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

FEB 27 1992

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 - ANNUAL OPERATING REPORT FOR JANUARY 1, 1991 - DECEMBER 31, 1991

In accordance with the provisions of 10 CFR 50.4, to satisfy the requirements of 50.59, and Regulatory Guide 1.16 Sections 1.b(1), (2), (3), and in order to satisfy BFN Technical Specifications Sections 6.9.1.2 and 6.9.2.1, we are hereby submitting this annual operating report for BFN Units 1, 2, and 3 for the period of January 1, 1991 through December 31, 1991.

The enclosed report contains a summary of plant conditions, occupational exposure data, reactor vessel fatigue usage, liquid and gaseous releases, and challenges to, or failure of, main steam relief valves. Also contained in the report are safety evaluations for Final Safety Analysis Report revisions, new procedures, procedure revisions, special operating conditions, special tests, temporary plant alterations, and plant modifications.

If you have any questions, please contact me at (205) 729-7570.

Sincerely,

Raul Baron

Raul R. Baron
Site Licensing Manager

Enclosure
cc: See page 2

JE47

ANNUAL OPERATING REPORT

BROWNS FERRY NUCLEAR PLANT

TENNESSEE VALLEY AUTHORITY

JANUARY 1991

through

DECEMBER 1991

DOCKET NUMBERS 50-259, 50-260, AND 20-296

LICENSE NUMBERS DPR-33, DPR-52, AND DPR-68

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OPERATIONAL SUMMARY
JANUARY-DECEMBER 1991

UNIT 1

Unit remains on administrative hold to resolve various TVA and NRC concerns.

UNIT 2

Unit remained on administrative hold to resolve various TVA and NRC concerns until startup on May 24, 1991. Outage work was performed in the following areas:

1. Environmental Qualification and Electrical Issue Modifications.
2. Appendix R Modifications
3. Drywell Structural Steel Modifications
4. Diesel Air Start System Modifications
5. Fire Protection System Upgrades
6. Radwaste System Upgrades
7. SGTS System Upgrades
8. PASS Modifications
9. Core Reload Analysis
10. HPCI and Feedwater Systems Upgrades
11. Nitrogen Supply Systems Upgrades
12. Central Lube Oil Modifications
13. PSC Pump Seals Modifications
14. Various Drawing Corrections
15. Telephone Communications Upgrade
16. PSA and CAS Upgrades
17. Recirculation Pump Monitoring Instrumentation Upgrades
18. Various AHU Upgrades
19. RCIC Upgrades
20. RPV Instrument Reference Leg Re-Route
21. Addition of a Security Continuous Power Supply
22. $H_2O_2$ Sample Line Moisture Intrusion Prevention
23. TIP System Upgrade
24. RHRSW Line Repair, Discharge Temperature Increase, and Miscellaneous Sampling Modifications
25. SJAE System Improvement
26. RHR MOV Interlock Upgrade
27. EECW MI' Sampling Installation

Unit 2 went critical for startup purposes on May 24, 1991, following 48 startup training criticals. On June 5, 1991, for approximately four hours, Unit 2 primary containment was not maintained due to personnel error in that both drywell personnel airlock doors were allowed to be open simultaneously. The door interlocks were disarmed without proper approval. This occurred as personnel were visually performing a reactor vessel leak inspection. After this incident, startup testing resumed and the reactor went above the 80% startup plat... on July 24, 1991.

BFN U2 was shut down five times during the year; only two were unplanned automatic trips. The first manual trip was the result of an increase in torus water temperature due to prolonged operation of RCIC. The reactor remained sub-critical for one day. As part of the power ascension test program, the main turbine was tripped which caused reactor water to fluctuate. This resulted in a low water level trip. A two-day outage was entered primarily to replace the HPCI gland seal head gasket. The first unplanned automatic trip occurred as a result of a broken air line which caused closure of a condensate system valve. The Unit remained shutdown two days for corrective maintenance. When the level of unidentified leakage in the primary containment approached the TS limit of 5 gpm, the Unit was scheduled to be brought off line to accommodate an investigation and repairs. The cause for the leakage was determined to be a broken one-inch sensing line on 2B recirculation pump. The Unit remained shutdown for four days for this repair and other miscellaneous maintenance. A second unplanned automatic trip was caused by a switchyard breaker not closing as required and causing a power transient.

From these trips, no single or related factors were contributors. Overall, the startup and initial seven months of U2 operation went very smoothly with no other major problems.

UNIT 3

Unit remains on administrative hold to resolve various TVA and NRC concerns.

## OPERATIONAL SUMMARY

Docket No. 50-259
Prepared By S. A. Ratliff
Telephone 205-729-2937

OPERATING STATUS

1. Unit Name: Browns Ferry Unit One

2. Reporting Period: December 1991

3. Licensed Thermal Power (MWt): 3293

4. Nameplate Rating (Gross MWe): 1152

5. Design Electric Rating (Net MWe): 1065

6. Maximum Dependable Capacity (Gross MWe): 1098.4

7. Maximum Dependable Capacity (Net MWe): 1065

8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reason: N/A

9. Power Level to Which Restricted, if any (net MWe): N/A

10. Reason for Restrictions, if any: N/A

| | THIS MONTH | YEAR TO DATE | CUMULATIVE |
|---|---|---|---|
| 11. Hours in Reporting Period | 744.0 | 8760.0 | 152744 |
| 12. Number of Hours Reactor Was Critical | 0.0 | 0.0 | 59521 |
| 13. Reactor Reserve Shutdown Hours | 0.0 | 0.0 | 6997 |
| 14. Hours Generator on Line | 0.0 | 0.0 | 58267 |
| 15. Unit Reserve Shutdown Hours | 0.0 | 0.0 | 0 |
| 16. Gross Thermal Energy Generated (MWh) | 0.0 | 0.0 | 168066787 |
| 17. Gross Electric Energy Generated (MWh) | 0.0 | 0.0 | 55398130 |
| 18. Net Electric Energy Generated (MWh) | -2104.0 | -25980.0 | 53535490 |
| 19. Unit Service Factor | 0.0 | 0.0 | 38.1 |
| 20. Unit Availability Factor | 0.0 | 0.0 | 38.1 |
| 21. Unit Capacity Factor (Using MDC Net) | 0.0 | 0.0 | 32.9 |
| 22. Unit Capacity Factor (Using DER Net) | 0.0 | 0.0 | 32.9 |
| 23. Unit Forced Outage Rate | 100.0 | 100.0 | 56.5 |

24. Shutdowns Scheduled Over Next 6 Month (Type, Date, and Duration of Each):

25. If Shutdown at End of Reporting Period, Estimated Date of Startup: TO BE DETERMINED

# OPERATIONAL SUMMARY

Docket No. 50-260
Prepared By S. A. Ratlif
Telephone 205-729-2937

OPERATING STATUS

1. Unit Name: Browns Ferry Unit Two
2. Reporting Period: December 1991
3. Licensed Thermal Power (MWt): 3293
4. Nameplate Rating (Gross MWe): 1152
5. Design Electric Rating (Net MWe): 1065
6. Maximum Dependable Capacity (Gross MWe): 1098.4
7. Maximum Dependable Capacity (Net MWe): 1065
8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reason: N/A
9. Power Level to Which Restricted, if any (net MWe): N/A
10. Reason for Restrictions, if any: N/A

| | THIS MONTH | YEAR TO DATE | CUMULATIVE |
|---|---|---|---|
| 11. Hours in Reporting Period | 744.0 | 8760.0 | 147631 |
| 12. Number of Hours Reactor Was Critical | 701.3 | 4646.3 | 60506 |
| 13. Reactor Reserve Shutdown Hours | 0.0 | 0.0 | 14200 |
| 14. Hours Generator on Line | 696.1 | 4128.5 | 58467 |
| 15. Unit Reserve Shutdown Hours | 0.0 | 0.0 | 0 |
| 16. Gross Thermal Energy Generated (MWh) | 2145488.7 | 11830688.4 | 165075855 |
| 17. Gross Electric Energy Generated (MWh) | 733340.0 | 3898010.0 | 54669808 |
| 18. Net Electric Energy Generated (MWh) | 715130.0 | 3759138.0 | 52810259 |
| 19. Unit Service Factor | 93.6 | 47.1 | 39.6 |
| 20. Unit Availability Factor | 93.6 | 47.1 | 39.6 |
| 21. Unit Capacity Factor (Using MDC Net) | 90.3 | 40.3 | 33.6 |
| 22. Unit Capacity Factor (Using DER Net) | 90.3 | 40.3 | 33.6 |
| 23. Unit Forced Outage Rate | 6.4 | 51.8 | 53.5 |

24. Shutdowns Scheduled Over Next 6 Month (Type, Date, and Duration of Each):

25. If Shutdown at End of Reporting Period, Estimated Date of Startup: N/A

# OPERATIONAL SUMMARY

Docket No. 50-296
Prepared By S. A. Ratliff
Telephone 205-729-2937

## OPERATING STATUS

1. Unit Name: Browns Ferry Unit Three
2. Reporting Period: December 1991
3. Licensed Thermal Power (MWt): 3293
4. Nameplate Rating (Gross MWe): 1152
5. Design Electric Rating (Net MWe): 1065
6. Maximum Dependable Capacity (Gross MWe): 1098.4
7. Maximum Dependable Capacity (Net MWe): 1065
8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reason: N/A
9. Power Level to Which Restricted, if any (net MWe): N/A
10. Reason for Restrictions, if any: N/A

| | THIS MONTH | YEAR TO DATE | CUMULATIVE |
|---|---|---|---|
| 11. Hours in Reporting Period | 744.0 | 8760.0 | 130056 |
| 12. Number of Hours Reactor Was Critical | 0.0 | 0.0 | 45306 |
| 13. Reactor Reserve Shutdown Hours | 0.0 | 0.0 | 5150 |
| 14. Hours Generator on Line | 0.0 | 0.0 | 44195 |
| 15. Unit Reserve Shutdown Hours | 0.0 | 0.0 | 0 |
| 16. Gross Thermal Energy Generated (MWh) | 0.0 | 0.0 | 131868267 |
| 17. Gross Electric Energy Generated (MWh) | 0.0 | 0.0 | 43473760 |
| 18. Net Electric Energy Generated (MWh) | -1629.0 | -19727.0 | 41952809 |
| 19. Unit Service Factor | 0.0 | 0.0 | 34.0 |
| 20. Unit Availability Factor | 0.0 | 0.0 | 34.0 |
| 21. Unit Capacity Factor (Using MDC Net) | 0.0 | 0.0 | 30.3 |
| 22. Unit Capacity Factor (Using DER Net) | 0.0 | 0.0 | 30.3 |
| 23. Unit Forced Outage Rate | 100.0 | 100.0 | 61.0 |

24. Shutdowns Scheduled Over Next 6 Month (Type, Date, and Duration of Each):
25. If Shutdown at End of Reporting Period, Estimated Date of Startup: TO BE DETERMINED

ACRONYMS LISTING

| | |
|---|---|
| ACU | Air Conditioning Unit |
| ADS | Atmospheric Depressurization System |
| AEC | Atomic Energy Commission |
| AHU | Air Handling Unit |
| AOT | Anticipated Operating Transient |
| ARI | Alternate Rod Insertion |
| ARM | Area Radiation Monitor |
| ASME | American Society of Mechanical Engineers |
| ATU | Analog Trip Unit |
| ATWAS | Anticipated Transient Without a Scram |
| BFN | Browns Ferry Nuclear Plant |
| CAD | Containment Atmospheric Dilution |
| CAM | Continuous Air Monitor |
| CAQR | Condition Adverse to Quality Report |
| CAS | Control Air System |
| CASA | Common Accident Signal A |
| CASB | Common Accident Signal B |
| CCWS | Condenser Circulating Water System |
| CDWS | Condensate and Demineralized Water System |
| CFM | Cubic Feet per Minute |
| CIS | Containment Inerting System |
| CLIP | Core Limits Program |
| CPT | Control Power Transformer |
| CRD | Control Rod Drive |
| CRLD | Change Request Licensing Document |
| CS | Core Spray |
| CSCS | Core Standby Cooling Systems |
| DBA | Design Basis Accident |
| DBVP | Design Baseline Verification Program |
| DC | Direct Current |
| DCA | Drywell Control Air |
| DCN | Design Change Notice |
| DCRM | Document Control Records Management |
| DD | Drawing Discrepancy |
| DG | Diesel Generator |
| DSAS | Diesel Start Air System |
| ECCS | Emergency Core Coolant Systems |
| ECN | Engineering Change Notice |
| ECSA | Electrical Conductor Seal Assembly |
| EECW | Emergency Equipment Cooling Water |
| EHC | Electro-Hydraulic Controls |
| EOI | Emergency Operating Instruction |
| EPG | Emergency Procedure Guidelines |
| FCV | Flow Control Valve |
| FHA | Fuel Handling Accident |
| FIT | Flow Indicating Transmitter |
| FW | Feedwater |
| GE | General Electric |
| HCV | Hand Control Valve |
| HELB | High Energy Line Break |

## ACRONYMS LISTING
(Continued)

| | |
|---|---|
| HPCI | High Pressure Coolant Injection |
| HPFP | High Pressure Fire Protection |
| HVAC | Heating, Ventilation and Airconditioning |
| ICS | Integrated Computer System |
| ILRT | Integrated Leak Rate Test |
| IRM | Intermediate Range Monitor |
| LLRW | Low Level Radwaste |
| LOCA | Loss of Coolant Accident |
| LPCI | Low Pressure Coolant Injection |
| MAPLHGR | Maximun Average Planar Linear Heat Generation Ratio |
| MCFL | Maximum Combined Flow Limiter |
| MCPR | Minimum Critical Power Ratio |
| MCR | Main Control Room |
| MG | Motor Generator |
| MIC | Microbiological Induced Corrosion |
| MOV | Motor Operated Valve |
| MS | Main Steam |
| MSIV | Main Steam Isolation Valve |
| MSL | Main Steam Line |
| MSRV | Main Steam Relief Valve |
| NE | Nuclear Engineering |
| PASS | Post Accident Sampling System |
| PCIS | Primary Containment Isolation System |
| PCLD | Primary Coolant Leak Detection |
| PCS | Process Computer System |
| PDD | Potential Drawing Discrepancy |
| PMI | Plant Manager's Instruction |
| PORC | Plant Operations Review Committee |
| PRD | Problem Reporting Document |
| PRFO | Pressure Regulator Failure Open |
| PSA | Plant Service Air |
| PSC | Pressure Suppression Chamber |
| PSI | Pounds Per Square Inch |
| PSIG | Pounds Per Square Inch Gravity |
| PSP | Physical Security Plan |
| PSPDS | Phase I Safety Parameter Display System |
| PSTG | Plant Specific Technical Guidelines |
| QA | Quality Assurance |
| RADCON | Radiological Control |
| RBCCW | Reactor Building Closed Cooling Water |
| RCIC | Reactor Core Isolation Cooling |
| RCW | Raw Cooling Water |
| RETS | Radiological Effluent Technical Specification |
| RFW | Reactor Feedwater |
| RFWP | Reactor Feedwater Pump |
| RHR | Residual Heat Removal |

ACRONYMS LISTING
(Continued)

| | |
|---|---|
| RHRSW | Residual Heat Removal Service Water |
| RMOV | Reactor Motor Operated Valve |
| RMS | Radiation Monitoring System |
| RPS | Reactor Protection System |
| RPT | Recirculation Pump Trip |
| RPV | Reactor Pressure Vessel |
| RRRMS | Radiation Release Rate Monitoring System |
| RSA | Radwaste Service Air |
| RTD | Resistance Temperature Detector |
| RWCU | Reactor Water Cleanup System |
| SA | Service Air |
| SAS | Secondary Alarm Station |
| SDB | Short Delay Band |
| SDBR | Shutdown Board Room |
| SE | Safety Evaluation |
| SFSP | Spent Fuel Storage Pool |
| SGTS | Standby Gas Treatment System |
| SI | Surveillance Instruction |
| SIL | Service Information Letter |
| SLC | Standby Liquid Control |
| SPDS | Safety Parameter Display System |
| SRM | Source Range Monitor |
| SRO | Senior Reactor Operator |
| SSA | Safe Shutdown Analysis |
| SSP | Site Standard Practice |
| ST | Special Test |
| STPU | Short Time Pick-up |
| TACF | Temporary Alteration Control Form |
| TCV | Temperature Control Valve |
| TE | Temperature Element |
| TI | Technical Instruction |
| TIP | Traverse Incore Probe |
| TIS | Temperature Indicating Switch |
| TOL | Thermal Overload |
| TS | Technical Specification |
| UFSAR | Updated Final Safety Analysis Report |
| UNID | Unit Identification |
| UO | Unit Operator |
| WO | Work Order |
| WP | Work Plan |

**UFSAR 1.5.2 - Principal Design Criteria, System by System, and 1.6.4 - Process Control and Instrumentation - Units 1, 2, and 3**

Description/Safety Evaluation

In accordance with the original design of BFN, an offsite power dispatcher in Chattanooga could directly alter the power output from the turbine. The offsite power control was accomplished by changing the frequency of the current to the recirculation pump motor and thereby vary the flow through the reactor and reducing the steam flow to the turbine. The offsite dispatcher control was eliminated by pegging out the recirculation control from being put in the automatic mode and also removing the wiring from the turbine control EHC panel 9-7 so that the automatic dispatch system could not be accidentally initiated. This modification was performed in June 1978, by WP 9209 under DCR 1367 (dated 11-17-77).

The automatic dispatch control was not a safety related feature. However, it was described in the UFSAR and this CRLD removed it from the UFSAR in order to make the UFSAR consistent with current operating procedures. These revisions did not impact the function or performance of the affected system neither did it impact other systems ability to perform its function. Therefore, it is concluded that the change was safe from a nuclear safety standpoint. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 1.6.2 - Nuclear Safety Systems and Engineered Safeguards and 4.7, RCIC System - Units 1, 2, and 3**

Description/Safety Evaluation

This change revised the BFN UFSAR to clearly specify that the RCIC system is not essential for mitigating various AOTs or mitigating the Loss of Control Room Habitability Special Event. RCIC is the preferred system for certain reactor isolation events. This clarification will aid in defining the inputs used for determining the safety class of various RCIC components. The following changes were made:

Move paragraph 1.6.2.5 to become the new paragraph 1.6.1.3.6 and delete the current paragraph 1.6.2.5.

This change moved the RCIC system out of the "Nuclear Safety Systems and Engineered Safeguards" section and put it into the "Nuclear System" section.

Deleted "Reactor Core Isolation Cooling System" from table 1.4-2A sheets 2 and 4 (RCIC will now be included by default in Table 1.4-2B as "Power Generation System Type PG-2"). This change further clarified the classification of RCIC's role. See Table 1.4-2B (sheet 2) and UFSAR paragraph 1.5.1.3.3 for PG 2 criteria as it applies to RCIC operation.

Deleted Section 4.7.2.

This section was not supported by Section 14.6 of the UFSAR. The accident analysis did not rely on RCIC to mitigate "certain pipe break accidents." The remaining discussion of RCIC providing makeup water during shutdown and isolation is covered by the power generation objective. Additionally UFSAR definition 1.2.29 for "Safety Objective" stated that safety objectives are concerned with "conditions considered to be of primary significance to the protection of the public." This definition clearly indicates that safety objectives apply only to nuclear safety systems, and the deleted section was therefore determined to not be applicable to RCIC.

Deleted paragraph 4.7.4.1.

The existing paragraph 4.7.4.2 reiterates RCIC's safety-related role in maintaining reactor pressure vessel, primary containment, and secondary containment boundary integrity as further discussed in UFSAR Chapters 5 and 7. The only other active safety function is for the RCIC system ECCS analog trip Unit power supplies to provide power to the analog trip Units of the HPCI, reactor feedwater, primary containment isolation, and reactor water recirculation systems (see Design Criteria BFN-50-7071).

Revised UFSAR paragraph 4.7.5 subparagraph (3) references to "low-water-level" to: "water level 2"

This change was required to accurately reflect the instrumentation trip level nomenclature.

Deleted the first four sentences of Section 4.7.6 and revised the last sentence to read "The safety design basis is satisfied...."

This change was needed to eliminate any possible confusion that might again lead to the erroneous conclusion that RCIC, rather than ADS, is HPCI's redundant backup. These statements were provided to document that the deleted safety design basis was satisfied. The remaining safety design basis is still covered by the last sentence in the paragraph (which is being retained).
Revised Technical Specification 3.5.F bases as follows:

Delete "...and for certain pipe break accidents" from sentence 1.

Revised "...in which RCIC is required to provide core cooling" to state "...in which RCIC is necessary to maintain sufficient coolant in the reactor vessel so that CSCS are not required."

This change was required to accurately reflect RCIC's role.

This change clarified the classification requirements for RCIC components which are not associated with safety related functions. Since RCIC injection is a function which is designed for "defense in depth" protection of the core, the components associated with the injection function will be maintained as quality related. While these requirements are not as stringent as those for safety related components, they will assure that RCIC is maintained at a level of high reliability such that it can reasonably be expected to function when required to prevent unnecessary challenges to the low pressure CSCS. This is acceptable since RCIC injection is not required to mitigate any chapter 14 events. The safety actions required for the various AOTs are provided by the safety related HPCI and ADS systems, not by the RCIC system. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR is not increased. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 4.2.4.1 - Reactor Vessel Design - Units 1, 2, and 3**

Description/Safety Evaluation

The update of the UFSAR related to the "Reactor Vessel design" involved revisions of text in order to provide internal consistency and clarifications with design documents. The revisions of Table 4.2-1 and Table 4.2-2 and other minor changes provided consistency between manufacturer's data and design documentation. These revisions did not impact the function or performance of the affected systems neither did it change these or other system's ability to interact.

The Reactor Vessel shall be designed to withstand adverse combinations of loadings and forces resulting from operation under abnormal and accident conditions. The revisions to the UFSAR was not a design change but rather a matter of providing internal consistency within the UFSAR (Appendices K and L, Design Criteria, and Manufacturer's data). Consequently, the ability of the Reactor Pressure Vessel design and associated components to mitigate any DBA or AOT was not degraded. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Section 4.10.3.1 - Identified Leakage Rate - Unit 1, 2, and 3**

Description/Safety Evaluation

This change discusses a revision to Section 4.10.3.1 of the UFSAR and its impact on safety. The following sentence was deleted from Section 4.10.3.1.

"The collection chamber filling time is periodically timed during plant operation, and the flange gasket leakage rate is calculated."

The RPV head flange gasket consists of inner and outer stainless steel seals. Vessel integrity is maintained by these seals. Leakage which might occur past the inner seal but held in check by the outer seal, is piped through a solenoid valve to the fore mentioned collection chamber. GE SIL number 42 recommended that operation of the collection chamber fill valve be avoided once leakage is detected through the first inner head seal. Continual operation of the fill valve will only serve to increase the damage to the inner seal. Calculation of the leakage rate as described in the UFSAR is, therefore, no longer performed at BFN. Determination of this leakage rate is not required per the UFSAR or Technical Specifications. It serves primarily for information purposes only. Nuclear safety was in no way impacted.

None of the DBAs or AOTs as described in Section 14 of the UFSAR were affected by discontinuation of the collection chamber leakage rate calculation. This leakage rate calculation never prevented nor decreased the loss of vessel integrity which might occur past the inner seal of the RPV head gasket. On the contrary, GE SIL Number 42 stated this calculation could serve to increase the loss of vessel integrity past the inner seal. If leakage should occur it is detected in the control room as the collection chamber fills up. Actual loss of vessel integrity outside the vessel is not a concern since the outer seal and the collection chamber serve to maintain vessel integrity at the RPV head. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 5.2 - Primary Containment System - Units 1, 2, and 3**

Description/Safety Evaluation

This update was a deletion of information in Table 5.2-1 in regard to total volume of the Pressure Suppression Chamber at elevation 537. Calculations were performed to verify the volume of the Pressure Suppression Chamber (torus) at the minimum and maximum level as indicated in this table. The value given (133,240 $ft^3$) as Torus free volume in this table was incorrect. The volumes given as the minimum and maximum are the values used in verifying the structural integrity of the torus. Deleting the incorrect information did not impact the System Design Parameters and neither did it impact the nuclear safety of the torus. The deleted information was not used as design input or for any functional or operational purpose. The significant information of minimum and maximum water level will remain in the table.

In addition, CRLD BFEP-MN-91051 revised the UFSAR to update the information regarding the Unit 2 primary containment penetration isolation arrangements as described in the current Unit 2 Technical Specifications and Technical Specification change 284. Technical Specification change 284 identified additional containment isolation valves in the PASS, CAD System, and the DCA System which have been installed as a result of plant modifications implemented for Unit 2 restart. These primary containment isolation arrangements have been evaluated and found to be acceptable as documented in the NRC Safety Evaluation for Unit 2 Technical Specification Amendment 193 and the justification for the changes in Technical Specification Change 284. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 5.2.3.4.3 and 14.10.4 - Electrical Penetrations - Units 1, 2, and 3**

Description/Safety Evaluation

Safety Evaluation SEBUFSAR910091 provided justification for revisions to UFSAR Sections 5.2.3.4.3 and 14.10.4 and Design Criteria BFN-50-738 paragraph 4.1.3. The revisions to the UFSAR and Design Criteria removed the specific description of the electrical penetration seals as being of a ceramic material. In addition, the qualified radiological dose reported for the ceramic material was removed from the UFSAR. These changes were necessary because the descriptions were overly restrictive. Currently, some electrical penetration assemblies are used which have seals of a material other than ceramic. However, these penetration assemblies are ensured by 10 CFR 50.49, Environmental Qualification Program, to meet or exceed all qualification requirements. Therefore, the electrical penetration assemblies are ensured of providing equivalent assurance of maintaining primary containment integrity during normal and accident conditions. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 5.2.6.3 - Primary Containment System Design Evaluation - Units 1, 2, and 3**

Description/Safety Evaluation

The update constituted revisions to reflect actual calculations and clarifications of information. No equipment was added, modified, or removed from BFN as a result of the UFSAR revisions. Neither did any of the revisions affect the performance or function of any structures, systems or components. The revision was to update and clarify the information provided in the text of the UFSAR.

The Primary Containment system can not initiate any DBAs or AOTs. In addition, no new credible failure modes have been identified. No unreviewed safety question was created and no Technical Specification changes resulted.

**UFSAR 5.3 - Secondary Containment System - Units 1, 2, and 3**

Description/Safety Evaluation

This update involved revisions to the following subsections: 5.3.3.4, Relief Panels; 5.3.3.5, Locks and Penetrations; 5.3.3.7, Standby Gas Treatment System; 5.3.4.1, Secondary Containment Isolation; 5.3.4.2, Standby Gas Treatment Instrumentation and Control; 5.3.5.2, Standby Gas Treatment System.

The primary purpose of the SGTS is to maintain a negative secondary containment pressure subsequent to an accident and thereby mitigate the consequences of an accident.

The change in percentage of offsite dosage due to a 300 percent steam leak in the tunnel (from .0000005 percent to 10 CFR 100 guidelines to <.1 percent) is still bounded by allowable dose rates specified in 10 CFR 100 for a DBA.

The change in type of weather-stripping used on personnel locks and equipment locks did not significantly change the air infiltration into secondary containment. The secondary containment allowable leakage is still monitored and limited to 12000 cfm.

It has been demonstrated by analysis that the calculated increase in decay heat of the SGTS charcoal and HEPA filter will not increase the temperature to the point where it will ignite the charcoal or cause the desorption of radioactive iodine.

The acceptability and thereby the capacity of the blower fans of the SGTS is demonstrated at each refueling outage prior to refueling by maintaining the secondary containment at 1/4 inch of water vacuum with a system leakage rate not to exceed 12000 cfm.

The other changes to this section of the UFSAR did not constitute changes or alterations to any systems, structures, or components. The deletions and updates brought the UFSAR in line with other existing design documents and available as-built information.

The SGTS cannot initiate any DBAs or AOTs. On the basis of the above discussion, the revisions and updates did not affect the ability of the SGTS to mitigate any DBA. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 5.3.3.7 - SGTS - Units 1, 2, and 3**

Description/Safety Evaluation

This change added additional information to the description of the downstream HEPA filter for each SGT train in Section 5.3.3.7. The additional information clarified that the function of the downstream HEPA filter is to prevent particles, especially carbon, from the charcoal filter from passing into the stack.

The function of the downstream HEPA filter remained unchanged. Particles and radioactive iodine from the Reactor Building are removed by the upstream HEPA filter and the charcoal absorber and thus the filteration capability of the downstream HEPA filter is of little significance to the accident mitigation function of the SGTS. However, it does prevent charcoal fines from the charcoal absorber from being transported to the stack. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Chapter 6 and 7 - Core Standby Cooling Systems and Control and Instrumentation - Units 1, 2, and 3**

Description/Safety Evaluation

Increased stoke times for these valves was required because of environmental qualification design changes to the operators for these valves and to relax the acceptance criteria for the valve stroke response time for the RHR Torus Spray Injection Valves during surveillance testing. Also, the HPCI system design flow time was revised from 25 seconds to 30 seconds because the time required for the HPCI system to reach design flow into the reactor vessel is dependent on the stroke time of the HPCI pump discharge valves. The injection times for CS, RHR, and the vessel depressurization for a Recirculation Suction line break were also revised to reflect the new LOCA analysis.

This safety evaluation evaluated the UFSAR changes to Chapters 6 and 7 for Units 1 and 3 valve stroke time changes and for Unit 2 RHR Torus Spray Injection Valves stroke time changes. This evaluation only considers the impact to the UFSAR accident analyses and not all aspects of the actual design change, such as the thrust developed as a result of the valve regearing. The 10 CFR 50.59 review to incorporate the UFSAR change for the remainder of the Unit 2 valve stroke time changes was given by the safety evaluations for ECNs P3116, P3117 and P3118. Table 6.5-1 has been revised to specify the type of pipe break which resulted in the ECCS system response times given in Table 6.5-1.

A safety evaluation was performed by GE which examined the impact of valve stroke time changes on the plant safety analysis. A comprehensive LOCA analysis was performed with the new valve stroke times because the primary safety functions of these valves are to provide core cooling in case of a LOCA. This analysis was performed to ensure that plant response for a full spectrum of postulated pipe breaks and assumed single failures met all regulatory requirements for BFN. The safety evaluation also examined the impact of the extended valve stroke times on non-LOCA events (such as HELB and fire events), other safety functions of the valves (such as containment isolation), and offsite dose releases. The safety evaluation demonstrated that the extended valve stroke times will have an insignificant impact on all the analyses listed above. Furthermore, the extended valve stroke times did not result in any changes in the MAPLHGR for all fuel types in BFN. Also, the peak cladding temperature was found to be well below the 2200°F limit specified by 10 CFR 50.46. This change did not alter the intended safety functions of these valves. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 6.4 and 6.6 - Description, Inspection, and Testing for Core Standby Cooling Systems - Units 1, 2, and 3**

Description/Safety Evaluation

Three clarity type changes were made to UFSAR Sections 6.4 and 6.6. Changing the type of check valve did not impact the type of operational single failure such as opening or closing nor did it impact any assumed operator error. Deletion of the UFSAR statements in regard to extended HPCI operation in hot standby or the fact that records are not kept of number of thermal cycles of components did not impact any of the DBAs or AOTs as evaluated in Chapter 14 of the UFSAR.

No equipment was added, modified, or removed from BFN as a result of the UFSAR changes. The critical systems, structures and components are manufactured from ductile material and any cracks due to thermal cycling will be detected by monitoring of leaks in the drywell. Consequently, not monitoring the thermal cycles did not result in any new modes of failure. The text changes did not result in any new or different design input. However, the changes being clarifications made the UFSAR consistent with other design documents and surveillance instructions and did not introduce any new failure modes or alter existing failure modes. Technical Specifications 6.10 Station Operating Records and Retention specified the monitoring and recording requirements in regard to fatigue usage. At BFN the applicable codes require fatigue usage evaluation for the reactor pressure vessel only. TI-19 implements the requirements of the Technical Specifications 6.10.q. In accordance with the Technical Specification, plant operations are reviewed and a cumulative usage factor is determined and reported in the Annual Operating Report. No records are kept nor are any required specifically for design basis thermal cycles for individual components.

