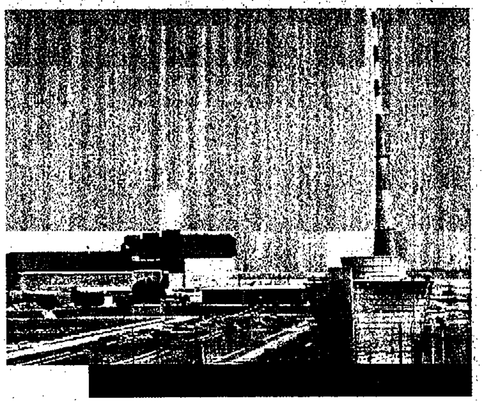
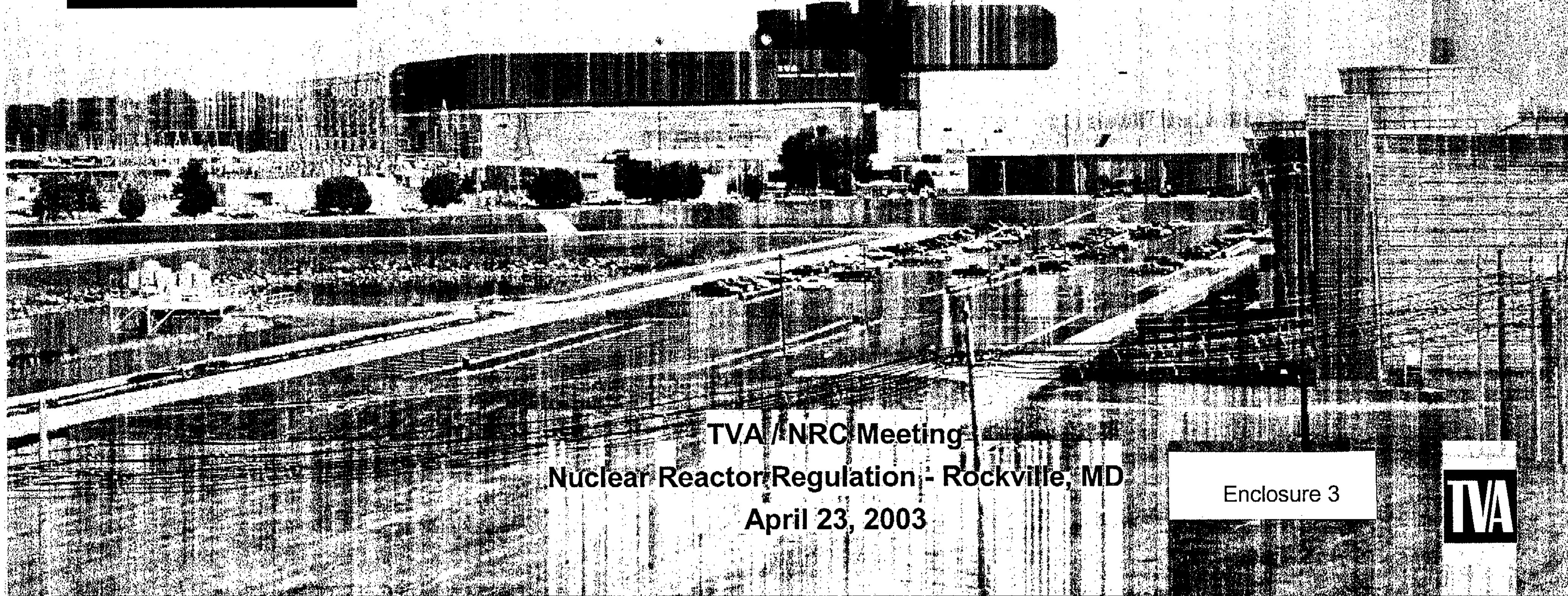
Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3



Meeting Handouts



TVA / NRC Meeting
Nuclear Reactor Regulation - Rockville, MD
April 23, 2003

Enclosure 3

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TVA / NRC Meeting
Nuclear Reactor Regulation - Rockville, MD
April 23, 2003

License Renewal Application
Browns Ferry Units I, 2, and 3

Handouts

Tab 1: Unit 1 Restart – Major Milestones

Tab 2: Application Examples, Chapters 2 and 3

- Main Steam System
- Reactor Recirculation System
- Residual Heat Removal System

Tab 3: Application Example, Chapter 4

- Neutron Radiation Embrittlement

Tab 4: Application Example, Appendix B

- XI.M17, Flow Accelerated Corrosion

Tab 5: Application Example, Appendix F

- Integration of Browns Ferry, Unit 1 Restart Activities and License Renewal

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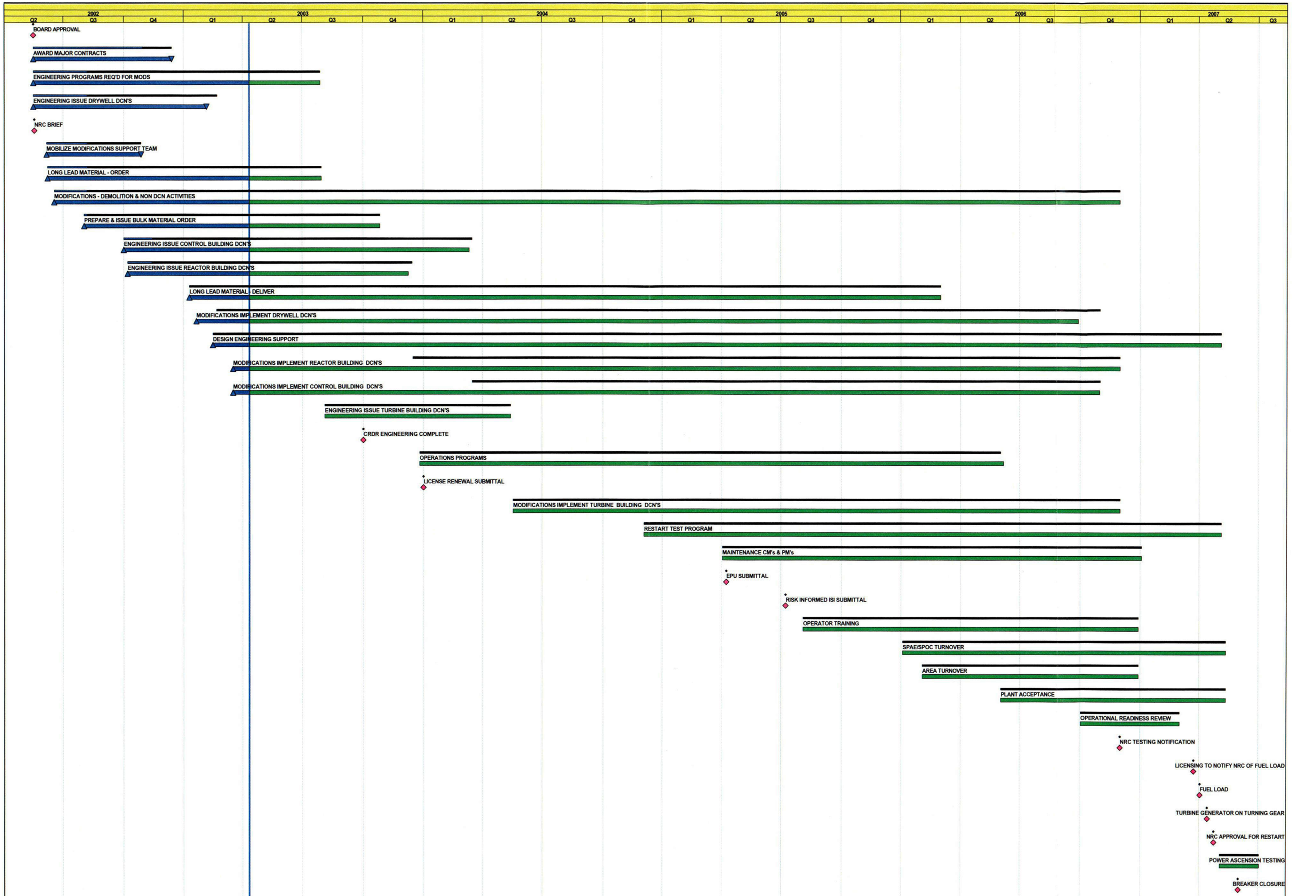

Start Date	01APR02
Finish Date	02JUL07
Data Date	10APR03
Run Date	10APR03 16:31

TARGET BAR
Early Bar
Progress Bar
Critical Activity

TVA BFN UNIT-1
MAJOR MILESTONES
REV 2

C01

2.3.4.1. Main Steam (System 001)

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System Description

The Main Steam system for each unit consists of four main steam lines that transfer steam from reactor vessel to the various steam loads in the turbine building during normal plant operation. Two Main Steam Isolation Valves are provided in each steam line to isolate the reactor coolant pressure boundary and the primary containment. Steam supply lines for the High Pressure Coolant Injection and the Reactor Core Isolation Cooling systems branch off the main steam lines between the reactor vessel and the Main Steam Isolation Valves. A flow restrictor is provided in each main steam line. The flow restrictor allows for measurement of steam flow and limits the steam flow rate in the event of a downstream steam line break. Thirteen main steam relief valves are provided on the main steam lines upstream of the flow restrictors for overpressure protection and for depressurization following Small Break Loss of Coolant Accidents. The Main Steam System for each unit shares no components with the other units. Main Steam components downstream of the Main Steam Isolation Valves are credited in analyses for Main Steam Isolation Valve *alternate leakage treatment*.

