UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

April 25, 1995

MEMORANDUM TO: Stewart D. Ebneter, Regional Administrator

FROM: Jon R. Johnson, Deputy Director [signature] FOR
Division of Reactor Projects

SUBJECT: MINUTES OF BROWNS FERRY 3 RESTART PANEL MEETING
APRIL 18, 1995

The Browns Ferry Unit 3 Restart Panel met in the Region II offices on April 18, 1995, to review the status of NRC and TVA activities for the restart of this unit. The next meeting of the NRC panel will be held in the Resident Inspector's offices at the Browns Ferry Nuclear Plant on May ~~24~~ 25th, 1995, at 9:00 a.m., CDT and the followup meeting with the licensee will be in the Administrative Building at Browns Ferry from 12:30 p.m., to 2:00 p.m., CDT. Meeting minutes are attached as Enclosure 1. A Unit 3 Task Checklist is provided as Appendix A and a Unit 3 Issues Checklist is provided as Appendix B. Appendix C is an executive Summary of the Browns Ferry Multi-Unit PRA.

Attachment: Browns Ferry Unit 3 Restart
Panel Meeting Minutes
w/Appendix A, B, and C

cc w/att:
L. A. Reyes, RII
E. W. Merschoff, RII
J. R. Johnson, RII
A. F. Gibson, RII
J. P. Stohr, RII
W. E. Cline, RII
T. A. Peebles, RII
C. A. Casto, RII
J. J. Blake, RII
M. B. Shymlock, RII
T. R. Decker, RII
W. H. Rankin, RII

cc w/atts: (Continued on page 2)

Attachment 4

cc w/atts: (Continued)
P. J.Kellogg, RII
M. S. Lesser, RII
B. Uryc, RII
J. W. York, RII
L. D. Wert, RII
K. P. Barr, RII
R. P. Zimmerman, NRR
S. A. Varga, NRR
G. M. Tracy, EDO
F. J. Hebdon, NRR
J. F. Williams, NRR
P. S. Koltay, NRR

BROWNS FERRY UNIT 3 RESTART PANEL MEETING MINUTES

APRIL 18, 1995

Meeting Date:	April 18, 1995
Meeting Location:	Region II Office
Members Present:	J. R. Johnson, Chairman, RII M. S. Lesser, RII C. A. Casto, RII W. E. Cline, RII K. P. Barr, RII J. F. Williams, NRR L. D. Wert, SRI

Summary:

The chairman reviewed the minutes of the previous meeting and the status and results of the assigned action items. The task checklist and selected items from the issues check list were discussed and updated checklists are provided as Appendix A and B. Appendix C is a copy of the executive summary for the Browns Ferry Multi-Unit PRA. The panel chairman announced that K. Barr will be replacing W. Cline on the panel.

Previously assigned actions:

1. NRR (Hebdon) Schedule ORAT.

   (Closed) Peter Koltay has been identified as the ORAT team leader. The inspection has been entered in the MIP for the end of October. J. Williams will invite P. Koltay to the next panel meeting.

2. DRP (Lesser) Arrange for a special (separate) meeting for public comment on the restart of BF3 (September).

   (Open) No action taken on this item yet.

3. DRP (Wert) Give a status of the number of area turnovers.

   (Closed) A list of the number of area turnovers was given. However, it was pointed out that at Browns Ferry, the licensee does not use this information for engineering or technical purposes (Watts Bar does) but uses the information for housekeeping purposes. The weekly status report will track the number of housekeeping areas turned over.

Attachment

4. DRP (Lesser) Schedule Ron Gibbs to perform Module 38703 for replacement components and parts for BF3.

   (Open) Arranged with R. Gibbs and management to perform this inspection and will place this on the MIPS for the week of July 10, 1995.

5. DRP (Lesser) Copy of each of our meeting minutes to Peter Koltay to aid him in scheduling the ORAT inspection.

   (Closed) York put P. Koltay on the panel distribution list.

6. DRP (Johnson) Arrange for a discussion of two unit operation with the licensee.

   (Open) The licensee will be requested to discuss this at the next meeting.

7. DRP (Lesser) Add CATD closeout letter to issues checklist.

   (Closed) We have added this to the Issues Checklist.

8. NRR (Hebdon) Discuss with NRR reviewer the possibility for finishing his review of licensee's Appendix R submittal sooner so that inspection can be performed sooner than July.

   (Closed) The NRR reviewer can't finish the review early and Casto will add this team inspection to the inspection schedule for early July.

9. DRP (Lesser/Wert) Compare TIs 2512/015 and 2515/074 (employee concerns) to ensure that all applicable points for both are covered for BF3.

   (Closed) TI 2515/074 was used to inspect employee concerns for both the Sequoyah and the Browns Ferry 2 startups and a number of inspections have been completed for the Browns Ferry 3 startup (with no apparent problems revealed). An inspection of seismic CATDs is to be scheduled by Casto using R. Chou. The review of TI 2512/015 revealed that it was only applicable to Watts Bar

10. DRSS (Barr/Decker) Schedule George Kuzo (because of his familiarity with the Watts Bar problems) to inspect or help Dan Jones with PASS or line sampling.

    (Open) This action is still being formulated, however the scope should include readiness of radiological instrumentation.

11. DRP (Lesser) Add four items from NPP identified by the SRI to the list. Also add Beta tape problem.

    (Closed) The items have been added.

12. DRS (Casto) Discuss the restart test program next panel meeting.

    (Open) Casto will assign someone to look into but cannot do this until first two weeks in June since the licensee is behind on procedures.

13. DRS (Casto/Peebles) Add inspections/ dates for inspection in EOPs, procedures, maintenance, TSs, etc. to the Master Inspection Plan.

    (Open) Still have to compile a list from the Operations Branch.

14. DRS (Casto) Check with Lenahan on a NRR Memo which apparently approved CONAN computer code and may provide information to close IFI 94-12-01.

    (Closed) Still have to close IFI, but the inspectors have the necessary information aand IFI is on the Issues Checklist.

Newly Assigned Action Items:

1. DRP (Lesser/Uryc) Discuss the status of DOL cases at Browns Ferry.

2. DRS (J. York) Distribute the executive summary for the Browns Ferry Multi-Unit PRA.

3. DRP (York/Turner) Determine problems with identifying inspections on the MIP. Review and ensure MIP is updated to reflect planned inspections.

These actions shall be completed by the next oversight meeting on May 24, 1995.

The following item was completed on the Task Checklist:

NRC/Licensee Agreement on Restart Issues-The licensee agreed with the Issues Checklist and provided a status report of all items during the April 19, 1995 meeting.

Appendix A

BROWNS FERRY 3 TASK CHECKLIST

Date Printed: April 25, 1995

TASK	RESP.	DATE	STATUS
Establish Restart Panel	RII, NRR	2/1/95	Complete
Develop Case Specific Checklist	RII, NRR	9/22/94	Complete
Develop Restart Action Plan	RII, NRR	2/1/95	Complete
Regional Administrator Approves Plan	RII	2/1/95	Complete
NRR Associate Director Approves Plan	NRR	2/1/95	Complete
Notification Restart Panel established; RON 509	Lesser	2/24/95	Complete
Licensee performs root cause analysis and develops corrective action plan	Licensee	7/10/91	Complete
NRC evaluates licensee's root cause determination and corrective action plan	NRR	4/1/92	Complete
Review licensee generated restart issues	Panel	3/21/95	Complete
Independent NRC identification of restart items (consider external sources)	Panel	3/21/95	Complete
NRC/Licensee agreement on restart issues	Panel	4/19/95	Complete
Obtain public comments; (press conf)	Lesser	3/21/95	
Obtain comments from State and Local Officials	Barr		
Obtain comments from applicable Federal agencies	Barr		
Evaluate licensee's readiness self assessment	RII		
Conduct Operational Readiness Assessment Team Inspection (ORAT)	Koltay		
Restart issues closed	Panel		
Issue augmented restart coverage ROI	RII		
Obtain staff comments on restart	RII, NRR		
Re-review MC 0350 generic restart checklist	Panel		

