completeness.
- The discussion of logic and valve testing during shut down will be deleted since such testing may be performed during modes other than cold shutdown.

The clarification of the wording in UFSAR Section 4.8 does not change testing requirements nor the method of testing RHR System. Since all testing will continue to be performed in accordance with the Technical Specifications, the safety margin cannot be reduced. Therefore, no unreviewed safety question is involved.

UFSAR SECTION 4.8.6.1

This safety evaluation addresses inaccurate information presented in UFSAR Section 4.8.6.1 concerning the point during normal unit shutdown at which Residual Heat Removal System (RHRS) shutdown cooling mode of operation is placed in service.

The existing description states, "When nuclear system temperature has decreased to the value where the steam supply pressure is not sufficient to maintain the turbine shaft gland seals, vacuum in the main condenser cannot be maintained and the RHRS is placed in the shutdown cooling mode of operation."

This information conflicts with plant operating instructions which direct switching the turbine gland seal supply to Auxiliary Boiler Steam at approximately 40% power.

This safety evaluation supports the following wording for a SAR change request: "The RHRS is typically placed in the shutdown cooling mode of operation when reactor vessel pressure has decreased sufficiently to clear the interlocks associated with the shutdown cooling suction valves."

This method of operation allows a normal shutdown to proceed without the risk of having to break main condenser vacuum due to loss of turbine gland seal supply. RHRS can then be placed in service independent of the status of the main condenser and turbine gland sealing steam. This is consistent with plant operating procedures and is a much better technique for normal plant shutdown.

The main turbine gland seal steam system is not covered by Technical Specifications. The method of normal plant shutdown contained in 2 and 3-GOI-100-12A do not conflict with any information in the Technical Specifications.

No unreviewed safety question is involved.

UFSAR SECTION 4.12

This UFSAR revision of Section 4.12, Inservice Inspection and Testing, updates references to American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection Codes of Record and incorporates references for the new containment inspections required by ASME Section XI, Subsection IWE.

This is an administrative change to update the UFSAR by clarifying 10CFR50 requirements. This change does not impact the plant design bases or operational parameters. Therefore, an unreviewed safety question does not exist.

UFSAR SECTION 5.2.6

The UFSAR is being changed to provide revised nitrogen requirements for Containment Atmosphere Dilution (CAD) operation post Loss of Coolant Accident (LOCA). This change does not introduce any new credible failure modes. The newly calculated CAD tank supply for nitrogen injection following a LOCA is generated in accordance with Atomic Energy Commission (AEC) Safety Guide 7. The Guide provides the percentage by volume of oxygen concentration to be maintained in containment by the CAD System. The original values given in UFSAR Section 5.2.6 were not accurately derived. This ensures that the change does not impair CAD nuclear safety functions.

This change was reviewed against Chapter 14 of the USFAR for impact on the CAD System. The CAD System is not the initiator of any design basis event, but it is designed to perform safety-related functions to mitigate accidents and events. The CAD System also allows for venting the drywell to the Standby Gas Treatment System and provides support for primary and secondary containment and primary containment isolation functions to mitigate a LOCA. This change supports CAD's nuclear safety function of containment inerting by determining a more accurate supply value for injection nitrogen following a LOCA. The capability of the system to perform primary containment isolation functions and to support primary and secondary containment is unaffected by this change. No other system's required safety functions are impacted. Therefore, this change has no affect on the Design Basis Accidents and Anticipated Operational Transients. No unreviewed safety question is involved.

UFSAR SECTION 5.3

This safety evaluation is in support of the SAR change request generated by a corrective action from PER-98-006702. A review of the portions of the SAR that addresses Standby Gas Treatment (SGT) identified several non-technical wording errors and some technical errors. The associated SAR Change Request corrects these errors and brings the SAR system description in agreement with the actual installation.

This change effects the SAR description of the SGT System and system/train operation. No actual changes to the plant are performed. Operation of the SGT System is in accordance with Technical Specifications and the Technical Requirements Manual. This change does not reduce the margin of safety as defined in the Technical Specifications basis. This change does not involve an unreviewed safety question.

UFSAR CHAPTER 7

This safety evaluation addresses changes to the UFSAR, Chapter 7. The changes are a result of Problem Evaluation Report numbers BFPER971248, BFPER980162 and BFPER971353. Changes were requested by the Integrated Standard Technical Specifications group to Section 7.4.3.5.4 and the level instrumentation listed in Table 7.4-2. These changes are clarifications of the wording and descriptions of the systems and subsystems to match the plant. There are no actual physical plant changes associated with this UFSAR revision.

The change from tripping the recirculation pumps at less than 100% to less than or equal to 90% open on the stop valve position still meets the requirement of reducing reactor power before the main turbine is lost. The recirculation pumps also still trip on low reactor water level, only they are now receiving the signal through Anticipated Transient Without SCRAM (ATWS).

None of the changes impact the Technical Specifications nor their basis. No unreviewed safety question is involved.

UFSAR CHAPTER 7

This safety evaluation is written in support of UFSAR clarifications, enhancements and administrative changes for PER BFPER980575. The changes are clarifications of the wording and descriptions of the systems and subsystems to match the plant. These changes include the following systems: Rod Block Monitor and the Control Rod Drive (CRD) subsystems.

There are no actual physical plant changes associated with this UFSAR revision. Most of the evaluated changes are clarifications and corrections of ambiguous and vague text. However, the following will be considered as physical changes for the purpose of this evaluation: the control rod block annunciator and buzzer, an automatic bypass of the SRM detector block when the IRMs are on a range 3 or above, and also the CRD temperature alarm/annunciator.

None of these changes to the UFSAR text impact the Technical Specifications or their basis. Therefore, this activity does not reduce the margin of safety as defined in the basis for any Technical Specification, and no unreviewed safety question is involved.

UFSAR TABLE 7.13-2

This safety evaluation addresses an update to the list of Area Radiation Monitors (ARMs) in the radwaste building listed in UFSAR Table 7.13-2 (TVA drawing 0-47E600-9).

Addition of the evaporator building area radiation monitor to the facility description does not impact any equipment required to operate or shut down the facility. The ARM is provided for ALARA, not nuclear safety consideration. Therefore, this activity is acceptable from a nuclear safety perspective.

The ARMs are not addressed in Technical Specifications and do not provide information used to determine the margin of safety.

No unreviewed safety question is involved.

UFSAR SECTION 7.16

This safety evaluation addresses changes to UFSAR Section 7.16. The subject changes are editorial in nature and assure proper information relative to the Process Computer System is contained in the SAR. Specifically, these changes involve:

- Removal of vendor model number information
- Removal of specifics regarding storage media
- Removal of “second person” verb tense

- Deletion of redundant words (e.g. "all")
- Wording clarifications

Operation of the process computer and any associated process are not changed.

No unreviewed safety question is involved.

UFSAR SECTION 8.1, 8.2, AND 8.7

This safety evaluation revises portions of UFSAR Section 8.1, 8.2, and 8.7 as part of the corrective actions for BFPER98-006244-000 and BFPER98-006417-000.

In the description of the 120-V Reactor Protection System (RPS) alternate source under Section 8.1, the statement "Each bus has an alternate source from a 480-120V regulation transformer." is changed to "Each bus has an alternate source from a single 480-120V regulating transformer, which may supply either bus." This change clarifies the description to make it consistent with UFSAR Section 7.2.3.2, and UFSAR Figures, 8.6-1d and 7.2-1 which indicate that there is only one regulating transformer for each unit.

In the description of the power system display information under Section 8.1, the statement "Shutdown system loading for all three units is either automatic or coordinated through an assigned licensed operator." is revised to delete the words "assigned licensed." he change is made for consistency to avoid the implication that some special level of assignment or qualification must be made for this task compared to all other tasks operators perform.

In the description of generator and transformer protective relays under Section 8.2, references to "the unit station service auxiliary power transformers" are revised to delete the term "auxiliary". This change is for editorial consistency and does not affect the content of the descriptions.

In the description of the 120-V AC power system under Section 8.7.3, the lead sentence indicates that six systems are listed below while there are actually seven systems listed. Changes made are editorial to correct this typographical error and do not affect the content of the descriptions.

Changes in the description of the 120-V RPS alternate source clarify the description to indicate that there is only one regulating transformer for each unit. This change makes the text agree with UFSAR Section 7.2.3.2, UFSAR Figure 8.6-1d, UFSAR Figure 7.2.1, Technical Specification Bases B3.3.8.2 and the system design criteria. The alternate supply from the regulating transformer (as well as the normal supplies from the motor generator sets) is provided with redundant isolators connected in series to protect the RPS buses from undervoltage, overvoltage, and underfrequency. The switching arrangement for the alternate power supply prevents feeding both RPS buses from the same source or paralleling of the normal and alternate sources. Changes in the description of shutdown system loading to remove reference to the operators as "assigned" and "licensed" do not change the described manual actions. Therefore, the proposed activity will not reduce the margin of safety as defined in the basis for any Technical Specification. This change does not involve an unreviewed safety question.

UFSAR SECTION 8.3, 8.3, 8.5, AND 8.6

This safety evaluation addresses the change to BFN UFSAR Sections 8.3., 8.4, 8.5, and 8.6 as part of the corrective actions for BFPER980129, BFPER980524, BFPER980231, and BFPER98-0008054-000. Examples of descriptions of changes. 1) In general description the transmission system under Section 8.3.1, revise the statement that the TVA transmission system network in Figure 8.3-3 to state that the Figure 8.3-3 is for historical references only. 2) In various places syntax errors in the callout for systems voltages are corrected. For example 4-KV is changed to 4-KV. These changes do not affect the content of the descriptions. 3) In the description of the location of the common station service and cooling tower transformers under Section 8.4.5.1, revise the term "outside the building" to say "outside". 4) In the safety design bases for the standby AC power system under 8.5.2 bases 13, the references to specific sections of the UFSAR for analysis are deleted. 5) In the list of 250-V DC connected loads Table 8.6-1, under unit battery boards, add "Distribution Board Panel 9-24 Alternate" as a load on the unit batteries.

Changing the reactor core cooling requirements in the safety design bases for the standby AC power system from the "EECS interim Criteria" to the "10CRF50 Appendix K Criteria" will not adversely affect the Standby AC Power Supply. This change is necessary to make Chapter 8 of the UFSAR agree with the present EECS Appendix K criteria in Chapter 6 and 14 of the FSAR. This change does not affect the loading sequence. Existing analysis for diesel generator loading performed prior to recovery of the Unit 2 and 3 were performed to the Appendix K criteria. Changes to the offsite and normal power supply preserve the independence of these systems and the redundancy of the offsite site sources. All changes to the description of the 161-KV and 500-KV transmission system will maintain compliance with existing design bases for the systems. Changes to Limestone line and changes to the planning map will not affect routing of transmission lines near the plant so a transmission tower failure will not cause the lost of both sources of offsite power.

Changes to the 250-V shutdown board and unit battery recharge times are consistent with the design criteria and the design bases calculations (ED-Q0248-870041 and ED-Q0248-870042). The shutdown board and unit batteries each have a spare charger that will be place in service in the event the normal charger fails. The length of these outages is controlled in the Technical Specification. These changes are consistent with the bases of the Technical Specification which require the battery charges to recharge the batteries from the design minimum change condition. Therefore, the proposed activity will not reduce the margin of safety as defined by the basis for any Technical Specification. There is no unreviewed safety question involved.

UFSAR SECTION 8.9

This safety evaluation addresses the changes in UFSAR Sections 8.9 as indicated below. Included in these discussions are the proposed changes to the applicable design criteria.

Section 8.9: Safety Systems Independence Criteria and Bases for Electrical Cable, Second paragraph, First sentence, Page 8.9-1 Change:

> **From:**
> "The electrical circuits associated with redundant or counterpart divisions, components, or subsystems of electrical systems important to safety are separated from each other by means of spacing or barriers."
> **To:**
> "The electrical circuits associated with redundant or counterpart divisions, components, or subsystems of electrical systems important to safety are separated from each other by means of spacing or barriers **or analysis to demonstrate functional redundancy (See Section 8.9.4)**."

The meeting of separations criteria by "functional redundancy" is allowed by General Electric's (GE) Topical Report NEDO-10139 and DC BFN-50-728. (References: BFN-50-728, Section 1.1.1 and Attachment D, and

Calculations ED-Q2002-880304, ED-Q0000-880437, ED-Q0999-910051, and ED-Q3999-930120). This change is considered a clarification and editorial change because it identifies another acceptable and established method for circuit design to meet single failure requirements. This change does not impact the design bases. DC BFN-50-728, Section 1.1.1 is being revised to add historical information dealing with the analysis for Unit 3 functional redundancy. This change is consistent with historical information provided in Section 1.1.1 for Units 1 and 2. The change does not impact the design bases.

Section 8.9.A: Reactor Protection System (RPS), Second sentence, Page 8.9-1 Change:

From:

"The instrument signal circuits for the RPS are separated into four channels (A1, A2, B1, and B2), identified as Divisions IA, IIA, IB, and IIB, respectively. The manual signal circuits are separated into Channels A and B, identified as Divisions IIIA and IIIB, respectively."

To:

"The instrument signal circuits for the RPS are separated into four channels (A1, A2, B1, and B2), identified as Divisions IA, IIA, IB, and IIB, respectively. The manual signal circuits are separated into Channels **A3** and **B3**, identified as Division IIIA and IIIB, respectively."

The terminology for "A3" and "B3" for the manual channels of the Reactor Protection System (RPS) is consistent with UFSAR Figure 7.2-3e (Drawing 2-47E611-99-5). UFSAR Figures 7.2-4 and 7.2-5 also utilize the "A3" and "B3" terminology for the RPS manual scram channels. This change does not impact the design bases.

DC BFN-50-728, Section 3.4.2.1 is being revised to clarify/add the"A3" and "B3" terminology for the RPS manual scram channels. This change does not impact the design bases and is consistent with issued design output.

Section 8.9.B: Primary Containment Isolation System (PCIS), First sentence, Page 8.9-1 Change:

From:

"Control and power circuits for the inboard Primary Containment Isolation System valves are all in Division I; similar circuits for the outboard valves are in Division II. Control and instrumentation of this isolation system are described in UFSAR Subsection 7.3."

To: (Delete **all** and add a new second sentence):

"**Except as noted,** control and power circuits for the inboard Primary Containment Isolation System valves are in Division I; similar circuits for the outboard valves are in Division II. **An exception to this requirement is the RHR System I LPCI inboard injection valve.** Control and instrumentation of this isolation system are described in UFSAR Subsection 7.3."

The concern is the statement that "all" PCIS outboard isolation valves receive power and control power from Division II. An exception to this statement is the Loop I RHR-LPCI Outboard Injection Valves 1-, 2-, and 3-FCV-074-0053 with RHR in shutdown cooling. This valve is a Division I valve. The inboard isolation valve associated with the subject loop and valve are testable check valves which do not require an electrical control power (process control). The assignment of Division I electrical control and power to this valve (all three units) meets the requirements of DC BFN-50-7074, Section 3.6.2.1.5 and design bases drawings which requires the redundant loop (Loop II) to be electrically separated so that no single design basis event can disable both loops (Reference Drawings 1-, 2-, 3-47E474 / UFSAR Figures 5.2-22, Sheets 1, 2, and 3, respectively and 1-47E610-74-1A, 2-47E2610-74-1 and 2, and 3-47E610-74-1 and 2). This change does not impact the design bases.

DC BFN-50-728, Section 3.4.2.2 is being revised to clarify a similar statement to agree with BFN-50-7074, Sections 3.6.2.1.4 and 3.6.2.1.5 and design bases drawings. This change does not impact the design bases.

Section 8.9.2.1: General Plant Area, Page 8.9-3, First paragraph Change:

From:

"Conduit was sized per TVA Electrical Standards Drawing 30AB09 RO, 30A810 RO, and 30A811 R1 prior to 1976."

To:

"Conduit was sized per TVA Electrical Standards Drawing 30A**809** R**0**, 30A810 R**0**, and 30A811 R1 prior to 1976."

This change is considered to be a typographical error (corrected B to an eight and Os to zeros). The change does not impact the design bases.

Section 8.9.4: Control Room and Local Panels, Second sentence, Page 8.9-7 Change:

From:

"If this protection cannot be provided by circuit design, the switches for controlling different divisions of subsystems are grouped in subpanels separated by fire-resistant separation barriers without penetrations that could propagate a fire between subpanels".

To:

"If this protection cannot be provided by circuit design **(i.e., principles of fault-resistant circuit design and functional redundancy analysis to satisfy the single failure requirement)**, the switches for controlling different divisions of subsystems are grouped in subpanels separated by fire-resistant separation barriers without penetrations that could propagate a fire between subpanels".

