TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT ANNUAL OPERATING REPORT

January 1, 1994- December 31, 1994

Docket Number 50-259, 50-260, and 50-296
License Number DPR-33, DPR-52, and DPR-68

This is a list of acronyms and abbreviations used throughout the 1994 Annual Operating Report.

ADS	Automatic Depressurization System; Atmospheric Dilution System
AFFF	Aqueous Film Forming Foam
ALARA	As Low As Reasonably Achievable
APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BFN	Browns Ferry Nuclear Plant
BPWS	Banked Position Withdrawal Sequence
BWR	Boiling Water Reactor
CAQR	Condition Adverse To Quality Report
CFR	Code of Federal Regulations
CISS	Containment Isolation Status System
CKV	Check Valve
COLR	Core Operating Limits Report
CRD	Control Rod Drive
CRDR	Control Room Design Review
CRLD	Change Request to a Licensing Document
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DC	Direct Current
DCN	Design Change Notice
DG	Diesel Generator
ECCS	Emergency Core Cooling System
ECN	Engineering Change Notice
EECW	Emergency Equipment Cooling Water
EFPD	Effective Full Power Days
ELLLA	Extended Load Line Limit Analysis
EMS	Equipment Management System
EOC	End of Cycle
F	Fahrenheit
FCV	Flow Control Valve
FDC	Floor Drain Collector
FDCN	Field Design Change Notice
FI	Flow Indicator
FIC	Flow Indicating Controller
FPC	Fuel Pool Cooling
FT	Flow Transmitter
ft	foot
GE	General Electric

GE SIL	GE Service Information Letter
GEMAC	General Electric Measurement and Control
gpm	Gallons per Minute
HELB	High-Energy Line Break
Hg	Mercury
HPCI	High Pressure Coolant Injection
HPFP	High Pressure Fire Protection
HS	Handswitch
HWC	Hydrogen Water Chemistry
ICS	Integrated Computer System
ILRT	Integrated Leak Rate Test
ISI	Inservice Inspection
I-Tabs	Instrument Tabulations
kV	Kilovolt
lbs	Pounds
LLRT	Local Leak Rate Test
LOCA	Loss of Coolant Accident
LPRM	Local Power Range Monitor
LS	Level Switch
LT	Level Transmitter
MCPR	Minimum Critical Power Ration
ME	Moisture Element
MG	Motor Generator
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MSLRM	Main Steam Line Radiation Monitor
MSRV	Main Steam Relief Valve
MWD/ST	Megawatt Days per Short Ton
MWe	Megawatt Electrical
MWt	Megawatt Thermal
NESSD	Nuclear Engineering Setpoint and Scaling Document
NFPA	National Fire Protection Association
NMS	Neutron Monitoring System
NRC	Nuclear Regulatory Commission
NUMAC	Nuclear Measurement Analysis and Control
NUMARC	Nuclear Utilities Management and Human Resources Committee
NUREG	Nuclear Regulatory Commission Regulation
PCIOMR	Preconditioning Interim Operating Management Recommendations
PCIS	Primary Containment Isolation System
PCV	Pressure Control Valve
PER	Problem Evaluation Report
PI	Pressure Indicator
ppb	Parts per Billion

ppm	Parts per Million
PS	Pressure Switch
psi	Pounds per Square Inch
PT	Pressure Transmitter
QA	Quality Assurance
RBCCW	Reactor Building Closed Cooling Water
RbNO3	Rubidium Nitrate
RCIC	Reactor Core Isolation Cooling
RCW	Raw Cooling Water
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RM	Radiation Modifier
RMOV	Reactor Motor Operated Valve
RMS	Radiation Monitoring System
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSW	Raw Service Water
RVLIS	Reactor Vessel Level Instrumentation System
RWCU	Reactor Water Cleanup
SBO	Station Blackout
SCFM	Standard Cubic Feet per Minute
SER	Sequential Events Recorder; Significant Events Report (INPO)
SGTS	Standby Gas Treatment System
SI	Surveillance Instruction
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SPDS	Safety Parameter Display System
SSP	Site Standard Practice
ST	Special Test
TACF	Temporary Alteration Control Form
TI	Technical Instruction
TIP	Traversing Incore Probe
TPM	Thermal Power Monitor
TRS	Temperature Recorder Switch
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
UPS	Uninterruptible Power Supply
V	Volt; Vanadium
VAC	Volts Alternating Current
VDC	Volts Direct Current
VHF	Very High Frequency

1994

OPERATIONAL SUMMARY

UNIT 1

Unit 1 remains on administrative hold to resolve various Tennessee Valley Authority (TVA) and Nuclear Regulatory Commission (NRC) concerns.

UNIT 2

On January 1, 1994, the unit's power level was a full power (3291 MWt and 1117 MWe).

On April 15, 1994, at 0218 the reactor scrammed on low scram air header pressure. The reactor remained shut down while maintenance was being performed on a compressed air system. The reactor was again critical on April 17 at 2233. After being placed in the run mode at 0226 on April 18, the reactor auto scrammed at 0355 on main steam isolation valve (MSIV) closure due to low reactor pressure. All bypass valves failed full open and then reclosed. The reactor was critical once again on April 20 at 2241 and at 100% power by 2015 on April 22.

Reactor coastdown began on July 24, 1994, with reactor shutdown starting October 1 for Cycle 7 refueling outage.

Following the refueling outage, control rod withdrawal began at 0507 on November 21, 1994, and the reactor declared critical at 0620 the same day. The reactor was manually scrammed and the turbine tripped at 0101 on November 28 to allow for the turbine to be balanced. The reactor was critical at 2242 on November 30 but scrammed again on December 2 at 0717 due to stator coolant system instrumentation failure. Criticality was achieved again by 2210 the same day.

On December 31, 1994, Unit 2 was at 3291 MWt and 1114 MWe.

UNIT 3

Unit 3 remains on administrative hold to resolve various TVA and NRC concerns.

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OPERATING STATUS

1. Unit Name: Browns Ferry Unit One
2. Reporting Period: Calendar Year 1994
3. Licensed Thermal Power (MWt): 3293
4. Nameplate Rating (Gross MWe): 1152
5. Design Electrical Rating (Net MWe): 1065
6. Maximum Dependable Capacity (Gross MWe): 0
7. Maximum Dependable Capacity (Net MWe): 0
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): 0
10. Reason for Restrictions, if any: Administrative Hold

		December 1994	Year to Date	Cumulative*
11.	Hours in Reporting Period	0	0	95743
12.	Hours Reactor Was Critical	0	0	59521
13.	Reactor Reserve Shutdown Hours	0	0	6997
14.	Hours Generator On Line	0	0	58267
15.	Unit Reserve Shutdown Hours	0	0	0
16.	Gross Thermal Generation (MWh)	0	0	168066787
17.	Gross Electrical Generation (MWh)	0	0	55398130
18.	Net Electrical Generation (MWh)	0	0	53796427
19.	Unit Service Factor	0	0	60.9
20.	Unit Availability Factor	0	0	60.9
21.	Unit Capacity Factor (MDC Net)	0	0	52.8
22.	Unit Capacity Factor (DER Net)	0	0	52.8
23.	Unit Forced Outage Rate	0	0	25.6

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each): N/A

25. If Shutdown at End of Reporting Period, Estimated Date of Startup: To Be Determined

*Excludes hours under administrative hold (June 1, 1985 thru end of reporting period)

Docket No.: 50-260

OPERATING STATUS

1. Unit Name: Browns Ferry Unit Two
2. Reporting Period: Calendar Year 1994
3. Licensed Thermal Power (MWt): 3293
4. Nameplate Rating (Gross MWe): 1152
5. Design Electrical 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

		December 1994	Year to Date	Cumulative*
11.	Hours in Reporting Period	744	8760	122071
12.	Hours Reactor Was Critical	729	7310	82127
13.	Reactor Reserve Shutdown Hours	0	0	14200
14.	Hours Generator On Line	711	7234	79859
15.	Unit Reserve Shutdown Hours	0	0	0
16.	Gross Thermal Generation (MWh)	2148854	22621314	231168915
17.	Gross Electrical Generation (MWh)	726260	7535260	76743178
18.	Net Electrical Generation (MWh)	708446	7345174	74589836
19.	Unit Service Factor	95.5	82.6	65.4
20.	Unit Availability Factor	95.5	82.6	65.4
21.	Unit Capacity Factor (MDC Net)	89.4	78.7	57.4
22.	Unit Capacity Factor (DER Net)	89.4	78.7	57.4
23.	Unit Forced Outage Rate	2.7	2.2	17.3

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each): None

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

*Excludes hours under administrative hold (June 1, 1985 to May 24, 1991)

Docket No.: 50-296

OPERATING STATUS

1. Unit Name: Browns Ferry Unit Three
2. Reporting Period: Calendar Year 1994
3. Licensed Thermal Power (MWt): 3293
4. Nameplate Rating (Gross MWe): 1152
5. Design Electrical Rating (Net MWe): 1065
6. Maximum Dependable Capacity (Gross MWe): 0
7. Maximum Dependable Capacity (Net MWe): 0
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): 0
10. Reason for Restrictions, if any: Administrative Hold

		December 1994	Year to Date	Cumulative*
11.	Hours in Reporting Period	0	0	73055
12.	Hours Reactor Was Critical	0	0	45306
13.	Reactor Reserve Shutdown Hours	0	0	5150
14.	Hours Generator On Line	0	0	44195
15.	Unit Reserve Shutdown Hours	0	0	0
16.	Gross Thermal Generation (MWh)	0	0	131868267
17.	Gross Electrical Generation (MWh)	0	0	43473760
18.	Net Electrical Generation (MWh)	0	0	42114009
19.	Unit Service Factor	0	0	60.5
20.	Unit Availability Factor	0	0	60.5
21.	Unit Capacity Factor (MDC Net)	0	0	54.2
22.	Unit Capacity Factor (DER Net)	0	0	54.2
23.	Unit Forced Outage Rate	0	0	21.6

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):
N/A

25. If Shutdown at End of Reporting Period, Estimated Date of Startup:
To Be Determined

*Excludes hours under administrative hold (June 1, 1985 thru end of reporting period)

1994

SUMMARY OF SAFETY EVALUATIONS FOR CORE COMPONENTS AND OPERATING LIMITS

Unit 2 Core Operating Limits Report (COLR)

Description/Safety Evaluation

This safety evaluation supports Revision 2 of the TVA Browns Ferry Nuclear Plant (BFN) Unit 2 COLR. The COLR contains the operating limits for the cycle determined from the reload licensing analyses as documented in the Supplemental Reload Licensing Report. Revision 2 of the COLR was made to incorporate revised end-of-cycle (EOC) minimum critical power ratio (MCPR) operating limits to allow a more bottom peaked EOC exposure distribution than assumed in the original licensing analyses. The revised MCPR limits were determined by General Electric (GE) from a reanalysis of the limiting transients. The reanalyses were performed using NRC approved methods and design bases.

Operation within the revised MCPR limits incorporated in Revision 2 of the COLR will ensure the MCPR safety limit specified in the Technical Specifications is not violated during any anticipated operational transient. Shutdown margins reported in the licensing analyses for Cycle 7 are adequate to accommodate any effects of the more bottom peaked exposure distribution ensuring Technical Specification limits on shutdown margin are met. By operating within the established limits, the more bottom peaked exposure distribution for Cycle 7 will not reduce the margin of safety as defined in the bases for any Technical Specification.

Since the core operating limits are contained in the COLR, revising the MCPR limits does not require a Technical Specification change. These changes will need to be incorporated into Appendix N of the Updated Final Safety Analysis Report (UFSAR).

No unreviewed safety question is involved.

Unit 2 COLR

Description/Safety Evaluation

This safety evaluation supports the BFN Unit 2 Cycle 8 reload core design and the cycle specific updates to the BFN Unit 2 COLR.

The reload core design and licensing analyses for this cycle were performed by GE with results documented in the Supplemental Reload Licensing Report. Operating limits for the cycle (i.e., Linear Heat Generation Rate, Minimum Critical Power Ratio, and Maximum

Average Planar Linear Heat Generation Rate) as determined by the licensing analyses are incorporated into the TVA BFN Unit 2 COLR.

The BFN 2 Cycle 8 core is a control cell core design with a predicted full power life of approximately 8900 megawatt days per short ton (MWD/ST) (equivalent to about 415 effective full power days (EFPDs)). Increased core flow, feedwater temperature reduction, and coastdown capability increase this to a maximum cycle burnup of approximately 9720 MWD/ST or 453 EFPDs assuming a 95% capacity factor is achieved (note: Cycle 8 is scheduled to operate from November 1994 to March 1996).

The fresh fuel types are GE7B and GE9B designs which are the same types as were loaded in Cycle 7. both are barrier cladding designs and have no Preconditioning Interim Operating Management Recommendations (PCIOMR) restrictions. The remaining twice-burnt fuel and once-burnt reinsert fuel from Cycle 6 does not contain barrier cladding and all PCIOMR constraints remain in effect for these bundles. The core will also include the 4 Westinghouse QUAD+ demonstration assemblies which were previously loaded in Cycles 6 and 7.

The cycle is analyzed for Extended Load Line Limit Analysis (ELLLA), Increased Core Flow, FFWTR, and Feedwater Heaters Out of Service. The cycle is also analyzed for Banked Position Withdrawal Sequence (BPWS) rod movement. The BPWS procedure must be followed in order to stay within the licensed Rod Drop Accident design basis.

Cycle 8 is designed for aggressive spectral shift operation. Spectral shift can extend full power operation by increasing the void content (spectrum hardening) during the first part of the cycle which increases plutonium production in the upper part of the core. Spectrum hardening is enhanced with operation at lower flow rates and by using rod patterns to obtain more bottom peaked power distributions.

No control blades or local power range monitor (LPRM) strings were replaced during the Fall 1994 outage.

The BFN Unit 2 Cycle 8 reload core design is acceptable from a nuclear safety standpoint. Due to necessary revisions to the UFSAR and the BFN Unit 2 COLR, a safety evaluation was required. No Technical Specification revisions are required.

No unreviewed safety question is involved.

Core Component Design Change Request No. 54

Description/Safety Evaluation

This safety evaluation addresses the use of modified original equipment control rods in BFN Units 1, 2, and 3. The control rods are being modified to replace the rollers with low cobalt spacer pads. The original equipment control rod roller and pin materials are cobalt bearing Stellite 3 and Haynes Alloy 25 respectively. The replacement spacer pad materials are low cobalt Inconel X-750 (spacer pads) and PH13-8Mo (retaining ring). The modification is being made for as low as reasonably achievable (ALARA) purposes to remove a large contributor of cobalt to the reactor coolant system thereby reducing dose rates to site personnel.