Prepare restart recommendation document and basis for restart to Regional Administrator	RII		
Restart meeting with licensee	Panel		
Restart Panel recommends restart	Panel		
Regional Administrator concurs in restart recommendation (SECY paper)	RII		
NRR Associate Director concurs in restart recommendation (SECY paper)	NRR		
EDO Concurs in restart recommendation (SECY paper)	NRR		
ACRS briefing	NRR		Not Required
Submit Commission paper	NRR		
Commission briefing	NRR, RII		
Commission restart authorization	Comm		
Notify Congressional Affairs of restart	NRR		
Notify ACRS of restart	NRR		
Notify FEMA of restart	RII, NRR		
Notify Public Affairs of restart	RII		
Notify State and Locals of restart	RII		
Monitor restart	RII		

Appendix B

BROWNS FERRY 3 ISSUES CHECKLIST

Date Printed: April 25, 1995

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
TMI ACTION ITEMS (TI 2515/065)							
I.D.1.2	Control Room Design Review TAC M56106; MPA F071	Peebles	SE 10/29/91 93-201 94-09 94-21	essentially complete	Inspection performed 9/94 reviewed program satisfactory		C
I.D.2.2	SPDS Installed TAC M74612; MPA F075 GL 89-06 TAC M73636 F072	SRI	SE 2/5/92 IR 95-22	field complete	Installation verified, open pending PMT		
I.D.2.3	SPDS Fully Implemented TAC M51225; MPA F009	SRI	SE 2/5/92	field complete; testing 4/96			
II.B.3.2	Post Accident Sampling - Corrective Actions TAC M74613; MPA F076	Decker	SE 5/27/87 94-33				
II.B.3.3	PASS - Procedures TAC M74614; MPA F077; M83122	Decker	SE 5/27/87; TS amend 6/21/94	2/22/95			
II.B.3.4	PASS - Modifications TAC M44425; MPA F012	Decker	SE 5/27/87	Design complete; 50% implement 12/94 75% implement 4/95			
II.E.4.1.2	Dedicated Hydrogen Penetrations - Review and Revise H2 Control Proc TAC R00003	SRI	SE 5/23/88 95-10	Complete; ready for closure	SIMS ready for closure		C
II.E.4.1.3	Dedicated Hydrogen Penetration - Install TAC M44763; MPA F018	SRI	SE 12/22/81 95-10	Complete; ready for closure	SIMS shows ready for closure		C
II.E.4.2.1-4	Containment Isolation Dependability - Diverse Isolation TAC M74615; MPA F078	SRI	SE 1/6/95 IR 95-16	TVA to provide completion status 8/95	Installation verified, open pending PMT		

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
II.E.4.2.6	Containment Isolation Dependability - Containment Purge Valves TAC M74616; F079	SRI	SE 1/6/95 95-16		installation verified, open pending PMT		
II.F.1.1	Accident Monitoring - Procedures TAC M74617; MPA F081	Decker	SER 8/17/90 94-33	Licensee tracking with individual instruments		Listed in NUREG 1435; not on other lists	
II.F.1.2.A	Accident Monitoring - Noble Gas Monitor TAC M44905; MPA F020	Decker	SER 12/22/81	4/18/95			
II.F.1.2.B	Iodine/Particulate Monitor TAC M44976; MPA F021	Decker	SER 12/22/81	4/18/95			
II.F.1.2.C	Containment High Range Monitor TAC M45047; MPA F022	Rankin	SER 1/8/82 IR 94-28 95-11	May 95			
II.F.1.2.D	Containment Pressure Monitor TAC M47584; MPA F023	SRI	SER 6/16/83	field complete 7/95			
II.F.1.2.E	Containment Water Level Monitor TAC M47655; MPA F024	SRI	SER 6/16/83	field complete 5/95			
II.F.2.4	Instrumentation for Detection of Inadequate Core Cooling GL 84-23 TAC M45118; MPA F026	SRI	SER 11/18/86 95-16	field complete 2/95	open pending review of procedures, PMT, training		
II.K.3.13.B	HPCI/RCIC Initiation Levels TAC M45534; MPA F043	SRI	SER 9/19/83 90-23	field complete 1/95			
II.K.3.18.C	ADS Actuation Modification TAC M45682; MPA F048	SRI	SER 5/29/90	field complete			
II.K.3.27	Common Reference Level TAC M45778; MPA F054	SRI	SER 12/3/82 95-16	T 12/3/82 stated licensee to track under CRDR as alternate method; field complete 2/95			C
II.K.3.28	Qualification of ADS Accumulators TAC M48262; MPA F055	SRI	SER 7/24/85	field complete 4/95			

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
II.K.3.57	Identify Water Sources Prior to Manual Actuation of ADS MPA F062	SRI			SIMS shows ready for closure		
III.D.3.4.3	Control Room Habitability - Implement Mods	SRI	SER 8/30/82 90-37		SIMS shows ready for closure	closed for all 3 units	C
TEMPORARY INSTRUCTIONS							
TI 2500/020	ATWS GL 83-28 TAC M07931; MPA D001	SRI	SER 1/22/90 IR90-29 IR90-33 95-22	design complete; 50% implement 12/94 75% implement 3/95	pending field completion and testing		
TI 2515/074	Employee Concerns Resolution	SRI	IR90-31 93-18 93-32 93-43 94-20 95-10				C
TI 2515/089	Stress Corrosion Cracking in BWR Piping	Blake			SIMS shows ready for closure	TI for GL 84-11; GL 88-01 superceded 84-11 and TI cancelled	
TI 2515/095	BWR Recirc Pump Trip	SRI	95-22	field complete 6/95			
TI 2515/099	BWR Power Oscillation IEB 88-07 TAC M72769; MPA X807	Kellogg	SER 4/4/90 TS 179 5/31/94	6/2/95			
TI 2515/109	MOV Testing GL 89-10	Casto	94-03	50% implement 6/95 75% implement 9/95			
TI 2515/111	EDSFI followup	Shymlock					
TI 2515/112	Eval Changes in Environs	SRI	93-44				C
TI 2515/118	Service Water System TAC M73972; MPA L917	Kellogg	SER 4/23/90				
TI 2515/119	Water Level Inst Errors GL 92-04	Lawyer	IR 93-16				C
TI 2515/120	Station Blackout TAC M68519: MPA A022	Shymlock	SER 9/16/92	> 75% implement complete 2/95			
TI 2515/121	Installation of Hardened Wetwell Vent GL 89-16	SRI	SER 8/16/91	field complete 6/95			
TI 2515/122	Loss of Fill Oil for Rosemont Transmitter IEB 90-01 TAC M85363 MPAB122	Shymlock			NRR to issue SER early 1995		
TI 2515/128	Plant Hardware Mods to Rx Vessel Water Level Inst.	Shymlock	SER 4/20/94 IR 93-201				