This change is consistent with DC BFN-50-728, Sections 1.1.1 and 3.6.2.5 and GE's Topical Report NEDO-10139. The change does not impact the design bases.

DC BFN-50-728, Section 3.5.2.3, last sentence is being revised to read as follows:
Reactor MOV boards D and E supply the LPCI system **inboard** injection valves, recirculation pump discharge valves, RHR test valves **(Unit 1 only)**, and RHR pump minimum flow bypass valves, which are the only ESS loads that may be assigned to **these boards.**

This change corrects a typographical error, corrects board to plural, and denotes the RHR test valves as being Unit 1 only to agree with design output drawings and plant configuration.

DC BFN-50-728, Section 3.6.2.5.f, first paragraph, last sentence is being revised to clarify the sentence to state that there is **no** requirement to separate or use barriers between devices or wiring of the same division.

This error of leaving the no out of the last sentence was introduced when Revision 2 of the DC was issued. This change is consistent with design philosophy and UFSAR Section 8.9.4.1. This change is considered to be a typographical error due to the inadvertent omission of the word "no" during the DC revision process. This change does not impact the design bases.

DC BFN-50-758, Section 3.1, first paragraph, after the fifth sentence, add two sentences dealing with; (1) the acceptability of no more than 10 uncoated non-IEEE 383 flame retardant cables on open cable trays and (2) that these uncoated cables shall be documented and tracked through the design process.

This addition is consistent with existing design philosophy and DC BFN-50-728,Section 3.6.3.7. The change does not impact the design bases, the text in Section 8.9.1 of the FSAR, nor the text in Volume 1 of the Fire Protection Plan.

These changes are consistent with the basis of the Technical Specifications. Therefore, the proposed activity will not reduce the margin of safety as defined in the basis for any Technical Specification.

This changes does not involve an unreviewed safety question.

UFSAR SECTION 9.2.4.3 AND 9.2.4.4

Sections 9.2.4.3 and 9.2.4.4 are being revised to remove an unnecessary restriction requiring the plant to discharge the contents of the Chemical Waste Tank, the Laundry Drain Tank, and the Cask Decontamination Tank into the circulating water canal. These tanks' contents are assumed to be of such unsuitable quality as to preclude plant reuse. This change will allow plant staff to evaluate tank contents to determine if processing to plant makeup quality by station or vendor supplied equipment is appropriate. This change requires no physical equipment changes to the plant.

There are no changes to system configuration, operation, or alignment of the Radwaste System. None of the UFSAR changes alter any existing limits or provisions for controlling the release of radioactive materials to the environment. These changes have no relationship to establishing or reducing any margin of safety as defined by these regulations. No unreviewed safety question is involved.

UFSAR SECTION 10.4

This safety evaluation addresses several changes to SAR Section 10.4, "Refueling Tools and Service Equipment". These changes provide clarification for updated vessel disassembly and refueling processes. Minor changes in equipment nomenclature are also included. These changes are consistent with currently approved BFN procedures.

These SAR changes do not affect Technical Specifications and have no impact on any margin of safety as defined in the basis for any Technical Specification. This change does not involve an unreviewed safety review.

UFSAR SECTION 10.5

This safety evaluation addresses several changes to SAR Section 10.5, "Fuel Pool Cooling and Cleanup System". These changes provide clarification and are consistent with currently approved BFN procedures. Minor changes in equipment nomenclature and terminology are also included.

Changes to UFSAR Section 10.5, Fuel Pool Cooling and Cleanup System, are listed below:

<u>Section/Page</u>	<u>Changes</u>
10.5.4 (pg 10.5-1)	Change: "The system cools the fuel storage pool by transferring the spent fuel decay heat <u>(see Table 10.5-1)</u> through heat exchangers to the Reactor Building Closed Cooling Water System." To: "The system cools the fuel storage pool by transferring the spent fuel decay heat through heat exchangers to the Reactor Building Closed Cooling Water System <u>(see Table 10.5-1, Fuel Pool Cooling and Cleanup System Specifications)</u>." Move table reference to end of sentence and add table title.

10.5.4 (pg 10.5-2)

Change:
"Four filter demineralizers are provided (one spare unit shared between the three active units), each with a design capacity equal to or greater than the design flow rate for a fuel pool. The pumps circulate the pool water in a closed loop, taking suction from the surge tanks, circulating the water through the heat exchangers and filter demineralizer, and discharging it through diffusers at the bottom of the fuel pool and reactor well. The water flows from the pool surface through skimmer weirs and scuppers to the surge tanks."
To:
"Four filter demineralizers are provided including one spare filter demineralizer shared between the three active units, each with a design capacity equal to or greater than the design flow rate for a fuel pool. The pumps circulate the pool water in a closed loop, taking suction from the surge tanks, circulating the water through the heat exchangers and filter demineralizer, and discharging it through diffusers at the bottom of the fuel pool and reactor well, (as required during refueling operations). The water flows from the pool surface through skimmer weirs and scuppers (wave suppressers) to the surge tanks."

Delete the parentheses, add "including", and change "unit" to "filter demineralizer" to better describe the spare equipment being discussed. Add "as required during refueling operations" to describe when a discharge would be made. Add "(wave suppressers)" to provide definition for "scuppers."

10.5.4 (pg 10.5-2)

Change:
"The heat exchangers are designed to remove the decay heat load of the normal discharge batch of spent fuel."
To:
"The heat exchangers are designed to remove the decay heat load of the normal discharge batch of spent fuel (see Section 10.5.5)."

Add "(see Section 10.5.5)" to provide reference to safety evaluation discussion and elaborate on the variations of core heat loads the system will have to dissipate.

10.5.4 (pg 10.5-3)

Change:
"The resin is replaced when the pressure drop is excessive or the ion exchange resin is depleted."
To:
"The resin is replaced when the pressure drop is excessive, the ion exchange resin is depleted, or as required by plant conditions."

Add "or as required by plant conditions" to recognize requirements of de-contamination instructions.

10.5.4 (pg 10.5-3)

Change:
"The filter demineralizer units are designed to operate with water flowing at normal 2 gpm/sq ft."
To:
"The filter demineralizer units are designed to operate with water flowing at approximately 2 gpm/sq ft."

Change "normal" to "approximately" to achieve better terminology.

10.5.4 (pg 10.5-3)

Change:
"The ion exchange resin is a mixture of finely ground, 300 mesh or less, cation and anion resins in proportions as determined by service requirements. The cation resin is a strongly acidic polystyrene with a divinylbenzene cross-linkage and is supplied in the fully regenerated hydrogen form. The anion resin is a strongly basic, Type I, quarternary ammonium polystyrene with a divinylbenzene cross-linkage and is supplied in a fully regenerated hydroxide form."

To:
"The ion exchange resin is a mixture of finely ground cation and anion resins in proportions as determined by service requirements. The cation resin is supplied in the fully regenerated hydrogen form. The anion resin is supplied in a fully regenerated hydroxide form."

Change allows flexibility in types of resin used in the Fuel Pool Cooling system. This description was originally derived from the GE specification from 1969.

10.5.4 Change:
(pg 10.5-4) "Pump low-suction pressure automatically turns off the pumps."
To:
"Pump low-suction pressure automatically turns off the pumps <u>in case of improper valve alignment</u>."

Add "in case of improper valve alignment" to explain the condition which stops the pump automatically.

10.5.4 Change:
(pg 10.5-4) "The controls for the remote manually controlled valves which discharge the fuel pool water to the condenser hotwell and condensate storage tank are located on the pump <u>room deck</u>. The open or closed condition of each of these valves is indicated by lights on the pump <u>room</u> panel."
To:
"The controls for the remote manually controlled valves which discharge the fuel pool water to the condenser hotwell and condensate storage tank are located on the pump <u>local panel</u>. The open or closed condition of each of these valves is indicated by lights on the pump <u>local</u> panel."

These changes more accurately agree with terminology of plant operations.

10.5.4 Change:
(pg 10.5-5 and fig 10.5-1a)
"A high rate of leakage through the refueling bellows assembly, drywell to reactor seal, or the fuel pool gates is indicated by lights on the instrument rack in the pump area and is alarmed in the Main Control Room."
To:
"A high rate of leakage through the refueling bellows assembly, drywell to reactor seal, or the fuel pool gates is indicated by lights on the instrument rack in the pump area and is alarmed in the Main Control Room. The refueling bellows drains have been welded closed."

Clarifies that refueling bellows drains are welded closed.

10.5.5 Change:
(pg 10.5-7) "These drainage paths are formed by welding channels behind the liner weld joints and are designed to permit free gravity drainage to the floor drain tank via the floor drain sump."
To:
"These drainage paths are formed by welding channels behind the liner weld joints and are designed to permit free gravity drainage to the floor drain <u>collection</u> tank via the floor drain sump."

Add "collection" to describe purpose of the tank.

This change does not affect system design or functional requirements; the technical content of text, tables, graphs, or figures in the SAR. No unreviewed safety question exists.

UFSAR SECTION 10.5.5 AND TABLE 10.5-1

This change removes all references to "maximum normal heat load/design heat load, and maximum possible heat load" from Section 10.5.5 and Table 10.5-1 of the UFSAR. These terms originated from the original General Electric design specification for Fuel Pool Cooling (FPC) and are outdated due to new fuel technology and updated outage management methods. The heat removal capacity of FPC is unchanged. Unloading the reactor core and the associated increase in fuel pool heat load is a controlled evolution. Removal of these terms does not affect the ability of FPC to safely cool the pool or maintain adequate water level.

This change eliminates inconsistencies by deleting inadequately defined terms and replacing them with administrative controls that are actually used to ensure that the heat load placed in the pool during refueling operations does not exceed available cooling capacities. Therefore, this change does not affect the design basis of FPC or Residual Heat Removal (RHR) assist but clarifies that unloading the core is a controlled evolution. Fuel pool level and temperature are unaffected by these changes.

This change also adds design heat capacities for the FPC and RHR heat exchangers in Table 10.5-1. Actual cooling capacities are unchanged. These capacities replace any reference to design and maximum possible heat loads referenced in the table. These capacities are used in conjunction with administrative controls to ensure anticipated heat loads do not exceed actual cooling capacities.

Hence, the activities per this UFSAR change are safe from a nuclear safety standpoint. No unreviewed safety question is involved.

UFSAR SECTION 10.18

This safety evaluation a revision to Section 10.18 of the FSAR. The following changes have been made:

- Group name changes due to reorganization.
- Selection of a vendor other than South Central Bell for commercial telephone service.
- Eliminated use of the insulated shield wire carrier system.
- Figure changes to reflect current plant configuration.

The communications equipment at BFN does not perform any safety-related function, nor does it interface with any equipment which does perform any safety-related function. The communications equipment does not interface with any plant logic. Therefore, any malfunction of plant communications equipment would be limited to communications equipment only. No unreviewed safety question exists.

UFSAR SECTION 12.2

This safety evaluation addresses changes to the BFN UFSAR, Section 12.2.9, "Principal Structures and Foundations." Sections 12.2.9.2.2 and 12.2.9.3.2 are being revised to correct the descriptions of the reactor building flood gate warning light and gate dogging device alarm.

The changes to the UFSAR text Section 12.2.9 do not affect the bases of the Technical Specifications since they are clarifications only. No operating parameters or components of the plant structures or foundations are changed; therefore, their ability to perform their safety functions are unaffected and there is no reduction in the margin of safety as defined in the bases of the Technical Specifications. No unreviewed safety question is involved.

UFSAR SECTION 12.2.2.1.6, AND 12.2.2.3.2, AND 12.2.8.2.1

This safety evaluation is written to support changes to the BFN UFSAR, Sections 12.2.2.1.6, 12.2.2.3.2 and 12.2.8.2.1. The change in Section 12.2.2.1.6 will delete the following sentence: "(Only one door can be opened at a time)". This section of the UFSAR provides a description of the personnel access locks which are part of the reactor building. The above sentence is correct when considering a personnel access lock which has only two doors. For access locks which have three doors, it is possible for two of the three doors to be open simultaneously, which would make the sentence literally incorrect. Since the paragraph which contains the sentence is discussing the inspection of personnel access lock doors in general, the sentence is not relevant or needed. Therefore, it can be deleted without affecting the content or intent of the paragraph.

The change in Section 12.2.2.3.2 will change two sentences from: "Where equipment occurs on floor slabs, the actual equipment load per square foot is computed to determine if it exceeds the uniform floor <u>design</u> load. The larger of these two values is used as the <u>design</u> load" to "Where equipment occurs on the floor slabs, the actual equipment load per square foot is computed to determine if it exceeds the uniform floor <u>live</u> load. The larger of these two values is used as the <u>live</u> load". The above two sentences are changed to clarify that the loading being discussed is live load and not dead load.

The change to Section 12.2.8.2.1 will change one sentence from: "Doors at the rear of rooms from the diesel generator units <u>connect these rooms to the CO2 room</u>." to "Doors at the rear of rooms for the diesel generator units <u>and the CO2 room connect to the pipe and electrical tunnel</u>."

These changes are for documentation only for the affected text in the three UFSAR sections. The changes will clarify the UFSAR text, will make the UFSAR consistent with the Design Criteria, describes more accurately past and current design practices, and makes a UFSAR description consistent with the actual arrangement of the buildings. This change does not involve an unreviewed safety question.

UFSAR SECTION 12.2.8.4.3

This change is being made to enhance the statement in Section 12.2.8.4.3, Inspection and Testing, of the UFSAR which states, "The bulkhead is to be inspected for deterioration of seals and structural members before each use." This statement does not give a specific time frame and the bulkhead has never been required to be used. The enhancement being made is due to the Preventive Maintenance Program placing the portable bulkhead inspections, for the above requirements, on a periodic basis. Reducing the time between inspections will ensure the bulkhead will perform its intended function when required. This is more conservative than the existing statement and will ensure that three out of the four Diesel Generators will remain operable during a flood, if one of the exterior doors is inoperable and cannot perform its intended function. 0-AOI-100-3 has steps which reference Technical Requirements Manual (TRM) Section 3.3.6 which states, "The unit shall be shutdown and placed in the cold condition when Wheeler Reservoir lake stage rises to a level such that water from the reservoir begins to run across the pumping station deck at elevation 565'. Also, the Improved Technical Specification Section 3.8 and Bases gives the requirements for one of the Diesel Generators being inoperable. Therefore, this change will not reduce the margin of safety as defined in the basis for any Technical Specification. This change does not involve an unreviewed safety question.

UFSAR SECTION 12.2.9.2.2

This safety evaluation, written to support a change to UFSAR Section 12.2.9.2.2 relative to emergency access from the equipment access lock, was summarized in the 1997 Annual Operating Report to NRC. A subsequent self-assessment identified that the safety evaluation was deficient in that the questions were not answered in a complete and thorough manner. The safety evaluation has, therefore, been revised to address these concerns. The conclusion that this change does not constitute an unreviewed safety question still exists.

UFSAR SECTION 13.7

This safety evaluation addresses a change to BFN UFSAR, Section 13.7, "Records". Since the requirements and responsibilities for records are contained in TVA-NAQ-PLN89-A, Revision 7, Section 6.3, they are being removed from the UFSAR and this section is being deleted.

The changes to the UFSAR has no affect on the basis of any Technical Specification. Therefore, no unreviewed safety question is involved.

UFSAR CHAPTER 14

BFN UFSAR Chapter 14 discussions regarding the radiological consequences of various design basis accidents (DBA) are not consistent with current design basis calculations and recent TVA submittals to NRC concerning power uprate. These discrepancies are documented in numerous Problem Evaluation Reports (PERs) as part of the UFSAR Verification Program. Prior to implementation of power uprate, it is necessary to insure the Chapter 14 radiological discussions accurately reflect pre uprate conditions.

The changes addressed in this safety evaluation will bring the UFSAR into agreement with other licensing and design documents concerning radiological consequences of various DBAs and will be more consistent with current regulatory guidance. The major changes concern the Main Steam Line Break (MSLB) and the method used for determining the source term. This methodology is consistent with Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors", and Standard Review Plan 15.6.4, "Radiological Consequences of a Main Steam Line Failure Outside Containment (BWR)". These changes dramatically increase the assumed source term and provide additional conservatism. Even with this additional conservatism, the calculated doses are well within the 10CFR100 and GDC 19 dose guidelines.

There are no new failure modes created by this change. Only the methodology and assumptions used in the performance of the dose calculations are changed. No unreviewed question is involved.

UFSAR SECTION 14.6.3

The phenomena of bypass leakage through the Main Steam Isolation Valves (MSIVs) into the turbine building and its associated effect on control room radiological doses was inappropriately added to the UFSAR in Amendment 11 via UFSAR Change 11-172 (RIMS R93 930927 800). MSIV leakage was not a part of the original licensing or design basis dose calculations for BFN (control room or offsite dose) but was voluntarily considered for control room doses as part of resolution of Condition Adverse to Quality Report (CAQR) BFP870591. Information relative to the resolution of the CAQR was provided to the NRC, and NRC concurrence was requested via a Safety Evaluation Report (SER). The information relative to MSIV leakage effects on control room doses should not have been added to the UFSAR until NRC concurrence was received. To date, NRC concurrence has not been received, and therefore, the information is included inappropriately and is thus being removed.