The Technical Specifications on BFN do not specify the HPCI system to be used in a hot standby condition neither does it identify the type of check valves used in the HPCI turbine exhaust line or the HPCI turbine drain line. Deletion of statement in regard to record keeping for the Standby Cooling Systems did not impact the margin of safety. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 6.4.1 - HPCI System - Units 1, 2, and 3**

Description/Safety Evaluation

The UFSAR was revised to provide the Analytical Limit corresponding to the Allowable Value that existed in the Technical Specifications. "The setpoint and Scaling Calculations" ED-Q2073-880071 R3, ED-2073-880072 R3, ED-Q2073-880073 R3 and ED-Q2073-880074 R3 demonstrated that with the existing setpoint, the Analytical Limit was not exceeded under all required operating conditions of the plant. The Process Safety Limit Calculation for HPCI

Turbine Exhaust Pressure (in between rupture discs) was performed to demonstrate that the safety equipment would perform its intended function under all conditions of operations. The calculation showed that the switches would actuate upon rupture of the rupture disc at a process value 52 psig or less. There was no change in performance of any equipment or its ability to perform its function. The analyzed DBAs and AOTs of Chapter 14 were not impacted by the UFSAR changes. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 7.12 - Process Radiation Monitoring - Units 1, 2, and 3**

Description/Safety Evaluation

This update constituted editorial changes and corrections. No equipment was added, modified, or removed from BFN as a result of the UFSAR changes. The revision of text did not result in any new or different design input. However, the changes provided clarifications and made the UFSAR consistent with other design documents and surveillance instructions and did not introduce any new failure modes or alter existing failure modes. The revision to the UFSAR did not impact the ability of the RMS to mitigate any DBA or AOT. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 7.16 - Processing Computer System - Units 1, 2, and 3**

Description/Safety Evaluation

The RWM baseline assessment was implemented to document the as-configured RWM system. The RWM system has recently undergone baseline testing and software modification per ST 90-11, RWM Operability, and Software Change Request BF-SCR-GE4020-030. The purpose of this assessment was to document actual system operation and to assess against the description found in UFSAR and document any discrepancies between the two. The process computer (RWM) is important to safety for the normal operations of achieving criticality, heatup, power operation, and achieving shutdown. The RWM is not important to safety for any transient or accident. The existing operators panel configuration as noted in the safety assessment did change the actual system description of the RWM from the UFSAR description. The RWM did not have a Shutdown Margin Test mode as described in the UFSAR, but rather had a Rod Test mode. The RWM also allowed only one sequence to be stored in its memory, so that the desired sequence could not be selected from the operators panel. The as-configured panel is more restrictive than that described in the UFSAR, since the Rod Test mode allowed only one rod to be withdrawn at a time instead of the two allowed by the Shutdown Margin sequence description, and only one sequence can be stored in memory at a time instead of two. Although the actual system operation is less flexible than the described system, these changes do not affect the overall reliability of the RWM system or the process computer. No other equipment important to safety is affected by the differences noted between the RWM operators panel and the UFSAR description.

Therefore, there was no increased probability of a malfunction of equipment important to safety over that previously evaluated in the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 7.2.3.8 - SCRAM Bypasses - Units 1, 2, and 3**

Description/Safety Evaluation

This update constituted corrections to the subsections fourth paragraph:

Page 7.2-13 the fourth paragraph in the subsection previously read:

> "The scram, initiated by placing the mode switch in SHUTDOWN, is automatically bypassed after a time delay of 10 seconds. The bypass is provided to restore the Control Rod Drive Hydraulic System valve lineup to normal. An annunciator in the control room indicates the bypassed conditions."

This paragraph was revised to read:

> "The scram, initiated by placing the mode switch in SHUTDOWN, is automatically bypassed after a time delay of 2 seconds. The bypass is provided to eliminate a sustained scram and to enable the scram to be reset with the mode switch in shutdown. An annunciator in the control room indicates the bypassed condition."

In order to eliminate the possibility of a sustained scram caused by the Reactor Mode switch being in the Shutdown position, an automatic bypass circuit has been provided. The circuit is designed so that the bypass is delayed for a minimum of two seconds after the reactor mode switch is place in Shutdown, to assure that the reactor scram is not interrupted.

The specified time of two seconds is in accordance with the original design and is not a design change. The original ten second number may have been confused with the 10 second delay in the scram reset circuits of the Reactor Protection System. Based on the above review, it is apparent that the change did not impact the function of any affected systems. Therefore, the change was safe from a nuclear safety standpoint. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Table 7.3-2, PCIS Instrument Specifications - Units 1, 2, and 3**

Description/Safety Evaluation

The lower trip setting of the temperature switches (2-TS-71-2A through -2S and 2-TS-73-2A through -2S) used to detect steam leaks in the steam supply piping for the HPCI and RCIC systems was changed by CRLD No. BFEP-EEB-90017R0. The request was forwarded to the NRC as TVA BFN-TS-290. The Technical Specification change has been approved by NRC. UFSAR Table 7.3-2 was revised per the approved Technical Specification changes.

UFSAR Section 7.3 was revised to remove the setpoint value for the Main Steam Tunnel Reactor and Turbine Building temperature switches (2-TS-001-17A-D, -29A-D, -40A-D, and -54A-D) from the text and add the 210°F Analytical Limit to Table 7.3-2.

The UFSAR Table 7.3-2 listed range and accuracy for the MS/HPCI/RCIC temperature switches. These values were removed and reference was made to the Setpoint and Accuracy Calculation where these values are used. This accuracy is acceptable.

An allowable value is the limiting value that the trip setpoint can have when testing periodically beyond which the instrument channel is declared inoperable and corrective action must be taken. An instrument loop analytical limit is a limit at or below the safety limit used in the system simulation to verify acceptable system operation. The safety limit is a limit on an important process variable to reasonable protect the integrity of certain physical barriers that guard against uncontrolled release of radioactivity. The changes to the UFSAR for the HPCI/RCIC steam supply line leak detection trip setting were the result of the approved Technical Specification. Range and accuracy values were changed to correspond to the temperature switches installed.

The changes to the UFSAR for the Main Steam Line leak detection temperature switches were a result of calculation ED-Q0001-880487 R0 to determine the setpoint required to support the Technical Specification. The setpoint value was removed from the UFSAR text and the correct range and accuracy values for the installed temperature switches were inserted. The documentation changes to the UFSAR did not affect any plant equipment or its performance. Therefore, it did not create any possibility of a malfunction of any equipment. No unreviewed safety question was created.

**UFSAR 7.3.5 - PCIS Safety Evaluation - Units 1, 2, and 3**

Description/Safety Evaluation

BFN Unit 2 Technical Specifications indicate that the trip setpoint associated with the RCIC flowmeters is 450 inches of water not 442 inches of water as indicated in the UFSAR. The Technical Specifications indicate that the trip setting for HPCI is less than or equal to 90 psi. There was no documentation to support the flow rates indicated at reduced pressure on either system. To change the limits to actual process values the statements were revised to read as follows:

> "The differential pressure trip setting for high flow through the redundant flow meters in the RCIC is less than or equal to 300 percent of rated steam flow at 1140 psia. This trip point was selected to provide sufficient margin to prevent isolation during normal startup transient pressure measurements associated with the particular flow meters utilized (elbow taps). At lower steam pressures, the trip setting in percent of rated flow is conservatively lower. A time delay relay in the trip

circuit prevents isolation during normal startup. The differential pressure trip setting for high flow through the HPCI flow meter is less than or equal to 225 percent of rated steam flow at 1140 psia. This trip point was selected to provide sufficient margin to prevent isolation during normal startup transient pressure measurements associated with the particular flow meter utilized (venturi). At lower steam pressures, the trip setting in percent of rated flow is conservatively lower. A time delay relay in the trip circuit prevents isolation during normal startup."

No equipment was added, modified, or removed from BFN as a result of the UFSAR changes. The rewriting and updating of this UFSAR section did not result in any new or different design input. However, the changes provided clarifications and made the UFSAR consistent with other design documents and site procedures and did not introduce any new failure modes or alter existing failure modes. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Table 7.4-1 - HPCI System Instrumentation - Units 1, 2, and 3**

Description/Safety Evaluation

As of the result of Calculation MD-Q2073-870205 R0, "The Process Safety Limits for HPCI Turbine Exhaust Pressure for 2-PS-073-22A and 2-PS-073-22B", the upper process safety limit was calculated to be 165 psig. The UFSAR Table 7.4-1 was also revised to reflect the new calculated Analytical Limit for HPCI Turbine exhaust pressure switches 2-PS-73-22A and 2-PS-73-22B.

As of the result of Calculation ED-Q2073-880296 R1, "The Setpoint and Scaling Calculation" for HPCI pump discharge flow switch, which is used to open/close the minimum flow bypass valve FCV-73-30, the process Analytical Limit was calculated to be 500 GPM. UFSAR Table 7.4-1 was revised to reflect the new calculated Analytical Limit for 2-FS-73-33 as 500 GPM.

The above revisions to the UFSAR tables were documentation only. No physical work was involved. No changes to the Technical Specification was required.

There was no change in performance of any equipment or its ability to perform its function. The Demonstrated Accuracy Calculation and the Process Safety Limit for HPC Turbine Exhaust Pressure Calculation, which necessitated the UFSAR revisions, were performed to demonstrate that the safety equipment would perform its intended function under all conditions of operations.

The DBAs liste in Section 3.2 of the Restart Design Criteria for HPCI System BFN-50-7073 wer evaluated to determine the acceptability of the changes. The analyzed DBAs an AOTs of Chapter 14 were not impacted by the UFSAR changes. No unreviewed saf ty question was created.

**UFSAR Table 7.4.3.2.5 - HPCI Valve Control - Units 1, 2, and 3**

Description/Safety Evaluation

This update constituted an update of the HPCI instrumentation system as described:

Page 7.4-8...third paragraph stated:

"All automatic valves in the HPCI are equipped with remote-manual test capability, so that the entire system can be operated locally or from the Main Control Room. Motor-operated valves are provided with appropriate torque switches to turn off the motors when the full-open or full-closed positions are reached. Certain valves are automatically closed on isolation or turbine trip signals. All essential components of the HPCI controls operate independent of AC power."

The "full-open or" was deleted in this paragraph.

TVA ECN M208 required the removal of torque switches in the opening circuit of all valves presently wired for torque seating in the open direction. The removal of the torque switch will improve equipment reliability by assuring the valves will be fully opened, no new failure modes are created, no change in system design parameters, and adequate safety injection/core cooling capability. This update made the UFSAR consistent with design documentation and the as-built design.

This modification was a system improvement that assured that any automatic valve in the HPCI system wired for torque seating in the open direction will open to the fully opened position and consequently provide full flow of steam to the HPCI turbine during mitigation of any DBAs or AOTs.
The changes did not alter the function of the HPCI system or how it interacts with other systems and consequently did not create the possibility for an accident of a different type. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Table 7.4-4 - LPCI Instrumentation - Units 1, 2, and 3**

Description/Safety Evaluation

CRLD BFEP-EEB-91029 R0 requested to delete recirculation pump differential pressure switches PDS-68-65 and PDS-68-82 from BFN UFSAR Table 7.4-4. This change was evaluated against the DBAs listed in Section 3.2 of Restart Design Criteria BFN-50-7068 R2, Reactor Water Recirculation System, Unit 2, to determine the acceptability of the change.

This did not affect any allowable values documented in Technical Specification. Existing setpoint, range and accuracy values were not changed. The analyzed DBAs and AOTs of Chapter 14 were not impacted by the UFSAR changes. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 7.5 - Neutron Monitoring Systems, 8.4 - Normal Auxiliary Power System; 8.5 - Standby AC Power Supply and Distribution; and 8.8 - Auxiliary DC Power Supply and Distribution - Units 1, 2, and 3**

Description/Safety Evaluation

The UFSAR changes can be summarized as minor documentation changes required to bring the UFSAR into agreement with other plant documentation, per the UFSAR verification program. These changes did not represent physical modifications being made to the plant.

BFN UFSAR Section 7.5.9.2.2, Section 7.5.9.2.3 and Figure 7.5-22 Traversing Incore Probe, was rewritten to reflect editorial changes, design criteria, site procedures, and as-built information.

This change replaced a list of diesel generator start signals with a more descriptive list without changing any of the diesel generator start signals, revised the description of the degraded voltage logic relays, the LOP/LOCA diesel generator loading sequence table, corrected the LPRM detector potential power supply adjustment range to agree with other plant documentation, and removed unsupported claims from the section describing the 48-VDC annunciator and telephone power supply system. UFSAR Section 8.8.1.3 and Table 8.8-1 describe the 48-VDC annunciator and telephone power supply system. The 48-VDC power supply and its loads are not important to safety and cannot initiate malfunctions of equipment important to safety. The updated description of the degraded voltage logic relays, the changed LOP/LOCA diesel generator loading sequence table, and the corrected LPRM detector potential power supply adjustment range simply reflect changes already accepted in calculations, the Technical Specifications, design criteria, relay setting sheets, and other plant documentation. The list of diesel generator start signals contains the same signals with a more detailed description.

No equipment was added, modified, or removed from BFN as a result of the UFSAR changes. The rewriting and updating of these UFSAR sections did not result in any new or different design input. However, the changes did provide clarifications and made the UFSAR consistent with other design documents and site procedures and did not introduce any new failure modes or alter existing failure modes. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Figure 7.5-2 - SRM/IRM Neutron Monitoring Uni' - Units 1, 2, and 3**

Description/Safety Evaluation

CRLD BFEP-EEB-91032 R0 replaced Figure 7.5-2 of the BFN UFSAR. This Figure was a sketch of the SRM/IRM Neutron Monitoring Unit. This sketch was replaced with controlled drawing 729E946-1. This drawing shows the same equipm..nt. There were no differences between the dimensions or other information presented on the sketch and the information presented on the drawing.

The Figure replacement did not alter the function or performance of the affected system. It did not impact other systems ability to perform their function nor did it change this or other systems abil'ty to interact. The analyzed DBAs and AOTs of Chapter 14 were not impacted by the UFSAR change. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Section 7.7 - React r Manual Control System - Units 1, 2, and 3**

Description/Safety Evaluation

This revision changed the description of the charging water to the accumulator pressure alarm from low to high in UFSAR Section 7.7.4.4 and changed the indicated setpoint for 2-PS-85-13 from 1410 psig decreasing to 1510 psig increasing in UFSAR Table 7.7-1. The indicated section and table discussed and listed the instrumentation associated with the CRD. Section 7.7.4.4 described the control room indicating lights and alarms which let the operator know the conditions of the CRD Hydraulic System. A subsection (k) associated with this description described the alarm for the charging water to accumulator pressure condition. Table 7.7-1 listed the control room instrumentation associated with CRD and identified the Accumulator Header Charging Pressure Alarm as 2-PA-85-13.

No equipment was added, modified, or removed from BFN as a result of the UFSAR changes. The rewriting and updating of this UFSAR section did not result in any new or different design input. However, the changes provided clarifications and made the UFSAR consistent with other design documents and site procedures and did not introduce any new failure modes or alter existing failure modes. The analyzed DBAs and AOTs of UFSAR Chapter 14 were not impacted by the UFSAR changes. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Appendix 7.7.B - RSCS - Units 1, 2, and 3**

Description/Safety Evaluation

This Safety Evaluation supported the UFSAR revision to modify Appendix 7.7.B. This appendix provided a description of the BFN Rod Sequence Control System for Units 1, 2 and 3 including the system's objective of preventing out-of-sequence control rod movement within preset power levels. By preventing out-of-sequence control rod movement or restricting control rod movement to a notch mode of operation, a postulated rod drop accident would not result in peak enthalpies in excess of 280 cal/gm over the range of plant operation and core exposures.

In addition, to the RSCS concept and technical descriptions, the appendix presents original cycle analysis of maximum worth in-sequence control rod drops and the maximum peak enthalpies that result. These are presented in substantiation of the RSCS function. Also included in the appendix is a detailed basic description of the original BWR control rod drop calculational methods, geometry, and nuclear constants. However, the detailed analysis information presented is only for the original cycle of operation of the BFN reactors.

The UFSAR change provided deletion of outdated descriptive information, tables, and figures and added necessary textual reference to the current core reload licensing amendment for each Unit's rod drop analysis results to substantiate the RSCS purpose. Current rod drop nuclear analysis methods and modeling are described in the present reload analysis (TVA-RLR-001, Appendix A) of UFSAR, Appendix N.

Actions performed by safety related systems during a DBA as analyzed did not change as a result of this UFSAR change. Removing the historical initial analytical rod drop accident information, replacing the previous UFSAR section by the current analytical methods, and the results of the current core reload analyses did not change any existent system function and no equipment was added or changed. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 7.9 - Recirculation Flow Control System - Units 1, 2, and 3**

Description/Safety Evaluation

This was an editorial change to Section 7.9.4.5.5 of the BFN UFSAR. This section discusses the RPT logic and previously read in part:

> "Capability is provided for testing and calibrating the system logic quarterly, and circuit breakers once per refueling outage."

BFN Unit 2 Technical Specifications indicate that the RPT 2 logic is to be tested once per month. The plant instructions which govern this testing also indicate that this testing is performed once each month. UFSAR paragraph 7.9.4.5.5 was changed to read:

> "Capability is provided for testing and calibrating the system logic monthly, and circuit breakers once per refueling outage."

The RPT logic section, UFSAR Section 7.9.4.5.3 was also changed. This section also deals with RPT logic. This section previously read in part:

> "The power for the RPT system logic is from the same power supply sources used for the Reactor Protection System. Wiring for the two-pump trip system requires special isolation, routing, and protection considerations and is in accordance with the design specification "Electrical Equipment Separation for Protection System."

Review of the design drawings for the RPS System and the RPT system revealed that the RPS system is powered from the 120V-AC RPS buses, while the RPT logic is powered in part from the 250V-DC RMOV boards. Some of the logic inputs to the RPT system are indeed developed from the RPS system but the RPT logic is clearly powered from a different source than RPS. To eliminate possible confusion regarding the power source for the RPT logic this paragraph was revised to read:

> "Wiring for the two-pump RPT system requires special isolation, routing, and protection considerations and is in accordance with the design specification "Electrical Equipment Separation for Protection Systems."

The update of the UFSAR sections related to the RPT logic involved updates and editorial revisions of the text. The revisions did not alter the function or performance of the affected system. It did not impact other systems ability to perform their function nor did it change this or other systems ability to interact. This UFSAR change did not involve the addition of any new equipment or alteration of any existing equipment. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Table 8.5-1 - DBA Loss of Coolant on One Unit - Units 1, 2, and 3**

Description/Safety Evaluation

UFSAR 8.5.4.1 refers to UFSAR Table 8.5-1 for the order and time at which loads are applied to a typical DG under accident conditions. Table 8.5.4.1 incorrectly identifies the time (20 seconds) at which some loads may be applied to the DGs. The change revised BFN UFSAR Table 8.5.1 per CRLD BFEP-EEB-89005 R0 to reflect that the closed cooling water pumps, drywell blowers, and SGTS be permitted to be applied to the DGs 40 seconds after a common accident signal combined with DG power available.

The change from 20 seconds time delay to 40 seconds did not modify the 480V load shedding time delay relay setpoints, but is a documentation change only. The DBVP analysis, which is based upon the 40 second time delay did not identify any events which were beyond the presently bounded UFSAR accident analyses for the closed cooling water pumps, drywell blowers, and SGTS and equipment which could be affected by this equipment. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Figures 8.5.2 - Standby AC Power and Distribution, Safety Design Basis - Units 1, 2, and 3**

Description/Safety Evaluation

UFSAR Amendment 8 Section 8.5.2 Safety Design Basis 13 previously stated that the DG capacity shall be within the limits of Safety Guide 9. This statement did not clearly reflect BFN compliance to Safety Guide 9.

The change revised Basis 13 to state that although the DGs are not required to meet the specific load, voltage and frequency response limits of Safety Guide 9, their capacity, and capability shall be adequate to meet the intent of Safety Guide 9.

Safety Guide 9 describes an acceptable basis for the selection of DG sets of sufficient capacity and capability to adequately supply the design basis load. It also sets specific numerical limits (which include margin) for load, voltage and frequency response of the DGs that the NRC staff considers adequate to generically meet this basis. The results of restart tests have documented the DG responses. Although the BFN diesels met the overall intent of Safety Guide 9, the specific numerical limits were not met.

The basis for the acceptability of the DG capacity and capability for Unit 2 operation has been established by a program of tests, calculations and analysis. Based on the results of this program the capacity of the DGs has been shown acceptable and the analysis documented to the NRC. The NRC has reviewed this analysis and concluded in the SER that the DGs have the demonstrated capability to adequately supply the safety related loads.

Furthermore, the design criteria reflects the actual basis and compliance to Safety Guide 9 as established above. A summary of the results of these analysis were documented in UFSAR Sections 8.5.4.2, 8.5.4.2.1 and 8.5.4.2.2. Therefore, this change was only a clarification of actual existing basis and analysis that have shown acceptability of the DG capacity and capability as required by Safety Guide 9. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Figures 8.7-2, 8.8-1 and 8.8-3 - Units 1, 2, and 3**

Description/Safety Evaluation

This revision deleted UFSAR Figures 8.7-2, 8.8-1 and 8.8-3 from the UFSAR. Sections 8.7, 8.8, Table 1.3-1, and Table 1.3-2 were also revised to remove references to the deleted Figures.

The subject UFSAR Figures were singleline drawings for the 120V-AC Plant Preferred and Non-preferred power system (731E758-1), the 48V-DC Annunciator and Communications power system (731E704), and the 24V-DC power system (731E717). These drawings were not classified as primary or critical drawings. The UFSAR description of each of these power systems is brief, but adequate for non-safety related power systems. No DBAs or AOTs, as evaluated in Chapter 14 of the UFSAR, were adversely affected by revising the UFSAR to delete the subject UFSAR Figures. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 9.2 - Liquid Radwaste System - Units 1, 2, and 3**

Description/Safety Evaluation

The UFSAR text change (Section 9.2.5, fourth paragraph, third sentence) replaced the word "all" with "most" to indicate that most pipe connections in the liquid radwaste system are welded and few connections are physically threaded, with the balance consisting of flange connections for ease in maintenance. The use of the word "most" is the proper way to identify the welded pipe connections in the Liquid Radwaste System. The threaded connection still complies with the system pressure/temperature/pipe class. Numerous other systems (some much larger than the liquid radwaste) are not welded, and for those non-welded connections in the Radwaste Building, flooding aspects are addressed in Safety Evaluation Number SEBUFSAR890095, Revision 1. Based on this, the potential leaks from the few non-welded connections in the Liquid Radwaste System did not decrease nuclear safety.

The current revision to the UFSAR (page 9.2-4, Section 9.2.4.2, fourth paragraph) states that the tritium level in the plant effluent is less than 1E-8 uCi/ml and that the MPC for the Tritium in drinking water is 3E-8 uCi/ml. The change, obtained from operating data, revised the paragraph to read: "Tritium is typically present in the effluent at an average quarterly concentration of less than 2E-7 uCi/ml. Since the 10 CFR Part 20 limit for tritium (soluble) is 3E-3 uCi/ml, the incremental contribution of the plant release is considered insignificant." In addition, UFSAR Table 9.2-2 required revision to reflect the average quarterly concentration for tritium which, in turn, resulted in changing the tritium release rate and the fraction of the part 20 limit.

The use of tritium concentration based on operating data and release limits currently presented in UFSAR Table 9.2-2 simply clarified and provided consistent UFSAR information, and as such, did not decrease nuclear safety. Replacing the drinking water limits with 10 CFR 20 limits for tritium in the plant effluent was proper and consistent with release requirements. Tritium concentration in the plant discharge is covered by the 10 CFR 20 requirements and therefore, no effect on the plant Environmental Impact Statement was involved. No physical change was made to the liquid radwaste system and components, or Technical Specification. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Change; Section 9.2.6 - Liquid Radwaste System - Units 1, 2, and 3**

Description/Safety Evaluation

The UFSAR change reflects the results of the liquid radwaste volumetric calculations and the liquid radwaste spill study. In the calculations, the quantity of liquid radwaste was derived and summarized for the maximum operating volume of 383,060 gallons. The concern raised on the CAQR is that the operating liquid levels in the radwaste tanks may be above the design safety analysis described in the current UFSAR Section 9.2.6. In order to resolve this concern, a study was conducted to evaluate the impact of spillage from the worst offending tank in the radwaste building. To consider the total rupture of all tanks, piping and components and the subsequent liquid release outside the radwaste building would be a less conservative assumption. The dilution effect of the low level liquid radwaste would tend to lower the activity of the higher level radwaste, thereby reducing the impact of the combined liquid release.

During the study, NUREG-0800 served as the guide in considering the rupture of the worst offending tank which was identified as the waste collector tank. Input data for the study include the maximum volume of approximately 38,000-gal. and the isotopic distribution as Table 9.2-4 in the UFSAR change.

This UFSAR change incorporated the results obtained from the liquid radwaste volumetric calculations and the conclusions derived from the radwaste spill study. The study concluded that the resulting radioactivity at the nearest public water supply was within the limits of 10 CFR 20. This UFSAR change did not affect the design or operation of any equipment or structures which could initiate any previously evaluated accidents. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 10.4 - Tools and Servicing Equipment, 10.3 - Spent Fuel Storage, 10-2 - New Fuel Storage - Units 1, 2, and 3**

Description/Safety Evaluation

This update constituted some editorial changes and the deletion and replacement of figures. The update of the UFSAR related to the Fuel Handling and Storage System sections involved deletion and revision of figures and text changes. The deleted figures (drawings) from the UFSAR will still remain TVA controlled documents.

The use of the light weight water-tight gate has been discontinued as it was demonstrated that the removable concrete blocks provided the required leak tightness and shielding. As a consequence, eliminating the use of the light weight water-tight gate in the storage pool will not impact the water level in the fuel pool.

The criticality analysis for the spent fuel assemblies is based on the thickness of .056" as given in the UFSAR. The minimum Boral Neutron Absorber Insert is .071". Consequently the thickness used in the criticality analysis is approximate and will result in conservative $k_{eff}$.

Removal of the low density storage rack tiedown lugs and the design and installation of a different type of restraint device have been demonstrated by analysis to meet all applicable requirements.

As indicated in the above discussion, the systems affected by this UFSAR revision will continue to meet all applicable regulatory requirements and will perform their function as designed. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 10.5.4 - Fuel Pool Cooling and Cleanup System - Description - Units 1, 2, and 3**

Description/Safety Evaluation

The update of the UFSAR related to the FPC System section involved deletion and revision of text. These revisions did not impact the function or performance of the affected system. In accordance with the Technical Specifications, the water level is required to be 8 1/2 feet or greater above the top of the spent fuel when irradiated fuel is stored in the spent fuel pool. In addition, when irradiated fuel is in the fuel pool, the pool water temperature shall be $\leq 150°F$. The margin of safety as indicated was not impacted by indicating the proper location of water level switches, location of valves, and correcting a temperature that was verified by a calculation.

No equipment was added, modified, or removed from BFN as a result of the UFSAR changes. The deletion and revision of text did not result in any new or different design input. However, the changes provided clarifications and made the UFSAR consistent with other design documents and surveillance instructions and did not introduce any new failure modes or alter existing failure modes. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 10.10 - EECW - Units 1, 2, and 3**

Description/Safety Evaluation

This update constituted revisions to reflect actual system operation, editorial changes, current design criteria, site procedures, and as-built information.

The EECW system cannot initiate any DBAs or any AOTs. The EECW system can however, mitigate the consequences of DBAs and AOTs as evaluated in Chapter 14 of the UFSAR. The EECW system supplies cooling water to the following safety related systems; Standby Diesel Engine Coolers, RHR pump seal heat exchangers, Shutdown Board Room Chillers, Control Bay Chillers and $H_2O_2$ Analyzers. Sufficient redundancy is provided in the EECW system such that a single failure of any EECW component will not jeopardize the function of the EECW system. In the event of loss of offsite power, the EECW system will continue to provide cooling water to the essential systems and components required to perform a function. It is concluded that the changes to the UFSAR did not impact the ability of the EECW system to mitigate the consequences of any DBA or AOT in accordance with Chapter 14 of the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 10.11 - Fire Protection Systems - Units 1, 2, and 3**

Description/Safety Evaluation

UFSAR Section 10.11, paragraphs 10.11.2(1) and 10.11.2(2), discusses the design basis of the HPFP and $CO_2$ Systems. These paragraphs were revised to include other fire protection features, their design basis, and remove inconsistencies. Paragraph 10.11.3.4.1 discusses the portable fire protection equipment. The statement regarding fire extinguisher capacities in the paragraph was revised.

The accidents and transients of UFSAR Chapter 7.18, 10.11, and 14 were reviewed. No accidents/transients (except fire mitigation) are impacted by this UFSAR text revision, since no safety-related systems are involved, or failure of the affected portions do not initiate or contribute to mitigation/prevention of any accidents/transients except a fire. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 10.12 - Heating, Ventilation, and Air Conditioning Systems - Units 1, 2, and 3**

Description/Safety Evaluation

This update constituted updates in the description of the A/C system. No equipment was added, modified, or removed from BFN as a result of the UFSAR changes. The text changes did not result in any new or different design input. However, the text changes provided clarification, made the UFSAR consistent with other design documents, surveillance instructions, and did not introduce any new failure modes or alter existing failure modes. The UFSAR change that deleted the statement related to the Unit 3 chill-water return temperature did not impact the function or performance of the affected system. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR Figure 10.12-2 - Ventilation and Air Conditioning Flow Diagram - Units 1, 2, and 3**

Description/Safety Evaluation

Air flow deficiencies were identified in the Unit 1 and 2 250V battery areas within the SDBR. One of the safety concerns inside a battery room was to ensure that hydrogen concentration was maintained below two percent. It was determined that the time required to reach a hydrogen concentration of two percent in Unit 1 and 2 250V shutdown board battery rooms, assuming the ventilation system was inoperable, would be 209 hours, or 13.4 days during normal plant operation. To deal with this problem, operating instructions have been revised to institute ventilation monitoring and to take appropriate administrative actions if the ventilation system is not functional for Units 1 and 2 250V shutdown board battery rooms during normal and accident conditions. Hence, these measures will ensure that even if the ventilation fans are inoperable, the hydrogen concentration in the battery rooms will remain below two percent and will not pose a safety concern.

Air flow deficiencies were identified in the fresh/makeup air quantities for the various air handling units serving the main control rooms. As shown in calculation, the fresh air intakes are only used during normal operation of the plant and are isolated during a LOCA. Hence, these deficiencies are not a concern during a LOCA. Some control room AHUs were shown to exceed the design supply or the exhaust air quantities. Any variations in the air flow quantities is sensed by the AHU control scheme and modulates the chilled water flow rates to maintain the desired set points. Hence, the environmental conditions in the control room will not be affected by the flow deficiencies and habitability will be maintained.

The accidents and transients of UFSAR Chapter 7.18, 10.12, and 14 were reviewed. No accidents/transients were impacted by revision of UFSAR Figure 10.12-2, since the discrepant air flows have been evaluated and were found to be acceptable as is. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 10.14 - Control and Service Air Systems - Units 1, 2, and 3**

Description/Safety Evaluation

Safety evaluation CRLD BFEP EEB-90030 RO was prepared to revise UFSAR Section 10.14.4.1. This was initiated as corrective action for CAQR BFP900367.

The CAQR documented that DCN S14032 created a discrepancy between the UFSAR and the as built facility because the setpoint for pressure switches 2-PS-85-35A1, 35A2, 35B1, -35B2, section 10.14.4.1 had not been revised.

UFSAR Section 10.14.4.1 was revised to remove the setpoint from that section since instrument information is provided in the UFSAR Table 7.2-1.