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
NRC BULLETINS							
IEB 79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts TAC R00017	Blake				* Refer to Large Bore Piping and Supports Program	*
IEB 79-12	Short Period Scrams at BWRs	SRI	81-18	field complete 9/95		Project plan	
IEB 79-14	Seismic Analysis for As-built SR Piping Systems TAC R00017	Blake				* Refer to Large Bore Piping and Supports Program	*
IEB 79-18	Audibility Problems	Barr	IR93-23	TVA 6/22/93 closed for U2; Licensee to submit closure letter for U3		TVA 10/21/94 certified that IEB was previously completed on all 3 units	C
IEB 80-06	ESF Reset Controls	SRI	95-22	field complete		Project Plan	C
IEB 83-02	Stress Corrosion Cracking in Large Dia SS Pipe	Blake	IR83-55 IR86-03			* Refer to GL 88-01	C
IEB 83-08	Elect Circuit Bkrs with UV Trip Feature in SR applications other than RPS	SRI	95-22			Project Plan	C
IEB 84-02	Failures of GE HFA relays in 1E Safety systems	SRI				Project Plan	
IEB 85-03	MOV Common Mode Failures; GL 89-10	Casto	88-32			* Refer to GL 89-10	*
IEB 86-02	Static O Ring DP Switches	Casto	94-31	field complete		Project Plan	
IEB 88-03	Inadequate Latch Engagement in HFA relays by GE TAC M73854; MPA X803	NRR	SER 8/2/90	TVA 4/11/94 reported completion of requirements; inspections complete for U3; no problems found		NUREG 1435; not on licensee list; NRR check if issue is closed	
IEB 88-04	SR Pump Loss TAC M69890; MPA X807	SRI	SER 4/4/90	field complete 6/95		Project Plan; TI 2515/087 issued but not required for BF	
IEB 88-07	Power Oscillations in BWRs; TI 2515/099	Kellogg				* Refer to TI 2515/099	*

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
IEB 90-01	Loss of Fill-Oil in Rosemount Transmitters TAC M85363; MPA B122	Shymlock			SER scheduled for early 1995	* Refer to TI 2515/122	*
IEB 92-01	Thermolag TAC M83850; MPA X201	Casto	SER 11/13/92			* Refer to GL 92-08	*
IEB 93-02	Debris Plugging of ECCS Suction Strainers TAC M86537; MPA X302 TAC M89279; MPA B124	SRI	SER 6/28/93 7/19/94	field complete 8/95		new TI	
IEB 93-03	Issues related to Reactor Vessel Water Level Inst. TAC M86884; MPA X303	Shymlock	SER 4/20/94 IR 93-201	complete 3/95 modifications		refer to TI 2515/128	*
NRC GENERIC LETTERS							
GL 82-33	Inst to follow course of Accident; RG 1.97 TAC M51075; MPA A017	Shymlock Decker	SER 2/8/90 IR90-32 93-201 94-33	June 95		Project Plan; TI 2515/087 closed IR 90-32	
GL 83-28	Salem ATWS; TI 2500/020	SRI				* refer to TI 2500/020	*
GL 83-36	NUREG 0737 TS	NRR	94-33; TS change 6/21/94 on PASS				C
GL 88-01	IGSCC in BWR Aust SS Piping TAC M85296	Blake	SER 12/3/93	field complete		Project Plan	
GL 88-11	Radiation Embrittlement of Reactor Vessel; RG 1.99 TAC M71469; MPA A023	NRR	SER 6/29/89	licensee states that TS amendments 190, 205 162 of 1/8/93 satisfied GL		NUREG 1435; not on licensee list; NRR to review for closure	
GL 88-14	Instrument Air Affecting SR Systems TAC M71633; MPA B107	SRI	SER 5/9/89	field complete 9/95		listed in NUREG 1435 and Project Plan	
GL 88-20	IPE	NRR		T 9/1/93 submitted IPE for all units; Expanded PRA of 10 systems for multiunit ops due 5/95		SER due 9/30/94 Seismic Evaluation Report due to NRC 3/19/96	NR
GL 89-06	SPDS TAC 73636; MPA F072	SRI	SER 2/5/92			* Refer to TMI Item 1.D.2	*

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
GL 89-08	Erosion Corrosion monitoring program TAC M73459; MPA L908	Blake	SER 8/21/89	10/9/94	DRS to schedule module 49001	NUREG 1435 and Project Plan	
GL 89-10	MOV Testing And Surveillance TAC M75637; MPA B110	Casto				* Refer to TI 2515/109	*
GL 89-13	Service Water Systems TAC M73972; MPA L913	Kellogg				* Refer to TI 2515/118	*
GL 89-16	Installation of Hardened Wetwell Vent TAC M74860; MPA B112	SRI	SER 8/16/91.	10/31/94		* Refer to TI 2515/121	*
GL 89-19	USI A-47 Safety Implication of Control Systems TAC M74917; MPA B113	NRR	SER 6/28/94; SE 9/22/94				C
GL 92-01	Reactor Vessel Structural Integrity TAC M83440; MPA B120	NRR	SER 4/19/94	T 9/2/93 5/23/94 7/28/94 identify commitments; licensee to add to Project Plan	Plant specific reviews being performed on all units	NUREG 1435	NR
GL 92-04	Reactor Vessel Water Level Instrument TAC M84271; MPA B121	SRI	SER 3/25/93 IR93-16	11/25/94		TI completed; Further review of mods under IEB 93-03	C
GL 92-08	Thermolag	Casto	SE 5/11/94	TVA 3/22/95		RHRSW cables use thermolag; and will be upgraded to configurations as tested at WB prior to rx vessel hydro;ampacity/combustible analysis by 12/22/95 and abandoned material removed by 6/20/96	
GL 94-02	Long Term Soln for Thermal Hydraulic Instabilities	Peebles	94-11	TVA 12/22/94 procedures to be revised prior to restart		action 1.a closed	
GL 94-03	IGSCC of Core Shroud TAC M90083	Blake	94-16	Licensee inspections performed June and July 1994; TVA 9/23/94 concludes U3 can be operated for at least 1 cycle		SE 1/13/95 concluded structural integrity will be maintained for at least 1 cycle without need for mod; TVA to reinspect OG	C

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
UNRESOLVED SAFETY ISSUES							
A-7	Mark I Long Term Program TAC M07931; MPA D001	Blake	88-19			* Refer to Long Term Torus Integrity Program; TI 2515/085 closed 88-19	*
A-9	ATWS	SRI				* Refer to TI 2500/020	*
A-24	Qualification of Class 1E Equipment TAC M42483; MPA B060	Shymlock	88-11			* Refer to EQ Program; TI 2515/076 closed 88-11	*
A-36	Control of Heavy Loads Near Spent Fuel Pool	SRI	94-12			Project Plan	
A-42	Pipe Cracks in BWRs	Blake				* Refer to GL 88-01	*
A-44	Station Blackout	Shymlock				* Refer to TI 2515/120	*
A-46	Seismic Qualification of Equipment in Operating Plants TAC M69432; MPA B105	NRR		Seismic Eval Rpt/ Seismic IPEEE due 3/96		Identified not to be a restart item in SE e.g. conduit support SE 3/20/92	NR
A-47	Safety Implication of Control Systems	NRR				* Refer to GL 89-19	*
A-48	Hydrogen Control Measures and Burn Effects TAC M55955; MPA A019	SRI	SER 9/9/86			Project Plan in WISP	
GENERIC SAFETY ISSUES							
GSI 40	Safety Concerns Associated with Pipe Breakes in BWR Scram Syst TAC M43736; MPA B065	Blake	SER 1/7/86			NUREG 1435 and Project Plan	
GSI 41	BWR Scram Discharge Volume System TAC M51014; MPA B058	SRI	SER 6/24/83	field complete 6/95		NUREG 1435; TI 2515/090 closed in 87-13; Project Plan	
GSI 43	Reliability of Air Systems	SRI				* Refer to GL 88-14	*
GSI 51	Improving SW system Reliability	Kellogg				* Refer to TI 2515/118	*
GSI 67.3.3	Improved Accident Monitoring; RG 1.97	Shymlock				* Refer to GL 82-33	*