Deletion of the MSIV leakage path from the current UFSAR will make the document consistent with the current design and NRC approved licensing basis descriptions/assumptions. This change does not affect the actual integrity of the primary containment, secondary containment boundary or effluent release paths. The reactor coolant boundary isolation, primary/secondary containment isolation and Standby Gas Treatment System filtering/exhausting functions will be maintained. Since this leakage path does not initiate any events, it cannot affect the probability of an event occurring. The leakage paths considered will be in conformance with the current design and licensing basis and thus will not affect the consequences of an event. The leakage path is not new and thus does not represent a new phenomena. Therefore, since the UFSAR will be in conformance with the current design and licensing basis, it is concluded that the proposed activity does not create an unreviewed safety question.

UFSAR SECTION 14.6

A discrepancy between the UFSAR assumed Main Steam Isolation Valve (MSIV) closure time for the Main Steam Line Break (MSLB) radiological consequence evaluation and the MSIV closure time used in the design basis calculation for MSLB radiological consequences exists. The Custom Technical Specifications (CTS) and Improved Technical Specification (ITS) MSIV required maximum closure time is 5 seconds, the design basis assumed MSIV closure time is 5.5 seconds for the MSLB dose evaluation and the UFSAR assumption for the MSLB dose evaluation closure time is 10.5 seconds.

This change from an MSIV assumed closure time of 10.5 seconds to 5.5 seconds for the MSLB dose assessment is consistent with SRP 15.6.4 and Regulatory Guide 1.5. The ITS, CTS, FSAR, and the design criteria will be consistent requiring the MSIV closure to be less than or equal to 5 seconds plus allow 0.5 seconds of additional time for instrumentation response time.

This change supports the MSIVs safety-related functions, safety margin, to prevent exceeding radiation release rates excess of 10CRF100 guidelines in the event of a steam line break outside of primary containment. This change does not alter the MSIVs ability to provide this engineered safety feature. This evaluation and the safety assessment discusses the affected portion of the MSIV closure time and shows the CTS and ITS operability of the system is not affected. The change in methodologies from the current design basis to a design basis that is consistent with the actual MSIV operating times is not an unreviewed safety question, and the radiological consequences from this change are well within the General Design Criteria 19 and 10CFR100 limits. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

UFSAR APPENDIX M

High energy line breaks in the reactor building are detected by either high flow sensors or high area temperature detectors. The setpoints for this instrumentation is influenced by both the location and size of postulated pipe breaks. In the past for 10CFR50.49 environmental qualification of electrical equipment, BFN has postulated pipe breaks to occur in high energy lines based solely on pipe routing. Breaks were postulated to occur at any location along a pipe. Furthermore, a complete spectrum of break sizes (double-ended, intermediate sized and critical crack) were assumed at each break location.

The NRC has provided guidance via Generic Letter 87-11 that allows plants to base the location and size of postulated breaks for 10CFR50.49 environmental qualification of electrical equipment on pipe stress criteria. This guidance has been previously inserted into BFN Design Criteria BFN-50-C-7105 and implemented for Civil pipe break studies as documented by CEB Report 88-06-C and calculation CD-Q3999-950377. The changes to Appendix M addressed in this safety evaluation denote that this guidance is applicable to the postulation of high energy line breaks for the purposes of 10CFR50.49 environmental qualification of electrical equipment.

This change is based upon guidance issued by the NRC. The Generic Letter states that these new criteria may be applied without prior NRC approval unless such changes conflict with the license or Technical Specifications. BFN license Item 2.C(4) pertains to actions related to piping integrity; however, the condition allowed the plant to not be in complete conformance to the final report on piping integrity until the first refueling outage. The UFSAR change being evaluated in this safety evaluation does not apply to this license condition. The BFN Technical Specifications do not address postulated pipe break locations. Thus, a review of the license and Technical Specification has concluded the provisions of the Generic Letter do not cause any conflicts and the implementation of the Generic Letter does not constitute an unreviewed safety question.

UFSAR SECTION F.4.f

This change revises UFSAR Appendix F, Section F.4.f, to delete language referring to "incremental fuel damage" that might be associated with long term post-accident core cooling. This is a descriptive enhancement to remove language that is not necessarily correct on a technical basis and which might create misunderstanding about the real nature of long term post-accident core cooling.

This change affects UFSAR text only and does not change or affect any system design parameters associated with any margin of safety as provided in the Technical Specifications or its bases. Therefore, no unreviewed safety question is involved.

UFSAR APPENDIX M, SECTIONS M.2 AND M.6.9

Appendix M of the UFSAR is being revised to clarify that current pipe break rules do not require a coincident break of Reactor Core Isolation Cooling (RCIC) and main steam in the valve vault. Current civil pipe break rules and evaluation results, coupled with the physical configuration of the piping and supports in the main steam valve vault, demonstrate that the rupture of the 26" main steam line will not result in the consequential rupture of the 4" RCIC line. A more recent valve vault pressure analysis evaluated the pressure transient without the coincident RCIC line rupture. The UFSAR was updated to reflect the peak pressure results of the new analysis; however, the update failed to clearly denote that the RCIC steam line rupture was not considered in the analysis. Similarly, the current Environment Qualification (EQ) analyses do not consider the coincident RCIC break; however, this assumption was inadvertently not removed from the text in UFSAR Section M.6.9. Also, the UFSAR was revised to clarify that the older pressure analysis retained from the Preliminary Safety Analysis Report (PSAR) and initial startup assumed the RCIC steam line contributes to the pressure transient; however, neither the current pressure analysis nor the current EQ analysis consider the RCIC steam line as a potential additional steam source.

Therefore, since the UFSAR document and its supporting calculations will be in conformance with the current design and licensing basis, it is concluded that the proposed activity does not create an unreviewed safety question.

UFSAR APPENDIX M, SECTIONS M.5.4 AND M.6.1

The text in BFN UFSAR Section M.5.4 is being revised to remove the reference to the postulated locations of critical cracks since the methodology to locate critical cracks is contained in Section M.6. The text change to Section M.6.1 provides clarification as to where critical cracks were assumed to occur in all high energy piping.

This change only clarifies the existing text relative to pipe rupture and does not create any new failure modes nor does it challenge the ability of any safety-related system to perform its intended function. No unreviewed safety question is involved.

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SUMMARY OF OTHER SAFETY EVALUATIONS

DRAWING 0-48N954 REVISION 002

This safety evaluation evaluates the interim plant configuration associated with the temporary removal of all three of the top layer reactor well shield blocks (Plug E) as permitted by Design Drawing 0-48N954 Revision 002. The drawing will allow for the removal of the top layer of reactor well shield blocks prior to beginning outage activities and when reactor cool down is initiated or for a period of up to 10 days when the reactor is at full power. Additionally, the note will state that it is unacceptable to lift and remove the second layer of reactor well shield blocks (Plug F) until the reactor is in cold shutdown and primary containment is not required. The reactor well shield blocks are located in the reactor building and centered above the Units 1, 2, and 3 reactors at Elevation 664' on the refuel floor. The drawing being revised is classified as a Category 4 drawing, does not depict configuration information, and is not necessary to be maintained as-constructed, but is invoked only with output documents that reference it.

One layer of shield blocks is capable of performing all of the required plant safety functions. The second layer is redundant. Therefore, removal of the top layer of the shield blocks for the time period specified in the safety assessment/safety evaluation has been demonstrated to be acceptable from a nuclear safety standpoint.

This change does conflict with a process summarized in Section 5.2.4.6 of the SAR but does not require changes to procedures or instructions described in the SAR and does not create any new procedures or instructions. Neither does this proposed note and drawing revision affect the operational characteristics of any system described in the SAR nor do they affect the plant's ability to comply with the requirements of the Technical Specifications. No unreviewed safety question is involved.

EMERGENCY OPERATING INSTRUCTION PROGRAM MANUAL

This safety evaluation addresses the use of the Boiling Water Reactor Owners' Group (BWROG) Emergency Procedure and Severe Accident Guidelines (EPG/SAG) and its Appendices Revision 1 to revise the BFN Emergency Operating Instructions Program Manual (EOIPM). Specifically the update of the EOI flowcharts, calculations, support procedures, and the implementation of the new Severe Accident Management Guideline (SAMG) flowcharts. This safety evaluation also addresses the draft Plant Specific Technical Guideline (PSTG)/SAMG which is part of the EOIPM.

This change to the EOIPM and EOI flowcharts and implementation of SAMG flowcharts is neither a special test nor an experiment not described in the SAR. The changes to the EOIPM and EOI flowcharts and implementation of the SAMG flowcharts do not affect any information in the SAR or deviate from the description in the SAR by changing the system design or functional requirements, the technical content of text, tables, graphs, or figures. This is because the systems are used in accordance with the system design and functional requirements as described in the SAR as long as the events fall within the design basis. The Integrated Computer System (ICS) displays will also be changed to provide consistency with the revised EOIs.

No unreviewed safety question (USQ) is involved. Where a USQ might have been involved, the procedures have been changed to address the SAR assumptions.

TEMPORARY STRUCTURE CONTROL FORM 1-97-001-RB

The safety evaluation for Temporary Structure Control Form 1-97-001-RB (see TVA BFN 1997 Annual Operating Report) was revised to remove the statement for painting an area on the floor around the existing cage on the refuel floor and to complete screening review questions to add justifications to Section C. The determination that no unreviewed safety question exists is still valid.

TEMPORARY STRUCTURE CONTROL FORM 1-98-001-RB

This safety evaluation addresses a Temporary Structure Control Form (TSCF) written to document the temporary structure used for an Operations Field Shack located on elevation 639 of the Unit 1 reactor building. This facility is used to house operations personnel during their daily duties and anytime they are required to be in the plant area for extended periods of time.

The temporary structure is constructed of sheet metal panels approximately 3'-8" wide by 7'-4" high with a 2" rolled flange which is welded at the corners. The sheets are approximately 3/32" thick. The interior panels are attached together with American Society for Testing Materials (ASTM) A307 minimum bolts by 1/4" long and located 6" on centers. The interior joints of the roof panels are attached together in the same manner as the interior joints of the side panels. The corner panels are attached together with 2' x 2' x 1/4" angle bolted to the two side panels with ASTM A307 minimum bolts located 6" on centers. The roof panels are attached to the side panels in the same manner as the corner panels. The metal wall panels are attached directly to the floor. The seismic justifications are detailed in Engineering Work Request EWR 98-1-303-063. There is an air conditioner in one of the interior panels on the South wall of the structure. There are windows made of plexiglass in frames on the North and South sides of the structure and in the door.

This temporary structure is located in proximity of the Unit 1 Recirculation Motor Generator Set 1A, Pressure Suppression Chamber Head Tank, and Panel 25-180 located on Elev. 639 of the Unit 1 reactor building. Temporary power is being furnished to the structure by a 480-V welding receptacle located on the South side of the column between R3 and R4 and S-Line. This temporary structure will not affect any equipment in its vicinity, because it has been installed in accordance with the significant requirements (construction details and electrical installation) and seismically justified by EWR 98-1-303-063. The EWR states how the temporary structure shall be installed to provide for its seismic justification, which is required for any temporary structure located in the reactor building. The 480-V welding receptacle is shown on Drawing 1-45N1756-8 and is fed from 480-V reactor building vent board 1B, Compartment 11D1, shown on Drawing 1-45N1756-4. Also, the phone is a non-safety-related system. These supplies are from existing plant features and are not required to have any permanent or temporary change made to them. Even if the circuit were to fail it would only affect the other receptacles located on this circuit, which is fed from the 480-V reactor building vent board 1B, Compartment 11D1. This temporary structure is not close to any Technical Specification equipment and cannot move to the vicinity of any Technical Specification equipment, and therefore, will not reduce the margin of safety as defined in the Technical Specifications. This change does not involve an unreviewed safety question.

TEMPORARY STRUCTURE CONTROL FORM 3-98-002-RB

This safety evaluation addresses a Temporary Structure Control Form (TSCF) which requests a temporary enclosure for Integrated Leak Rate Test Group (ILRT) use during the Unit 3 Cycle 8 outage. This enclosure is required to be climatically controlled, because it will house all of the computers and test equipment for use prior to and during the outage. During the testing it will be used as a controlled test station. The temporary enclosure requested is to be a Tube-Loc (Tube and Coupler) scaffold frame, with Herculite attached to the frame, structure (11'-0" x 15'-0" x 8'-0" high) and is to be located on elevation 593 of the Unit 3 reactor building between R16 and R17 at S and T-line.

The frame for the enclosure will be erected with Tube-Loc (Tube and Coupler) scaffolding members with a galvanized finish. The frame will be tied together with heavy duty couplers made especially for this purpose. The framing will be as shown on the sketches and as described in Engineering Work Request (EWR) 98-3-303-50. The temporary structure will be laterally braced by cables from each corner of the frame. The cables will be 3/16" Steel Aircraft Cable or 3/8" steel cable, as used for scaffolding restraints. The cables will run from each corner of the frame to existing anchors in the walls/columns. The frame shall be anchored to the floor with four ½" diameter or larger SDI concrete anchors located on the frame. The anchor locations are one anchor in each corner. The frame is to be connected to the SDI anchors by bolting the Tube-Loc clamps through eyebolts installed in the SDI shells. The framed Tube-Loc structure shall be sufficiently braced to ensure the frame behaves in a rigid manner.

The temporary enclosure is seismically qualified by instructions given in EWR98-3-303-50. This temporary enclosure cannot move from its location once it is tied down in accordance with the EWR. The electrical power supply is coming from a non-safety-related welding receptacle connected to a breaker in the reactor building vent board 3B, Compartment 9D2. This temporary enclosure is not close to any Technical Specification equipment and cannot move to the vicinity of any Technical Specification equipment, and therefore, it will not reduce the safety margin of safety as defined in the Technical Specifications. Therefore, no unreviewed safety question is involved.

TEMPORARY STRUCTURE CONTROL FORM 3-98-003-RB

This safety evaluation addresses a Temporary Structure Control Form (TSCF) which requests a temporary enclosure for InService Inspection (ISI) Group use during the Unit 3 Cycle 8 outage. This enclosure is required to be climatically controlled, because it will house all of the computers and test equipment for use prior to and during the outage. During the testing it will be used as a controlled test station. The temporary enclosure requested is to be a Tube-Loc (Tube and Coupler) scaffold frame, with Herculite attached to the frame, structure (11'-0" x 15'-0" x 8'-0" high) and is to be located on elevation 593 of the Unit 3 reactor building between R17 and R18 at T and U-line.

The frame for the enclosure will be erected with Tube-Loc (Tube and Coupler) scaffolding members with a galvanized finish. The frame will be tied together with heavy duty couplers made especially for this purpose. The framing will be as described in Engineering Work Request (EWR) 98-3-303-49. The temporary structure will be laterally braced by Tube-Loc members at the NE, SE, and SW corners of the frame. There will be an additional cable or Tube-Loc brace located from the NW corner of the frame. The cables will be 3/16" Steel Aircraft Cable or 3/8" steel cable, as used for scaffolding restraints. Additional Tube-Loc framing shall be provided, as required, to support the air conditioning unit. This frame shall be attached to the main framing sufficiently to prevent overturning from accidental bumping or a seismic event. The air conditioner shall be located on the East end of the temporary structure to keep it away from the safety-related equipment located close to the east end of the structure.

This temporary enclosure cannot move from its location once it is tied down in accordance with the EWR. The electrical power supply will be coming from the non-safety-related loop line through disconnect switch 3-SW-276-0020 or a welding receptacle connected to a breaker in the reactor building vent board 3B, Compartment 9D2.

The power supplies are from existing plant features and are not required to have any permanent or temporary change made to them. This temporary enclosure is close to Technical Specification equipment, but cannot move to the vicinity of any Technical Specification equipment, and therefore, will not reduce the margin of safety as defined in the Technical Specifications. Therefore, no unreviewed safety question is involved.

TEMPORARY STRUCTURE CONTROL FORM 3-98-004-RB

This safety evaluation addresses a Temporary Structure Control Form (TSCF) which requests a temporary enclosure for General Electric (GE) Group use during the Unit 3 Cycle 8 outage. This enclosure is required to be climatically controlled because it will house all of the computers and test equipment for use prior to and during the outage. During the testing, it will be used as a controlled test station. The temporary enclosure requested is to be a Tube-Loc (Tube and Coupler) scaffold frame, with Herculite attached to the frame, structure (12'-0" x 12'-0" x 8'-0" high) and is to be located on elevation 565 of the Unit 3 reactor building at the North end of the East bank control rod drive hydraulic control units between R20 and R21 at N - P-line.