The control rod blade contains rollers at the top and bottom to guide the control rod as it is inserted and withdrawn from the core. Only the top rollers were replaced. The original rollers together with the portion of the retaining pins inside the rollers were removed and replaced with low cobalt spacer pads. The spacer pads consist of two halves threaded together and independently locked together by a snap ring. Each half of the assembly consists of a round washer conically tapered to a flat contact surface that interfaces with the fuel channels. The thickness of the assembled spacer pads is the same as the diameter of the rollers they replace.

It should be noted that the described modifications were performed on irradiated blades by remote underwater operations. The spacer pad design affords easier and faster replacement over the original pin and roller design under these conditions. GE procedure BN1-SWP-009 describes the modification process.

The modified control rod configuration was interchangeable with existing control rod assemblies and was compatible with existing nuclear steam system hardware.

The use of modified original equipment control rods having the upper rollers replaced with spacer pads is acceptable from a nuclear safety standpoint. Changes to UFSAR Section 3.4.5.1.1 and Figure 3.4-4 are necessary.

The control rod modification does not significantly affect control rod reactivity worth, scram insertion performance, or drop velocity. There is no impact on shutdown margins, scram times, or operating limits and no changes to the Technical Specifications are required.

No unreviewed safety question is involved.

1994

SUMMARY OF SAFETY EVALUATIONS FOR FIELD COMPLETED PLANT MODIFICATIONS

Safety Evaluations or Unreviewed Safety Question Determinations (USQDs) for the following plant modifications, which were field completed during 1994 were summarized in previous Annual Operating Reports. Therefore, they are not included in this report.

ECN/DCN No.	Description	See Annual Operating Report for Year
P0161	Remove Automatic Initiation Opening Logic from RCIC Steam Line Valves - Unit 3	1989
P0533	Install Torus Temperature Monitoring System - Unit 3	1988
P0596	Control Rod Drive (CRD) Flow Control Valve Replacement - Unit 3	1988
P0652	Replacement of Flow Control Valve (3-FCV-71-40 with Pneumatic Operated Soft Seated Check Valve - Unit 3	1988
P0706	Replacement of Power Supplies for Analog Trip System - Unit 3	1988
P0730	Modify Residual Heat Removal (RHR) Head Spray - Unit 3	1989
L2079	Replacment of Oxygen and Hydrogen Analyzer - Unit 3	1988
P3023	Replacement of Pressure Switches - Unit 3	1988
P3092	Replacement of Flow Indicating Switches 3-FS-74-50 and -64 - Unit 3	1988
P3104	Replacement of Flow Transmitter (FT) 3-FT-73-33 with Environmentally Quallified Transmitter - Unit 3	1989
H7054	Modification to Packing Configuration - Unit 2	1989
W13294	Replacement of Door Interlocks - Unit 0	1990
W14155	Installation of Carrier Heat Pump for the Instrument Maintenance Shop - Unit 0 Revision 1 (approved 07/21/94) of this safety evaluation was prepared to address the changes initiated by F28698. This Field Design Change Notice (FDCN) removes the requirements to replace the existing circuit breaker trip unit in compartment 2C of 480V service building main board with a GE Radiation Monitoring System (RMS)-9 unit. Calculation ED-N0215-910084 R14 was issued to show that the existing EC-1 trip device is adequate.	1992
W15365	Conduit, Cable, and Multiplier Equipment Setting for Unit 3 Process Computer - Unit 3 Revision 1 of this safety evaluation was prepared to delete the revision level for Safety Assessment SABFEDCN910032.	1992

ECN/DCN No.	Description	See Annual Operating Report for Year
W15724	Upgrade Evacuation Alarm System, Code Call, and Paging Systems - Unit 3	1992
W16710	Uninterruptible Power Supply (UPS) Building - Unit 0 Revision 1 of this safety evaluation was issued to address changes made by F21845. The changes to the safety evaluation included a format change, reference change, and minor editorial changes due to a revision to the Change Request to a Licensing Document (CRLD).	1992
W16713	Contractor Facilities - Unit 0	1992
W16726	Control Room Design Review (CRDR) Modifications for Panel 3-9-4 - Unit 3 - Revision 1 (approved 08/05/93) of this safety evaluation was issued to address the removal of position indication from Panel 3-9-4 for the recirculation loop equializer valves and to delete reference to a recently deleted CRLD. This was considered a minor change.	1992
W16960	CRDR Modifications for Panel 3-9-25 - Unit 3	1992
W17040	CRDR Modifications for Panel 3-9-7 - Unit 3	1992
W17041	CRDR Modifications for Panel 3-9-8 - Unit 3	1992
W17044	CRDR Modifications for Panel 3-9-3 - Unit 3 Revision 1 (approved 06/02/94) of this safety evaluation clarifies implementation restrictions and interim configuration associated with this DCN. This was considered a minor revision.	1992
W17057	CRDR Modifications for Panel 3-9-53 - Unit 3	1992
W17082	CRDR Modifications for Panel 3-9-47 - Unit 3	1992
W17133	CRDR Modifications for Panel 3-9-5 - Unit 3	1992
W17215	CRDR Modifications for Panel 3-9-54 and 3-9-55 - Unit 3	1992
W17251	Installation of Main Control Room Workstations - Unit 0 Revision 1 of this safety evaluation incorporated the changes associated with FDCN F20395 which deleted the very high frequency (VHF) radio console from the Unit 0 workstation but retained the capability to reinstall the console at a future time. Additionally, the FDCN installed conduit, added utility power and installed signal cables from the Unit 1 operators desk to the common area workstations.	1992
W17252	Installation of Main Control Room Workstations - Unit 2	1992

ECN/DCN No.	Description	See Annual Operating Report for Year
W17257	Control Bay Elevation 593' Air Conditioning - Unit 0	1992
W17310	Replacement of Obsolete GE Measurement and Control (GEMAC) Transmitters with Rosemount Transmitters - Unit 2	1991
W17347	Motor Operated Valve (MOV) Thrust Requirements - Unit 2 Revision 1 of this safety evaluation (approved 10/13/94) was prepared to address the changes initiated by F31688. F31688 deleted valve FCV 2-FCV-78-68 from W17347 and from the GL 89-10 program. This valve is normally closed, stays closed for all design basis earthquakes (DBEs) and is not required for any DBE. Therefore, this valve does not have an active safety function as defined in GL 89-10 and the GL 89-10 scoping calculation has deleted this valve from the scope of GL 89-10.	1992
W17427	CRDR Modifications for Panel 3-25-32 - Unit 3	1992
W17447	Replacement of Fuel Pool Cooling (FPC) Pump 1/4" Seal Water Line with Stainless Steel Tubing and Add Throttle Valve - Units 1, 2, 3	1992
W17514	Replacement of Drywell Control Air System Dewpoint Temperature Monitoring Loops - Unit 2	1992
W17536	Modifications to Drywell Platform - Unit 3	1992
W17545	Recirculation Ringheader, Risers, Safe Ends, and Jet Pump Instrumentation Nozzle Safe Ends Replacement - Unit 3	1992
W17725	Addition of Station Battery No. 5 - Units 1, 2, 3	1992
W17904	Fire Alarm and Detection System Upgrade - Unit 0	1992
W18207	Installation of Hydrogen Water Chemistry (HWC) System - Unit 2	1992
W18209	Installation of HWC System - Unit 2	1992
W18554	Modification to Transformer TS3E - Unit 3	1992
W18685	Reroute/Replace Cables - Unit 1, 2, 3	1992
W18812	Modifications to Condensate Transfer System Piping - Unit 2	1992

ECN P0112 - Average Power Range Monitor (APRM) Simulated Thermal Trip Modification - Unit 3

Description/Safety Evaluation

This ECN added a thermal power monitor (TPM) to the APRM subsystem of the neutron monitoring system (NMS). The addition of the TPM was in response to the operating history of Boiling Water Reactors (BWRs) which shows numerous scrams resulting from momentary neutron flux spikes. These spikes are typically caused by power increase anomalies such as disturbances in the recirculation system, disturbances during large flow control load maneuvers, transients during turbine stop valve tests, etc. During a power increase transient, the neutron flux leads the reactor thermal power because of fuel time constants. This situation can result in neutron flux trip levels before the reactor thermal power has increased significantly, thereby causing an unnecessary reactor scram. The flux spikes typically represent no danger to the fuel since they are only one or two seconds in duration and are less than the 120% flux trip limit.

The TPM should eliminate this problem by providing a measurement which is more representative of the reactor thermal power during a transient than the previous design. The TPM utilizes the APRM output signal as its input and provides an output signal which closely approximates the average heat flux (thermal power) during a transient or steady-state condition. This is accomplished by the use of a time constant which is representative of the fuel dynamics. This time constant is sufficiently long so that flux spikes such as those described above are averaged over a longer time period and hence do not result in the generation of a trip signal from the TPM.

The addition of the TPM resulted in the replacement of the flow-referenced APRM trip function with a flow-referenced TPM trip, but preserved the existing trip channel on straight APRM flow utilizing a nonflow-referenced set point. The APRM signal which previously was an input to a flow-referenced trip unit would now provide the input to the fixed 120% trip unit and also to the TPM time constant module. Total recirculation drive flow is used to provide the flow referencing to the TPM. The TPM then provides a signal to the flow-referenced APRM thermal power trip unit. A channel trip would result from either a fixed APRM trip unit or the flow-referenced TPM trip unit.

The Technical Specifications already incorporate the necessary changes to reflect this ECN, however, the UFSAR requires revision to reflect the design. No unreviewed safety question is involved.

ECN P0244 - Replacement of Differential Pressure Transmitters - Unit 2

Description/Safety Evaluation

The ECN replaced differential pressure transmitters PDT-68-65 and PDT-68-62 of the reactor water recirculation system. It also removed load resistors that are not needed when Rosemount transmitters are used and changed the installation requirement from seismic Class I to seismic Class II.

The original Foxboro Model 611DM transmitters were replaced with Rosemount Model 1151 HP 7B22PB transmitters. The affected components are not safety related and no function of the reactor water recirculation system is changed. No other system or equipment is effected.

No Technical Specification changes are required. The UFSAR is not affected.

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

ECN P0511 - Replacement of Reactor Building Emergency Lighting Transformer - Unit 2

Description/Safety Evaluation

This ECN replaced the reactor building emergency lighting transformer with a newer model due to discontinuance of the existing older model. This ECN also relocated the transformer from reactor building Elevation 608.3' to Elevation 593'.

The new transformer will perform the same function as the present transformer. The transformer was seismically qualified to prevent damage to safety-related equipment.

No Technical Specification changes are required.

No unreviewed safety question is involved.

ECN P0583 Revision 1 - Individual Test Points for 3-FCV-71-40 - Unit 3

Description/Safety Evaluation

The testable check valve (CKV) on the injection line to the RPV had two test connection points, one on each side of the valve. Originally, the test lines were connected to one common test point. This ECN removed the inter-tie piping to valves 74-536 and 71-534 and provides separate lines with individual test points. It also changed the valve number prefixes for 74-536 and 74-535 to 71-536 and 71-535, respectively. Those modifications will correct the mislabeled valve tags and improve the testing of feedwater valve 3-568 where the reactor core isolation cooling (RCIC) connects to the feedwater system. It will also allow a temporary pressure hose to be installed across FCV 71-40 to remove any differential pressure when cycling the valve for a test.

A revision to the Technical Specification is not required. The margin of safety as defined in the Technical Specification is not reduced. UFSAR Figure 4.7-1A requires a revision to indicate that the change is applicable to Unit 3 only. No unreviewed safety question is involved.

DCN P0597 - Replacement and Modification of Reactor Feedwater Pump Flow Instrumentation - Unit 3

Description/Safety Evaluation

This ECN provided the documentation to replace and modify the reactor feedwater pump flow instrumentation (FT-3-6, -13, -20, and FI-3-6, -13, -20) located in the reactor feedwater system. The existing nonsafety-related flow transmitters, Bailey Meter Company Type 555 weighing 23 lbs. were replaced with lighter (12 lbs.) Rosemount Model 1151DP transmitters and their indicating range was increased to enable a greater flow rate indication.

This change allows the operator to monitor the actual feedwater flow of any two feedwater pumps, while the other one pump is out of service. With this change, the feedwater pumps can provide a flow of water to the reactor, equivalent to approximately 90% of normal flow during full power operation and stay within the operating range of the flow instrumentation.

The setpoints for opening the minimum recirculation valve and the low reactor vessel level signal to ramp the pumps to a speed corresponding to 75% of full power are not affected.

The feedwater system is not safety related and is not required for safe shutdown of the reactor. The instruments do not impact the seismic qualifications of the panels they are located on, because the new transmitters are placed in the old location, occupy approximately the same space, and are approximately half the weight of the replaced old transmitters. Also, they do not degrade any Class 1E system.

No Technical Specification change is required. However, Section 11.8.3.1 of the UFSAR is affected.

No unreviewed safety question is involved.

ECN P0737 - Replace Air Lock Electrical Penetration- Unit 3

<u>Description/Safety Evaluation</u>

This ECN replaced existing penetration with one meeting environmental qualification. The penetration contains cables for airlock light, telephone, and door status circuits. A failure of this penetration would cause leakage into secondary containment if the inner air lock door was open during a loss of coolant accident (LOCA) or High-Energy Line Break (HELB) inside primary containment.

The new electrical penetration is qualified to American Society of Mechanical Engineers (ASME) Code Section III primary containment penetrations requirements. Also, the new penetration is welded into the existing bulkhead sheaves and are seismically qualified.

No Technical Specification change is required.

No unreviewed safety question is involved.

ECN P0852 - Modification to Offgas Gas Reheater Effluent Moisture Loop - Unit 3

<u>Description/Safety Evaluation</u>

ECN P0852 deals with a modification to the offgas system and involves instruments which measure the moisture content of the effluent gases from the gas reheater before it enters the prefilters and the charcoal adsorbers. The original scope of ECN P0852 was to replace the

moisture sensor ME-66-110, transmitter MT-66-110, and recorder TRS-66-108 in Units 1, 2, and 3 due to equipment obsolescence and unavailability of spare parts. However, the transmitter was later taken out of the ECN scope and the ECN only provided the design for the replacement of the recorder and the moisture element. The ECN has been implemented in Unit 2.

The replacement of the recorder 3-TRS-66-108 is addressed by CRDR DCN W17057. DCN F18188 was generated to remove the recorder from the scope of ECN P0852 for Unit 3. Hence, for Unit 3, the scope of ECN P0852 involves changes associated with the replacement of the existing moisture element, 3-ME-066-0110, with a moisture element with an expanded range.

DCN F21442 was generated to address the document deficiencies and the downscoping of ECN P0852 for Unit 3.

This change involves a modification to the offgas system which is considered part of the radwaste system, i.e., gaseous radwaste. The modification does not impact offgas system functional or operational characteristics, therefore nuclear safety is not decreased. The modification is in a section of the Unit 3 offgas system that is fully isolable from the other units and not required for the function of either of the other units.

