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

November 1, 2006

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
OWFN P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket No. 50-259
Tennessee Valley Authority	)	

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 1 - SUPPLEMENTAL INFORMATION - CHANGES TO INSTRUMENTATION SURVEILLANCE TEST INTERVALS (STIs) AND ALLOWED OUT-OF-SERVICE TIMES (AOTs)

The purpose of this letter is to provide the BFN Unit 1 plant specific information associated with the NRC approved generic methodology (BWR Topical Reports) supporting instrument surveillance intervals and allowed outage times as discussed in TVA's September 7, 2006 letter (Reference 1). TVA's September 7, 2006 letter responded to NRC's July 7, 2006 letter (Reference 2) pertaining to License Condition 2.C(4).

Specifically, in BFN's Improved Technical Specification (ITS) conversion, TVA referred to several BWROG Licensing Topical Reports (LTRs) used to justify the Surveillance Test Intervals (STIs) and Allowed Outage Times (AOTs) for instrument systems. The BWROG LTRs established the generic basis for supporting the plant specific TS change. TVA's December 11, 1997 letter provided plant specific assessments for Units 2 and 3 that concluded that the generic analyses were applicable to BFN Units 2 and 3. BFN has performed a similar analysis for Unit 1 and is it provided in the Enclosure. In the ITS conversion, TVA referred to several BWROG Licensing Topical Reports used in justifying the STIs and AOTs for particular instrument systems. These were:

D030

1. NEDC-30851P-A, Technical Specification Improvement Analysis for Reactor Protection System; NEDC-30851P-A Supplement 1, Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation; and NEDC-30851P-A Supplement 2, Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation.

2. GENE-770-06-1, Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications

3. GENE-770-06-2, Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications

4. NEDC-31677P-A, Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation

5. NEDC-30936P-A, Technical Specification Improvement Methodology, Parts 1 and 2 (With Demonstration for BWR ECCS Actuation Instrumentation).

TVA previously performed a review of the methodology and determined it was applicable to Unit 3 based on its configuration. For Unit 1, TVA performed a similarity analysis between the Unit 3 and Unit 1 instrumentation systems and concluded that the configuration of the two units is functionally the same and thus the evaluation was likewise applicable to Unit 1. Note that the setpoint methodology used for Unit 1 is the same as that used for Units 2 and 3. Therefore, TVA has concluded that the BWROG generic analyses are applicable to BFN Unit 1 and the STI and AOT extensions in the current Unit 1 TS are acceptable.

There are no new commitments contained in this letter. If you have further questions on this submittal, please contact me at (205) 729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 1st day of November, 2006.

Sincerely,

William D. Crouch

William D. Crouch
Manager of Licensing and
Industry Affairs

References:

1. TVA Letter dated September 7, 2006, BFN - Unit 1 - Response to NRC Letter, Dated July 7, 2006 - Review of Pending License Amendment Request (TAC No. MC4797)(TS-432)

2. NRC letter dated July 7, 2006, Review of Pending License Amendment Request (TAC No. MC4797)(TS-432)

3. TVA Letter dated December 11, 1997, BFN - Units 1, 2, and 3 TS-362 - ITS - Supplemental Information - Changes to Instrumentation STIs and AOTs.

Enclosures
cc: See page 5

Enclosure
cc (Enclosure):

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ENCLOSURE

BROWNS FERRY NUCLEAR PLANT

Unit 1

Comparison to

Unit 3

Support for Applicability Analysis
BWR Owner's Group
Technical Specification Improvement Analysis

BROWNS FERRY NUCLEAR PLANT

Unit 1 W77 061030 001

Comparison to

Unit 3

Support for Applicability Analysis
BWR Owner's Group
Technical Specification Improvement Analysis

Preparer	Tommy R Trudell For **George A. Washburn** per Telecon 10/30/06	**Date 10/30/06**
Checker	Richard B Sparks **Richard B. Sparks**	**Date 10/30/06**

BROWNS FERRY NUCLEAR PLANT
Unit 1 Comparison to Unit 3

History

The generic study performed by GE provides the technical basis to modify the surveillance test frequencies and allowable out-of-service time of the RPS/ECCS/Rod Block/PCIS/ATWS&EOC RPT from the Generic Tech Specs. This generic study was extended by GE to apply to BFN Unit 2 based upon their analysis of the differences between Unit 2 and the generic plant previously evaluated.

Purpose

The purpose of this analysis is to compare the configuration of BFN Unit 1 to the present configuration of BFN Unit 3 that was evaluated by TVA against the Unit 2 evaluation of the generic plant configuration, with the intent to conclude that the units are identical and therefore the previously issued TVA analysis of Unit 3 can be directly applied to Unit 1. The following pages are the comparison documented in the same format as the TVA analysis for Unit 3.

Method

The method of analysis consisted of the following steps:

- Review TVA Unit 3 analysis
- Pull Unit 1 and 3 drawings and documented references for affected systems.
- Compare Unit 1 configuration against current Unit 3 configuration
- Document and justify differences.
- Develop conclusion.

Conclusion

Based upon the comparison of Unit 3 to Unit 1 as documented on the following pages it is concluded that the configuration of the two units is functionally the same and that the TVA analysis of Unit 3 is directly applicable to Unit 1.

References

- Various Drawings and documents as documented in the following sections.
- Unit 1 Technical Specification Change requests, TS-430, TS-433, TS-434, TS-436, TS-437, TS-438, TS-443, TS-447.
- Units 1, 2, 3 ITS Conversion request TS-362.
- Units 1, 2, 3 Technical Specification Change request TS-424
- Unit 3 Technical Specifications through Amendment 253.
- Unit 1 Technical Specifications through Amendment 255.

BROWNS FERRY NUCLEAR PLANT UNIT 1
INSTRUMENTATION DATA

- ECCS INSTRUMENTATION
- RPV LEVEL 8 FEEDWATER PUMP/MAIN TURBINE TRIP
- CREV INITIATION INSTRUMENTATION
- END-OF-CYCLE RPT & ATWS RPT INSTRUMENTATION
- ROD BLOCK INSTRUMENTATION
- CONTAINMENT ISOLATION INSTRUMENTATION
- RPS INSTRUMENTATION

ECCS EVALUATION CHECKLIST FOR

BROWNS FERRY NUCLEAR PLANT – Unit 1

Comparison to Unit 3

Support for applicability analysis
BWR Owners' Group
Technical Specification Improvement Analyses
for
ECCS Actuation Instrumentation

The ECCS configuration for UNIT 1 was compared to the current configuration of UNIT 3. The comparison is as documented as follows.

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section I – ECCS Plant Specific Data Sources

Source Number		
1.	Main Steam (ADS) Flow Diagram:	1,3-47E801-1
2.	RHR Flow Diagram:	1,3-47E811-1
3.	HPCI Flow Diagram:	1,3-47E812-1
4.	RCIC Flow Diagram	1,3-47E813-1
5.	Core Spray Flow Diagram:	1,3-47E814-1
6.	EECW Flow Diagram:	1,2,3-47E859-1
7.	EECW System Design Criteria:	BFN-50-7067
8.	RCIC System Design Criteria:	BFN-50-7071
9.	HPCI System Design Criteria:	BFN-50-7073
10.	RHR System Design Criteria:	BFN-50-7074
11.	Core Spray System Design Criteria:	BFN-50-7075
12.	Technical Specifications, BFNP Unit 1 through Amendment 257.	
13	NEDC-30936P-A, Part I, Technical Specification Improvement Methodology (With Demonstration For BWR ECCS Actuation Instrumentation), December 1988.	
14.	Browns Ferry FSAR: Section 4.4	
15.	Browns Ferry FSAR: Chapter 8	
16.	1-SR-3.5.1.3.5(CS I), (CS I), (RHR I), (RHR II), RHR & CS SYSTEM MOV OPERABILITY LOOP II	
17.	Technical Specifications, BFNP Unit 3 through Amendment 254	

Section II – ECCS Configuration Data

A. ECCS System

		BFN-1(3)	Difference U3 Vs U1 (Yes/No)	Data Source[1]
1.	Number of:			
	Core Spray Pumps/Loops	4/2(4/2)	No	5
	LPCI Pumps	4(4)	No	2
	ADS Valves	6(6)	No	1
	HPCI Pumps	1(1)	No	3
2.	Needed for Success, Number of:			
	Core Spray Pumps/Loops	2/1(2/1)	No	13
	LPCI Pumps	1(1)	No	13
	ADS Valves	2(2)	No	13
3.	Number of:			
	Diesel Generators	4[2](4)	No	15
	DC Power Divisions	2(2)	No	15
	AC Power Divisions	2[3,4](2[3,5])	No	15
4.	IC or RCIC	RCIC(RCIC)	No	8
	Loop Selection Logic	No(No)	No	

[1]The numbers shown in the Data Source column refer to the documents listed in Section I.

[2]The Standby AC System for Units 1/2 consists of four diesel generators. The Standby AC system for Unit 3 is separate and consists of four diesel generators that supply power to the ECCS equipment for Unit 3. Some of the safety-related support equipment required for operation of Unit 3 (e.g., EECW pumps, CREVS) is powered from the four Unit 1/2 diesel generators.

[3]There are two 4KV Shutdown Boards in AC Power Division I and two 4KV Shutdown Boards in AC Power Division II.