The frame for the enclosure will be erected with Tube-Loc (Tube and Coupler) scaffolding members with a galvanized finish. The frame will be tied together with heavy duty couplers made especially for this purpose. The framing will be as described in Engineering Work Request (EWR) 98-3-303-53. The temporary structure will be laterally braced by Tube-Loc members at the corners of the frame. There will be an additional Tube-Loc brace located from the SW corner of the frame over to within 12" of the eye bolt which is bracing the corner. The Tube-Loc braces will run from the NW and SW corner of the frame to SDI anchors and eyebolts installed in the walls. The Tube-Loc braces from the NE and SE corners will be attached to the handrail which goes down to the NE Quad. All lateral restraints shall be attached to the top of the temporary structure to maximize the resistance to any potential overturning of the structure. No anchors are required to be placed in the floor, due to no close proximity to any safety-related equipment. The framed Tube-Loc structure shall be sufficiently braced to ensure the frame behaves in a rigid manner. Additional Tube-Loc framing shall be provided, as required, to support the air conditioning unit. This frame shall be attached to the main framing sufficiently to prevent overturning from accidental bumping or a seismic event. The air conditioner shall be located on the North end of the temporary structure.

The temporary enclosure is seismically qualified by instructions given in EWR 98-3-303-53. This temporary enclosure cannot move from its location once it is tied down in accordance with the EWR. The electrical power supply will be coming from the non-safety-related loop line through disconnect switch 3-SW-276-0020 or a welding receptacle connected to a breaker in the reactor building vent board 3B, Compartment 9D2. The power supplies are from existing plant features and are not required to have any permanent or temporary change made to them. This temporary enclosure is close to Technical Specification equipment, but cannot move to the vicinity of any Technical Specification equipment, and therefore, it will not reduce the margin of safety as defined in the Technical Specifications. Therefore, no unreviewed safety question is involved.

TECHNICAL REQUIREMENTS MANUAL (UNITS 1, 2, AND 3)

This change involved taking the designated content of the Custom Technical Specifications (CTS) and incorporating the contents into the new Technical Requirements Manual (TRM). The designated content of the CTS is based on the Justification for Changes (also referred to as the DOCs) included in License Amendment TS-362, Improved Standard Technical Specifications. The relocation of the designated CTS requirements to the TRM and the changes in CTS content which are directly based on the new ITS (TS-362) requirements will be covered by the NRC Safety Evaluation Report for TS-362 and are thus outside the scope of the

10CFR50.59 process. The content of the new TRM which is the same as the original CTS content is covered by the SERs for the pre-existing CTS. The changes in content which were not directly based on the new ITS requirements are addressed by this safety evaluation and associated safety assessment.

Administrative changes were made in the conversion of the CTS to the TRM. These changes include the format of presentation of the requirements and the associated Bases. Reformatting and renumbering in the TRM was made consistent with the ITS (which is in accordance with BWR Standard Technical Specifications, NUREG 1433). As a result, the TRM will be more readily understandable by plant operators as well as other users. The reformatting includes replacement of CTS Limiting Conditions for Operation (LCOs) with TRM LCOs and ACTIONS (Conditions, Required Actions, and Completion Times) and CTS Surveillance Requirements with Technical Surveillance Requirements (TSRs). The content of the Bases was expanded as a result of the reformatting. The additional content is considered an administrative change since the Bases only clarifies the LCOs, Applicability, Required Actions, and Technical Surveillance Requirements and was developed based on the CTS Bases, the UFSAR, ITS Bases, and other appropriate documents. Additional administrative changes made during the conversion includes the addition of information for proper use and administration of the TRM (section added describing logical connectors, etc.), use of different but equivalent terminology (utilization of ITS terminology for Modes, change in Frequency from daily to 24 hours, etc.), combining (deletion) of redundant requirements, adding surveillances currently performed to satisfy the CTS requirements although not directly specified in the CTS, and adding clarification for requirements (adding clarifying Notes, etc.).

Technical changes were also made in the conversion. These technical changes were required due to the requirements of the ITS or were otherwise deemed appropriate. Technical changes were made to be consistent or not conflict with the requirements of the ITS. These changes included establishing TRM LCO Applicability consistent with the ITS LCO Applicability for the supported feature and establishing Required Actions, Surveillance Frequencies, and Notes to implement directives of the ITS. Technical changes were made to Required Actions as required due to changes in Applicability, confusion in implementing the CTS actions, or in cases where actions other than the CTS actions were more appropriate. Technical changes to the Completion Times for Required Actions were made as appropriate for conditions where no completion time was provided for the CTS action or where the Completion Time allowances in the ITS for similar conditions were more appropriate. Revisions to surveillances were made as appropriate based on similar ITS allowances and engineering judgment. Revisions to surveillances included changes in Frequencies and allowed out of service times for surveillances. Technical changes were made to the Applicability of LCOs as appropriate for conditions where no specific Applicability was provided for the CTS LCO or in cases where an Applicability other than the CTS Applicability was more appropriate. Additionally, Allowable Values were provided in the TRM in lieu of trip settings as similarly done in the ITS. The required trip settings are an operational detail not directly related to the operability of the instrumentation and are included in the appropriate plant procedures. The Allowable Value is the required limitation of the parameter and this value is retained in the TRM. TVA's methodology for determination of setpoints utilized the CTS "trip level settings" as the Allowable Value in establishing the nominal trip setpoints.

Most of the changes made to the CTS requirements during the conversion to the TRM involved administrative changes, were required to prevent conflicts with the ITS, or were based on similar requirements of the ITS. The changes to the CTS requirements have been evaluated and determined (1) not to increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR); (2) not to create a possibility of an accident or malfunction of a different type than evaluated previously in the SAR; and (3) not to reduce a margin of safety as defined in the basis for any Technical Specification. Therefore, the proposed change does not involve an unreviewed safety question.

UNIT 2 TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 3.3.1.1.5 BASES

This evaluation for Technical Specification (TS) Surveillance Requirement (SR) 3.3.1.1.5 requires verification of Source Range Monitor/Intermediate Range Monitor (SRM/IRM) overlap prior to withdrawing the SRMs from the fully inserted position. The intent of this SR as stated in Technical Specification Bases 3.3.1.1 is to ensure that reactor power will not increase into a neutron flux region without adequate indication. Technical Specification SR 3.3.1.1.5 Bases provide the definition of what constitutes acceptable SRM/IRM overlap. The Bases currently state that overlap between SRMs and IRMs exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are above mid-scale on Range 1 before SRMs have reached the upscale rod block. The Bases are being revised utilizing NADP-6 as provided in Technical Specification 5.5.10 to state that overlap between SRMs and IRMs exists when previously verified IRM downscale indications have cleared and IRM count rates are on-scale and trending higher prior to the SRM count rates reaching the SRM HI (rod block) setpoint allowable value of 10^5.

The IRMs are assumed in the safety analyses to mitigate neutron flux excursions caused by positive reactivity additions. The IRM upscale trip function is explicitly assumed to actuate to terminate the positive reactivity addition due to a continuous rod withdrawal event. As stated in Technical Specification Bases B 3.3.1.1, the IRMs provide diverse protection from the Rod Worth Minimizer (RWM). The proposed change does not impact the ability of the IRMs to perform that safety function. In addition to the SRM/IRM overlap surveillance, operability of the IRMs is verified prior to each plant startup by performing a logic system functional test (LSFT) and channel calibration (quarterly performance) if not performed within the previous 18 months and by a channel functional test if not performed within the previous 7 days. Operability of the IRMs is also verified every 24 hours while in Mode 2 by performing a channel check. These surveillance requirements along with the Unit Operator (UO) practice of periodically monitoring the IRMs during a startup ensure that the IRMs are capable of performing their intended safety function.

This Technical Specification Bases change does not alter any acceptance limits previously approved by the NRC. In the accident analyses, the Neutron Monitoring System (IRMs and APRMs) is assumed to actuate to prevent exceeding fuel thermal limits during assumed accidents and transients. Specifically, the IRM upscale trip is assumed to terminate a neutron flux excursion caused by a continuous control rod withdrawal event during reactor startup while below the normal power range. The proposed Technical Specification Bases change has no impact on the requirements. As such, the fuel thermal acceptance limits are not impacted by this change. Since there is no change in either the acceptance limit or the failure point, and the margin of safety is defined as the difference between these two parameters, then the margin of safety will also not be altered. Therefore, no unreviewed safety question is involved.

UNIT 3 TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 3.3.1.1.5 BASES

This evaluation for Technical Specification (TS) Surveillance Requirement (SR) 3.3.1.1.5 requires verification of Source Range Monitor/Intermediate Range Monitor (SRM/IRM) overlap prior to withdrawing the SRMs from the fully inserted position. The intent of this SR as stated in Technical Specification Bases 3.3.1.1 is to ensure that reactor power will not increase into a neutron flux region without adequate indication. Technical Specification SR 3.3.1.1.5 Bases provide the definition of what constitutes acceptable SRM/IRM overlap. The Bases currently state that overlap between SRMs and IRMs exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are above mid-scale on Range 1 before SRMs have reached the upscale rod block. The Bases are being revised utilizing NADP-6 as provided in Technical Specification 5.5.10 to state that overlap between SRMs and IRMs exists when previously verified IRM downscale indications have cleared and IRM count rates are on-scale and trending higher prior to the SRM count rates reaching the SRM HI (rod block) setpoint allowable value of 10^5.

The IRMs are assumed in the safety analyses to mitigate neutron flux excursions caused by positive reactivity additions. The IRM upscale trip function is explicitly assumed to actuate to terminate the positive reactivity addition due to a continuous rod withdrawal event. As stated in Technical Specification Bases B 3.3.1.1, the IRMs provide diverse protection from the Rod Worth Minimizer (RWM). The proposed change does not impact the ability of the IRMs to perform that safety function. In addition to the SRM/IRM overlap surveillance, operability of the IRMs is verified prior to each plant startup by performing a logic system functional test (LSFT) and channel calibration (quarterly performance) if not performed within the previous 18 months and by a channel functional test if not performed within the previous 7 days. Operability of the IRMs is also verified every 24 hours while in Mode 2 by performing a channel check. These surveillance requirements along with the Unit Operator (UO) practice of periodically monitoring the IRMs during a startup ensure that the IRMs are capable of performing their intended safety function.

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UNIT 3 TECHNICAL REQUIREMENTS MANUAL BASES 3.3.4

This safety evaluation addressed Unit 3 Technical Requirements Manual (TRM) TR 3.3.4 which is being revised to accommodate DCN W39479A modifications which installs a new General Electric NUMAC microprocessor-based Power Range Neutron Monitoring System (PRNMS) and Rod Block Monitor System (RBMS). Implementation of the design modifications requires corresponding changes to the TRM TR 3.3.4 and associated bases. The TRM changes have been grouped into categories according to the type of change (e.g., simple format) being implemented. The affected TRM section/table is shown in brackets after each category description.

Simple formatting changes which are the result of the TRM revisions. [LCO 3.3.4 CONDITIONS D AND E, Table 3.3.4-1 Function 7, TSR 3.3.4.9]

Formatting changes involving the scram discharge volume water level instrument channel calibration surveillance frequency description are also being incorporated. [3.3.4.8 and Table 3.3.4-1 Functions 4a and 4b]

A note has been added to TSR 3.3.4.7 for calibration of Average Power Range Monitors (APRMs) which states that neutron detectors are excluded. This is considered an administrative change since the channel calibration definition of CTS 1.V.11 excluded non-calibratable components. This change is consistent with notes provided for these instruments in TS 3.3.1.1 and 3.3.1.2. [TSR 3.3.4.7]

The functional test surveillance frequency for APRM functions was changed from once per month to once per 184 days. The calibration frequency for APRM functions requiring calibration was changed from once per 92 days to once per operating cycle. [TSR 3.3.4.2, 3.3.4.6, and 3.3.4.7]

The flow bias comparator and upscale table functions and their surveillance requirements were deleted as a result of the PRNMS modification. The PRNMS modifications have been reviewed and approved by NRC via TS Amendment No. 213 and its associated Safety Evaluation Report. [Table 3.3.4-1 Functions 6a and 6b]

The PRNMS modification also allows the channel check requirements for RBM functions to be deleted. [Table 3.3.4-1 Functions 5a, 5b, and 5c]

These TRM changes do not alter any acceptance limits previously approved by the NRC. In the accident analyses, the Neutron Monitoring System (IRMs and APRMs) is assumed to actuate to prevent exceeding fuel thermal limits during assumed addidents and transients. These TRM changes are not postulated to have any impact on the operability of this instrumentation which continues to be verified operable by Technical Specification surveillance requirements. As such, fuel thermal acceptance limits are not impacted by this change. The rod block monitor function is not assumed in any safety analyses. The intent of the RBMS is to ensure the local core reactivity will not increase in such a manner so as to cause fuel damage. The RBMS also serves to provide Operations personnel with an indication of local relative power during control rod movement. This change continues to ensure that this intent is maintained. Additionally, this change does not add, remove, or otherwise modify any plant components or systems, or change any current methods of operation. The margin of safety will not be altered by these TRM changes because there is no change in any factors (e.g., trip logic configuration) that influence the margin of safety. This change does not involve an unreviewed safety question.

UNIT 3 TECHNICAL REQUIREMENTS BASES FOR TR 3.6.4 AND TR 3.6.5

This safety evaluation supports a revision to the BFN Unit 3 Technical Requirements Manual (TRM) Bases. The TRM Bases revision is being made to support implementation of 5% power uprate on BFN Unit 3. On Unit 3 the rated thermal power is being increased from 3293 MWt to 3458MWt. NRC has issued amendments allowing operation of BFN Units 2 and 3 at 3458 MWt and approved the changes to Technical Specifications to implement the uprated power operation. This change requires the stated maximum peak containment pressure to be increased from 49.6 psig to 50.6 psig in the affected Bases.

This change revises the Bases discussion of certain TRM sections, but no Bases sections of the Technical Specification are affected by this change. The maximum peak containment pressure remains within the structural design limits of the primary containment; thus, the margin of safety provided by operation within the existing structural design limit is not affected. This activity does not reduce the margin of safety as defined in the basis for any Technical Specification. This change does not involve an unreviewed safety question.

IMPROVED TECHNICAL SPECIFICATIONS BASES B 3.6.4.3

Improved Technical Specification (ITS) Bases Section B 3.6.4.3 , "Background", is being revised to clarify the discussion concerning Secondary Containment leakage limit. The Bases will be revised to state that the Secondary Containment membrane limits infiltration to not more than design flow requirements for the Standby Gas Treatment System (SGTS) to be able to evacuate the volume of the Secondary Containment to a negative pressure 0.25" water gage. This system's nuclear safety function is to provide a means of limiting the offsite dose due to a Design Basis Accident to an insignificant portion of the dose limits specified in 10CFR20 and 10CFR100. This function is achieved by a low leakage membrane which encloses the air volume of the three reactor buildings and the refuel floor. The SGTS takes suction from the Secondary Containment and provides an elevated filtered release. The SGTS is sized to maintain the volume contained within the entire secondary containment membrane (all units), to a subatmospheric pressure of at least 0.25" of water gage relative to outside the membrane.

The revised Bases Section B 3.4.6.3 does not affect the function of the Secondary Containment System, nor does it affect any other system to mitigate their applicable design basis events. A physical modification is not required. The change meets the design requirements for the system as specified in Design Criteria, BFN-50-7064C, for Secondary Containment. The ITS and ITS Bases were reviewed for impact by the revision to ITS Bases B 3.6.4.3. This revision clarifies the discussion of allowable leakage into Secondary Containment. The change does not affect the description of any safety-related equipment. This change does not modify the function of any component or operation of the Secondary Containment System. Calculation, ND-Q0999-890031, "Secondary Containment Pressure Transient Analysis", documents secondary containment volume as 6,954,000 ft^3 per day which equates to a flow rate of 4829 cfm. Two of the three trains of SGTS are sized to maintain at least 0.25" of vacuum water gauge in secondary containment at a flow rate of ≤12,000 cfm. This is specified by Surveillance Requirement 3.6.4.1.4. Therefore, limiting secondary containment leakage to 100% volume per day is not required. An allowable surveillance inleakage value is specified in the ITS. The design basis margin is flow required to compensate for increase in leakage following a design basis event. The design basis margin calculated value of the increased leakage through the secondary containment boundary, due to a design basis event, is not altered by the Bases change. The ITS Bases section requiring revision does not affect or change the systems surveillance requirements. Therefore, the margin of safety is not reduced as defined in the basis for any ITS document. Therefore, no unreviewed safety question is involved.