A special procedure, "Setpoint Calculations (EEB-fI-28)" and tracking system, "Implementation of the Calculation Cross Reference Information System for BFN" (BFEP PI 87-76) has been established for the purpose of performing, documenting, and tracking calculations. This establishes a more exacting, complete, and technically accurate method of performing Setpoint Calculations was used previously at BFN. These methods and procedures were used in calculating the setpoints (Calc. ED-Q2085-890159) for the subject pressure switches. Therefore, changing the setpoint of 60 psig (UFSAR 10.14.4.1) is acceptable from a nuclear safety standpoint. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 10.16 - Equipment and Floor Drainage Systems - Units 1, 2, and 3**

Description/Safety Evaluation

This update supported the UFSAR revision to downgrade the equipment and floor drainage systems from safety related, and to no longer require the operation of the Reactor Building floor drain sump pumps during a DBA. UFSAR Section 10.16, "Equipment and Floor Drainage Systems," stated that the Reactor Building floor drainage system was required in a DBA to remove normal drainage plus minor leakage which might have been caused by the accident in order to avoid water-level buildup in the areas containing the pumps of the CSCS. The UFSAR also required that the associated pumps and piping for the Reactor Building floor drainage system be designed to seismic Class I requirements, and that the power supply to the sump pumps be from independent, diesel backed, Class IE electrical sources. The seismic and electrical requirements ensure that the Reactor Building floor drainage system is available to operate during a DBA. However, several of the requirements of the UFSAR were not implemented in the system design.

There are no "qualified" means available to the reactor operator and supporting personnel for determining if a flood level exists in the Reactor Building following a DBA. However, several methods of information gathering are available. There are six flood level switches which exist in the Reactor Building basement; one in each of the four corner rooms, one in the HPCI room, and one in the Torus area. These switches actuate if water accumulates to a two inch depth in the Reactor Building basement. A second method of assessing leakage is by the operation of the Reactor Building floor drain sump pumps. The duration which the pumps are running and the increase of water in the Radwaste Floor Drain Collector Tank will be two indications of leakage rates in the Reactor Building.

Although these methods of flood detection are not qualified, they are likely to be available. The level switches have been previously determined not to be safety related. While the sump pumps and associated components are being downgraded to non-safety related, the sump pumps can be reasonably expected to be available since the normal power source is DG backed. If in fact the pumps were operational, they could be relied upon to remove several hundred gallons of water from the Reactor Building per minute; therefore, no accumulation of water would be expected to occur.

If not available, the failure of the Reactor Building floor drain sump pumps to operate will also serve as an indication that the expected water leakage in the Reactor Building would be accumulating in the basement area and appropriate actions should be initiated. Additionally, if the environment in the Reactor Building is not as severe as postulated for the DBA, personnel could be dispatched to visually assess water level in the basement.

The only safety design basis of the Reactor Building floor drainage system as evaluated by the UFSAR is to provide for the removal of normal drainage plus minor leakage which might have been caused by the accident and thus avoid water-level buildup in the areas containing the pumps of the CSCS. Safety related equipment required to mitigate accidents/transients will not be adversely affected by this UFSAR change. The flooding of this equipment is shown not to occur, even without the operation of the Reactor Building floor drainage system. Therefore, the possibility for a new malfunction was not created.

As per the discussion above, BFN minimizes the leakage that can occur in the Reactor Building during a DBA. In conjunction with minimal leakage, several means of information gathering are available which would indicate appropriate actions to be taken. The basement can contain sufficient volume to contain the minimal leakage expected to occur without any imminent damage to CSCS equipment. The potential for leakage exceeding 17 inches would only exist after some extended period following the accident, which would allow sufficient time to develop and implement a long term plan for post accident recovery, including the removal and processing of radioactive water from the Reactor Building if necessary. Based on this, the sump pumps will no longer be designated safety related in accordance with the SRP because this safety evaluation documents that no credit is taken for their operation in the mitigation of appropriate events. The sump pumps are still reliable equipment and are powered from a DG supplied source. If the sump pumps operate during a DBA, the events discussed above will be less severe since the pumps will ensure no accumulation of water in the basement, except for the HELB-OPC event.

Based on this determination, UFSAR Section 10.16 was revised to remove the required operation of the floor drainage system during a DBA. Also, UFSAR Sections 10.16 and F.7.14 were revised to remove the seismic and independent Class I electrical requirements for this function. UFSAR Section F.7.14 was moved from Section F.7 to F.6 and the listings in F.5 were revised according. Additionally, UFSAR Sections 6.6, 10.9.5 and 10.10.5 were revised to add information pertaining to leakage inspection performed each operating cycle for CSCS, RHRSW and EECW piping and components in the Reactor Building. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 10.18 - Plant Communications System - Units 1, 2, and 3**

Description/Safety Evaluation

This update of BFN UFSAR Section 10.18, Plant Communications System, involved deletion of figures and text changes. The figures deleted are copies of existing full size drawings. These drawings are very detailed and contain a considerable amount of detailed information that did not lend itself to the degree of copy reduction required for inclusion in the UFSAR. The text in this section of the UFSAR did not contain any reference to these figures neither do any other sections of the UFSAR refer to these figures. Drawings corresponding to these figures can be retrieved from the TVA Drawing Records Control Unit.

Based on the above review it was apparent that the changes to the UFSAR Section 10.18 did not impact the design or function of the affected system. Therefore, it was concluded that the deletions were safe from a nuclear safety standpoint. No unreviewed safety question was created and no Technical Specification changes resulted.

**UFSAR 11.6.3 - CCWS System Description - Units 1, 2, and 3**

Description/Safety Evaluation

This revision constituted updating the description of the CCWS system. The update of the UFSAR related to the CCWS system se..ion involved deletion and revision of text descriptions. The volume of water dispersed to individual units was properly defined. These revisions did not impact the function or performance of the affected system neither did it impact other systems ability to perform their function or performance.

The affected system did not serve a nuclear safety function, but rather a protective safety function. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 12.2 - Principal Structures and Foundations - Units 1, 2, and 3**

Description/Safety Evaluation

This change constituted an update of the reference to the applicable American Institute of Steel Construction Specification. The code of record is the 1967, 6th edition, and the change allowed use of the 1972, 8th Edition for re-evaluations and re-designs.

No equipment was added, modified, or removed from BFN as a result of the UFSAR change. However, the text changes allowed the use of a more current design input. The test changes provided consistent acceptance criteria for allowable stresses for all structural steel and miscellaneous steel and is within the current controlling margins of safety. Therefore, the change did not introduce any new failure modes or alter any existing failure modes. Based on this, no new credible failure modes are required to be postulated. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 12.3 - Shielding and Radiation Protection - Units 1, 2, and 3**

Description/Safety Evaluation

The UFSAR change involved design parameter changes to the plant shielding design. Plant shielding is designed and provided to allow personnel access to the plant areas during shutdown and normal plant operation in order to perform maintenance and carry out operational duties without exceeding occupational radiation exposure limits as set forth in 10 CFR 20 and the site Radiation Protection Plan. Shielding is also provided to reduce the radiation to certain equipment which might deteriorate under prolonged exposure to high radiation. The shielding and radiation protection design criteria consider the radiation conditions following a DBA in order to ensure that personnel can safely inhabit the control room to shutdown the reactor and control the plant following an accident. Shielding must be in place at all times in order to perform its shielding function; thus, all modes of plant operation including any DBA or ALT event is applicable to this UFSAR change.

These UFSAR changes did not adversely affect nuclear safety. Changing the shielding design parameters did not result in any shielding analysis falling below acceptable design margin nor were any estimated post-accident off-site doses or control room doses forced outside the limits specified in 10 CFR 20, 10 CFR 100, and 10 CFR 50, Appendix A, GDC-19.

Revising the UFSAR to reflect updated shielding design input parameters did not create any new failure modes, did not increase the radiation dose consequences of any accidents, or reduce the margin of safety to any personnel radiation exposure limits. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 14.10.5 - Control Room Dose Calculation - Units 1, 2, and 3**

Description/Safety Evaluation

This UFSAR change affected only the analysis of post-accident radiological effects. The change relocated control room dose analysis to a more appropriate section of the UFSAR and updated the information based on more recent dose calculations. These calculations showed that the radiological consequences of a LOCA and FHA are within acceptable limits. The section deleted from the UFSAR contained outdated analysis of the post-accident radiation effects on equipment and components. These changes in no way affect the radiological outcome of a LOCA or FHA.

This UFSAR change did not create any changes to any plant structures, systems, components, features, procedures, or instructions. It did not require any physical field work, so the plant as-constructed configuration remained unaltered. The change only affected UFSAR text concerning radiological source terms used in analyzing post-accident radiation effects to plant equipment and dose to control room personnel. This change was made to update the design basis documentation of the plant. Based on this, there were no credible equipment failure modes created by this activity. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR 14.5.1.1 and 14.5.4.3 - Generator Trip and Loss of Feedwater Flow - Units 1, 2, and 3**

Description/Safety Evaluation

CRLD BFEP-MN-91062 revised the Generator Trip and Loss of Feedwater Flow transient descriptions in the UFSAR.

The Generator Trip with Bypass Valves Failure was included (was not included previously) since it is more severe than Turbine Trip with Bypass Failure. This meets the intent of Section 14.4 for describing the more severe transients and was consistent with reload licensing submittals which present Generator Load Reject Without Bypass results. This change was only to provide a typical transient description since this transient is reanalyzed for each reload.

The Loss of Feedwater Flow transient was revised to describe the effects of lowering the MSIV water level isolation setpoint. GE provided a SE of the setpoint change in December 1982 and NRC approved the lowering of the setpoint from 470 inches above vessel zero to 378 inches in September 1984. Subsequently, this setpoint has been raised for Unit 2 to 398 inches and approved by NRC. The current approved setpoint continues to provide reduced isolations and safety relief valve challenges. This change was only to provide an updated description of this transient.

The remaining changes were editorial in nature and are not discussed. No unreviewed safety question was created and no Technical Specification change resulted.

**UFSAR - Appendix F - Unit Sharing and Interactions - Units 1, 2, and 3**

Description/Safety Evaluation

This revision of BFN UFSAR Appendix F, Unit Sharing and Interactions, constituted updates and clarifications related to the systems that serve the purpose of sharing function and interaction between Units.

These changes did not impact the function or performance of the affected systems neither did it impact on other system's ability to perform its function or performance. These UFSAR changes did not involve the addition of any new equipment or alteration of any existing equipment. The changes did not impact how the affected systems interact with other systems.

The deletion and revision of text did not result in any new or different design input. However, the changes provided clarifications and made the UFSAR consistent with other sections of the UFSAR and other design documents and surveillance instructions.

Appendix F describes the function of the shared systems and indicate whether the system penetrate primary and/or secondary containment. These changes to Appendix F of the UFSAR did not impact on any UFSAR Chapter 14 Plant Safety Analysis. Therefore, it was concluded that the changes were safe from a nuclear safety standpoint. No unreviewed safety question was created and no Technical Specification change resulted.

**W. O. #91-28977-01 - Establishing a RWCU Jumper - Unit 2**

Description/Safety Evaluation

This WO installed a jumper across temperature indicating switch, 2-TIS-69-11, to eliminate spurious isolations of the RWCU system to facilitate troubleshooting of the TIS. This temperature switch automatically isolates the system upon high temperature downstream of the nonregenerative heat exchangers to protect the demineralizer ion exchange resin from damage.

2-TE-69-10 provided main control room annunciator of high temperature (130°F) downstream of the non regenerative heat exchanger. This is the same point that 2-TIS-69-11 monitors. Operations personnel were directed by the WO to either bypass the RWCU demins or isolate the system manually upon receipt of the high temperature alarm.

After completion of troubleshooting and vessel hydro testing, the jumper was removed. This change to jumper out the high temperature RWCU isolation interlock was only in place when the reactor temperature was less than 212 degrees F. RWCU operation below this temperature will require minimal cooling from the RBCCW system. If the RWCU temperature reaches 130 degrees between the heat exchangers and the demineralizers, the operator will manually isolate the system.

This change did not affect the isolation capabilities of the RWCU system in the event of Reactor vessel low water level; therefore, did not increase the consequences of a DBA. No unreviewed safety question was created and no Technical Specification change resulted.

**SSP 1.51 - Unit 1 and 3 Restart Administration and Control - Units 1, 2, and 3**

Description/Safety Evaluation

This procedure defined the Unit 1 and 3 Restart principle organization duties, responsibilities and authorities, and its interface with the BFN Unit 2 organization.

This procedure did not create any accident initiator r failure because it did not affect the function or performance of any safety system. All maintenance, modification, operation, and work control activities will be controlled by other approved plant instruction.

This procedure did not degrade the performance or increase the challenges to safety systems assumed to function in the accident analysis. No unreviewed safety question was created and no Technical Specification change resulted.

**SI-4.2.E.1-03 - Drywell Floor Drain and Drywell Equipment Drain Calibration - Unit 2**

Description/Safety Evaluation

This Surveillance Instruction provided for the calibration of the Drywell Floor Drain and Drywell Equipment Drain sump flow integrators (2-FQ-77-6 and 2-FQ-77-16). This test verified loop accuracy and component calibration of Equipment Drain and Floor Drain discharge flow loops and ensured these systems were in compliance with Technical Specification 4.2.E. This change consisted of a general revision to bring the procedure up to standards according to PMI 2.3, Style Guide for writing instructions and PMI 2.5, Surveillance Instruction Writers Guide. This change incorporated Walk Down and Validation comments.

The components affected by this activity did not adversely impact the operation, function, or qualification of the Drywell Equipment Drain, Floor Drain Sump Pumps or the Liquid Radwaste System. The components will accurately and reliably measure the total flow of its respective sump pump. The change relating to non-safety-related components will not degrade the performance of safety-related systems assumed to function in the accident analysis nor will it increase the challenges to the safety systems.

The calibration did not involve an initiator or failure different than these considered in the UFSAR whose effects are bound by other events analyzed in the UFSAR, nor did it increase the probability of malfunction of equipment, or involve a newly discovered malfunction of equipment, previously thought incredible, to the point where it became credible. No unreviewed safety question was created and no Technical Specification change resulted.

**2-SI-4.8.A.3-04 - Auto Isolation of Radwaste Discharge Canal Valve 2-FCV-77-61 - Unit 2**

Description/Safety Evaluation

This Surveillance Instruction provided for the automatic isolation logic functional testing of Canal Discharge Valve, 2-FCV-77-61, of the Liquid Radwaste System for the initiating signals from 0-RM-90-130 with less than two CCW pumps running. This instruction along with 0-SI-4.8.A.3, 0-SI-4.2.D.1 and 0-SI-4.2.D.1Ft. satisfy entirely the requirements of Technical Specification 4.8.A.3.

The components affected by this activity did not impact the qualification, function, and operation of the Radwaste System or its associated equipment. The activity to non-safety related components did not cause the acceptance limits for any accident to be exceeded, nor does it change the margin of safety. Radioactive materials present in liquid effluents will not be increased, dilution flow rates will not be adversely affected and material concentrations in effluent streams released to the environs will remain within administrative limits set by the RETS to ensure compliance with 10 CFR 20. No unreviewed safety question was created and no Technical Specification changes resulted.

**2-EOI-1 - Emergency Operating Instructions - Unit 2**

Description/Safety Evaluation

The changes to EOI-1 consist of editorial changes and the following:

Reformatted logic statements for each step to correct human factor deficiencies. This change had no effect on any DBAs or AOTs.

Reformatted C5 to resolve referencing and branching concerns. This change had no effect on any DBAs or AOTs

Changed methods of word emphasis to be consistent and correct human factor deficiencies.

- Capitalized logic words (e.g., IF, THEN, WHEN).
- Bolded logic conjunctions (e.g., **AND**, **OR**).
- Capitalized and underlined other words for emphasis (e.g., CAN, CANNOT, NOT, ONLY, BEFORE).
- Bolded all action verbs.

Changed the requirements to reset the ADS timer to require inhibiting the timer using the keylock switches. This was done because a modification had been performed on the ADS which installed the inhibit switches. The basis for this step contained in the EPG, Revision 004, and the EOI Program Manual, Revision 002 permits this change to be made. This change may affect the intermediate line break accidents.

Deleted the allowance to line up plant control air to the drywell. The procedure only allowed use of CAD to supplement drywell control air. This change had no effect on any DBAs or AOTs.

Added a clause that allowed the procedure to be exited with all control rods not at 02 provided shutdown margin is guaranteed. These actions are allowed in EPG Revision 004 and are appropriately provided with an evaluation performed by Engineering Support personnel to ensure the reactor will remain subcritical. This condition is during ATWAS events and is outside the Design Bases.

Deleted specific valve and component manipulation sequence and wrote individual appendices for these actions to allow the user to have the procedure in hand and relieve the SRO of having to explain these operations verbally. This change had no effect on any DBAs or AOTs.

Changed requirement to maintain RPV water level between +11 inches and +54 inches to between -150 inches, and +54 inches. This was done in order to be consistent with the EPG Revision 004, and allowed increased flexibility for level control during an ATWAS event, without compromising adequate core cooling. This is a concern with ATWAS events which are outside the Design Basec.

Allow restoration of RPV level from -150 inches to +11 inches to 54 inches if the Reactor is subcritical and no boron has been injected. Directions are provided to raise level slowly. The override before the step directs the operator to step 5-3.1 if power starts to rise. This step would stop level restoration if power problems were encountered. This is a concern with ATWAS events which are outside the Design Bases.

The UO will inhibit the ADS rather than reset the timer if the operator determines that reactor water level cannot be maintained above the ADS initiation setpoint. The reason for this is that ADS actuation imposed a severe thermal transient on the reactor vessel and may complicate efforts to restore and maintain reactor water level as specified in step RC/L-2. In certain cases (e.g., HPCI/RCIC available but LPCI/LPCS injection valves closed and control power not available) ADS actuation may directly lead to loss of adequate core cooling and subsequent core damage. Further, the conditions assumed in the design of the ADS are not present (e.g., no operator action for ten minutes after event initiation) when the actions are being carried out. Finally, an operator can draw upon much more information than is available to ADS logic and can better judge when to depressurize the reactor. None of the systems will be operated, in response to an event, in any way likely to increase the probability of a malfunction. No unreviewed safety question was created and no Technical Specification change resulted.

**EOI Program Manual - Units 1, 2, and 3**

Description/Safety Evaluation

This Safety Evaluation addressed Revision 003 to the EOI Program Manual. The EOI Program Manual consists of PORC reviewed, controlled documents containing the technical source material used in the process of developing the EOIs. These documents include the PSTG, Appendices A, B, C, and D to the PSTG, and the Deviations Cross Reference.

The PSTG is developed from the BWROG EPG Revision 003. Revision 004 of the EPG also was used in developing the PSTG. Both revisions have been reviewed by the NRC and approved for implementation. Although portions of Revision 004 have been used, Revision 003 is the primary source for the PSTG and the deviations from the generic guidance are compared to Revision 003. The PSTG was developed from the generic EPG and provided the specific actions, cautions, entry conditions, and action limits or controlling reactor parameters such as Drywell temperature, Primary Containment pressure, Suppression Pool level and temperature, Secondary Containment levels, temperatures, and radiation levels, and radiation release rates in order to mitigate an entire spectrum of events, including less than Design Basis, Design Basis, and beyond Design Basis accidents.

This Safety Evaluation addressed changes to the EOI Program Manual which provided the technical information necessary to prepare symptom based EOIs. These changes did not affect the Radwaste system, nor constitute a special test or experiment.

These changes did not change any system design or functional requirements, nor change any text, tables, graphs, or figures presented in the UFSAR.

The changes listed as item 9 and item 10 in PC/P-6 of the EOI Program Manual that make the lower pressure limit 30 psig, and require emergency venting before design pressure limits exceed 55 psig differed from the discussion of Primary Containment venting contained in the UFSAR.

Items 9 and 10 were concerned with venting the Primary Containment following an accident or transient. This is a concern only for Design Basis LOCA events that result in stable pressures greater than those analyzed in the UFSAR. Item 9 gave guidelines to reduce the Primary Containment pressure to 30 psig when pressure reaches 55 psig by venting using the 18" line. Item 10 gave guidance for emergency venting of the Primary Containment before exceeding the design pressure limit. Stable pressures above 30 psig are indicative of multiple equipment failures such as containment sprays and the 2 inch hardened vent system (CAD). These concerns are not addressed in the UFSAR, as analysis of the Design Basis accidents do not show containment pressure to reach these values. For those scenarios beyond the Design Basis envelope, the EOI Program Manual is consistent with Revision 003 and Revision 004 of the EPG's. No unreviewed safety question was created and no Technical Specification change resulted.

**Lost Article 2-90-2 - Ball of Tape - Unit 2**

Description/Safety Evaluation

This safety evaluation addressed the possible effects of the lost article documented on form PMI-63, Lost or Unsecured Article Recovery form, for lost article number 2-90-2. This article was initially described as a shiny object about the size of a golf ball, which was later determined to be a ball of tape, and was located at about 45 degrees on the core spray header in the vessel cavity. Duct tape has a silvery appearance and is commonly used on the refueling floor on tools, poles, etc., which may then be immersed in the SFSP and vessel cavity area as needed for underwater maintenance work. It is not known when this particular object was dropped into the vessel, nor what piece of equipment it had been attached to. The article was initially identified by an NRC-licensed SRO who was supervising core unload work at the time of the discovery. The article was discovered on January 7, 1990, but could not be recovered at the time. On September 10, 1990 this area of the vessel was searched by Operations and Technical Support personnel from the refueling bridge, and the object could not be located. Operations personnel searched the area again on September 13, 1990, and after no object was found the tape ball was declared a lost article in the reactor vessel.

The loss of tape into the Unit 2 vessel is of concern because it is possible, in theory, for the tape to be swept by coolant flow up to a fuel support orifice or to a lower fuel tie plate nosepiece. In such a position, the tape might block flow to the fuel bundle such that the bundle is damaged due to the onset of transition boiling brought on by inadequate coolant flow. However, tests have shown that a piece of duct tape is unable to survive the high temperature (545° F) environment which is existent at rated conditions for BFN Unit 2. Autoclave tests show that the material of which duct tape is made has little mechanical strength in boiling water after exposure to simulated reactor conditions. The residue from such tests is easily broken up into small fragments upon agitation as the primary reinforcing textile (cotton scrim) is dissolved by hydrolysis at the high temperatures typical of a reactor environment. Therefore, it is not likely that a piece of tape could block coolant flow through a bundle at operating conditions. GE studies have shown that during the heatup to rated conditions individual bundle powers would be low enough that transition boiling would be avoided even with complete blockage at the fuel support piece orifice or the lower tie plate nosepiece. Additionally, since tape deteriorates at reactor rated operating temperatures it need not be considered in any analysis of the cumulative effects of all lost articles in the vessel. Therefore, it is assured that no fuel damage would result from inadequate cooling during heatup prior to the deterioration of the duct tape.

Concerns also exist with lost articles about any possible adverse effect they may have on the reactor water chemistry. The list of chemicals which may be found in duct tape includes some elements which are undesirable to have in the reactor coolant. However, in this case the relatively small size of this piece of tape compared with the large volume of water in the vessel available for dilution of the constituent materials of the tape make the effects on reactor water chemistry negligible. Any change in water chemistry induced by the deterioration of this piece of tape would be easily within the cleanup capacity of the RWCU system filter/demineralizers. Therefore, it is not expected that this piece of tape will have any adverse impact on the reactor water chemistry.

Operation of BFN Unit 2 with this lost article is acceptable from a nuclear safety standpoint. No unreviewed safety question was created and no Technical Specification change resulted.

**Lost Article 2-91-1 - 1/4" Nylon Rope " Long - Unit 2**

Description/Safety Evaluation

This safety evaluation addressed the possible effects of the lost article documented on form PMI-63, Lost or Unsecured Article Recovery form, for lost article number 2-91-1. This article was described as a section of quarter-inch nylon rope about 6 to 12 inches in length, and was located on the vessel surveillance sample holder support in the vessel cavity. Nylon rope is commonly used on the refueling floor to secure items to the pool rails, and on tools, poles, etc., which may then be immersed in the SFSP and vessel cavity area as needed for underwater maintenance work. It is not known when this particular object was dropped into the vessel, nor what piece of equipment it had been attached to. The article was initially identified by an inspection of jet pumps and other vessel internals and is documented on a videotape record of this inspection. The article was discovered in December, 1990, but no recovery was attempted at the time. Due to the depth below the water surface of the lost article, the material involved (nylon rope, which tends to fray and come apart after being immersed in water for a long time period), and the confined area surrounding it, it is not likely that the item can be recovered.

The loss of nylon rope section into the Unit 2 vessel is of concern because it is possible, in theory, for the rope to be swept by coolant flow up to a fuel support orifice or to a lower fuel tie plate nosepiece. In such a position, the rope might block flow to the fuel bundle such that the bundle is damaged due to the onset of transition boiling brought on by inadequate coolant flow. However, experience has shown that nylon rope material tends to degrade when immersed in water for an extended period of time such that it retains no structural strength. Furthermore, at the high temperature (545 ° F) environment which is existent at rated conditions for BFN Unit 2 degradation of the rope material would be accelerated. The constituent materials of nylon rope will soon disintegrate after exposure to reactor conditions. The residue from such material is easily broken up into tiny fragments when subjected to the smallest amounts of stress. Since the initial diameter of the rope is smaller than the flow area between individual fuel pins within a bundle, it is not likely that a piece of rope could block coolant flow through a bundle at operating conditions. GE studies have shown that for boiling transition to occur, 79 percent area blockage is required at the orifice and 86 percent area blockage at the lower tie plate is required. The small initial size of the lost article compared to the size of the flow areas involved makes this amount of blockage impossible even before the rope disintegrates. Additionally, since rope deteriorates when immersed in water, especially at elevated temperatures, it need not be considered in the vessel. Therefore, it is assured that no fuel damage would result from inadequate cooling induced by flow blockage caused by this rope.

Concerns also exist with lost articles about any possible adverse effect they may have on the reactor water chemistry. The chemicals which may be found in nylon rope includes some elements which are undesirable to have in the reactor coolant. However, in this case the relatively small size of this piece of rope compared with the large volume of water in the vessel available for dilution of the constituent materials of the rope make the effects on reactor water chemistry negligible. Any change in water chemistry induced by the deterioration of this piece of rope would be easily within the cleanup capacity of the RWCU system filter/demineralizers. Therefore, it is not expected that this piece of rope will have any adverse impact on the reactor water chemistry.

Operation of BFN Unit 2 with this lost article is acceptable from a nuclear safety standpoint. No unreviewed safety question was created and no Technical Specification change resulted.

**W.O. 91-27559-00 - Disabling RCIC Overspeed Trip Circuitry - Unit 2**

Description/Safety Evaluation

During performance of RCIC System flow testing, it was discovered that the RCIC stop valve trip solenoid is continuously energized making it impossible to perform the required testing. It is believed that the electrical overspeed trip circuitry is spuriously energizing the trip solenoid. The subject work order removed the electrical overspeed trip function by lifting wires which enable the circuit. The electrical overspeed trip function is presently pending removal from the plant design via DCN number 3441.

The activity removed one of two RCIC overspeed trip devices. The removal of the electric overspeed trip device did not cause a malfunction of a different type as evaluated in the UFSAR. RCIC turbine overspeed trip protection will be achieved by the mechanical overspeed trip device that was tested satisfactorily. No unreviewed safety question was created and no Technical Specification change resulted.

**CAD Storage Capacity - Units 1, 2, 3**

Description/Safety Evaluation

On October 16, 1991, a vacuum leak was discovered in the CAD system nitrogen storage tank B. The tank is a cyrogenic tank with vacuum between an inner and outer shell serving as the insulating medium. The vacuum leak causes the nitrogen to boil off at a higher rate than usual necessitating more frequent refills. A temporary deviation from the UFSAR is being taken to reduce the site's required CAD nitrogen storage capacity from seven days for post-accident operation to three days. This change will serve as a basis for maintaining OPERABLE status of the CAD system considering the need to compensate for reduced nitrogen storage capacity due to the increased boil off rate. This change will be effective until the tank is either repaired or replaced.

Section 5.2.6.1.f of the UFSAR requires a seven day CAD nitrogen storage capacity. This will temporarily be reduced to three days.

Changing the site storage capacity of the CAD nitrogen system to provide for three days of post-accident CAD operation instead of seven days was acceptable on the following basis:

The CAD tanks are designed with enough capacity in each tank to store the amount of nitrogen required (2260 gal.) to maintain the drywell and torus of one unit below 5% oxygen by volume during the seven days after a LOCA assuming the oxygen generation rates given in AEC Safety Guide 7 plus tank boil off losses which would be expected to occur. Maintaining 2500 gal. in each tank per Technical Specifications (TS) ensures that this requirement is met. The degraded vacuum insulation on CAD nitrogen tank B affects its ability to store liquid nitrogen over time because it results in greater ambient heat input to the tank resulting in a greater boil off rate.

The CAD tanks can each perform their safety function as long as 2260 gal. is available for use between tank refills. Therefore, the only affect that this change has is to reduce the time allowed between tank refills in a post-accident situation from seven days to three days. The current seven day requirement does not have a technical basis, but appears to have been chosen as a practical timeframe for receiving shipments of consumables (such as fuel oil, for example) after an accident.

Section 5.2.6.2 of the UFSAR identifies three local suppliers of nitrogen all of whom are within one day travel distance of Browns Ferry Nuclear Plant. Experience with routine nitrogen delivery has shown that obtaining a shipment within three days can be assured, particularly when considering the resource allocations and coordination that would be established by the radiological emergency plan during a radiological emergency. It is concluded that CAD nitrogen replenishment is assured in three days and that reducing the minimum CAD storage capacity to three days supply will not decrease nuclear safety.

The Technical Specifications do not address delivery of CAD replenishment nitrogen. The TS and its BASES only specifies the minimum amount of nitrogen to be kept onsite (2500 gal.) and this amount is not being changed by this UFSAR change. Since TS operability of the CAD system is a function of the amount of nitrogen stored and not the rate of usage, three days is adequate time to allow for delivery of replenishment nitrogen without threatening operability of the CAD system. Therefore, this UFSAR change will have no impact on the TS.

This activity involved changes to the plant documentation only. No physical change was made to any plant structure, system, equipment or component. The CAD system is used to mitigate a LOCA and in no way can cause an accident to occur. Nor can the CAD system cause a fire. The CAD system provides the pneumatic medium for MSRV operation during safe shutdown operations following a fire event. Based on this, there is no possibility of affecting the probability of occurrence of any accident or fire as previously evaluated in the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**2-ST-91-01 - Turbine - Generator Torsional Response Testing - Unit 2**

Description/Safety Evaluation

The purpose of this procedure was to provide instructions for performance of a torsional response test on the Unit 2 turbine-generator. This test was performed in response to General Electric Technical Information Letter 1012-2, "Effects of Electrical System Variations on Turbine-Generator Torsional Response." The turbine EHC System and Excitation System was altered for this test in order to provide additional speed and excitation control. These instrumentation were set up in a turbine test center located on the turbine floor and monitored turbine torsional response through sensors located at the #4, #6 and #8 bearings. Turbine speed and shaft position were monitored through detectors mounted at the turbine front standard. The generator output was instrumented to provide voltage and current data during the performance of the test. This data was recorded during this test and was analyzed to determine if a resonance problem exists within the turbine.

The turbine torsional test consisted of two parts which were performed prior to the normal synchronization of the turbine-generator. Part 1 was the single phase generator torsional excitation-acceleration test and was performed with the unit isolated from the utility power system. Most of the generator protective relaying was removed to allow for testing under these abnormal conditions with all parameters being monitored by operations personnel. Negative sequence current was generated by grounding the output of the A phase transformer and providing a controlled auxiliary excitation to the generator while operating the turbine over a speed range of 100 to 1940 RPM. Part 2 required the EHC and Excitation systems be returned to normal and the transformer ground lifted. A generator synchronization was performed per approved plant instructions to complete the data collection. The plant was restored to normal electrical configuration at the conclusion of the test and the unit available for operation.

This procedure was developed to test turbine torsional responses at various speeds. Reactor power was controlled at less than 25 percent and the turbine was operated at virtually no loads. Should a turbine trip occur all main steam will be bypassed to the condenser as designed without a reactor SCRAM. The UFSAR Section 14.5.1.3 provided turbine trip analysis at design power output. This test was conducted at a low power level and was acceptable from a nuclear safety standpoint. Additionally, the turbine-generator is non-safety related and provides no safety function.

Because performance of this test required that potentially damaging torsional vibrations be induced on the turbines, the potential for a turbine failure resulting in the ejection of external missiles did exist. BFN had been analyzed to withstand such a failure as discussed in UFSAR Section 11.2.2.