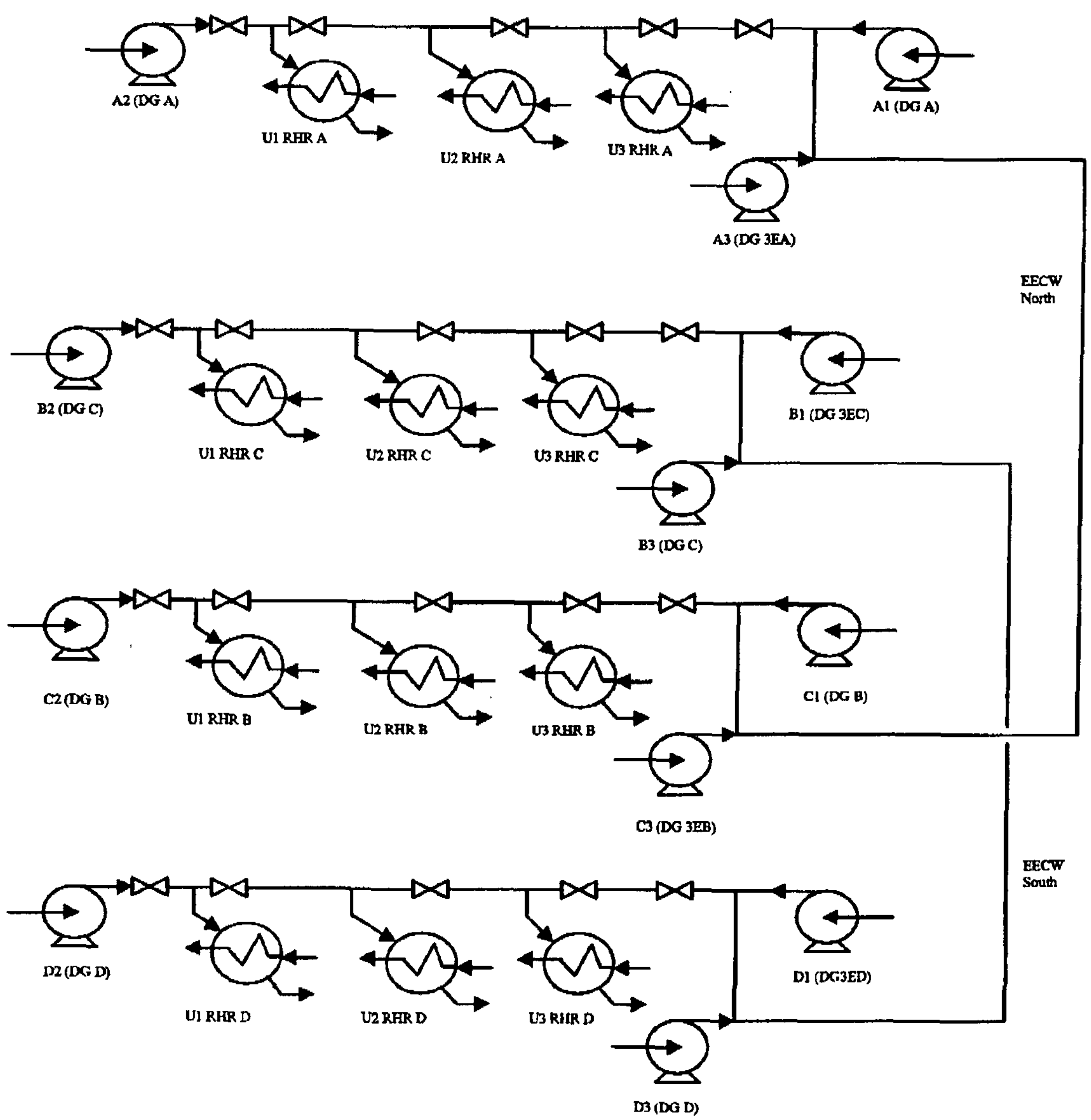
[4]Each of the four Units 1/2 4KV shutdown Boards is electrically separated and powered from a separate diesel generator with a Shutdown Board of each division dedicated to each unit.

[5]Each of the four Unit 3 4KV Shutdown Boards is electrically separated and powered from a separate diesel generator.

Section II – ECCS Configuration Data

B. Support System Dependencies

The figure below shows the arrangement of the EECW/RHRSW pumps along with the support Diesel Generators needed.

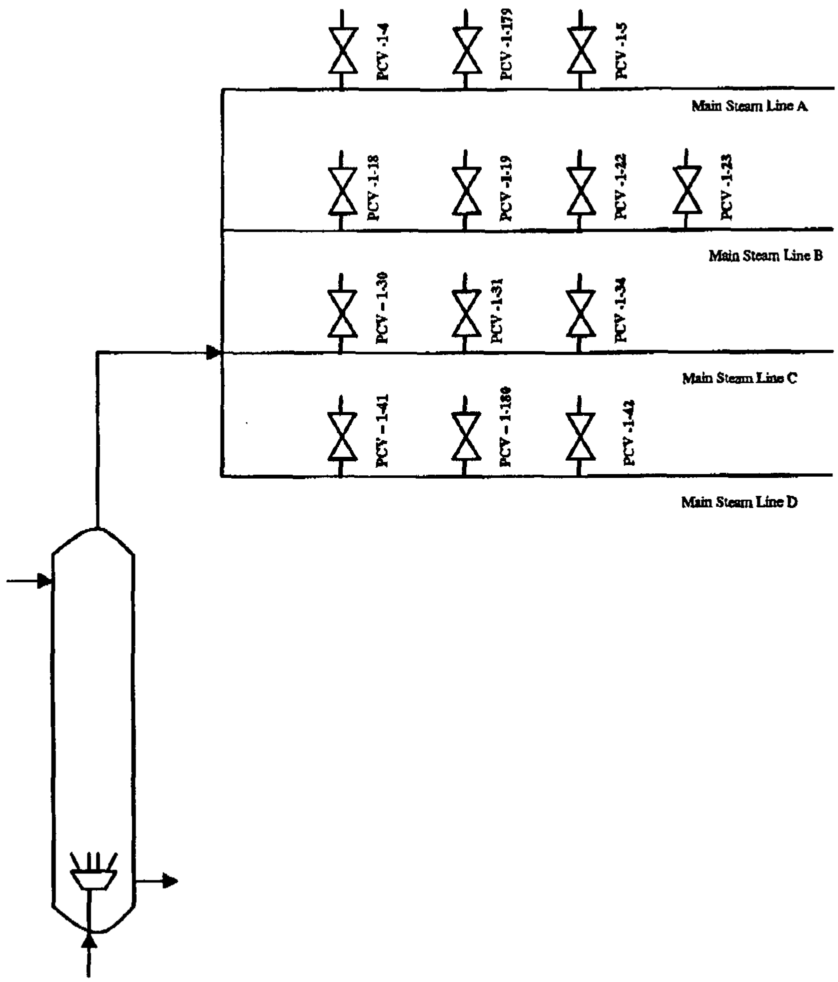


The above system is shared between Units 1/2 and Unit 3 already and therefore was evaluated under the Unit 3 analysis.

Section II – ECCS Configuration Data

B. Support System Dependencies

The Automatic Depressurization system is arranged as shown below.



The following table documents the ADS valve dependencies on power (logic not shown).

	II	I	I
	Batt 1	Batt 3	Batt 2
	250 V RMOV 1A	250 V RMOV 1B	250 V RMOV 1C
1-34			1M, 2M
1-31		1M, 2M	
1-30	1A	2A	
1-22	1A	2A	
1-19		1M, 2M	
1-5			1M, 2M

1M is primary power and 2M is secondary, with manual transfer from 1 to 2 and back.
1A is primary power and 2A is secondary with automatic transfer from 1A to 2A and back.

Section II – ECCS Configuration Data

B. Support System Dependencies

The table below represents the dependencies of the diesel generator/AC power system upon the 250 VDC power system.

	Batt Bd 1	Batt Bd 2	Batt Bd 3	SB A	SB B	SB C	SB D	4KV Shtdn Bd A	4KV Shtdn Bd B	4KV Shtdn Bd C	480V Shtdn Bd 1A	480V Shtdn Bd 1B
4KV Shtdn Bd A		2		1								
4KV Shtdn Bd B		2			1							
4KV Shtdn Bd C	2					1						
4KV Shtdn Bd D			2				1					
480V Shtdn Bd 1A				S/D				D	A			
480V Shtdn Bd 1B						S/D			A	D		
RMOV Bd 1A	S							D			D	
RMOV Bd 1B			S							D		D
RMOV Bd 1C		S								D		D

The following are the explanations of the annotations. The A is 480VAC alternate feed, D is 480VAC direct feed, S/D represents that a support supply is both a secondary support and direct (e.g., 480 V Shtdn Bd 1A is fed from 4KV Shtdn Bd A which has Batt 1 as its support therefore SB-A is S for the Shtdn Bd 1A, 1 is primary, 2 is alternate (manual transfer).

Section II – ECCS Configuration Data

B. Support System Dependencies

	LPCI RHR PUMPS				CORE SPRAY		ADS		RCIC	HPCI	DIESELS			
	LPCI I		LPCI II		CS I	CS II								
	A	C	B	D	A/C	B/D	A	B			A	B	C	D
Off Site Power	X	X	X	X	X	X			X	X				
On Site AC Power														
Div I (SD Bd A)	X		1		A									
Div I (SD Bd B)		X			C									
Div II (SD Bd C)	1		X			B								
Div II (SD Bd D)				X		D								
On Site DC Power														
Div I	X	X			X		X	X	X	X				
Div II			X	X		X		2	X	X				
Service Water[1]														
EECW North (pumps A1 & C1)	X	X	X	X	X	X					(SD BD A)	(SD BD B)		
EECW South (pumps B1 & D1)	X	X	X	X	X	X							(3EC)	(3ED)

[1]The LPCI injection valve for loop I is normally powered from DG 3EA. The LPCI injection valve for loop II is normally powered from DG 3EC. The 1 represents the secondary support DG automatically transferred to power the valve on loss of primary power.

[2]Two of the six ADS valves are normally powered from a Division II power source. However, these valves will automatically transfer to a Division I power source if the Division II power source fails. Should Division I fail the two Division II ADS valves will be available.

[3]The U1 and U3 systems utilize the same type relays between the units. (CS/RHR/ADS/RCIC/HPCI all use HFA/HGA/CR120/CR2820/Agastat)

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[1]The following alternate EECW pumps can be used if the primary fails. A1 alternate for A3, B1 for B3, C1 for C3, D1 for D3. See previous figure that shows diesel dependency of various pumps.