TECHNICAL REQUIREMENTS MANUAL SECTION 3.5.4 AND BASES SECTIONS 3.3.3.1 AND 3.5.4

Technical Requirements Manual (TRM) Section 3.5.4 and Bases Sections 3.3.3.1 and 3.5.4, "Maintenance of Filled Discharge Piping", for Units 1, 2 and 3 are being revised to specify additional actions when the charging pressure at the discharge of the Residual Heat Removal (RHR) and Core Spray pumps falls below 48 psig. Currently the only action specified in the TRM if the charging pressure decreases below the 48 psig requirement is to immediately declare the affected Emergency Core Cooling System (ECCS) subsystem (loop) inoperable. The proposed additional actions will require immediate partial restoration of pressure to a newly specified value to maintain the immediate operability of the ECCS loop; an action to restore the pressure at 48 psig in 1 hour; and an action to vent the affected loop within 4 hours if pressure was below 48 psig for more that 10 minutes.

Since the RHR and Core Spray Systems are not adversely affected by this change, this activity cannot reduce the margin of safety as defined in the basis for any Technical Specification. The changes are extremely conservative and are strictly administrative. The requirement to maintain 48 psig in the ECCS discharge piping has not been changed. Actions have been added in the event pressure drops below 48 psig to immediately restore the discharge pressure to a value which ensures the piping is maintained filled and to vent the ECCS piping if pressure was below 48 psig for more than 10 minutes. Therefore, these changes do not adversely impact any ECCS function.

This change does not involve an unreviewed safety question.

RADIOACTIVE CONTAMINATION OF THE NORMALLY NONRADIOACTIVE DEMINERALIZED WATER SYSTEM, AUXILIARY DECAY HEAT REMOVAL SECONDARY SIDE AND STATION SUMPS

This safety evaluation is being written due to the radioactive contamination of the normally nonradioactive Demineralized Water System (System 2), Auxiliary Decay Heat Removal (ADHR) secondary side (System 72) and Station Sumps (System 40). A safety assessment/safety evaluation is required by NRC IE Bulletin No. 80-10, "Containment of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment", dated May 6, 1980, when it is considered necessary to continue operation of a non-contaminated system as contaminated. Although there is no intention to continue operating the recently contaminated non-contaminated systems as contaminated, this safety assessment/evaluation is being written as a conservative action and may also by used to cover continued system operation when future contamination events occur. The systems affected are not important to safety and have no safety functions other than ADHR meeting the requirements for secondary containment penetrations. This condition does not contribute to any of the design basis accidents and radioactive releases are much less severe (within 10CFR20 limits).

Continued operation of these systems as radioactively contaminated is acceptable from a nuclear safety standpoint. No unreviewed safety question is involved.

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT

JANUARY 1 - MAY 31, 1999
SUMMARY OF
SAFETY EVALUATIONS

JANUARY 1 - MAY 31, 1999

SUMMARY OF SAFETY EVALUATIONS FOR CORE COMPONENT DESIGN CHANGE REQUESTS AND CORE OPERATING LIMITS REPORTS

UNIT 2 CYCLE 10 CORE OPERATING LIMITS REPORT

This safety evaluation addresses revisions to the BFN Unit 2 Cycle 10 Core Operating Limits Report (COLR). The COLR is being revised to extend the Average Planar Linear Heat Generation Rate (APLHGR) operating limit data (COLR - Figure 8) for the BP8DRB299 fuel type. The BF2 3D Monicore system indicates that the plant will exceed the last APLHGR exposure point for this fuel type before the unit shuts down on 4/11/99. General Electric (GE) has performed an analysis to support extending the allowable nodal exposure from 42.131 out to 44 GWD/ST. The GE analysis takes credit for inherent margins existing in the APLHGR limits originally generated for the GE7B (barrier) fuel type. Operation within the revised APLHGR limits will ensure that the fuel continues to operate within the analyzed and allowable envelope of power versus exposure. Thus, all design and licensing criteria are satisfied including those which address response to Abnormal Operational Transients and Design Basis Accidents.

There is no increase in the probability or consequences of an accident or malfunction of equipment from that previously evaluated in the SAR. The proposed activity does not create the possibility of a different type of accident or malfunction from that previously evaluated in the SAR. The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, it does not involve an unreviewed safety question.

UNIT 2 CYCLE 11 CORE OPERATING LIMITS REPORT

This safety evaluation addresses the BFN Unit 2 Cycle 11 reload core design, incore fuel shuffle, Core Operating Limits Report (COLR), and associated FSAR revisions. The reload core design and licensing analyses for Cycle 11 were performed by General Electric (GE) with results documented in the Supplemental Reload Licensing Report. The GE analyses were performed in accordance with NRC approved methodology as described in GE licensing topical report GESTAR II (NEDE-24011-P-A-13). Operating limits for the cycle (i.e., Linear Heat Generation Rate, Minimum Critical Power Ratio, and Average Planar Linear Heat Generation Rate) are incorporated into the Unit 2 COLR.

The Cycle 11 core is a Modified Control Cell Core design which includes 300 fresh GE13 fuel assemblies and 17 new Marathon control blades. Cycle 11 is the initial 5% power-uprate cycle for Unit 2. Operating flexibility options analyzed include Increased Core Flow, Final Feedwater Temperature Reduction, Feedwater Heaters Out of Service, End of Cycle Recirculation Pump Trip Out Of Service, Turbine Bypass Valves Out Of Service, and Recirculation Single Loop Operation. Consistent with NRC commitments, GE also performed a cycle specific Safety Limit Minimum Critical Power Ratio (MCPR) analysis.

The Unit 2 Reload 10 refueling will be accomplished by an incore fuel shuffle using a TVA developed shuffle sequence. Shutdown margin calculations have been performed for each unique core configuration in the planned shuffle sequence.

There is no increase in the probability or consequences of an accident or malfunction of equipment from that previously evaluated in the SAR. The proposed activity does not create the possibility of a different type of accident or malfunction from that previously evaluated in the SAR. The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, it does not involve an unreviewed safety question.

UNIT 3 CYCLE 9 CORE OPERATING LIMITS REPORT

This safety evaluation addresses revisions to the BFN Unit 2 Cycle 10 Core Operating Limits Report (COLR) and to the Unit 3 Cycle 9 COLR. Both COLRs are being revised to implement a Technical Specification change allowing Single Loop Operation (SLO). Analyses have been performed to determine required changes to operating limits and instrument setpoints necessary to support SLO. Additional uncertainties in core flow indication and incore instrument readings associated with SLO require an increase in the Safety Limit MCPR (SLMCPR). Therefore, an MCPR operating limit adder applicable to SLO is incorporated into the COLRs to reflect the increased SLMCPR. The LOCA design basis accident also becomes more severe with SLO due to a longer reflood time. To maintain acceptable peak clad temperatures, an APLHGR multiplier applicable to SLO is included in the COLRs. The inaccuracy in jet pump flows due to reverse flow through the inactive jet pumps during SLO affects the accuracy of the APRM flow-biased rod block trip setpoint. This revision to the COLRs includes a correction to the recirculation drive flow term to correct for this inaccuracy.

There is no increase in the probability or consequences of an accident or malfunction of equipment from that previously evaluated in the SAR. The proposed activity does not create the possibility of a different type of accident or malfunction from that previously evaluated in the SAR. The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, it does not involve an unreviewed safety question.

JANUARY 1 - MAY 31, 1999

SUMMARY OF SAFETY EVALUATIONS FOR FIRE PROTECTION REPORT REVISIONS

FIRE PROTECTION REPORT CHANGE NOTICE

This safety evaluation addresses Fire Protection Report Volume 1 which is being revised to a) delete a commitment and change associated responsibility of the Shift Manager, b) change titles to the Administrative Control Program (to support standardization with all TVAN sites), c) clarify responsibilities/qualifications of the Incident Commander (IC) under the Fire Emergency Response Organization, d) annotate Section 9.0 (Testing and Inspection) for relevant NRC commitments - providing cross reference between LCO's and NRC Commitments, and e) describe the new implementing documents in Section 10.0 (Fire Protection Commitments). Neither the administrative changes nor the text changes have an impact on nuclear safety. Henceforth, this change does not reduce the margin of safety as defined in the basis of any Technical Specification. This change does not involve an unreviewed safety question.

FIRE PROTECTION REPORT CHANGE NOTICE

This safety evaluation for changes in Fire Protection Report Volume 1 (FPR-1) is administrative in nature. The change to FPR-1, Appendix R Safe Shutdown Program, Section V - Testing and Monitoring - involves the removal of a reference to an instruction that is no longer required by Technical Specifications. The revision to FPR-1, Fire Protection Plan, Section 9.4.11.G.1 involves 1) a change in frequency of penetration inspections associated with the Unit 2 and 3 Steam Tunnels from once per 18 months to once per operating cycle, and 2) the elimination of surveillance requirement 9.4.11.G.1.c. This change does not 1) affect equipment important to safety, 2) increase the probability or consequences of an accident, 3) create the possibility of an accident, 4) reduce or affect the margin of safety, or 5) have any involvement with or impact on safety. This activity has been reviewed against the criteria of 10CFR50.59 and does not constitute an unreviewed safety question.

JANUARY 1 - MAY 31, 1999

SUMMARY OF SAFETY EVALUATIONS FOR PLANT MODIFICATIONS

DESIGN CHANGE NOTICE S40957A

This safety evaluation for DCN S40957A revises the design temperature of portions of the Condensate and Condensate Storage and Supply Systems. This drawing change documents the increased design temperature based on the calculations and evaluations documented in the safety assessment for this DCN. The design basis accidents which the Condensate System supports required that the condensate ring header and condensate storage tanks (CST) provide supply to emergency core cooling system (ECCS) pumps. Even though the CSTs are the preferred source of water, they are not required for accident mitigation because the suppression pool is the assumed source. However, the condensate ring header shall contain sufficient water to supply the HPCI pumps while transfer to the suppression pool is being made.

This documentation change to increase the design temperature of portions of the Condensate System does not affect the operation or function of the system. The changes do not prevent the Condensate System from performing its required functions during an accident or (AOT) abnormal operational transient and therefore the probability of occurrence of the consequences of an accident or malfunction are not increased. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE S41104A

DCN S41104A revises various drawings, including several drawings contained in the FSAR, to reflect plant conditions after the 5% uprate of BFN Unit 2. The proposed drawing changes are completely consistent with data and evaluations submitted to the NRC in support of the power uprate license amendment request, and they are consistent with the NRC safety evaluation in support of BFN Unit 2 TS Amendment 254. The proposed drawing changes have been evaluated against the criteria of 10 CFR 50.59, and this activity does not constitute an unreviewed safety question.

DESIGN CHANGE NOTICE S41105A

This safety evaluation supports DCN S41105A. The change being evaluated is the revision of Figure 1.6-29 of the FSAR. This figure shows the heat balance (thermal cycle) for the turbine-generator and other Balance of Plant (BOP) equipment. This figure is based on drawing 2-47K1110-13, which is being revised to show the effects of a 5% increase in thermal power on BFN Unit 2.

Chapter 14 of the FSAR lists several abnormal operational transients (AOTs) which could be affected by a change in BOP thermal cycle parameters. These events, which directly involve BOP equipment, include the following:

- Generator or Turbine Trips
- Bypass Valve Failures
- Loss of a Feedwater Heater
- Loss of Condenser Vacuum
- Loss of Feedwater Flow

None of the design basis accidents (DBAs) are potentially affected by these changes in BOP thermal cycle parameters. This change is simply a reflection of changed operating conditions which remain within the design envelopes of the affected equipment, no new credible failure modes are created by this change. Therefore, no unreviewed safety question is involved.

DESIGN CHANGE NOTICE S41164A

This SA/SE is written in support of DCN S41164A. This DCN provides documentation to allow for the use of any of three separate styles for the reactor head O-ring seals.

The reactor vessel is a vertical, cylindrical pressure vessel with hemispherical heads of welded construction. The reactor vessel is designed and fabricated for a useful life of 40 years based upon the specified design and operating conditions. The vessel for each unit is designed, fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 edition, Summer 1965 addenda (for Unit 2) or Summer 1966 addenda (for Unit 3).

The reactor vessel top head is secured to the reactor vessel by studs and nuts which are designed to be tightened with a stud tensioner. The vessel flanges are sealed by two concentric metallic O-rings designed for no detectable leakage through the inner or outer seal at any operating condition. The O-rings currently used at BFN are inconel tubes with 0.004-0.006" of silver coating. This DCN will revise the paperwork necessary to allow any of the three different O-rings supplied by the vendor to be used. FSAR Section 4.2.4.1 will be revised to remove the material designator of "stainless steel" from the description of the O-rings, since the O-rings can also be fabricated from other suitable metallic materials. No changes to any design criteria are required as a result of this DCN. This DCN does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE S41176A

DCN S41176A revises notes in drawings 2,3-47E858-1 based on revised Residual Heat Removal Service Water (RHRSW) flow data from calculation MD-Q0023-87106. This design basis calculation was revised due to calculation errors identified in BFNPER-9806960. The affected drawings appear in the FSAR as Figures 10.9-1a Sheet 2 and Sheet 3. The plant drawings and the FSAR are revised by this DCN to reflect the calculation revision. This a documentation change only, no physical change to the plant is required Therefore, no unreviewed safety question is involved.

DESIGN CHANGE NOTICE T26014A

This safety evaluation is written in support of DCN T26014A which installs a flow limiting orifice in the discharge line of each Reactor Water Cleanup (RWCU) pump, 2A and 2B, to prevent flow run-out when starting a pump. Also, the disc and trim in the RWCU demineralizer bypass valve, 2-FCV-69-8, and the RWCU return throttle valve, 2-FCV-69-12, are being modified and upgraded to provide better throttling characteristics on these valves.

The orifice flanges, orifice plate and the throttle valves' internals are installed in accordance with applicable design specifications and material requirements of the original design to ensure pressure boundary integrity under all modes of RWCU system operation and during accidents or transients. Therefore, the changes implemented by this DCN create no new failure modes that affect nuclear safety. This change does not constitute an unreviewed safety question.

DESIGN CHANGE NOTICE T39736A

The safety evaluation for DCN T39736A removes pressure indication switches 2-PIS-66-21A and 2-PIS-66-21B and temperature indicating switches 2-TIS-66-22A and 2-TIS-66-22B from the electrical trip interlock logic of the steam jet air ejectors. This will be accomplished by disconnecting the contacts of control relays 2-RLY-66-R1, 2-RLY-66-R2, 2-RLY-66-R3, and 2-RLY-66-R4 from the interlock logic inside panel 9-36. The control relays used for annunciation will remain intact.

Neither the steam jet air ejectors nor the Offgas System are classified as safety related. These changes do not degrade the design basis nor do they challenge safety systems assumed to function during an accident. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40199A

This safety evaluation is written in support of DCN T40199A which provides the design for the installation of a bypass line around the Residual Heat Removal (RHR) pump seal injection water heat exchangers and shutoff valves between the pumps and the heat exchangers located in the Unit 2 Reactor Building. This installation will provide a means to service RHR pump seal heat exchangers without removing the associated pumps from service. The bypass valve is closed during normal operation and will be open only when isolating the heat exchangers for servicing. One-half inch globe valves and test connections are added for local leak rate testing of the new isolation valves.

The design is installed in accordance with ASME Section III Class 2 requirements. No other safety related equipment is affected by this change. Therefore, the current Unit 2 Technical Specifications are not affected nor is the margin of safety reduced as a result of this design change. This does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40280A

This safety evaluation addresses the Steam Jet Air Ejectors (SJAEs) which provides offgas to the Offgas System. Upon loss of steam, the SJAEs are isolated. The Offgas System is isolated upon a high radiation signal from the offgas post treatment radiation monitors. These functionalities are not altered by the proposed modification. The proposed activity removes a possible failure point from a loop. The proposed activity will increase the efficiency of the Offgas System moisture separation function during periods in which the moisture separation function is strained due to an increase in environmental temperatures. Removing the moisture alarm and allowing both cooler condenser/moisture separators to be used at once for the Offgas System does not alter its function in any way nor does it impact its ability to processes gaseous radioactive wastes and controls their release to the atmosphere through the plant stack so that the total radiation exposure to persons outside the controlled area is as low as reasonably achievable and does not exceed applicable regulations. Therefore, this change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40346A

This safety evaluation is written in support of DCN T40346A which installs test connections for the Residual Heal Removal (RHR) and Core Spray systems for the purpose of pressure differential testing across 2-FCV-74-71 and 2-FCV-75-9 as part of the long term Motor-Operated Valve (MOV) program associated with the NRC generic letter 96-05. A ¾" test connection will be installed on the upstream side of 2-FCV-74-71 and on the downstream side of 2-FCV-75-9, consisting of piping, valves, and supports. The 2-FCV-74-71 valve is located on top of the torus, bay 14 area at an elevation of 555', while the 2-FCV-75-9 valve is located just off the 541' elevation, NW quad, at an elevation of 540'. Both of the points in these respective systems, at which these test connections will be added, are within the primary containment pressure boundaries of these systems. This indicates that the modification be performed during a Primary Containment system outage.