SAFETY EVALUATION FOR
SPECIAL TESTS

The evaluations performed in support of this analysis were presented in the responses to questions B-4-1 of Amendment 3 and C-8-1 of Amendment 6 to the BFN Units 1 and 2 "Design and Analysis Report". These evaluations concluded that:

"In consideration of the low probability of turbine-generator failures which could generate missiles, the low ratio of potential critical target area to total target area, the massive concrete structures which surround the primary containment and other critical equipment, it is concluded that there is no significant probability of missile damage from main turbine-generator failure that could lead to a hazardous release of fission products. (B-4-1 of Amendment 3)

All failures other than a failure resulting in an external missile are bounded by the evaluation for external missiles. As shown by this evaluation, such a failure will not result in a designed basis accident as defined in Section 14.6 of the UFSAR.

(1) Generator Trip

Because this test was performed below the 25 percent power level, a generator trip would result in a mild reactor transient as the turbine control valves close and the bypass valves open. Unless bypass valve failures result in the inability to vent steam to the condensers, no reactor scram will occur.

(2) Loss of Condenser Vacuum

In the event of a loss of condenser vacuum due to damage incurred from an ejection of external missiles from the turbine, a turbine trip without bypass valves would occur. This would result in the relief valves lifting to limit the reactor pressure rise and sequentially reclosing as the stored energy is dissipated.

(3) Turbine Trip

As with the generator trip described above, with reactor power below 25 percent the bypass valves will open and minimize the affect of the reactor transient, with a high probability that a reactor scram will be avoided.

(4) Bypass Valves Failure Following Turbine Trip, Low Power

When the bypass valves fail to relieve reactor pressure, a scram will be initiated and the relief valves will open to relieve the pressure transient.

BFN has been analyzed for the above transients as documented in Section 14.5.1 of the UFSAR.

This test did not introduce any failure modes that were not already evaluated in UFSAR Sections 11.2.2 and 14.0. No unreviewed safety question was created and no Technical Specification change resulted.

**3-ST-91-02 - Fuel Inspection - Unit 3**

Description/Safety Evaluation

The purpose of this test was to improve the reliability of the fuel by identifying Unit 3 fuel rods that meet acceptable corrosion criteria needed for continued operation. When unacceptable fuel bundles were identified they were recorded as such and administratively prevented from future use at BFN. This will reduce the expected number of fuel failures next cycle that will reduce plant radiation levels and thus increase plant safety. As discussed later, a fuel bundle or individual fuel rod may be damage during this test but the consequences will be bounded by the design basis fuel handling accident per UFSAR Section 14.6.4.

The activity involved the inspection of the Unit 3 cycle 6 fuel bundles. This test involved the disassembly of fuel bundles and the inspection of selected individual fuel rods to determine their acceptability for use. This test was performed in the Unit 3 SFSP.

If fuel rods are bent and cannot be reinserted into the fuel bundle, they will be stored in a GE storage can, a GE defective fuel storage container, or stored on the floor of the SFSP. Broken fuel rods may be temporarily stored in a GE temporary storage can until they can be moved to the failed fuel rod canister or to a GE defective fuel storage container. The GE defective fuel storage containers will be stored in the defective fuel storage rings located on the periphery of the SFSP racks. The fuel canisters will be stored either in the SFSP racks or in a GE defective fuel storage container. The movement of a GE defective fuel storage container that contains a failed fuel rod canister or a fuel bundle was prohibited during this test. Loose fuel pellets and debris was recovered and stored in stainless steel buckets in the SFSP.

During a fuel inspection, no plant configurations were introduced that were not described in the UFSAR accident consideration for fuel handling activities. The criticality analysis for the spent fuel storage racks in the fuel pool assumes the full storage with fuel (FSAR Section 10.3.5). The design of the spent fuel storage provides for a $K_{eff} \leq 0.95$ for both normal and abnormal storage conditions. The placement of the fuel bundles into the FPMs is bounded by UFSAR criticality input assumptions. All required load handling (fuel and non-fuel) during the inspection operations were accomplished through existing plant procedures. Therefore, no new accident potential was introduced by the installation or moving of fuel to the fuel inspection equipment. The UFSAR Fuel Handling Accident event analysis has considered the failure modes for fuel servicing equipment and subsequent accident concerns.

A dropped bundle accident bounds a dropped or broken fuel rod incident. The design basis dropped bundle assumes that at least 125 fuel rods are damaged, where as at a minimum only two rods could be damaged at the same time if one bundle is under inspection. Normally only one rod would be expected to be damaged or dropped based on GE experience. Dropping an individual fuel rod according to the definition of DBA in UFSAR Section 1.2.16, is the same type of accident (but less serious) as the design basis fuel handling accident of dropping a fuel bundle as analyzed in UFSAR Section 14.6.4. No unreviewed safety question was created and no Technical Specification change resulted.

**3-ST-91-03 - Fuel Inspection/Sipping for Unit 3 - Unit 3**

Description/Safety Evaluation

Fuel handling and inspection equipment was used for the normal activities for which they were designed. The fuel sipping process was conducted using routine fuel bundle handling procedures and the generic GE fuel sipping procedure. This special test did not create an increase in the probability of a fuel handling accident, because there was no change in the design or use of the fuel handling equipment.

The DBA for fuel handling is dropping one fuel bundle onto the top of the spent fuel pool storage racks as analyzed in section 14.6.4 of the UFSAR. A fuel bundle weights approximately 670 pounds and is assumed to fail all 62 fuel rods on impact. The total number of failed rods resulting from the accident is 125. The heaviest piece of GE equipment that could be dropped into the SFSP is a sipping canister that weights less than 250 pounds. Based on this weight, a dropped fuel bundle is the worst accident that could occur during this test. A dropped fuel bundle could fail the same number of fuel rods assumed in the fuel handling DBA but the fission product release and radiological effects would be considerably less due to the approximately 6.0 years of radioactive decay since shutdown. Most of the major contributors to gaseous radioactive releases, the noble gases and iodines, have essentially decayed away due to their short half lives. The only contributor of concern is the KR-85 and any that might be released should be detected and handled by site RADCON. No unreviewed safety question was created and no Technical Specification change resulted.

**3-ST-91-04 - Trending SFSP Heatup Rate - Unit 3**

Description/Safety Evaluation

The purpose of Special Test 3-ST-91-04 was to trend water heatup rate and monitor chemistry when Fuel Pool Cooling and Cleanup System was not in operation.

SAFETY EVALUATION FOR
SPECIAL TESTS

The BFN Unit 3 reactor has been shutdown since September 1985. During recent Unit-2 Fuel Inspection, Spent Fuel Pool Cooling and Cleanup System was in operation to remove the decay heat and maintain water clarity. The loose corrosion products and the crud burst debris migrated and deposited on the heat exchangers during Spent Fuel Pool Cooling and Cleanup System (FPC&CS) Operation. Thus causing the heat exchangers, pumps and other system components to be classified as high radiation equipment. To prevent similar migration of loose corrosion products, and crud burst, it was decided to secure the Unit 3 Fuel Pool Cooling and Cleanup System and monitor the Fuel Pool heatup rate during Unit 3 Fuel Inspection. Conservative administrative controls in the special test restores the FPC&CS to operation based on the heatup rate and the water chemistry analysis. The data collected during the heatup study may be utilized in future Fuel Inspection Operations at BFN.

This test caused the Unit 3 FPC&CS to be shutdown and therefore required temperature and chemistry monitoring per Technical Specification 3.10.C/4.10.C. The duration of this Special Test was at the discretion of the Test Director, not exceeding one (1) week.

The only design basis accident remotely applicable to this test was the fuel handling accident or the dropping a fuel bundle or other heavy weight onto the top of the spent fuel pool storage racks. This test did not handle or require the moving of weights over the spent fuel racks. Therefore, there was no applicable design basis accident for this special test.

The FPC&CS was taken out and returned to service using existing plant procedures. It was not expected that any component of the FPC&CS system would fail or malfunction due to the activities of the special test. In the event that the system could not be returned to service due to systematic problems 3-4 days would be available before the 150°F Technical Specification limit is approached. This was based on the heatup rate of 5-6°F a day, as seen during the Unit 2 FPC&CS shutdown, and the 120°F admin limit for stopping the test.

The SFSP water conditions was monitored carefully and was not allowed to exceed administrative limits which were below the Technical Specification temperature and chemistry limits. When one of these limits was approached, the FPC&CS was returned to service and the pool conditions returned to an acceptable value before resumption of the test. No unreviewed safety question was created and no Technical Specification change resulted.

**TACF 02-91-1-303 - Temporary Building Inside Secondary Containment - Unit 2**

Description/Safety Evaluation

A temporary building was erected for storage of equipment required to perform the ILRT. It was required to be climatically controlled for care of computers prior to and during the testing. During the testing, it was used as a controlled test station. The temporary building was constructed of metal panels which were attached to each other by 5/16 inch bolts located 5-1/2 inches on center. At the corners and top, the panels were attached together with 2 x 2 x 3/16 inch angle with 5/16 inch bolts located 5-1/2 inches on center.

The building was located in the southwest corner of the Unit 2 Reactor Building between R8 and R9 and T & U-lines on elevation 593. The building was attached to the floor slab by two 1/2 inch SSD anchors located at the approximate 1/3 points of each panel. After completion, there was two (2) 3/8 inch cables in the north-south and the east-west directions placed across the top of the building at approximately the 1/3 points were attached to the concrete walls with 1/2 inch SSD anchors and eyebolts. The other end of the cables were attached to 1/2 inch eyebolts located on the top angles of the building. Also, there was a frame built between the building and the U-line wall for protection of a conduit which runs to the RHR 2A and 2C pump room cooler motors.

The erection of this temporary building did not cause the malfunction of any equipment because of the location, which is on elevation 593 in the Unit 2 Reactor building between R8 and R9 and T & U-line. The only features located in this area were a fire hose station, a two inch conduit which runs to the RHR 2A and 2C pump room cooler motors and the MG oil drain. The building was erected in such a manner that it did not affect these features nor did it affect any other equipment. No unreviewed safety question was created and no Technical Specification change resulted.

**TACF 3-91-1-303 R1 - Personnel Access Barriers Between Units 2 and 3 Reactor Zones - Units 2 and 3**

Description/Safety Evaluation

Temporary personnel access barriers were provided at doors providing access/egress between the Unit 2 and Unit 3 reactor zones. The barriers were required to control personnel movement and access across the new secondary containment boundary which isolates the Unit 3 reactor zone from the remainder of the reactor building during the Unit 3 recovery effort.

The barriers were of modular construction, consisting of a welded structural steel frame and expanded steel mesh fence 8 foot - 5 inch high (maximum). The barriers were generic in design and were installed on both the Unit 2 and Unit 3 sides of the floors common to the Unit 2 and Unit 3 reactor zones and the Control Bay along column line R14/R15 at elevations 565.0 foot, 593.0 foot, 621.25 foot, and 639.0 foot.

These barriers do not affect the integrity or operability of the Secondary Containment system. Therefore, this TACF did not introduce any credible failure modes associated with the capability of the secondary containment system to mitigate the design basis events identified above. No unreviewed safety question was created and no Technical Specification change resulted.

**TACF 2-91-2-3 - Plugging a Drain Line to 2A RWFP - Unit 2**

Description/Safety Evaluation

A threaded pipe plug was used to temporarily plug the existing drain of the 2A RFWP casing. This method of temporarily plugging the drain line until a permanent solution is achieved was verbally approved by the manufacturer. Additionally, the pressure and structural integrity of the pump casing was maintained.

The activity had no effect on any accident previously evaluated in the UFSAR. The RWFP is not a component used to mitigate accidents. The activity as described by this TACF had no effect on any DBAs and AOTs. No unreviewed safety question was created and no Technical Specification change resulted.

**TACF 2-91-3-73 - HPCI Time Delay Relay Installation - Unit 2**

Description/Safety Evaluation

This change installed time delay relays in the HPCI low suction pump trip circuit. The function of this trip was to protect the HPCI pump from damage caused by extended periods of cavitation. Installation of a time delay in this circuit did not impede the initiation of the HPCI pump and allowed greater pump availability by enhancing its ability to continue to operate during short term negative suction pressure transients incurred after pump initiation.

Installation of the time delay into the HPCI low suction pressure pump trip circuitry did not affect HPCI functions or ability to operate. The failure modes associated with the installation of time delay relays in the pump trip circuit were the same as those for the existing trip circuitry (i.e., failure of the pressure transmitter 2-PT-73-29-1 or failure of the analog trip unit). These failure modes are a failure to send a trip signal to the HPCI pump turbine control or to send a spurious signal to the HPCI pump turbine controls. As a result, HPCI's ability to mitigate the consequences of any DBAs or AOTs remained unaffected. Therefore, the failure modes associated with this change were bounded by the existing failure modes of the pump trip circuit. No unreviewed safety question was created and no Technical Specification change resulted.

**TACF 2-91-4-2 - Mechanical Positioner for 2-FCV-2-190 - Unit 2**

Description/Safety Evaluation

This temporary alteration placed a mechanical positioner on 2-FCV-2-190 which is normally positioned by a pneumatic positioner. This provided Operations the capability of manually positioning 2-FCV-2-190.

2-FC-2-190 normally functions to position 2-FCV-2-190 to maintain a differential pressure across the steam packing exhauster. This device will provide a means for Operations to manually position 2-FCV-2-190 with 2-FC-2-190 out-of-service. Since the condensate system does not perform a safety function, this activity did not decrease nuclear safety.

On a reactor scram or any load reductions, 2-FCV-2-190 would normally throttle in the closed direction to maintain the differential pressure across the steam packing exhauster within limits. Installation of this mechanical device would defeat that automatic throttling capability and give 2-FCV-2-190 a failure mode of "as-is" on any load changes until Operations responded and manually repositioned the valve.

With a failure mode of "as-is", 2-FCV-2-190 would be unable to respond to system flow changes. This inability to respond would result in reducing the cooling water available to the steam packing exhauster, steam jet air ejector and the off-gas condenser. The lack of cooling water to the steam jet air ejectors and off-gas condenser introduces the possibility of losing the condenser as an available heat sink.

However, a loss of condenser vacuum has been recognized and analyzed as an abnormal operational transient in Chapter 14 of the UFSAR. Since this transient has been analyzed and found to be safe from a nuclear safety standpoint, this activity did not reduce nuclear safety. No unreviewed safety question was created and no Technical Specification change resulted.

**BF2 Cycle 6 Cycle Management Report - Revision 2, BCD 446 - Unit 2**

Description/Safety Evaluation

The Browns Ferry Unit 2 Cycle 6 reload core design licensing analyses are documented in the Reload Licensing Report and have been reviewed and accepted by the NRC. The Cycle Management Report specified the final loading pattern for cycle 6, including the results of the 1988 Fuel Inspection and Reconstitution Program. The safety evaluation concluded that the final loading pattern, including the use of reconstituted fuel assemblies, involves no unreviewed safety question.

This change revised the final loading pattern to increase flexibility to operate the cycle longer within fuel exposure licensing limits. The revised loading pattern used the same fuel assemblies as the previous loading pattern, but changed the locations of six of the high exposure assemblies.

In addition, conservatisms to account for potential channel bow effects on MCPR and an improved modeling of the effect of the long shutdown on core reactivity were added to the cycle management analyses. Although these changes will impact the data presented in the Cycle Management Report, the results of the licensing analyses (and thus the licensing basis) was not affected.

The proposed activity did not create a possibility for an accident of a different type than any evaluated previously in the UFSAR. The mechanical, neutronic, and thermalhydraulic characteristics of the revised core loading pattern have been reviewed, and are bound by the current UFSAR. No Technical Specification change resulted.

**ECN P0007 R0 - Secondary Alarm Station - Units 1, 2, and 3**

Description/Safety Evaluation

ECN P0007 relocated the SAS and associated security equipment from the Unit 1 and 2 control room to the Unit 3 control room to facilitate ongoing modifications to the Unit 1 and 2 control room. Also addressed, the utilization of 120 VAC fed from existing non-Class 1E lighting cabinet LC308, for relocated SAS security intrusion detection cabinets 4 through 10.

In addition to the above modifications, the closed circuit television and the security computer were relocated to the Unit 3 control room utilizing new and existing, non-Class 1E conduit and raceways. Review of Sections 14.5 and 14.6 of the BFN UFSAR indicated that this modification had no effect on the listed DBA and AOT. No unreviewed safety question was created and no Technical Specification change resulted.

**ECN P0286 RO - Security Continuous Power Supply - Units 1, 2, and 3**

Description/Safety Evaluation

ECN P0286 provided an uninterruptible yard security lighting system, including lighting fixtures, diesel generator, and the housing structure for the diesel located inside the protected area. The security diesel generator supplied power for certain communications and miscellaneous security loads in the central alarm station and SAS as delineated in the PSP. Specifically, ECN P0286 provided an uninterruptable yard lighting system that will maintain a horizontal light level of 0.2 foot candles throughout the protected including 20 feet beyond the perimeter fence. ECN P0286 also modified the plant security computer, including its associated power supply, to bring it into compliance with the PSP.

Modification of the yard lighting system and installation of a backup source of onsite power brings BFN into compliance with 10 CFR 73.55 as delineated in the PSP. The security system is not an initiator of any postulated accident. Review of Sections 14.5 and 14.6 of the BFN UFSAR indicated that this modification had no effect on the listed DBAs and AOTs. No unreviewed safety question was created and no Technical Specification change resulted.

**ECN P0314 RO - Pass Building Construction - Unit 2**

Description/Safety Evaluation

This ECN installed a building inside the turbine building and supplied Service Air to this new building. This change did not adversely affect the function of the Service Air system. ECN P0314 documented the addition of service air piping in a new building inside the turbine building. The Service Air system, which includes the additional piping and connections inside the new building, accomplishes the same function as prior to the modification. The Service Air System provides backup control air through a check valve and a backpressure control valve which opens if control air pressure drops below a certain setpoint. Thus, the Control and Service Air systems are normally separate, with the Service Air system acting as a backup to the Control Air system. The most credible failure mode for the new Service Air piping and connections would be pipe failure. This event is bounded by the credible failure modes of the existing Service Air system. Therefore, the new equipment failure modes are enveloped by the failure modes of the existing Service Air system and do not adversely impact nuclear safety. Furthermore, the design basis accidents and anticipated operational transients described in Sections 14.5 and 14.6 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DD 1-88-0369 - Drawing Corrections, Drawing 1-47E610-32-1 and 2-47E812-1 - Units 1 and 2**

Description/Safety Evaluation

Drawing 1-47E610-32-1 was revised to add Unit designator "0-" to components 0-FSV-33-1, 0-PA-33-1, and 0-PCV-33-1 to clarify that these components are common for all three Units in accordance with DD 1-88-0369.

Also, a typographical error which denotes 0-PCV-33-1 as 0-FCV-33-1 was corrected on drawing 1-47E610-32-1. The Instrument Tabs for system 32 presently show the affected valve as 0-PCV-33-1. Drawing revision 002 of 1-47E610-32-1 shows the component labeled as "PCV" and a review of subsequent revisions of drawing 1-47E610-32-1 shows that no physical modifications were performed on this valve.

Drawing 2-47E812-1 was revised to add a continuation flag to the condensate supply system. This change was an Exception A Change Control per NEP-6.1.

The documentation change was to resolve DDs with the installed configuration. HPCI, Control Air and HVAC system function and operation remained unchanged. Therefore, the design basis accidents and operational transients described in Chapter 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**ECN P0407 R0 - Construction of New West Gatehouse - Unit Common**

Description/Safety Evaluation

This ECN provided for construction of a building for the plant access control portal of the security system that meets the requirements of 10 CFR 55. The facility is known as the West Gatehouse. It is located approximately 450 feet from the nearest safety related seismic structure. The only function of this facility is to control access to the plant.

This modification involves fabrication of the gatehouse superstructure, installing conduit, non-Class 1E cables and potable water. ECN P0407 also disconnects and caps off the 2" piping for the yard lawn sprinkler piping to facilitate the installation of the West Gatehouse. The downstream piping is left as is. All involved piping is buried in the ground. Review of Sections 14.5 and 14.6 of the BFN UFSAR indicates that this modification would have no effect on DBAs or AOTs associated with the construction of a building to control access to the protected area or the disconnection of the yard lawn sprinkler from the Raw Service Water piping. No unreviewed safety question was created and no Technical Specification change resulted.

**ECN P0418 - Addition of Control Air Pressure Switches - Units 1, 2, and 3**

Description/Safety Evaluation

ECN P0418 provided for the addition of pressure switches to monitor the control air system pressure at the pressure control station from the CRD System. An output signal from the pressure switches will interface with the RPS to initiate reactor scram when air pressure drops to the set point of the switches.

The modifications associated with ECN P0418 were made in response to IE Bulletin 80-17, Supplement 3 which outlines the actions required by NRC to be taken at BFN to assure safe operation of the scram function. NRC requirements were initiated by a failure of the contr l air system during a manual scram at BFN Unit 3 on June 28, 1980. It was noted that sustained low pressure in the control air system could result in complete or partial opening of multiple scram outlet valves before opening of the scram inlet valves. This could cause the Scram Discharge Volume to fill rapidly, thus leaving a relatively short time for the operator to take corrective action before scram capability is lost. This modification provides a continuous monitoring system which will automatically scram the reactor if control air header pressure drops too low.

The low pressure trip system added per ECN P0418 is an independent, Class 1E, Seismic class I system connected to the RPS but is not considered part of the RPS.

The normal operating pressure of the control air system is 90 psig. This pressure is sufficient to hold the SCRAM valves closed during normal reactor operation. The SCRAM valves begin to unseat when the pressure on the diaphragm is between 40 to 50 psig. Below the unseating pressure, valve position is proportional to the air pressure on the diaphragm. Prior to ECN P0418 should the Operator receive a low control air pressure alarm (alarm was set at 60 psig) his operating instructions were to initiate a SCRAM of the Unit. Upon issuance of ECN P0418, any failure of the control air system which results in the system pressure dropping to 53 psig will cause an automatic SCRAM. No unreviewed safety question resulted. A Technical Specification change related to this ECN was approved.

**DCN W0524A - Fuel Preparation Machine Upgrade - Units 1, 2, and 3**

Description/Safety Evaluation

This DCN W0524A added an oiler to the service air line that feeds the air hoist motor of the fuel preparation machines. This was added in order to increase the reliability of the air hoist motors. A short stroke shut off valve replaced the existing valve to provide a quick response emergency shut off of the service air to the fuel preparation machines. The oil used was based on recommendations from GE and related manufacturer data which ensured improved motor performance without adverse impact on the service air and fuel pool related systems. To further assure no impact to water chemistry by introduction of oil into the fuel pool, an oil-separating coalescing filter was fitted on the motor's air exhaust line, such that virtually all oil introduced to the air inlet will be subsequently extracted.

The scope of this change included the air hoist motors of the two fuel preparation machines, for each Unit 1, 2, and 3. These machines are used to remove channels from spent fuel assemblies and to install used channels on new fuel assemblies located in the fuel storage pool, and are non-safety related equipment.

This DCN affected Figure 10.14-2b in the UFSAR by the addition of oilers and shutoff valves in the service air lines. However, it did not affect any related system or equipment.

The mechanical failure of an oiler in an existing service air line will not render this system inoperable. The replacement of one type of manual valve with another did not change the failure modes of the system. Therefore, no new credible failure modes were created by this modification.

This modification did not affect the structural integrity of any equipment. The equipment modified was not safety related and was not required for the safe shutdown of the plant. There was no radioactive leakage path created by this modification. The FUEL HANDLING and STORAGE SYSTEM shall mitigate the consequences of the Fuel Handling Accident. In addition, the system shall provide safe storage and maintain the fuel covered for all AOTs, DBAs and Special Events. The UFSAR changes did not impact the ability of the system to mitigate any DBA or AOT. No unreviewed safety question was created and no Technical Specification change resulted.

**ECN P0742 - Telephone Communication Upgrade - Units 1, 2, and 3**

Description/Safety Evaluation

Replaced the existing Stromberg Carlson Crossreed GPABX and the Dimension 400 PABX with a single PABX. Reliable backup emergency power will be provided for the PABX. The existing BFN telephone system which consists of a leased Dimension 400 system and a TVA-owned Stromberg Carlson Crossreed system did not meet the regulatory requirements of the Radiological Emergency Plan. 10 CFR 50 requires the telephone communication system be provided with a backup power source.

The modification will bring the telephone system into compliance with 10 CFR 50 requirements. No safety related function will be adversely affected. No unreviewed safety question was created and no Technical Specification change resulted.

**ECN P0859, DCN F12950A - Drywell Access Platform Attachment Changes - Unit 2**

Description/Safety Evaluation

The drywell access platforms at Elevations 563 foot 2 inches and 584 foot 11 inches compromising of 1-1/2 inch grating x 3/16 inch load bars are presently attached to the floor grating to the supporting beams by tack welding. This ECN allowed an alternate attachment method (mechanical fastening) of the grating to its supporting steel. The alternate method was needed to reduce modifications delays inherent when welding is performed (e.g., ALARA considerations) and expedite QC inspections. The Buildings and Structures Systems cannot provide a potential initiating cause of threats to the fuel or the nuclear system process barrier. No system operation was affected, and no equipment failure was possible since the holddown devices preclude interaction of the grating with safe shutdown equipment.

The modification affected only the method of attachment of the grating. Calculations provide qualification that the alternate methods (mechanical fastening) adequately restrains the grating as does the present method (welding). No unreviewed safety question was created and no Technical Specification change resulted.

**ECN P0919 - PASS - Unit 2**

Description/Safety Evaluation

In order to comply with NUREG 0737, BFN installed a Post Accident Sampling Facility for each unit in the respective Turbine Building. The purpose and scope of this ECN was to provide piping/tubing tie-in connections with manual isolation in each of the following lines for the future connection to a PASS for Unit 2 only:

a. RHR Liquid Sample Line
b. Torus Gas Sample
c. Drywell Gas Sample Line
d. Reactor Recirculation Sample Line
e. Reactor Building Closed Cooling Water Supply Line (Cooling only)
f. Demineralized Water Line (Flushing only)
g. Liquid and Gas Sample Return Line to Torus

The PASS was not made operational by this ECN, since a plugged socket weld pipe coupling was welded to the open end of the tie-in piping for each line downstream of installed manual isolation valves. For the Liquid and Gas sample return line to the Torus, two locked closed manual isolation valves were installed in addition to the plug to satisfy containment isolation requirements during installation of the permanent PASS. These lines will be kept plugged until the final tie-in connection is made, via ECN P0916.

The activity did not adversely affect safe operation of the plant. None of the systems involved was compromised by providing tie-in piping/tubing connections for the PASS. PASS provided improved monitoring of plant conditions after an accident. These tie-ins tap off of existing systems outside of primary containment outboard of any primary containment isolation valves/devices, except for the Pass Return to Torus and the RHR Liquid Sample lines which are designed to meet containment isolation requirements. The tie-ins will maintain the parent system integrity via a closed manual valve (two manual valves for the Torus tie-in) in series with a welded plug. The tie-ins meet the same seismic and piping classification as the system to which they connect. In addition, design calculations for pipe stress were performed to document the integrity of the tie-ins and the parent systems. The adequacy of the pipe supports was documented and was validated by the resolution of the special requirements.

In addition to the above, and taking into consideration the fact that the future PASS did not perform any safety function, these tie-ins did not prevent any of the systems involved from performing their intended safety function. Further, these tie-ins provided future capability for taking and analyzing liquid and gas samples from the containment so plant conditions can be better monitored in the event of a DBA.

The equipment added by this modification met the original system requirements and did not introduce any new failure mechanisms that are not analyzed in the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**ECN P0956 - Unit 2 Shutdown Board Room C and D HVAC Seismic Qualifying - Unit 2**

Description/Safety Evaluation

ECN P0956 provided new seismically and environmentally qualified air conditioning and ventilation equipment for Unit 2 SDBR C and D at elevation 621 foot 3 inches and 593 foot respectively. This modification provided adequate cooling capacity to handle the increased heat loads which will result from the installation of new equipment described in ECN P0399. This modification assured that a redundant and qualified HVAC system for SDBR C and D were installed in accordance with the applicable seismic, environmental and Appendix R fire protection requirements.

This Safety Evaluation only evaluated the effects of blanking off the ventilation supply air ductwork for shutdown board rooms C and D. This ventilation flow path was disabled as part of ECN P0956 however, the effects of this portion of the modification were not explicitly evaluated in the previous safety evaluations. The new recirculation air conditioning units will ensure that adequate cooling capacity is available to handle the heat loads generated in the SDBR C and D. Therefore, removing the ventilation supply air did not affect the ability to maintain the temperature of the electrical board rooms within acceptable limits for operation of instruments and for uninterrupted safe occupancy under all plant conditions. This ensured that all SDBR equipment required to limit the consequences of an equipment malfunction will be available. The lack of SDBR pressurization and the elimination of the room fresh air supply did not affect the ability to use the SDBRs for backup control when the MCR must be evacuated. The UFSAR does not describe or take credit for SDBR pressurization for limiting the consequences of equipment malfunctions. No other plant systems or features that limit the consequence of an equipment malfunction were affected by removing the SDBR ventilation supply air. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W1290 - EQ Upgrade of Various CS Limit Switches - Units 1, 2, and 3**

Description/Safety Evaluation

Replacement of these limit switches (Radwaste, primary and secondary containment, control air and containment inerting) with environmentally qualified switches had no effect on the probability of occurrence of any previously evaluated accident at BFNP. The replacement switches perform exactly the same function as the previous switches. There was no change to the function of any affected system, or in the manner in which these systems performed their functions. The limit switches only indicate the valve position and are not involved in the valve logic. The only credible failure mode for this activity was a failure of one of the limit switches which could lead to incorrect indication of valve position.

The valves were not modified in any way other than slight modification to the switch mounting bracket. The switches are not a part of the valve logic, they provide position indication only. The new limit switches were more reliable than the previous switches because the replacement switches were qualified for their environment. During the unlikely event of an accident, the new switches will continue to function as designed; the continued operation of the previous switches were questionable. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W1514A - Modify Control Air Dryer Circuitry to Meet Scram Reduction Goals - Units 1 and 2**

Description/Safety Evaluation

DCN W1514A was a modification for 0-FCV-32-90 and 2-FCV-32-90 associated with the Standby and Unit 1 Control Air dryers. The control air tubing connected to solenoid valves 0-FSV-32-90 and 2-FCV-32-90 was modified. This allowed flow control valves 0-FCV-32-90 and 2-FCV-32-90 to remain open on loss of electrical power to solenoid valves 0-FSV-032-90 and 2-FSV-032-90 and close on loss of control air.

The previous installation provided for valve closure upon a loss of control power (non-Class 1E source), which also resulted in a reactor SCRAM in that Unit (except the common CA system), due to low pressure in the CA system.

The modification to the tubing of the solenoid valves permitted the continuous supply of CA upon loss of control power to the loop. A CA line break downstream of the low control loop would cause a decrease in CA pressure. A decrease in CA pressure would cause alarms in the control room via low pressure switches. While it is true that a loss of pressure would be detected, it is unreasonable to assume that the operator could take any action within the short time (less than 30 seconds) it takes to reach the scram setpoint. A line break in one Unit downstream of the FCV will also be detected by a flow switch upstream of the valve which initiated closure of the FCV to prevent depressurization of the receiver tanks through the common header. Prior to this modification, the FCV would close on a loss of control power thus protecting the common CA header from an additional failure in the same Unit such as a stuck open purge valve or a possible line break. A loss of control power to the Unit 2 dryer FCV (or the Unit 0 backup to any Unit if its in service) now leaves all three Units vulnerable to a low pressure scram if that same Unit should experience an unlikely second failure that vents control air out downstream of the FCV, since the FCV would remain open and bleed down the common header. However, the possibility for a low CA pressure scram of all three Units has always existed if the common header ruptured upstream of the FCVs at any time and the effects of this failure mode is the bounding case since none of the CA piping from the compressors past the dryers to the Reactor Building boundary is seismically qualified.