The Main Steam system is in the scope of 10 CFR 54 because it contains components that meet the criteria of 10 CFR 54.4 for the following paragraphs:

(a)(1)	(a)(2)	(a)(3) FP	(a)(3) EQ	(a)(3) ATWS	(a)(3) SBO
Yes	No	Yes	Yes	No	Yes

The Main Steam system meets 10 CFR 54.4(a)(1) because it is relied upon for 1.) reactor vessel overpressure protection, 2.) reactor vessel depressurization during SBLOCAs using the main steam safety relief valves and by providing a motive force to the High Pressure Coolant Injection system, 3.) limiting offsite dose by establishing primary and secondary containment. Main Steam system components also limit off site dose by restricting flow from the reactor vessel following a steam line break and by providing alternate treatment of main steam isolation valve leakage. In addition, portions of the Main Steam system form the reactor coolant pressure boundary. The Main Steam system meets the requirements of 10 CFR 54.4(a)(3) because it contains components that are relied upon in plant evaluations to perform a function that demonstrates compliance with *10 CFR 50.49*. The Main Steam system meets the requirements of 10 CFR 54.4(a)(3) because it contains components that allow removal of reactor residual heat 1.) to meet the safe shutdown requirements of *10 CFR 50 Appendix R* during and following fires, and 2.) to meet the safe shutdown station blackout requirements of 10 CFR 50.63.

The portion of the Main Steam system that contains components subject to an AMR extends from the reactor vessel through the outboard Main Steam Isolation Valves to the secondary containment boundary. It then extends to the first isolation valve for each load in the turbine building and through drains to the main condenser.

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

FSAR References

Additional details for the Main Steam system are found in FSAR *3.7, 4.1, 4.4, 4.5, 4.6, 4.11, 5.2.3, 5.3, 6.4.2, 7.2, 7.3, 7.4, 7.10, 7.11, 7.12, 7.18, 11.2, 11.5, M.6.1, M.6.7* and in the Fire Protection Report Safe Shutdown Analysis 3.6 and the Appendix R Safe Shutdown Program, Sections 3 & 4.

License Renewal Drawings

The license renewal drawings for the Main Steam system are listed below.

Unit 1	Unit 2	Unit 3	Shared
1-47E801-1-LR	*2-47E2847-9-LR*	*3-47E3847-9-LR*	None
1-47E817-1-LR	*2-47E801-1-LR*	*3-47E801-1-LR*	
	2-47E801-2-LR	*3-47E801-2-LR*	
	2-47E807-1-LR	*3-47E807-1LR*	
	2-47E807-2-LR	*3-47E807-2-LR*	
	2-47E817-1-LR	*3-47E817-1-LR*	
	2-47E2847-9-LR	*3-47E3847-9-LR*	

Components Subject to AMR

The component types that require aging management review are indicated in Table *2.3.4.1*, Main Steam System.

The aging management review results for these component types are provided in Table *3.4.2.1* – Steam and Power Conversion Systems - Main Steam System – Summary of Aging Management Evaluation.

Table 2.3.4.1 - Main Steam System

Component Type	Intended Function
Bolting – Class 1	MC
Bolting – Non Class 1	MC
Miscellaneous Appurtenances - Class 1	PB TH
Piping and Fittings - Class 1	PB
Piping and Fittings - Non Class 1	PB
Valves - Class 1	PB
Valves - Non-Class 1	PB

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	AMP	NUREG -1801 Vol. 2 Item	Table 1 Item	Notes
Bolting – Class 1	MC	Carbon & Low Alloy Steel	Internal: None External: Air with metal temperature up to 550°F	Loss of Material due to wear	Bolting Integrity (XI.M18) App. B B.2.11	IV C1.3-e	3.1.1. 26	A
				Loss of pre-load/ due to stress relaxation	Bolting Integrity (XI.M18) App. B B.2.11	IV C1.3-f	3.1.1.26	A
				Fatigue	TLAA Ch 4.x.x.	IV C1.3-g	3.1.1 .26	A
Bolting Non-Class 1	MC	Carbon & Low Alloy Steel	Internal: None External: Air with metal temperature up to 550°F	Loss of material due to general corrosion	Bolting Integrity (XI.M18) App. B B.2.11	VIII H2-a	3.4.1.8	A
				Crack initiation and growth due to cyclic loading and SCC	Bolting Integrity (XI.M18) App. B B.2.11	VIII H2-b	3.4.1.8	A
Miscellaneous Appurtenances - Class 1 (Instrument fittings at flow restrictors)	PB	Carbon steel	Internal: Treated water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chemistry Control Program (XI.M2) App. B B.1.4	None	None	H, 1
Miscellaneous Appurtenances - Class 1 (Venturi inserts for MS Flow Restrictors	TH	CASS	Internal: Treated water	Change in material properties/ reduction in fracture toughness due to thermal aging:	Thermal Aging Embrittlement of CASS (X1.M12) App B B.2.51	None	None	F, 2
Miscellaneous Appurtenances – Non Class 1	TH, PB	CASS and Wrought	Internal: Treated water	Loss of material due to crevice and pitting corrosion: Crack initiation/ growth due to SCC.	Chemistry Control Program (XI.M2) App. B B.1.4	None	None	F, 3
Piping and Fittings - Class 1	PB	Carbon & Low Alloy Steel	Internal: Treated water	Wall Thinning due to FAC	Flow Accelerated Corrosion (XI.M17) App. B B.1.6	IV C1.1-a	3.1.1.25	A
				Loss of material due to general, pitting and crevice and galvanic corrosion	Chemistry Control Program (XI.M2) App. B B.1.4	None	None	H, 4
				Fatigue	TLAA Chapter 4.3.12	IV C1.1-b	3.1.1.1	A

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB

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Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	AMP	NUREG -1801 Vol. 2 Item	Table 1 Item	Notes
Piping and Fittings Class 1	PB	Small bore piping less than NPS 4 CASS and Wrought Stainless steel	Internal: Treated water	Crack initiation and growth due to SCC, IASCC, thermal and mechanical loading	ASME Section XI Inservice Inspection, (XI.M1) App B B.1.7 One time inspection (XI.M32) App B B.2.11	IV C1.1-i	3.1.1.7	A
Piping and Fittings-Non Class 1	PB	Carbon & Low Alloy steel	Internal: Treated water	Loss of material due to pitting and crevice corrosion	Chemistry Control Program (XI.M2) App. B B.1.4	VIII B2.1-a	3.4.1.7	A
				Wall thinning due to flow-accelerated corrosion	Flow Accelerated Corrosion (XI.M17) App. B B.1.6	VIII B2.1-b	3.4.1.6	A
				Cumulative fatigue damage due to fatigue	TLAA Chapter 4.3.22	VIII B2.1-c	3.4.1.1	A
Valves - Class 1	PB	Carbon & Low Alloy steel	Internal: Treated water	Wall thinning due to FAC	Flow Accelerated Corrosion (XI.M17) App. B B.1.6	IV C1.3-a	3.1.1.25	A.
				Loss of material due to general, crevice and pitting corrosion	Chemistry Control Program (XI.M2) App. B B.1.4	None	None	H, 5
		Stainless Steel (CASS and Wrought)	Internal: Treated water	Crack Initiation/Growth due to stress corrosion cracking (SCC):	Chemistry Control Program (XI.M2) App. B B.1.4	IV C1.3-c	3.1.1.29	A
				Loss of Material due to crevice and pitting corrosion	Chemistry Control Program (XI.M2) App. B B.1.4	None	None	H, 6

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB

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Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	AMP	NUREG -1801 Vol. 2 Item	Table 1 Item	Notes
Valves - Class 1	PB	Carbon & Low Alloy steel, Stainless Steel (CASS and Wrought)	Internal: Treated water	Cumulative fatigue damage due to fatigue	TLAA Chapter 4 .3.14	IV C1.3-d	3.1.1.1	A.