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
GSI 75	Salem ATWS 4.5.2 and 4.5.3 RPS Test Alternatives TAC 53966; MPA B093	Shymlock	SER 8/17/90	licensee tracking under GL 83-28			
GSI 75	Salem ATWS 1.2 Data Capability TAC M53573; MPA B085	SRI	SER 6/12/85	licensee tracking under GL 83-28			
MULTI PLANT ACTION ITEMS							
MPA A004	Appendix J Cont Leak Testing M08717	Casto	SER 10/24/84				
MPA B118	IPE External Events	NRR				* Refer to GL 88-20	NR
MPA B41	Fire Protection Final TS M48136	NRR	SER 10/12/83			Removal of TS complete, NRR reviewing Fire Protection Plan submittal	
MPA C-10	Heavy Loads Phase I	NRR	SER 6/6/84			Project Plan	
MPA C011	RPS Power Supply M08931	NRR	SER 6/27/85	6/13/94		Project Plan	
INSPECTOR FOLLOWUP SYSTEM							
URI 84-29-01	Failure to adequately control welding	SRI	IR 94-18				C
IFI 84-32-02	Torus Level Instrumentation	SRI		field complete 5/95			
IFI 84-41-04	Relocation of HPCI Emerg Control Boxes	SRI	94-27				
IFI 85-09-02	Bolts inadequate on Limitorque motors	Casto					
URI 85-26-03	Interim Acceptability of Plant Operation for IEB 79-02	Blake	95-03				C
VIO 85-41-01	Cable Tray Supports	Blake					
IFI 85-51-01	Cable Tray Support Criteria Seismic	Blake					
LER 85-20	Failure to install core spray hanger	SRI	94-32				C

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
LER 85-32	Reevaluation of Design Criteria for FSAR 8.6.2.1	Shymlock	91-06				
LER 85-33	Nonstandard 4" pipe penetrations thru sec cont	SRI	88-28				
LER 86-47	Improper SLC heat trace tape	SRI	89-50 94-32				C
URI 86-06-02	Rx Bldg control Bay HVAC inadequate design	Blake	89-20				
URI 86-14-03	Overstress of Drywell Beams	Blake					
IFI 86-28-05	Addition of EECW/RBCCW Xconn valves to IST	SRI	94-27				C
IFI 86-40-03	IRM Power supply and procedure changes per SIL 445	SRI	89-35 94-32				C
LER 86-10	Design Engineers identified connection of unqualified piping to containment sensing lines	SRI	88-28				
LER 86-16	Fluid leakage problem with large bore snubbers for Torus dyn restraint	Blake	88-28				
LER 86-18	Neutron monitor surv test deficiencies	SRI	88-33 94-36				C
URI 87-02-02	Limitorque gear ratio	Casto	88-16				
URI 87-26-03	RHR pump suction and nozzle load allowances possibly exceeded	Blake	88-32 90-08				
IFI 87-33-02	Failure of drywell control air isol valves to fail closed on air loss	SRI	89-35 95-16	field complete 2/95			C
IFI 87-37-03	Reactor Water Level Sensing lines	SRI	89-35 95-16	field complete 1/95			C
IFI [illegible]	Temp Alterations change forms large number outstanding	SRI	94-17				C

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
VIO 88-04-03	Failure to Correctly translate design requ into drawings	SRI	91-02 91-16	field complete 4/95			
LER 88-12	Battery failure concurrent with LOP/LOCA prevent auto start	Shymlock	89-35				
LER 88-16	Unplanned manual start of ESF due to pers error	SRI	89-50			Placard mods needed	
LER 88-25	CR operator dose may exceed design limits after accident because of design error	SRI	94-24				C
LER 88-32	Elect sep requ violated due to design controls	Shymlock	90-25				
LER 88-37	Inadequate design control process discrep in HVAC duct work	Blake					
LER 88-40	Inadequate design controls results in backup control system not meeting design req	Shymlock	91-16				
IFI 89-11-03	Deteriorated GE cables	SRI	90-03 95-22				C
IFI 88-17-05	Followup on ATWS mods	SRI	95-22				C
IFI 89-20-02	CRD seismic analysis	Blake	95-03				C
URI 88-56-02	Use of closed manual valves in ECW line to control bay chiller	SRI	91-10 94-17				C
LER 89-03	Design of suppression pool vacuum relief system not provide single failure proof	SRI	91-10 95-22	field complete 4/95			C
LER 89-11	Design error in EECW anti syphon check valves	SRI	91-02 95-22				C
LER 89-07	Cable deterioration causes inoperable neutron monitoring	SRI	95-22				C
LER 89-25	Design errors in 250VDC results in unanalyzed cond	Shymlock	90-03				

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
URI 90-33-06	RPS/ARI diversity	SRI	93-32 95-22				C
IFI 90-60-01	Deficiencies identified during integrated ESF test	SRI	95-10				C
DEV 91-41-01	Control of Contruction	SRI	95-10			PER 827	C
LER 91-01	Failure of Two trains of Standby power system to load sequence	Shymlock	91-10 95-10				C
LER 91-15	HPCI did not fulfill safety function from low suction pressure during fast start	SRI	92-11 95-10			Mod complete, post mod testing sch for 9/95	
URI 92-07-01	Large bore walkdown insp & docu checking problems	Blake	92-19 93-29 95-03				C
IFI 92-30-03	Circuit Breaker Coordination	SRI	94-18				C
IFI 92-32-01	Design Problems in Spring Supports	Blake	94-15				C
VIO 92-37-01	Failure to verify secondary containment isolation	SRI	94-06 94-32			LER 92-03	C
LER 92-02	ESF actuation from relay failure	SRI	94-20				C
LER 92-03	Failure of Reactor zone isol damper to close	SRI	94-32		vio 92-37-01		C
LER 92-05	Design Deficiency allowed secondary containment atm to be released thru RCW	SRI	95-10				C
URI 93-11-01	Weld Differerences between the welds assumed in support	Blake					
IFI 94-04-02	Verify Method used to install wedge anchors	Blake					
IFI 94-07-03	Verification of SBO Functional Requirements	NRR	95-16				
IFI 94-07-03	System operational boundary test identification	NRR					

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
IFI 94-11-02	Response to GL 94-02	Peebles					
IFI 94-12-01	Rosemont Transmitter Drift Problems	Shymlock					
VIO 94-17-01	Instrumentation Calibration Deficiencies	SRI					
IFI 94-18-02	Condition of Containment Coating :	SRI					
IFI 94-29-01	Review of Conan Concrete Capacity	Blake					
IFI 94-29-02	Design Methods for Anchor Location Tolerance	Blake					
URI 94-29-04	Amplification factors for anchor loads	Blake					
VIO 94-33-01	Failure to maintain spacing in low voltage cable tray	Shymlock					
VIO 95-03-01	Spring can installation	Blake					
URI 95-10-01	Inadequate Second party check by foremen (Beta Tape)	Shymlock	94-35				
URI 95-16-01	Work Performed on incorrect equip	SRI					
LICENSE AMMENDMENTS							
TS 359	Scram Pilot Air Header Pressure Switchs TAC	NRR					
TS 337	Appendix R License Ammendment TAC M87902	NRR					
TS 320	RWCU Temperature Switches TAC M88085	NRR	SE 3/15/95				C
TS 340	Diesel Generator Load Shedding TAC M89245	NRR	SE 2/14/95				C
TS 319	HPIC/RCIC Temp and Channel Checks TAC M89247	NRR	SE 3/16/95				C
TS 318	Analog Transmitter/Trip Systems TAC M89250	NRR					