The implementation of this modification necessitates removing power from the control circuits associated with flow control valves 0-FSV-32-90 and 2-FSV-32-90. With the plant in a cold shutdown condition, this modification did not impact safety limits. The safety related portion of the CA System was not impacted by this modification (the protective safety function as accomplished by accumulators). Based on the above considerations, the modification had no adverse effect on the associated systems nor on any other system. Therefore, this modification had no impact on nuclear safety. No unreviewed safety question was created and no Technical Specification change resulted.

**ECN P5250 RO - HPCI and Reactor Feedwater Inverters Setpoint - Units 2 and 3**

Description/Safety Evaluation

This ECN changed the (low input voltage trip) setpoint of the HPCI and the RFW inverters from 200 VDC +/- to 185 VDC +/- 5V. The reason for reducing the setpoint was to eliminate the possibility of worst case voltage transient conditions shutting down the inverters. Any shutdown of the inverters jeopardizes the HPCI and RFW systems ability to perform their design functions and has a potential adverse impact on nuclear safety. This modification did not change any circuits, equipment, or method of performing design functions; therefore, no new failure modes were introduced. The setpoint change did not adversely affect any system. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN H5860A - Reactor Recirculation Pump Monitoring Instrumentation Additions - Unit 2**

Description/Safety Evaluation

As a result of industry operating experience, the possibility exists for thermal fatigue cracks forming in the shafts of vertical reactor pumps which utilize mechanical seal water injection. Observed cracks have been localized in the vicinity of seal water cooling temperature gradients. Therefore, it is essential to monitor vertical shaft pumps for shaft and pump housing crack development.

This DCN installed vibration, proximity, and velocity sensors on each of the reactor recirculation pumps to provide data and annunciation signals relative to crack formation. Vibration sensors, one per pump, are mounted on the pump housing and will be utilized as acoustic monitors. This measurement will initiate an alarm when an X period noise (transducer orientation) occurs at a specified value above background.

Two velocity sensors per pump were mounted 90° apart at the top of each pump motor housing. These sensors will measure motor vibration and initiate an alarm upon sensing excess pump vibration.

Four proximity probes per pump were mounted in pairs, 90° apart to monitor shaft eccentricity. One set was mounted to monitor shaft rotation at the bottom of the motor and the other set was mounted to monitor shaft rotation at the top of the pump. When eccentricity exceeds a preset limit in either X, Y direction or a combination of an alarm is initiated.

The existing vibration switches were electrically disconnected and abandoned in place. A common alarm from the new system annunciated at the same location as the previous alarm.

The existing tachometer was electrically disconnected and abandoned in place. The speed signal in the control room was replaced with the speed signal from the phaseometer.

Existing cables were replaced and rerouted from the Control Room to the station monitor rack (2-PNLA-068-25/412) to the Reactor Building. New cables were added from the station monitor rack through Penetration EH to the vibration transducer interconnection boxes on Reactor Recirculation Pumps A and B. A computer will be located in the Reactor Building with associated cables to connect the computer with the station monitors. A modem will allow realtime analysis from other TVA locations.

The activity did not prevent the reactor recirculation pumps or the Auxiliary Power System from performing their function or operation as described in the Unit 2 Technical Specification. Because the proposed change had no effect on any safety related system or any analysis of DBA's or AOT's, it did not reduce the margin of safety as defined in the basis for any Technical Specification. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W6377B - Modifications to Drywell Platform - Unit 2**

Description/Safety Evaluation

This DCN W6377B released design changes for the modification to Unit 2 Reactor Building drywell platform at elevation 584 foot. Modifications can be categorized in three groups:

a. Modifications to structural steel due to increased confirmatory piping loads from 79-14 analysis, and CRDH frame attachment loads;

b. Additional bracing for horizontal rigidity,

c. Removal and subsequent reinstallation of conduit and/or conduit supports to facilitate the above (a. and b.) modifications.

DCN W6377A provided modifications to the Drywell Platform at elevation 584 foot based upon "Hanger Guidance Loads" from the increased seismic loadings due to the 79-14 analysis. DCN W6377B provided additional modifications to the platform beams and connections due to the final pipe support loads and provided lateral bracing to ensure horizontal rigidity. Some modifications initially provided in W6377A were removed due to more sophisticated analysis performed at this time.

DCN W6377B and DCN F11357A provided for the removal and reinstallation of drywell conduits and conduit supports where necessary to perform and facilitate the platform modifications.

Since this modification introduced no new credible failure modes, the effects of this change were enveloped by the credible failure modes for the existing design. This modification had no adverse impact on systems important to nuclear safety and therefore, created no possibility for a malfunction of equipment of a different type than any evaluated previously in the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN H6558A - Central Lube Oil Purifier Chiller Replacement - Unit 1, 2, and 3**

Description/Safety Evaluation

The lube oil purifier in the non-safety related Central Lubricating Oil System is designed to maintain lube oil quality by removal of entrapped moisture and suspended solids. The lube oil purifier chiller assembly, which cools the lube oil condenser and provides the necessary vacuum in the lube oil purifier, had been a maintenance problem resulting in unreliable operation. This DCN replaced the existing lube oil purifier chiller assembly shown on vendor drawings with a mechanical vacuum pump design shown on vendor drawings. This modification cooled the lube oil condenser with raw cooling water. The new components were installed on the existing lube oil purifier system in the turbine building.

In addition to the change to a mechanical vacuum pump design, additional electrical modifications were required. The lube oil 200 amp safety switch has been replaced with a 150 amp fusible disconnect safety switch to provide proper circuit protection. Non Class 1E power and control cables for the new components are also provided. Six control cables from the condensate tank level switches and solenoids will be routed in dedicated conduit to a new junction pull box 0-JBOX-020-8965. From the pull box they are routed in conduit to the lube oil purifier electrical enclosure located on the skid. This modification also installed thermal overload heaters in the vacuum pump motor power circuit. The power cable for the vacuum pump motor (skid mounted) is routed by conduit to the electrical enclosure. This modification resulted in a new RCW cooling water interface which affected UFSAR Figure 10.7-1a.

The Raw Cooling Water system and the Central Lube Oil System perform no nuclear safety functions. The only protective safety function involving the Raw Cooling Water System pressure boundary is that secondary containment integrity must be maintained at RCW secondary containment penetrations. This DCN had no adverse impact on the function of either system and improved the reliability of the Central Lube Oil System. Therefore, this modification had no adverse impact on the function or performance of any safety related systems. No unreviewed safety question was created and no Technical Specification changes resulted.

**ECN P7005 - $H_2O_2$ Sample Line Moisture Intrusion Prevention - Unit 2**

Description/Safety Evaluation

This modification affected the CAM system, a subsystem of the CIS. These modifications were required to prevent moisture from condensing in the Hydrogen/Oxygen Analyzer instrument lines and entering the $H_2O_2$ analyzer and associated in-line instrumentation. Class 1E heat tracing, a filter coalescer, float trap, solenoid valve, miscellaneous tubing fittings and needle valves, and cable to the solenoid valves were added to each Unit 2 $H_2O_2$ analyzer. The changes placed a filter coalescer on each $H_2O_2$ analyzer instrument lines downstream from both the $O_2$ sample pump and the $H_2$ sample pump. This removed any moisture located in the sample sense line. A float trap was connected to the filter coalescer to drain off the water collected. A solenoid valve was connected to the drain of the float trap to allow the trap to empty. This solenoid valve was controlled by the existing auto timer associated with the $H_2O_2$ analyzers.

This change did not alter or otherwise impact the normal operational characteristics of the $H_2O_2$ analyzers or the CIS. The CAMs shall provide the capabilities to determine the primary containment atmospheric oxygen and hydrogen concentrations during normal plant conditions and hydrogen concentrations during and after a LOCA. The CAMs and the identified modifications did not interface with equipment capable of initiating an accident nor with the reactor.

The new components (filter coalescer, float trap, solenoid valve, miscellaneous tubing fittings and cable for each CAM) were qualified to the same requirements (i.e., Class 1E and Seismic Class II requirements) as the existing components, were installed to the same procedures as the existing components and were functional tested after installation. The CAMs, including power supplies for the heat tracing and solenoid valves, consists of redundant physically separated sampling loops as described in the Unit 2 UFSAR, Section 5.2. There was no unreviewed safety question created and no Technical Specification change resulted.

**ECN E-2-P7031 - RCIC Lube Oil Temperature Indicating Switch Replacement - Unit 2**

Description/Safety Evaluation

This ECN addressed modifications to the RCIC Lube Oil System. Three temperature indicating switches (TIS-071-023, TIS-071-045, and TIS-071-046) were replaced with three temperature indicators and three temperature switches that provided the same function as the TISs (monitor the oil temperature and annunciate in the control room on high temperature). Dual element thermocouples replaced the three existing thermocouples. The first element of the thermocouples was connected to an existing temperature recorder while the other element was connected to a temperature switch installed in a seismically mounted box. These switches perform no tripping function, only annunciation in the control room. In addition, three temperature indicators were installed to provide local readout. The previous TISs were derated and like replacements were no longer available. Without this modification, damage to the RCIC turbine bearings may occur which could ultimately lead to RCIC turbine shaft failure.

Replacing the temperature indicating switches did not affect the function or operation of the RCIC Lube Oil System as the oil temperature is still monitored. RCIC Lube Oil System pipe loads were reduced by replacing the existing TISs with lighter temperature indicators. The thermocouples did not affect the load on the pipe.

The seismically mounted box containing the temperature switches was mounted on a wall adjacent to the piping in order to prevent adverse interaction with seismically qualified piping. No unreviewed safety question was created and no Technical Specification change resulted.

**ECN E-0-P7068 R0 - PSA Changes - Units 1, 2, and 3**

Description/Safety Evaluation

TACFs 0-84-94-77, 0-84-99-33, 0-84-100-33 and 0-84-110-33 separated service air for the Radwaste and Unit 1 and 2 Diesel Generator Buildings from the PSA System by installing three additional air compressors and receiver tanks (one 25 HP compressor with a 16 $ft^3$ receiver tank located in the Waste Demineralizer Fill and Access Room, and two 10 HP compressors each with a 10.7 $ft^3$ receiver tank located in the Chemical Waste Tank Room) and associated equipment in the Radwaste Building and Diesel Generator Building, and disconnecting the PSA supply to these buildings from that for the remainder of the plant.

ECN E-0-P7068 superseded the above TACFs and incorporated those modifications into the permanent plant design with some additional changes. As a result of the potential contamination of the service air in the Diesel Generator Building, it was decided to isolate the RSA from the rest of the PSA system. The service air header in the DG Building was disconnected from the Radwaste Building air system and reconnected directly to the PSA System. The RSA system operated in the range of 95 to 125 psig with a design pressure of 150 psig.

The existing supply piping and wall penetration into the DG Building was used to reconnect the DG Building to the PSA System. A drain line was added to the DG Building portion of the Service Air system at the low point for removal of condensation. ECN E-0-P7068 also installed non-Class 1E circuit breakers in 480V Radwaste Board 2 Compartments 1C2L and 1C2R and 480V Radwaste Board 1 Compartment 9F and non-Class 1E cables with their associated conduit to supply the new service air compressors.

This change did not adversely affect the function of the PSA system. ECN E-0-P7068 documented the addition of air compressors, receiver tanks and their associated piping, controls and electrical connections in the Radwaste Building. The PSA, which included the new separate RSA system, accomplished the same function as prior to the modification. The new receiver tanks were supplied with relief valves for overpressure protection. There are no safety related components in the vicinity of the new receiver tanks which could be impacted by the possible breach of a receiver tank.

The postulated rupture of the RSA receiver tank is an extremely low probability event. Rupture of a receiver tank will not result in a substantial increase in room pressure due to the volumes of the rooms (the volume of the receiver tanks is less than 1% of the room volume). Rupture of the 16 $ft^3$ receiver tank located in the Waste Demineralizer Fill and Access Room could cause a projectile to be sent out the open door and impact a Waste Sample Tank. The probability of a Waste Sample Tank penetration is low given the configuration of the room and the distance between the receiver tank and the Waste Sample Tanks which is greater than 30 feet. This event is enveloped by the discussion provided in UFSAR Section 9.2. The new equipment failure modes are enveloped by the failure modes of the existing PSA system and did not adversely impact nuclear safety.

The new RSA system added three new compressors and the associated equipment needed to provide an independent RSA system. The RSA system did not perform a safety function and is not essential for the operation of any safety-related system. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN E-1-P7114 R1 - DG Air Dryers - Units 1 and 2**

Description/Safety Evaluation

This DCN added air dryers and aftercoolers to the Unit 1 and 2 DGs that were upstream of the check valve in the non-safety related portion of the system. The Unit 3 DG's air dryers were installed in 1990 by ECN E-3-P7113. The piping modifications and electrical conduit supports associated with the addition of the air dryers and aftercoolers were seismic Category II qualified and had no impact on the safety-related (seismic Category I) portions of the DSAS or any other system. In addition, these air dryers will eliminate future problems associated with corrosion products in the starting air system, increasing long term reliability. The installation of the air dryers and aftercoolers did not affect the operability or function of the DSAS, and did not impact nuclear safety. No unreviewed safety question was created and no Technical Specification change resulted.

**ECN E-2-P7161 - Unit 3 120 VAC Regulating Transformer Relocation - Unit 3**

Description/Safety Evaluation

This revision relocated the Unit 3 regulating transformer into the SDBR and required installation prior to Unit 2 fuel load.

The possible failure modes associated with this modification consist of failure of the regulating transformers added by this ECN to Unit 3, electrical faults, inadequate circuit protection, regulation, coordination, or the circuitry failing open. These failure modes will result in annunciation in the control room for abnormal voltage on Unit 1 and 3

Panel 9-9 and automatic transfer to an alternate power feeder. The regulating transformers, breakers, cables, switches and raceway modified by this ECN are seismically qualified by calculations and were installed in compliance with all applicable design codes, specifications and procedures. It is verified that the electrical design of breakers and cables is acceptable from ampacity, voltage drop, and short circuit considerations. The electrical modifications were implemented in compliance with UFSAR Section 8.9 to assure satisfactory electrical separation/isolation and physical separation. The use of the maintenance bypass will be controlled administratively in accordance with Technical Specifications. The possible failure modes were enveloped by the existing analysis. No new failures were introduced which could adversely affect the function or performance of the 120 VAC Instrument and Control Power Supply, 480V Shutdown Boards, or any other safety related system.

The proposed ECN did not alter any assumptions previously made in evaluating the radiological consequences of the accidents evaluated in the UFSAR. All affected systems are fully qualified and will respond as described in the UFSAR to mitigate the postulated events. This modification did not impact radioactivity discharge; therefore, there will be no increase in radiation dose to the public or plant personnel. No unreviewed safety question was created and no Technical Specification change resulted.

**ECN E-0-P7195 R1 - Power Supply to Diesel Fire Pump - Units 1, 2, and 3**

Description/Safety Evaluation

Connect the 4160-480 volt bladder tank substation to the 4160 volt north loop line, which is fed from 4160 volt cooling tower switchgear D, panel 7. The normal power source for 4160 volt cooling tower switchgear D is 161KV/4160V cooling tower transformer 1, and the alternate power source is 161KV/4160V cooling tower transformer 2, which serves switchgear C.

The purpose of this activity is to provide an adequate source of electrical power for the diesel fire pump house 480 volt distribution panel.

Connecting the bladder tank substation to the north loop line has no adverse effect on safety. The additional load is within the upstream electrical equipment continuous current rating, and is within the upstream 4160 volt circuit breakers' trip ratings (per calculation ED-N0205-890081 R0). There was no change to the redundant, seismically qualified 4160 volt cooling tower switchgear incoming circuit breakers' trip circuits. Therefore, the capability to trip the cooling tower lift pumps to prevent pumping of uncooled water from the warm water channel to the cold water channel was not adversely affected by this modification. None of the loads supplied power from 4160 volt cooling tower switchgear C or D, the loop lines, or any downstream electrical equipment perform safety functions. Connecting the bladder tank substation to the 4160 volt north loop line has no direct or indirect effect which could initiate an accident. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN H7844A - DSAS Relief Valve Reduction - Units 1 and 3**

Description/Safety Evaluation

DCN H7844A removed four of the five relief valves originally provided on each bank of the Unit 1/2 DSAS. On each tank with a relief valve removed, the relief valve and a 1 x 3/4 inch hex head bushing was deleted from the piping configuration and a one inch square or hex head plug inserted to plug the tank where the relief valve was attached. This weight removal did not adversely affect the seismic response of the tank. The upper tank with the remaining relief valve had no modifications. Thus, the seismic qualification of the air tanks and relief valves was not adversely affected and no formal civil calculation was needed.

Calculation MD-Q0086-900074, Revision 1, showed that one relief valve per air bank will be adequate to protect the DSAS from overpressure. Eliminating four out of five relief valves per bank reduced the probability of system failure due to relief valve malfunction.

The DSAS is provided as a redundant system (two starting air systems per diesel). This DCN modification did not alter the starting air system redundancy, consequently, single failure criteria was maintained as considered in the original safety analysis. Also, this piping modification was installed to the original Seismic Class I requirements. Thus, the consequences of an accident previously evaluated in the UFSAR was not changed. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN H8367A - PSC Head Tank Pump Seal Cooling Piping Change - Unit 2**

Description/Safety Evaluation

Previously, the PSC Head Tank pumps' mechanical seals were cooled by flushing their stuffing boxes with condensate water from the Condensate and Demineralized Water System and then draining to Dirty Radwaste. This provided approximately 3-5 gpm inleakage to the Radwaste floor drain system. To eliminate this processing burden, this modification seals off the Unit 2 PSC Head Tank pumps. A flush line, including an isolation valve, was installed from each pump discharge and was routed to the seal gland housing and stuffing box and then returned to the pump suction. This provided cooling to both the upper and lower halves of the double seals which were previously inadequately cooled when seal piping was only routed to and from the stuffing boxes. The existing condensate water supply and drain lines are capped off.

The PSC Head Tank pumps are not safety related and are used to provide makeup water from the Torus to the PSC Head Tank for the purpose of pressurizing the CS and RHR pump discharge piping to prevent water hammer upon pump startup. This modification had no significant impact on the performance of the PSC head tank pumps and did not result in any new credible failure modes which could impact safety related systems.

The rerouting of the seal piping of the PSC head tank pumps, in accordance with design criteria, did not impact the operability of any safety related system and will not result in unnecessary safety component actuations. The accidents identified in Chapter 14 of the UFSAR cannot be initiated by any credible failure modes involving the PSC head tank pumps. No unreviewed safety question was created and no Technical Specification changes resulted.

**DCN W9276A - Telecommunications Upgrade - Units 1, 2, and 3**

Description/Safety Evaluation

DCN W9276 involved the replacement of telecommunications equipment in the Communications Room on elevation 593 foot of the Control Building. This modification affected only the Communication System.

This modification changed the Telephone System from a centralized system to a distributed system. In other words, the telephone switching equipment is no longer in one central location but is distributed among three locations. This concept allowed expansion of the Telephone System without a commensurate expansion of cable pairs in the Communications Room. New telephone switching equipment replaced existing equipment in the communications room and will primarily serve all powerhouse areas such as the Reactor Building, Control Building, Turbine Building, switchyard, etc.

There are no accidents evaluated in UFSAR chapter 14 that can be initiated or directly mitigated by the components of this design change. The Telephone System is not a safety-related system and is not required to operate during or after a DBA. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W9277A - Telephone Power Source Upgrade - Unit Common**

Description/Safety Evaluation

There are three credible failure modes for the telephone system, (1) loss of power to the system, (2) equipment malfunction, and (3) cable failure which remain unchanged from before. The new modification of the telephone system power supply was relevant only to the first two failure modes. In the event of loss of the four diverse sources of power to the chargers or the failure of both chargers, the telephone battery will provide adequate power to operate Node 1 for three hours. In the event of loss of power to Node 2 telecommunications equipment, an uninterruptible power supply will provide the required power for three hours.

The telephone-system is not a safety related system and is not required to function during or after UFSAR Chapter 14 events. The modification did not alter the function of the telephone system and did not impact any nuclear safety-related system, structure, or component. Only the telephone system power supply was altered, but the cables were routed in dedicated conduit and the present isolation of the non-1E communication loads from the Class 1E power system by qualified breakers remained unchanged. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W9951A - Load Dispatcher Load Information Updates - Common**

Description/Safety Evaluation

This DCN provided for replacement of the existing GETAC remote terminal with a new SCADA RTU for transmitting data to the Wilson area dispatcher. Also provided was a remote terminal for transmitting generation data to the new digital control and monitoring system to the load coordinator. This new equipment will provide faster transmission and more accurate data, such as BFN generation levels, loading of transmission lines and status of switchyard equipment to the load coordinator and the area dispatcher. Ready access to this data is critical to the operation of the TVA power grid. This modification is part of a valley-wide upgrade to improve the reliability of the TVA generation and transmission control and monitoring system.

The effects of the credible failure modes of this change were within the bounds of the credible failure modes for the existing design. This change facilitated communications between BFN and TVA's area and load dispatchers. This modification had no adverse impact on systems important to nuclear safety and therefore, created no possibility for a malfunction of equipment of a different type than any evaluated previously in the UFSAR. This modification did not adversely impact any of the analysis of DBA's or AOT's described in the UFSAR. No unreviewed safety question was created and no Technical Specification changes resulted.

**DCN S10069A - Radwaste Flow Drawing Corrections - Units 1, 2, and 3**

Description/Safety Evaluation

DCN D10069A was initiated to update drawing 0-47E830-3 to resolve ten discrepancies with the installed configuration. These discrepancies were walked down and were evaluated to determine the acceptability of the installed configuration.

This change did not result in any physical changes to the existing plant configuration. The change corrected drawing discrepancies to reflect as-built condition. No new credible failure modes associated with this documentation change are different from those enveloped by the existing design. These documentation changes did not adversely affect system function or operation, nor can they be the initiator of an accident. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W10224A - Offgas Loop Seal Solenoid Valve Replacement, Units 1, 2, and 3**

Description/Safety Evaluation

The offgas dehumidification drain has a water seal in it to prevent leakage of the offgas from the line. The water seal is in a pipe loop in the turbine building condensate pump pit equipment drain sump. The loop is submerged when the water level is at or above 4 feet - 6 1/2 inches below the flange of the sump. If the water level drops below this point, LS-77-19 closes contacts to alarm and energize FSV-77-19, which opens to admit water from the gland seal system to the loop to seal it.

DCN W10224A replaced the existing normally open (fail open) solenoid valves 1, 2 and 3-FSV-77-19 with normally closed (fail closed) solenoid valves. These valves were originally purchased normally open. However, the original design, required these valves to be normally closed and fail closed. These valves are the Radwaste System Offgas Loop Seal Condensate make-up vales. These valves will fail closed to prevent continuous dumping of gland seal water into the sump and will allow the system to function as originally designed.

This modification did not affect any safety related portion of the Radwaste System or any other system and is required only to satisfy design requirements for valve operation. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W10416A - SGTS Crosstie Valve Removal - Units 1, 2, and 3**

Description/Safety Evaluation

DCN 110416 covered the analyses and replacement of the SGTS decay heat removal cross-tie vales (FSV-065-002, -024, and -066) which are not properly qualified to perform their safety related functions. Since there are no qualified direct replacements for these valves, mechanical dampers are installed in their place. These four inch replacement dampers, DMP-065-002, -024, and -066, have provisions for attaching a possible future automatic actuator. After these dampers are adjusted for minimum required flow, the dampers are locked in that position. These dampers can be manually closed for periodic testing and maintenance. The purpose of replacing these normally closed solenoid valves with these locked in position dampers is to ensure decay heat removal from contaminated charcoal filters in a non-operating SGTS filter train. The existing cross-tie piping has been enlarged to ensure sufficient air flows through the filters' charcoal sections. The additions of the test ports in the cross-tie piping and the filters' supply and exhaust ducting will help in balancing the SGTS for the PMT.

This modification increased the reliability of this system by replacing unqualified valves with qualified dampers. Once the replacement dampers were locked in position, no electrical and/or operator actions were needed. This modification did not introduce any new failure modes. The addition of seismically designed test ports had no adverse impact on safety. As a result of this modification, the flow capability for the system, when operating in decay heat removal mode, may decrease. However this depends on the damper settings required to achieve adequate decay heat removal flow. These damper settings, the decay heat removal flow, and the total system flow capacity will be tested and verified by the required PMT.

The SGTS is not involved in initiating any DBA but does serve to mitigate several DBAs. This DBA mitigating function will be maintained by this modification by locking the crosstie dampers in position to allow cooling air flow for decay heat removal from the charcoal filter. The previous design required operator action to open the crosstie solenoid valves for this cooldown function.

All cables associated with solenoid valves and limit switches were removed back to the termination junction box. In addition, remote handswitches and panel indicating lights were removed and associated wiring was deleted back to the terminal block. This ensured that the previous electric wiring did not adversely affect other safety components.

The manually operated dampers installed by this modification met the same seismic class I requirements as that specified for the SGTS system. No unreviewed safety question was created and no Technical Specification changes resulted.

**DCN W10422A and W10423A - Nitrogen Supply Isolation Valves Addition - Units 1 and 3**

Description/Safety Evaluation

This modification involved the elimination of the supply of nitrogen to Units 1 and 3 while these Units were not in run mode.

The supply of nitrogen to Unit 2 will be uninterrupted as a result of this modification. Two isolation valves have been added in order to eliminate the nitrogen supply to Units 1 and 3. However, a Special Requirement which specified that the added isolation valves would be locked in the closed position in order to prevent any unintentional opening while Unit 2 is in operation assured the Unit 2 uninterrupted supply.

All performed modifications met class I requirements in accordance with the UFSAR. Leakage of one isolation valve will not prevent the CAD system from operating on Unit 2 (meets single failure criteria). The existing CAD piping penetration in the reactor wall and the addition of a pipe plug will assure secondary containment isolation for Units 1 and 3. Thus no increase in offsite radiation exposure will occur as a result of this modification. Isolation of Unit 1 and 3 secondary containment is provided by pipe plugs on the ends of the piping in tunnels 1C and 1D. The remaining buried portion of the Units 1 and 3 CAD system will maintain its seismic qualification and so will the added isolation valves. All associated piping, supports, and equipment in support of Unit 2 will maintain class I qualification in accordance with the licensing commitments in the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W10618A - PASS Tie-ins - Unit 2**

Description/Safety Evaluation

Connections for the sampling points were issued in ECN P0919. Sample line connections and manual root valves were provided by this ECN from the jet pump number 1 instrument line for sampling reactor coolant, the RHR system (Heat Exchanger "C" vent lint) for sampling torus liquid and the $H_2/O_2$ monitoring system for sampling containment atmosphere in the drywell and torus.

Primary containment isolation is required for the RHR liquid sample line and the liquid/gas return line to the torus. Primary containment isolation valves for these lines met the requirements for primary containment isolation. The primary containment isolation valves for the PASS are normally closed and fail closed. These valves are manually opened from the main control room following an accident to allow sampling. The use of qualified components and seismically analyzed piping assured no adverse impact to safety involving primary containment isolation and safety related syst m 'solation boundaries. The failure of the reactor recirculation sample pip in ombination with the failure of the excess flow check valve during testing occurring under normal plant operating conditions would result in a small LOCA whose consequences are enveloped by the evaluation of a LOCA in UFSAR Section 14.6. A Technical Specification change was submitted to add 4 isolation valves to the appropriate valve listing. No unreviewed safety question was created.

**DCN S10620A - Drawing Discrepancy Valve Location Corrections - Units 1, 2, and 3**

Description/Safety Evaluation

This DCN addressed PDD 90-009 which identified an inconsistency between the field conditions and drawing 791E242-2 concerning the position indicating lights' labeling on panel 25-17 for primary containment outboard isolation floor drain valves 1-FCV-77-2B, 2-FCV-77-2B and 3-FCV-77-2B and primary containment inboard isolation equipment drain valves 1-FCV-77-15A, 2-FCV-77-15A and 3-FCV-77-15A. DCN S10620A updated drawing 791E242-2 to show the proper position indicating light location for the primary containment outboard isolation floor drain valves and the primary containment inboard isolation equipment drain valves as installed on panel 25-17. Also, DCN S10620A updated drawing 0-47E610-77-1 to add a note on the unitization of the affected component numbers. The revised control diagram matched the labeling in the field for these valves. This documentation only change to correctly reflect the as-built condition of some Radwaste System primary containment valves did not affect system function or operation. Therefore, the DBAs and AOTs described in Chapter 14 of the UFSAR are unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W12579, Revision A - Replacement of RMS Flow Control Valves - Unit 2**

Description/Safety Evaluation

DCN H7926A replaced electric motor actuators for Class 1E flow control valves FCV-90-254A and B, -255 and -257A and B in the Unit 2 RMS with identical electric motor actuators borrowed from the corresponding Unit 3 valves. However, the qualified life of these actuators expires in October of 1992. This DCN replaced the above Unit 2 ball valves and electric motor actuators with new Class 1E solenoid operated gate valves. The FCV prefix for valves 90-254A and B, -255 and 257A and B were changed to FSV. These containment isolation valves close on receipt of a containment isolation signal. The existing piping and replacement valves were seismically analyzed and support modifications implemented to assure acceptability.

The existing power and control cables 2PC629-I, 2PC632-I, 2PC635-I, 2-PS644-II and 2PC647-II, between JB 3335 and the subject valves were replaced with environmentally qualified valve pigtail leads which were supplied with the new valves. Calculation ED-Q2090-900073 evaluated the adequacy of the pigtail leads and the existing cables for voltage drop, ampacity, short circuit and Appendix R high and low impedance fault considerations, and found them acceptable. The existing panel 9 circuit breakers 203 and 303 were replaced with Class 1E GE type TED 15 circuit breakers, in order to provide adequate circuit protection.

The only difference in credible failure modes for these replacement valves is power loss. Valves with electric motor operators will fail in the as-is position. However, these solenoid valves will fail in the closed position. Since these valves served as primary containment isolation valves and must close on containment isolation signal, the fail close position is safer than the fail as-is position. These primary containment valves are designed to close on containment isolation signal and they are fail close valves. This modification did not adversely impact the qualification, function, or operation of the affected system or any other system. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S12735A - Radwaste Drainline Capped - Unit 2**

Description/Safety Evaluation

DCN S12735A was issued to resolve PDD 90-158 on drawings 0-47E835-1 and 0-47E830-3. Revisions were made to the Potable Water Distribution and Radwaste Systems.

Changes to the Potable Water Distribution System under DCN S12735A did not require a safety evaluation and is therefore not addressed. The capping of a drain line to Radwaste as shown on 0-47E830-3 (FSAR Figure 9.2-3c) is the subject of the evaluation.

The change had virtually no effect on the system. Capping the drain pipe prevents the escape of radioactive liquids or gases. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W14096 - Addition of Backup Motive Air Supply for FCV 64-20 and 21 - Unit 2**

Description/Safety Evaluation

DCN W14096A documents the addition of a backup motive air supply to the Suppression Chamber/Reactor Building vacuum breaker butterfly valves FCV-64-20 and FCV-64-21. A three-way pressure control valve, PCV-84-654, PI-84-708 and pressure regulator, PCV-84-706, were added to provide CAD system nitrogen at the required pressure to the butterfly valves if the normal air supply from the CAS is not available.

This change was made to mitigate a failure mode of the butterfly valves which could prevent achieving primary containment isolation when it is required. The butterfly valves are designed to fail open upon loss of supply air. During postulated LOCA conditions the Control Air supply to these valves could be lost, thereby preventing them from performing their secondary safety function of containment isolation. Addition of a qualified backup nitrogen supply from the CAD system will improve the reliability of these butterfly valves in performing this secondary safety function. This modification will not affect in any way the primary safety function of these vacuum breakers, that is, to automatically open to protect the torus from experiencing a negative differential pressure in relation to the Reactor Building.

The additional nitrogen load on the CAD system added by this modification is insignificant compared to the CAD flow to the drywell/torus during containment air dilution activities. This tap-off is practically a dead-end user since its main purpose was to serve as a backup. The backup nitrogen was supplied through pressure regulator PCV 84-706 at approximately the original Control Air supply pressure and relief valve RFV 84-704 will protect from over pressurization in this line. Consequently, solenoid valves, FSV 64-20 and 21, which actuate the Torus/Reactor Building vacuum breakers and the piston actuators on FCV 64-20 and 21 continued to operate at the original design air supply pressure.