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Note: Text enclosed in bold border is specific to Units 2 and 3 CLB

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	AMP	NUREG -1801 Vol. 2 Item	Table 1 Item	Notes
Valves - Non Class 1	PB	Carbon and Low Alloy Steel	Internal: Treated water	Wall thinning due to FAC	Flow Accelerated Corrosion (XI.M17) App. B, B.1.6	VIII A2-a	3.4.1.6	A, 7
				Loss of material due to general, pitting, and crevice corrosion	Chemistry Control Program (XI.M2) App. B, B.1.4	VIII A2-b	3.4.1.2	A,
			Air/Gas	None	None	None	None	J, 8

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Note: Text enclosed in bold border is specific to Units 2 and 3 CLB

Notes:

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General:

Note A Consistent with NUREG-1801 item for component, material, environment, and aging effect. The AMP is consistent with NUREG-1801.

Note B Consistent with NUREG-1801 item for component, material, environment, and aging effect. The AMP takes some exceptions to NUREG-1801.

Note C Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. The AMP is consistent with NUREG-1801.

Note D Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. The AMP takes some exceptions to NUREG-1801.

Note E Consistent with NUREG-1801 item for material, environment, and aging effect, a different aging management program is credited.

Note F Material not in NUREG-1801 item for this component.

Note G Environment not in NUREG-1801 item for this component and material.

Note H Aging effect not in NUREG-1801 item for this component, material and environment combination.

Note I. Aging effect in NUREG-1801 item for this component, material and environment combination is not applicable.

Note J Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant Specific:

1 This grouping applies to carbon steel instrument piping where it contacts the stainless steel steam lines and flow restrictors. The aging effects identified are consistent with carbon and low alloy steels in treated water results in other NUREG-1801 sections.

2 This grouping applies to the venturi inserts for the main steam line flow restrictors. Management of additional aging effect identified is consistent with cast austenitic stainless steel aging effect results in other NUREG-1801 sections.

3 This material group includes pipe fittings and flow limiting venturis.. The NUREG-1801 Tables do not include stainless steel piping and fittings for the Main Steam Systems, however, the aging effects identified are consistent with stainless steel in treated water results in other NUREG-1801 sections.

4 Additional aging effects identified are consistent with carbon and low alloy steels in treated water results in other NUREG-1801 sections.

5 Additional aging effects identified are consistent with carbon and low alloy steels in treated water results in other NUREG-1801 sections.

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

6 Additional aging effects identified are consistent with stainless steel in treated water results in other NUREG-1801 sections.

7 The only turbine control valves included in the scope of license renewal at BFN are the turbine stop valves. The main steam safety/relief valves listed in this item are included as Class 1 valves in NUREG-1801 Item IV.C1.3.

8 Consistent with industry guidance documents, no aging effects requiring management were identified for carbon and low alloy steel in a control air environment.

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Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

2.3.1.4. Reactor Recirculation System (System 68)

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System Description

The Reactor Recirculation system provides:

- Subcooled water to the reactor core during normal operations to maintain normal core operating temperatures.
- Control of reactor power by varying recirculation flow during normal operations.
- A flow path for Low Pressure Coolant Injection flow from the Residual Heat Removal system to the reactor vessel during design basis accidents.
- A flow path to and from the Residual Heat Removal system for decay heat removal at low temperatures.

The Reactor Recirculation system for each unit shares no components with the other units. The Reactor Recirculation system consists of two piping loops connected to but external to the reactor vessel. The recirculation loop piping is stainless steel that is either IGSCC resistant or has been treated to improve resistance to *IGSCC*. Each loop has a single variable-speed motor-driven pump with pump suction and discharge valves. Each pump takes a suction from the reactor vessel downcomer region and discharges into a manifold that supplies flow to ten jet pumps internal to the reactor vessel. Reactor Recirculation system components internal to the reactor vessel are described and discussed in Section 2.3.1.2, Reactor Vessel Internals. Except for instrumentation piping that penetrates the primary containment, the recirculation system is located inside the primary containment.

The Reactor Recirculation system is in the scope of 10 CFR 54 because it contains components that meet the following criteria of 10 CFR 54.4.

(a)(1)	(a)(2)	(a)(3) FP	(a)(3) EQ	(a)(3) ATWS	(a)(3) SBO
Yes	Yes	Yes	Yes	Yes	No

The Reactor Recirculation system meets 10 CFR 54.4(a)(1) because it is included in the reactor coolant pressure boundary and is relied upon to remain functional during and following design basis events to ensure low pressure coolant injection flow into the core region of the reactor vessel. The Reactor Recirculation system meets 10 CFR 54.4(a)(2) because it contains non-safety-related components that are required to ensure the satisfactory performance of safety related components. The Reactor Recirculation system meets 10 CFR 54.4(a)(3) because it contains components that are relied upon in plant evaluations to perform a function that demonstrates compliance with *10 CFR 50.49*. The Reactor Recirculation system also meets 10 CFR 54.4(a)(3) because it contains components that 1.) ensure removal of reactor residual heat to meet the safe shutdown requirements of *10 CFR 50 Appendix R* during and following fires, and 2.) mitigate the consequences of a failure to scram that meet the anticipated transient without scram requirements of *10 CFR 50.62*.

The portion of the Reactor Recirculation system that contains components requiring an AMR extends from the reactor vessel recirculation outlet nozzle to the reactor vessel

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

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recirculation inlet nozzle. The Reactor Recirculation system components internal to the reactor vessel are presented in Section *2.3.1.2*, Reactor Vessel Internals.

FSAR References

Additional details of the Reactor Recirculation system are contained in FSAR *3.7.6, 4.3, 5.2.3, 7.8, 7.9,* and *7.19* and in the Fire Protection Report Appendix R Safe Shutdown Program, Sections 3 & 4.

License Renewal Drawings

The license renewal drawings for the Main Steam system are listed below.

Unit 1	Unit 2	Unit 3	Shared
1-47E817-1-LR	*2-47E817-1-LR*	*3-47E817-1-LR*	None
1-47E822-1-LR	*2-47E822-1-LR*	*3-47E822-1-LR*	

System Components/Commodities Requiring AMR

Components Subject to AMR

The component types that require aging management review are indicated in Table *2.3.1.4*, Reactor Recirculation System.

The aging management review results for these components are provided in Table *3.1.2.4* – Reactor Coolant Systems - Reactor Recirculation System – Summary of Aging Management Evaluation.

Table 2.3.1.4, Reactor Recirculation System.

Component Type	Intended Function
Bolting	MC
Piping and Fittings	PB
Recirculation Pump Casing	PB
Valves	PB

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

Table 3.1.2.4: Reactor Recirculation System (System 068) Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG -1801 Vol. 2 Item	Table 1 Item	Notes
Bolting	MC	Carbon & Low Alloy Steel	Internal: None External: Drywell atmosphere with metal temperature up to 550°F	Loss of Material/ Wear	Bolting Integrity (XI.M18) App. B, B.2.11	IV C1.3-e IV C1.2-d	3.1.1.26	A
				Loss of pre-load/ stress relaxation	Bolting Integrity (XI.M18) App. B B.2.11	IV C1.3-f IV C1.2.e	3.1.1. 26	A
				Fatigue	TLAA Ch. 4.3.11	IV C1.3-g IV C1.2-f	3.1.1 .26	A
Piping and Fittings	PB	Stainless Steel (e.g., type 304, 316, or 316NG)	Treated water	Crack initiation and growth due to SCC and IGSCC	BWR stress corrosion cracking (M1.M7) App. B, B.1.32 Chemistry Control (XI.M2) App. B, B.1.4	IV C1.1-f	3.1.1.29	A
				Crack initiation and growth due to SCC and IGSCC in small bore piping	ASME Section XI Inservice Inspection, (XI.M1) App B, B.2.11 One time inspection (XI.M32) App B, B.2.11	IV C1.1-i	3.1.1.7	A
Recirculation Pump Casing	PB	CASS	Treated water	*Cumulative fatigue damage*	TLAA Ch. 4.3.86	IV C1.2-a	3.1.1.1	A
				Crack initiation and growth due to SCC and IGSCC	BWR stress corrosion cracking (M1.M7) App. B, B.1.32 Chemistry Control Program (XI.M2) App. B, B.1.4	IV C1.2-b	3.1.1.29	A
				Loss of fracture toughness from thermal aging embrittlement	ASME Section XI Inservice Inspection, (XI.M1) App B, B.1.7	IV C1.2-c	3.1.1.23	A