The test connections do not perform any active safety function. Design calculations have been performed to ensure that these lines maintain system pressure boundary during all design basis conditions. The test connections do not affect the operation or response of the RHR, Core Spray, or Primary Containment systems with respect to their design functions and no new credible failure modes are created. This activity does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40610A

This safety evaluation for DCN T40610 revises the Unit 2 and 3 Amertap Ball Cleaning System. One portion of the change is to install flow meters in the supply air lines to Pressure Differential Indicating Switches 2 and 3-PDSI-27-89, -91, -93, -95, -97 and 99. The other part of the change is to plug the Control Air shutoff valves which feed the PDIS's and field route a filtered, Service Air line to them through a new shutoff valve.

Neither the Control Air System nor the Service Air System piping components in the Turbine Building are required to mitigate any accident postulated in the Browns Ferry FSAR. The replacement of the Control Air supply to the PDIS's with Service Air will still provide a reliable, functional system which will perform adequately. This will reduce challenges to the Control Air System by ensuring that the heater dump valves have sufficient air supply for normal operation. Therefore, an unreviewed safety question does not exist.

DESIGN CHANGE NOTICE T40617A

This safety evaluation is written in support of DCN T40617, which allows for plant modifications to the Unit 2 and 3 Reactor Water Cleanup (RWCU) System. The design change will replace existing 6" motor operated flow control gate valves FCV-69-1 and 2 with similar 6" gate valves with double disc design.

There are no new failure modes created upon completion of the installation of Unit 2 and 3 FCV 69-01 and -02 valves. The torque switch was removed from the valve closure circuit because it was less reliable in parallel with the limit switch than with the limit switch alone. No new failure modes are created during the valve installation process because of the actions and precautions specified in the DCN. These actions and precautions are intended to minimize any additional risk of system failure (e.g., uncovered irradiated fuel, inadvertent draining of the vessel, localized flooding) during modification.

The changes made to the RWCU systems by this DCN do not add any interfaces with other high systems or adversely affect the operability or integrity of the RWCU system. Therefore, these changes do not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40646A

This safety evaluation for DCN T40646A will install a Debris Filter and associated equipment and its corresponding backwash line in the Condenser Circulating Water (CCW) inlet pipe to the 2C2 Condenser inlet waterbox. The Debris Filter is being installed to eliminate the accumulation of debris in the Condenser 2C2 waterbox. This accumulation reduces the thermal performance of the condenser, and increases maintenance activities by requiring periodic cleaning of the 2C2 waterbox to remove the debris accumulating against the condenser tubesheet during Unit operation. This modification will provide a means of collecting and removing the debris before it enters the condenser waterbox, therefore improving the reliability and operability of the 2C Condenser.

This modification is installing components that will require non-safety related power supplies to operate the Debris Filter, the backwash line flow control valve and the associated instrumentation. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40795A

DCN T40795A is replacing the existing actuator motors for valves 2-FCV-01-55, -56, 2-FCV-059-01, 2-FCV-071-02, 2-FCV-073-02, 2-FCV-074-60, -74 and 75. The DCN is also adjusting the torque switch settings on 2-FCV-071-08 & 39, 2-FCV-073-03, -26 & 36, 2-FCV-074-47, and 2-FCV-75-09. Replacing the motors results in the need to replace thermal overload heaters and cables for selected valves. Also, local splice boxes, Motor-Operated Valve (MOV) housing terminal blocks, and reduced wire size cables sections will be required for some valves. These changes increase the thrust capability of the valves and are necessary to ensure the valves ability to perform their design basis safety functions, as required by MOV Program Generic Letter 89-10. These changes are necessary to reflect the impact of the 105% Power Uprate Program (PUR) on selected Unit 2 GL 89-10 safety-related MOVs, and to incorporate the latest corporate design input parameters from the NRC audit/TVA methodology changes. The engineering requirements established by this DCN ensure that each of the affected MOVs are capable of performing their design basis function.

This evaluation does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40842A

The safety evaluation addresses DCN T40842A which makes TACF 2-97-4-90 a permanent change. The TACF replaces the automatic flow control for 2-RM-90-250 sample lines with manual control. The automatic flow control equipment was obsolete and had become unreliable. The TACF was used to allow time to confirm that the manual control provided adequate control of the sample line flows prior to making it permanent. The TACF has been in effect for over six months and the manual control has been determined to be acceptable, therefore it is being made permanent. The scope of the change includes the addition of manual valves to throttle the sample flow, flow indicating switches for monitoring the flow and providing alarm contacts on low flow, relays and indicating lights to provide indication and annunciation of low flow. The change to 2-RM-90-250 does not alter its alarm on high release rates. Since the equipment will still perform the same function there is no reduction in the margin of safety. This change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T40856A

This safety evaluation for DCN T40856A revises the Mechanical Setpoint for the Main Steam Relief Valves (MSRVs) to the following valves and provides the pressure and flow capacities for revision to the MSRV Code Data Plates to accommodate the Power Uprate Program. The new pressure relief valves are as follows: 1) 4 valves originally @ 1105 PSIG will be set @ 1135, 2) 4 valves originally @ 1115 PSIG will be set @ 1145, and 3) 5 valves originally @ 1125 PSIG will be set @ 1155. The new MSRV pressure setpoints addressed in Calculation ND-Q0999-980003 and the new flow capacities determined in Calculation MD-Q0001-870133 are a follows:

RELIEF VALVE SETPOINTS
4 - CAP @ 1135 PSIG - 905,477 LB/HR
4 - CAP @ 1145 PSIG - 913,356 LB/HR
5 - CAP @ 1155 PSIG - 921,234 LB/HR

Increasing the Main Steam Relief Valves setpoints does not create any new accidents of any type that would represent an unreviewed safety question.

DESIGN CHANGE NOTICE T40918A

This safety evaluation for DCN T40918A will upgrade the existing discrete analog component Reactor Recirculation Control System (RRCS) and jet pump instrumentation with modern digital controls. The modified control system will incorporate the functionality of the existing system, with additional enhancements for redundancy, reliability, and ease of operation . The modified system failure modes are bounded by those of the existing system as presently analyzed in the FSAR. The modification does not increase the probability of any accident, transient, or equipment malfunction previously analyzed in the FSAR, does not create the possibility of a new accident, transient, or equipment malfunction not previously analyzed in the FSAR, and has no impact on any Technical Specification margin of safety. Therefore, this modification does not constitute an unreviewed safety question.

DESIGN CHANGE NOTICE T40975A

DCN T40795A is replacing the existing actuator motors for valves 2-FCV-01-55, -56, 2-FCV-059-01, 2-FCV-071-02, 2-FCV-073-02, 2-FCV-074-60, -74 & 75. The DCN is also adjusting the torque switch settings on 2-FCV-071-08 & 39, 2-FCV-073-03, -26 & 36, 2-FCV-074-47, and 2-FCV-75-09. Replacing the motors results in the need to replacing thermal overload heaters and cables for selected valves. Also, local splice boxes, Motor-operated Valve (MOV) housing terminal blocks, and reduced wire size cables sections will be required for some valves. These changes increase the thrust capability of the valves and are necessary to ensure the valves ability to perform their design basis safety functions, as required by MOV Program Generic Letter 89-10. These changes are necessary to reflect the impact of the 105% Power Uprate Program (PUR) on selected Unit 2 GL 89-10 safety-related MOVs, and to incorporate the latest corporate design input parameters from the NRC audit/TVA methodology changes. The engineering requirements established by this DCN ensure that each of the affected MOVs are capable of performing their design basis function.

This evaluation does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T41153A

This safety evaluation is written in support of DCN T41153A. FSAR Figure 8.7-4b Sheet 2 (which shows the utilization of 120V AC Unit Non-preferred and Plant Preferred circuit breakers) is affected by the change. The change provides automatic redundant AC power for the Unit 2 Off Gas system circuits supplied from panel 2-9-53. These include the Unit 2 Off-Gas Condenser Cooler Inlet and Outlet Valves (2-FCV-66-97, -98, -122, -123). An under voltage relay is installed in panel 2-9-53 to provide automatic transfer to the alternate source if the normal source fails. The existing normal power source is from the 120V AC Unit Non-preferred bus panel 2-9-9 and the new alternate power source is from the Plant Preferred bus panel 2-9-9. A new cable is provided for the alternate source of power. This provides a more reliable power supply for the Off-Gas system valves supplied from panel 9-53.

Failure of the 120V AC power supply will result in loss of the Off-Gas flow and may after a period of time result in a unit trip. The purpose of this change is to improve the reliability of the 120V AC power supply and thus reduce the likelihood of a unit trip. The failure modes for loss of power for the equipment supplied with the new redundant 120V AC power are not affected. The existing Unit Non-preferred normal power source is non-safety related, and the new Plant Preferred alternate power source is also non-safety related. Therefore, the change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T41194A

This safety evaluation has been prepared in support of DCN T41194A. DCN T41194A supports modification of the Unit 2 Main Generator protection circuitry to eliminate an identified single point failure that could cause an unnecessary Unit 2 generator and turbine trip. The circuit for Generator 2 overcurrent relay 251G will be modified to block a trip signal which could be generated improperly by the 251G relay on opening of a supplying potential transformer (PT) fuse. Generator 2 overcurrent relay 251G, a General Electric (GE) Type IJCV Time-Overcurrent relay with voltage restraint, is equipped with a voltage restraining coil connected to one set of PT's supplied from the Generator 2 bus. Loss of potential to the 251G overcurrent relay due to a supplying PT fuse clearing could result in generation of a false trip signal. With the Unit 2 Generator under load, loss of potential to the 251G relay may result in a Unit 2 generator and turbine trip. The existing circuit as configured will annunciate on opening of a PT fuse associated with relay 251G. In order to prevent the possibility of false tripping due to loss of potential to the 251G overcurrent relay, a spare normally closed "blocking" contact of Generator 2 "Loss of Potential" relay 260 will be connected in the trip circuit of the 251G overcurrent relay to block a possible trip initiation by 251G when a supplying PT fuse clears.

The proposed change does not result in any physical changes which could adversely affect the function or operation of any plant equipment. Therefore, there would not be any reduction in the margin of safety as defined in the basis of any Technical Specification. This does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE T41313A

This safety evaluation for DCN T41313A provides for replacement of the sudden pressure relays on the Unit 2 Main Bank (Phase A, B and C) and Unit 2 Station Services Transformers 2A and 2B. In addition, a second sudden pressure relay will be added in series with the replacement sudden pressure relay and will be connected in a two-out-of-two logic trip scheme. The new (and replacement) relays will be mounted to a pedestal which will be installed in the ground near the transformer. The relays will be piped to the transformers with high pressure flexible hose. This arrangement will reduce the potential for spurious operation due to transformer vibration, as will also the two-out-of-two trip logic. Both relays must operate in order to produce the trip which will improve Unit 2 reliability by minimizing spurious trips which result in a loss of one of the offsite power sources and a generator trip. There will not be any adverse impact on the safe shutdown capability of the plant and thus no adverse impact on nuclear safety. This modification has been reviewed against the criteria of 10CRF50.59 and does not constitute an unreviewed safety question.

DESIGN CHANGE NOTICE T41381A

This safety evaluation addresses DCN T41381. The change is to swap the leads of the cable that provides the temperature signal from the RHR Pump motor thermocouples to a recorder from one thermocouple to a similar thermocouple leads in the same junction box. The potential impact of this change is the loss of temperature monitoring for the RHR Pump motor windings. The temperature monitoring is equipment protection only and therefore the change will not negatively impact any equipment important to safety. Because equipment important to safety will not be negatively affected the probability of malfunction is not increased. Therefore, this change does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE W18208A

This safety evaluation for F-DCN 40706A addressed the installation of the Hydrogen Water Chemistry (HWC) central control panel, modules and piping design which is covered by DCN W18208A. The entire HWC system will be implemented through five (5) DCNs:

W18207A - HWC Outage Modifications
W18208A - HWC Control Panels, Cable Routing, Supply Header to 100 year Flood Ditch
W18209A - HWC Offgas Modification
W19201A - HWC Liquid H2/02 Storage Facility.
W40707A - HWC Supply Header into plant.

The HWC system is not nuclear safety related. Equipment and components are not redundant (except when required by prudent engineering practice), or environmentally qualified. The equipment is designed as seismic Category II, non-Class 1E. All piping and panels located within the reactor building are seismic II/I so as not to impact operation of other safety related components in that area.

The HWC system implementation and operation does not constitute an unreviewed safety question.

DESIGN CHANGE NOTICE W39882A

This safety evaluation is written in support of DCN W39882A that authorizes changes which will enable the scram trip function of the Oscillation Power Range Monitor (OPRM) on BFN Unit 2. In addition, DCN W39882A is issuing all Neutron Monitoring System (NMS) Nuclear Engineering Setpoint and Scaling Documents (NESSDs) required to support power uprate, 24-month cycle operation, and Hydrogen Water Chemistry (HWC) implementation on BFN Unit 2.

The OPRM is a subsystem of the Power Range Neutron Monitor (PRNM) upgrade which was installed on Unit 2 in the Fall 1997 refueling outage. At the time of installation, and consistent with NRC commitments, the OPRM scram trip function was not enabled. One full cycle of operation with the OPRM scram trip function disabled was allowed so that equipment performance could be monitored without the risk of spurious scrams from OPRM trips.

The PRNM upgrade uses General Electric (GE) Nuclear Measurement Analysis and Control (NUMAC) components to provide the Average Power Range Monitor (APRM), Rod Block Monitor (RBM) and OPRM functions. The OPRM functions are performed by software algorithms and electronic components contained in the NUMAC chassis which also perform the APRM functions. The OPRM trip functions implement the stability long-term solution designated as BWR Owner's Group (BWROG) Option III.

The stability long-term solution consists of software algorithms and electronic hardware which provide for reliable, automatic detection and suppression of stability related power oscillations. The Option III solution automatically initiates control rod insertion (scram) to terminate the power oscillation while it is still small. The combination of hardware, software, and system setpoints provide protection against violation of the Minimum Critical Power Ratio (MCPR) safety limit for anticipated oscillations. The Option III solution includes three separate algorithms for detecting stability related oscillations: the Period Based Algorithm (PBA), the Growth Rate Detection Algorithm (GRDA) and the Amplitude Based Algorithm (ABA). All three algorithms perform calculations on OPRM cell signals to determine if a trip is required, and each algorithm is capable of initiating a trip. However, only the PBA is credited in licensing evaluations with providing the required protection. The GRDA and the ABA are classified as defense-in-depth functions which provide additional protection against unanticipated oscillations.

This activity does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, this does not involve an unreviewed safety question.

DESIGN CHANGE NOTICE W40116A

This safety evaluation for DCN W40116A installs a new Auxiliary Decay Heat Removal (ADHR) system into Units 2 and 3, capable of removing decay heat from the reactor core and the spent fuel pool. The ADHR system is designed primarily for use during Refuel Mode but can be used for cooling of the Spent Fuel Pool (SFP) in other reactor operating conditions. During decay heat removal using the ADHR system the cooling function of one train of the Fuel Pool Cooling and Cleanup System (FPCCS) is required to supplement ADHR for the first 116 hours after reactor shut down.

The ADHR system, with natural circulation between the fuel pool and the vessel cavity, is an alternate to RHR shutdown cooling. The RHR shutdown cooling mode is not a safety-related function, and the ADHR system is classified as non safety-related. The ADHR system does not provide an alternate for the clean-up function of the FPCCS, and operation of the FPCCS to maintain water clarity will normally be required. This change does not constitute an unreviewed safety question.

DESIGN CHANGE NOTICE W41190A

DCN W41190A replaces an existing old instrument with latest instrument from the same manufacturer on Reactor Water Cleanup (RWCU) panel 2-LPNL-25-9 and installs a new dissolved oxygen sensor in the panel hood area. A new oxygen analyzer monitor is mounted on the panel. In addition to modifying RWCU panel, the DCN replaces existing old instruments with the latest instruments from the same manufacturer on condensate feedwater panel 2-LPNL-25-103. The modification installs new dissolved oxygen and hydrogen sensor in the panel hood area. A dual channel oxygen/hydrogen analyzer monitor is installed on the panel. A provision for continuous nitrogen purge facility is provided for the hydrogen sensor located in the condensate feedwater panel hood. The dissolved oxygen and hydrogen instruments are non safety and non-essential instruments. The power to these instrument is supplied from the spare utility power sockets located in the panels. The modification does not require change to the Technical Specification or Technical Requirement Manual. Therefore, no unreviewed safety question is involved.