Loss of the nitrogen air supply will meet single failure criteria and will return the system to the original single air source configuration. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W14099A - Radwaste Drain Modification - Units 1, 2, and 3**

Description/Safety Evaluation

The modifications made on the radwaste system by DCN W14099A were safe from a nuclear standpoint. The changes provided flow blockage to potential radioactive ground release paths; thus, effectively eliminating the potential to exceed main control room and off-site federal dose limits. Dose calculations associated with the off-site Dose Calculation Manual did not require revision.

The closed valve installed on the three inch off-gas stack drain line and the added loop seal piping and check valves just inside the Radwaste Building on the SGTS underground header drain line provide a static pressure boundary and are rated for the system service pressure. Since the seismically qualified stack drain valve and piping segment and the loop seal, and check valves provide static pressure boundary only, they do not have the potential to increase the occurrence of a malfunction of the SGTS or any other associated equipment. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S14269B - Instrument Labeling - Units 1, 2, and 3**

Description/Safety Evaluation

There are no DBAs or AOTs associated with labeling panels 2-9-10, 2-9-23-7, 2-9-23-8, 3-9-23A, 3-9-23B, 3-9-23C and 3-9-23D instruments to meet RG 1.97 requirements for unique identification. This change had no impact on the qualification, function or operation of any of the affected systems.

This DCN replaced existing labels with new labels to reflect current labeling requirements for radiation monitoring wide range gaseous effluent radiation monitors recorder and new UNID numbers for Diesel Generators A, B, C, D, 3A, 3B, 3C and 3D Amps, Vars and Volts indicators. The new labels were not initiators of any postulated accidents.

This DCN did not create any new credible failure modes which were not enveloped by the existing design. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W14487A - SPDS Upgrades - Units 1, 2, and 3**

Description/Safety Evaluation

DCN W14487A supports the design and installation for the SPDS and the ICS upgrade modification. This modification is required to support TVA's commitment to NRC to implement NUREG 0696 requirements. The original plant computer at BFN is a GE4020, with a single computer serving both Unit 1 and Unit 2. Unit 3 is also currently served by a separate GE4020. The function of the PCS is to provide a quick and accurate determination of core thermal performance, to improve data reduction, accounting, and logging functions for both the nuclear boiler and balance of plant equipment, and to supplement procedural requirements for control rod manipulation during reactor startup and shutdown. The new ICS/SPDS upgrade modification provided a separate computer system for each Unit.

The worst case failure mode of the architectural features (new walls, ceiling, and raised floor) would be their collapse. Since there are no safety-related components located within the drop zones associated with these items, and since the failure of these features does not degrade the ability of the Class I and Class II walls located in the existing instrument shop to perform their required functions, the loss of these components will not prevent the actuation/initiation or impact the performance of any safety-related component or function.

The addition and modification of control room penetrations, associated with the installation of this modification, were evaluated. The evaluation determined no new failure modes were created by the installation of this DCN since all work was performed in accordance with approved design criteria and procedures and was consistent with existing plant design.

This review also determined that a fire in the new computer will not degrade any safety function. The addition of the fire suppression system was also evaluated and no failure modes were identified due to the utilization of Halon 1301.

The failure modes associated with the electrical scope of this DCN are consistent with the design of existing plant systems. These failures modes are the short and open circuit conditions. Based on a review of the DCN, it has been determined that the implementation of this design change did not create any unanalyzed failure modes or cause the degradation of any safety system or function. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D14994B - Drawing 2-47E822-1 Correction - Unit 2**

Description/Safety Evaluation

DCN D14994A was issued to update drawing 2-47E822-1 to correct a DD. Drawing 2-47E822-1 was revised to correctly depict the one inch size of non-safety related valve 70-567 and the associated non-safety related one inch drain pipe from the Reactor Building equipment drain sump heat exchanger. The documentation change did not affect system function or operation, nor can it be the initiator of an accident. Therefore the DBAs and AOTs described in Section 14 of the UFSAR are unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S15468A - HPCI Fire Detector Drawing Discrepancies Corrections - Unit 2**

Description/Safety Evaluation

This DCN corrected drawing discrepancies on Unit 2 connection and schematic diagrams involving HPCI room fire detectors TS-26-37 A, B, and C. These fire detectors were changed from rate of rise to rate compensated in accordance with the BFN Fire Protection Plan, and a verification walkdown. This change was previously submitted to the NRC.

This documentation change did not add any new credible failure modes and did not have any adverse impact on the function or operation of the Fire Protection System or any other system. These drawing changes were in accordance with the BFN Fire Protection Plan.

These rate compensated fire detectors have a setpoint between 190°F and 225°F. The steam line leak detectors installed in the Unit 2 HPCI room have a setpoint greater than or equal to 190°F. Three of the steam line leak detectors are located approximately 3 feet from the ceiling of the HPCI room and are strategically placed following the steam line path, while the fire protection heat detectors are located directly on the ceiling for Unit 2, and away from the nearest point of the steam line path by approximately 25 feet.

The fourth steam line leak detector is located at a much lower elevation and above the turbine lube oil system pumps. Due to the distance between the fire protection heat detectors and the steam line leak detectors in the Unit 2 HPCI room, it is unlikely that the fire protection heat detectors would actuate before the steam leak detectors due to a steam line break. In addition, the fire protection spray nozzles are located below the HPCI steam line directly over the HPCI turbine lube oil system. Thus, if somehow, the fire detectors actuated first and initiated spray onto the lube oil system, this water would have little, if any effect on the upper room air temperature masking steam released from the break itself. Thus, the temperature near the steam line break detectors would continue to rise and actuate these detectors to isolate the break. Based on this, no safety concern or risk of operation is present with the existing configuration.

The documentation change did not affect the Fire Protection System performance from that described in the UFSAR. The change eliminated a discrepancy involving the fire detector type located in the HPCI pump room. Fire detection system failures are not initiators of any accident described in the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S15533A - HVAC Drawing Corrections - Units 1 and 3**

Description/Safety Evaluation

DCN S15533A was issued to update drawings 1-47E865-1, 3-47E865-12 and 0-47E366-64 to correct drawing discrepancies. These drawings did not show isolation valves 1-64-6002, 1-64-6003, 3-64-6002, and 3-64-6003 which are installed in the field. These root valves are a part of the vendor supplied assembly for pressure differential indicators 1-PDI-64-22 and 3-PDI-64-22 respectively. Root valves 2-64-6002, and 2-64-6003 are provided on the corresponding Unit 2 indicator 2-PDI-64-22, and are shown on the Unit 2 HVAC Flow Diagram. Walkdown data confirms that root valves exist for these indicators for all three units. Therefore, the subject drawings were revised to reflect the Unit 1 and 3 as-built configuration and to be consistent with the Unit 2 design drawing which shows the equivalent components.

System function and operation remain unchanged. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change. There were no credible failure modes associated with this documentation change. There was no unreviewed safety question created and no Technical Specification changes resulted.

**DCN W15674A - HPCI Gland Seal Modification - Unit 2**

Description/Safety Evaluation

This DCN provided changes to the HPCI Gland Seal Condenser lower flange cap drain lines which improved the draining function of this equipment, facilitated disassembly of the drain lines during condenser maintenance activities, minimized radioactive leakage, and reduced ALARA concerns.

The failure mode of concern for this DCN was loss of pressure boundary. The Seismic Class I portion of the drain piping was not postulated to fail during a DBA, while the seismic Class II portion may be postulated to fail during a DBA. Disassembly of the drain piping during maintenance on the HPCI Gland Seal Condenser may result in some contaminated liquid spillage; however, the 1/2 inch capped drain Tee was provided by this DCN for manual draining in order to minimize this type of spillage.

This DCN did not involve any equipment that could cause an accident, the HPCI Gland Seal Condenser itself did not serve any direct function in mitigating the consequences of accidents or AOT's, the piping being modified remains qualified to retain pressure boundary integrity. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W15755A - 480V Breaker Upgrade - Units 1, 2, and 3**

Description/Safety Evaluation

DCN W15755A provided the design to replace existing breaker trip devices for circuit breakers located on 480V Common Board 1 (Compartments 3C and 7C), 480V Common Board 3 (Compartments 4B and 9D), 480V Shutdown Board 1A (Compartment 5A) and 480V Shutdown Board 2A (Compartment 5A). The new RMS-9 trip devices eliminated nuisance trips that occurred with the previous trip units.

This change replaced existing breaker trip devices (EC-2 and EC-2A) with GE RMS-9 MircoVersa units. The affected breaker circuits supplied power for the Control and Service Air Compressors A, B, C, D, E and F.

Circuit breakers for compressors B, C, E and F are nonclass 1E devices (480V Common Boards 1 and 3) while breakers for Compressors A and D are Class 1E (480V Shutdown Boards 1A and 2A). The 1E trip devices for compressors A and D are electrically qualified per electrical calculation, and all the trip devices are seismically qualified per civil calculation. Setpoint and seismic calculations justify the change, and address device setpoints and seismic requirements.

Changing the affected Common Boards 1, 3 and Shutdown Boards 1A, 2A breakers' trip devices did not degrade the performance of the affected circuit breakers and improved circuit reliability. The modification had been designed in accordance with all applicable design criteria to ensure that the qualification, function, and operation of affected and associated safety-related systems were not adversely affected. The changes did not affect any parameters described in the basis for any Technical Specification. The modification did not cause the exceeding of any acceptance limit for any accident analysis, nor did it reduce the margin of safety. Therefore, this modification did not reduce any margin of safety for the systems included or any other systems. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D15822B - 1-47E610-64-1 Drawing Discrepancy Correction - Units 1, 2, and 3**

Description/Safety Evaluation

This DCN resolved drawing discrepancy with the installed configuration. TVA A/C drawing 1-47E610-64-1, incorrectly showed Flow Control Dampers FCO-64-65C and FCO-64-65D downstream of the Secondary Containment Equipment Access Lock Exhaust Fan. This drawing was revised to show the dampers upstream of the exhaust fan as indicated in PDD 90-392 and primary drawing 1-47E865-1. This change affected the Secondary Containment.

This documentation only change resolved a drawing discrepancy with the installed configuration. Secondary Containment system function and operation remained unchanged. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S15865A - Numerous Instrumentation Tabulation Drawing Errors Corrections - Units 1, 2, and Common**

Description/Safety Evaluation

This DCN revised instrument tabulation drawings for the indicated instruments as follows:

Flow annunciator 1-FA-066-048 was changed to 0-FA-066-048 to reflect correct UNID as indicated on the control diagram.

Flow indicating switch 0-FIS-066-048 information was changed to correct the indicated setpoint of 675 CFM to 2700 CFM and to indicate that it corresponds to a calibration differential pressure of 0.03 inches WC. This reflects the way the instrument has been previously operated and is consistent with the function of this switch, which is to indicate filter cubicle exhaust low flow.

Pressure differential indicator switch 2-PDIS-066-053 information was changed to add missing system setpoint of 8 inches WC.

Pressure indicating switches 2-PIS-066-021C and 2-PIS-066-021D setpoint information was changed from 35 PSIG to 3.5 PSIG to correct a drawing discrepancy. The instrument range is 0-15 PSIG.

Temperature indicating controller 2-TIC-066-109 information was changed to add missing system nominal control setpoint of 77°F.

Temperature switch 2-TS-066-108 was re-identified as 2-HS-66-108 because research has determined that this device is actually a temperature select switch and has no setpoint as indicated on elementary and connection drawings.

Temperature switch 2-TS-066-109 setpoint was changed from 77°F to 200°F in accordance with a GE Specification. This switch functions as a high temperature cutoff to protect the associated temperature element.

This change had no impact on the function or operation of the Offgas system or any other system. The Offgas Instrument Tabulations changes are in accordance with the related electrical calculations and system documentation. Therefore, the modification resulted in no new credible failure modes which were beyond the existing design.

The DCN did not invalidate any assumptions used in the UFSAR with respect to mitigation of DBAs. This change did not adversely affect the function or operation of any system used to mitigate the consequence of postulated accidents. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W15873A - Reactor Zone Supply Fans Sheave Size Modification - Unit 2**

Description/Safety Evaluation

During a performance of the Reactor Zone System Flow Verification Test, low individual grille flows in the Reactor Zone ventilation ductwork were identified. Design Change Request, DCR-3632, changed the sheave size to increase the flow through this ductwork. PRD BFP 910045 specifies corrective action which required that a revision be made to the flow diagram to include the Reactor Zone ventilation fans within the scope of Note 4 on this drawing. DCN W15873A implemented these changes to change the supply fans' motor and fan sheave sizes and revised the ventilation flow diagram to include both the Reactor Zone supply and exhaust fans within the scope of Note 4. This note explained that the individual grille flows are not critical as long as the total flows measured in the main branch ductwork are within specifications.

Air flow rates in the Reactor Building ventilation system are non-safety related parameters defined only in the design output documents. These flow rates are neither listed nor discussed in the Technical Specification; however, Reactor Building ventilation flow can affect secondary containment integrity/operability. The increased Reactor Zone air supply flows will provide for better flow balancing in the Reactor Building ventilation system and, thus, help ensure proper secondary containment pressure control during normal operation as specified by Technical Specification section 3/4.7.C. Better secondary containment pressure contr[illegible] will improve door operation between ventilation zones (i.e., door closure between the turbine building, outside entrances and between isolated reactor or refueling zones); thereby, enhancing the ability to maintain secondary containment integrity as defined in Technical Specification section 1.0.P. These actions improve ventilation system operation; therefore, no margin of safety was decreased.

The UFSAR Appendix G, SSA calculation ND-Q2000-88115 and the Design Criteria, section 3.2, require the Reactor Building ventilation system to mitigate the consequences of certain DBAs and AOTs by providing secondary containment isolation and RHR/CS cooling when required during these events. DCN W15873A changed the sheave size of the Reactor Zone ventilation supply fans (affects air flow rates) and revised the associated design basis documents. This DCN action affected only the non-safety related function of the Reactor Building ventilation system. In that the ventilation supply and exhaust fans are stopped during any event requiring secondary containment isolation and the SGTS will provide the ventilation/exhaust function to mitigate radioactive releases. Consequently, only the secondary containment isolation function and RHR/CS cooling functions are required post-accident. Therefore, based on these findings, DCN W15873A did not affect the ability of the Reactor Building ventilation system to mitigate the consequences of any DBA's or AOT's. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16025A - Drawing Corrections - Units 1, 2, and 3**

Description/Safety Evaluation

DCN D16025A was initiated to update drawings 0-47E830-1 through -5 and drawings 0-47E851-1 and -2 to resolve twenty discrepancies with the installed configuration. These DDs were walked down and were evaluated to determine the acceptability of the installed configuration.

This documentation only change was to correctly reflect the as-built condition of the Radwaste System as shown on drawings 0-47E830-1 through 5 and drawings 0-47E851-1 and -2. System function and operation remained unchanged. The only nuclear safety function performed by the Radwaste System is to maintain primary and secondary containment integrity. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16053A - Condensate Pipe Tunnel Sump Level Switch Setpoint - Units 1, 2, and 3**

Description/Safety Evaluation

This change established the high level setpoint for level switch 0-LS-77-353, revised the Instrumentation Tabulations to eliminate non existing alarms, and correctly identified level alarm 0-LA-77-353. Therefore, the response of the radwaste system to any DBA or AOT was unchanged.

The electrical calculation has established and justified the setpoint for the condensate pipe tunnel sump A level switch. This documentation change did not have any adverse impact on the function or operation of the level switch and resulted in no new credible failure modes which are beyond the existing design. No unreviewed safety question was created and no Technical Specification change resulted.

**DCNs D16060A, D16061A and D16062A - Drawing Corrections for Reactor Feedwater - Unit 1, 2, and 3**

Description/Safety Evaluation

DCN D16061A updated various drawings to resolve discrepancies with the installed configuration as follows:

1. In conflict with drawing 2-47E610-3-1, only one handswitch, 2-HS-3-188, is installed in panel 2-9-3 as shown on secondary connection diagrams 791E489 Sh 9 and SH 12, secondary wiring diagrams 45N2641-7 and 2-45N2631-8 and primary elementary drawing 2-730E927RF Sh 19. Drawing 2-47E610-3-1 was revised to eliminate these discrepancies. The Instrument Tabulations were reviewed to reflect installed handswitch 2-HS-3-188.

2. In conflict with drawing 2-47E803-5, valve 2-FCV-3-188B is a normally closed fail closed valve as shown on primary elementary diagram 2-730E927RF Sh 19 and primary control diagram 2-47E610-3-1. Drawing 2-47E803-5 was revised to eliminate this discrepancy.

3. In conflict with drawing 2-730E927RF Sh 19, valve 2-FCV-3-188A is a normally open fail open valve as shown on primary control diagram 2-47E610-3-1 and primary flow diagram 2-47E803-5. Drawing 2-730E927RF Sh 19 was revised to eliminate this discrepancy.

DCNs D16060A and D16062A updated drawings for Units 1 and 3 for similar drawing discrepancies.

This is a documentation only change to resolve drawing discrepancies with the installed configuration. Reactor feedwater system function and operation remained unchanged. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16071A - CAD Nitrogen Storage Tank Pressure Controller Change - Units 1, 2, and 3**

Description/Safety Evaluation

DCRs 2684 and 2719 changed the setpoints and calibration ranges for the CAD nitrogen storage tank pressure controllers (from 100 psig to 110 psig) and level transmitters (from 0-54 inch WC to 0-44.5 inch WC) respectively. However, 100 psig setpoint is specified on the Instrument Tabulations for pressure controllers 0-PC-84-4 and -15 for nitrogen storage tanks A and B. The calibration range for level transmitters 0-LT-84-2 and -13 is not shown on the Instrument Tabulations. DCN S16071A was a documentation only change, which revised appropriate Instrument Tabulations and issued NESSD setpoint documents to change the pressure controller's setpoint and add the level transmitter calibration range.

Design Criteria BFN-50-7084 required the nitrogen storage tanks be maintained at a minimum of 100 psig. Resetting the tank pressure controllers from 100 psig to 110 psig assured that this requirement was met. The new operating pressure of 110 psig is below the CAD system design pressure of 150 psig.

Technical Specifications require that 2500 gallons of liquid nitrogen be maintained in each storage tank. The level transmitter calibration range of 0-44.5 inches WC spans the full tank height. This calibration range resulted in more accurate level readings.

Additionally, based on walkdown information, this DCN corrected components numbers for instruments and valves shown on drawing 1-47E610-84-1.

This change had no adverse impact on the qualification, function, or operation of the CAD system. Therefore, the DBAs and AOTs described in Chapter 14 of the UFSAR were not adversely affected by this change. No unaffected safety question was created and no Technical Specification change resulted.

**DCN SW16097A - AHU Upgrades for Unit 3 Elevation 617 Relay Room - Unit 3**

Description/Safety Evaluation

This DCN provided redundant safety related cooling to the Relay Room on elevation 617 foot of the Control Building. Safety related and redundant cooling to the Relay Room is required to keep the Unit 3 cable spreading room on elevation 606 foot from exceeding its calculated mild environment qualification temperature during a design basis event. This is required to replace the existing safety related cooling cross-tie from the Unit 1/2 Main Control Room AHUs which no longer have sufficient capacity to cool both the MCRs and the Relay Room. The Relay Room contains the relays required to keep the switchyard (offsite power sources) operable and/or to de-energize selective switchgear to prevent damage from abnormal events. The functioning of these relays is highly desirable but are not a safety related function.

One redundant train of safety related cooling was provided by the existing Relay Room AHU which is being qualified for safety related service by this DCN. This unit is supplied with chilled water from the Unit 1/2 control bay chilled water (safety related and redundant) systems. In addition, the piping, valves, ductwork, and electrical supply system (480V Control Bay Vent Board A) for this AHU was upgraded to safety related service. The system operation is identical to the past operation.

A second train (standby) of safety related cooling was added via a new AHU located in the Relay Room. This unit was supplied chilled water from cross-ties from the Unit 3 control bay chilled water systems A and B. This AHU tapped into the existing HVAC supply ductwork for the Relay Room and will draw suction from the base of the unit. Class 1E power was supplied from 480V Control Bay Vent Board B. The system operation was based on continuous chilled water flow with a local temperature switch to control fan operation and a high temperature alarm on local panel 25-165.

In addition, as part of this change, the existing partial-height wall (temporary) that separates the Relay Room relay area from the operators lunchroom area was removed and replaced with another partial height wall. This new wall was a different configuration to accommodate the new AHU and to enlarge the operators lunchroom area. A door through this wall was configured such that sufficient airflow can pass from the relay area back to the suction of the AHU.

The failure mode for the existing AHU is the same as before, and the failure mode for the new AHU is similar to the failure modes analyzed for the Control Bay Cooling Units which are served by the same chilled water supply and are enveloped by existing UFSAR analyses. Appropriate procedural revisions were made to ensure a common mode failure of both Unit 3 Control Bay Chilled Water Systems will not occur.

The accident evaluations in the UFSAR did not address the Relay Room directly, only the ability of the room to be cooled. Therefore, since this modification ensured the cooling of the Relay Room was maintained, it cannot initiate any reactor transients or accidents. No ur viewed safety question was created and no Technical Specification change ulted.

**DCN W16119A - Radwaste Pump Impeller Modification - Units 1, 2, and 3**

Description/Safety Evaluation

DCN W16119A was a modification to the Waste Collector and Waste Surge pumps by downsizing the pump impellers from 9 1/16 inch diameter to 8 1/2 inch diameter.

The Waste Collector and Waste Surge Pumps are located in the Radwaste Building of the BFN plant. The original design flow rate for the Waste Collector and Waste Surge Pumps was 440 gpm. In order to facilitate efficient operation of the Waste Collector/Surge System Processing, the current filtration flow rate range has been lowered to 100 - 150 gpm and this will result in extended exposure time of fluids in the resin bed for better ion exchange and longer service period.

With larger impeller size, the reduction in flow rate resulted in an operating pressure range of 160 - 180 psig, which exceeds the 150 psig design pressure. This causes the potential to overpressurize the Waste Collector/Surge Piping System. The use of smaller pump impellers to match the 150 psig design pressure maintains the operation of the Waste Collector/Surge System at the lower flow rates and avoids damaging piping and associated equipment/components in the system. This DCN also replaced the pressure gauges downstream of these pumps. These gauges were upgraded from 0-150 psig range to 0-200 psig range. The portion of Liquid Radwaste System covered by this DCN was not safety-related and is not required for the safe shutdown of BFN Unit 2.

The Liquid Radwaste System has a feature of cross connections between the subsystems and providing additional flexibility for processing of the Radwaste. A mechanical failure by a single pump will not render the system inoperable. Therefore, it can be concluded that the changes will not create new credible failure modes. Credible mechanism of failure for the components involved under the DCN that can result in release of radioactive materials to the plant environment is the postulated piping rupture. The design temperature and pressure of Waste Collector/Surge system is 140°F and 150 psig respectively, which meet the criterion of moderate energy piping in NUREG-0800, section 3.6.1. The probability of a postulated high energy pipe break is therefore, not credible. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16134A - Drawing Correction for 1-47E610-43-2 - Units 1, 2, and 3**

Description/Safety Evaluation

DCN S16134A updated drawing 1-47E610-43-2 to resolve discrepancies with the installed configuration as identified by PDD 91-049. Specifically, conductivity transmitters 2-CIT-43-25B and 3-CIT-43-25C, as labeled in the plant in accordance with SDSP 12.3 were incorrectly depicted with Unit 1 prefixes on drawing 1-47E610-43-2. Conductivity elements 2-CE-43-25B and 3-CE-43-25C, which were shown on 2-47E610-43-2 and 3-47E610-43-2, respectively, provide signals to the above transmitters. The above components provide water quality information for the outlet filters on the Unit 1, 2, and 3 fuel pool demineralizers. Drawing 1-47E610-43-2 was revised to eliminate these discrepancies. The Instrument Tabulations were also revised to reflect the proper unit prefix for these components.

Sample and water quality system function and operation remained unchanged. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16164A - Documentation Changes to Drawings 2-47E610-43-1 and -2 - Unit 2**

Description/Safety Evaluation

DCN S16164A was issued to update drawings 2-47E610-43-1 and 2-47E610-43-2 to correct drawing discrepancies identified in PDD 91-053. This DCN incorporated documentation changes only and did not change the existing plant configuration. The Sample and Water Quality System was affected by this DCN.

TVA A/C drawing 2-47E610-43-2 has been revised to correctly reflect the turbidity measurement instrument numbers as shown on Instrumentation Tabulation drawing 0-47B601-043 or 2-47E601-43-1; deleted the Main Steam Startup Sample Station and showed the four sample lines to this sample station as capped; the line with Manometer 43-15 and associated Note 2 removed and capped downstream of valve 43-631; sample bomb for non-condensibles downstream of valve 43-631A removed and line capped.

No physical work was performed. System function and operation remained unchanged. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16165A - Sample and Radwaste System Drawing Discrepancies - Unit 1, 2, and 3**

Description/Safety Evaluation

DCN S16165A was issued to update drawings 1-47E910-43-2, 0-47E830-3, and 7-47A365-43-16 to correct drawing discrepancies with the installed configuration. This DCN incorporated documentation changes only and did not change the installed plant configuration. TVA drawing 1-47E610-43-2 has been revised to correctly reflect the installed pipe configuration from the waste demineralizer outlet valve 0-77-810 to CE-43-25E and valve 0-43-742. Also, to correctly reflect the installed pipe configuration from the floor drain filter outlet to CE-43-25F and valve 0-43-743. TVA drawing 0-47E830-3 has been revised to assign a new UNID for valve 0-77-2369 in the floor drain filter outlet line downstream of valve 0-77-861. TVA drawing 7-47A365-43-16 has been superseded by drawing 0-47A365-43-16 and the UNID for valve 1-43-743 is changed to 0-43-743. System function and operation was not adversely affected by this change. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16196A - Drawing Upgrade for Offgas Temperature Controllers - Unit 2**

Description/Safety Evaluation

DCN S16196A addressed PDD 91-056 which involved the replacement of non-Class 1E temperature controllers 2-TC-66-76 and 2-TC-66-90 and updated various drawings to correctly show the terminal points on the new non-Class 1E temperature controllers.

Temperature controllers 2-TC-66-76 and 2-TC-66-90 were originally Ogden ETR-15-TA temperature controllers. TVA has since replaced the original ETR-15-TA controllers with Ogden ETR-20-2A temperature controllers. The terminal points of the control wires for these temperature controllers are shown on vendor drawings with wire numbers instead of terminal points. To facilitate maintenance, this DCN added unique numbers to the affected drawings for the associated terminal blocks. In short, this DCN evaluated the replacement temperature controllers and clarified wire terminations to the new temperature controllers 2-TC-66-76 and 2-TC-66-90.

This documentation only change resolved drawing discrepancies with the installed configuration. Offgas system function and operation were not adversely affected by the use of replacement temperature controllers. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16212A - HPFP Cooling Water Regulator Valve Setpoint Change - Units 1, 2, and 3**

Description/Safety Evaluation

The diesel driven fire pump pressure regulating valve (0-PCV-26-111) controls the volume and pressure of raw water to the diesel engine. The diesel engine block was required to be maintained at 170° F + 5° F when operating. The periodic surveillance of the diesel driven fire pump indicated that the regulating valve had to be kept fully open to allow maximum water flow through the engine block to maintain the desired engine temperatures. The pressure gauge (PI-26-112) downstream of the regulating valve read approximately 12 psig during maximum flow (valve fully open). This falls outside the range to 15-20 psi, as shown on the flow diagram 1-47E836-1 revision 015. Hence the operating range for the regulating valve was revised to 5-20 psi. Note that by closing the regulating valve, the flow decreased and the gauge downstream indicated lower pressure. A fully open valve will deliver maximum flow and indicate maximum pressure on the gauge. The upper limit is not being revised because even with the valve fully open, the pressure downstream did not exceed 12 psi. The revised lower limit accommodates the present operating conditions and leaves sufficient margin to reduce flow if required.

This change did not affect the operating capability of the regulating valve, which can be operated from a fully open to partially closed position to maintain the desired engine block temperature. The change did not impose additional pressures on the system which may cause equipment failure. Additionally, adjustment of the pressure regulator valve position did not affect the diesel driven fire pump capability to deliver required flows/head to meet the Technical Specification performance requirements. Hence, this change did not have any adverse impact on the HPFP system nor did it affect the Appendix R analyses. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W16229A - Vacuum Pump Installation for Containment Inerting Nitrogen Tank - Units 1, 2, and 3**

Description/Safety Evaluation

The scope of DCN W16229A was to provide the engineering design and documentation for the east yard Containment Inerting System Liquid Nitrogen Tank vacuum pump installation. The electrical system for the vacuum pump installation was redesigned to meet the National Electric Code and applicable TVA standards. The pump was connected to the tank through the evacuation valve provided with the tank. This piping was documented as built. A weather enclosure and pump skid was designed to protect the pump from the elements of the weather.

The Containment Inerting System is used to purge the primary containment until the atmosphere contains less than 4 percent oxygen, prior to each start up.

The inerting system also continued to supply makeup gas, required by temperature changes and leakage, during planned operations. The primary containment is held at a slight positive pressure by the inerting system as a means of leak-rate monitoring. The purpose of the vacuum pump is to provide an insulation carrier for the liquid nitrogen tank to prevent the nitrogen from boiling off through the relief valves.

These changes did not alter the design basis of the CIS provided in TVA Design Criteria BFN-50-7076 and cannot be an initiating event for the DBA described in the UFSAR. Since the activity was designed and constructed in accordance with the established criteria, did not alter the design basis of the CIS, and cannot initiate a DBA, there was no increase in the probability of an accident previously evaluated in the UFSAR.

A failure associated with the activity would result in a loss of vacuum jacketing the nitrogen tank, boiling off the nitrogen through relief valves and an inability to supply nitrogen to the containment. This is the same result which would occur with the previous design; therefore, the activity cannot cause a new type of accident to occur. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W16318A - Upgrading Operator's Lunchroom - Unit 3**

Description/Safety Evaluation

DCN W16318A involved the relocation and upgrade of the operators lunchroom. The new location is on elevation 617.0 foot in the control bay at the existing women's restroom and locker area outside the Unit 3 control room. The facility has a refrigerator, stove, ice maker, sink and adequate counter and storage space. Also, this DCN involved the modification of the present janitor's closet, located outside the Unit 3 control room on elevation 617.0, and the shower in the women's restroom (same as above), into a new women's restroom.

These facilities are contained within a Class I Structure, but are classified as non-safety related. They are not essential for preventing an accident which would endanger the public health and safety, and are not essential for the mitigation of the consequences of these accidents. The block walls removed are not seismic and the existing seismic block wall remained qualified. The electrical supply to these facilities was from non-safety related source (240V Lighting Board 3A, to LC 307). The piping and plumbing to these facilities were connected to non-safety related systems existing in this area. The HVAC for these facilities were connected to the existing Control Bay Air Conditioning System. The exhaust ducts were connected to a non-safety related duct and the supply duct were connected to the Main Control Room Air Handling Unit 3A/3B supply duct.

For Figure 1.6-12 and the associated tables, the existing facilities preformed no safety-related function and the rearranged facilities perform no safety-related function.

For Figure 10.12-2, the Control Bay Air Conditioning System performs no nuclear safety functions, only protective safety functions. This system is designed to maintain the required environmental conditions for plant operators and safety-related control equipment as its protective safety function. The function of this modification was non-safety related and this additional supply duct portion of the system was designed not to prevent the safety-related portion from performing its function.

The capability of the fire protection system in the rearranged areas was adequate to detect and extinguish a fire for the amount of combustibles present in this area. Therefore, this new fire load will not interfere with the ability of the control building to retain its present fire rating. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W16318B - Modifying Operations Lunchroom - Unit 3**

Description/Safety Evaluation

Modified the operations lunchroom size, i.e., reconfiguration of the non-safety related area, and the Control Bay HVAC System which can not initiate a DBA or create no new failure modes.