Section II – ECCS Configuration Data

C. Support System Dependencies

	LPCI RHR PUMPS				CORE SPRAY		ADS		RCIC	HPCI	DIESELS			
	LPCI I		LPCI II		CS I	CS II								
	A	C	B	D	A/C	B/D	A	B			A	B	C	D
Water Supply														
Suppression Pool	X	X	X	X	X	X			X	X				
Condensate Storage Tank[2]									X	X				
River											X	X	X	X
Air														
Drywell Control Air[3]							X	X						
Control Air: Div 1														
Control Air: Div 2														
Containment Instrument: Div 1														
Containment Instrument: Div 2														
Room Cooling[4]														
LPCI	X	X	X	X										
CS					X	X								
RCIC									X					
HPCI										X				
Diesels											X	X	X	X

[2]The RHR (LPCI) and Core Spray can be manually aligned to take suction from the Condensate Storage Tank by repositioning hand-operated valves in the torus room.
[3]The ADS valves have a backup tie to CAD and have accumulators sized for 5 openings.
[4]Room cooling also requires EECW to be functional (justified elsewhere).

Section II – ECCS Configuration Data

C. Support System Instrumentation Dependencies

	LPCI RHR PUMPS				CORE SPRAY		ADS		RCIC	HPCI	DIESELS			
	LPCI I		LPCI II		CS I	CS II								
	A	C	B	D	A/C	B/D	A	B			A	B	C	D
RPV Water Level 1 (Low Low Low)														
LS-3-58A	X	X	X	X	X	X	X							
LS-3-58B	X	X	X	X	X	X	X							
LS-3-58C	X	X	X	X	X	X		X						
LS-3-58D	X	X	X	X	X	X		X						
RPV Water Level 2 (Low Low)														
LIS-3-58A, C									X	X				
LIS-3-58B, D									X	X				
RPV Water Level 2 (Low)														
LIS-3-184							X							
LIS-3-185								X						
RPV Water Level 8 (High)														
LIS-3-208A, C									X					
LIS-3-208B, D										X				

Section II – ECCS Configuration Data

C. Support System Instrumentation Dependencies

	LPCI RHR PUMPS				CORE SPRAY		ADS		RCIC	HPCI	DIESELS			
	LPCI I		LPCI II		CS I	CS II								
	A	C	B	D	A/C	B/D	A	B			A	B	C	D
Drywell Pressure High														
PIS-64-57B, D							X							
PIS-64-57A, C								X						
PS-64-58A	X	X	X	X	X	X				X				
PS-64-58B	X	X	X	X	X	X				X				
PS-64-58C	X	X	X	X	X	X				X				
PS-64-58D	X	X	X	X	X	X				X				
RPV Pressure Low														
PIS-3-74A	X	X	X	X	X									
PIS-3-74B	X	X	X	X		X								
PIS-68-95	X	X	X	X	X									
PIS-68-96	X	X	X	X		X								
Injection Valve DP														
The instruments noted above are used in same manner as they are used in Unit 3.														

Section II – ECCS Configuration Data

C. Support System Instrumentation Dependencies

	LPCI RHR PUMPS				CORE SPRAY		ADS		RCIC	HPCI	DIESELS			
	LPCI I		LPCI II		CS I	CS II								
	A	C	B	D	A/C	B/D	A	B			A	B	C	D
LPCI Pump Discharge Pressure High[5]														
PS-74-8A&B							X	X						
PS-74-19A&B							X	X						
PS-74-31A&B							X	X						
PS-74-42A&B							X	X						
CS Pump Discharge Pressure High[6]														
PS-75-7							X	X						
PS-75-16							X	X						
PS-75-35							X	X						
PS-75-44							X	X						
ADS Drywell Pressure Bypass Timer[7]														
2-1-58A2							X							
2-1-58B2							X							
2-1-58C2								X						
2-1-58D2								X						

[5] 3-730E938-6, 9
[6] 0-730E930-24, 23
[7] 3-730E929-2

Section II – ECCS Configuration Data

C. <u>Support System Instrumentation Dependencies</u>

	LPCI RHR PUMPS				CORE SPRAY		ADS		RCIC	HPCI	DIESELS			
	LPCI I		LPCI II		CS I	CS II								
	A	C	B	D	A/C	B/D	A	B			A	B	C	D
ADS Timer[8]														
2E-K34&35							X	X						
Manual Initiation Switch														
LPCI Pump Discharge Flow Low														
FS-74-50	X		X											
FS-74-64		X		X										
Core Spray System Discharge Flow Low														
FS-75-21					X									
FS-75-49						X								
CST Level Low[9]														
LS-73-56A, B[10]										X				
Suppression Pool (Torus) Level High														
LS-73-57A, B[8]										X				

[8] 3-730E929-2
[9] The BFN-1 RCIC System Does <u>NOT</u> have automatic transfer of the pump suction path.
[10] 56 OR 57 LOOPS Transfer

Section II – ECCS Configuration Data

C. <u>Support System Instrumentation Dependencies</u>

	LPCI RHR PUMPS				CORE SPRAY		ADS		RCIC	HPCI	DIESELS			
	LPCI I		LPCI II		CS I	CS II								
	A	C	B	D	A/C	B/D	A	B			A	B	C	D
ADS Inhibit Switch														
XS-1-159A							X							
XS-1-161A								X						
HPCI Pump Discharge Flow Low														
FIS-73-33										X				
HPCI Turbine Exhaust Pressure High														
PS-73-22A&B										X				
HPCI Pump Suction Pressure Low														
PIS-73-29-1										X				
HPCI Turbine Exhaust Rupture Disc Pressure High														
PS-73-20A,B,C,D										X				
High Steam Supply Pressure Low														
PS-73-1A,B,C,D										X				

Section II – ECCS Configuration Data

D. Support System & Instrumentation Dependencies

Summary

A review of plant drawings and documents has concluded that the Unit 1 systems are functionally identical to the Unit 3 systems. There is no difference in the UFSAR described design functions.

Section II – ECCS Configuration Data

D. <u>Minimum Number of Sensors, Channels, or Components for Failure, BFN-1.</u>

A = MINIMUM NUMBER OF SENSOR FAILURES REQUIRED TO FAIL TRIP FUNCTION
B = MINIMUM NUMBER OF SENSOR FAILURES REQUIRED TO FAIL FUNCTION – TOTAL
C = MINIMUM NUMBER OF SENSOR TYPES REQUIRED TO FAIL FUNCTION

		Unit 1		Different From Unit 3	
Trip Function	**A**	**B**	**C**	**B**	**C**
CS Pump Initiation[1]	(2 Drywell Pressure OR 2 Low RPV Pressure) **AND** 2 RPV Level 1 (LOLOLO)	4	2	No	No
CS Injection Valve[2]	2 RPV Low Pressure	2	1	No	No
LPCI Pump Initiation[1]	(2 Drywell Pressure OR 2 Low RPV Pressure) **AND** 2 RPV Level 1 (LOLOLO)	4	2	No	No
LPCI Injection Valve[3]	2 RPV Low Pressure	2	1	No	No
ADS Initiation	2 RPV Level 1 (LOLOLO)[4] **OR** 2 RPV Level 3 (LOW)[5] **OR** 1 RPV Level 1 AND 1 RPV Level 3[6] **OR** 2 Drywell Press High AND 2 Drywell Press Bypass Timers	2	1	No	No
ADS Time Delay	2 timers	2	1	No	No
HPCI Level 8[8]	2 RPV Level 8 (High)	2	1	No	No
HPCI Initiation[7]	2 Drywell Pressure (high) **AND** 2 RPV Level 2 (LOLO)	4	2	No	No
HPCI Injection Valve	2 Drywell Pressure (high) **AND** 2 RPV Level 2 (LOLO)	4	2	No	No
HPCI Water Source[7]	2 Condensate Header Level	2	1	No	No
RCIC Initiation[9]	2 RPV Level 2 (LOLO)	2	1	No	No
RCIC Level 8[8]	2 RPV Level 8 (High)	2	1	No	No
RCIC Injection Valve[9]	2 RPV Level 2 (LOLO)	2	1	No	No

[1] 1 of 2 twice level OR (1 of 2 twice hi DWP and 1 of 2 twice Lo RVP).
[2] 1 of 2 for bus A, 1 of 2 for bus B.
[3] 1 of 2 twice RPV low pressure.
[4] (one on bus A & one on bus B).
[5] (one on bus A & one on bus B).
[6] (one on bus A & one on bus B).
[7] 1 of 2 twice level or 1 of 2 twice high DW pressure.
[8] 2 of 2 to trip.
[9] 1 of 2 twice on RPV level

Section II – ECCS Configuration Data

E. Surveillance Requirements, ECCS Instrumentation and Related Subsystems[1]

	BFN-1	BFN-3[4]	DIFFERENCE (Yes/No)
CORE SPRAY SYSTEM			
REACTOR WATER LEVEL 1 (LOW-LOW-LOW)	Q	Q	No
DRYWELL PRESSURE HIGH	Q	Q	No
REACTOR PRESSURE LOW	Q	Q	No
MANUAL INITIATION	N/A	N/A	No
LPCI			
REACTOR WATER LEVEL 1 (LOW-LOW-LOW)	Q	Q	No
DRYWELL PRESSURE HIGH	Q	Q	No
REACTOR PRESSURE LOW	Q	Q	No
INJECTION VALVE DIFFERENTIAL PRESSURE LOW	N/A	N/A	No
MANUAL INITIATION	N/A	N/A	No
HPCI			
REACTOR WATER LEVEL 2 (LOW-LOW)	Q	Q	No
DRYWELL PRESSURE HIGH	Q	Q	No
CST LEVEL LOW	Q	Q	No
SUPPRESSION POOL LEVEL HIGH	Q	Q	No
REACTOR WATER LEVEL 8 (High)	Q	Q	No
MANUAL INITIATION	N/A	N/A	No
ADS			
REACTOR WATER LEVEL 1 (LOW-LOW-LOW)	Q	Q	No
DRYWELL PRESSURE HIGH	Q	Q	No
ADS DRYWELL PRESSURE BYPASS TIMER	24M	24M	No
ADS TIMER	24M	24M	No
CORE SPRAY PUMP DISCHARGE PRESSURE	Q	Q	No
LPCI PUMP DISCHARGE PRESSURE	Q	Q	No
REACTOR WATER LEVEL (LOW)	Q	Q	No
MANUAL INITIATION	N/A	N/A	No
INJECTION VALVE STROKE TEST	Q	Q(ref21)	No
DIESEL GENERATOR	M	M	No
ELECTRIC POWER[3]			
ESSENTIAL AC	W	W	No
ESSENTIAL DC	W	W	No
ESSENTIAL AC Buses	W	W	No

	BFN-1	BFN-3[4]	DIFFERENCE (Yes/No)
RCIC			
REACTOR WATER LEVEL 2 (LOW LOW)	Q	Q	No
REACTOR WATER LEVEL 8 (HIGH)	Q	Q	No
CST LEVEL LOW	N/A	N/A	No
SUPPRESSION POOL LEVEL HIGH	N/A	N/A	No
MANUAL INITIATION	N/A	N/A	No

[1]Based on Technical Specifications (Reference 12, 17)
[2]M = Monthly, W = Weekly, R = Refueling, Q = Quarterly
[3]1-SR-3.3.8.1.3(3EC).a covers the weekly Essential DC. SI-2 (TS section 4.9.A.4.d) does a verification of 4KV Shutdown Board voltages every 12 hours which could be construed to mean Essential AC and Essential AC Buses are verified more than once per week.
[4]If Unit 3 took no credit for a function then Unit 1 also will not (N/A).