No Technical Specification change is required and the UFSAR is not affected.

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

ECN L1937 - Installation of Backwash Connections on the Core Spray and RHR Room Coolers - Unit 3

Description/Safety Evaluation

This ECN provided 1" flush connections along with associated isolation valves for the RHR room coolers, core spray room coolers, and the core spray motor bearing coolers. The flushing lines were installed between the cooler and their respective isolation valves. These modifications will facilitate, from a maintenance perspective, the connection of the backwash lines and the isolation and flushing of the coolers. The new connections will provide a means of accomplishing the cooler cleaning program without cutting and welding pipe each time a cooler is cleaned.

The new flushing connections along with associated isolation valves are safety related and designed to TVA Class P requirements. The manual flushing connection isolation valve on the inlet and outlet line of each cooler are normally closed, thus maintaining the integrity of the emergency equipment cooling water (EECW). Since the new connections are used only for maintenance, and on an annual basis, the required flow of EECW cooling water to each respective cooler to achieve safe shutdown of the plant is not compromised.

No Technical Specification revision is required.

No unreviewed safety question is involved.

ECN L2050 - Modifications to Automatic and Manual Controls for Steam Jet Air Ejectors (SJAEs) - Unit 3

Description/Safety Evaluation

The automatic controls on the steam supply to the SJAEs are subject to instability during startup transfers from auxiliary boiler steam to nuclear process steam and automatic SJAE switchovers. As a result, gross excess dilution steam is often being provided by the SJAEs to the offgas system during these transients. Therefore, this ECN was written as a short-term fix to allow manual operator action to minimize the control instability problem. (The long term fix to be covered by another ECN will involve alleviating the problems with the existing automatic control system.)

The modifications covered by this ECN constitute an interim fix to the SJAE steam supply pressure controller instability and alleviate the offgas system operational problems caused by the controller instability. No safety-related system or function is affected by these modifications.

No Technical Specification changes are required.

No unreviewed safety question is involved.

ECN P3025 - Replacement of High Pressure Coolant Injection (HPCI) Pressure Switches - Unit 3

Description/Safety Evaluation

This ECN replaced HPCI pressure switches (PS) PS-73-22A and PS-72-22B with new pressure switches to meet environmental qualifications. The new equipment meets the same requirements and performs the same function as the original equipment.

No Technical Specification change is required.

No unreviewed safety question is involved.

DCN H4277 - Removal of Thermal Overloads on RHR and Residual Heat Removal Service Water (RHRSW) Valves - Unit 2

Description/Safety Evaluation

DCN H4277 implements a removal of thermal overload protection for selected MOVs by bypassing the thermal overload relays and removing the thermal overload heater elements from the starters of the following valves:

- 2-FCV-23-34
- 2-FCV-23-40
- 2-FCV-23-46
- 2-FCV-23-52
- 2-FCV-74-59
- 2-FCV-74-73

This safety evaluation was revised to address DCN F33607. This FDCN removes the special requirements from the safety evaluation and documents the test requirements on the single line drawings for 480V reactor motor operated valve (RMOV) Boards 2A and 2B. Additionally, the frequency of the testing was changed to coincide with BFN's valve diagnostic testing commitments for NRC Generic Letter 89-10.

Thermal overload protection is intended to provide motor protection from the harmful consequences of operational overloads and motor stall conditions. This protection is not intended to provide, nor will it provide, valve system boundary integrity or flow integrity protection. The valves involved in this DCN had experienced tripping of their overload

protective devices as a result of the valves duty requirement for providing throttling of their respective system flows.

The RHRSW valves are the throttle valves on the discharge side of the Unit 2 RHR heat exchangers. These valves are positioned by the operator as needed to establish and control cooling water for the RHR heat exchangers as needed for the applicable RHR operating mode.

The RHR valves are the test return valves and are normally closed. These valves require closure if they are not in their normal position at the onset of a DBE LOCA. These valves are also required to open in support of the RHR system torus cooling mode of operation, and they must maintain their position integrity for long term decay heat removal.

The Technical Specification does not directly discuss thermal overload protection for MOVs. However, assuring the integrity of required MOV actuations is a necessary element of assuring the basis for Technical Specifications. Therefore, assuring the capability of plant operators to position valves when necessary to obtain required system configurations and flows will enhance the existing margins of safety. Thus, this modification did not reduce the margins of safety as defined in the basis for any Technical Specification.

No Technical Specification revisions are required as a result of this modification. UFSAR figures have been revised to depict the removal of the thermal overload protection as implemented by this DCN and the text and figures require revision to reflect the testing requirement note being added by DCN F33607.

No unreviewed safety question is involved.

DCN H5586- Rewiring of Output Switch Contacts - Unit 2

Description/Safety Evaluation

DCN H5586 provided wiring changes in auxiliary instrument room Panel 2-9-19 to allow for surveillance testing of the drywell differential pressure alarm and drywell compressor control loops without violating the intent of Technical Specification Sections 3.2.F and 3.7.A.6.

The existing circuitry was rewired to allow the control and alarm function to remain intact for one channel while the other channel is being calibrated utilizing a lifted lead to isolate the circuit function under test.

The function of the differential pressure alarm and control circuitry modified by this DCN is not degraded in any manner.

No Technical Specification change is required.

No unreviewed safety question is involved.

DCN H5614 - Replacement and/or Rerouting of Normal Lighting Circuits - Unit 1

Description/Safety Evaluation

This DCN replaced and rerouted two normal lighting circuits to provide corrective action for CAQR BFP890261P. This DCN did not involve modifications to the circuits' associated end devices. The cables required modification as a result of cable damage due to settling of the Unit 1 air intake structure.

The cables replaced by this modification have no safety related function and their replacement has no affect on the function, operation, nor qualification of any component, system, or structure required to ensure nuclear safety.

No Technical Specification changes are required.

No unreviewed safety question is involved.

DCN W6846 - Reroute Cable to Appropriate Fire Zones - Unit 3

Description/Safety Evaluation

This safety evaluation was revised to incorporate DCN F20203A which provided design to reroute cable 3PP733-I3B out of fire zones 3-3 and 3-4. This modification only involved cable and raceway in the Unit 3 reactor building and diesel generator (DG) building. This DCN did not involve modifications to the sources nor the end devices associated with these cables.

Cable 3PP733-I was replaced with environmentally qualified cable end to end and routed in fire area 22, fire zone 3-2, and fire zone 3-1 in a new 4" conduit. The reroute of the cable in

fire areas was required to assure Appendix R separation. Old conduit was removed and existing penetrations spared. Existing cable 3PP733-I was retagged as 3ABN15, capped, and abandoned in tray CE-I.

The replacement of cable 3PP733-I required the opening of breaker number 9 in 4kV shutdown board 3EB. This disabled alternate feed to 480V shutdown boards 3A and 3B. However, the normal feed to shutdown boards 3A and 3B were not affected by the implementation of this modification and were available to comply with Technical Specification 3.9.C/4.9.C.

During the conduit installation, the breaching of fire barriers (walls, floors, and seal assemblies) complied with Technical Specification 3.11.G/4.11.G. The new seals meet the requirements for secondary containment, flood protection, and fire resistance, as applicable.

There are no impacts on nor potential changes to the Technical Specifications resulting from implementation of this modification. No unreviewed safety question is involved.

ECN P7019 - Upgrade of Reactor Water Cleanup (RWCU) Sampling Station - Unit 2

Description/Safety Evaluation

This modification was made to upgrade the inline coolant chemistry instrumentation in order to better control coolant chemistry. This modification replaced or rerouted sample lines, replaced RWCU sampling panel and chiller, installed an online ion chromatograph and associated computer with a UPS in the RWCU sampling subsystem of the sampling and water quality system.

No Technical Specification change is required. UFSAR figures will require revision.

No unreviewed safety question is involved.

DCN W15367 - Integrated Computer System (ICS) Upgrade - Unit 3

Description/Safety Evaluation

This DCN provided equipment mounting details, cable routing, and cable terminations to complete the Unit 3 ICS upgrade modification. Also included in this DCN was the removal of the Unit 3 GE4020 Plant Process Computer. This modification was required to support TVA's commitment to the NRC to implement Nuclear Regulatory Commission Regulation (NUREG) 0696 requirements. The ICS upgrade modification will ultimately provide a separate computer system for each unit. This DCN addressed the Unit 3 upgrade only.

The process computer system provides a quick and accurate determination of core thermal performance, improves data reduction, accounting, and logging functions for both the nuclear boiler and balance of plant equipment, and supplements procedural requirements for control rod manipulation during reactor startup and shutdown. The new Unit 3 system performs all current nuclear steam supply system and balance of plant functions provided by the GE4020 computer as well as the following additional functions:

- Safety parameter display system (SPDS)
- Sequence of events
- Rod scram time recording
- Transient recording analysis
- Rod worth minimizer

No revision to plant Technical Specifications is required as a result of the implementation of this DCN. UFSAR Sections 7.16, 7.16.5.3, and Appendix 7.7B require revision to reflect the ICS installation. In addition, numerous UFSAR figures require revision to reflect changes made to their TVA source drawings.

This modification does not result in a reduced margin of safety as defined in the basis for any Technical Specification. No unreviewed safety question is involved.

DCN W16435 - Reactor Vessel Level Instrumentation System (RVLIS) Instrumentation - Unit 2

Description/Safety Evaluation

DCN W16435B was originally issued providing for a periodic backfill of the RVLIS instrument reference leg condensing pots at local Panels 25-5 and 25-6. The reference legs of the RVLIS pressure/level transmitters were to be arranged such that similar reactor pressure vessel (RPS)/emergency core cooling system (ECCS) channels shared the same condensing pot, thus 'channelizing' the condensing pots.

NRC Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Instrumentation In BWRs", required a modification to ensure the level instrumentation was of a high reliability for long-term operation, to be in place following the first cold shutdown after July 30, 1993. To meet this commitment, the scope of DCN W16435B was reduced to address installing a continuous backfill to each RVLIS instrument reference line.

DCN W16435C was issued as Phase I to add continuous backfill to each condensing pot by injecting backfill from the CRD system into the reference leg headers at Panels 25-5 and 25-6. Also, only the reference leg for level transmitter (LT) 2-LT-3-53 was moved from condensing pot 3-820 to 3-821, via internal tubing reroute inside Panel 25-5, to ensure that feedwater level control is maintained and high water level trip of feedwater and main turbine is not disabled. A transient or perturbation of the reference leg could potentially cause a scram or ECCS initiation.

DCN F29466 was written against DCN W16435C to reinstate the arrangement of the RVLIS instrument reference legs as originally designed per DCN W16435B. The scope of this work was considered as Phase II implementation. The 'channelization' of the instrumentation to the condensing pots will reduce a perturbation or transient in a reference leg to actuation of half the RPS/ECCS logic. 2-LT-3-53 was returned to condensing port 3-820 and transmitter 2-LT-3-207 and 2-PT-3-207 reference legs were moved to 3-821.

Calculation and the Electrical Calculation Checklist have concluded that this activity does not affect the accuracy of the RVLIS instrumentation. No permanent change to the Technical Specification is required. A temporary change to the Technical Specification (Temporary Technical Specification No. 343T) was required to allow implementing the modification with reactor head on and fuel in the vessel.

UFSAR figures will require updating as a result of this modification.

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

DCN W16809 - CRDR Modifications to Panel 3-9-20 - Unit 3

Description/Safety Evaluation

This DCN consisted of modifications to Panel 3-9-20 for resolving identified human engineering discrepancies between the design of the Unit 3 control room and TVA's human factors standards. These modifications are applications of human engineering principles to improve man-machine interface characteristics and, thus, enhance operator response during abnormal and emergency conditions of the plant.

DCN F25936 deletes the addition of redundant EECW sectionalizing valve position indications originally intended to be installed in Panel 3-9-20.

In general, this DCN performed the following:

- Rearranged control switches and instruments;
- Replaced switch escutcheons and switch handles with black handles;
- Replaced meter and/or meter scales with color banding, as applicable;
- Provided hierarchical and component labeling; and
- Implemented panel repair maintaining seismic integrity of Panel 3-9-20.

In addition to the general modifications listed above, this DCN performed the following:

- Duplicated standby gas treatment system (SGTS) outlet flow indication, 0-FI-65-50B/3 and 0-FI-65-71B/3, from Panel 3-9-25 to Panel 3-9-20;
- Provided SGTS train operability indication, 0-XI-65-18B/3, 0-XI-65-40B/3, and 0-XI-65-69B/3 at Panel 3-90-20;
- Duplicated control, indication, and annunciation associated with the service air crosstie valve (Handswitch (HS) 0-HS-33-1A/1 and 0-PA-33-1A/1) from Panel 1-9-20 to Panel 3-9-20;
- Modified control air header pressure instrument loop 3-P-32-88 to provide a wider range;
- Added an instrument loop (0-P-33-3) for indication of service air header pressure, pressure indicator (PI) 0-PI-33-3A/3, at Panel 3-9-20;
- Replaced existing indication of DG cooler high discharge temperature with annunciation at 3-XA-55-20, Panel 3-9-20;

- Added an instrument loop (0-P-26-44) for indication of high pressure fire protection header pressure, 0-PI-26-44A/3, at Panel 3-9-20;
- Removed the CO_2 system capability for cable spreading rooms A and B.

Revision 1 of this safety evaluation was issued to incorporate changes made by DCN F27244 to the CO_2 storage, fire protection, and purging system. DCN F27244 removes the CO_2 system capability for cable spreading rooms A and B. The National Fire Protection Association (NFPA) sprinkler modifications installed by DCNs W17821 and W17822 provide adequate fire protection for cable spreading rooms A and B, and the existing CO_2 system is no longer required.

These modifications do not change any Technical Specification requirements. UFSAR figures and Fire Protection Report figures and text will require updating.

These changes do not reduce nuclear safety and no unreviewed safety question is involved.

DCN W16810 - CRDR Modifications to Panel 3-9-6 - Unit 3

Description/Safety Evaluation

This DCN consisted of modifications to Panel 3-9-6 for resolving identified human engineering discrepancies between the design of the Unit 3 control room and TVA's human factors standards. These modifications are applications of human engineering principles to improve man-machine interface characteristics and, thus, enhance operator response during abnormal and emergency conditions of the plant.

DCN F25936 deletes the addition of redundant EECW sectionalizing valve position indications originally intended to be installed in Panel 3-9-20.

In general, this DCN performed the following:

- Rearranged and/or replaced control switches, indicating lights, and meters;
- Provided new labels for all components with improved functional descriptions;
- Provided hierarchical labels for identification of systems and their associated components;
- Replaced switch handles with tactile and shape coded black handles, as applicable;
- Replaced switch position escutcheons;
- Replaced indicating light lenses to conform with BFN standards;
- Provided color banding for specific scales;
- Replaced meter scales to conform with BFN human factors design standards;

- Replaced existing analog recorders with functionally identical recorders; and
- Implemented modifications while maintaining the seismic integrity of the panel.