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Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

Table 3.1.2.4: Reactor Recirculation System (System 068) Summary of Aging Management Evaluation

Component	Intended Function	Material	Environment	Aging Effect Requiring Management	AMP	NUREG -1801 Vol. 2 Item	Table 1 Item	Notes
Valves -	PB	CASS	Treated water	Wall thinning due to FAC	Flow Accelerated Corrosion (XI.M17) App. B, B.1.6	IV.C1.3-a	3.1.1.25	A
				Loss of fracture toughness from thermal aging embrittlement	ASME Section XI Inservice Inspection, (XI.M1) App B, B.1.7	IV C1.3-b	3.1.1.23	A
				Crack initiation and growth due to SCC and IGSCC	BWR stress corrosion cracking (M1.M7) App. B, B.1.32 Chemistry Control Program (XI.M2) App. B, B.1.4	IV C1.3-c	3.1.1.29	A
				Cumulative fatigue damage	TLAA Ch. 4.3.86	IV C1.3-d	3.1.1.1	A

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Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

Table 3.1.2.4: Reactor Recirculation System (System 068) Summary of Aging Management Evaluation

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Table 3.1.2.4 Notes:

General:

Note A Consistent with NUREG-1801 item for component, material, environment, and aging effect. The AMP is consistent with NUREG-1801.

Note B Consistent with NUREG-1801 item for component, material, environment, and aging effect. The AMP takes some exceptions to NUREG-1801.

Note C Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. The AMP is consistent with NUREG-1801.

Note D Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. The AMP takes some exceptions to NUREG-1801.

Note E Consistent with NUREG-1801 item for material, environment, and aging effect, a different aging management program is credited.

Note F Material not in NUREG-1801 item for this component.

Note G Environment not in NUREG-1801 item for this component and material.

Note H Aging effect not in NUREG-1801 item for this component, material and environment combination.

Note I Aging effect in NUREG-1801 item for this component, material and environment combination is not applicable.

Note J Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant Specific

None

2.3.2.4. Residual Heat Removal (System 074)

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System Description

The Residual Heat Removal system functions are:

- Shutdown cooling (residual heat removal with the reactor at low temperature) during normal shutdown operations. The system circulates water from the recirculation pump suction through the RHR heat exchangers back to the recirculation pump discharge into the reactor vessel and through the core for decay heat removal. This mode of operation is manually initiated when required
- Core flooding to limit, in conjunction with the other ECCS systems, the peak fuel clad temperature over the complete spectrum of possible break sizes in the reactor coolant pressure boundary during design basis accidents. In the Low Pressure Coolant Injection mode, the system pumps water from the suppression pool into the reactor vessel through the recirculation pumps discharge line.
- Long term cooling following loss of coolant accidents - Following low pressure coolant injection during a LOCA, cooling is manually initiated using the low pressure coolant injection flowpath through the Residual Heat Removal heat exchanger for long term cooling.
- Primary containment pressure and temperature control. In containment spray mode, the system pumps water from the suppression pool to spray headers in the primary containment to condense steam and control pressure in the primary containment. This mode of operation is manually initiated when required following a LOCA. In suppression pool cooling mode, the system pumps water from the suppression pool through the heat exchangers back to the suppression pool. This mode of operation is manually initiated when required during both normal and accident conditions.
- Containment flooding post accident - Water can be pumped from non-accident unit suppression pools or from the Residual Heat Removal Service Water system (Standby Coolant Supply) to flood the containment. This mode of operation is manually initiated.
- Suppression pool level control - Provisions are provided for both makeup and reject to maintain the suppression pool level within required limits.
- Supplemental fuel pool cooling - Cross-connections with the Fuel Pool Cooling system allow the Residual Heat Removal heat exchangers to be used for spent fuel residual heat removal.

Each unit has two Residual Heat Removal system loops with each loop having two Residual Heat Removal pumps and two Residual Heat Removal heat exchangers. The Residual Heat Removal stainless steel piping on the reactor coolant pressure boundary is either IGSCC resistant or has been treated to improve resistance to *IGSCC*. The pump suction header and heat exchanger discharge header of one loop in Unit 1 and one loop in Unit 2 can be cross-connected. A similar cross-connection is provided between Unit 2 and Unit 3. Provisions are provided to ensure the integrity of the reactor coolant boundary, primary containment, and secondary containment. Major components are located in the Reactor Building.

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

The Residual Heat Removal system is in the scope of 10 CFR 54 because it contains components that meet the following criteria of 10 CFR 54.4.

(a)(1)	**(a)(2)**	**(a)(3) FP**	**(a)(3) EQ**	**(a)(3) ATWS**	**(a)(3) SBO**
Yes	Yes.	Yes	Yes	No	Yes

The Residual Heat Removal system meets 10 CFR 54.4(a)(1) because it is relied upon to remain functional during and following design basis events to remove decay heat from the reactor vessel and to control primary containment pressure and temperature. The Residual Heat Removal system limits off-site radiation dose by maintaining primary containment integrity. In addition, portions of the Residual Heat Removal system form the reactor coolant pressure boundary. The Residual Heat Removal system meets 10 CFR 54.4(a)(2) because it contains non-safety-related components that are required to ensure the satisfactory performance of safety related components. The Residual Heat Removal system meets 10 CFR 54.4(a)(3) because it contains components that are relied upon in plant evaluations to perform a function that demonstrates compliance with *10 CFR 50.49*. The Residual Heat Removal system meets 10 CFR 54.4(a)(3) because it contains components that allow removal of reactor residual heat 1.) to meet the safe shutdown requirements of *10 CFR 50 Appendix R* during and following fires, and 2.) to meet the safe shutdown station blackout requirements of 10 CFR 50.63.

The portion of the Residual Heat Removal system that contains components requiring an AMR extends from the its suctions points (suction strainers in the suppression pool, interconnections with the reactor recirculation system and the fuel pool cooling and cleanup system), through the pumps and heat exchangers to its various discharge points (the torus and drywell spray headers, the suppression pool for cooling and test, the recirculation system for low pressure coolant injection and shutdown cooling, fuel pool cooling and cleanup system for assisted cooling).

FSAR Reference

Additional descriptive information for the Residual Heat Removal system is found in FSAR *4.1, 4.8, 5.2.3, 5.3, 6.4.4, 7.3, 7.4, 7.18, 9.2, 10.5, 10.9, 10.10, 10.17, F.7.15*, and the Fire Protection Report Safe Shutdown Analysis *3.6* and Appendix R Safe Shutdown Program Sections *3* & 4.

License Renewal Drawings

The license renewal drawings for the Residual Heat Removal system are listed below.

Unit 1	**Unit 2**	**Unit 3**	**Shared**
1-47E811-1-LR	*2-47E811-1-LR*	*3-47E811-1-LR*	None

Components Subject to AMR

The component types that require aging management review are indicated in Table *2.3.2.4*, Residual Heat Removal System.

The aging management review results for these components are provided in Table *3.2.2.4* – Engineered Safety Feature Systems - Residual Heat Removal System - Summary of Aging Management Evaluation.