RFPT & Main Turbine RPV High Water Level Trip Instrumentation

EVALUATION CHECKLIST FOR

BROWNS FERRY NUCLEAR PLANT – Unit 1

Comparison to Unit 3

Support for applicability analysis
BWR Owners' Group
Technical Specification Improvement Analyses
for
RFPT & Main Turbine RPV High Water Level Trip Instrumentation

The RFPT & Main Turbine RPV High Water Level Trip Instrumentation configuration for UNIT 1 was compared to the configuration of UNIT 3 as evaluated by drawing and document review. The comparison is as documented as follows.

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section I – RFPT & Main Turbine RPV High Water Level Trip Instrumentation Plant Specific Data Sources

Source
Number

1.	RFPT Schematic:	1,3-45E612-1
2.	Turbo Generator Auxiliaries Schematic:	1,3-45E602-1
3.	Feedwater Control System	1,3-729E895
4.	ECCS Div II ATU Schematic	1-45E670/1 series 3-45E670-36 , -30
5.	Technical Specification, BFNP Unit 1	Amendment 257
6.	Technical Specifications, BFNP Unit 3	Amendment 254
7.	Unit 3 data as documented in TS-362	

Section II – ECCS Configuration Data

A. RFPT & Main Turbine RPV High Water Level Trip Instrumentation System

The instrumentation listed in the table below initiates a signal to trip the three reactor feedwater pump turbines (RFPTs) and the main turbine from redundant trip channels A or B. Two out of Two logic in either trip channel will initiate a RFPT and Main Turbine Trip. Trip Channel A is initiated by coincident actuation of instrument loops A and C. Trip Channel B is initiated by coincident actuation of instrument loops B and D.

Table 1

RFPT & Main Turbine High Water Level Trip Instrumentation

	Function	Instrument Number	Function Test Frequency	Cal Frequency	Common to ECCS	Common to RPS
Unit 3	RPV Level 8 RFPT & Main Turbine Trip	LS-3-208A LS-3-208B LS-3-208C LS-3-208D	92 Days	24 Months	Yes[1]	No
Unit 1	RPV Level 8 RFPT & Main Turbine Trip	LS-3-208A LS-3-208B LS-3-208C LS-3-208D	92 Days	24 Months	Yes[2]	No

Conclusion:

As can be seen above the Unit 1 configuration is identical to the Unit 3 configuration previously analyzed and determined to be adequate.

[1]Instrument loops B and D are common to ECCS (e.g. HPCI high level trip). A and C provide RCIC high level trip.
[2]Instrument loops B and D are common to ECCS (e.g. HPCI high level trip). A and C provide RCIC high level trip.

Control Room Emergency Ventilation/Isolation Instrumentation

EVALUATION CHECKLIST FOR

BROWNS FERRY NUCLEAR PLANT – UNIT 1

Comparison to Unit 3

Support for applicability analysis
BWR Owners' Group
Technical Specification Improvement Analyses
for
Control Room Emergency Ventilation/Isolation Instrumentation

The Control Room Emergency Ventilation/Isolation Instrumentation configuration for UNIT 1 was compared to the configuration of UNIT 3 as evaluated by drawing and document review. The comparison is as documented as follows. The CREVs is a shared system previously evaluated for Unit 3; therefore, it equally applies to Unit 1.

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section I – Control Room Emergency Ventilation/Isolation Instrumentation Plant Specific Data Sources

Source
Number

1.	Control Bay Emergency Isolation Schematic:	0-45E769-11
2.	Primary Containment Isolation System:	1,3-730E927-18
3.	CREVS Wiring Diagrams	0-55E725 0000
4.	CREVS Wiring Diagrams	0-92D510-09 series
5.	Technical Specifications, BFNP Unit 1	Amendment 257
6.	BFNP Control Bay and Reactor Building Board Rooms Environmental Control System Design Criteria:	BFN-50-7030A
7.	Unit 3 data as documented in TS-362	
8.	Technical Specifications, BFNP Unit 3	Amendment 254

Section II – Control Room Emergency Ventilation/Isolation Instrumentation Configuration Data

A. Control Room Emergency Ventilation/Isolation Instrumentation System

The instrumentation listed in the table below initiates isolation of the Control Room and initiates the Control Room Emergency Ventilation (CREV) system due to high radiation in the Control Room air intake. A PCIS Group 6 isolation signal will also provide a Control Room isolation and CREVs initiation. One of two redundant control room emergency pressurization units will be initiated from the radiation monitors or PCIS group 6 isolation signal. The redundant control room pressurization unit will start upon failure or loss of the first unit. It should be noted that this is not a Unit 3 versus Unit 1 comparison. The two CREVS previously analyzed are shared by both Units.

Table 1

Control Room Emergency Ventilation/Isolation Instrumentation

Function	Instrument Number	Function Test Frequency	Cal Frequency	Common to ECCS	Common to RPS
Control Room Air Supply Radiation Monitors	RM-90-259A RM-90-259B	92 Days	92 Days	No	No

Conclusion:

It should be noted that this is not a Unit 3 versus Unit 1 comparison. The two CREVS previously analyzed are shared by both Units.

EOC RPT & ATWS RPT

EVALUATION CHECKLIST FOR

BROWNS FERRY NUCLEAR PLANT – UNIT 1

Comparison to Unit 3

Support for applicability analysis
BWR Owners' Group
Technical Specification Improvement Analyses
for
EOC RPT & ATWS RPT Instrumentation

The EOC RPT & ATWS RPT Instrumentation configuration for UNIT 1 was compared to the configuration of UNIT 3 as evaluated by drawing and document review. The comparison is as documented as follows:

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section I – EOC RPT & ATWS RPT Instrumentation Plant Specific Data Sources

Source
Number

1.	Reactor Recirculation System:	1,3-45E611-68-5
2.	Recirculation Pump Trip Schematic Diagram:	1,3-45E763-11 R003, -12
3.	BFN Reactor Water Recirculation System Design Criteria:	BFN-50-7068
4.	Technical Specifications, BFNP Unit 1	Amendment 257
5.	Unit 3 data as documented in TS-362	
6.	Technical Specifications, BFNP Unit 3	Amendment 254

Section II – EOC RPT & ATWS RPT Instrumentation Configuration Data

A. EOC RPT Instrumentation System

There are two divisions of EOC RPT logic. Each division receives signals indicating turbine stop valve closures (two-out-of-two logic) or turbine control valve fast closure (two-out-of-two logic) from the reactor protection system (RPS) to trip the recirculation pump motor breakers. Either of these signals will trip both recirculation pumps. Signals indicating first stage turbine pressure greater than 30% are provided from RPS to permit the recirculation pump trip.

Table 1

EOC RPT Instrumentation

Function	Instrument Number	Function Test Frequency[3]	Cal Frequency	Common to ECCS	Common to RPS
Turbine Control Valve Closure	(PS-47-142 AND PS-47-144) **OR** (PS-47-146 AND PS-47-148)	92 Days	24 M	No	Yes
	Above **OR** Below				
Turbine Stop Valve Closure	(ZS-1-74F AND ZS-1-78F) **OR** (ZS-1-84F AND ZS-1-88F)	92 Days	24 M	No	Yes

Allowed Out-of-Service Times

Allowed out-of-service times for EOC RPT instrumentation are addressed in the Technical Specification requirements. Inoperable instrument channels must be restored to operate status within 4 hours[4]s or additional actions must be taken (e.g. insert control rods or reduce power). The TS for Unit 1 states that one channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition. The Unit 3 TS also states 6 hours.

[3]The RPT breakers are tested once per operating cycle.
[4]TS 3.3.4.1.C

B. ATWS RPT Instrumentation System

There are two divisions of ATWS RPT logic. Each division receives signals indicating low reactor water level (two out of two logic) or high reactor pressure (two-out-of-two logic) to trip the recirculation pump motor breakers. Either of these signals will trip both recirculation pumps. These signals are independent of the EOC RPT trip signals transmitted from RPS.

Table 1

ATWS RPT Instrumentation

Function	Instrument Number	Function Test Frequency[5]	Cal Frequency	Common to ECCS	Common to RPS
RPV Water Level 2 (lo-lo)	(LS-3-58A1 AND LS-3-58B1) **OR** (LS-3-58C1 AND LS-3-58D1)	92 Days	24 M	Yes	No
	Above **OR** Below				
RPV Pressure High	(PS-3-204A AND PS-3-204B) **OR** (PS-3-204C AND PS-3-204D)	92 Days	24 M	Yes	No

Allowed Out-of-Service Times

Current Technical Specifications allow one channel of ATWS RPT instrumentation in only one trip system to be placed in an inoperable status for up to 6 hours for required surveillance provided the other channels in that trip system are operable.

If a channel is found to be inoperable or if the surveillance/maintenance/calibration period for one channel exceeds 6 consecutive hours, the trip system will be declared inoperable or the channel will be placed in a tripped condition.

C. Conclusions

Review of the Unit 3 information (reference 5) shows that the Unit 1 data is identical.

[5] The breaker is tested once per operating cycle.