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
TS 339	Extended Load Line Limit and Revised RBM Operability TAC M89253	NRR	SE 2/24/95				C
N-416-1	Pressure testing Relief Reqest	NRR					
N-498-1	Pressure testing Relief Reqest	NRR					
	Standby coolant supply	NRR					
MISC ISSUES							
	Procedures Upgrades	Kellogg	93-36				
	TS Changes	SRI					
	Plant Simulator Cert	Peebles		TVA ltr 12/17/91			C
NPP pg IV-17	HPCI Controller Improvements	SRI	SSER2 App E	NC0860270003			
NPP pg III-57	Shroud Head Bolts (IGSCC)	Blake	SSER2 Sec 3.6	NC0860326118			
NPP pg II-94	Online Chem Instrum	Decker	SSER2 Sec 4.10	NC0860326292			
NPP pg II-58	Unresolved CAQs	SRI	SSER2	NC0860326143		periodic insp of PERs	
	ERDS link to HQ operational	Barr					
	Review licensee CATD closeout letter	SRI					
PROGRAMS IMPLEMENTED IN ACCORDANCE WITH UNIT 2 PRECEDENT							
	Cable Ampacity	Shymlock	94-35	> 50% implement 75% implement 4/95 complete 5/95		NUREG 1232 V3 S2 reviewed program and NRC ltr of 6/23/93 reviewed followup items	
	Cable Tray supports M 80684	Blake	SER 12/17/91	TVA ltr 3/27/91; >90% implement complete 2/95			

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
	Component and Piece Parts Qualification M 83828	Blake	SER 12/7/93	TVA ltr 6/12/92		Program has changed; possibly perform module 38703 on procurement	
	Containment Coatings	SRI	94-01 94-09 94-18 94-27		IFI 94-18-02 opened to track repairs to U3		
	CRD Insert and Withdrawal Piping	Blake	95-03				
	Design Calc Review	Casto	94-31				
	EQ TAC M42482; MPA B060	Shymlock	94-06 94-27 94-35	<75% implement		TI 2515/076 closed 92-03	
	Flexible Conduits	Blake					
	Fuses	Shymlock	SER 13.6 93-02 93-08				
	HVAC Duct Supports TAC R00300, M82127	Blake	SER 10/24/89 7/16/92 IR 93-201	50% implement 2/95 complete 5/95	Inspect as part of open item closure		
	IGSCC	Blake	92-31 93-05			* Refer to TI 2515/089 DRS Reviewed program and found acceptable	*
	Large Bore Piping and Supports (IEB 79-02 & 79-14) TAC R0017	Blake	SER 10/24/89 94-15 94-29 95-03	75% implement 10/94 complete 7/95			
	Misc Steel Frames TAC R00297 M80620	Blake	SER 10/24/89 94-15 94-29	>50% implement 75% implement 11/94 complete 1/95			
	Moderate Energy Line Break	Blake	SER 4/20/94				
	Platform Thermal Growth TAC R00297, M80620	Blake	SER 10/24/89 4/20/94 IR 93-201 94-29				
	PRA	NRR		T 1/19/95		TVA to submit multi unit PRA 4/95, IPEEE 6/95, Internal fires IPEEE 120days after refueling	NR
	Q-List	SRI					
	Seismic Class II/I TAC M80015	Blake	SER 12/17/91	TVA ltr 2/27/91 50% implement 12/94 75% implement 5/95			

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
	Splices	Shymlock	90-22 95-14			SSER2 3.13 found program acceptable and IR satisfactory for all units	
	Thermal Overloads	Shymlock	SER 13.4				
PROGRAMS WHICH DEPART FROM THE UNIT 2 IMPLEMENTATION PRECEDENT							
	Cable Installation M80682	Shymlock	SER 4/8/92 7/1/94 93-34 94-27 94-35			revised bend radius for medium voltage cables	
	Conduit Support TAC R00024 M80690	Blake	SER 10/24/89 3/30/92	>50% implement 75% implement 10/94 complete 5/95			
	Configuration Mgmt/Design Baseline M80688	Casto	94-07 SER 11/21/91 94-07 94-20 94-31				
	Instrument Tubing TAC M80036	Blake	SER 2/4/92 95-03	TVA2/27/91 TVA12/12/91		Licensee has combined inst tubing and small bore piping programs	
	Instrument Sensing Lines TAC M80017	SRI	SER 12/10/92 IR 94-24	TVA ltr 2/13/91			
	Long Term Torus Integrity TAC M80686	Blake	SER 2/10/92 94-15	TVA ltr 4/29/91 >75% implement complete 5/95		URI 94-15-01, Spring Can Settings	
	Restart Test Program TAC M81791	SRI Casto	SER 8/30/94	TVA ltr 2/2/94 50% implement 6/95		SRI - Review administrative program; Casto - identify electrical/mechanical tests and inspector to review.	
	Small Bore Piping TAC M80013 R00306	Blake	SER 10/24/89 2/4/92 95-03	TVA2/27/91 TVA12/12/92			
PROGRAMS WHICH DEPART FROM UNIT 2 CRITERIA PRECEDENT							
	Fire Protection; App R. TAC M85254	Casto	94-27 95-04 95-07	Licensee submittal of 12/20/94, status > 50% implement 75% implement 4/95	NRR to write SER by 7/95		
	Lower Drywell Platforms and Misc Steel TAC M80620 R00303	Blake	SER 7/26/88 10/24/89 3/19/92 4/20/94 IR94-15, 93-201 94-29	TVA 6/12/91 >90% implement complete 1/95		Long term design criteria implements 1978 AISC spec	
PROGRAMS COMPLETED ON ALL THREE UNITS							

ISSUE	DESCRIPTION	NRC LEAD	IR/SER	LICENSEE STATUS	NRC ACTION	COMMENTS	STATUS
	Heat Code Traceability	Blake	SER 5/31/90			NURUG 1232 V3 S1 sec 2.3 and NRC SER of May 31, 1990 reviewed program for all 3 units.	
	Secondary Containment Penetrations	Blake	SER 4/11/88			Program evaluated by April 11, 1988 addressed all 3 units	
	Welding Program	Blake	SER 5/31/90			Welding concerns adequately addressed per NUGEG 1232,V3,S1	
	Pipe Wall Thinning (GL 87-01)	Blake	SER 8/31/88			SER addressed all 3 units	

Sources for issues include: IFS, SIMS, WISP, NUREG 1435 (Status of Safety Issues at Licensed Power Plants), BFNPP, NUREG 1232 (SER for Browns Ferry NPP)

1. EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) policy statement on severe accidents in nuclear power plants was published in the Federal Register on August 8. 1985. The severe accident policy statement of the NRC concluded that existing plants did not pose an undue risk to the public health and safety. However, the NRC stated that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements. The NRC's plan for implementing the severe accident policy statement was published on May 25, 1988. The first step in this plan was to request that licensees complete an Individual Plant Examination (IPE). The IPE was intended to be "an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributors that might be plant specific and might be missed absent a systematic search." On November 23, 1988, licensees were requested by Generic letter No. 88-20 to perform an IPE/probabilistic risk assessment (PRA) that addressed each plant in order: "(1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant, (3) to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) if necessary, to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents."

A PRA is the usual and preferred method of performing an IPE. A PRA is a structured analysis of postulated events, equipment failures, operator errors, or various combinations of each, which could result in a degraded core and/or a major offsite release of radioactivity. In response to Generic Letter No. 88-20, TVA committed to model BFN Unit 2 and perform a PRA and containment analysis.

However, in August 1990, the NRC noted that the three units at BFN share many important safety systems. The NRC expressed a concern with the potential safety implications of shared systems in the various operating modes of the BFN units; e.g., all three units operating, Units 1 and 2 operating with Unit 3 shutdown, etc. In response to this concern, TVA committed to perform a multi-unit PRA, which bounds the various combination of units in operation and evaluates the impact of the shared systems on the probability of a degraded core calculated by the BFN PRA.

The single unit BFN PRA was submitted for NRC review on September 1, 1992, and approved by the NRC on September 28, 1994. As part of the commitment of TVA to maintain the BFN PRA current over the life of the plant, the PRA that was submitted to the NRC review was revised as a result of plant modifications and to refine previously modeled plant features.