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

Table 2.3.2.4 – Residual Heat Removal System

Component Type	Intended Function
Bolting	MC
Heat exchangers	HT, PB
Nozzles and Orifices	TH, PB
Piping and fittings	PB
Pumps	PB
Valves	PB

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Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

Table 3.2.2.4: Residual Heat Removal System (System 074) Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	AMP	NUREG -1801 Volume 2 Item	Table 1 Item	Notes
Bolting	MC	Carbon & Low Alloy Steel	Internal: None External: Air with metal temperature up to 550°F	Loss of Material due to wear	Bolting Integrity (XI.M18) App. B, B.2.11	IV.C1.3-e V.E.2-a	3.1.1.26 3.2.1.18	A
				Loss of pre-load due to stress relaxation	Bolting Integrity (XI.M18) App. B, B.2.11	IV.C1.3-f V.E.2-b	3.1.1.26 3.2.1.18	A
				Fatigue	TLAA Ch. 4.3.22	IV.C1.3-g	3.1.1.26	A
Heat Exchangers	PB, HT	Carbon & Low Alloy Steel	Primary side: treated water Secondary side: Raw water	Loss of material due to general corrosion, pitting, crevice and MIC corrosion	Open Cycle Closed Cooling Water System (X1.M12) App B, B.2.33	V.D.2.4-a	3.2.1.12	A
				Buildup of deposits due to biofouling	Open Cycle Closed Cooling Water System (X1.M12) App B, B.2.33	V.D.2.4-b	3.2.1.12	A
			Primary side: treated water Secondary side: Closed cycle cooling treated water	Loss of material due to general corrosion, pitting, and crevice corrosion	Closed Cycle Closed Cooling Water System (X1.M21) App B, B.2.34	V.D.2.4-b	3.2.1.13	A
Nozzles and orifices (Drywell and suppression chamber spray)	PB, TH	Carbon & Low Alloy Steel	Drywell Air	Loss of material due to general corrosion	Spray Nozzle Surveillance Program App B, B.2.40	V.D.2.5-a	3.2.1.3	A
				Plugging due to general corrosion	Spray Nozzle Surveillance Program App B, B.2.40	V.D.2.5-b	3.2.1.9	A

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Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

Table 3.2.2.4: Residual Heat Removal System (System 074) Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	AMP	NUREG -1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings	PB	Stainless Steel - CASS	Treated water	Crack initiation and growth due to SCC and IGSCC	BWR stress corrosion cracking (M1.M7) App. B, B.1.32 Chemistry Control Program (XI.M2) App. B, B.1.4	V.D.2.1-c	3.2.1.16	A
				Loss of fracture toughness from thermal aging embrittlement	ASME Section XI Inservice Inspection, (XI.M1) App B, B.1.7 Thermal Aging embrittlement of CASS (X1.M12) App B, B.2.51	IV.C1.1-g V.D2.1-d	3.1.1.24 3.2.1.11	A
				Crack initiation and growth due to SCC and IGSCC (Small Bore)	ASME Section XI Inservice Inspection, (XI.M1) App B, B.1.7 One time inspection (XI.M32) App B, B.2.11	IV.C1.1.i	3.1.1.7	A
		Carbon and low alloy steel	Internal: Treated water	Loss of material due to general, pitting, and crevice corrosion	Chemistry Control Program (XI.M2) App. B, B.1.4 One time inspection (XI.M32) App B B.2.41	V.C.1-b V.D2.1-a V.D2.1-a	3.2.1.6 3.2.1.2 3.2.1.4	A, 1
			External: Air	Loss of material due to general corrosion	Inspection of External *Surfaces Program* App B, B.2.13	V.E.1-b	3.2.1.10	A
Pump Casing	PB	Carbon and low alloy steel	Internal Treated water	Loss of material due to general, pitting, and crevice corrosion	Chemistry Control Program (XI.M2) App. B, B.1.4 One time inspection (XI.M32) App B, B.2.11	V.D2.2-a	3.2.1.2 3.2.1.4	A
			External: Air	Loss of material due to general corrosion	Inspection of External Surfaces Program App B, B.2.61	V.E.1-b	3.2.1.10	A

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

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Table 3.2.2.4: Residual Heat Removal System (System 074) Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	AMP	NUREG -1801 Volume 2 Item	Table 1 Item	Notes
Valves	PB	Stainless Steel - CASS	Treated water	Wall thinning due to FAC	Flow Accelerated Corrosion (XI.M17) App. B, B.1.6	IV.C1.3-a	3.1.1.25	A
				Loss of fracture toughness from thermal aging embrittlement	ASME Section XI Inservice Inspection, (XI.M1) App B, B.1.7 Thermal Aging embrittlement of CASS (X1.M12) App B, B.2.51	IV.C1.3-b	3.1.1.23	A
				Crack initiation and growth due to SCC and IGSCC	BWR stress corrosion cracking (M1.M7) App. B, B.1.32 Chemistry Control Program (XI.M2) App. B, B.1.4	IV.C1.3-c V.D2.3-c	3.1.1.29 3.2.1.16	A
				Cumulative fatigue damage	TLAA Ch. 4.6.17.	IV.C1.3-d	3.1.1.1	A
		Carbon and low allow Steel	Internal: treated water	Loss of material due to general, pitting, and crevice corrosion	Chemistry Control Program (XI.M2) App. B B.1.4 One time inspection (XI.M32) App B, B.2.11	V.C.1-b V.D2.1-a V.D2.1-a	3.2.1.6 3.2.1.2 3.2.1.4	A
			External: air	Loss of material due to general corrosion	Inspection of External Surfaces Program App B, B.2.61	V.E.1-b	3.2.1.10	A

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Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

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Notes:

General:

Note A Consistent with NUREG-1801 item for component, material, environment, and aging effect. The AMP is consistent with NUREG-1801.

Note B Consistent with NUREG-1801 item for component, material, environment, and aging effect. The AMP takes some exceptions to NUREG-1801.

Note C Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. The AMP is consistent with NUREG-1801.

Note D Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. The AMP takes some exceptions to NUREG-1801.

Note E Consistent with NUREG-1801 item for material, environment, and aging effect, a different aging management program is credited.

Note F Material not in NUREG-1801 item for this component.

Note G Environment not in NUREG-1801 item for this component and material.

Note H Aging effect not in NUREG-1801 item for this component, material and environment combination.

Note I. Aging effect in NUREG-1801 item for this component, material and environment combination is not applicable.

Note J Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant Specific:

1. For NUREG-1801 item V.C.1.b, the only inside surface environment applicable to the RHR system is treated water

4.2 Neutron Irradiation Embrittlement

Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to ensure adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. For the RPV, Appendix G to 10 CFR Part 50 requires an evaluation of the Charpy Upper Shelf Energy (USE) and an evaluation of the Adjusted Reference Temperature (ART) to determine pressure-temperature (P-T) limits for the RPV. Neutron irradiation causes a decrease in the Charpy USE and an increase in the ART of the RPV beltline materials.

This section presents TVA's evaluation of the impact of irradiation during the period of extended operation on the Charpy USE for the BFNP reactor vessels. In addition, this section presents TVA's evaluation of the impact of irradiation during the period of extended operation on the BFNP RPV-temperature limits.

A calculation(Ref. 1) to determine the RPV ID neutron flux using the approved GE flux calculation methodology (Ref. 2) has been performed for the extended power uprate thermal power limit of 3952 MW_t. The results of the calculation are that the peak RPV ID neutron flux is $1.4e^9$ n/sec-cm^2. The axial and azimuthal (at core midplane) flux distributions were also obtained. *

The flux calculation results were used as the input to the revised ART, Charpy USE, and the P-T limits. In addition, the calculations will assume that, at the end of the period of extended operation (60 calendar years), the end of life effective full power years (EFPY) will be 52. This determination is based on assumptions that the capacity factor will be 90% and the rated power level will be 3952 MW_t for Unit 1 after [Unit 1 specific number] EFPPY, Unit 2 after 18.1 EFPY, and Unit 3 after 13 EFPY. The rated power level assumed for the units until implementing the power uprate is the currently licensed power level of 3458 MW_t.

4.2.1 RTNDT

The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RTNDT), the mean value of the adjustment in reference temperature caused by irradiation (ΔRTNDT), and a margin (M) term. The ΔRTNDT is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RTNDT is a plant-specific or a generic value and whether the chemistry factor (CF) was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RTNDT , the copper and nickel contents, the fluence, and the calculation methods. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.
The 52 EFPY ART for the limiting beltline material for Unit 1 is [Unit 1 specific number]. The 52 EFPYs ART for the limiting beltline material for Unit 2 (Heat C2463-1) at 1/4T

has been determined to be 157.4 °F. The 52 EFPY ART for the limiting beltline material for Unit 3 (Shell # 2, Heat C3222-2) at 1/4T is 157.4 °F. These values for ART are based on a neutron fluence value of 1.6E18 n/cm^2, the initial RTNDT values of 23.1°F for the units, the limiting Cu content of 0.24%, and the limiting Ni content of 0.37% for the units.