Rod Block

EVALUATION CHECKLIST FOR

BROWNS FERRY NUCLEAR PLANT – UNIT 1

Comparison to Unit 3

Support for applicability analysis
BWR Owners' Group
Technical Specification Improvement Analyses
for
Rod Block

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section I – Rod Block Instrumentation Plant Specific Data Sources

Source Number		
1.	Reactor Protection System Schematic:	1,3-730E915-series
2.	Reactor Manual Controls Schematic	1,3-730E321-series
3.	Start-up Range Neutron Mon Sys Schematic	1,3-730E237-series
4.	Scram Discharge Volume Instruments	1,3-47E610-85-5
5.	GE Analysis of ROD Block	GENE-A31-0002-03
6.	Technical Specifications, BFNP Unit 1	Amendment 257
7.	Unit 3 data as documented in TS-362	
8.	Technical Specifications, BFNP Unit 3	Amendment 254

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Browns Ferry Nuclear Plant – Unit 1

Section II – Rod Block Instrumentation Configuration Data

A. Rod Block Instrumentation System

The instrumentation used for Rod block signal generation are detailed in the table below. It should be noted that the equipment and configuration is the same as Unit 3.

Function	Functional Test Frequency	Calibration Frequency	Common to ECCS	Common to RPS
BFN Unit 1 Rod Block Instrumentation				
APRM[6]			NO	YES
▪ upscale (flow biased)	184 Days	24M		
▪ upscale (startup)[7]	184 Days	24M		
▪ downscale[8]	184 Days	24M		
▪ inoperative	184 Days	N/A		
RBM			NO	NO
▪ upscale (flow biased)	184 Days	24M		
▪ downscale[9]	184 Days	24M		
▪ inoperative[15]	184 Days	N/A		
IRM[10]			NO	YES
▪ upscale	7 Days	92 Days		
▪ downscale[11, 12]	7 Days	92 Days		
▪ detector not in start position[13,14]	7 Days	24M		
▪ inoperative[15]	7 Days	N/A		
SRM[16]			NO	YES
▪ upscale	7 Days	92 Days		
▪ downscale[17]	7 Days	92 Days		
▪ detector not in start position	7 Days	24M		
▪ inoperative	7 Days	N/A		
Recirc Flow Bias Comparator	M	24M	NO	YES
Recirc Flow Bias Upscale	M	24M	NO	YES
West SDV High Water Level LS-85-45L	92 Days	24M	NO	NO

[6]The functional test frequency and calibration frequency specified for the APRMs and RBMs should be revised as part of the PRNM Upgrade Modification for Unit 1.
[7,10,16]Bypassed when Mode Switch placed in RUN.
[8,9]Active when mode switch is in RUN, Bypassed when IRMs are operable but not high.
[11,13]IRM downscale is bypassed on its lowest range.
[12]All SRM rod Block functions are bypassed when all the IRMs are on range 8 or above.
[14,15]All SRM rod Block functions are bypassed when all the IRMs are on range 8 or above.
[17]SRM Downscale functions are bypassed when IRMs are above range 2. SRM detector not in startup position is bypassed when the count rate is >= 100 CPS or the condition above is satisfied.

BFN Unit 1 Rod Block Instrumentation				
Function	**Functional Test Frequency**	**Calibration Frequency**	**Common to ECCS**	**Common to RPS**
East SDV High Water Level			NO	NO
LS-85-45M	92 Days	24M		
SDV High Water Level Scram Bypass	N/A	N/A	NO	YES
Reactor Mode Switch HS-99-5A-S1	24 M	N/A	NO	YES

Scram Discharge Volume High Water Level Rod Block

There is one level instrument in each scram discharge volume which provides a rod block signal. Each level instrument provides a rod block signal to one of the two rod block channels.

ROD Block Logic

The Rod Block logic for BFN Unit 1 is the same as for Unit 3 based upon drawing review.

Allowed Outage Times for Rod Block Instrumentation

Current Technical Specifications specify that during repair or calibration of equipment, not more than one SRM or RBM channel nor more than two APRM or IRM channels may be bypassed. Bypassed channels are not counted as operable channels to meet the minimum operable channel requirements specified in the TS Table 3.3.2.1.

Allowed outage time is specified as once per outage in the current Technical Specifications Basis for calibration and functional testing of the Recirculation Flow Bias Comparator or Flow Bias Upscale instrumentation. With the number of operable channels less than required by the minimum operable channels per trip function requirement, at least one inoperable channel must be placed in the tripped condition within one hour.

No specific allowed outage time is provided in the current Technical Specifications for calibration and functional testing of the SDV High Water Level Rod Block instrumentation. If this function is not operable at a time when operability is required, the channel must be tripped or administrative controls must be immediately imposed to prevent control rod withdrawal.

CONCLUSION

The Unit 1 Rod block system is the same as the evaluated Unit 3 system.

Containment Isolation Instrumentation Data

EVALUATION CHECKLIST FOR

BROWNS FERRY NUCLEAR PLANT – UNIT 1

Comparison to Unit 3

Support for applicability analysis
BWR Owners' Group
Technical Specification Improvement Analyses
for
Containment Isolation Instrumentation

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section I – Containment Isolation Instrumentation Plant Specific Data Sources

Source Number		
1.	PCIS Elementary Diagrams	1,3-730E927-Series
2.	Technical Specifications, BFNP Unit 1	Amendment 257
3.	GE Analysis of PCIS	GENE-A31-0002-04 R1
4.	Unit 3 data as documented in TS-362	
5.	Technical Specifications, BFNP Unit 3	Amendment 254

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section II – Containment Isolation Instrumentation Configuration Data

A. Containment Isolation Instrumentation System

The instrumentation for the containment isolation logic groups (table 1) initiates signals to perform various containment isolation functions as follows:

Group 1	Main steam isolation valve closure is initiated by one-of-two twice logic, and main steam drain valve by one-of-two-twice from Group 1 instrumentation.
Group 2	The RHR shutdown cooling suction and LPCI injection valves, the drywell floor and equipment sump drain valves, and the torus drain valves are isolated by one-of-two twice logic from Group 2 instrumentation. Instrumentation used for Group 2 isolation is common to RPS. A LPCI initiation signal will override the Group 2 isolation signal for the LPCI injection valves. Group 2 isolation also initiates Group 8 (TIP withdrawal).
Group 3	RWCU supply valve isolation is initiated by one-of-two-twice logic from Group 3 instrumentation.
Group 4	HPCI steam supply and pump suction isolation is initiated by one-of-two-twice logic (with the exception of high steam line flow (one-of-two) from Group 4 instrumentation.
Group 5	RCIC steam supply and pump suction isolation is initiated by one-of-two-twice logic (with the exception of high steam line flow (one-of-two) from Group 5 instrumentation.
Group 6	Group 6 isolation is initiated by four sets of instrumentation, two of which are common to RPS (RPV water level and high drywell pressure). The RPV water level and high drywell pressure isolation are both one-of-two-twice. Radiation monitor HIGH logic is one-of-(two-of-two twice). Radiation monitor DOWNSCALE logic is one-of-two-twice. The Group 6 isolation signal activates SGTS, isolates the reactor zone, and refueling zone secondary containment boundaries, and isolates the reactor building main vent.

The Group 6 logic also isolates the following primary containment isolation valves: Containment purge and exhaust, containment inerting makeup/purge, drywell control air, post accident sampling, drywell/torus differential pressure compressor line, hydrogen/oxygen analyzers sample and return, CAD exhaust to SGTS, and airborne radiation monitor sample lines.

The relays used are a mixture of GE HFA, and CR 120 relays (as are the Unit 3 relays).

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section II – Containment Isolation Instrumentation Configuration Data

B. Containment Isolation Instrumentation System

PCIS Group 1 (Main Steam Line Isolation)						
Function	**Instrument Number**	**Function Test Frequency**[18]	**Calibration Frequency**	**Common to ECCS**	**Common to RPS**	**Same as Unit 3**
RPV Level (low-low-low)	LIS-3-56A LIS-3-56B LIS-3-56C LIS-3-56D	92 Days	24 M	No	No	Yes
MS Line Pressure Low	PIS-1-72 PIS-1-76 PIS-1-82 PIS-1-86	92 Days [19]	24 M	No	No	Yes
Main Steam Line High Flow	PDIS-1-13 (A-D) PDIS-1-25 (A-D) PDIS-1-36 (A-D) PDIS-1-50 (A-D)	92 Days [20]	24 M	No	No	Yes
Main Steam Tunnel Space High Temp	TS-1-17 (A-D) TS-1-29 (A-D) TS-1-40 (A-D) TS-1-54 (A-D)	92 Days	24 M	No	No	Yes

[18] The Logic is tested as noted above (actual actuation is tested each refueling outage).
The Logic is ((A OR C) AND (B OR D))
[19] The functional test frequency was decreased to once per 3 month to reduce challenges to relief valve settings per NUREG 0737 Item II.K3.16. TS TBL 3.3.6.1-1.
[20] The functional test frequency was decreased to once per 3 month to reduce challenges to relief valve settings per NUREG 0737 Item II.K3.16. TS TBL 3.3.6.1-1.