JANUARY 1 - MAY 31, 1999

SUMMARY OF SAFETY EVALUATIONS FOR PROCEDURE REVISIONS

EOIPM AND EOI/SAMG FLOWCHARTS

Revision 1 of this safety evaluation addresses the changes resulting from the U2C10 outage in April 1999 and the changes resulting from the revised station blackout (SBO) analysis and other enhancements based on operator input. Additionally, enhancements have been made based on a recent Quality Assurance audit. This safety evaluation addresses the use of the Boiling Water Reactor Owners' Group (BWROG) Emergency Procedure and Severe Accident Guidelines (EPG/SAG) and its Appendices Revision 1 to revise the BFN Emergency Operating Instructions Program Manual (EOIPM). Specifically the update of the Emergency Operating Instruction (EOI) flowcharts, calculations, support procedures, and the implementation of the new SAMG flowcharts. This safety evaluation also addresses the draft Plant Specific Technical Guideline/Severe Accident Management Guideline (PSTG/SAMG) which is part of the EOIPM.

The current revisions of the EOIPM and EOI flowcharts are based upon the BWROG Emergency Procedure Guidelines (EPGs) Revision 4. This was reviewed and approved by the NRC as documented by its issuance of the Safety Evaluation Report (SER). In addition, the NRC has reviewed and approved certain modifications to the EPG Revision 4 that addressed Anticipated Transient Without Scram (ATWS)/Instability events. An SER was issued for these changes. EPG/SAG Revision 0 and Revision 1 were sent to NRC for information only. The staff was not requested to review and approve these revisions or to issue an SER since the BWROG executives did not want the EPG/SAG to receive NRC review and approval. Therefore, the actions in EPG/SAG Revision 1 (and hence, the EOIPM and EOI and SAMG flowcharts and the draft PSTG/SAMG) will be reviewed and approved under the 50.59 process. Therefore, the actions in the EPG/SAG (and BWROG draft PSTG/SAMG and EOI and SAMG flowcharts) will be compared to the BFN design basis events, to EPG Revision 4 and the ATWS Instability modifications as reviewed and approved by the NRC (and documented within those SERs) to ensure that the changes have not altered the basis on which the NRC reviewed and approved the actions prescribed in the EPGs.

Based upon the review, the procedure changes do not constitute an unreviewed safety question (USQ). Where a USQ might have been involved, the procedures have been changed to address the SAR assumptions.

SURVEILLANCE INSTRUCTION 3-SI-4.6.H-2B

This safety evaluation supports a revision to Surveillance Instruction (SI) 3-SI-4.6.H-2B regarding the testing of snubbers. The revision allows Site Engineering to evaluate and accept snubber test results, on a case specific basis, that are outside the standard acceptance criteria. This in effect establishes case specific acceptance criteria. The revision also revises the acceptance criteria for three snubbers to specify only those requirements which are applicable to these snubbers. The relaxed test requirements will still ensure that the pipe stresses will be within the code allowables. The systems will still be capable of performing their design and licensing functions for seismic events, accidents and transients. There is no increase in the likelihood of pipe breaks and thus no impact on the environmental qualification of equipment or any effect on the reactor coolant or primary containment pressure boundaries. This procedure revision has been reviewed against the criteria of 10CRF50.59 and does not constitute an unreviewed safety question.

JANUARY 1 - MAY 31, 1999

SUMMARY OF SAFETY EVALUATIONS FOR TEMPORARY ALTERATIONS

TACF 2-98-8-47-0

This safety evaluation is in support of the TACF 2-98-8-47-0 which defeats the Stop Valve Load Limit (SVLL) logic from the Electro-Hydraulic Controls (EHC) control logic on Unit 2. In conjunction with the defeating of the logic, the Turbine Bypass Out of Service (TBOOS) critical power ratio corrections provided in the Core Operating Limits Report (COLR) shall be applied as part of the TACF.

The SVLL position switches are describe in the FSAR to initiate Turbine Bypass Valve (TBV) opening in the event of a turbine trip. This SVLL logic is being removed by the TACF because intermittent failure of the limit switches has caused inadvertent TBV opening during the Turbine Stop Valves (TSV) test in accordance with the Operating Instruction (OI) for system 47 (OI-47). The additional position switches on the TSVs that initiate a reactor scram and recirculation pump trip are not affected by the TACF. Therefore, this TACF does not affect any margin of safety as defined in the TS or its bases. This change does not involve an unreviewed safety question.

JANUARY 1 - MAY 31, 1999

SUMMARY OF SAFETY EVALUATIONS FOR UPDATED FINAL SAFETY ANALYSIS REPORT REVISIONS

FSAR CHAPTER 8

This safety evaluation revision is issued to support the corrective action for Problem Evaluation Report (PER) 98-009978-000. The previous revisions of the safety evaluation addressed changes to Chapter 8 of the FSAR identified by review in accordance with 0-TI-353. The PER in part identified an inconsistency in the ratings shown for the Diesel Generators (DG) on drawings 0-15E500-1 (FSAR Figure 8.4-1b, Standby Auxiliary Power System Key Diagram) and the DG rating shown on 0-45E724-3 (FSAR Figure 8.5-4d, 4160-V Shutdown Board C - Single Line). For consistency, the referenced drawings/ FSAR Figures will be revised to show the Diesel Generator's continuous rating as shown in FSAR Table 8.5-6 (Diesel Generator Ratings) and Section 8.5.3.2 (Diesel Generators). Additionally, the following drawings/FSAR figures similarly require revision for the same reason identified in the referenced PER: 3-15E500-3 (FSAR Figure 8.4-2, Normal and Standby Auxiliary Power System Key Diagram), 0-45E724-1, -2 & -4 (FSAR Figures 8.5-4a, -4c & -4e, 4160-V Shutdown Board A, B & D - Single Lines), 3-45E724-6, -7, -8 & -9 (FSAR Figures 8.5-4b, -4f, -4g & -4h, 4160-V Shutdown Board 3EA, 3EB, 3EC & 3ED - Single Lines). The inconsistency between the drawings/FSAR Figures identified by the referenced PER was not recognized by the earlier assessment/enhancement of FSAR Chapter 8.

This change does not reduce the margin of safety as defined in the Technical Specification basis. This change does not involve an unreviewed safety question.

UFSAR SECTION 14.5.8 AND 14.5.10

This safety evaluation for USFAR 14.5.8 and 14.10.8 discusses the Loss of Habitability of the Control Room that are not in agreement with discussion in Section 7.8 (Backup Control System). Section 7.18 is reflected in current plant operating procedures 2-AOI-100-2 and 3-AOI-100-2 for Control Room Abandonment. Since Sections 7.18, 14.5.8 and 14.10.8 are dealing with the same subject matter the following changes are being proposed to eliminate redundancy:

1. Add a sentence in Section 14.5.8 and 14.10.8 to refer to Section 7.18 for details on plant operation for outside the control room.
2. Delete Sections 14.5.8.1, 14.5.8.2, 14.5.8.3, 14.5.8.4, 14.10.8.1, 14.10.8.2, 14.10.8.3, and 14.10.8.4.

This change does not affect any margin of safety in the Technical Specification or its bases. This change does not involve an unreviewed safety question.

FSAR 6.4, TS BASES 3.5.1 AND 3.5.3

This safety evaluation for FSAR 6.4, TS Bases 3.5.1 and 3.5.3 supports a revision to the UFSAR and the Units 1, 2 and 3 Technical Specification Bases to clarify the expected sequence for transfer of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) suction source from the condensate storage tank to the suppression pool. This revision does not represent any changes to the capability or timing of the system in response to a HPCI or RCIC system initiation. The system transfer is based upon existing level switches which are not altered by this documentation revision. The changes to these documents are for clarifying or expanding upon existing information and do not represent a change to the system capabilities, setpoints requirements. The equipment will continue to contain redundant, highly reliable instrumentation for both low condensate header level and high suppression pool level. The HPCI and RCIC suction valves will continue to operate in the same manner as before. This modification has been reviewed against the criteria of 10CRF50.59 and does not constitute an unreviewed safety question.

FSAR SECTIONS 11.3, 11.7, 11.8 AND 11.9

This safety evaluation for FSAR sections 11.3,11.7,11.8 and 11.9 which do not involve any physical modifications or changes to the function or operation of the system. Most of the changes are minor in nature and provide clarification of incorrect, unclear or misleading statements. Any significant changes are backed up by existing analyses (i.e., calculations or design criteria) or other design basis documents. The changes are within the current design bases and physical limitations of the system. Therefore, the changes do not create the possibility for a malfunction of a different type than any evaluated previously in the SAR. This change does not involve an unreviewed safety question.

FSAR CHAPTER 2

This safety evaluation revises FSAR Chapter 2.2 to indicate Chapter 2.2 text except for Section 2.2.3, (Land Use), Table 2.2-10 (Hazardous River Traffic), and Figure 2.2-4 (Plant Systems General Plan) is considered historical. Hence, these historical sections will not be periodically updated. This includes all Chapter 2.2 population distribution Tables, employment Tables, statistical Tables, and General Site Location Figures 2-2-1 and 2.2-2.

Offsite dose calculations from reactor accidents are based on defined distances from the plant as described in the FSAR. These are the Exclusion Area Boundary (1400 meters) and the Low Population Zone (3200 meters). Hence, the population tables in the FSAR do not affect these analyses or margins of safety defined in the Technical Specifications. Therefore, this change does not involve an unreviewed safety question.

FSAR CHAPTERS 7.8 AND 12.2

This safety evaluation is prepared in support of changes to BFN Safety Analysis Report, sections 7.8, "Reactor Vessel Instrumentation and 12.2, "Principal Structures and Foundations". These changes are listed as follows by FSAR sections numbers:

FSAR section 7.8.5.2 Reactor Vessel Water Level

Revised the 2nd paragraph to remove the number of Main Control Room Indicators and only reference the types of indication displayed. The vessel levels are displayed on various control panels using different meter types (analog/digital). Only the types of redundant vessel level indications need to be listed to clearly define the level indication requirements. The present text lists the number of level indicators as 14; however, DCNs W26520A and W25841A added level indicators 2-LI-3-253 and 3-LI-3-253 to Main Control Room panels 2-9-5 and 3-9-5 which increased the number of level indicators to 15. The addition of these indicators is covered by the 50.59 prepared for these DCNs.

This change removes the reference to setpoints being discussed in subsections of the FSAR which address various level measuring components in the 3rd paragraph. Only a reference to the appropriate FSAR section is necessary instead of including a mention of the setpoints since these sections will contain the design description for that system.

FSAR Section 7.8.5.4 Reactor Vessel Internal Pressure

Corrected the number of Control Room Indicators which are driven by three pressure transmitters used in the Feedwater Control System (from 3 to 4). SAR Change Package 17-056 corrected the number of Feedwater Control Pressure transmitters (from 2 to 3). ECN E-2-P7152 and DCN W17133 Stage 2 added pressure indicators 2-PI-3-253 and 3-PI-3-

253 to Main Control Panels 2-9-5 and 3-2-5. This increased the number of indicators from 3 to 4 for the Feedwater Control System transmitters. The addition of these indicators is covered by the 50.59 prepared for these DCNs.

FSAR Section 12.2.2.7.3 Drywell Temperature Effects

Revised the 6 th and 8 th paragraphs to reference 10 CFR 50.49 Program as the program controlling environmental specifications for electrical penetrations and safety components located inside the drywell. These temperatures and pressure parameters are enveloped by the present Environmental Qualification (EQ) requirements as revised by DCNs S40792 and S40793 (Refs. 2.20 and 2.21). The texts are revised to reflect the EQ program which maintains the appropriate EQ for devices which are used in the drywell for safety applications.

The documentation change does not change the design basis, nor does it affect the components' design function of the systems operability. Therefore this change does not introduce any new credible failure modes and does not introduce an unreviewed safety question.

FSAR SECTION 10.17, PROCESSING SAMPLING SYSTEMS

This safety evaluation is being written to address the revision of FSAR Section 10.17, Process Sampling Systems. This revision updates and clarifies information needed to give a more accurate description of the process sampling systems and removes details that are not useful or are not necessary in describing the process sampling systems. This revision does not adversely affect the ability of the process sampling systems to perform its primary containment isolation function.

The FSAR revision does not physically modify any plant structure, system or component or change any plant approved procedure. However, this revision does change details given in the FSAR on how the plant was analyzed to work and information relied upon by NRC to understand how the plant works. Therefore, the effects of these changes to the FSAR are evaluated as if the changes are actually being made to the plant.

This change does not involve an unreviewed safety question.

UFSAR CHAPTER 7.13

This safety evaluation addresses the following revisions to UFSAR Section 7.13:

Paragraph 7.13.2.3 - Paragraph has been clarified by adding the description of UFSAR Section 7.12.5 already referenced and by noting that there is a difference between the Area Radiation Monitoring System and the Reactor Building Ventilation Radiation Monitoring System.

Paragraph 7.13.3.1 -The word *these* has been changed to reflect the actual title of the trips for clarity.

Paragraph 7.13.4 - The detailed description of the check source used to calibrate the area radiation monitors has been deleted. This check source is a calibration standard and not a system, structure or component per se. This change eliminates unnecessary detail without removing the function of the calibration process.

Paragraph 7.13.5.3.2 - The subject paragraph is revised to correctly identify the ratemeters as count ratemeters (friskers). The references to specific areas of the plant already covered in the statement *plant area* have been deleted. Radiation has been identified as radioactive contamination level because that is the intent of the ratemeters.

Paragraph 7.13.5.4 - Information has been added to paragraph concerning the functional testing of the high radiation area door alarms.

Table 7.13-2 -The column designation of RE-90-3 on all three units has been corrected. The designation on all three units is changed from q-line to p-line per walkdown by System Engineer.

The description of RE-90-35 is changed from *Service Water Booster Pump Area* which does not exist to *Laundry Drain Tank* which matches the field nameplate.

No setpoints or system operating parameters are changed by the UFSAR text and table changes. The changes do not result in any equipment change or alter the operating characteristics of the plant. This change does not involve an unreviewed safety question.

UFSAR CHAPTER 4.3.4

This safety evaluation evaluates changes to UFSAR Section 4.3.4 concerning the description of the design life for the recirculation pump seals. The current wording of the UFSAR indicates that the minimum design life of the recirculation pumps seals is 18 months. During the review for possible UFSAR changes associated with the license amendment for 24 month cycles (Amendments 235, 255, and 215) the use of the term "18 months" in this description was identified as inconsistent with the 24 month operating cycle.

The change evaluated by this safety evaluation revises the current wording of the UFSAR to remove the reference to "18 months". The currently installed recirculation pump seals on Units 2 and 3 are an improved design that has an increased design life from the originally installed seals. Currently, the Unit 2 recirculation pump seals are on their second cycle of operation and the Unit 3 seals are on their third cycle of operation. This change does not change the design or operation of the recirculation pump seals. The change does not involve an unreviewed safety question.

UFSAR CHAPTERS 3, 4, 7, AND 10

The evaluations performed in this safety evaluation are keyed to the following item descriptions. For example, justifications associated with Change Group No.1 Subitem (a) would deal with the 105 versus 100 percent flow control range wording change. The changes evaluated by this safety evaluation (SE) involve UFSAR text clarifications, corrections, deletions, and additions as noted below. The changes have no design basis accidents or credible failure modes associated with them that have not already been evaluated by the SAR.