4.2.2 Charpy Upper Shelf Energy

Section IV.A.1a of Appendix G to 10 CFR Part 50 requires, in part, that the RPV beltline materials have Charpy USE in the transverse direction for base metal and along the weld for weld material of no less than 50 ft-lb (68J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will ensure margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

By letter dated April 30, 1993, the Boiling Water Reactor Owners Group (BWROG) submitted a topical report entitled "10 CFR Part 50 Appendix G Equivalent Margins Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," to demonstrate that BWR RPVs could meet margins of safety against fracture equivalent to those required by Appendix G of the ASME Code Section XI for Charpy USE values less than 50 ft-lb. In a letter dated December 8, 1993, the staff concluded that the topical report demonstrated that the evaluated materials have the margins of safety against fracture equivalent to Appendix G of ASME Code Section XI, in accordance with Appendix G of 10 CFR Part 50. In this report, the BWROG derived through statistical analysis to derive the unirradiated USE values for materials that originally did not have documented unirradiated Charpy USE values. Using these statistically derived Charpy USE values, the BWROG predicted the end-of life (40 years of operation) USE values in accordance with RG 1.99, Rev. 2. According to this RG, the decrease in USE is dependent upon the amount of copper in the material and the neutron fluence predicted for the material. The BWROG analysis determined that the minimum allowable Charpy USE in the transverse direction for base metal and along the weld for weld metal was 35 ft-lb.

General Electric (GE) performed an update to the USE equivalent margins analysis, which is documented in EPRI TR-113596, "BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," BWRVIP-74, September 1999. The staff review and approval of EPRI TR-113596 is documented in a letter from C. I. Grimes to C. Terry dated October 18, 2001. The analysis in EPRI TR-113596 determined the reduction in the unirradiated Charpy USE resulting from neutron radiation using the methodology in RG 1.99, Revision 2. Using this methodology and a correction factor of 65% for conversion of the longitudinal properties to transverse properties, the lowest irradiated Charpy USE at 54 EFPYs for all BWR/3-6 plates is projected to be 45 ft-lb. The correction factor for specimen orientation in plates is based on NRC Branch Technical position MTEB 5-2. Using the RG methodology, the lowest irradiated Charpy USE at 54 EFPY for BWR non-Linde 80 submerged arc welds is projected to be 43 ft-lb. EPRI TR-113596 indicates that the percent reduction in Charpy USE for the limiting BWR/3-6 beltline plates and BWR non-Linde 80 submerged arc welds are 23.5% and 39%, respectively. Since this is a generic analysis, TVA has performed a plant-specific analysis to demonstrate that the

beltline materials of the BFNP Units 1, 2, and 3 RPVs meet the criteria in the report at the end of the license renewal period.

The BFNP analysis determined the predicted percent decrease of the beltline material USE values at 1/4T and 54 EFPYs was estimated using BWRVIP-74 and RG 1.99, Revision 2. The equivalent margin analysis was performed using information presented in Tables B-4 and B-5 of EPRI TR-113596. RG 1.99, Revision 2, predicted percent decrease in USE for the limiting beltline plate material at the end of the license renewal period is [Unit 1 specific number] for Unit 1, 17% for Unit 2, and 16% for Unit 3. Both predicted values of USE are less than the generic value of 23.5% reported in EPRI TR-113596. Similarly, the RG 1.99, Revision 2, predicted percent decrease in USE for limiting weld material (Electroslag Weld at both units) at the end of license renewal period is [Unit 1 specific number] for Unit 1 and 25.5% for both Unit 2 and Unit 3, which is less than the generic value of 39% reported in EPRI TR-113596. The 52 EFPYs neutron fluence at 1/4T for the limiting beltline plate and weld materials of the units is 1.6E18 n/cm 2. The Cu contents for the limiting beltline materials are 0.24 wt% for the units.

While the margin has decreased, the BFNP analysis results are acceptable because the percent decrease in USE for limiting plate and weld materials at BFNP Units 1, 2, and 3 is bounded by the corresponding generic results obtained by the equivalent margin analysis presented in EPRI TR-113596 as mentioned above. Therefore, the Charpy USE values at 52 EFPY for the limiting plate and weld materials at BFN Units 1, 2, and 3 are greater than the minimum allowable value of 35 ft-lb, which demonstrates that the evaluated materials have the margins of safety against fracture equivalent to Appendix G of Section XI of the ASME Code, in accordance with Appendix G of 10 CFR Part 50, throughout the license renewal period.

4.2.3 Pressure/Temperature Limits

BFN Units 1, 2, and 3 Technical Specifications, Section 3.4.9, contain P/T limit curves for heatup, cooldown, criticality, and inservice leakage and hydrostatic testing. The curves are currently calculated for the operating periods ending with 12 EFPY, 17.2 EFPY, and 13.1 EFPY for Units 1, 2, and 3 respectively. The P/T Limit curves are calculated using Code Case N-640.

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

Analysis and Conclusion

The P-T limit curves developed for this TLAA evaluation are based on the following NRC regulations and guidance:

- 10 CFR Part 50, Appendix G; Generic Letter
- (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations";
- GL 92-01, "Reactor Vessel Structural Integrity, " Revision 1; GL 92-01, Revision 1, Supplement 1;
- RG 1.99, Revision 2
- Standard Review Plan (SRP) Section 5.3.2, "Pressure-Temperature Limits and Pressurized Thermal Shock."

Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by the methodology of Appendix G Section XI of the ASME Code. The ASME Code Appendix G methodology requires that applicants determine the ART at the end of the operating period.

TVA has performed P/T limit curve calculations for the period of extended operation using methodologies base on RG 1.99 Revision 2. Use of RG 1.99 requires that an allowance for margin be included in the bounding ART value. This ensures that adequate safety margins are maintained. The calculations show that the RPVs will be in compliance with regulatory requirements and adequate safety margins can be maintained during the period of extended operation.

Therefore, operation of the BFNP RPVs to 52 EFPY (60 calendar years) will not have an adverse affect on reactor vessel fracture toughness.

Disposition: - 10 CFR 54.21(c)(1)(ii)

Amendments to the Technical Specifications to revise the reactor vessel P/T limit curves will be requested and implemented as current P/T curves reach their operational limits.

B.2.1.12 XI.M17 FLOW-ACCELERATED CORROSION (FAC)

Program Description

The FAC aging management program is an existing program implemented by a administrative procedures that are applicable to all three units. The program is based on the EPRI guidelines in NSAC-202L-R2. The program predicts, detects, and monitors wall thinning in piping, fittings, and valve bodies due to FAC in the following systems: Main Steam (System 001), Condensate (System 002), Reactor Feedwater (System 003), Extraction Steam (System 05), Heater Drains and Vents (System 06), Aux. Boiler (System 12).

Program activities include analyses using the predictive CHECWORKS computer code to determine critical locations, baseline inspections to determine the extent of thinning at these critical locations, and follow-up inspections to confirm the predictions. Repairs and replacements are performed as necessary.

Exceptions to NUREG-1801

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None

NUREG-1801 Consistency

With enhancements the FAC AMP is consistent with the ten elements of aging management program XI.M17, "Flow-Accelerated Corrosion," specified in NUREG-1801.

Enhancements

One enhancement to implement the program on Unit 1 is required. This enhancement affects the following program element:

- Scope

  The scope of the program should include the specific structures and components of which the program manages the aging.

The enhancement is scheduled for completion prior to the fuel load of Unit 1 prior to startup from its current outage.

Operating Experience

Industry

Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC IE Bulletin No. 87-01 and NRC Information Notices (INs) 81-28, 92-35, and 95-11), in two-phase piping in extraction steam lines (NRC INs 89-53 and 97-84), and in moisture separator reheater and feedwater heater drains (NRC INs 89-53, 91-18, 93-21, and 97-84).

Browns Ferry

The TVA experience with its flow-accelerated corrosion aging management program activities has shown that the program can determine susceptible locations for flow-accelerated corrosion, predict the component degradation, and detect the wall thinning in piping and valves due to flow accelerated corrosion. In addition, the program provides for reevaluation, repair or replacement for locations where calculations indicate an area will reach minimum allowable thickness before the next inspection. When FAC problems have been identified, corrective actions have been taken to prevent recurrence. For example:

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

Extraction steam, heater drain, and heater vent lines have experienced wall thinning on Units 2 and 3 due to FAC and this piping has been replaced.

Conclusion

Based on the use of industry guidelines, NRC requirements, and BFN operating experience, there is reasonable assurance that the BFN FAC program will continue to adequately manage the aging effects due to flow accelerated corrosion such that the piping and components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

Note: Text enclosed in bold border is specific to Units 2 and 3 CLB.

APPENDIX F

INTEGRATION OF BROWNS FERRY UNIT 1 RESTART AND LICENSE RENEWAL ACTIVITIES

F.0 Introduction

HISTORY

In 1985, TVA shut down all three units at BFN to address management and technical issues. With the resolution of management issues and Unit 2 technical issues, Unit 2 was restarted in 1991. Unit 3 was restarted in 1995. Consistent with Unit 2 and 3, TVA has committed not to restart that unit without prior approval from the NRC. With the exception of Unit 1 systems and components that are required to be in-service to support Unit 1, or to support the operation of Units 2 and 3, Unit 1 has remained shutdown with key systems and components placed in layup. TVA has initiated a restart plan to return Unit 1 to service.