The HVAC system serving the main control room area is being modified to allow 450 CFM to be diverted to the new lunchroom. The loss of 450 CFM, out of approximately 9000 CFM being delivered by AHU's 3A/3B will not affect the environmental conditions of the safety related areas including the main control room. The AHU's have sufficient excess capacity to accommodate the additional load. The added supply duct has been qualified and meets the applicable code requirements for the system. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16385A - Reactor Building Ventilation Drawing Discrepancies - Unit 2**

Description/Safety Evaluation

DCN S16385A was issued to update drawing 2-47E2865-12 identifying the train orientation of each core spray pump room air cooling unit and to document the cooling air flow capacity of each unit. The drawing was revised to indicate that cooling unit A (capacity 12700 cfm) is located in the core spray A/C and CRD pump room, and cooling unit B (capacity 10000 cfm) is located in the Core Spray B/D pump room. This was shown on the drawing based on a walkdown.

System function and operation remained unchanged. The nuclear safety functions performed by this system remained unaffected. There were no new credible failure modes associated with this documentation change beyond those enveloped by the existing design. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16386A - HPCI Drawing Discrepancies - Units 1 and 2**

Description/Safety Evaluation

DCN S16386A was issued to update 1-47E850-1 and 2-47E850-1 to resolve three discrepancies with the installed configuration. These discrepancies were evaluated to determine the acceptability of the installed configuration as follows:

1. Drawing 1-47E850-1 did not reflect a hose station 1-26-1242 (Coord E-6) which was installed at elevation 617 foot 0 inches in the Turbine Building. The hose station was shown on the drawing based on its depiction of the As-Designed drawing.

2. Drawing 1-47E850-1 did not agree with the installed piping configuration. The three inch line to the pre-action system (Coord A-8) was shown to branch off from the 6 inch line before the 2 inch line to the spreading room. This condition was reversed in the field. The drawing was revised to reflect the as-installed condition.

3. Drawing 2-47E850-1 did not show a vent line (Coord A-6) which is installed. This drawing was revised to show the installed vent line configuration. The materials used were acceptable from a pressure and temperature standpoint and the piping configuration was acceptable from a stress standpoint. The vent line was acceptable per the NFPA code.

There were no new credible failure modes associated with this documentation change that were not enveloped by the existing design. These documentation changes did not adversely affect system function or operation, nor can they be the initiator of an accident. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W16409A - RCIC Electronic Overspeed Elimination - Units 1, 2, and 3**

Description/Safety Evaluation

The modification resulted in the removal of the electronic overspeed trip function and related devices from the RCIC turbine control system at BFN. The previous design of the RCIC turbine overspeed detection/protection system consisted of mechanical overspeed trip which was set at 125% of rated turbine speed, and a supplemental electronic overspeed trip feature set at 110% of rated speed. The subject modification affected the electronic overspeed trip function only.

The electronic trip, which was remotely resetable, was originally incorporated in the design in the expection that it would provide turbine trip activation below the limiting speed, thereby avoiding actuation of the mechanical trip, which cannot be reset remotely.

Consistent with BFN experience, the trip initiating acceleration transient is typically so rapid that the electronic trip cannot terminate the transient before the mechanical trip also actuates. Therefore, the electronic overspeed device did not perform its intended function, was not specifically required for overspeed protection, and was a potential source for spurious trips. GE has eliminated this feature from subsequent designs and recommended its removal on existing designs.

The installed electronic overspeed trip monitor had failed and caused the RCIC trip and throttle valve (2-FCV-71-09) and trip solenoid (2-XX-71-9) to energize. Energizing this solenoid prevented the reset of the trip and throttle valve inhibiting further system testing.

The removal of the electronic overspeed trip device did not cause malfunction of a different type as evaluated in the UFSAR. RCIC turbine overspeed trip protection will be achieved by the mechanical overspeed trip device that was tested satisfactorily. The current overspeed trip setting of 125% rated speed remained unchanged. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W16435A - RPV Instrument Reference Leg Re-route - Unit 2**

Description/Safety Evaluation

There are two RPV instrument taps that each supply two condensate pots in the drywell, giving a total of four different reference legs exiting containment. The previous configuration had different divisions of RPV level and pressure instruments on common reference legs. As a consequence, any perturbations or transients on one reference leg (such as caused by maintenance activities which manipulate isolation valves on the reference legs) may cause a reactor scram, and several scrams in the past have been attributed to this. As part of the Scram Frequency Reduction Program, it was recommended that the RPV RPS instrumentation be modified to place each RPS logic channel for the subject instruments on a separate reference leg. With this configuration, a perturbation or transient on any one reference leg will at worst cause a half-scram from the subject pressure and level instrumentation. This modification did reroute the reference legs of instruments 2-PT-3-22D, 2-PT-3-22AA, 2-LT-3-203C, and 2-LT-3-203B.

The change involved minor instrument piping modifications (less than ten feet per reroute). This did not affect the subject instrumentation operability or operational characteristics, nor impact any existing Nuclear Engineering generated Setpoint and Scaling documentation related to the subject instrumentation.

The activity was a piping configuration change and did not introduce any new failure modes or increase the probability of a failure. The change did not alter the in-service operational characteristics of the subject instrumentation in any manner, and will reduce the possibility of spurious scrams due to maintenance activities; therefore, reduce the challenges to plant safety systems. In addition, the material and installation was of equal or better qualification than the previous piping. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D16529A - HVAC Drawing Corrections - Units 1, 2, and 3**

Description/Safety Evaluation

This was a documentation only change to resolve DDs with the as-installed configuration. The DCN had no adverse impact on the air conditioning (cooling-heating) system function or operation. Therefore, the design basis accidents and operational transients described in section 14 of the UFSAR were unaffected by this change. This documentation change did not affect system function or operation, nor can it be the initiator of an accident.

There were no new credible failure modes associated with this documentation change. The installed plant configuration remained unchanged. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D16566A - Instrument Number Corrections - Unit 3**

Description/Safety Evaluation

DCN D16566A addressed PDD 91-147 which identified a discrepancy involving instruments located in the Unit 3 diesel generator building which were incorrectly shown with a Unit 0 designation on the Instrument Tabulations and drawings 3-47E850-4, 3-47E850-10 and 3-45E643-10. These instruments are tagged in the field with a Unit 3 designator.

This was a documentation only change to resolve DDs with the as-installed configuration. Fire protection system function and operation remained unchanged. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16591A - Drawing Discrepancy Corrected - Unit 2**

Description/Safety Evaluation

DCN S16591A addressed PDD 91-155 which identified an inconsistency between drawings 2-47E610-3-1 and 2-47E803-5 and the as-installed configuration for a capped tee downstream of penetration X-26B (outside the drywell) on a Unit 2 reactor vessel level indication sensing line. The capped tee was not installed in the field.

The capped tee was added to drawings 2-47E610-3-1 and 2-47E803-5 for resolution of DDP 003-001. A recent walkdown performed for PDD 91-155 found no capped tee. A review of DCDTS for drawings 2-47E610-3-1 and 2-47E803-5 found no indication of modifications made to this line since the issuance of DDP 003-001. Therefore, DDP 003-001 was determined to be incorrect for its resolution of the capped tee and the capped tee never existed. This DCN corrected this drawing discrepancy by revising flow diagram 2-47E803-5 and control diagram 2-47E610-3-1 to eliminate the capped tee. This was a documentation only change to resolve a DD with the installed configuration. Reactor feedwater system function and operation remained unchanged. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D16593A - Drawing Valve Number Correction - Unit 2**

Description/Safety Evaluation

DCN D16593A was issued to update drawing 2-47E844-1 to correct a DD. This flow diagram showed the same valve number (2-24-1180) for both the vent and drain valves for Hydrogen Cooler 2A. The installed valves were tagged correctly as 2-24-1181 ("2A H2 CLR Drain") and 2-24-1180 ("2A H2 CLR Vent"). Therefore, the subject flow diagram was revised to reflect the installed valve tagging. This change affected the RCW system. System function and operation remained unchanged. There are no nuclear safety functions performed by this system. Therefore, the DBAs and AOTs described in Chapter 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D16594A - Drawing 3-47E844-1 Corrections - Unit 3**

Description/Safety Evaluation

This was a documentation only change to correctly reflect the vent and drain valves 3-24-1180 through 3-24-1187 of the Hydrogen Coolers 3A through 3D on drawing 3-47E844-1 to reflect the as-installed condition of the RCW system. System function and operation remain unchanged. There were no new credible failure modes associated with this documentation change that were not enveloped by the existing design. There are no nuclear safety functions performed by this system. Therefore, the DBAs and AOTs described in Chapter 14 of the UFSAR were unaffected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D16601A - Unit Preferred 120 VAC Drawing Corrections - Unit 2**

Description/Safety Evaluation

This was a documentation only change to resolve minor drawing discrepancies with the installed configuration. Unit preferred 120 VAC system function and operation remained unchanged. Therefore, the DBAs and AOTs described in Section 14 of the UFSAR were unaffected by this change.

The installed plant configuration remained unchanged. Since this documentation change did not add any new credible failure modes or adversely impact the function or operation of any plant system it cannot increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16651A - Units 1, 2, and 3 Cable and Loads Restrictions - Units 1, 2, and 3**

Description/Safety Evaluation

The DCN involved placing restrictions on operation of Unit 1, 2, and 3 cables and loads. These restrictions will prevent on-going work in Units 1 and 3 and operations in Unit 2 from affecting Unit 2's ability to achieve and maintain safe shutdown. The restrictions did not alter the function or performance of the affected Unit 2 safety systems as they have been analyzed to assure safe operation of Unit 2. They did not impact other systems ability to perform their safety function nor did they change this or other systems ability to interact as required to achieve and maintain safe shutdown and no safety related loads were affected. No equipment was added, modified, or removed from BFN as a result of the activity. The restrictions placed on various loads by this activity did not introduce any new failure modes or alter existing failure modes. The only credible failure mode of this activity is a personnel error as failing to follow unit separations procedures and beginning work on one of the affected circuits without obtaining proper authorization. Personnel error is an analyzed Chapter 14 UFSAR AOTs. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W16666A - TIP Modifications - Unit 2**

Description/Safety Evaluation

The modification constituted a one for one exchange of selected equipment, e.g., gamma detectors replaced thermal neutron detectors, triaxial cables replaced coaxial cables, and high gain Flux Probing Monitors replaced existing Flux Probing Monitors. The failure mode of the system was the same as before the modification.

The direct replacement of the neutron TIP detectors and the Flux Probing Monitors upgraded the TIP system with more reliable and accurate equipment than those previously installed. The function of the TIP system is to provide signals to the process computer which represent core axial power distribution. This TIP function is not directly safety-related. The modification did not alter the function or degrade the performance of these or other systems. Additionally, the TIP system performs no function, other than the primary containment isolation, in any accident described in Chapter 14 of the UFSAR. Since the hardware modification did not affect the primary containment isolation function of the TIP system, and the only software changes were the process computer databank which were tested to assure that the correct changes were made to the TIP databank, this modification did not increase the probability of occurrence of an accident previously evaluated in the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16699A - Mechanical Logic Drawing Updates - Unit 2**

Description/Safety Evaluation

This was a documentation only change to the following logic diagrams: MS System, CDWS, SLC System. RWCU System, HPCI System, and RPS. This DCN updated these logic diagrams as required by licensing commitment control number NCO 900118001. Previously, these logic diagrams have been frozen for several years and were not maintained. TVA committed to revising these drawings to ensure that they reflect the latest control circuitry of the affected systems. CRLD BFEP EEB 91037 R0 was processed to replace the existing UFSAR figures with these revised drawings. These Mechanical Logic Diagrams were prepared from the latest revisions of Mechanical Control Diagrams, Schematic diagrams, and wiring diagrams. The DBAs and AOTs for each of the systems affected by this DCN were listed in the respective design criteria. This documentation only change did not affect any DBA or AOT requirements for any of the affected systems. This DCN change affected only the logic diagrams for these systems which were updated to reflect the existing control configurations for these systems. There was no credible failure mode created by the activity since there were no physical changes to the plant and no additional contamination release points created. Therefore, this document change did not impact the capability of these systems to perform their design basis functions. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16700A - Mechanical Logic Drawing Updates - Units 1, 2, and 3**

Description/Safety Evaluation

The modification was a documentation only change to numerous Mechanical Logic Diagrams. The affected systems by this DCN were: RHRSW, EECW, RBCCW, CS, Fuel Pool Cooling and Demineralizing. TVA committed to revising these drawings to ensure that they reflected the latest control circuitry of the affected systems. Thirteen different drawings were revised.

This change did not affect any DBA or AOT requirements for any of the affected systems. This DCN change affected only the logic diagrams for these systems which were updated to reflect the existing control configurations for these systems. Therefore, this document change did not impact the capability of these systems to perform their design basis functions. There was no credible failure mode created by the activity since there were no physical changes made to the plant and no additional radiological release points were created. The design inputs used to develop the logic diagrams were not changed. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16754A - EECW Valve Alignment to Control Bay Chillers - Units 1, 2, and 3**

Description/Safety Evaluation

The activity ensured that continuous cooling is provided to the control bay chillers by the safety related EECW system during normal operation and all abnormal events. This change eliminated a credible failure mode that existed in that a single failure of one of the check valves that provide the system interface pressure boundary between RCW and EECW could have led to a loss of the EECW system by providing a drainage path through the non-safety related RCW system. Furthermore, the design analyses associated with the EECW system assumes that the Control Bay chillers are continuously fed by the EECW system and has determined the configuration to be acceptable. The Restart Testing Program also accounted for the chillers being in continuous service by EECW and has found the configuration to be acceptable. Since the EECW system will be operating in a mode already determined to be acceptable, there are no new failure modes created by this change; and therefore there are no credible failure modes associated with this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16803A - Condensate Pipe Tunnel Sump Alarm Correction - Units 1, 2, and 3**

Description/Safety Evaluation

DCN D16803A was issued to provide a means for Nuclear Engineering to resolve PDD 91-243. A review of DCN D16803A has determined that no design input exists to support the field configuration. Therefore, DCN D16803A has been changed to DCN S16803A, which requires the issuance of a calculation to create a design basis for the setpoints associated with the plant configuration. In addition, applicable Instrument Tabulations have been revised to incorporate the new setpoints.

No credible failure modes exist for this DCN since this DCN only provided a justification by analyses of the actual field configuration setpoints associated with level switch 0-LS-77-350. None of the DBAs as described in Section 14.6 of the UFSAR were impacted by this DCN. None of the AOTs as described in Section 14.5 were impacted by this DCN. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN Q16813A - SGTS Dresser Coupling Repair - Units 1, 2, and 3**

Description/Safety Evaluation

The use of elastomer repaired the Dresser coupling seal; therefore, stopped the water inleakage into the SGTS exhaust piping. This use did not affect the structural integrity of the SGTS piping nor the ability of the SGTS to perform its safety function.

The SGTS provides mitigation of the consequences of an accident by minimizing the release of radioactive materials to the environment. The use of elastomer did not affect the ability of SGTS to provide filtering, nor did it reduce the ability of SGTS to maintain secondary containment. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN [illegible]5A - HVAC Drawing Change - Units 1, 2, and 3**

Description/Safety Evaluation

This activity was a documentation change only which allowed a broader range of air flow into the communication room from the control bay cooling system during flow balancing. The design previously called for 1500 cfm +/-10% and this change increased the acceptance band to +15% -10%, by adding a note to drawing 0-45E865-4.

The control bay cooling system is a support system for various other safety related systems which have controls and instrumentation located in control bay spaces. These systems which include RHR, CS, RPS, and PCIS are utilized to mitigate all of the accidents and transients in Chapter 14 of the UFSAR. Since this change did not alter the design cooling of any safety related equipment space, none of these accidents or transients were affected in any way. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W16867A - RHRSW Piping Temperature Upgrade - Unit 2**

Description/Safety Evaluation

This change increased the design temperature of the RHRSW Water piping at the discharge of the heat exchangers downstream of the heat exchanger outlet valves out to the point where two heat exchangers form a common discharge line. The previous design temperature of 150° F was increased to 350° F. This change affected A, B, C, and D heat exchangers on Unit 2.

A note was added to the flow diagram to indicate that sets of companion heat exchangers (A and C, B and D) shall be operated together whenever the outlet temperature of one heat exchanger exceeds 150° F in order to reduce the temperature of the water flowing through the piping downstream of the common point. The purpose of this change was to protect piping and components downstream of the common point from temperatures in excess of their 150° F qualification temperature that are anticipated during shutdown cooling operation. Some buried piping components and the effluent radiation monitors are not qualified above 150° F.

This change also replaced a support on the Unit 2 RHRSW piping associated with Stress Problem N1-223-5R. This support (2-B450-H051) was located on the discharge line from the 2B RHR Heat Exchanger. The previous support was utilized until the new support was installed so as to ensure the continued operability of the associated piping throughout the implementation of the change. The result of this modification ensured the seismic qualification of the RHRSW discharge piping for all applicable operating modes.

This change affected only the shutdown cooling mode of RHR and associated RHRSW operation. The functions relied upon in accidents do not involve the higher temperatures or the changed operating procedure associated with this change and are thus not affected. No credit is taken in accident events for shutdown cooling. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S16873A - Loading Limitations and Breaker Alignments for 480V RMOV Boards - Units 1, 2, and 3**

Description/Safety Evaluation

Prior to the issue of this DCN, use of the alternate power supplies for 480V RMOV Boards 1A, 1B, 1C, 2A, 2B, 2C, 3A, 3B, and 3C were restricted by DCN S16651 because their use was outside of analyzed conditions. The calculations supporting this DCN analyzed the safety-related 480V power distribution system for electrical adequacy during use of the alternate power supplies. Based upon these calculations, this DCN removed the restrictions placed upon the use of the alternate power supplies, and documented maximum loading limits, and required 480V shutdown board breaker alignment for normal operation of these boards while being supplied power from their alternate power supplies (there is one case when the loading limit is reduced for operation on diesel generator power). The operation of these boards on their alternate power supplies could adversely affect Unit 2 safe shutdown capability by the possibility of overloading the alternate power supplies. Loading limitations and limitation on circuit breaker alignment were imposed for Unit 2 normal reactor power operation to ensure that the capability of these boards to supply power to all automatically started safety-related loads was maintained. Supporting calculations were revised to analyze the capability of the alternate power supplies to supply power to the affected 480V RMOV boards, providing the maximum loading limits, and 480V shutdown board breaker alignments. Notes specifying the maximum loading limits, and required 480V shutdown board breaker alignments were incorporated into design output documents (480V board singleline drawings). The maximum loading limits imposed by this DCN are applicable when any of the subject 480V RMOV boards alternate breakers are closed. The closing of the alternate breaker of the operating unit's associated RMOV boards results in limits being placed upon continued normal reactor power operation per the special requirement. The difference between the loading limits and the power system maximum calculated capability has been reserved for the automatic start of safety-related loads in response to accident conditions, since an accident must be postulated to occur while a subject 480V RMOV board alternate breaker is closed. Thus, with the loading limits observed, the 480V safety-related power distribution system will still have sufficient reserve power capacity to automatically start and run required affected 480V RMOV board safety-related loads.

The reactor operator may have to trip loads at his discretion to maintain 480V loading at or below the specified limits during normal reactor operation when the affected 480V RMOV boards are being supplied power from their alternate power supplies. Closing 480V RMOV board 2A and 2B alternate breakers is acceptable because the transfer to an alternate power supply will result in a time limit on continued normal reactor power operation defined in the special requirements of the safety evaluation. Closing the 480V RMOV board 2C alternate breaker for an indefinite amount of time is acceptable since there are no safety-related loads on the board. Closing one of the non-operating unit's associated RMOV boards alternate breakers for an indefinite amount of time is acceptable since the loads fed by these boards are not immediately needed in the event of a DBA on the operating Unit.

The safety-related loads supplied power from the affected 480V RMOV boards and its upstream opposite division 480V shutdown board will continue to automatically start and operate after the transfer to the alternate power supply just as they did before the transfer. Nuclear safety of the operating Unit is not affected by closing one of its 480V RMOV board alternate breakers provided that continued normal reactor power operation is time limited to the period allowed by the special requirement. Nuclear safety of the operating Unit is not affected if one of the non-operating Unit's associated RMOV boards alternate breaker is closed indefinitely since any of the operating Unit's support loads fed by the RMOV boards are not immediately needed in the event of a DBA on the operating Unit.

In order to use the alternate feeders to the 480V RMOV boards 1A, 1B, 1C, 2A, 2B, 3A, 3B, and 3C the following criteria must be adhered to. These criteria were incorporated into operating procedures. The procedures annotated the source of these requirements as being this safety evaluation.

1. During normal power operation of a given Unit, closing of the alternate feeders to the associated RMOV boards is only permissible when there is a failure of the normal supply. Return to the normal supply shall be within twelve hours or the reactor shall be in cold shutdown within the next twenty-four hours. Indefinite operation of the alternate feeders of the RMOV boards in the other Units is permissible only when that Unit's reactor is in cold shutdown. In any case, the loading limitations shown on the 480V shutdown and 480V RMOV boards single lines shall not be violated.

2. Preventative maintenance on the normal feeder breakers for the RMOV boards or their feeders is permitted only during shutdown condition.

3. When a 480V shutdown board is being fed by its emergency transformer, the alternate feeder breaker to an RMOV board fed from that shutdown board cannot be closed.

No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W16924A - Hypochlorite Building Fire Protection Sprinkler Addition - Unit Common**

Description/Safety Evaluation

The use of the Raw Water Fire Protection System by the Hypochlorite Building for sprinklers did not impact the demand for the system during a special event fire in the Reactor Building or at a main transformer. The worst case demand is bounded by these areas and a simultaneous fire at another location is not postulated to occur.

The Design Basis for the Raw Water Fire Protection System is to supply a minimum of 200 gpm at 65 psig to the Reactor Building roof wye-hose connections or to supply water to a burning main transformer and the two adjacent main transformers. This modification was on non-safety-related site facilities and therefore, outside the scope of "Fire Protection of Safety Shutdown Capability". Only a fire event at this facility is applicable and it will have no impact on the DBAs or AOTs.

Since the piping was designed for the system pressure/temperature conditions at the location of the modifications, no credible failure modes which could impact nuclear safety were introduced. A fire event at the facility will not impact any safety-related components. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D16926A - Radiation Monitors Drawing Correction - Units 1, 2, and 3**

Description/Safety Evaluation

Potential Drawing Discrepancy number 91-264 identified that the TVA unique identifiers assigned to Vendor (GE) identifiers are shown incorrectly on drawings 2-828E307-1 R3, 2-828E307-2 R2, 2-45N2631-25 R2, 2-45N2684 R1, 828E469 RF Revision 5, 828E470 RF Revision 5 and 729E530-1 Revision D. RM-90-272A should be GE# 16-62A but shows 16-63A, RM-90-272B should be GE# 16-63A but shows 16-62A, RM-90-273A should be 16-62B but shows 16-63B and RM-90-273B should be GE# 16-63B but shows 16-62B.

The recorder pen assignments per ECN P0324 (Work Plan 2008-85 and 2009-85) are inconsistent with pen assignments per ECN P7134 (WP# 2709-88). Instruments RE-90-272A/273A are Drywell Radiation Sensors and RE-90-272B/273B are Suppression Chamber Radiation Sensors. The instrument recorder RR-90-272CD shows Division I red pen as Drywell High Range Radiation RM-90-272B and -273C, blue pen Drywell operating Range Radiation RM-90-272A and -273A and the green pen Suppression Chamber Radiation RM-90-272B/-273B (typical for RR-90-273CD).

DCN D16926A reviewed design output documents to reflect the as-installed configuration for Radiation instrumentation of the Drywell and Suppression Chamber. This D-DCN corrected GE UNID Numbers to agree with field installation.

This D-DCN revised a UFSAR drawing which did not change qualifications of existing equipment. The D-DCN did not degrade equipment reliability of Primary Containment System. This change did not involve any equipment that can itself cause a DBA or AOT. Therefore, this change did not increase the probability of occurrence of previously analyzed accident. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W16945A - RHRSW Discharge Line Repair - Units 1 and 2**

Description/Safety Evaluation

DCN W16945A installed a welded flange with a bolted on blind flange in the RHRSW system discharge prior to the discharge pipe passing under the cool water channel. This modification was required to isolate the piping because a piping failure had allowed warm RHRSW discharge water to escape into the cool water channel. The system normally discharges into the Wheeler Reservoir through a bank of twelve 14 inch pipes. This pipe is the one most western of the four Unit 1 discharge lines which are required for Technical Specification operability of Unit 2. This DCN isolated the failed pipe such that three of the four lines are available to discharge RHRSW flow to the river. Design calculations show that the function of the system was not affected. The flange was designed to meet the pressure/temperature requirements of the system and the piping configuration is seismically qualified.

The RHR system was designed to mitigate the consequences of the DBA's listed in Design Criteria BFN-50-7023, section 3.2. The changes made to the plant configuration did not affect the ability of the system to perform its intended functions. The discharge piping was repaired to ensure that the system will continue to function as designed without allowing RHRSW to discharge into the cool water channel. The failure of the piping components added by this modification did not cause any DBA's to occur. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S17008A - Drawing 2-47E2865-12 Update - Unit 2**

Description/Safety Evaluation

This change revised the design air flow requirements on Drawing 2-47E2865-12. This change was necessary to document the changes in Reactor Building ventilation exhaust design flows from the main steam pipe vault (steam vault), elevation 565 foot general area, and the Torus room.

During performance of the power ascension program for Unit 2, it was noted that the temperature in the steam vault (approximately 170°F as observed remotely from the main control room) was approaching the reactor trip setpoint of 189°F. As a result, TI, Main Steam Tunnel Ventilation Adjustment, was performed to determine if the design air flow from the steam vault could be increased by re-balancing the exhaust flows in other areas of the Unit 2 Reactor Building. Specifically, exhaust flow from the steam vault was increased while exhaust flows from the Northeast quadrant of elevation 565 foot and the torus room were reduced. The temperature in the affected areas was monitored at intervals throughout the duration of the TI. The results of the TI indicate that the temperatures in areas where ventilation flows were reduced were unaffected. The temperature in the steam vault (as indicated by 2-TS-1-60A in the main control room) stabilized at 150°F during performance of the TI. This temperature had been recorded at 167°F just prior to the flow adjustment discussed above.

Based upon the results of the above referenced TI, the design air flow rate from the steam vault was increased from 6,000 to 8,000 CFM. The design flow rates for the two branch lines (upstream of damper 2-64-513) which take suction from the area above the Torus were reduced from 3,100 to 2,600 CFM each. Also, the design flow rate of the branch line upstream of damper 2-64-538 which takes suction from the Northeast quadrant of elevation 565 foot were reduced from 5,500 CFM (2,750 per grille) to 4,500 CFM (2,250 per grille). These changes were incorporated onto Drawing 2-47E2865-12 by this DCN.

The UFSAR and the Design Criteria require the Reactor Building ventilation system to mitigate the consequences of certain DBAs and AOTs by providing secondary containment isolation and RHR and CS cooling. This change increased the design air flow from the Unit 2 main steam pipe vault (steam vault) and decreased design air flows from the Unit 2 Reactor Building elevation 565 foot and the Torus room. This action only affected the non-safety related portion of the Reactor Building ventilation system. The ventilation supply and exhaust fans are stopped during any event requiring secondary containment isolation. Consequently, only the secondary containment isolation and RHR/CS cooling functions are required post-accident. Therefore, this change did not affect the Reactor Building ventilation system's ability to mitigate the consequences of any DBAs or AOTs. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W17010A - MIC Sampling for EECW - Units 1, 2, and 3**

Description/Safety Evaluation

MIC is a problem causing a decrease in the useful life of carbon steel piping. Monitoring the rate of corrosion of a system is important so that the rate of corrosion inhibitor injection can be controlled.

DCN W17010A provided internal monitoring capability of the EECW while the system was in full operation. Access fitting assemblies, in which the monitoring device was placed, were permanently attached to the process pipe. The monitoring devices were installed in the EECW common discharge header piping for RHR pump room coolers A/C (Units 1, 2, 3), EECW common supply headers for Unit 3 DG engine coolers, EECW North/South supply headers and EECW North/South Diesel Generator Building supply headers.

Three different types of corrosion monitors will be used at each general location. One type monitor, a polarization resistance probe, can provide instantaneous or continuous corrosion monitoring. This method is based on a measurement of the apparent resistance of a corroding electrode when it is polarized by a small voltage. The resistance being inversely proportional to corrosion. A second type, strip coupon holder, provides a visual indication of the type of corrosion which may be occurring in the monitored system. This monitoring method uses a piece of metal (coupon), of the same chemical composition as the process system. which is exposed to the corrosive medium of the process system for a period of time, then removed and analyzed. A third type monitor, injection strip coupon holder, consists of a standard metal coupon holder with a tee access fitting, whereby the coupon can be flushed or corrosion inhibitors can be injected into the system.

The access assembly was designed to maintain its structural integrity under the system design and corrosion conditions. Although it is highly unlikely, an improperly attached coupon could come loose from its holder. Normally the coupon would fall and remain stationary until its loss was discovered during changeout. However, even under high flow conditions, the small flatshape (3 x 3/4 x 1/8 inch) would not allow it to be picked up and act as a missile nor block flow in the pipe. Thus, this assembly will not degrade the reliability of the equipment which must operate during an emergency.

The access fittings did not interfere with the required flows nor decreased the structural integrity of the EECW piping during a DBA. In fact, this monitoring feature will increase the reliability of the EECW piping system by detecting problems related to piping corrosion. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W17015A - HPCI Time Delay Relay Installation - Unit 2**

Description/Safety Evaluation

DCN W17015A installed a time delay into the HPCI low suction pump trip circuit. A time delay relay replaced the existing interposing control relay, identified as 2-63-73-29-1. The trip signal from the ATU will be delayed for 4.7 seconds by the time delay relay. The time delay relay plugged into the existing mounting block, which the previous control relay utilized.

This DCN removed TACF 2-91-3-73. All wiring changes preformed under TACF 2-91-3-73 were removed. The modification installed a time delay relay qualified to Class 1E requirements so that its qualifications were equivalent to those of the existing loop components. The function of the low suction pressure trip circuit is to trip the HPCI pump if the pump was started with closed suction valves. This modification did not degrade equipment reliability of the HPCI system by permitting the pump to mitigate a short low suction pressure transient in lieu of tripping instantaneously on low suction pressure. In fact, this modification decreased nuisance trips of the HPCI system and thus increases HPCI reliability.

The failure modes associated with the installation of the time delay relay in the pump trip circuit were the same as those for the previous trip circuitry (i.e., failure of the pressure transmitter 2-PT-73-29-1 or failure of the analog trip unit). These failure modes were a failure to send a trip signal to the HPCI pump turbine controls or to send a spurius trip signal to the HPCI pump turbine controls. As a result, HPCIs ability to mitigate the consequences of any DBAs or AOTs remained unaffected. Therefore, the failure modes associated with this change were bounded by the existing failure modes of the pump trip circuit. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S17018A - Drawing Corrections - Units 1, 2, and 3**

Description/Safety Evaluation

This Safety Evaluation was written to support updating Mechanical Logic Diagrams. This was a documentation only change to the logic diagrams. RHR system, TIP System, SLC system, RFW System, PCIS, Reactor Water Recirculation system, 250 VDC Power System, Radiation Monitoring system and MS System were affected by this DCN. This DCN updated these logic diagrams as required by licensing commitment control number NCO 900118001. Previously, these logic diagrams have been frozen for several years and were not maintained. TVA committed to revising these drawings to ensure that they reflect the latest control circuitry of the affected systems. CRLD BFEP EEB 91048 R0 was processed to replace the existing UFSAR figures with these new drawings. These Mechanical Logic Diagrams were prepared from the latest revisions of Mechanical Control Diagrams, Schematic diagrams, and wiring diagrams.

There was no credible failure mode created by the activity since there were no physical changes made to the plant and no additional radiological release points were created. The design inputs used to develop the logic diagrams have not changed. No hardware or function was changed by this DCN. The revised logic diagrams provided clarifications of system function along with other design documents and procedures and did not introduce any new failure modes or alter existing failure modes. Therefore, this document change did not impact the capability of these systems to perform their design basis functions. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S17019A - Numerous Drawing Updates - Units 1, 2, and 3**

Description/Safety Evaluation

DCN S17019A issued numerous updated Mechanical Logic Diagrams. This was a documentation only change to the logic diagrams. The affected systems by this DCN were: MS System; PCIS; Reactor Water Recirculation System; RCIC; Radwaste System; CRD System; and Neutron Monitoring System. Previously, these logic diagrams have been frozen for several years and were not maintained. TVA committed to revising these drawings to ensure that they reflect the latest control circuitry of the affected systems.
CRLD BFEP EEB 91041 R0 was processed to replace the existing UFSAR figures with these new drawings. These Mechanical Logic Diagrams were prepared from the latest revisions of Mechanical Control Diagrams, Schematic diagrams and wiring diagrams.