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section II – Containment Isolation Instrumentation Configuration Data

B. Containment Isolation Instrumentation System

PCIS Group 2 (RHR, Drywell Floor & Equipment Drain Valve Isolation)						
Function	**Instrument Number**	**Function Test Frequency[21]**	**Calibration Frequency**	**Common to ECCS**	**Common to RPS**	**Same as Unit 3**
RPV Level 3 (low)	LIS-3-203A LIS-3-203B LIS-3-203C LIS-3-203D	92 Days	24 M	No	Yes	Yes
Drywell Pressure High	PIS-64-56A PIS-64-56B PIS-64-56C PIS-64-56D	92 Days	24 M	No	Yes	Yes

[21] The Logic is tested as noted above (actual actuation is tested each refueling outage). The Logic is ((A OR C) AND (B OR D))

Section II – Containment Isolation Instrumentation Configuration Data

B. Containment Isolation Instrumentation System

PCIS Group 3 (RWCU System Isolation)						
Function	**Instrument Number**	**Function Test Frequency[22]**	**Calibration Frequency**	**Common to ECCS**	**Common to RPS**	**Same as Unit 3**
RPV Level 3 (low)	LIS-3-203A LIS-3-203B LIS-3-203C LIS-3-203D	92 Days	24 M	No	Yes	Yes
Main Steam Tunnel Temp High	TIS-69-834A TIS-69-834B TIS-69-834C TIS-69-834D	92 Days	24 M	No	No	Yes
RWCU Pipe Trench Temp High	TIS-69-835A TIS-69-835B TIS-69-835C TIS-69-835D	92 Days	122 Days	No	No	Yes
RWCU Pump Room 2A Temp High	TIS-69-836A TIS-69-836B TIS-69-836C TIS-69-836D	92 Days	122 Days	No	No	Yes
RWCU Pump Room 2B Temp High	TIS-69-837A TIS-69-837B TIS-69-837C TIS-69-837D	92 Days	122 Days	No	No	Yes
RWCU Exchanger Room Temp High	TIS-69-838A TIS-69-838B TIS-69-838C TIS-69-838D	92 Days	122 Days	No	No	Yes
RWCU Heat Exchanger Room Pipe Chase Temp High	TIS-69-839A TIS-69-839B TIS-69-839C TIS-69-839D	92 Days	122 Days	No	No	Yes

[22]The Logic is tested as noted above (actual actuation is tested each refueling outage).
The Logic is ((A OR C) AND (B OR D))

Section II – Containment Isolation Instrumentation Configuration Data

B. Containment Isolation Instrumentation System

PCIS Group 4 (HPCI System Line Isolation)						
Function	**Instrument Number**	**Function Test Frequency**	**Calibration Frequency**	**Common to ECCS**	**Common to RPS**	**Same as Unit 3**
HPCI Steam Line High Flow	PDIS-73-1A PDIS-73-1B	92 Days	24 M	No	No	Yes
HPCI Steam Supply Pressure Low	PS-73-1A PS-73-1B PS-73-1C PS-73-1D	92 Days	24 M	No	No	Yes
HPCI Turbine Exhaust Rupture Disc Pressure High	PS-73-20A PS-73-20B PS-73-20C PS-73-20D	92 Days	24 M	No	No	Yes
HPCI Steam Line Space Temp High (HPCI Room)	TS-73-2A TS-73-2B TS-73-2C TS-73-2D	92 Days	92 Days	No	No	Yes
HPCI Steam Line Space Temp High (Torus Room)	TS-73-2 (E,F,G,H) TS-73-2 (J,K,L,M) TS-73-2 (N,P,R,S)	92 Days	92 Days	No	No	Yes

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section II – Containment Isolation Instrumentation Configuration Data

B. Containment Isolation Instrumentation System

PCIS Group 5 (RCIC System Line Isolation)						
Function	**Instrument Number**	**Function Test Frequency**	**Calibration Frequency**	**Common to ECCS**	**Common to RPS**	**Same as Unit 3**
RCIC Steam Line High Flow	PDIS-71-1A PDIS-71-1B	92 Days	24 M	No	No	Yes
RCIC Steam Supply Pressure Low	PS-71-1A PS-71-1B PS-71-1C PS-71-1D	92 Days	24 M	No	No	Yes
RCIC Turbine Exhaust Rupture Disc Pressure High	PS-71-11A PS-71-11B PS-71-11C PS-71-11D	92 Days	24 M	No	No	Yes
RCIC Steam Line Space Temp High (RCIC Room)	TS-71-2A TS-71-2B TS-71-2C TS-71-2D	92 Days	92 Days	No	No	Yes
RCIC Steam Line Space Temp High (Torus Room)	TS-71-2 (E,F,G,H) TS-71-2 (J,K,L,M) TS-71-2 (N,P,R,S)	92 Days	92 Days	No	No	Yes

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section II – Containment Isolation Instrumentation Configuration Data

B. Containment Isolation Instrumentation System

PCIS Group 6						
Function	**Instrument Number**	**Function Test Frequency**	**Calibration Frequency**	**Common to ECCS**	**Common to RPS**	**Same as Unit 3**
RPV Level 3 (low)	LIS-3-203A LIS-3-203B LIS-3-203C LIS-3-203D	92 Days	24 M	No	Yes	Yes
Drywell Pressure High	PIS-64-56A PIS-64-56B PIS-64-56C PIS-64-56D	92 Days	24 M	No	Yes	Yes
Reactor Zone Radiation High	RM-90-142A RM-90-142B RM-90-143A RM-90-143B	92 Days	24 M	No	No	Yes
Refuel Zone Radiation High	RM-90-140A RM-90-140B RM-90-141A RM-90-141B	92 Days	24 M	No	No	Yes

Section II – Containment Isolation Instrumentation Configuration Data

B. Containment Isolation Instrumentation System

Allowed Out of Service Times

Current Technical Specifications allow one channel of containment isolation instrumentation to be placed in an inoperable status for up to six hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. For the reactor building ventilation system, one channel may be inoperable for up to 6 hours for functional testing or for up to 24 hours for calibration and maintenance, as long as the downscale trip of the inoperable channel is placed in the tripped condition.

C. Summary

As noted in the tables the logic is the same, and the relays used are the same in the logic. The instrumentation is the same (TS Settings are the same) between Unit 1 and Unit 3 therefore any conclusions drawn for Unit 3 are applicable to Unit 1 also.

RPS Instrumentation Data

EVALUATION CHECKLIST FOR

BROWNS FERRY NUCLEAR PLANT – UNIT 1

Comparison to Unit 3

Support for applicability analysis
BWR Owners' Group
Technical Specification Improvement Analyses
for
RPS Instrumentation

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section I – RPS Instrumentation Plant Specific Data Sources

Source
Number

1.	RPS Elementary Diagrams:	1,3-730E915-Series
2.	RPS Circuit Protector Elementary Diagrams:	1,3-45E641-5
3.	RPS Design Criteria:	BFN-50-7099 Rev 7
4.	FSAR Section 7	
5.	Technical Specifications, BFNP Unit 1	Amendment 257
6.	BFN SIs	(e.g., 1-SR-3.3.1.1.9(IRM A), 1-SR-3.3.1.1.15(B1)
7.	Main Steam Flow Diagram	1,3-47E801-1
8.	Unit 3 data as documented in TS-362	
9.	Technical Specifications, BFNP Unit 3	Amendment 254

Section II – RPS Instrumentation Configuration Data

A. RPS Instrumentation System

Item	Data	Same as Unit 3	Data Source (see references)
Number of trip systems	2	Yes	1
Number of logic channels per trip system: For Automatic Scram For Manual Scram	 2 1	Yes	 1 1
Power Supply source for each channel	MG Set[23]	Yes	2
Operation Mode De-energize to trip	Yes	Yes	1
Logic Arrangement One-of-two twice	Yes[24]	Yes	1
Electrical Protection Assemblies (EPAs)	Yes	Yes	2
Design Requirement	IEEE 279	Yes	3, 4

[23] There is an alternate feed from a regulating transformer (also circuit protected).
[24] Trip logic is one-of-two twice except for (1) Turbine stop valve closure which is 3-of-4, and (2) MSIV closure scram which is 3-of-4 steam lines less than 90% open.

Section II – RPS Instrumentation Configuration Data

B. RPS Sensors

1. Identify the type, total number, and number per RPS channel for the following RPS sensors

	Type	Total Number	Number per RPS Channel	Reference	Same as Unit 3
APRM	Analog	6	2	2	Yes
Turbine Stop Valve	Position switch	4	2	2	Yes
Turbine Control Valve	Pressure switch	4	1	2	Yes
MSIV Position	Pressure switch	8	4	2	Yes
RPV Low Level 3	ATU	4	1	2	Yes
SDV Level Type 1	Heated RTD	4	1	2, 5	Yes
SDV Level Type 2	Float switch	4	1	2, 5	Yes
High Reactor Pressure	ATU	4	1	2	Yes
High Drywell Pressure	ATU	4	1	2	Yes
Manual Trip[25]	Switch	2	1	2	Yes
Mode Switch Trip	Switch	1	1	2	Yes
Low Condenser Vacuum	NA				Yes
Low Scram Air Header Pressure	Pressure switch	4	1	2	Yes
IRM	Analog	8	2	2	Yes

[25]There are two manual trip switches one for each of the two RPS trip systems.

Section II – RPS Instrumentation Configuration Data

B. RPS Sensors

2. Turbine stop valve closure logic arrangement is closure of 3-out-of-4 valves initiates scam — Reference 2
3. Turbine stop valve closure monitoring is via position switches — Reference 2
4. Turbine control valve fast closure monitoring is via oil pressure switches — Reference 2
5. MSIV closure logic arrangement is closure of 3-out-of-4 mainsteam lines initiates scam — Reference 2
6. Diversity in SDV level sensors is via float switches and heated RTD level sensors — Reference 5
7. BFN Unit 1 has 4 steam lines as does Unit 3 — Reference 7
8. List of available bypasses — Reference 1, 3, 4

IRM trip bypass	yes
APRM trip bypass	yes
Noncoincident neutron monitoring system trip bypass	yes[26]
RPV high level RPS trip bypass	NA
Turbine stop valve RPS trip bypass	yes[27]
Turbine control valve RPS trip bypass	yes[28]
MSIV closure RPS trip bypass	yes[29]
SDV high level trip bypass	yes[30]
Reactor mode switch "shutdown" mode trip bypass	no

Technical Specification Table 3.3.1.1-1 for both units shows the same settings for the trip instrumentation.

[26] Shorting links
[27] Bypassed at <30% power
[28] Bypassed at <30% power
[29] In refuel/shutdown
[30] SDV bypassed in shutdown

Section II – RPS Instrumentation Configuration Data

C. Sensor Relays

1. Types of relays used are GE type HFA and CR120A. Contactors are GE type CR105 per Reference 1.
2. Number of pairs of contacts per relay in trip channel is 2 per Reference 1.
3. List type of relay for each RPS sensor

	CR105	HFA	CR120
• APRM/IRM		X	X
• Turbine Stop Valve		X	
• Turbine Control Valve		X	
• MSIV Position		X	
• RPV Level 3		X	X
• SDV Level RTD		X	
• SDV Level Float Sw		X	
• High RPV Pressure		X	
• High Drywell Pressure		X	
• Manual Trip	X		
• Mode Switch Trip		X	

D. Scram Contactors

1. Scram contactors are GE CR105 per Reference 1.
2. Total number of scram contactors is 8 for auto trip and 4 for manual trip per Reference 1.
3. The number of contactors per channel is 2 for auto per Reference 1.