These modifications do not change any Technical Specification requirements. UFSAR figures will require updating.

These changes do not reduce nuclear safety and no unreviewed safety question is involved.

DCN W17185 - Installation of Containment Isolation Status System (CISS) - Unit 3

Description/Safety Evaluation

This DCN installed and interconnected the CISS for Unit 3. The CISS uses programmable logic controllers to monitor the position of primary containment isolation system (PCIS) valves and the status of PCIS isolation initiations. This information is processed to provide a summary of PCIS isolation completions on Control Room Panel 3-9-4.

In addition, this modification relocated two PCIS logic reset handswitches (16A-S32 and 16A-S33) and four PCIS Group 1 isolation logic status indicating lights (16A-DS250, 16A-DS251, 16A-DS252, and 16A-DS253) from Panel 3-9-5 to Panel 3-9-4. To make room for the CISS status indications, the drywell floor drain sump flow totalizer (3-FQ-77-6) was relocated on Panel 3-9-4.

This DCN also modified the PCIS Group 8 (traversing incore probe (TIP)) isolation circuitry and installed a relay and Group 8 reset pushbutton on Panel 3-9-13. This modifies the circuitry to remove the existing auto-reset circuitry on the TIP valves and requires an operator to manually reset a Group 8 isolation before the TIP ball valves can be opened.

No Technical Specification changes were required. UFSAR figures will require updating as a result of this modification.

By maintaining the current design, function, and performance of PCIS control and indication components, it is assured that the installation of CISS will not reduce the margin of safety as defined in the basis for any Technical Specification. No unreviewed safety question is involved.

DCN W17531 - ADS Modification - Unit 3

Description/Safety Evaluation

DCN W17531 made the following changes to the design of the ADS:

- Replaced the existing ADS timer (120 seconds) with a seismically qualified time delay relay of similar manufacture;
- Revised the setpoint of the 120 second timer from 120 seconds $\pm$ 5 seconds; to 95 $\pm$ 7 seconds in order to comply with BFN UFSAR Appendix N Section N.6.5.10 and BFN Unit 3 Technical Specification Section 3.2.B;
- Added an ADS inhibit switch into both trains of the ADS initiation logic;
- Added a time delay relay to each train of ADS logic to bypass the high drywell pressure initiation signal after a low-low-low (Level I) RPV water level signal occurs and the timer times out;
- Established the setpoint of the time delay relay at 265 seconds in order to maintain an analytical limit of six minutes.

The modification changed setpoints and electrical design features of the ADS function. The setpoints presently stipulated in BFN Unit 3 Technical Specification Table 3.2.B require revision as a result of this DCN. This change is in a conservative direction and supported by calculation and setpoint and scaling documents. This Technical Specification revision was a requirement for return to operation of DCN W17531 and its associated FDCNs.

The hardware changes did not remove any safety functions described or inferred in the bases for Technical Specification Sections 3.2.B and 3.6 without proper administrative control. Additionally, the modifications ensure that a valid design basis event (i.e., main steam line break outside primary containment with loss of high pressure makeup) applicable to BFN Unit 3 can be successfully mitigated without operator intervention. Therefore, the margin of safety associated with Technical Specification Sections 3.2.B and 3.6 is in no way reduced.

This modification is safe from a nuclear safety standpoint and no unreviewed safety question is involved.

DCN W17803 - Change Valve Operating Medium from Water to Air - Unit 3

Description/Safety Evaluation

This modification changed the valve actuation medium for 3-FCV-67-50 and 3-FCV-67-51 from hydraulics supplied from the EECW header to pneumatics from the control air system. Additionally, the operating logic for the valves was slightly modified such that the valves will no longer cycle on EECW header pressure with a demand from raw cooling water (RCW), but upon shutting from low EECW header pressure will remain shut and require a manual reset to be reopened. This change was necessary to reconcile Condition Adverse to Quality Report (CAQR) BFP900232 which was written to document the failure of these valves to perform their intended safety function of closing in a timely manner upon low EECW header pressure to ensure adequate flow to essential EECW loads. These valves were operated with raw water from the EECW header as the actuating medium. The failure of these valves to operate correctly was attributed to silt blockage in the actuator lines due to the use of raw water for hydraulic actuation. These control valves are normally closed, backup supply valves to the portion of the RCW system which supplies the Unit 3 reactor building closed cooling water (RBCCW) heat exchanges.

No change to Technical Specifications is required as a result of the implementation of this modification. This modification does require revisions to the UFSAR to properly depict plant configuration.

This modification is safety from a nuclear safety standpoint and no unreviewed safety question is involved.

DCN W18052 - NFPA Upgrades To Unit 3 Reactor Building Elevation 639' - Unit 3

Description/Safety Evaluation

W18052 modified the Aqueous Film Forming Foam (AFFF) system which supplies the fire suppression for the lube oil system of the recirculation motor generator (MG) sets located on the 639' elevation of the Unit 3 reactor building. All existing pipe, pipe supports, valves, and sprinkler heads from the branch off from the 4" raw service water (RSW) supply header between valves 3-26-1279 and 3-26-1280 was removed and replaced by this modification. The size of the main header of this AFFF system was increased to 4". An air supervision system was installed with the modification. This design change only installed the air supply hardware and pressure switch. This DCN did not provide the signals and associated alarms for the air supervision system. The power, signals, and associated alarms for the air

supervision pressure switch are provided in DCNs W17905 and W17908. Also, the smoke and heat detection system which is associated with the AFFF preaction system is installed by DCNs W17905 and W17908. In addition to the fire protection hardware modifications, the curb design on the 639' elevation was modified to improve the containment of lube oil in case of a spill. The containment of the lube oil to the curbed areas limits the area required to be covered by the fire suppression system.

This design change was provided to upgrade the fire suppression system to meet the design requirements of NFPA codes 16A and 13 and design criteria.

No Technical Specification change is required.

This change does not have any impact on the Unit 2 Appendix R analysis nor does it affect the capability, performance, or function of any component important to safety. This modification is safe from a nuclear safety standpoint. No unreviewed safety question is involved.

DCN W19260 - Anticipated Transient Without Scram (ATWS) Standby Liquid Control (SLC) Modification - Unit 3

Description/Safety Evaluation

DCN W19260 modified the Unit 3 SLC system to comply with the Code of Federal Regulations (CFR) 10CFR50.62 ATWS equivalency requirements. The modification ensures that the SLC system has the capability to inject a borated water solution into the reactor vessel at a flow rate, level of boron concentration, and Boron-10 isotope enrichment that will control reactivity to at least the equivalent of that resulting from the injection of 86 gallons per minute (gpm) of 13 weight percent sodium pentaborate solution with a natural boron concentration within the reactor core of at least 660 parts per million (ppm).

The modification enriches the SLC boric acid to 92 atom percent Boron-10 and decreases the sodium pentaborate solution concentration to ≤ 9.2 weight percent. The modification also replaced and rescaled temperature switches 3-TS-63-3 and -4, rescaled temperature control loop 3-T-63-2, and rescaled level loops 3-L-63-1 and -1B.

Enriching the SLC boric acid and decreasing the sodium pentaborate solution concentration improves the ability of the SLC system to bring the reactor from full power to a cold shutdown and brings the system into compliance with the 10CFR50.62 ATWS equivalency requirements as specified in the Technical Specifications.

The temperature and level alarm setpoint changes, as well as the heater control setpoint changes, are commensurate with the chemical composition changes and continue to alert the operator of solution temperature and volume changes that might indicate a possible solution concentration change. Nuclear Engineering Setpoint and Scaling Documents (NESSDs) are issued to ensure the instrument setpoints, scaling, and accuracy requirements are implemented and controlled.

Increasing the minimum amount of required Boron-10 in the Unit 3 SLC tank adds to the margin of safety as defined in the basis of the Technical Specifications.

This modification involved a change to the Unit 3 Technical Specifications. A design calculation for Unit 3 has determined that 186 lbs. is the minimum amount of Boron-10 needed for injection into the reactor coolant to achieve cold shutdown. To maintain consistency between all units, the minimum amount of Boron-10 needed for each unit was calculated and the most limiting value calculated was used for all units. Thus, a change will be performed for the Units 1, 2, and 3 Technical Specifications. Also, revision to the text of the UFSAR is required.

The modifications to the Unit 3 SLC system do not reduce nuclear safety and involves a change to the plant as described in the UFSAR.

No unreviewed safety question is involved.

DCN W20206 - Removal of PCIS and RPS Trip Signals from Main Steam Line Radiation Monitors (MSLRMs) - Unit 2

Description/Safety Evaluation

DCN W20206 deleted the following safety-related reactor scram and PCIS functions associated with the Unit 2 MSLRMs:

- Reactor scram
- Closure of the MSIVs
- Closure of the main steam line drain valves
- Closure of the reactor water recirculation loop sample valves

Nonsafety-related automatic trips to deenergize the main condenser mechanical vacuum pumps and to initiate closure of the vacuum pump suction line isolation valves will remain active.

Calculations performed in support of this DCN demonstrate that with these changes, offsite radiation doses will remain well within (<25%) the limits of 10CFR100 and the ability to safely shutdown the plant is not degraded. The operation and ability of the MSLRM system to perform its required function (as changed by this DCN) are not adversely affected by this modification. This modification will not prevent any associated systems from performing their safety-related functions and is therefore acceptable from a nuclear safety standpoint.

Technical Specification changes required to support this modification were approved (Technical Specification Amendment 322) and no additional Technical Specification changes are required. This DCN requires changes to the UFSAR for Unit 2.

This modification is acceptable from a nuclear safety standpoint and no unreviewed safety question is involved.

DCN W22478 - MSLRM Replacement - Unit 3

<u>Description/Safety Evaluation</u>

This DCN replaced the MSLRMs with more accurate models. DCN W22478 also deleted the following safety-related reactor scram and primary containment isolations initiated by the MSLRMs. The replacement of the MSLRMs is similar to modifications performed on Unit 2 by DCN H1263.

MSLRMs (GE Model 194X629), located in main control room Panel 3-9-10, were replaced with GE Nuclear Measurement Analysis and Control (NUMAC) digital radiation monitors. The following MSLRMs are affected:

- 3-RM-90-136 MSLRM Channel A
- 3-RM-90-137 MSLRM Channel C
- 3-RM-90-138 MSLRM Channel B
- 3-RM-90-139 MSLRM Channel D

MSLRM system operation and protective functions required by the existing design basis were changed by this DCN. Calculations performed by this DCN demonstrate that the changes made by this DCN will not increase offsite radiation doses above the limits of 10CFR100. The operation and ability of the MSLRM system to perform its required functions (as changed by this DCN) are not adversely affected by this modification. This modification will not prevent any associated systems from performing their safety-related functions and is therefore acceptable from a nuclear safety standpoint.

This modification did require a Technical Specification change and the system design and functional requirements as described in the UFSAR were significantly affected by this DCN.

No unreviewed safety question is involved since NRC approval was obtained prior to implementation and return to operation of this modification.

DCN W22500 - Addition of Filter Capacitors to Signal Circuit for Rosemount Transmitters - Unit 3

Description/Safety Evaluation

This DCN installed capacitors across GEMAC flow modifiers (square root converters) in flow loops to filter process noise fluctuations. The new capacitors were installed in Panels 9-19, 9-29, and 9-38 in the auxiliary instrument room for the feedwater, recirculation, RHR, and radwaste system flow loops. A similar modification was performed on BFN Unit 2 as an addition to the scope of ECN P0381 by Field Change Request 86-204. ECN P0381 had been worked and closed for all three units without the filter capacitors being installed on Unit 3.

The components and associated circuits affected by this modification provide non-safety related indication to the main control room and signal inputs to the reactor feedwater and recirculation pump control circuitry. This modification had no affect on any safety related components or system operability and functions. Therefore, the change is acceptable from a nuclear safety standpoint.

The components and associated circuits affected by this modification are not listed or described in the Technical Specifications. Therefore, no Technical Specification change was required. No unreviewed safety question is involved.

DCN W22767 - Installation of Helium Leak Test Connections at Condenser Vacuum Pumps - Unit 3

Description/Safety Evaluation

DCN W22767 provided the design to install offgas system air inleakage test connections on the suction and discharge piping of main condenser vacuum pumps 3A and 3B. The new test connections were used to install a portable helium leak detector on the condenser vacuum

pump piping to conduct offgas system air inleakage testing. The test connections consist of a welded pipe nipple with an isolation valve and a 1/4 turn quick connect hose coupling and are located on the suction and discharge piping of each vacuum pump between the pump and the inlet and outlet check valves.

Offgas system air inleakage testing is performed in accordance with Technical Instruction (TI) 2-TI-55 to identify sources of air inleakage in systems penetrating the main condenser, including flanges, valves, penetrations, and other components exposed to condenser vacuum. Installation of the test connections affords an alternate means of conducting air inleakage testing by allowing use of the condenser vacuum pumps to maintain condenser vacuum during testing instead of the steam jet air ejectors. Similar test connections have been installed for the Unit 2 condenser vacuum pumps by Temporary Alteration Control Form (TACF) 2-84-87-66 and have been documented by ECN P5332.

This modification has no affect on the normal operating characteristics of the offgas system nor does it affect any safety related equipment.

No Technical Specification change is required.

No unreviewed safety question is involved.

DCN T25331 - Installation of Chart Paper and Pen Supply Cabinets- Unit 2

Description/Safety Evaluation

This DCN permanently mounted two recorder chart paper and pen supply cabinets in a former janitor's closet above the stairwell at the P-line wall in the Unit 2 control room.

By moving the supply cabinets into the room, the total paper quantity was increased to 16 ft^3. Combustible load calculations have been revised. The fire severity increases from 57 minutes to 182 minutes. However, the paper is enclosed in steel cabinets. Therefore the fire severity will be significantly less than the estimated values. The room is also protected with a photo-electric smoke detector. Hence, a fire would be promptly detected and annunciated in the main control room and manual fire extinguishing measures can be taken. Also, alternative shutdown capability is available for a fire in this area in accordance with Appendix R safe shutdown procedures. The safe shutdown capability of the plant is not affected.

No Technical Specification change is required. The Fire Protection Report, Volume 1, will be revised to reflect the combustible loading change in the area.

No unreviewed safety question is involved.

DCN T25455 - Replacement of Unit 1/Unit 2 Battery and Battery Board Room Flow Switches - Unit 0

Description/Safety Evaluation

An exception to the system design criteria down-graded the ventilation system associated with the Units 1 and 2 250VDC battery rooms to nonsafety related. System evaluations and administrative controls have been provided to assure the 250VDC batteries can still achieve their safety-related function.