OVERVIEW OF THE RESTART PLAN

The basic TVA principle for the Unit 1 Restart is that all three BFN units will be operationally identical upon completion of Unit 1 restart activities.

To meet this principle, TVA plans for the Unit 1 current licensing basis (CLB) at restart to be identical to the CLB of Units 2 and 3. The starting point for development of design and programmatic changes required for Unit 1 restart began with the changes required for the restart of Unit 3 and changes implemented or planned for Units 2 and 3 since their restarts. Technological improvements and regulatory changes since implementation of the Unit 3 restart have also been considered in the development of the Unit 1 restart plan.

The intent of the restart plan is to maximize the reliability of Unit 1 for the duration of a combined current and renewed operating license term. To that end, the restart plan incorporates extensive activities to replace, upgrade, and refurbish of components to minimize future aging effects and to implement lessons-learned from Units 2 and 3 operating experience.

The restart plan ensures compliance with TVA commitments made during the shutdown and with regulatory requirements that changed during the extended shutdown.

RELATIONSHIP OF THE RESTART PROGRAM TO LICENSE RENEWAL

The Unit 1 restart program will result in three operationally identical BFN units, providing assurance that the Unit 1 CLB changes implemented prior to restart will result in the same aging management programs (AMPs) for each unit.

BFN has a single FSAR. The License Renewal FSAR Supplement, Appendix A, identifies and describes the AMPs that are required for all three units. No unique AMPs are required for Unit 1.

The BFN procedures for AMPs are applicable site-wide. BFN procedures for new AMPs and existing AMP enhancements will be issued in accordance with the license conditions and commitments associated with the renewed licenses.

F.0 Introduction, continued

To assure the effective integration of restart plan activities with license renewal activities, TVA has established programs and processes to ensure license renewal requirements are considered by the design process:

1. BFN site engineering personnel incorporate license renewal results into design changes. TVA procedures require consideration of license renewal results during the design change process.

2. BFN license renewal project personnel evaluate planned or implemented BFN design changes and existing BFN AMP changes to incorporate these changes into license renewal evaluations.

TVA is required to submit updates to the LRA prior to the issuance of the renewed license in accordance with 10 CFR 54.21(b). If plant licensing or design basis changes are implemented during the NRC review of the BFN LRA, this update will include changes to Appendix F to reflect the Unit 1 CLB and an updated plan for the resolution of any unit differences.

DESCRIPTION OF INFORMATION PRESENTED IN APPENDIX F

This Appendix contains no new commitments. It provides TVA's plans and current schedules for BFN LRA activities affected by the Unit 1 restart.

Whenever text enclosed in a bold border appears in the LRA symbolizing a licensing or design basis that is only applicable to Units 2 and 3, a link is provided to the appropriate Appendix F section.

Appendix F summarizes the resolution of the difference as it pertains to Unit 1 and its impact on the application. For each difference, the following information is presented:

- Description – A detailed description of the difference. Links are provided to source documents if they have been included on the electronic submittal.

- Difference Resolution – This includes an explanation of the methodologies and activities that TVA plans to use to employ to disposition each licensing or design basis difference.

- LRA Impact -This summarizes changes that would be expected to the LRA, if the condition were resolved prior to issuance of the renewed licenses. Links are provided to commitments made in the LRA related to the resolution activities and schedule.

F.0 Introduction, continued

- Schedule for completion – In general, this will be related to milestones rather than specific dates. The following milestones are defined:
  - Prior to renewed license issuance – TVA expects the resolution activities to be complete prior to the expected issuance date of the renewed licenses (23 to 25 months from the submittal date)
  - Prior to restart – TVA will complete the resolution activities prior to Unit 1 restart.
  - Permanent - The difference is acceptable as is for license renewal. No changes related to license renewal are necessary or are planned for the condition.
  - If a submittal is required, the expected dated of the submittal will be stated.
- Systems Impacted – The impacted systems (or component types) are identified with links to the appropriate Chapter 2 sections and the appropriate Chapter 3 sections.
- AMPs/TLAAs Impacted - The impacted AMP and TLAAs are identified with links to the appropriate Chapter 4 sections and the appropriate Appendix B sections.
- References - Unit 1 and Units 2 and 3 documents as appropriate.

F.1 Main Steam Isolation Valve (MSIV) Alternate Leakage Treatment

DESCRIPTION

The current licensing basis for MSIV leakage does not incorporate an alternate leakage treatment pathway utilizing main steam system piping and the main condenser. The application of this methodology has been included in the Units 2 and 3 licensing basis as described in References 2 and 3.

RESOLUTION

The Unit 1 main steam piping from the outermost isolation valve up to the turbine stop valve, the bypass/drain piping to the main condenser and the main condenser is being evaluated and modified as required to ensure the structural integrity is retained during and following a safe-shutdown earthquake (SSE). This will allow use of methodology that assumes plateout and holdup in the piping and condenser in LOCA offsite and control room dose calculations for radioactive leakage past the main steam isolation valves.

This issue will be resolved by the approval of Unit 1 Technical Specification Change TS-436, *Main Steam Isolation Valve Leakage Rate Limits and Exemption from 10 CFR 50, Appendix J,* by NRC.

LRA IMPACTS

The only change to the LRA if this activity were to be completed during the NRC review of the LRA would be to remove the boxes indicating the Unit 1 CLB difference in the scoping and screening results and the AMR Tables for affected systems. LR Drawings for impacted systems will be changed to reflect the extension of the portions of the Unit 1 systems that contain components subject to an AMR from the outboard MSIVs to the turbine stop valves and to the condenser.

The component types and environments for the components added to Unit 1 will be the same as already evaluated for Units 2&3.

SCHEDULE

Unit 1 Restart Activity	Scheduled Completion
Submittal of TS-436	Q2, 2004
Program Implementation on Unit 1	N/A
Physical Modification Completion	Q2, 2006

UNIT 1 SYSTEMS IMPACTED

Main Steam (2.3.4.1, Table 3.4.2.1)	Condensate (2.3.4.2, Table 3.4.2.2)	Turbine Drains and Miscellaneous Piping (2.3.4.5, Table 3.4.2.5)	Heater Drains and Vents (2.3.4.4,Table 3.4.2.4)

AMP / TLAA IMPACTS

None

REFERENCES

1. FSAR Chapter *14.6*

2. TVA letter to NRC dated March 14, 2000, Brown Ferry Nuclear Plant, Units 2 And 3 - Issuance Of Amendments Regarding Limits On Main Steam Isolation Valve Leakage (TAC Nos. MA6405 And MA6406)

3. NRC letter to TVA dated September 28, 1999, Browns Ferry Nuclear Plant (BFN) - Units 2 And 3 – Technical Specifications (TS) Change 399 - Increased Main Steam Isolation Valve (MSIV) Leakage Rate Limits And Exemption From 10 CFR 50 Appendix J

F.2 Flow Accelerated Corrosion

DESCRIPTION

A site-wide Flow Accelerated Corrosion Program (FAC) in response to GL 89-08 has been developed for all three BFN units. FAC inspection and examination activities have not been performed on Unit 1 during the extended shutdown period. The Unit 1 restart plan includes the replacement of FAC susceptible piping; therefore, FAC inspections will not be required prior to Unit 1 restart.

RESOLUTION

The LRA AMP description contains a commitment to implement the site FAC AMP procedures on Unit 1 prior to restart.

Replacement of all Unit 1 FAC susceptible piping with FAC resistant piping is underway. The component type of the replacement piping will be of the same component type and in the same environments already evaluated for Units 2&3.

IMPACT ON LRA

No changes to the description of the FAC AMP presented in the LRA will be required.

SCHEDULE

Unit 1 Restart Activity	**Scheduled completion**
Analyses	N/A
Submittal	None
Program Implementation on Unit 1	Q2, 2006
Physical Modification Completion	Q2, 2006

UNIT 1 SYSTEMS IMPACTED

Main Steam (Table 3.4.2.1)	Feedwater (Table 3.4.2.3)	Reactor Vessel Vents and Drains (Table 3.1.2.3)	Heater Drains and Vents (Table 3.4.2.4)	HPCI (Table 3.2.2.3)
RCIC (Table 3.3.2.23)	Additional systems (later)			

AMP / TLAA IMPACTED

BFN Flow Accelerated Corrosion (*B.2.1.15*)
BFN License Renewal Application, Appendix A, Updated Final Safety Analysis Supplement (A.*1.15*)

REFERENCES

Generic Letter 89-08, Erosion/Corrosion-Induced Pipe Wall Thinning

F.3 Fire Protection

DESCRIPTION

TVA is required by 10 CFR 50 Appendix R to have the capability to maintain safe shutdown during and after fires. The TVA Fire Protection Report Vol. 1 (incorporated by reference into FSAR Chapter *10.11*) states that the Appendix R requirements for operating units have only been established and implemented for Units 2 and 3.