Change Group No. 1
This group combines the changes to UFSAR Section 4.3 dealing with Reactor Recirculation System (RRS). The following changes have been proposed:

(a) The Reactor Recirculation System (RRS) designated flow control range is being changed to read *105* percent versus *100* percent. [UFSAR Pages 4.3-1]

(b) The RRS pump discharge and suction valve closure discussion has been modified by replacing the words *The only* with the word *A*. [UFSAR Page 4.3-2]

(c) The abbreviation cps is being replaced with the abbreviation *Hz.* [UFSAR Page 4.3-4 and Table 4.3-1]

Change Group No. 2

This grouping correlates the changes to UFSAR Section 7.7 dealing with the Reactor Manual Control System (RMCS). The following UFSAR text revisions are proposed:

(a) Section 7.7.2.2 has been revised to reference accident analysis information for resolution of Operations comments. Specifically, the velocity limiter and RPS 120% flux trip are now referenced for information. [UFSAR Page 7.7-1]

(b) Section 7.7.2.3 has been revised to reference accident analysis information for resolution of Operations comments. Specifically, references to the definition of the word erroneous have been added. [UFSAR Page 7.7-1]

(c) The reversion to the NO ROD SELECTED condition statement at the end of Section 7.7.4.2.1 has been revised to note that rod movement is possible under certain conditions. This revision ensures that the UFSAR correctly reflects the EMERG ROD IN function when using the CRD NOTCH OVERRIDE switch and has been analyzed accordingly. [UFSAR Page 7.7-3]

(d) The RONOR switch description is being revised to agree with the plant labeling of the switch. Specifically, the RONOR switch is labeled as CRD NOTCH OVERRIDE. [UFSAR Page 7.7-4]

(e) The words *and two solenoid-operated stabilizing valves* are being replaced with the words *and four solenoid-operated stabilizing valves in parallel with only two operating*. Also, UFSAR Section 3.4 text changes have been implemented to add the backup stabilizing valve set description. [UFSAR Pages 3.4-16, 3.4-29, and 7.7-4]

(f) The SRM rod block descriptions have been clarified to note under which conditions the rod blocks are bypassed by IRM range positions. [UFSAR Page 7.7-8]

(g) Text has been added to indicate that a licensed operator will be present whenever control rods movement is initiated with the RWM function bypassed for maintenance purposes. [UFSAR Page 7.7-10]

(h) The words *into one of the two* are being replaced with *to both* in order to correct RMCS input descriptions regarding scram discharge volume high level trip signals being provided to both RMCS rod block logic trains. [UFSAR Page 7.7-10]

(i) The words *alarms and indications* are being added after the *words The following control room lights*. The lighting label *WITHDRAWAL NOT PERMISSIVE* is being revised to read *WITHDRAW PERMISSIVE*. The words *and indication* are being added after the words *high pressure (alarm only)*. [UFSAR Page 7.7-13]

(j) The U2/U3 RMCS rod block logic diagrams (Figure 7.7-6a and 7.7-6b) are being revised to indicate under what conditions the IRMs can bypass the rod block function. Also, Figure 7.7-6a is being revised to show the correct unit designation in the figure title block.

<u>Change Group No. 3</u>
This change group deals with UFSAR Section 7.9 text changes dealing with the Recirculation Flow Control System (RFCS). The following UFSAR text revisions are proposed:

(a) The second sentence in the first paragraph of Section 7.9.4.1 has been changed to read as follows: [UFSAR Page 7.9-1]

The change in flow then affects changes in reactor power level.

(b) The following sentence is being added to the end of the third paragraph in Section 7.9.4.1 to emphasize the U2/U3 difference: [UFSAR Page 7.9-1]

Unit 3 has a different recirculation pump discharge piping arrangement which does not have equalizing valves.

(c) The last sentence in Section 7.9.4.2.1 is being clarified to discuss what constitutes a different RRS bus arrangement for each unit. The normal and alternate power supplies are discussed in more detail. [UFSAR Page 7.9-2]

(d) The word *are* is being changed to read *include*. [UFSAR Page 7.9-2]

(e) The word *automatic* is being replaced with the words *dual pump manual* in order to clarify the UFSAR text. [UFSAR Page 7.9-3]

(f) The word *speed* is being replaced with the word *flow* in order to ensure the technical accuracy of the UFSAR. [UFSAR Page 7.9-4]

(g) A new Item 11 (DRIVE MOTOR WINDING TEMP NOT HIGH) interlock description is being added to ensure the technical accuracy of the UFSAR text. [UFSAR Page 7.9-5]

(h) The administratively-controlled unit operator manual trip has been removed from the RRS MG set drive motor trip logic discussions. [UFSAR Page 7.9-7]

(i) The wording *Start of control valve closure* has been revised to read *Low Emergency Trip System (ETS) Fluid Pressure*. [UFSAR Page 7.9-9]

(j) The RFCS testing description contains an error in that annunciators instead of indicating lights are furnished in the control room to alert operators to the presence of a RFCS bypass condition. [UFSAR Page 7.9-10]

(k) The phrase *Any failing component* is being replaced with the phrase *Most failing components*. [UFSAR Page 7.9-11]

(l) The separate descriptions of the MG sets' controls have been condensed into the one word *controls* rather than listing each subcomponent separately. [UFSAR Page 7.9-12]

(m) The phrase *All the* has been deleted. [UFSAR Page 7.9-12]

<u>Change Group No. 4</u>

The last sentence in the second paragraph of Section 7.13.3.1 is being deleted.
[UFSAR Page 7.13-1]

<u>Change Group No. 5</u>

The UFSAR Section 10.19 lighting system description is being revised to reflect the plant configuration for UFSAR evaluation purposes. [UFSAR Page 10.19-1]

The proposed text and figure changes to UFSAR Chapters 3, 4, 7, and 10 have been reviewed in accordance with 10CFR50.59 requirements. None of the proposed changes constitute an unreviewed safety question.

FSAR SECTION 8.6

This safety evaluation addresses changes to FSAR Section 8.6 which is being revised to clarify the description of inspection and testing for the 250V DC system. This revision clarifies that a battery may be removed from service without loss of capability to supply DC loads required for shutdown and cooldown of all three units in the event of the loss of offsite power and a design basis accident in any one unit do not constitute an unreviewed safety question. These changes clarify the description to indicate compliance with the safety design bases of the 250V DC power distribution system. For this configuration, the capability to supply all required Essential Safeguard System (ESS) loads is maintained.

UFSAR SECTION 2.3.7.1

This safety evaluation for UFSAR Section 2.3.7.1 changes the error description of the aspirated radiation shield for temperature in UFSAR section 2.3.7.1 to correspond to the equipment currently on the measurement tower. The corrected error specification results in a higher level of uncertainty than previously stated in the UFSAR. The total system error calculated for direct measurements of air temperature has increased from +0.36 degree F to +0.54 degree F due to the corrected aspirated temperature shield error. However, it is still below the R.G. 1.23 requirement of +0.9 degree F.

Correcting the stated error for the aspirated temperature shield does not alter the data collected, does not reduce the margin of safety defined in any bases, and does not cause any requirements to be exceeded. Therefore, this FSAR change does not constitute an unreviewed safety question.

FSAR SECTIONS 7.11 AND 11.2

This safety evaluation for FSAR Sections 7.11 and 11.2 describe in detail proposed changes to SAR wording in an effort to enhance compatibility between plant systems and the SAR wording. Most of the changes identified were clearly of an editorial or clarification nature. Prior to categorizing any of the changes as editorial or as a clarification, each change was examined from a perspective of whether it could possibly be construed as a change to the plant. After passing this scrutiny those changes were categorized as an editorial or as clarifications. These change are minor and are necessary only to enhance the consistency of the SAR wording. The two changes that did not pass the scrutiny of possibly being changes to the plant were the increase in pressure channel separation from 2 psi to 3 psi and the changing of the speed channels from independent to redundant. These changes have been thoroughly reviewed and have proved to have no impact on Electro-Hydraulic Controls (EHC) system performance, any design basis accidents, or any plant transients previously analyzed. It is therefore concluded that the proposed changes do not represent an unreviewed safety question.

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SUMMARY OF OTHER SAFETY EVALUATIONS

TOE 2-99-073-5576 AND WO 99-005575-000

This safety evaluation addresses Technical Operability Evaluation (TOE) 2-99-073-5576 and Work Order (WO) 99-005575-000. This activity will allow the High Pressure Coolant Injection (HPCI) system to be considered operable with the steam supply line drain pot level switch malfunctioning. Since the steam trap bypass valve will be administratively controlled in the open position, the steam trap bypass valve will be able to fulfill its function of providing a path to drain condensate from the steam line to the main condenser, thus helping to prevent water hammer when the HPCI system is initiated. This will allow continued operability of the HPCI system, and will allow troubleshooting to determine the cause of the level switch failure. This activity does not constitute an unreviewed safety question.

PLANT OFFICE BUILDING (POB) RENOVATION

This safety evaluation addresses the renovation of the Plant Office Building (POB) reconfigured office areas and added office features which enhanced working environments. The POB is classified as Class II non seismic structure designed and constructed under industry codes and standards. The POB does not affect accident conditions, create new types of accidents, or does it affect or interface with equipment important to safety or shutdown of the reactor units. Therefore, renovation of office interiors, reconfiguring of office spaces , and the addition of an elevator do not create the potential for accidents or consequences of an accident or challenge the functionality of equipment important to safety or shutdown of the reactor units. This building is an exclusion area as defined in the TVAN Business Practice BP-239 dated 1/22/99. Therefore, the renovation of the POB does not constitute an unreviewed safety question.

TEMPORARY STRUCTURE CONTROL FORM 2-99-002 RB

This safety evaluation for TSCF 2-99-002-RB has been prepared to implement Work Order 99-003692-000, which request a temporary enclosure for RadCon Control Point use during the Unit 2 Cycle 10 outage. This enclosure is required to be climatically controlled, because it will house the computers for RadCon control in the drywell during the outage. The temporary enclosure requested is to be a Tube-Loc (Tube and Coupler) scaffold frame, with Herculite attached to the frame structure (11'-0" x 15'-0" x 8'-0" high). There will also be 2-4'x 8' pieces of plexiglass located on the North and East walls, approximately 2' above the floor and located as shown on the sketches. There will be SONEX acoustical panels wired to the outside of the Herculite for sound proofing needed for communications in the drywell during the outage. This temporary structure is to be located on elevation 565 of the Unit 2 Reactor Building between R8 and R9 at U-line.

This temporary enclosure is not close to any Technical Specification equipment and cannot move to the vicinity of any Technical Specification equipment and therefore, it will not reduce the margin of safety as defined in the Technical Specifications. This change does not involve an unreviewed safety question.

TECHNICAL SPECIFICATION SECTION B3.3.3.1

This evaluation for Technical Specification (TS) Section B3.3.3.1 purposes a change to add a clarification to the Technical Specification Bases documents for units 2 and 3 for BFN PER 98-008829-000. Section B3.3.3.1 for Post Accident Monitoring (PAM) Instrumentation, item 5 specifies the control room recorders RR-90-272 and RR-90-273 as

PAM instruments for drywell radiation monitors. However, these recorders also have the Torus radiation as inputs, which are not PAM devices. This proposed change would clarify the TS bases by adding the UNID's for the "Drywell High Range radiation monitors", which are RM-90-272A, RM-90-273A (unit 3) and RM-90-272A, RM-90-273A (unit 2). Also this proposed change will not be implemented for unit 2 until completion of DCN 41156A, which will replace the RM-90-272C/273C and identifies them the same UNID's as unit 3 (RM-90-272A/273A). The proposed change to the TS bases does not alter the drywell radiation monitors or its equipment setting, operating range or setpoint(s), therefore the margin of safety is not affected by this proposed change. This change does not involve an unreviewed safety question.

MISCELLANEOUS BASES CLARIFICATIONS

This safety evaluation for Miscellaneous Bases Changes are: Base Change 1 ensures the details of calibration methods of the primary containment radiation monitors are maintained in the Improved Technical Specifications (ITS) Bases (to carry forth a commitment to NRC). Change 2,3 and 4 are purely editorial. Change 5 adds "design basis" to the Bases text to distinguish in the Bases that the valve times presented in the ITS 3.6.1.3 Bases are the design basis timing requirements. Change 6 in the B 3.6.2.5 RHR Drywell Spray, SR 3.6.2.5.2 Bases revises the existing Bases text which currently indicates that either air or water can be used for the 5-year drywell spray nozzle surveillance. In practice, air is always used since the use of water would involve the potential to wet down drywell equipment and cause equipment damage. Hence, this Change 6 corrects the Bases text to reflect actual practice by eliminating the option of using water. These Bases revisions do not directly affect operability of the equipment nor plant operation. This change does not involve an unreviewed safety question.

TRM SECTION 3.6.1

This safety evaluation for Technical Requirements Manual Section 3.6.1 will change the Primary Containment Purge System APPLICABILITY from "at all times" to "Anytime system being used for purging Primary Containment". The purge system is not a safety related system and is not used in the prevention or mitigation of any design basis accidents. The proposed activity will not reduce the margin of safety as defined in the basis of any Technical Specification. This change does not involve an unreviewed safety question.

TRM SECTION 3.7.4 AND BASES

The proposed revision for TRM Section 3.7.4 and Bases will allow online functional snubber surveillance testing and it will ensure that the required systems will remain operable so they may perform their safety function when required. This proposed revision also adds additional clarification to the TRM and Bases so they will be better understood by all personnel using them. It also does not affect or change any analysis of the Design Basis Accidents or Abnormal Operational Transients discussed in the FSAR.

This revision is due in part to the realization that some safety systems, such as Residual Heal Removal (RHR), Core Spray (CS) and Emergency Core Cooling Systems (ECCS) are required to be operable during Modes 4 and 5. For systems such as these, the impact of removal of snubbers for testing during any mode of plant operation is no less than in any other mode. Therefore, the prior requirement that snubbers on these systems be tested during Modes 4 and 5 offered no relative improvement or offset in the potential impact on overall plant safety. Therefore this revision does not involve any unreviewed safety question.

TECHNICAL SPECIFICATION B 3.10.4 AND B 3.10.5

This evaluation for Technical Specification Limiting Condition for Operation (LCO) 3.10.4 and 3.10.5 clarifies that a control rod withdrawal block may be initiated by either the reactor manual control system logic or by disarming the control rods. Disarming of control rods consists of disconnecting the individual directional control valves or isolation of the associated hydraulic control unit such that no reactor manual control system signal of drive water pressure can be applied to the control rods. A clearance is used to control the configuration of the rods in this condition. This provides an equivalent means of blocking control rod withdrawal such that no more than one control rod can be withdrawn with the reactor mode switch in Refuel in order to ensure compliance with the reactivity safety design basis as described in the SAR. This change does not affect the analysis described in FSAR section 14.5.3.3, control rod removal error during refueling. This change to the TS bases provides for an equivalent way of blocking control rod withdrawal and ensuring that only one control rod can be withdrawn under the conditions described in the Limiting Condition for Operation (LCO). This change has been evaluated against the criteria of 10CFR50.59 and found to not involve any unreviewed safety question.

TECHNICAL SPECIFICATION UNIT 2 AND 3 BASES B.3.3.3.2

This safety evaluation involves a revision to Technical Specification (TS) Unit 2 and 3 Bases B.3.3.2. This revision will allow the backup controls for a component to be considered operable even if the end device is not operable. Furthermore, this revision will remove the backup controls for the 4 kV fire pumps and for the Residual Heat Removal (RHR) system 1-2 crosstie valve from Table B3.3.3.2-1, and will increase the requirements for backup controls for Residual Heal Removal System (RHRS) pumps and header pressure indicators from 1 to 2.

This revision to the TS Bases will not make any physical changes to the plant equipment, and will not require any changes to procedures for shutting down the plant from outside the control room. Equipment important to safety will still be subject to operability requirements of the appropriate section of the TS, the Technical Requirements Manual , or the Fire Protection Report. Therefore, this activity does not constitute an unreviewed safety question.

TRM CHANGES UNIT 2

This is a change to the BFN Unit 2 Technical Requirements Manual (TRM) which makes revisions required to support implementation of power uprate and 24-month operating cycles on BFN Unit 2. Supporting changes in the Unit 1 and Unit 3 TRMs are also made for certain shared or common equipment. Changes to support power uprate implementation revise the definition of RATED THERMAL POWER and revise the calculated maximum peak containment pressure following a design basis Loss of Coolant Accident as stated in two TRM Bases section. Changes to support implementation of 24-month operating cycles revise from “once per 18 months” to “once per 24 months” the frequencies for certain TRM surveillance requirements. These changes are consistent with BFN Unit 2 Technical Specification amendments 254 and 255. This activity has been reviewed against the criteria 10 CFR 50.59 and does not constitute an unreviewed safety question.

UNIT 3 TRM FOR 24-MONTH CYCLE

This is a change to the BFN Unit 3 Technical Requirements Manual (TRM) which makes revisions required to support implementation of 24-month operating cycles on BFN Unit 3. Changes to support implementation of 24-month operating cycles revise from "once per 18 months" to "once per 24 months" the frequencies for certain TRM surveillance requirements. These changes are consistent with BFN Unit 3 Technical Specification amendment 215. For the requirement related to frequency of snubber inspection, identical changes are also made in the Unit 1 and Unit 2 TRMs to maintain consistency among the three units. This activity has been reviewed against the criteria of 10 CFR 50.59 and does not constitute an unreviewed safety question.

TRM UNIT 2 TSR 3.3.9.3

This safety evaluation for Unit 2 TRM TSR 3.3.9.3 Hydrogen Water Chemistry (HWC) revises Unit 2 Technical Requirement Manual so that the calibration frequency for the Offgas Hydrogen Analyzer is changed from once per OPERATING CYCLE to once per 92 days. This change is consistent with design characteristics of the new Offgas Hydrogen Analyzer equipment and is supported by issued setpoint and scaling calculations. This activity has been reviewed against the criteria of 10 CRF50.59 and does not constitute an unreviewed safety question.