Since the outdoor air supply and exhaust ventilation system associated with the Unit 1 and Unit 2 250VDC battery rooms no longer have the protective safety function of maintaining a negative pressure with respect to surrounding areas, air flow switches associated with that ventilation system was removed by this DCN.

DCN T25455 rewired the control circuits associated with the Units 1 and 2 250VDC battery ventilation system blowers to remove the flow switches and the time delay relays and place an electrical interlock between each set of blowers (i.e., between 1A and 1B exhaust blowers, between 1A and 1B supply blowers, between 2A and 2B exhaust blowers, and between 2A and 2B supply blowers). The new electrical interlock will start the opposite train blower upon the deenergization of the selected train blower. The blower train is selected via the existing handswitches. Air flow difficulties will activate the existing alarm as before.

DCN T25455 revised the System 31 Equipment Management System (EMS), mechanical control diagram 0-47E931-6, mechanical flow diagram 0-47E865-4, electrical schematic diagrams 0-45E779-18 and 2-45E779-18, and associated connection diagrams.

The subject instrumentation or associated ventilation equipment is not described in the Technical Specifications. Consequently, a change to the Technical Specifications is not required.

UFSAR Figures 8.5-7b, 8.5-8a, and 10.12-2a require a revision as a result of this modification. This change had no adverse affect on nuclear safety. No unreviewed safety question is involved.

DCN W25875 - Replacement of Demineralizer Vessel Internals - Unit 2

This DCN involved modifications of the condensate filter-demineralizer system related to the internal parts and elements of each condensate demineralizer vessels.

The condensate filter-demineralizer system for each reactor unit consists of nine filter-demineralizer units, a backwash system, a precoat system, and a body feed system. The condensate filter-demineralizer system is used to remove ionic and particulate material from feedwater so as to maintain a high reactor feedwater quality. The system minimizes corrosion products entering the reactor which could affect fuel performance and accessibility to primary system components, and reduce the capacity required of the RWCU system. The equipment is also used to protect the primary system against intrusion of foreign materials, especially chlorides, which could occur due to condenser leakage.

The condensate water requires a high degree of purity in order to meet fuel warranty requirements. These requirements by GE specify a low level of iron content <5 parts per billion (ppb) and a low level of conductivity <.1 micromho/cm. A high level of iron in the condensate water may cause plate out on the fuel rods with subsequent hot spots.

The existing condensate filter-demineralizer system at BFN had become very inefficient (short run time between precoats) with consequent excessive generation of resin. Replacement resin and its disposal is expensive.

By implementing this modification, the efficiency of the system will increase and the run length between each precoat will be extended resulting in significant savings in precoat material and resin disposal cost and an overall reduction in the handling of radioactive waste.

This modification installed bigger filter elements but this did not change the function of the system.

This change is acceptable from a nuclear safety standpoint. No Technical Specification changes are required. A change is required to the UFSAR in regard to the filtration area of the new elements in the demineralizer vessels.

No unreviewed safety question is involved.

DCN T26769 - Floor Drain Collector Pump Impeller Changeout - Unit 0

Description/Safety Evaluation

This change completed the design work needed to document the installation of the 8 1/4" impellers in the floor drain collector (FDC) pumps in association with Special Test (ST) 89-06. The original design of these pumps called for 7 7/8" impellers. The larger impellers were installed as part of the ST to determine if they would result in improved efficiency of the floor drain filters. This change also corrected the applicable drawings to indicate that the Cation Floc addition line to the FDC tank is actually 1/2" O.D. tubing instead of 3/8" as was currently shown.

Implementation of this change does not adversely affect the function or operation of the radwaste system. This change is intended to increase the capacity of the FDC pumps and also improve the efficiency of the floor drain filters. The functions and flow paths of the radwaste system remain unchanged. The portions of the radwaste system affected by this change are nonsafety related and cannot cause an accident. These portions of the radwaste system are adequately designed for the increased pressures that will result from the larger diameter impellers.

This change does not affect any information presented in the Technical Specifications. This change does however affect radwaste flow diagrams 0-47E830-2 and -3 which are the parent drawings for UFSAR Figures 9.2-3b and 9.2-3c.

This modification does not have an adverse affect on nuclear safety and does not involve an unreviewed safety question.

DCN T27897 - Modification of Circuits and Setpoints on Offgas Flow Instrument Loops-Unit 2

Description/Safety Evaluation

This safety evaluation addresses the non-class 1E 2-F-66-111A/B instrument loops that measure and record the Unit 2 offgas system flow rates at the 6-hour holdup pipe.

This modification removed the low flow alarm associated with 2-FS-66-111B (20 standard cubic feet per minute (SCFM) decreasing on range 0 to 300 SCFM) to prevent nuisance alarms from this instrument. The low setpoint for 2-FS-66-111A was changed from 6 SCFM to 8 SCFM for greater margin at that low flow.

To eliminate process noises displayed on indicator 2-FI-66-111A, a vendor supplied capacitor was replaced with a higher capacitive value.

These instruments do not provide any safety function or any control functions. The above loops provide only visual indication (local and main control room), recorder output, and alarm annunciation in the main control room. These instruments do not affect the operation of the offgas system.

No Technical Specification change is required.

No unreviewed safety question is involved.

DCN T27975 - Settings for Limit Switch (LS), LS-5 - Unit 2

Description/Safety Evaluation

This DCN revised the settings for the LS-5 limit switches for the Unit 2 MSIVs from 90% open to 85% open as recommended by GE Service Information Letter (SIL), GE SIL 568, and further discussed in NRC Information Notice 94-08. In effect, the limit switches will be actuated with the valve 1/2" further closed than was currently the case (BFN's MSIVs have total stem travel of 10"). This change does not affect the settings for the MSIV 90% open limit switches (LS-3 and LS-4) that initiate a reactor scram via the RPS. This change also removes the limit switch setting for the affected switches from the main steam instrument tabulations (I-Tabs) (0-47B601-001 Series) and mechanical control diagram 2-47E610-1-1. This information will continue to be presented on 2-730E927RF-6 and 2-45N2631-3 and -4.

The LS-5 limit switches affected by this change are provided only to turn on and off green position indication lights. The settings for these limit switches are not discussed in the Technical Specifications. Therefore, this change does not affect any information presented in the Technical Specifications.

UFSAR Section 4.6 describes the design of the MSIVs including a discussion of the function of the affected limit switches and their setting at 90% open. Therefore, this change does affect this text.

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

DCN W29505 - Install Lube Oil Purifier for the Unit 3 Main Turbine Oil Tanks - Unit 3

Description/Safety Evaluation

DCN W29505 installed a permanent turbine lubricating oil purifier for the Unit 3 main turbine and reactor feed pump turbine oil tanks. Revision A of DCN W29505 implemented Stage 1 of the design change which installed a welded pipe connection and isolation valve on the 4" drain piping for the Unit 3 main turbine oil tanks. Revision B implemented Stage 2 of the design change which installed the remainder of the modifications including the lube oil purifier skid, associated piping and fire detection and suppression features.

This design change met the design, material, and construction standards applicable to the affected systems and structures. The modification does not affect any safety-related equipment.

This modification is safe from a nuclear safety standpoint. No Technical Specification change is required. A change to the Fire Protection Report is required to reflect minor changes to the high pressure fire protection (HPFP) piping serving the main turbine oil tanks and the surrounding area.

No unreviewed safety question is involved.

DCN T30397 - Change Feedback on Recirculation Flow Control Loops - Unit 2

Description/Safety Evaluation

DCN T30397 was issued to delete the speed feedback signal for the MG set tachometer to the Error Limiter circuit in the recirculation flow control system. The existing system controlled generator speed using the feedback signal and had exhibited unstable operation at higher pump speeds.

This change eliminates the speed feedback loop and replaces it with scoop tube position demand feedback. This will convert the recirculation control system from a speed controller to a scoop tube position controller. This removes a feedback loop which facilitates self sustaining oscillations initiated by an electrically or mechanically induced perturbation.

The recirculation flow control system is not discussed in the Technical Specifications or its Bases, therefore, no change is required to the Technical Specifications. This change revises the description of the recirculation flow control system circuit function and operation discussed in the UFSAR.

This modification does not decrease nuclear safety and no unreviewed safety question is involved.

DCN S30691 - Relocation of Outboard Containment Isolation Boundary - Unit 2

Description/Safety Evaluation

DCN S30691 was issued for Unit 2 to change the outboard containment isolation boundary in feedwater line B (penetration X-9B) which includes RWCU return check valve 2-CKV-69-579 and CRD return check valve 2-CKV-85-576 to a single boundary valve, RWCU return check valve 2-CKV-69-630.

This change will improve 10CFR50 Appendix J Local Leak Rate Test (LLRT) on penetration X-9B. Leakage on the current containment isolation boundary valve, RWCU return check valve 2-CKV-69-579, usually requires extensive maintenance in order to get it to pass LLRT leakage criteria. A redundant check valve, 2-CKV-69-630, was installed in this line under DCN W18298 to preclude single check valve failure blowdown of the feedwater system, an environmental analysis concern. This valve is correctly configured (including installed LLRT test vent/drain connections) to function as a containment isolation valve and replaces two parallel containment boundary leakage paths, check valves 2-CKV-69-579 and

2-CKV-85-576. This modification required no changes to any piping system design pressure or temperature ratings nor did it change seismic qualifications of any systems or components. Check valve 2-69-630 was procured and installed to ASME III, Class I and Seismic I safety-related criteria, thus, this valve meets all design and Quality Assurance (QA) requirements to serve as a containment isolation valve.

This DCN changed the piping classification downstream of check valves 2-CKV-69-579 and 2-CKV-85-576 to the upstream side of valve 2-CKV-69-630 from TVA Piping Class B to Class C. The ASME Section XI Inservice Inspection (ISI) boundary was changed to reflect the same boundary. Containment isolation valve drawings were revised to delete check valves 2-CKV-69-579 and 2-CKV-85-576 from the containment isolation function and add check valve 2-CKV-69-630 as the new isolation valve. These are design document changes only, no field work was required.

No Technical Specification changes are required. The UFSAR required revision to document the change in the containment isolation boundary.

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

DCN T31546 - Removal of Low Condenser Vacuum Scram - Unit 2

Description/Safety Evaluation

This modification eliminates the possibility of an inadvertent scram from "Low Condenser Vacuum" initiated by pressure switches 2-PS-2-1A/1B/5A/8A. The pressure switch setpoints were reduced by DCN M00074A to 0.8 in Hg (Mercury) vacuum eliminating any sensing capabilities of the switches to scram the reactor, but the switch circuits were left intact which could cause inadvertent scrams.

The basis for the turbine condenser low vacuum scram was to provide an anticipatory scram to reduce the reactor vessel pressure increase, caused by a turbine trip on low condenser vacuum. The low vacuum setpoint of 2-PS-2-1A/1B/5A/8A (25 in Hg vacuum decreasing) was selected to initiate a scram before closure of the turbine stop valves on low condenser vacuum, initiated by pressure switches 2-PS-47-72A/72B/73A/73B/74A/74B (setpoint 21.8 in Hg vacuum decreasing).

In the BFN, Unit 1, 2, 3 Safe Shutdown Analysis, ND-Q0999-910033 R9, no credit is taken for the low vacuum scram anticipatory signal provided by pressure switches

2-PS-1A/1B/5A/8A. The NRC issued Amendments 89, 113, and 118 to BFN Units 1, 2, and 3 Facility Operating License Nos. DPR-33, DPR-52, and DPR-68. These amendments revised the Technical Specifications requirements, tables, and bases to eliminate the main condenser low vacuum scram initiated by pressure switches 2-PS-2-1A/1B/5A/8A; therefore, no Technical Specification changes are required. The UFSAR text and figures require revision to delete these switches from Unit 2.

This change does not decrease nuclear safety and no unreviewed safety question is involved.

DCN T31916 - Setpoint Change for 2-TA-66-108B - Unit 2

Description/Safety Evaluation

The offgas system is part of the gaseous radwaste system, which collects and processes gaseous radioactive wastes from the main condenser air ejectors, the startup vacuum pumps, and the gland seal condensers, and controls their release to the atmosphere through the plant stack so that the total radiation exposure to persons outside the controlled area is ALARA and does not exceed applicable regulations.

Troubleshooting on Work Order 94-10708-00 and Problem Evaluation Report (PER) BFPER940343 addressed low temperature nuisance alarms that affect the low temperature alarm for loop 2-TRS-66-108. GE made the recommendation that the low temperature alarm (TRS-66-108) be set at 39°F. This new setpoint allows for better than design performance of the cooler condenser component upstream of the moisture separator. This DCN lowers the existing setpoint value of 42°F (2-TA-66-108B), annunciated in 2-XA-55-53, Window 17. The system still functions as designed but the nuisance alarms are reduced.

Recorder 2-TRS-66-108 and its associated loop components are not described in the Technical Specifications. Therefore, a change is not required to the Technical Specifications. The UFSAR is not affected.

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

DCN S32301 - Relocation of Outboard Containment Isolation Boundary - Unit 3

Description/Safety Evaluation

DCN S32301 was issued to change the outboard containment isolation boundary for Penetrations X-9A and X-9B. For Penetration X-9A, the boundary was relocated from RWCU check valve 3-CKV-69-624 to check valve 3-CKV-69-628. For Penetration X-9B, the boundary was relocated from RWCU check valve 3-CKV-69-579 and CRD system check valve 3-CKV-85-576 to RWCU check valve 3-CKV-69-629.

Valves 3-CKV-69-628 and 3-CKV-69-629 were originally installed to preclude a single check valve failure blowdown of the feedwater line, an environmental equipment qualification concern. However, they are suited for the Appendix J containment boundary function.

These valves are correctly configured (including installed LLRT test connections) to function as containment isolation valves. Additionally, the relocation of the LLRT boundary for Penetration X9-B to check valve 3-CKV-69-629 will eliminate a potential leak path through the CRD system. This modification did not require changes to any piping system design pressure or temperature ratings nor did it change seismic qualifications of any systems or components. Check valves 3-CKV-69-628 and 3-CKV-69-629 were procured and installed to ASME, Class I and seismic safety-related criteria, thus they meet all design and QA requirements to serve as containment isolation valves.

This DCN changed the piping classification downstream of check valves 3-CKV-69-579 and 3-CKV-85-576 to the upstream side of 3-CKV-69-629 from TVA piping Class B to Class C. Also, the piping classification downstream of check valve 3-CKV-69-624 to the upstream side of 3-CKV-69-628 changed from Class B to Class C. The ASME Section XI ISI boundary was changed to reflect the same boundaries. This DCN updated containment isolation valve drawings. These are design document changes only, no field work was required.

No Technical Specification change is required. This change does affect UFSAR Figure 5.2-22 Sheet 3 which identifies the Unit 3 containment isolation valves.