E. Air Pilot Solenoids Valves

1. There are 2 solenoid valves per control rod per Reference 1, 4.
2. There are 2 scram solenoid valves per control rod drive. Both scram solenoid valves must de-energize to vent the control air header which opens the associated scram outlet valve and opens the associated scram inlet valve to insert the control rod.

F. Backup Scram Valves

1. Scram contactors for the Backup scram valves are GE type CR105 per Reference 1.
2. There are 6 scram contactors for each backup scram valve (4 automatic, 2 manual) per Reference 1.
3. The scram contactors are the same as those used for RPS per Reference 1.
4. The backup scram valves are energized to actuate per Reference 1.
5. The technical specifications do not specify any tests for the backup scram valves.

Tennessee Valley Authority

Browns Ferry Nuclear Plant – Unit 1

Section II – RPS Instrumentation Configuration Data

G. RPS Technical Specification Requirements

1. Calibration frequency for LPRMs is every 1000 effective full power hours. Per Reference 5.
2. Calibration frequency for trip units is not specified directly but a functional check of the setpoints is performed monthly as part of 1-SR-3.3.1.1.9(IRMA-H) and Reference 7 (If outside bounds calibration occurs).
3. Frequency of logic system functional are not specifically called out in the technical specifications but are performed as part of the ATU calibrations in item 2 (up to half scrams). See item 2 references.
4. The allowable time to place an inoperable channel or trip system in tripped condition when the number of operable channels is less than the required minimum operable channels for one trip systems is 1 hour. Reference 5.
5. There is an exception to item 4, six hours is allowed to perform required surveillances. (TS 3.3.1.1, Reference 5).
6. Allowable time to restore a trip system when the number of operable channels is less than the required minimum operable channels for both trip systems where placing the channel in trip will trip the plant is 1 hour per Reference 5.
7. There are no exceptions to item 6.

Functional Unit	Channel Check	Functional Test	Channel Calibration	Operable Channels per Trip System
APRM				
• Flow biased simulated thermal power high	N/A	184 Days	24 M	3
• Neutron flux high		184 Days	24 M	3
• Inoperative		184 Days	N/A	3
• Downscale		W	24 M	3
Reactor Vessel Steam Dome Pressure High	N/A	92 Days	184 Days	2
Reactor Vessel Water Level Low Level 3	N/A	92 Days	24 M	2
Reactor Vessel Water Level High Level 8	N/A	92 Days	24 M	2
MSIV Closure	N/A	92 Days	24 M	8
Main Steam Line High Radiation	N/A	N/A	N/A	N/A
Drywell Pressure High	N/A	92 Days	24 M	2
Main Condenser Low Vacuum	N/A	N/A	N/A	N/A
SDV High Level (RTD and Float SW)	N/A	92 Days	24 M	2
Turbine Stop Valve Closure	N/A	92 Days	24 M	4
Turbine Control Valve Fast Closure (oil pressure low)	N/A	92 Days	24 M	2
Reactor Mode Switch Shutdown Position	N/A	24 M	N/A	1
Manual Scram	N/A	92 Days	N/A	1
Scram Air Header Pressure Low	N/A	184 Days	24 M	2
IRM				
• Neutron Flux High		7 Days	92 Days	3
• Inoperative		7 Days	N/A	3

W = Weekly • Q = Quarterly • M = Monthly • R = Refueling Outage • WR = Weekly During Refueling Outage

Section II – RPS Instrumentation Configuration Data

H. RPS Surveillance Test Procedures

1. The following components are all tested as part of an individual channel functional test: (Reference 8)
   a. Individual channel sensor(s), e.g., transmitters and trip units, switches, flux or radiation sensors.
   b. Associated logic relay(s)
   c. Associated scam contactors

   List any plant specific differences from the above.

   RESPONSE

   Transmitters are calibrated once per fuel cycle. The trip unit's setpoints are verified monthly and calibrated if outside bounds (1-SR-3.3.1.1.9(IRMA-H)).

2. When an individual sensor channel is in test or repair, is associated logic channel tripped or is the sensor channel jumpered? State which of two conditions applies to your plant. If any other condition exists in your plant, describe. Reference 8.

   RESPONSE

   The channel is not tripped prior to the test, or jumpered. The test will trip the channel to verify proper functioning.

3. For those plants which do not place individual channels in a tripped condition during test or repair, it is assumed in the GE analysis that only the individual sensor and associated logic relay is placed in an inoperable condition during test or repair of the individual channel. If this assumption is not true for your plant, list the components (from sensor to scram contactors) which are placed in inoperable condition during test or repair (Reference 8).

   RESPONSE

   This assumption is true for BFNP-Unit 1.

4. The following number of individual scram contactor actuations are assumed in the GE analyses for each channel functional test: Reference 8.

   a. APRM channel functional tests 2 actuations per scram contactor pair in each trip logic channel.
   b. MSIV closure channel functional tests 4 actuations per scram contactor pair in each trip logic channel.

Section II – RPS Instrumentation Configuration Data

H. RPS Surveillance Test Procedures

c. Other channel functional tests 1 actuation per scram contactor pair in each trip logic channel.

List any differences from the above for your specific plant.

RESPONSE

Turbine Stop Valve Functional Tests – 2 actuations

5. Do plant procedures allow simultaneous inoperable conditions (failed condition) of diverse sensors in a given logic channel?

RESPONSE

Yes, provided the associated RPS logic channel or the affected instrument channels are placed in the tripped condition.

I. Summary

The Unit 1 RPS configuration is the same as the Unit 3 RPS configuration.

BROWNS FERRY NUCLEAR PLANT
Unit 1 Comparison to Unit 3

Unit Differences

ECCS INSTRUMENTATION

Differences Based on Comparison of Unit 1 & Unit 3:

Results of Review of ECCS-RHR (Residual Heat Removal) Instrumentation

Design Criteria Document BFN-50-7074, R17

1. RHR logic circuitry provisions in U1 to initiate U2 ECCS Preferred Pump Logic; nothing in U3 because U4 does not exist.

2. For a LOCA with normal power available in either U1 or U2, RHR logic circuitry provisions in U1 to initiate U2 ECCS Preferred Pump Logic; nothing in U3 because U4 does not exist.

3. A spurious accident signal from a non-accident unit combined with a real accident signal from the other unit, the U1/U2 ECCS Preferred Pump Logic shall generate a signal to the U1 RHR pump start logic to dedicate it to U1 and same for U2; nothing in U3 because U4 does not exist.

4. For Unit 1 RHR loop isolation, the RHR loop crosstie header isolation valve has been removed and the pipe capped per DCN 51199. A Technical Specification Change Request and Design Criteria revision are in process for this change. For Unit 2, a normally closed electrically disabled isolation valve isolates the loops. For Unit 3, either an electrically disabled isolation valve or a locked manual shutoff valve is maintained closed to provide loop isolation.

5. LPCI operation for U1 / U2 different than U3.

6. RHR logic circuitry provisions in U1 to initiate U2 ECCS Preferred Pump Logic on RV level 1 or high Drywell pressure with low RV pressure.

7. Core spray logic circuitry provisions in U1 to initiate U2 ECCS Preferred Pump Logic to trip running RHR pumps and to divisionally assign RHR pumps so that Div. 1 to U1 and Div. II to U2.

8. For a LOCA with normal power available in either U1 or U2, RHR logic circuitry provisions in U1 to initiate U2 ECCS Preferred Pump Logic to trip any running RHR and Core Spray Pumps and other selected 4kv loads in the non-accident unit.

9. A spurious accident signal from a non-accident unit combined with a real accident signal from the other unit, the U1/U2 ECCS Preferred Pump Logic shall generate a signal to the RHR pump start logic to automatically lockout the U1 Div II pumps and the U2 Div 1 pumps…; nothing in U3 because U4 does not exist.

10. RHR to provide a signal to the ECCS Preferred Pump Logic to trip any Core Spray pumps and to divisionally assign Core Spray pumps Div. 1 to U1 and Div. II to U2.

RESOLUTION:

The above unit differences are associated with the sharing of diesel generators between Units 1&2 and the preferred pump logic. The NRC has approved the Unit's configuration in the SER associated with TS-424.

Results of Review of ECCS-CSCS (Core Spray Cooling System) Instrumentation

Design Criteria Document BFN-50-7075, R10

1. Units 1 and 2 only: A spurious accident signal from the non-accident unit combined with a real accident signal from the other unit is a design basis single failure. To compensate for this single failure, the Core Spray logic shall provide a signal to the opposite unit to initiate the Unit 1/2 ECCS Preferred Pump Logic to dedicate the Division I Core Spray and RHR pumps to Unit 1 and the Division II Core Spray and RHR pumps to Unit 2 (References 8.1.12, 8.1.17, and 8.1.18). This prevents unanalyzed loading of the Unit 1 and 2 4kV Shutdown Boards following a spurious accident signal with a real accident signal due to a LOCA with a loss of offsite power (see Section 3.9-1).

2. Units 1 and 2 only: For a LOCA with normal power available in either Units 1 or 2, the Core Spray logic circuitry shall provide a signal to the opposite non-accident unit to initiate the Unit 1/2 ECCS Preferred Pump Logic to trip any running RHR or Core Spray pumps, and other selected 4kV loads in the non-accident unit (Reference 8.1.17 and 8.1.18). This allows all four RHR and Core Spray pumps to start in the accident unit without overloading the 4kV Shutdown Buses and maintains the reliability of the normal offsite power distribution system.

   In the event of a real accident signal combined with a spurious accident signal from the opposite unit with normal power available, the Unit 1/2 ECCS Preferred Pump Logic dedicates the Division I RHR and Core Spray pumps to Unit 1 and the Division II RHR and Core Spray pumps to Unit 2. This prevents overloading the 4kV Shutdown Buses and maintains the reliability of the normal offsite power distribution system while ensuring that at least one division of ECCS pumps are available in the unit with the real accident.