The enclosed report provides the BFN Multi-Unit PRA. The results of this multi-unit analysis indicated that the most limiting site configuration is with all three BFN units in operation. The resulting core damage frequency for Unit 2, with three units operating, of 2.8E-05 is

approximately a factor of 4 higher than the revised single unit estimate of 7.6E-06; however, the multi-unit core damage frequency still represents a very low risk from severe accidents. As shown below, no single initiating event was found to dominate the total frequency of core damage.

No plant vulnerabilities were identified for BFN when multiple units are in operation. Therefore, no additional enhancements are required to address vulnerabilities.

1.1 BACKGROUND

This report documents the work performed by Tennessee Valley Authority (TVA) and its contractor, PLG, to investigate the influence on the core damage frequency (CDF) at Browns Ferry Nuclear Power Plant associated with the bounding configuration of all three units operating.

TVA has previously submitted to the U.S. Nuclear Regulatory Commission a plant-specific probabilistic risk assessment (PRA) for Browns Ferry Unit 2 in September 1992 (Reference 1). That analysis, referred to as Rev. 0, represented the plant conditions at the time of the submittal; namely, Unit 2 operational and Units 1 and 3 defueled. TVA subsequently performed updates, the latest denoted as Rev. 1A, to the Unit 2 PRA to reduce some of the initial modeling conservatisms, to incorporate the effects of design changes at the plant made since the original analysis, and to incorporate selected plant-specific data. In the Rev. 0 and Rev. 1A PRA for Unit 2, plant systems and features shared among units were considered to support Unit 2, as appropriate.

TVA committed to the NRC (Reference 2) to perform an expanded PRA that considers the shared plant systems and features, and considers in this study a particular bounding configuration in which all three units are in operation. This report presents the results of what is referred to as the Multi-Unit PRA.

The methodology used in this study is summarized in Section 2 and is a straight forward extension of the methodology used with previous PRAs on Browns Ferry. The main difference is that this study considers a comprehensive set of multi-unit interactions that was not addressed in the previous PRAs. Potential system and unit interactions are first identified. Next, a bounding plant configuration is determined. This bounding configuration specifies the initial status of the three units. Initiating events that are specific to multi-unit operation are then identified. In addition, system and operator action success criteria specific to multi-unit operation is determined.

The models developed for the Unit 2 Rev. 1A PRA were used as a starting point in the current analysis. The additions and changes to these models that were necessary are documented in Section 3.

This report also presents the impact of expanding the PRA models developed for the Rev. 1A analysis to explicitly consider the effects of the loss of control bay ventilation.

As part of the Multi-Unit PRA, dependency matrices similar to the ones developed for the Unit 2 Rev. 0 PRA were updated for Unit 2 and new ones developed for Units 1 and 3. These matrices document the intersystem dependencies that exist between plant systems considered in the PRA and are also provided in Section 3.

Section 4 documents the results of an investigation of the multi-unit interactions to verify no risk significant vulnerabilities were overlooked in selecting the bounding plant configuration.

Section 5 describes TVA's participation in review and performing the Multi-Unit PRA.

Section 6 documents the unique strengths of the Browns Ferry Nuclear Plant and the assessment of plant vulnerabilities and potential enhancements.

Section 7 summarizes the final conclusions of this Multi-Unit PRA.

The references of the report are provided in Section 8. and the detailed backup calculations and documentation are provided in the appendices.

1.2 RESULTS

The quantitative findings of the Browns Ferry Multi-Unit PRA are presented in this section, and are compared to the results of the Unit 2 Rev. 1A PRA model. The results delineate the principal contributors to risk. The basis for the multi-unit analysis and, therefore, the basis of the comparison of the Multi-Unit PRA results to those of the single unit PRA, is the frequency of core damage.

For the Multi-Unit PRA, an initial plant configuration, which is bounding with respect to the availability of systems to avert core damage, is selected. In this manner, the consideration of the CDF results of the single unit Rev. 1A model and the Multi-Unit PRA model provides lower and upper bounds, respectively, for the CDFs that would be applicable to the other possible initial plant configurations at Browns Ferry. The same initiating events were used for both models, plus six additional ones for the multi-unit model quantification. The baseline configuration date for both the Multi-Unit PRA and Unit 2 Rev. 1A is May 31, 1993.

The mean value of the uncertainty distribution for the total CDF for Browns Ferry Unit 2 under the conditions that all three units are initially operating at power was found to be 2.8E-05 per reactor-year.* For the Rev. 1A model, corresponding to Unit 2 initially at power and Units 1 and 3 defueled, the mean value of the distribution describing the CDF was determined to be 7.6E-06 per reactor-year. For both analyses, core damage is assumed for any sequence in which sustained core uncovery occurs. Per the vulnerability criteria specified for the IPE Rev. 0 report and provided here in Section 6, no vulnerabilities were identified. The results for CDF were developed in terms of a mean point estimate, as required in

*The unit for the CDF is events per nuclear-powered electric generating unit per calendar year. This definition is abbreviated to "per reactor-year."

NUREG-1335 (Reference 3), as well as the previously cited uncertainty distribution. The presentation of the total CDF in terms of the uncertainty distribution is shown in Figure 1-1 for the Multi-Unit PRA and Unit 2 Rev. 1A PRA. Note that the Monte Carlo process used to determine the uncertainty distributions yields a slightly different value for the mean than the point estimate mean reported elsewhere in this report. This deviation between point estimate and Monte Carlo means is normal and results from small numerical uncertainties associated with the Monte Carlo sampling process.

Descriptive parameters of the uncertainty distributions are as follows:

PRA	5th Percentile	50th Percentile	Mean	95th Percentile
Multi-Unit	4.5E-06	1.5E-05	2.8E-05	8.2E-05
Unit 2 Rev. 1A	1.6E-06	4.5E-06	7.6E-06	2.3E-05
Unit 2 Rev. 0	5.6E-06	—	4.8E-05	1.1E-04

In the quantification of the Level 1 event sequence models, the principal contributors to the CDF were identified from several vantage points. The results and contributors are summarized in this section for the multi-unit model and compared to the Rev. 1A model.

The Multi-Unit PRA was initially based on Unit 2 Rev. 1 PRA model. In the process of developing the Multi-Unit PRA, refinements to the Unit 2 Rev. 1 model were provided, and the Unit 2 PRA was updated to Rev. 1A by TVA.

1.2.1 IMPORTANT CORE DAMAGE SEQUENCE GROUPS

The importance of initiating events was examined by determining the contributions of core damage sequences grouped by type of initiating event. The ranked results are shown in Table 1-1 and Figure 1-2 for major initiating event categories.

As can been seen, the mean CDF corresponding to the multi-unit configuration, while still small, is about a factor of 4 greater than the corresponding mean CDF of the single unit configuration. The reason for the increase is the change in success criteria for shared systems for initiating events that could impact two or three reactor units concurrently. Specifically, the impact of the change in success criteria for such shared features as diesel generators, emergency equipment cooling water system (EECW), and residual heat removal service water system (RHRSW) is evident for initiator categories such as loss of offsite power that could impact all three units concurrently. For initiators (such as those that comprise the category "transients with reactor not isolated") that involve essentially a single unit to respond, the impact of shared features is much more modest.

A detailed listing of the contribution of each initiating event to the CDF is given in Appendix C, and is summarized below for Unit 2 in the Multi-Unit PRA and compared to the Unit 2 Rev. 1A PRA:

- Scenarios initiated by a loss of offsite power contribute 39% of the CDF in the Multi-Unit PRA as compared to 20% for the Unit 2 Rev. 1A PRA.