There was no credible failure mode created by the activity since there were no physical changes made to the plant and no additional radiological release points were created. The design inputs used to develop the logic diagrams did not change. No hardware or function was changed by this DCN. The revised logic diagrams provided clarifications of system function along with other design documents and procedures and did not introduce any new failure modes or alter existing failure modes. This documentation only change did not affect any DBAs or AOT requirements for any of the affected systems. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D17021A - Deletion of Non-Existant RHR Pressure Switch - Units 1, 2, and 3**

Description/Safety Evaluation

The DCN change involved deletion on the UFSAR Figure 7.4-6B, Instrument Tabulations, Q-List, and Mechanical Control Diagram of a switch which has never existed; but, was added to these items due to an error on the Instrument Tabulations. The switch 2-XS-74-127 was erroneously added to the control loop for 2-FCV 74-53, but serves no function for valve control, has not been in the schematic diagram, has no described function in the UFSAR, and did not exist in the plant; but, showed up by error on the UFSAR Figure. Deletion of the switch had no functional change on the control of valve 2-FCV-74-53. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W17036A - Radwaste Drum Compactor Installation Corrections - Units 1, 2, and 3**

Description/Safety Evaluation

CAQR BFP890122 was initiated due to the removal of a Radwaste baler and installation of a new radwaste drum compactor without proper documentation as required by SDSP-8.12, Plant Modification - Overall Process. Installation of the compactor created several electrical and mechanical drawings errors (primary and critical drawings). Implementation of the corrective actions listed in this DCN were to be used to close the stated CAQR.

This DCN also documented the permanent installation of the radwaste Steam Jenny Power Supply. The Steam Jenny is used for in-line cleaning of the radwaste floor drain and waste collector filter elements.

The scope of this DCN was to complete the following:

a. Completely remove the radwaste drum compactor from the Radwaste Building. This included revising plant electrical and mechanical drawings to reflect removal of the waste baler. These components were no longer required since the wastes are being shipped offsite for compaction; a backup compactor is available in the decontamination area if it again becomes necessary to compact materials onsite.

b. Electrical supply to existing drum compactor was modified by replacing the existing 20 amp breaker in compartment 3D1 of 480V Radwaste BD 1 with a 100 amp breaker, cable and general purpose receptacle in the radwaste baler room.

The newly installed 100 amp breaker, cable and receptacle will be used by the radwaste steam jenny (90 amps) whenever radwaste in-line filter element cleansing is required. Previously, the steam jenny was connected to 8D1 480V radwaste board 1, which contains a 100 amp breaker. Whenever the steam jenny is in service, no other loads can be connected to the remaining six wall receptacles. Installation of a wall receptacle in the radwaste baler room for steam jenny use eliminated the need to caution order the control wall receptacle usage.

The waste baler and the drum compactor were installed to compact compressible dry solid wastes and small non-compressible wastes into containers. These components are no longer required since the wastes are being shipped uncompacted in approved containers for offsite compaction by a qualified vendor.

The guidance provided in IE Circular 80-18 is intended to prevent inadequacies in evaluations which have allowed radiological safety hazards to occur unidentified and, therefore, to remain unevaluated and uncorrected. In two particular cases, the inadequately evaluated system changes resulted in failures that caused an uncontrolled release of radioactivity to the environment. This activity only eliminated the non-safety equipment used for compaction of dry solid wastes such that each container shipped from the site contains a smaller quantity of wastes (be weight). This change did not involve a waste treatment process, it did not alter the existing provisions for controlling releases of radioactive materials, for monitoring and/or sampling of the process prior to shipment, not did it involve potentially explosive mixtures which require evaluation in accordance with the Standard Review Plan Section 11.3, "Gaseous Waste Management System." For these reasons, it was concluded that the activity did not create any potential for uncontrolled releases beyond those previously evaluated by the NRC. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W17046A - MIC Sampling for RHRSW - Units 1, 2, and 3**

Description/Safety Evaluation

MIC is a problem causing a decrease in the useful life of carbon steel piping. Monitoring the rate of corrosion of a system is important so that the rate of corrosion inhibitor injection can be controlled.

DCN W17046A provided internal monitoring capability of the RHRSW while the system was in full operation. Access fitting assemblies, in which the monitoring device was placed, were attached to the process pipe. The monitoring devices were installed in the RHRSW common discharge header piping for RHR heat exchangers (HX) A/C (Units-1,2,3) & B/D (Units 1,2), RHRSW supply piping to RHR HX B (Units 1,2,3) and RHRSW pump A2, B2, C2, and D2 discharge piping.

Three different types of corrosion monitors will be used at each general location, with the exception of the Unit 1 common discharge header for HX A/C, where space allows for only two types of monitors. One type monitor, a polarization resistance probe, can provide instantaneous or continuous corrosion monitoring. This method is based on a measurement of the apparent resistance of a corroding electrode when it is polarized by a small voltage. The resistance being inversely proportional to corrosion. A second type, strip coupon holder, provides a visual indication of the type of corrosion which may be occurring in the monitored system. This monitoring method uses a piece of metal (coupon), of the same chemical composition as the process system, which is exposed to the corrosive medium of the process system for a period of time, then removed and analyzed. A third type monitor, injection strip coupon holder, consists of a standard metal coupon holder with a tee access fitting, whereby the coupon can be flushed or corrosion inhibitors can be injected into the system.

The access assembly was designed to maintain its structural integrity under the system design and corrosion conditions. Although it is highly unlikely, an improperly attached coupon could come loose from its holder. Normally, the coupon would fall and remain stationary until its loss was discovered during changeout. However, even under high flow conditions, the small flatshape (3 x 3/4 x 1/8 inch) would not allow it to be picked up and act as a missile nor block flow in the pipe. Thus, this assembly will not degrade the reliability of the RHR HXs and RHRSW piping system.

The access fittings did not interfere with the required flows nor decreased the structural integrity of the RHRSW piping during a DBA. In fact, this monitoring feature will increase the reliability of the RHRSW piping system by detecting problems related to piping corrosion. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN G17047A - 480V AC Circuit Breaker Upgrade - Units 1, 2, and 3**

Description/Safety Evaluation

The existing GE 480 Volt Ak-15 and Ak-25 circuit breakers primarily employ either "EC" trip devices, "SST" trip devices, or a "Power Sensor" trip device. These trip devices are obsolete and represent excessive maintenance and repair items with the "EC" device (most commonly used) having an estimated life of approximately three to five years. The current GE product for this application is the MicroVersa Trip RMS-9 unit (sensing and tripping system). The RMS-9 Conversion Kit is a pre-engineered unit representing product improvement and has been identified by GE as providing improved flexibility, accuracy, reliability, and long life.

DCN G17047A provided engineering support to use the generic equipment replacement process to replace the obsolete trip devices with MicroVersa Trip RMS-9 Conversion Kits. This was accomplished by revising BFN Generic Substitution Engineering Requirements Specification (ER-BFN-NTB-001) to document procurement and installation requirements. These requirements were applicable for circuit breaker conversions accomplished through a contract with GE, using GE facilities, or for conversions implemented by BFN maintenance personnel. The settings for the RMS-9 Conversion Kits were established based upon conformance with design requirements and criteria.

The circuit breaker trip device is used to initiate breaker tripping when the circuit current exceeds predetermined values. This represents a protective function to limit the scope and extent of equipment involvement during circuit faults or overloads. The installation of trip devices assured to meet existing design requirements and qualified for their application and location did not represent a fault initiator and hence did not increase the probability of an accident previously evaluated in the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W17065A and W17066A - Division I and II RHR MOV Interlocks - Unit 2**

Description/Safety Evaluation

The scope of DCN W17065A was to design and install electrical interlocks between the Division I RHR Shutdown Cooling Suction MOVs, 2-FCV-74-2 and 2-FCV-74-13, and the RHR Pressure Suppression Chamber Isolation MOV, 2-FCV-74-57. Also for Division II, RHR Shutdown Cooling Suction MOVs, 2-FCV-74-25 and 2-FCV-74-36, and the RHR Pressure Suppression Chamber Isolation MOV, 2-FCV-74-71.

The interlocks were designed to provide a means by which the RHR Shutdown Cooling Suction MOVs cannot be opened if the RHR PSC Isolation MOV is open. The interlocks will also prevent the opening of the RHR PSC Isolation MOV if either of the RHR Shutdown Cooling Suction MOVs is open.

The interlocks were designed to prevent inadvertent draining of the reactor vessel in response to an INPO SOER.

Failure of the interlocks would result in loss of control power to the MOV and the MOV will remain in its last position. This result is the same as the previous design as these valves were not required to have redundant power sources. Loss of power to these valves and their subsequent inoperability is not an initiating event for the DBAs listed in the UFSAR.

The activity utilized spare limit switches on the valves which were identical to other limit switches already in use on this system. The electrical cable and connections were designed and were constructed to the same standards and of the same materials as existing Class IE systems. As such the interlocks have the same response characteristics as existing interlocks on the RHR system. The addition of the interlocks did not alter the operational characteristics of the system since no new flow paths were created and there were no changes in the system flow characteristics.

This change did not cause system vibration, water hammer, corrosion, thermal cycling, or degradation of the environment in which the equipment operates and thus cannot result in the system operating beyond its design limits. The only change in the system interfaces was that the interlocks will prevent an improper alignment of valves which could have resulted in inadvertent draining of the reactor. These valves are not required to be open at the same time and the present operating instruction, also prohibit their being opened simultaneously. Thus, the activity only changed the proper valve alignment from an operator controlled activity to an automatically controlled activity which will lessen the likelihood of an AOT. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S17071A - SJAE Pressure Switch Setpoint Reduction - Units 1, 2, and 3**

Description/Safety Evaluation

DCN S17071 reduced the setpoints for the SJAE pressure switches. SJAE Pressure Switches PS-1-150, -152, -166, and -167 provide an isolation function to prevent volatile concentrations of hydrogen in the Offgas System. GE SIL Number 497 recommended a reduction in the setpoint value to decrease the loss of steam through the Offgas System. This type of setpoint change also reduced inadvertent isolations of the Offgas System.

The non-safety related SJAEs continuously remove noncondensable offgas from the main condenser during operation. This offgas includes hydrogen which is produced from the radiolytic dissociation of water. Operating steam supplied to the second and third stages of the SJAEs is provided to dilute the hydrogen concentration of the SJAE effluent. Calculation MD-N2001-910158 determined analytically the minimum steam operating pressure required to dilute the hydrogen below flammable concentrations.

Revising the setpoints for the steam supply to the SJAEs pressure switches did not adversely affect system function or operation. Therefore, the failure modes associated with this change were bounded by the existing design. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN G17099A - PACO Sump Pump Bearing's Cooling Upgrades - Units 1, 2, and 3**

Description/Safety Evaluation

This DCN G17099A was a generic change to allow field modification to existing PACO sump pumps to install water-lubricated bearings in place of self-lubricated bearings.

The revised design included the installation of a short line running from the impeller housing to the intermediate bearing holder, conveying sump pumpage water through the bearing for cooling and lubrication. This change assured adequate bearing cooling/lubrication without impact to pumps operation, or the need to pipe cooling water from an independent source. This design configuration has been used successfully by the vendor on similar applications without adverse effects.

The activity affected the non safety-related portion of the Radwaste System. The sump pumps remove potentially radioactive liquids from floor and equipment drain sumps, and pump them to collector tanks in the Radwaste Building. They are not required for primary or secondary containment and are not required to prevent or mitigate the consequences of any DBA or ATS.

The credible failure modes of the proposed activity consist of the tubing line becoming clogged or broken thereby preventing the supply of cooling water to the bearing. This could result in overheating of the bearing and loss of the pump. This scenario is not considered likely since the debris is screened out prior to entering the housing and the loads on the tubing (other than internal pressure) are negligible. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D17103A - Correction of 4kV Common Boards A & B's Feeder Designation-Units 1, 2, and 3**

Description/Safety Evaluation

The DCN change involved correction of the "NC" and "NO" designation on the board incoming feeder breaker to designate the supply from the Unit Station Service Transformer as the normal supply to 4KV Common Boards A & B. This change corrected the UFSAR Figure 8-4-1a to agree with the single-line diagram, schematic, connection diagr...., and UFSAR description. The error was created when the Key Diagram, which is the UFSAR Figure, was As-Constructed out of sequence for the As-Designed revisions.

The change did not alter the function or performance of any safety systems. It did not impact other systems ability to perform their safety function nor did it change this or other systems ability to interact as required to achieve and maintain safe shutdown. This DCN did not involve the addition of any new equipment or alteration of any existing equipment. Therefore, this DCN did not create a possibility for an accident of a different type than any evaluated previously in the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S17117A - Drawing Corrections - Units 1, 2, and 3**

Description/Safety Evaluation

DCN S17117A was initiated in order to make miscellaneous revisions to drawings 47E200-18 and 3-47W200-19. These drawings provide the location and laydown requirements for the placement of refueling equipment and disassembled Reactor components on the Refueling Floor, elevation 664'-0", of the Reactor Building. The individual revisions are identified as follows:

Relocated laydown space for one 39 ton dryer-separator shield plug from the north side of each Reactor to the south side of each Reactor.

Relocated laydown space for the four 5 1/2 ton refueling slot plugs for each Unit from the north side of the Reactor to the south side of each Reactor. The present laydown space will remain as an alternate laydown space to provide some flexibility.

Corrected drawing 3-47W200-19 to show the correct weight of the 50 ton dryer-separator shield plug.

Added Note G to drawing 3-47W200-19 allowing the Reactor head strongback, dryer-separator sling, RPV service platform and support, and the stud tensioner and support frame to be placed on the refuel floor as required to facilitate refueling floor activities.

Added Note H requiring that Nuclear Engineering be contacted to assess floor loading and cribbing requirements for equipment to be placed on the refueling floor which are not addressed by drawings 47E200-18 and 3-47W200-19.

Added Note J which allows the alternate laydown space for the four 5 1/2 ton refueling slot plugs to be used as alternate laydown space for one 39 ton dryer-separator shield plug when space is not needed for the refueling slot plugs.

The loadings on the refueling floor and the lifts by the reactor building crane have been evaluated as addressed by the UFSAR. These changes did not represent new loadings to either the Reactor refuel floor or to the Reactor Building crane. The lifts were already identified in MMI-199. No other safety-related equipment was affected. Therefore, the changes did not increased the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D17118A - Updated UFSAR Figure 8.7-4c - Units 1, 2, and 3**

Description/Safety Evaluation

The DCN change involved correction of the UFSAR Figure which is vendor drawing 2-731E753-3, but did not involve any physical change to the plant. This change corrected the UFSAR Figure to agree with changes made for ECN P0916, DCN W1073, and DCN W6842 which included safety evaluations in each change package to justify the physical changes made to the plant. Previous changes have been documented on TVA initiated drawings 2-45E2647-2 and 2-45E2647-3 which were kept up to date, but vendor drawing 2-731E753-3 (which is the UFSAR Figure) was overlooked and not updated when changes were made. This caused the need to update the UFSAR Figure which was the only purpose of DCN D17118A.

No equipment was added, modified, removed, or operated in any different manner as a result of the activity. Therefore, no new or unanalyzed failure modes were introduced.

The change is a documentation change only and did not impact any systems ability to perform its safety function nor did it change this or any other systems ability to interact as required to achieve and maintain safe shutdown. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN S17127A - HPFP End Cap Drain Holes - Units 1, 2, and 3**

Description/Safety Evaluation

This change was to the HPFP System. It allowed for small diameter holes (1/8 inch) to be drilled into two each 2 1/2 inch diameter threaded end caps. The purpose of these lines is for attaching fire protection hoses in case of fire, or for testing. The end caps are used only to protect the piping when the hoses are not being used. These lines are located outside the building on the turbine building roof. The holes in the caps will allow water that may enter the lines because of problems such as leaking valves to drain and prevent freezing when ambient temperature falls below 32 degrees Fahrenheit. Therefore, this change enhanced the reliability of the affected HPFP lines during periods of outside temperatures below 32 degrees Fahrenheit. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN D17183A - Drawing Corrections - Units 1, 2, and 3**

Description/Safety Evaluation

This documentation only change removed the locking requirements depicted on the flow diagrams for various valves in the Units 1 and 2 and Unit 3 DSAS.

Revising the locking requirements for the DSAS valves could result in inadvertent misalignment of one of the affected valves in the safety-related portion of the DSAS which could result in a loss of system pressure. This condition could result in the initiation of an alarm locally and in the main control room. This system response is consistent with that which would occur subsequent to failures of the non-safety-related portions of this system. Therefore, the failure mode associated with this change is bounded by the existing design.

TS section 4.9.A.1.a. requires testing of the DSAS air compressors to check for operation and their ability to recharge the air receivers. However, the subject valves themselves neither perform, nor can their mispositioning prevent, any post-accident mitigation functions. The affected valves are neither listed nor discussed in the TSs. Consequently, there is no Tech Spec safety margin affected by this change. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W17184A - Correction of Cable and Conduit Separation Suffixes - Units 1, 2, and 3**

Description/Safety Evaluation

The nature of this DCN was to correct the separation suffix for cables and conduits to reflect their correct separation group requirements. Additionally, the UFSAR description and figure related to conduit and cable tagging were changed to provide the appropriate level of detail. These changes were not specific accident or operational transient related.

The credible failure modes were not equipment related but were instead designer or installer related. These failures modes were:

1. Incorrect identification of the cable or conduit number (including the separation suffix).
2. Incorrect tag selection for a particular installation.
3. Incorrect installation of the conduit or cable tag.

There were no credible failure modes associated with the UFSAR change.

The cable and conduit separation suffix corrections implemented by this DCN enhanced Browns Ferry's compliance with its separation criteria, thus insured the integrity of safety-related equipment.

This DCN did not result in circuit or cable configuration changes (excluding the replacement of existing tags); therefore, there was no equipment function, operation, or failure mode changes. The field implementation of this DCN was to replace existing cable and conduit tags with tags containing the corrected separation suffix.

The field implementation was controlled by design engineering specifications, plant drawings, and plant implementing procedures; thus, the field implementation did not result in equipment function, operation, or failure mode changes.

Additionally, revising the UFSAR to remove inappropriate detail information did not result in changes to the tagging requirements or tagging practices. Therefore, implementing this DCN did not directly or indirectly reduce the margin of safety as defined in the basis for any Technical Specification. No unreviewed safety question was created and no Technical Specification change resulted.

**DCN W17310A - Replacement of Obsolete Flow Transmitters - Unit 2**

Description/Safety Evaluation

Previously, flow transmitters 2-FT-074-076, 2-FT-077-006, and 2-FT-077-016 were GE (GEMAC) Type 555 differential pressure transmitters. These transmitters were obsolete and were maintained by using parts from spared transmitters. This DCN replaced the above mentioned transmitters with Rosemont 1151 Series transmitters.

Per Electrical Calculation ED-N0999-910252 R0, Rosemont Series 1151DP differential pressure transmitter was an acceptable replacement for the GEMAC Model 555.

The scope of this modification was to remove the GEMAC transmitters from the panels and mount the Rosemont transmitters in t eir place. The instrument sense lines from the panel isolation valves to the panel drain valves and the conduit connections were reconfigured to accommodate the new transmitters.

This modification did not change the function of the instrument loops. Flow transmitter 2-FT-74-76 displays the RHR flow to the fuel pool cooling system. Flow transmitters 2-FT-77-006 and -016 totalize drywell leakage flows. The transmitters were installed to TVA Seismic Class II ensuring pressure boundary integrity. The probability of an accident previously evaluated in the UFSAR is not increased. No unreviewed safety question was created and no Technical Specification change resulted.

## 1991 RELEASE SUMMARY

| MONTH | Gaseous Releases: Fissions & Activation Products (Ci) | Gaseous Releases: Iodines (Ci) | Gaseous Releases: Particulates >8 day half-lives (Ci) | Gaseous Releases: Tritium (Ci) | Liquid Releases: Fission & Activation Products (Ci) | Liquid Releases: Tritium (Ci) | Liquid Releases: Dissolved Noble Gases (Ci) | Liquid Releases: Gross Alpha (Ci) |
|---|---|---|---|---|---|---|---|---|
| January | ND | ND | ND | 8.72E-02 | 1.67E-02 | 1.83E-02 | ND | ND |
| February | ND | ND | 1.01E-04 | 4.26E-02 | 9.88E-03 | 1.50E-02 | ND | ND |
| March | ND | ND | 2.51E-04 | 7.61E-03 | 2.01E-02 | 1.73E-02 | ND | ND |
| April | ND | ND | 3.86E-05 | 1.20E-02 | 1.56E-02 | 1.67E-02 | ND | ND |
| May | 7.06E-07 | ND | 1.58E-05 | 3.23E-02 | 4.91E-02 | 4.27E-02 | ND | ND |
| June | 4.75E+01 | ND | 6.08E-05 | 4.18E-02 | 3.60E-02 | 1.82E-01 | 8.82E-03 | ND |
| July | 2.16E+02 | 7.17E-03 | 1.07E-02 | 1.69E-02 | 3.77E-02 | 1.34E-01 | 1.19E-02 | ND |
| August | 4.53E+02 | 3.07E-02 | 3.35E-03 | 2.85E-01 | 1.23E-01 | 6.34E-01 | 3.04E-02 | ND |
| September | 6.10E+02 | 2.10E-02 | 2.87E-03 | 5.07E-01 | 1.47E-01 | 7.40E-01 | 2.27E-02 | ND |
| October | 2.71E+02 | 8.35E-03 | 7.92E-04 | 4.95E-01 | 3.38E-01 | 2.27E+00 | 7.05E-02 | ND |
| November | 2.60E+02 | 1.70E-02 | 2.05E-04 | 1.91E-01 | 9.31E-02 | 1.04E+00 | 7.20E-03 | ND |
| December | 2.42E+02 | 9.80E-03 | 1.01E-03 | 1.07E+00 | 1.02E-01 | 8.50E-01 | 7.54E-03 | ND |

ND is for non-detectable.
Variation in the data for gaseous releases have been correlated with the numbers of operating fans. There were no excursion of interest nor releases which exceeded Tech Spec limits.

T E N N E S S E E   V A L L E Y   A U T H O R I T Y
BFN RADIATION EXPOSURE SYSTEM



NUMBER OF PERSONNEL AND MAN-REM BY WORK JOB FUNCTION
TOTAL NUMBER OF INDIVIDUALS

MO=REACTOR OPS SURVEILLANCE

| GROUP | NUMBER OF PERSONNEL (> 100 M-REM) STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL PERSONS | TOTAL MAN-REM STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL M-REMS |
|---|---|---|---|---|---|---|---|---|
| MAINTENANCE PERSONNEL | 406 | 10 | 117 | 533 | 19.913 | 0.129 | 3.592 | 23.634 |
| OPERATING PERSONNEL | 71 | 0 | 2 | 73 | 9.992 | 0.000 | 0.077 | 10.069 |
| HEALTH PHYSICS PERSONNEL | 96 | 0 | 15 | 111 | 15.073 | 0.000 | 0.231 | 15.304 |
| SUPERVISORY PERSONNEL | 25 | 0 | 7 | 32 | 0.551 | 0.000 | 0.228 | 0.779 |
| ENGINEERING PERSONNEL | 78 | 9 | 149 | 236 | 4.709 | 0.148 | 61.308 | 66.165 |
| MO | 676 | 19 | 290 | 985 | 50.238 | 0.277 | 65.436 | 115.951 |

MO=ROUTINE MAINTENANCE

| GROUP | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL PERSONS | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL M-REMS |
|---|---|---|---|---|---|---|---|---|
| MAINTENANCE PERSONNEL | 483 | 13 | 171 | 667 | 102.021 | 2.734 | 28.440 | 133.195 |
| OPERATING PERSONNEL | 71 | 0 | 0 | 71 | 3.166 | 0.000 | 0.000 | 3.166 |
| HEALTH PHYSICS PERSONNEL | 97 | 0 | 7 | 104 | 10.939 | 0.000 | 0.052 | 10.991 |
| SUPERVISORY PERSONNEL | 24 | 0 | 6 | 30 | 1.807 | 0.000 | 0.295 | 2.102 |
| ENGINEERING PERSONNEL | 88 | 11 | 35 | 134 | 12.487 | 1.332 | 2.947 | 16.766 |
| MO | 763 | 24 | 219 | 1006 | 130.420 | 4.066 | 31.734 | 166.220 |

MO=IN-SERVICE INSPECTION

| GROUP | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL PERSONS | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL M-REMS |
|---|---|---|---|---|---|---|---|---|
| MAINTENANCE PERSONNEL | 16 | 1 | 14 | 31 | 0.078 | 0.017 | 0.038 | 0.133 |
| OPERATING PERSONNEL | 3 | 0 | 0 | 3 | 0.044 | 0.000 | 0.000 | 0.044 |
| HEALTH PHYSICS PERSONNEL | 17 | 0 | 1 | 18 | 0.121 | 0.000 | 0.000 | 0.121 |
| SUPERVISORY PERSONNEL | 2 | 0 | 1 | 3 | 0.000 | 0.000 | 0.000 | 0.000 |
| ENGINEERING PERSONNEL | 11 | 0 | 1 | 12 | 0.135 | 0.000 | 0.005 | 0.140 |
| MO | 49 | 1 | 17 | 67 | 0.378 | 0.017 | 0.043 | 0.438 |

MO=SPECIAL MAINTENANCE

| GROUP | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL PERSONS | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL M-REMS |
|---|---|---|---|---|---|---|---|---|
| MAINTENANCE PERSONNEL | 5 | 0 | 0 | 5 | 0.014 | 0.000 | 0.000 | 0.014 |

1991 OCCUPATIONAL EXPOSURE DATA

REXP"219
RUN DATE: 02-08-92
RUN TIME: 06:28:52

T E N N E S S E E   V A L L E Y   A U T H O R I T Y
BFN RADIATION EXPOSURE SYSTEM



NUMBER OF PERSONNEL AND MAN-REM BY WORK JOB FUNCTION
TOTAL NUMBER OF INDIVIDUALS

| | NUMBER OF PERSONNEL (> 100 M-REM) | | | | TOTAL MAN-REM | | | |
|---|---|---|---|---|---|---|---|---|
| OPERATING PERSONNEL | | | | | | | | |
| HEALTH PHYSICS PERSONNEL | 2 | 0 | 0 | 2 | 0.000 | 0.000 | C.000 | 0.000 |
| SUPERVISORY PERSONNEL | 0 | 0 | 1 | 1 | 0.000 | 0.000 | U.025 | 0.025 |
| ENGINEERING PERSONNEL | 0 | 0 | 0 | 0 | 0.000 | 0.000 | 0.000 | 0.000 |
| MO | 7 | 0 | 1 | 8 | 0.014 | 0.000 | 0.025 | 0.039 |

MO=WASTE PROCESING

| GROUP | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL PERSONS | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL M-REMS |
|---|---|---|---|---|---|---|---|---|
| MAINTENANCE PERSONNEL | 45 | 1 | 24 | 70 | 0.265 | 0.000 | 1.426 | 1.691 |
| OPERATING PERSONNEL | 42 | 0 | 2 | 44 | 0.277 | 0.000 | 0.325 | 0.602 |
| HEALTH PHYSICS PERSONNEL | 77 | 0 | 15 | 92 | 3.931 | 0.000 | 4.282 | 8.213 |
| SUPERVISORY PERSONNEL | 2 | 0 | 2 | 4 | 0.011 | 0.000 | 0.008 | 0.019 |
| ENGINEERING PERSONNEL | 28 | 1 | 5 | 34 | 0.852 | 0.001 | 0.002 | 0.855 |
| MO | 194 | 2 | 48 | 244 | 5.336 | 0.001 | 6.043 | 11.380 |

MO=REFUEL

| GROUP | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL PERSONS | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL M-REMS |
|---|---|---|---|---|---|---|---|---|
| MAINTENANCE PERSONNEL | 99 | 1 | 24 | 124 | 8.080 | 0.000 | 0.857 | 8.937 |
| OPERATING PERSONNEL | 23 | 0 | 0 | 23 | 0.616 | 0.000 | 0.000 | 0.616 |
| HEALTH PHYSICS PERSONNEL | 28 | 0 | 0 | 28 | 1.336 | 0.000 | 0.000 | 1.336 |
| SUPERVISORY PERSONNEL | 8 | 0 | 4 | 12 | 0.087 | 0.000 | 0.005 | 0.092 |
| ENGINEERING PERSONNEL | 20 | 4 | 15 | 39 | 0.637 | 0.051 | 0.861 | 1.549 |
| MO | 178 | 5 | 43 | 226 | 10.756 | 0.051 | 1.723 | 12.530 |
| | ======== | ======== | ======== | ======== | ======== | ======== | ======== | ======== |
| | 1867 | 51 | 618 | 2536 | 197.142 | 4.412 | 105.004 | 306.558 |

REXPR219
RUN DATE: 02-08-92
RUN TIME: 06:28:52

T E N N E S S E E   V A L L E Y   A U T H O R I T Y
BFN RADIATION EXPOSURE SYSTEM



NUMBER OF PERSONNEL AND MAN-REM BY WORK JOB FUNCTION
TOTAL NUMBER OF INDIVIDUALS

| | NUMBER OF PERSONNEL (> 100 M-REM) | | | | TOTAL MAN-REM | | | |
|---|---|---|---|---|---|---|---|---|
| GROUP | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL PERSONS | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT AND OTHERS | TOTAL M-REMS |
| MAINTENANCE PERSONNEL | 1054 | 26 | 350 | 1430 | 130.371 | 2.880 | 34.353 | 167.604 |
| OPERATING PERSONNEL | 210 | 0 | 4 | 214 | 14.095 | 0.000 | 0.402 | 14.497 |
| HEALTH PHYSICS PERSONNEL | 317 | 0 | 38 | 355 | 31.400 | 0.000 | 4.565 | 35.965 |
| SUPERVISORY PERSONNEL | 61 | 0 | 21 | 82 | 2.456 | 0.000 | 0.561 | 3.017 |
| ENGINEERING PERSONNEL | 225 | 25 | 205 | 455 | 18.820 | 1.532 | 65.123 | 85.475 |
| | ======== | ======== | ======== | ======== | ======== | ======== | ======== | ======== |
| | 1867 | 51 | 618 | 2536 | 197.142 | 4.412 | 105.004 | 306.558 |

TENNESSEE VALLEY AUTHORITY
BFN RADIATION EXPOSURE SYSTEM



NUMBER OF PERSONNEL AND MAN-REM BY WORK JOB FUNCTION
TOTAL NUMBER OF INDIVIDUALS

| GROUP | STATION | UTILITY | CONTRACT | TOTAL |
|---|---|---|---|---|
| MAINTENANCE PERSONNEL | 470 | 13 | 87 | 570 |
| OPERATING PERSONNEL | 71 | 0 | 2 | 73 |
| HEALTH PHYSICS PERSONNEL | 69 | 0 | 10 | 79 |
| SUPERVISORY PERSONNEL | 18 | 0 | 3 | 21 |
| ENGINEERING PERSONNEL | 77 | 3 | 136 | 216 |
| | ======== | ======== | ======== | ======== |
| | 705 | 16 | 238 | 959 |

## CHALLENGES TO OR FAILURES OF MAIN STEAM RELIEF VALVES

Unit 1

None

Unit 2

None

Unit 3

None

Units 1 and 3 were shut down during the entire reporting period. Unit 2 was operating for approximately seven consecutive months.

# REACTOR VESSEL FATIGUE USAGE EVALUATION

The cumulative usage factors for the reactor vessels are as follows:

| Location | Unit 1 | Unit 2 | Unit 3 |
|---|---|---|---|
| Shell at water line | 0.00620 | 0.00542 | 0.00431 |
| Feedwater nozzle | 0.29782 | 0.21638 | 0.16139 |
| Closure studs | 0.24204 | 0.20790 | 0.14360 |