3. A control circuit shall be provided at the 4kV shutdown boards (four for units 1 and 2, four for unit 3) for its respective core spray pumps to establish control at the boards of all CS pumps (to trip and lock out the pumps) independent of the condition of the control room or spreading room circuits in accordance with reference 8.3.24 (to prevent potential overloading of these 4kV buses/diesel). (C/R BFNEEBGRR1411)

4. Following an initiation of a Common Accident Signal from either Unit 2 or Unit 3 (which trips the diesel beakers), a second diesel breaker trip initiated for the RHR system on a "unit

priority" basis ensures that the diesel supplied buses are stripped prior to starting the ECCS pumps and other required loads.

5. Units 1 and 2 only: The ECCS Preferred Pump Logic shall provide a signal to lock the Unit 1 initiated Unit Priority Re-Trip of the Division II diesel generator breakers and the Unit 2 initiated Re-Trip of the Division I diesel generator breakers (see BFN-50-7075, section 3.9.11). This prevents unanalyzed loading of the Unit 1/ 2 4kV shutdown Boards and diesel generators while maintaining the minimum number of required RHR pumps for each unit.

6. Units 1 and 2 only: The Core Spray logic circuitry shall provide a signal to the opposite unit on RV water Level 1 on high drywell pressure concurrent with low RV pressure to initiate the Unit 1/ 2 ECCS Preferred Pump Logic. The ECCS Preferred Pump Logic will trip any running core Spray pump in the opposite unit. After 60 seconds, the trip signal shall be removed so that the opposite unit's operators may manually restart the Core Spray pumps when the 4 kV Shutdown Board electrical loading conditions will allow the restoration of the pumps.

7. An ECCS inhibit key-lock switch shall be provided to block the automatic start of the Core Spray pumps and automatic opening of the Core Spray injection valves (Units 1 and 2 only). Only the automatic initiation functions are impacted. Manual control of the Core Spray system is not affected by these key-lock switches. Units 1 and 2 only: The ECCS inhibit key-lock switch will also block the initiation of the opposite Unit's ECCS Preferred Pump Logic.

8. Units 1 and 2 only: The RHR system shall provide an initiation signal to the ECCS Preferred Pump Logic (redundant to the Core Spray initiation signal) to trip any running Core Spray pumps and divisionally assign the Core Spray pumps so that the Division I pumps are dedicated to Unit 1 and Division II pumps are dedicated to Unit 2 (see section 3.9(12)).

9. Units 1 and 2 only: The Core Spray system shall provide an initiation signal to the ECCS Preferred Pump Logic (redundant the RHR initiation signal) to trip any running RHR pumps and divisionally assign the RHR pumps so that the Division I pumps are dedicated the Unit1 and the Division II pumps are dedicated to Unit 2 (Reference 8.3.3).

RESOLUTION:

The above unit differences are associated with the sharing of diesel generators between Units 1&2 and the preferred pump logic. The NRC has approved the Unit's configuration in the SER associated with TS-424.

Results of Review of ECCS-EECW (Emergency Equipment Cooling Water) Instrumentation

<u>Design Criteria Document, BFN-50-7067, R15</u>

1. U1/U2 & U3 fed from primary and emergency EECW headers. For U1/U2, RCW is available to the RHR HXs & Pump Room Coolers and Control Bay Chillers.

2. RCW serves as an alternate cooling water supply for the U3 Shutdown Board Chillers and U3 H2 & O2 Analyzers/Panels.

3. The EECW serves as cooling water supply to the U1/U2 Control Bay emergency condensing unit.

4. U1/U2 Emergency Cooling Unit is valved in and out of service if the U1/U2 Chillers are out of service and valved out. U3 does not have these provisions.

5. U1 and U3 have different sources of Class 1E power to their components. For example, the 480v Reactor MOV Board supplies power to sectionalizing valves in the reactor and U3 DG buildings. U1 receives power from the 480v Diesel Auxiliary Board.

6. Piping from the two loop headers are routed differently for the U1/U2 and U3 DG buildings.

RESOLUTION:

The EECW system is shared among all three units and was previously evaluated by Unit 3. Therefore, the above items have been previously deemed acceptable.

The following drawings associated with the ECCS Instrumentation have the listed differences between Unit 1 and Unit 3:

1-45E670-3, -5, & -11

The Unit 1 MSRV Auto Actuation logic derives its reactor pressure signal from master trip units 1-PIS-3-244A, 244B, 244C, and 244D and associated slave trip units. These loops are independent in that they have no other design functions than to provide signals for the MSRV Auto Actuation logic. This is a Unit difference from Units 2 and 3 which derive their signals from the P-3-204A, 204B, 204C, and 204D loops associated with ATWS, ARI, EHC, etc. The independence of the loops is an enhancement that reduces the likelihood of system interface errors. The safety related features of the MSRV's designed to prevent reactor pressure from exceeding the technical specification limits has not changed by the addition of this MSRV Auto Actuation logic.

RPV LEVEL 8 FEEDWATER PUMP/MAIN TURBINE TRIP

Differences Based on Comparison of Unit 1 & Unit 3:

Results of Review of RPV Level 8 Feedwater Pump/Main Turbine Trip Instrumentation

The drawings associated with the RPV Level 8 Feedwater Pump/Main Turbine Trip Instrumentation were reviewed and no functional differences were found between Unit 1 and Unit 3.

CREV INITIATION INSTRUMENTATION

Differences Based on Comparison of Unit 1 & Unit 3:

<u>Design Criteria Document BFN-50-7030A, R12</u>

1. SR & Non-SR HVAC do not supply HVAC to the same areas for U3 & U1:
    a. Computer Room is SR, while U1 Computer Room is not
    b. SR HVAC to U3 Shutdown Board in DG Bldg.

2. The Chilled Water systems between U1 and U3 differ.
    c. U1 MCR AHUs have non-SR cooling coils served by a single cooling water condensing unit.
    d. Flow switches provided with U3 water chiller. U1 abandoned in place.

3. U1 MCR AHUs operation and chilled water interconnections not identical to U3.
    e. U1/U2 standby AHU cooled by U3 chilled water
    f. U1/U2 standby AHUs automatically switch on loss of power, U3 assumed to only have one (1) AHU.

4. U3 water chillers and condensing units receive cooling water from EECW. Raw Cooling Water supplies U1 direct expansion refrigerant condensing units w/U3 backup.

5. U1 250v battery vent system furnishes air to control H2 concentrations.
    g. U1&U2 250v Battery Room downgraded to non-SR.

RESOLUTION:

Since Unit 1 and Unit 2 have a shared control room, the above items were resolved during the Unit 3 evaluation and are therefore, not a concern.

END-OF-CYCLE RPT & ATWS RPT INSTRUMENTATION

Differences Based on Comparison of Unit 1 & Unit 3:

Results of Review of RWRS (Reactor Water Recirculation System) & RFCS (Recirculation Flow Control System) Instrumentation

<u>Design Criteria Document BFN-50-7068, R11</u>

1. U1 Recirculation Pumps are constant speed pumps and U3 and U2 have variable frequency drives.
    a. Controls associated with each should be physically different as will be the wiring.

RESOLUTION:

DCN 51219 will be installing variable frequency drives in Unit 1.

CONTAINMENT ISOLATION INSTRUMENTATION

Differences Based on Comparison of Unit 1 & Unit 3:

Results of Review of Containment Isolation Instrumentation

The following drawings associated with the Containment Isolation Instrumentation have the listed differences between Unit 1 and Unit 3:

1-730E927-10, & -11

The Containment Isolation Status System (CISS) has been designed as a non-safety related system for Unit 1 with class-1E isolation for the safety related system interfaces. Units 2 and 3 CISS systems were designed as safety related. The differences in the Unit 1 CISS are due to the safety related / non-safety related interfaces. The design function of the CISS remains the same as Units 2 and 3 except that valve stroke times are monitored by computer to verify stroke times for surveillances.

RPS INSTRUMENTATION

Differences Based on Comparison of Unit 1 & Unit 3:

Results of Review of RPS (Reactor Protective System) Instrumentation

<u>Design Criteria Document BFN-50-7099, R11</u>

1. The RPS shall generate a reactor scram upon receipt of : (1) high SDV water level or (2) low scram discharge air header pressure trip signals. Either of these conditions will prevent effective insertion of the control rods into the active core region. Note: Not applicable to Unit 2 per DCN 50897 or Unit 3 per DCN 50729.

    The RPS shall generate a reactor scram upon receipt of high SDV water level. This condition will prevent effective insertion of the control rod drives into the active core region. Applicable to Unit 2 per DCN 50897, Unit 3 per DCN 50729.

2. The NTSP value used for the control air header low pressure scram shall be selected low enough to avoid spurious scrams and high enough to prevent unseating the scram inlet and outlet values. Note: Not applicable to Unit 2 per DCN 50897 or Unit 3 per DCN 50729.

3. Scram discharge low air header pressure- Not applicable to Unit 2 per DCN 50897 or Unit 3 per DCN 50729.

4. The reactor scram function for a high Scram Discharge Volume (SDV) level or low scram discharge air header pressure trip shall be bypassed by the combination of both the SDV scram bypass switch and the reactor mode switch being placed in the shutdown or refuel position. A keylock switch in proximity to the scram reset switches shall be furnished to bypass these two scram initiators as required under administrative control. Note: Not applicable to Unit 2 per DCN 50897 or Unit 3 per DCN 50729.

   The reactor scram function for a high Scram Discharge Volume (SDV) level trip shall be bypassed by the combination of both the SDV scram bypass switch and the reactor mode switch being placed in the shutdown or refuel position. A key switch in proximity to the scram reset switches shall be furnished to bypass these two scram initiators as required under administrative control. Note: Applicable to Unit 2 per DCN 50897 and Unit 3 per DCN 50729.

5. The CRD system shall monitor control air header pressure to the HCU and SDV equipment. Pressure switches shall initiate scram signal outputs to the RPS trip logic if the air pressure is too low. Note: Not applicable to Unit 2 per DCN 50897 or Unit 3 per DCN 50729.

RESOLUTION:

Unit 1 DCNs 51080 and 51206 will install modifications to Unit 1 that will be equivalent to the changes in Units 2&3 and the Units will then be the same.

The drawings associated with the RPS Instrumentation were reviewed and no functional differences were found between Unit 1 and Unit 3.