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

DCN T33096 - Appendix R Manual Actions - Unit 2

Description/Safety Evaluation

This safety evaluation was written in support of DCN T33096 which was generated to provide corrective action for PER BFPER940763 and in support of an associated revision to Volume 1 of the Fire Protection Report. The purpose of the DCN was to transmit revised Appendix R manual action requirements to the plant which will be incorporated into the Safe Shutdown Instructions which are used to safety shut down the plant in the event of an Appendix R fire. These revised manual actions simply ensure that the equipment and power alignments relied upon for the safe shutdown of the plant are available for a fire in any given area of the plant. The manual actions consist of aligning normal or alternate power supplies (depending on which is available for a given area) and manually starting/stopping equipment for certain areas. These manual actions did not result in any physical changes being made to the plant.

The transfer of plant equipment to its alternate power supply is within the original design basis of the plant and does not affect the function or operability of any plant systems.

No Technical Specification change is required.

Volume 1 of the Fire Protection Report will require revision to update the Appendix R Safe Shutdown Analysis and the Appendix R Safe Shutdown Program.

No unreviewed safety question is involved.

DCN S33326 - Drawing Revision to Allow Existing Coating on Stainless Steel Torus to Remain in Place - Unit 2

Description/Safety Evaluation

This DCN revised design drawings to allow the existing coating on stainless steel in the Unit 2 torus to remain in place.

Coating has been found applied to the following stainless steel structures within the Unit 2 suppression chamber:

- T-quenchers
- Main vent bellows
- Miscellaneous support steel on the catwalk
- Electrical conduits
- Electrical junction boxes
- Small bore piping and valve bodies

The coating applied to the stainless steel components inside the suppression chamber is Valspar 78, which is qualified for design basis accident (DBA) conditions when applied to carbon steel surfaces.

Stainless steel does not require any coating. The coating that was applied (Valspar 78) is a DBA qualified and approved coating system for Browns Ferry Coating Service Level I use over carbon steel. It has not been tested and qualified over stainless steel. The items listed above were initially coated with Valspar 78 in the early 1980's time frame and were subsequently recoated during follow up coating activities in the torus. However, the stainless steel components were sandblasted and coated by qualified individuals using approved procedures for the surface preparation and application of this coating on carbon steel.

Due to the adhesion of the coating to the stainless steel T-quenchers and the other structures and components, the coating will be left in place and DCN S33326 was issued to show this condition on applicable drawings.

TVA has performed tests and technically justified the existing coating on stainless steel structure/components within the suppression chamber. It will not dislodge during any anticipated accident or transient. Any disbonding of the coating that might occur will be in the form of small particles that cannot impair the flow of recirculation water through the ECCS strainers.

Consequently, leaving the coating on the structures/components is safe from a nuclear safety standpoint and does not require any changes to Technical Specifications. The UFSAR will require a revision to identify the coating situation within the suppression chamber.

No unreviewed safety question is involved.

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SUMMARY OF SAFETY EVALUATIONS FOR FIRE PROTECTION REPORT REVISIONS

Fire Protection Report Volume 1

Description/Safety Evaluation

This safety evaluation addresses a revision to the SSDP in the Fire Protection Volume 1. This change was made to incorporate the corrective actions as identified in PER BFPER930143. The corrective actions required review of Section III (Required Safe Shutdown Equipment) and Section V (Testing and Monitoring) of the SSDP. The review identified safe shutdown components for which testing requirements are either not provided or incorrectly stated. Appropriate changes to the SSDP were made based on these reviews. The system descriptions provided in Section VI of SSDP were also revised to reflect the current system configurations. Section VII of the SSDP (Revision/Control of the Program) was revised to reflect the current method of controlling changes to the Fire Protection Report. Changes to the Fire Protection Report are now controlled in accordance with SSP-12.15.

These changes are acceptable from a nuclear safety standpoint.

Fire Protection requirements are not described in the Technical Specifications. Therefore, no Technical Specification change was required. UFSAR text and tables are affected by these changes.

No unreviewed safety question is involved.

Fire Protection Report Volume 1

Description/Safety Evaluation

This safety evaluation addresses a revision to the SSDP in the Fire Protection Volume 1.

The SSDP portion of the Fire Protection Report identifies equipment required to ensure a safe shutdown of Unit 2 following an Appendix R fire. Section III of the SSDP provides a listing of the equipment required for safe shutdown following and Appendix R fire. This section also identifies the required compensatory measures to be taken if a required piece of equipment is not able to perform its Appendix R function. The technical basis for the compensatory actions described by Section III are justified in Section V of the program.

This change was made to clarify the compensatory actions to be taken in the event that a component identified by Section III of the SSDP is not considered capable of performing its Appendix R function, while still Technical Specification operable. The change will require that when a piece of Appendix R equipment does not meet spacial separation requirements (i.e., failure to meet 10CFR50 Appendix R Section III.G.2.b or where safe shutdown components are identified outside their designated area), compensatory measures shall be established in accordance with Paragraph 9.3.11.G.1.a of the Fire Protection Plan.

These changes are acceptable from a nuclear safety standpoint.

Fire Protection requirements are not described in the Technical Specifications. Therefore, no Technical Specification change was required. UFSAR text and tables are affected by these changes.

No unreviewed safety question is involved.

Fire Protection Report Volume 1

Description/Safety Evaluation

This safety evaluation addressed a change to the Fire Protection Report Volume 1. The change will update, clarify, enhance, remove discrepancies and errors in text or tables.

These changes are acceptable from a nuclear safety standpoint.

Fire Protection requirements are not described in the Technical Specifications. Therefore, no Technical Specification change was required. UFSAR text and tables are affected by these changes.

No unreviewed safety question is involved.

Fire Protection Report Volume 1

Description/Safety Evaluation

This safety evaluation addressed a change to the firewatch/compensatory measure requirements associated with the following preaction system areas of protection (i.e., those areas containing redundant safe shutdown equipment):

- 1,2,3-FCV-26-77 (Reactor Building General Area Suppression System, Elevations 541' [Water Curtain], 565' 593', and 621')
- 1,2,3-FCV-26-88 (Recirculation MG Set AFFF Suppression System, Elevation 639')

Presently, when one or more of the required sprinkler systems is partially or completely impaired (i.e., part or all of the area of protection is declared inoperable), a continuous/area fire watch is established within one hour for each of the above listed impaired systems (2 systems per unit, 3 units, therefore, possibly 6 individual fire watches). The responsibilities of these fire watches are to move throughout the area of impairment (within the same preaction protection area) once each hour (he/she is not to be stationed in one specific area but is to travel throughout the entire area of impairment, and he/she is to not leave the area of impairment without proper relief). This connotates that the firewatch is to be "dedicated" to this particular impairment and have no other assigned function, but the firewatch is not restricted in his/her deployment. For clarification, under the present requirements of the Fire Protection Report, this firewatch can only act as a compensatory measure for specified areas under one of the affected preaction coverage areas (he/she under no circumstances, can act as the required compensatory measure for an area outside the preaction area of protection he/she is presently monitoring). This change is to document (in Table 9.3.11.B of the Fire Protection Report Volume 1) the ability of this firewatch to perform the functions of compensatory measures as a continuous/area fire watch for these preaction areas of impairment (as mentioned above) in areas protected by one or more of the associated preaction sprinkler systems. This will provide (i.e., crossing from one area of protection to another in the same unit and/or crossing unit boundaries, if feasible) to act as compensatory measures and still satisfy the requirement of the Fire Protection Report to monitor each area of impairment at least once per hour.

This change will also address detection systems serving the same areas of protection as indicated above (and will be documented in Table 9.3.11.A of the Fire Protection Report Volume 1).

This change is acceptable from a nuclear safety standpoint. This change is administrative in nature and has no impact on the defined margins of safety in the Technical Specifications. The existing Technical Specifications and the UFSAR (referenced over to the Fire Protection Report) are adequate in permitting safe implementation of this change. The function of the firewatches as defined in this change are not modified.

No unreviewed safety question is involved.

Fire Protection Report Volume 1

Description/Safety Evaluation

This safety evaluation addressed an increase in total combustible loading for the main control room areas. The increase in combustible loading resulted due to installation of control room work stations and carpeting.

The increase in combustible loading in the main control rooms due to installation of work stations and carpets has minimal impact on the fire hazards of the area. The change does not affect the analyzed fire event (Unit 2 Appendix R safe shutdown analysis). The alternate safe shutdown capability is not compromised in the event of a control building fire.

The modifications to the control rooms were accomplished by DCN W17256.

The existing Technical Specifications are adequate to permit safe implementation of this change.

No unreviewed safety question is involved.

Fire Protection Report Volume 1

Description/Safety Evaluation

The purpose of this safety evaluation was to determine the safety implications of changes described in Fire Protection Change Notice FPR-94003 and affect the description of the sprinkler system. The revision reflects upgrades to the Unit 3 reactor building sprinkler systems to meet the NFPA code requirements. The changes have no impact on the Unit 2 Appendix R analysis nor does it affect the capability, performance, or function of any component important to safety.

The modifications to the automatic preaction sprinkler system in the Unit 3 reactor building were accomplished by DCNs W18048, W18049, and W18050.

The existing Technical Specifications are adequate to permit safe implementation of the change.

No unreviewed safety question is involved.

Fire Protection Report Volume 1

Description/Safety Evaluation

The purpose of this safety evaluation was to determine the safety implications of changes described in Fire Protection Change Notice FPR-94004. Changes are being made in Section 5.2 of the Fire Hazard Analysis to reflect the method in which the adequacy of the fire pumps is determined. Flow tests are being performed in lieu of pressure drop calculations to determine the adequacy of fire pumps.

The Appendix R Safe Shutdown Program (Section III) Compensatory Measures are being revised for the shutdown board rooms and the cable spreading rooms. The continuous fire watch requirement is being changed to an hourly roving fire watch since NFPA code compliant fire detection systems have been installed in the shutdown board rooms and the cable spreading rooms.

This change does not affect the Appendix R required equipment or other Technical Specification systems. The existing Technical Specifications are adequate to permit safe implementation of this change.

The changes to be implemented by this revision do not involve, influence, or change any system operational characteristics as described in the UFSAR.

No unreviewed safety question is involved.

Fire Protection Report Volume 1

Description/Safety Evaluation

This safety evaluation evaluates a change to revise the combustible loading (heat of combustion per unit area - BTU/ft^2), area square footage, and fire severity information for Fire Areas 16, 18, 19, 21, and 25 in the Fire Protection Report. These changes resulted from a review as part of the corrective action to BFPER940949.

The combustible fire loading information in the Fire Protection Report is utilized to assess the magnitude of a potential fire in the area, and the corresponding capability of the fire suppression systems to extinguish such fires. The safety evaluation shows that the changes in combustible loading are either insignificant or present no significant challenge to the fire barriers. The potential fire is still contained within the fire barriers and does not affect the redundant safe shutdown capability.

The existing Technical Specifications are adequate to permit safe implementation of this change. No Technical Specifications change is required.

No unreviewed safety question is involved.

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SUMMARY OF SAFETY EVALUATIONS FOR NEW INSTRUCTIONS OR PROCEDURE REVISIONS

Surveillance Instruction (SI) 2-SI-4.7.A.2.a-f - Primary Containment Integrated Leak Rate Test (ILRT)

Description/Safety Evaluation

2-SI-4.7.A.2.a-f implements the requirements of 10CFR50, Appendix J, and Technical Specification 4.7.A.2. The requirements include leak testing primary containment at accident pressure once ever 40 ± 10 months during cold shutdown condition. This leak rate test involves measuring temperature, pressure, and dewpoint of the primary containment atmosphere with the reactor vessel vented over an 8-hour (minimum) period at a minimum differential pressure of 49.6 pounds per square inch (psi). The actual test pressure is 50.8 ± .4 psi. The data is used to correlate the mass leak rate of primary containment atmosphere at accident pressure over a 24-hour period. This instruction is performed in conjunction with 2-TI-179 that inspects primary containment, leak tests the core spray pump seals, and bubble tests air leakage paths from primary containment.

The revision to 2-SI-4.7.A.2.a-f will have no negative effects on the UFSAR described safety functions. This SI will ensure the leak tight integrity function of primary containment as required by the Technical Specifications.

No unreviewed safety question is involved.

2-TI-275D Revision 4 - Drywell Leak Investigation - Chemistry

Description/Safety Evaluation

This procedure provides information to assist personnel in investigating unidentified drywell leakage. The procedure outlines sampling and analyses designed to yield information as to the sources(s) of drywell leakage. The procedure was revised to include the use of a tracer chemical which when injected at the appropriate point(s) will assist in identifying the source(s) of drywell leakage when the usual methods fail.

The specified tracer is Rubidium Nitrate ($RbNO3$). $RbNO3$ was selected because of its relatively low impact on plant systems and its prior use as a tracer at another utility. After injection, samples of drywell sumps are to be analyzed for the rubidium cation by ion chromatography. The presence of rubidium cation in the drywell floor drain sump will indicate that leakage originated from the tested source. The quantity of rubidium cation in the drywell floor drain sump may allow the quantity of leakage from the tested source to be estimated.

When the tracer is used, samples of the drywell floor drain and drywell equipment drain sumps will be obtained by Chemistry (with assistance from Operations) as the sumps are pumped over to radwaste. Work requests will be submitted to connect temporary sample lines for sampling the sumps, as required. Injection of the tracer will be performed by Technical Support. Details concerning the injection apparatus are outside the scope of this procedure. Work requests will be submitted to make connection for injection.

The amount of tracer which is to be added will result in only very slight increases in conductivity and activity of the reactor coolant. These increases will be insignificant in comparison with the limits of Technical Specification 3.6.B/4.6.B. No increase in chloride concentration or change in pH of the reactor coolant are anticipated. No damage to 304 stainless steel or Zircaloy cladding are anticipated. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

No Technical Specification changes are required. No unreviewed safety question is involved.

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SUMMARY OF SAFETY EVALUATION FOR SPECIAL OPERATING CONDITION

Final Feedwater Temperature Reduction (FFWTR) Operation

Description/Safety Evaluation

This safety evaluation addresses the use of a mode of operation with FFWTR. The primary use of FFWTR is intended to be continuous operation with FFWTR at end of cycle; however, FFWTR operation can be used as a contingency mode of operation during the cycle.

With normal feedwater heating, a reactor reaches the end of full power capability and begins coasting down in power. The feedwater temperature for BFN is calculated to be reduced by approximately 45°F at rated power conditions by intentionally valving out the last stage high pressure feedwater heaters. By reducing the feedwater temperature, the reactor can be kept at full thermal power for approximately 2 weeks after normal end of full power capability. This mode of operation for cycle extension purposes is called FFWTR operation.

The use of FFWTR operation for cycle extension is described in GE Standard Application for Reactor Fuel, GE Licensing Topical Report - NEDE-24011-P-A and the concept was generically approved by the NRC. However, a plant specific safety evaluation of the impact of FFWTR operation must be performed. In addition, a safety evaluation of the impact of FFWTR operation on cycle dependent reload analysis is required. The FFWTR operation option has existed for several years and has been utilized by many BWRs.