RESOLUTION

The BFN Fire Protection Program to ensure the capability to maintain safe shutdown during and after fires will be revised on Unit 1 to ensure compliance with 10 CFR 50 Appendix R. The BFN FPR will be revised in accordance with Unit 1 License Condition 2.C.13.

IMPACT ON LRA

This will require that impacted Unit 1 systems be included in scope for criterion 10 CFR 54.4(a)(3). There will be no impact on SSC scoping, screening, or AMPs since the existing SSCs required to meet 10 CFR 54(a)(3) for fire protection are in scope for license renewal as required by other scoping criteria.

Component types added to the plant are will be the same as those currently evaluated for Units 2 and 3 in the same environments.

SCHEDULE

Unit 1 Restart Activity	**Scheduled Completion**
Analyses	1Q, 2004
Submittal	None
Program Implementation on Unit 1	Q2, 2006
Physical Modification Completion	Q2, 2006

UNIT 1 SYSTEMS IMPACTED

MS (2.3.4.1)	RCIC (2.3.3.23)	HPCI (2.3.2.3)	Condensate (2.3.4.2)	RHR (2.3.2.4)
Additional systems (Later)				

AMP / TLAA IMPACTED

None

F.3 Fire Protection, continued

REFERENCES

1. BFN FSAR Chapter *10.11*, Fire Protection Systems
2. BFN Fire Protection Report, Vol. 1.
3. 10 CFR 50, Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

F.5 Environmental Qualification

DESCRIPTION

A site-wide Environmental Qualification Program required by 10 CFR 50.49 has been developed for BFN, has been implemented on Units 2 and 3, and will be implemented on Unit 1.

RESOLUTION

The BFN EQ Program will be implemented on Unit 1 to ensure compliance with 10 CFR 50.49. If the renewed license for Unit 1 is issued prior to the implementation of 10 CFR 50.49 on Unit 1, consideration must be given to the plant operating license of 60 years from the date of issuance of the original license when the program is implemented.

IMPACT ON LRA

This will require that affected systems be included in scope for criterion 10 CFR 54.4(a)(3). There will be no impact on SSC scoping, screening, or AMR results since the SSCs required to meet 10 CFR 54.4(a)(3) for EQ will be in scope for other reasons.
There will be no impact on the AMP.

Component types for Unit 1 are expected to be the same as those currently evaluated for Units 2 and 3 in the same or similar environments.

SCHEDULE

Unit 1 Restart Activity	**Scheduled completion**
Analyses	Complete
Submittal	N/A
Program Implementation on Unit 1	Q2, 2006
Physical Modification Completion	Q2, 2006

UNIT 1 SYSTEMS IMPACTED

MS (3.4.2.1)	HPCI (3.2.2.3)	RCIC (3.3.2.23)	RHR (3.2.2.4)	RHRSW (3.4.2.3)
Additional systems (later)				

AMP / TLAA IMPACTED

BFN Environmental Qualification Program
(TLAA *4.4*, App. *A.3.3*, App. *B.3.1*)

REFERENCES

10 CFR 50.49, Environmental Qualification Of Electric Equipment Important To Safety For Nuclear Power Plants.

F.5 Inter-Granular Stainless Steel Stress Corrosion Cracking (IGSCC)

DESCRIPTION

TVA has submitted and implemented plans for addressing IGSCC in accordance with Generic Letter 88-01 and Supplement 1 for Units 2 and 3. In accordance with the Unit 1 restart plan, IGSCC susceptible piping is being replaced.

RESOLUTION

To comply with GL 88-01, TVA will implement its plans for addressing IGSCC. Susceptible piping and other components will be replaced with IGSCC resistant materials to provide the greatest degree of assurance against future cracking problems. The material planned for Unit 1 is the same as material already used in Units 2 and 3 in the same environment.

IMPACT ON LRA

No changes are expected to the scoping and screening results, AMRs, or AMPs presented in the LRA.

The replacement components are expected to be of the same material in the same environments as already evaluated in the LRA.

SCHEDULE

Unit 1 Restart Activity	**Scheduled completion**
Analyses	Complete
Submittal	Prior to Restart
Program Implementation on Unit 1	N/A
Physical Modification Completion	Q3, 2006

UNIT 1 SYSTEMS IMPACTED

Reactor Vessel (2.3.1.1)	Reactor Recirculation (2.3.1.4)	RWCU (2.3.3.21)	Core Spray (2.3.1.5)	Residual heat Removal (2.3.1.4)

AMP / TLAA IMPACTED

None

REFERENCES

1. Generic Letter 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, dated Jan. 25, 1988 and Supplement 1 dated February 24, 1992

F.6 Reactor Vessel Pressure-Temperature (P-T) Curves

DESCRIPTION

The current P-T curves for Units 2 and 3 utilize ASME B&PV Code Case N-640. Code Case N-640 had not been issued when the current Unit 1 P-T curves were approved by NRC.

RESOLUTION

When required, the TS 3.4.9 P-T curves will be updated for Unit 1. The methodologies that will be used for that update will comply with the Unit 1 TS and 10 CFR 50, Appendices G and H. This will ensure that the latest guidance for complying with 10 CFR 50 Appendices G and H is considered.

IMPACT ON THE LICENSE RENEWAL APPLICATION

None

SCHEDULE

Unit 1 Restart Activity	**Scheduled completion**
Analyses	Prior to restart
Submittal	None required for license renewal
Program Implementation on Unit 1	Permanent
Physical Modification Completion	None

UNIT 1 SYSTEMS IMPACTED

Reactor Vessel *(2.3.1.1)*				

AMP / TLAA IMPACT

Neutron Embrittlement *(4.2)*

REFERENCES

1. NRC Letter, "Browns Ferry Nuclear Plant, Units 2 And 3 - Issuance Of Amendments Regarding The Pressure-Temperature Limits For The Reactor Pressure Vessel (TAC Nos. MB2753 And MB2754)" dated March 28, 2002.
2. Regulatory Guide 1.99, Revision 2, ""Radiation Embrittlement of Reactor Vessel Materials"

F.7 Anticipated Transient Without Scram

DESCRIPTION

10 CFR 50.62 requires licensees to reduce the risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants. FSAR Chapter *7.19* describes the BFN design features that are required to ensure compliance with 10 CFR 50.62. Technical Specification 3.3.4.2 provides the requirements for the Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation. Technical Specification 3.1.7, Standby Liquid Control (SLC) System, provides requirements for ATWS that satisfy 10 CFR 50.62.

These features have been installed on Units 2 and 3 and will be installed on Unit 1. The impacted Unit 1 systems (Main Steam, CRD Hydraulic, and Standby Liquid Control) are currently not in scope for 10 CFR 54.4(a)(3) for ATWS.

RESOLUTION

Design features described in FSAR Chapter *7.19* will be installed on Unit 1.

IMPACT ON THE LICENSE RENEWAL APPLICATION

The LRA scoping reason for the impacted systems will be changed to reflect their ATWS intended function. However, they are already in scope for other reasons. So there will be no change to scoping results. There will be no change to the screening results presented in the LRA since the added components for the recirculation pump trip and the alternate rod insert functions will be electrical components of the same commodity types as those installed for Units 2 and 3. Similarly, there will be no change to the AMR and AMP results for the impacted systems.

SCHEDULE

Unit 1 Restart Activity	Scheduled completion
Analyses	Q1, 2005
Submittal	N/A
Program Implementation on Unit 1	N/A
Physical Modification Completion	Q4, 2006

UNIT 1 SYSTEMS IMPACTED

Reactor Recirculation *(2.3.1.4)*	SLC *(2.3.3.18)*	CRD Hydraulic *(2.3.3.29)*		

F.7 Anticipated Transient Without Scram, continued

AMP / TLAA IMPACTED

None

REFERENCES

1. 10 CFR 50.62, Requirements For Reduction Of Risk From Anticipated Transients Without Scram (ATWS) Events For Light-Water-Cooled Nuclear Power Plants.
2. BFN FSAR *7.19*, Anticipated Transients Without Scram