- Scenarios initiated by internal floods contribute 22% to the CDF for the Multi-Unit PRA as compared to 15% for the Unit 2 Rev. 1A PRA. No internal flooding scenarios lead directly to core damage but require additional hardware failures. Flooding initiators were postulated in the Unit 2 reactor building, in the Unit 1 or 3 reactor building, in the turbine building, and at the intake pumping station. One flooding sequence, initiated by a flood in the turbine building, has a mean frequency greater than 1.0E-06 (1.2E-06) in the Multi-Unit PRA. No individual sequence in the Unit 2 PRA was greater than 1.0E-06 in frequency.

- Support system failure initiators (specifically, loss of plant air; loss of raw cooling water; loss of unit preferred power; loss of either instrumentation and control bus 2A or 2B; or instrument tap failures) contribute 21% to the total CDF for the Multi-Unit PRA as compared to 2% for the Unit 2 Rev. 1A PRA.

- Transients with the reactor not isolated contribute 8% to the CDF for the Multi-Unit PRA as compared to 28% for the Unit 2 Rev. 1A PRA. Turbine trip and loss of feedwater are two specific examples of initiators in this group.

- Transients with the reactor isolated as a result of the initiating event (initiator) contribute 7% to the CDF in the Multi-Unit PRA as compared to 25% for the Unit 2 Rev. 1A PRA. Closure of the main steam isolation valves (MSIV) and turbine trip without bypass are two specific examples of initiators in this group.

- Large and medium loss of coolant accidents (LOCA) and interfacing systems LOCAs (i.e., when the boundary between a high and a low pressure system fails and the lower pressure system overpressurizes) make up only a small part (2%) of the total CDF for the Multi-Unit PRA as compared to 7% for the Unit 2 Rev. 1A PRA. The absolute change in contribution to CDF actually decreased slightly (5.0E-08) due to modeling refinements incorporated into the Multi-Unit PRA but not into the Unit 2 Rev. 1A PRA.

- Scenarios initiated by the inadvertent opening of one or more relief valves contribute only a small part (1%) to the CDF for the Multi-Unit PRA as compared to 3% for the Unit 2 Rev. 1A PRA. Three distinct initiators are considered: opening of one safety relief valve (SRV), opening of two SRVs, and opening of three or more SRVs.

A review of the top 25 sequences leading to core damage provides some insight as to the varying nature of core damage scenarios for the Multi-Unit PRA. Twenty-one of these sequences were initiated by "multiple unit" initiators (plant disturbances that have the potential to impact more than one operating unit). Specifically, these initiators that appear in the top 25 sequences are Internal Flood in the Turbine Building (eight scenarios), Loss of Offsite Power (eight scenarios), and Loss of Raw Cooling Water (five scenarios). Of the four "single unit" scenarios in the top twenty-five, three were initiated by vessel isolation events (Closure of All MSIVs, Loss of Condenser Vacuum, and Turbine Trip without Bypass). The remaining "single unit" scenario was initiated by a Loss of Feedwater.

The top two sequences are of a similar nature. Both are initiated by a "multiple unit" initiator (Internal Flood in the Turbine Building and Loss of Raw Cooling Water) followed by hardware failure of all four RHR pumps. The total frequency for these two sequences is 2.28E-06 (or about 8% of the total CDF). Hardware failure of the four RHR pumps is common to ten of the top 25 scenarios. The increased importance of RHR failures in the Multi-Unit study is primarily due to the reduced availability of the interunit RHR crossties for multiple unit events.

Table 1-2 summarizes the functional failure group contributions to core damage frequency.

Failure of heat removal is characteristic of three additional sequences of the top twenty-five. In two sequences, all four RHR heat exchangers fail, and in the remaining sequence, the RHR pumps fail due to the loss of pump cooling (specifically, loss of the fan coolers).

The third sequence overall is initiated by a loss of offsite power followed by hardware failure of all diesel generators. This is the most limiting station blackout sequence and represents about 2% of the total CDF. Two other sequences in the top twenty-five are related: sequence 7 is a loss of offsite power followed by failure of the Unit 1/Unit 2 fuel oil transfer pumps; and, sequence 22 is a loss of offsite power followed by hardware failure of the four Unit 1/Unit 2 diesel generators only.

Transient initiators followed by loss of two vital DC power supplies characterize six of the top twenty-five scenarios.

Transients initiators followed by inadequate EECW flow characterize three of the top twenty-five scenarios.

1.2.2 ANALYSIS OF INDIVIDUAL SEQUENCES

No single core damage sequence was found to dominate the total frequency of core damage. A large number of sequences make up the total CDF. Table 1-3 provides information on the distribution of core damage sequences across the frequency range for the Multi-Unit PRA as compared to the Unit 2 Rev. 1A PRA. The noted decrease in the number of sequences in the highest frequency category is due to the added complexity of the Multi-Unit PRA model that results in additional split fraction branching; e.g., more sequences but at lower values. See Appendix C for further details.

1.2.3 IMPORTANT OPERATOR ACTIONS

The importance of a specific operator action was determined by summing the frequencies of the sequences involving failure of that action. and comparing that sum to the total CDF. The importance is the ratio of that sum to the total CDF. This analysis provides a relative importance of the operator action. as it only determines the CDF impact of sequences that include the operator action. but does not distinguish whether the sequence failure is due to the operator action or the component failures.

Table 1-4 summarizes the important operator action failures ranked in order of their impact on the total CDF for the Multi-Unit PRA and the Unit 2 Rev. 1A PRA.

The operator actions to recover offsite electric power are not included in Table 1-4 because they are a complex function of the time available and the specific equipment failures involved. The offsite power recovery actions split fraction importance is shown in Table C-13.

1.2.4 IMPORTANT PLANT HARDWARE CHARACTERISTICS

An importance analysis of plant system failure modes to the total CDF was also performed. Only hardware failures involving the system itself are considered in Table 1-5, which provides a ranking in order of their impact on the total CDF for the Multi-Unit PRA. The Unit 2 Rev. 1A PRA impacts are also shown in the table for comparison.

The system importance measure is the fraction of the CDF involving partial or complete failure of the indicated system. These importance measures are not strictly additive because multiple system failures may occur in the same sequence. The importance rankings account for failures within the systems that lead to a plant trip, or failures that limit the capability of the plant to mitigate the cause of a plant trip. Consequential failures resulting from dependencies on other plant systems (e.g., the loss of drywell control air due to failure of reactor building closed cooling water) are not included in this importance ranking.

Care must be taken when comparing the results of the multi-unit PRA to the Unit 2 PRA as gauged by the PRA importance since this quantity is merely a relative measure. For example, RPS system failures appear in 7% of the core damage scenarios in the Multi-Unit PRA; the corresponding importance measure for the Unit 2 PRA is 20%. The relative nature of the measure is apparent when 38% of the multi-unit CDF 2.8E-05 (or 1.96E-06 is compared to 20% of the Unit 2 CDF equal to 7.6E-06 (1.5E-06). RPS is "more important" in absolute CDF impact in the Multi-Unit PRA, than in the Unit 2 PRA, a fact not communicated solely by the importance measures. What is apparent in Table 1-5 is that systems that are shared among the units to a significant degree (such as the diesel generators, RHRSW, and EECW) are relatively more important in the Multi-Unit PRA, as compared to the Unit 2 PRA.

1.3 SENSITIVITY ANALYSIS: EXTENDED DC POWER AND ALTERNATE INJECTION CAPABILITY

An analysis was performed to determine the risk reduction potential of the following:

- Using the diesel-driven fire protection system pump to inject water into the reactor vessel upon loss of AC power.

- Providing an alternative source of power to the SRVs solenoid valves to permit depressurization of the reactor following loss of AC power and depletion of batteries.