The preferred method of temperature reduction is to close the turbine extraction steam flow path to high pressure heater string #1 (heaters A-1, B-1, and C-1). However, the analyses in GE Report NEDC-32356P show that the use of other methods, including removal of any single heater or combinations of low pressure and high pressure heaters, is acceptable as long as the final feedwater temperature is within the nominal 47°F reduction assumed in the analyses. In addition, the heaters can be removed from service by isolating extraction steam only or by isolating extraction steam, feedwater and condensate. In all cases, operation with feedwater heaters out of service is subject to the requirements of the vendor turbine manual.

The effects of FFWTR for the BFN units on accident events such LOCA, anticipated transients without scram, containment LOCA loads, the mechanical integrity of the reactor internal components and the feedwater nozzle/sparger fatigue usage have been analyzed by GE. The impact of FFWTR on plant operating limits and fuel thermal-mechanical performance is cycle dependent and has been analyzed by GE for the current Unit 2 Cycle 7. Future cycle dependent reload analyses will evaluate the impact of FFWTR on plant operating limits and fuel thermal mechanical performance.

The impact of FFWTR operation on transient and accident analyses is most limiting at end of cycle (all control rods fully withdrawn) and bound the impact of FFWTR operation during the cycle. Use of partially inserted control rods decreases the severity of transients. Since

FFWTR operation is intended primarily for cycle extension at end of cycle, end of cycle will be determined by Reactor Engineering and may occur before all control rods are fully withdrawn. Any use of FFWTR operation before the declared end of cycle is only for contingencies (such as maintenance) and may reduce the value of FFWTR operation at end of cycle.

Operation above the rated power rod line, e.g., in the Extended Load Line Limit Analysis (ELLLA) region of the power flow map, with FFWTR increases the containment loads during a design basis LOCA. The analyses by GE justify only short term operation in the ELLLA region that should not exceed 14 continuous days per occurrence. Contingency FFWTR operation may be used more than once in a cycle if needed as the time in the ELLLA region is not cumulative between separate occurrences.

The impact of FFWTR operation on fatigue of feedwater nozzles and sparger is cumulative and the time in FFWTR operation must be tracked. Operation is considered to be in FFWTR mode if the feedwater temperature is more than 10°F below the normal (all heaters in service) feedwater temperature. Approximately 600 days of FFWTR operation can be accommodated in the 16 year refurbishment cycle.

The GE analyses assumed a bounding reduction in feedwater temperature of 47°F at rated power conditions with an allowance of 10° for uncertainty in calculations and monitoring. Thus, the safety assessment is valid for an observed FFWTR of up to 57°F at rated power conditions. However, the intentional continuous operation with a temperature reduction greater than 47°F is not justified by the analysis and the feedwater temperature should be brought within the analyzed band.

The use of FFWTR mode of operation is acceptable from a nuclear safety standpoint. No Technical Specification revisions are required. A modification to Section 11.8.3.2 (Feedwater Heaters) of the UFSAR will be made to reflect the maximum feedwater temperature reduction justified.

No unreviewed safety question is involved.

1994

SUMMARY OF SAFETY EVALUATIONS FOR SPECIAL TESTS

Post Modification Test 2-PMT-BF-066.004 - Unit 2

Description/Safety Evaluation:

This procedure functionally tested modifications (relocation and/or rewiring of controls) made to control room Panel 2-9-8 by DCN W170360 Stage 1. This safety evaluation specifically addressed the testing of offgas system related components in the main steam system, offgas system, and radiation monitoring system which are associated with the steam jet air ejector (SJAE) operation and inlet/outlet/drain valves auto-isolation logic.

Functional testing of these components with the plant in shutdown or refuel mode required the installation of jumpers in auxiliary instrument room Panel 2-9-36 to simulate condenser vacuum and steam pressure adequate so that the auto-isolation logic for offgas channels A and B could be reset, thereby enabling the controls for these valves. The safety evaluation also addressed any necessary lifting of certain internal wires in Panel 2-9-36 to allow proper verification of contact configuration of control switches 2-HS-1-150 and 2-HS-1-152. These jumpers and wire lifts constituted a temporary alteration to a radwaste system, thus requiring the safety evaluation.

Appropriate administrative controls were used to assure that the as-designed configuration was maintained. The function and performance characteristics of components affected by this test were unchanged. The temporary alterations installed and removed by this test did not affect normal operational parameters, setpoints, calibration intervals, or functional test intervals nor did they affect any Technical Specifications or their bases. No Technical Specification change was required.

This test was acceptable from a nuclear safety standpoint and no unreviewed safety question was involved.

2-PMT-BF-066.006 - Unit 2

Description/Safety Evaluation:

This PMT was performed to verify the design functions of the affected components were unchanged and electrical faults were not introduced into the associated component's circuitry after the installation of DCN W17368 Stage 1. DCN W17368 consisted of modifications to control room offgas Panel 2-9-53. In general, the modification rearranged and/or replaced control switches, instruments, temperature recorders, and indicating lights. This procedure

was performed during the time period when the reactor was in cold shutdown and the offgas system was not required to be operable.

The components modified and tested were nonsafety related and were not required for safe shutdown of the plant. The offgas components tested were outside the boundary of the offgas stack and its associated ducting. Therefore, the offgas nuclear safety functions were not compromised or affected.

No Technical Specification changes were required and no unreviewed safety question was involved.

Special Test 0-ST-93-01 - Units 1, 2, and 3

Description/Safety Evaluation

The purpose of this special test was to perform electromagnetic interference (EMI) mapping at BFN. Phase one performed specific mapping of the Unit 1, 2, and 3 refuel floor and control room locations in proximity to the reactor and refuel zone radiation monitors' detectors and drawers. Subsequent phases will perform EMI mapping of additional areas in all three units to obtain an overall plant EMI profile. Each phase will be detailed in an appendix to this special test with additional appendices added as testing scope and requirements are expanded and defined.

There was no impact on plant systems. EMI mapping is not intrusive in that no cables or equipment is rendered inoperable during the collection of data. Current probes are clamped over power and signal cables to monitor the levels of EMI emanating from them while the plant equipment they interface with is in operation. These probes connect to receiving and recording instruments and do not inject signals into the cables they connect onto. Likewise, oscilloscope probes are attached to selected terminal points and data recorded. Plant handheld radio transceivers and repeaters are keyed on and off in permissible locations to simulate normal use and determine their effect on monitored plant equipment. Antennae connected to receiving and recording instruments are positioned at various locations and rotated to map the EMI profile present in each area surveyed while the plant is in operation. These receiving antennae do not transmit outgoing signals.

This was a special test that gathered data only. It in no way affected system operational characteristics. It did not affect compliance with any Technical Specification nor did it conflict with or alter anything contained in the UFSAR. Normal operational alignment of all

plant equipment was required for this data to be representative of the actual EMI environment by plant instrumentation.

This special test was acceptable from a nuclear safety standpoint and no unreviewed safety question was involved.

1994

SUMMARY OF SAFETY EVALUATIONS FOR TACFs

TACF 0-94-2-67

Description/Safety Evaluation

This safety evaluation addressed the removal of the valve disk from 0-FCV-67-88. This valve aligns/isolates the A1 RHRSW pump to/from the EECW system. This valve also has a limit switch that will align the A1 pump logic to start on an EECW pump initiation signal. This valve had been identified as having a broken disk. The valve had been removed and all the disk had been recovered from the system. This safety evaluation allowed the valve to be reinstalled without the disk and allowed the A1 pump to be used for EECW but inoperable for RHRSW.

This change did not reduce the margin of safety as defined in the basis for any Technical Specification. The alignment maintained while this alteration is in place is an alignment consistent with plant procedures. The alteration will neither affect the ability of the A1 or A3 pumps to supply water to the EECW system, nor will it impede the ability of the A2 pump to supply the RHRSW system.

No Technical Specification change is required.

No unreviewed safety question is involved.

TACF 0-94-04-77

Description/Safety Evaluation

As part of the Radwaste Improvement Program, ST 89-06 was performed to optimize water processing of the radwaste system. Included in this test was the installation of the following equipment:

A. Pall regimesh elements into the floor drain and waste collector filters
B. Addition of accumulators on the discharge and suction of the floor drain and waste collector filter aid pumps
C. Addition of a cation polymer injection system. This system includes a skid mounted tank and pump for the introduction of polymer (Betz polymer 1175) into the floor drain filter vessel. This polymer increases the run time and the effectiveness of the resin. This equipment affects only the floor drain filter system.

Initial plant design has filter aid systems and polymer injection systems installed. However, the filter aid system was not functional until the pulsation dampers were added to each pump by Item 2 above. The permanently installed polymer system has never been used due to incomplete wiring on the pumps. Addition of Item 3 above allowed polymer injection from a different injection path. Based upon the discussion, injection of filter aid and polymer are covered by existing UFSAR.

This test was successful in optimizing radwaste filter performances. Based upon results, Plant Management requested that ST 89-06 remain open to document the physical and operational changes while DCRs were being processed.

The Pall regimesh elements (Item 1) were documented under DCN H7889. DCR 3579, submitted to document Items B and C, has never been implemented.

It should be noted that plant operating procedures associated with this special test document operation of this equipment.

TACF 0-94-04-77 was written to document Items B and C.

This equipment is not required to support any margin of safety and is not considered able to impact any margin of safety as defined in the basis of Technical Specifications. No Technical Specification change is required.

This TACF does not have an adverse affect on nuclear safety and does not involve an unreviewed safety question.

1994

SUMMARY OF SAFETY EVALUATION FOR TEMPORARY SHIELDING REQUEST

Temporary Lead Shielding Within Drywell

Description/Safety Evaluation

This safety evaluation addressed the use of operability criteria as an appropriate basis for ensuring structural/system operability. This evaluation is limited to the installation of temporary lead shielding within the drywell. This evaluation was prepared on a generic basis as installation of lead shielding during various refueling outages may be different. The plant configuration associated with this evaluation is:

- Unit is in cold shutdown
- Drywell head removed
- RPV head removed
- Fuel is in the reactor vessel and the fuel pool gates may be in or out
- Piping systems and equipment remain available to support decay heat removal

The initial basis for this evaluation is Temporary Shielding Request 94-0001 Revision 0. This evaluation is appropriate for other shielding requests provided that the plant configuration discussed above is the same.

The plant systems encompassed by this evaluation include any system with piping in the drywell including associated equipment and structural systems.

The shielding is required to minimize dose during inservice inspections of pipe welds and to lower background exposure levels for other work performed in the drywell. The shielding, in the form of lead blankets, is to be installed on piping, valves, pipe whip restraints, and grating in the drywell during refueling outages.

The shielding request calls for the shielding to either be placed on or suspended from structural features adjacent to specific valves, pumps, and piping segments. In addition, the shielding package allows for roving shield areas in support of inservice pipe weld inspections.

This change is acceptable from a nuclear safety standpoint. This change does not affect the information presented in the Technical Specifications or the Fire Protection Report. The proposed activity does not permanently change any information presented in the UFSAR as the installation of lead shielding is temporary and shall be removed prior to RPV head replacement and the qualification of the temporary lead shielding is based on the use of either long term design basis requirements based on the UFSAR or NRC accepted operability based criteria that are appropriate for temporary conditions.

A safety evaluation was performed since acceptance of shielding packages is in part based on the use of NRC accepted operability criteria that were intended for evaluation of nonconforming and degraded conditions.

No unreviewed safety question is involved.

1994

SUMMARY OF SAFETY EVALUATIONS FOR UFSAR REVISIONS

UFSAR Sections 6.4.1, HPCI System, and 7.4.3.2, HPCI System Control and Instrumentation

Description/Safety Evaluation

This safety evaluation was written in support of corrective action promulgated by PER BFPER940570. This PER identified inaccuracies in the functional descriptions given for the HPCI system flow indicating controller (FIC), 2-FIC-73-33, in the UFSAR and the system design criteria.

The HPCI system is designed to automatically start upon receipt of an initiating signal (low-low reactor vessel level or high drywell pressure) and inject water into the vessel at full design flow rate (5000 gpm). The controls are arranged such that the system can automatically realign to inject into the reactor vessel to fulfill its safety function even though the system may initially be operating in the test mode. When testing the HPCI system, the flow controller could be in either MAN or AUTO with the flow demand signal adjusted to less than full design flow rate. If a HPCI initiation signal is received while in this mode, the HPCI system would automatically realign to its safety related injection position, however, the flow controller would remain in the mode (AUTO or MAN) and at the flow demand setting in use at the time the initiating signal was received. Operator action would be required to place the flow controller back in AUTO and readjust the flow demand signal for full rated design flow. Functional descriptions given in both the UFSAR and the design criteria currently state that when in the test mode and a HPCI initiate signal is received, the flow controller would automatically return to the AUTO mode with flow demand set at full design flow rate.

This change will revise the UFSAR to correctly describe the HPCI flow controller function and the design criteria to clarify the design functional requirements as reflected by the current as-built configuration of the HPCI control system. An additional change to UFSAR Section 7.4.3.2.4 is being made to correct a typographical error that describes the Unit 2 level switch configuration which trips the HPCI turbine. These are documentation changes only, no physical work will be done on the plant.

This change does not decrease nuclear safety. No change is required to the Technical Specifications and no radwaste system(s) are involved.

No unreviewed safety question is involved.

UFSAR Section 8.3, Transmission System

Description/Safety Evaluation

This safety evaluation was written in support of CRLD BFEP-EEB-94003 R0. This CRLD brings UFSAR Section 8.3 up to date with current transmission line/substation configuration and bulk plant loading conditions. It also updates voltage and freqüency graphs for various events, utilizing data collected during peak conditions in the summer of 1993 as baseline data.

Specifically, it revised the text to reflect Unit 2 only operation rather than 3 unit operation. It reflects the current configuration of transmission lines, including transmission line crossings, connections to offsite substations, and connections to onsite transformers. It recognizes the presence of capacitor banks in the 161KV switchyard and describes the revised Section 8.3 figures.

The figure revisions show peak electrical system conditions, electrical system response during a Unit 2 LOCA combined with various electrical equipment failures, electrical system response for loss of a Cumberland Steam Plant unit (one of TVA's largest 2 units), and for trip of or a fault on the BFN Unit 2 generator, and for a fault on the 500KV bus or transmission line.

These changes do not reduce the margin of safety as defined in the basis for any Technical Specification. The UFSAR changes affect the transmission line system, which is non-safety related and is located entirely outside of the nuclear plant buildings.

No Technical Specification change is required.

No unreviewed safety question is involved.

UFSAR Chapter 8 Section 8.10, Station Blackout (SBO)

Description/Safety Evaluation

This safety evaluation was written in support of BFN Unit 2 CRLD BFEP-EEB-94001 R0.

This CRLD added Section 8.10, Station Blackout, to Chapter 8, Electrical Power Systems, of the UFSAR. This section summarizes SBO background and the electrical power system's

capability to provide electrical power in support of coping with a 4 hour SBO event at BFN Unit 2.

A SBO Technical Specification is currently an open item. The NRC will notify BFN if a SBO Technical Specification is required. Therefore, there is no Technical Specification change required at the time of this report.

This change to the UFSAR is acceptable from a nuclear safety standpoint.

No unreviewed safety question is involved.

UFSAR Section 10.6, RBCCW System

Description/Safety Evaluation

This safety evaluation addressed a change to the UFSAR to remove mention of specific parameters/additives and associated limits (listed in Table 10.6-2, RBCCW System Heat Exchanger Operating Conditions) and replace with a statement that there will be additives to minimize corrosion. The parameters/additives and associated limits are listed in Site Standard Practice SSP-13.1, Chemistry Program. The UFSAR change will eliminate the need to revise the UFSAR every time SSP-13.1 is revised to incorporate the most current industry information for corrosion control.

SSP-13.1 reflects the most current information/guidance for the operation of the chemistry program based on regulatory requirements, industry practices, vendor recommendation, and technological advancements. The SSP is consistent with the TVA Nuclear Power Chemistry Manual which superseded the April 1985 DPM N79E2 as the most up to date current source for water chemistry information for BFN. The information contained in the SSP is more restrictive than that in the section of the UFSAR proposed to change.

This change is acceptable from a nuclear safety standpoint.

No Technical Specification changes are required.

No unreviewed safety question is involved.

UFSAR Section 13.1, Organizational Structure for the Conduct of Operations and Section 13.2, Organization and Responsibility

Description/Safety Evaluation

This safety evaluation addressed a total revision to UFSAR Sections 13.1 and 13.2 due to major organizational changes and as part of the annual update. Section 13.1 remains essentially the same except for a new statement which is added as a reference to the Nuclear Power responsibility for preoperational and startup testing programs as discussed in UFSAR Sections 13.4 and 13.5 and BFN Operations has been deleted from Amendment 10, Section 13.1.2 (now discussed in the TVA Topical Report). Section 13.2 organization description has been deleted and has been formatted to reference TVA Topical Report, TVA Nuclear Quality Assurance Plan, and BFN SSP-1.4.

These changes are administrative requirements associated with qualification and training of personnel. These changes are administrative in nature. The NRC SER allows TVA to reference the TVA Topical Report in lieu of the organization description normally found in the UFSAR. This change is acceptable from a nuclear safety standpoint.

No Technical Specification changes are required.

No unreviewed safety question is involved.

UFSAR Section 13.6.5.1, Radiological Emergency Plan

Description/Safety Evaluation

The UFSAR currently states "The plan contains information for control of emergency conditions modeled after those contained in NUREG-0654, Revision 1." In 1992, the NRC issued Regulatory Guide 1.101 Revision 3, Emergency Planning and Preparedness for Nuclear Power Reactors. The issuance of Regulatory Guide 1.101 allows nuclear power plants to utilize the methodology described within the guide to develop emergency action levels as an alternate method to that described within NUREG-0654 Revision 1. Following a review of the Nuclear Utilities Management and Human Resources Committee (NUMARC)/NESP-007 Revision 2 methodology it was determined that it would be utilized at BFN. Based upon this decision, new emergency action levels were engineered and prepared. A review of the UFSAR revealed that a change would be necessary to allow for the use of NUMARC/NESP-007 Revision 2 guideline. This revision provides for the allowance of NUMARC/NESP-007 Revision 2 as described in Regulatory Guide 1.101 Revision 3 dated 1992.

This change is acceptable from a nuclear safety standpoint.

A review of the Technical Specifications applicable to emergency planning were reviewed and no sections were noted effecting this change to the UFSAR. Therefore, no changes to the Technical Specifications are required.

No unreviewed safety question is involved.

UFSAR Appendix M, Report on Pipe Failures Outside Containment

Description/Safety Evaluation

This safety evaluation addressed changes to UFSAR Appendix M. These changes will update this section of the UFSAR with current information on the methods used to analyze outside containment pipe failures. The current method of analyzing high energy line breaks outside of containment meets the requirements of NUREG 0588. This change is not a change to the facility since no physical changes to the plant were made. This change does not affect any existing plant procedures nor requires any revision to any plant procedures.

This change cannot reduce the margin of safety as defined in the basis for any Technical Specification since the analysis in Appendix M uses the existing system design parameters as input and evaluates the effects of pipe failures outside of primary containment on the reactor building environment.

This change to the UFSAR does not impact any Technical Specifications. Therefore, no changes to the Technical Specifications are required.

No unreviewed safety question is involved.

1994

RELEASE SUMMARY

1994 RELEASE SUMMARY
ANNUAL OPERATING REPORT

	GASEOUS RELEASES				LIQUID RELEASES			
MONTH	FISSIONS & ACTIVATION PRODUCTS (CI)	IODINES (CI)	PARTICULATES >8 DAY HALF-LIVES (CI)	TRITIUM (CI)	FISSIONS & ACTIVATION PRODUCTS (CI)	TRITIUM (CI)	DISSOLVED NOBLE GASES (CI)	GROSS ALPHA (CI
JANUARY	8.98E+00	6.74E-04	4.02E-04	9.21E-01	2.16E-02	1.17E+00	1.03E-03	ND
FEBRUARY	6.58E+00	6.19E-04	7.52E-04	6.97E-01	2.14E-02	1.28E+00	6.80E-05	ND
MARCH	7.85E+00	3.17E-04	6.97E-04	8.38E-01	1.60E-02	1.20E+00	1.19E-03	ND
APRIL	1.45E+01	4.07E-04	5.30E-04	9.85E-01	2.01E-02	2.11E+00	1.40E-03	ND
MAY	1.73E+01	4.30E-04	9.12E-04	1.01E+00	2.32E-02	1.31E+00	9.44E-05	ND
JUNE	7.32E+01	4.56E-04	2.86E-04	1.05E+00	1.80E-02	6.93E-01	4.86E-05	ND
JULY	2.80E+01	3.76E-04	2.26E-04	1.31E+00	2.14E-02	7.07E-01	8.71E-05	ND
AUGUST	3.28E+01	3.96E-04	4.61E-04	1.34E+00	1.91E-02	6.22E-01	4.51E-05	ND
SEPTEMBER	2.37E+01	3.92E-04	3.29E-04	1.39E+00	2.14E-02	1.17E+00	4.09E-03	ND
OCTOBER	5.14E-02	4.33E-04	7.08E-04	9.50E-01	5.91E-02	1.42E+00	1.03E-03	ND
NOVEMBER	1.64E-01	6.27E-06	5.42E-04	8.67E-01	9.09E-02	1.76E+00	ND	ND
DECEMBER	1.31E+00	1.49E-05	1.97E-05	3.07E-01	3.05E-02	1.20E+00	2.30E-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 ODCM limits.

1994

OCCUPATIONAL EXPOSURE DATA

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

NUMBER OF PERSONNEL (> 100 MILLIREM) — TOTAL MAN-REM

MO=REACTOR OPS SURVEILLANCE

GROUP	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL PERSONS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL MAN-REM
MAINTENANCE PERSONNEL	204	29	423	656	10.184	1.876	3.378	15.438
OPERATING PERSONNEL	118	9	9	136	23.881	0.797	0.140	24.818
HEALTH PHYSICS PERSONNEL	55	6	24	85	7.004	0.217	0.728	7.949
SUPERVISORY PERSONNEL	45	3	69	117	3.732	0.019	3.810	7.561
ENGINEERING PERSONNEL	51	6	76	133	2.381	0.038	1.928	4.347
MO	473	53	601	1127	47.182	2.947	9.984	60.113

MO=ROUTINE MAINTENANCE

GROUP	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL PERSONS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL MAN-REM
MAINTENANCE PERSONNEL	220	29	889	1138	21.605	1.472	115.814	138.891
OPERATING PERSONNEL	94	3	13	110	7.220	0.024	0.243	7.487
HEALTH PHYSICS PERSONNEL	56	4	11	71	4.767	0.037	0.086	4.890
SUPERVISORY PERSONNEL	36	3	79	118	1.387	0.034	9.128	10.549
ENGINEERING PERSONNEL	52	13	89	154	2.659	0.098	3.301	6.058
MO	458	52	1081	1591	37.638	1.665	128.572	167.875

MO=IN-SERVICE INSPECTION

GROUP	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL PERSONS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL MAN-REM
MAINTENANCE PERSONNEL	21	0	189	210	9.705	0.000	80.869	90.574
OPERATING PERSONNEL	0	0	6	6	0.000	0.000	5.830	5.830
HEALTH PHYSICS PERSONNEL	3	0	1	4	0.015	0.000	0.065	0.080
SUPERVISORY PERSONNEL	2	1	8	11	0.382	0.302	4.130	4.814
ENGINEERING PERSONNEL	8	9	69	86	4.688	8.161	76.117	88.966
MO	34	10	273	317	14.790	8.463	167.011	190.264

MO=SPECIAL MAINTENANCE

GROUP	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL PERSONS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL MAN-REM
MAINTENANCE PERSONNEL	138	9	931	1078	10.576	0.042	202.641	213.259

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

	NUMBER OF PERSONNEL (> .100 MILLIREM)				TOTAL MAN-REM			
OPERATING PERSONNEL	37	6	5	48	0.532	0.076	0.067	0.675
HEALTH PHYSICS PERSONNEL	51	1	4	56	1.347	0.003	0.029	1.379
SUPERVISORY PERSONNEL	13	0	75	88	0.073	0.000	8.443	8.516
ENGINEERING PERSONNEL	20	6	88	114	0.611	0.015	8.080	8.706
MO	259	22	1103	1384	13.139	0.136	219.260	232.535

MO=WASTE PROCESING

GROUP	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL PERSONS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL MAN-REM
MAINTENANCE PERSONNEL	20	3	29	52	0.300	0.020	0.450	0.770
OPERATING PERSONNEL	17	0	1	18	0.452	0.000	0.180	0.632
HEALTH PHYSICS PERSONNEL	8	0	0	8	0.062	0.000	0.000	0.062
SUPERVISORY PERSONNEL	6	0	0	6	0.067	0.000	0.000	0.067
ENGINEERING PERSONNEL	0	0	0	0	0.000	0.000	0.000	0.000
MO	51	3	30	84	0.881	0.020	0.630	1.531

MO=REFUEL

GROUP	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL PERSONS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL MAN-REM
MAINTENANCE PERSONNEL	216	30	582	828	59.037	8.819	165.915	233.771
OPERATING PERSONNEL	83	6	12	101	9.383	0.456	3.104	12.943
HEALTH PHYSICS PERSONNEL	47	5	24	76	9.178	0.683	11.189	21.050
SUPERVISORY PERSONNEL	37	3	29	69	5.896	0.195	2.393	8.484
ENGINEERING PERSONNEL	44	10	90	144	5.370	1.009	6.792	13.171
MO	427	54	737	1218	88.864	11.162	189.393	289.419
	========	========	========	========	========	========	========	========
	1702	194	3825	5721	202.494	24.393	714.850	941.737

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 MILLIREM)				TOTAL MAN-REM			
GROUP	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL PERSONS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT AND OTHERS	TOTAL MAN-REM
MAINTENANCE PERSONNEL	819	100	3043	3962	111.407	12.229	569.067	692.703
OPERATING PERSONNEL	349	24	46	419	41.468	1.353	9.564	52.385
HEALTH PHYSICS PERSONNEL	220	16	64	300	22.373	0.940	12.097	35.410
SUPERVISORY PERSONNEL	139	10	260	409	11.537	0.550	27.904	39.991
ENGINEERING PERSONNEL	175	44	412	631	15.709	9.321	96.218	121.248
	========	========	========	========	========	========	========	========
	1702	194	3825	5721	202.494	24.393	714.850	941.737

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

GROUP	STATION	UTILITY	CONTRACT	TOTAL
MAINTENANCE PERSONNEL	265	15	1055	1335
OPERATING PERSONNEL	128	7	13	148
HEALTH PHYSICS PERSONNEL	56	1	24	81
SUPERVISORY PERSONNEL	48	1	85	134
ENGINEERING PERSONNEL	56	17	127	200
	========	========	========	========
	553	41	1304	1898

1994

CHALLENGES TO OR FAILURES OF MAIN STEAM RELIEF VALVES

UNIT 1

None

UNIT 2

During the operating cycle, increasing MSRV discharge tailpipe temperature trends were noted. The tailpipe temperature trends were indications that the MSRV pilot valves were leaking steam. During the same period, the safety-related acoustic monitor system indicated essentially no change. Therefore, the leakage rate was believed to be minimal.

Due to the proximity of the discharge temperature thermowells to the MSRV pilot valve discharge port, minimal leakage rates will be detected by the thermocouples. The existing plant equipment does not provide sufficient process parameters to support a detailed calculation to quantify minimal mass flow rates through the MSRV pilot valves.

Suppression Pool temperature (bulk and individual bays) can be used to determine gross leakage rates due to the increased heat load. During the operating period, Suppression Pool temperatures were routinely monitored. Unaccounted for heat loads were not observed in either the vicinity of the MSRV discharge point or in the bulk temperature.

In accordance with Target Rock Corporation Test Report 3892 (dated: August 5, 1993), MSRV pilot leakage rates of about 200 pounds per hour (lbm/hr) will result in a deviation of the MSRV pilot valve setpoint by approximately + 1 percent (BFN Technical Specification limit). Calculation of the theoretic bulk Suppression Pool temperature rise can be completed under the following assumptions and conditions:

- Assume a MSRV pilot valve leakage rate of 100 lbm/hr (one-half of 200 lbm/hr required for +1 percent setpoint change) of steam.
- Thermodynamic properties based upon saturated steam at 1005 psig with an enthalpy of 1192 BTUs/lbm is condensed in the Suppression Pool at a bulk temperature of about 74°F with an enthalpy 42 BTUs/lbm.
- Assume a Suppression Pool capacity of about 127,000 cubic feet (950,000 gallons) with a specific volume of 0.01605 cubic feet/lbm.
- Assume perfect mixing occurs.

Under these conditions, the bulk Suppression Temperature would increase about 1°F every three days. Localized thermal effects due to non-perfect mixing would have intensified the local bay temperatures and would have resulted much larger local temperature change rates.

Therefore, using the bounding condition of 100 lbm/hr with no Suppression Pool heating, and no acoustic monitor response, there appears to no basis for classifying the observed pilot valve leakage as a pilot valve failure.

UNIT 3

None

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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	0.00572	0.00431
Feedwater nozzles	0.29782	0.22754	0.16139
Closure studs	0.24204	0.22766	0.14360