These improvements are evaluated in conjunction with the hardened wetwell vent because of the interaction each improvement has on the other. Although separately each has benefit, taken together they provide an open loop cooling mode for the vessel with a flow path from the diesel driven fire pump into the vessel, through the SRVs into the suppression pool, and out of the hardened wetwell vent.

During the preparation of the Unit 2 PRA Rev. 0 (issued September 1992), TVA recognized the potential of using the diesel-driven fire pump for vessel injection or debris bed cooling and subsequently prepared a system notebook for the high pressure fire protection system. However, the results have not yet been incorporated into the PRA model. The pump is capable of removing decay heat only after about 4 hours, therefore successful initial vessel level control (such as provided by HPCI or RCIC) is required. The SRVs are capable of extended operation in that a nitrogen gas supply can be aligned. DC control power to the solenoid valves is still required. The valves required to open for the hardened wetwell vent path also have a backup nitrogen gas supply and require DC power. These valves are located outside containment and can be locally operated via handwheels (prior to any postulated core damage).

The analysis was performed by running a number of sensitivity cases using the Multi-Unit and Unit 2 PRA models. For each model, two cases were evaluated. The difference between the two cases is that one assumes the availability of a supplemental DC power supply for the SRV solenoid valves, where the other case requires that offsite power be restored within 6 hours in order to provide a DC source for the SRVs. Both models used the same probabilistic values for the availability of the diesel driven fire pump and the manual actions required to align the pump flow path for vessel injection and remote manual operation of the hardened wetwell vent.

For the Multi-Unit PRA, the supplemental DC power case produces a CDF of 2.6E-06 from the loss of offsite power initiator, while the offsite power recovery required case produces a CDF of 2.7E-06. The baseline Multi-Unit PRA CDF, due to the loss of offsite power initiator, is 1.1E-05. The Rev. 1A Unit 2 PRA showed similar results with the supplemental DC power case CDF due to the loss of offsite power of 5.7E-07, while the offsite power required case produced a CDF of 5.9E-07. The baseline Unit 2 CDF, due to the loss of offsite power initiator, is 1.5E-06.

These results indicated that the use of the diesel-driven fire pump in an open loop mode of core cooling reflects a reduction in the computed core damage frequency due to the loss of AC power. Most of the gain in risk reduction is achieved through use of the diesel-driven fire pump and the hardened wetwell vent, which are already in place. Providing an alternate source of power to the SRVs is not warranted. This is especially so once consideration is given to the fact that the 4-hour battery depletion time is based on a conservative calculation and that relatively low current is required to maintain a solenoid open to allow an SRV to function. Based on this, TVA has no plans to provide an alternate source of power to the automatic depressurization system solenoid valves. Use of the diesel-driven fire pump as an alternate low pressure injection source is already discussed in the Emergency Operating Instructions.

Table 1-1. Initiating Event Group Contributions to Core Damage Frequency				
Initiating Event Category	Multi-Unit PRA		Unit 2 PRA	
	Mean CDF (per Reactor-Year)	Percentage of Total	Mean CDF (per Reactor-Year)	Percentage of Total
Loss of Offsite Power	1.1E-05	39%	1.5E-06	20%
Internal Floods	6.1E-06	22%	1.1E-06	15%
Support System Failures	5.8E-06	21%	1.7E-07	2%
Transients with Reactor Not Isolated	2.3E-06	8%	2.1E-06	28%
Transients with Reactor Isolated	2.0E-06	7%	1.9E-06	25%
Loss of Coolant Accidents	4.6E-07	2%	5.1E-07	7%
Stuck-Open Relief Valves	1.9E-07	1%	1.9E-07	3%
Interfacing Systems LOCAs	4.6E-08	<< 1%	4.6E-08	< 1%
Total	**2.8E-05**	**100%**	**7.6E-06**	**100%**

Table 1-2. Functional Failure Group Contributions to Core Damage Frequency

Accident Sequence Group	Multi-Unit PRA		Unit 2 PRA	
	Mean CDF (per Reactor-Year)	Percentage of Total	Mean CDF (per Reactor-Year)	Percentage of Total
Loss of RHR	1.1E-05	39	—*	—*
Degraded EECW	3.8E-06	14	—*	—*
Transient followed by Loss of Vital DC Power (250V Boards 2 and 3)	3.3E-06	12	2.5E-06	33
Anticipated Transient without Scram	1.7E-06	6	1.6E-06	21
Station Blackout	1.7E-06	6	—*	—*
Transient with Vessel at High Pressure	8.1E-07	3	5.2E-07	7
Blackout of Unit 1/Unit 2	5.0E-07	2	—*	—*

*Not calculated.

Table 1-3. Breakdown of Core Damage Sequences in Each Frequency Range				
Frequency Range (Events per Year)	Multi-Unit PRA		Unit 2 PRA	
	Number of Sequences	Percentage of CDF	Number of Sequences	Percentage of CDF
10E-06 to 10E-05	2	9	0	0
10E-07 to 10E-06	22	17	6	14
10E-08 to 10E-07	299	32	11	35
10E-09 to 10E-08	2,817	31	1,071	38
10E-10 to 10E-09	9,240	11	2.199	13
10E-11 to 10E-10	1,030	0	**	
10E-12 to 10E-11	13	0	**	
10E-13 to 10E-12	4*	0	**	

*The number of sequences in this range may be reduced by truncation. No initiator was considered with a cutoff less than 1.0E-10.
**Not determined.

Table 1-4. Browns Ferry Unit 2 Important Operator Actions			
Operator Action	PRA Importance		
	Multi-Unit	Unit 2 PRA	Surrogate Split Fraction
1. Manual Alignment of Redundant DC Source given Loss of Battery Board 2	0.144	0.390	CPREC3
2. Manual Depressurization of the Reactor Vessel using the SRVs	0.028	0.069	RVDZ2
3. Manual Alignment of Redundant DC Source given Loss of Battery Board 3	0.027	0.070	CPREC1
4. Manual Start of Residual Heat Removal/Core Spray	0.010	0.220	ORP2, ORP3
5. Align Alternate Injection to Reactor Vessel via the Unit 1 to Unit 2 RHR Crosstie	*	0.045	U12
6. Manual Alignment of RHR for Suppression Pool Cooling	0.017	0.006	OSP1
7. Manual Alignment of Swing RHRSW Pumps to EECW	*	0.008	OEE1
8. Start Standby Liquid Control System, given ATWS with the Reactor Vessel Isolated	0.011	0.035	OSL1
9. Start Standby Liquid Control System, given ATWS with the Reactor Vessel Not Isolated	0.008	0.020	OSL2
10. Reactor Vessel Level Control using RHR/Core Spray	*	0.011	OLP1
11. Failure to Prevent ADS during ATWS	*	0.007	OAD1
12. Manual Manipulation of the Turbine Bypass Valves to Depressurize Vessel	*	0.006	OBD1

*Less than 0.005.

Table 1-5. Browns Ferry Unit 2 Important Systems

System	PRA Importance*	
	Multi-Unit	Unit 2 PRA
Residual Heat Removal Service Water System	0.61	0.09
Diesel Generators	0.40	0.15
Residual Heat Removal System	0.38	0.2
250V DC Battery Boards	0.21	0.51
Emergency Equipment Cooling Water System	0.12	**
High Pressure Coolant Injection System	0.09	0.07
Reactor Core Isolation Cooling System	0.08	0.06
Reactor Protection System	0.07	0.2
Shared Actuation Instrumentation	0.04	0.07
Main Steam System Including Turbine Trip	0.04	0.08
Standby Liquid Control System	0.02	0.04
Control Rod Drive Hydraulic System	0.01	0.04
Condensate and Feedwater System	**	.01

*Fraction of CDF associated with sequences in which the failures occur in the indicated system.
**Less than 0.01.

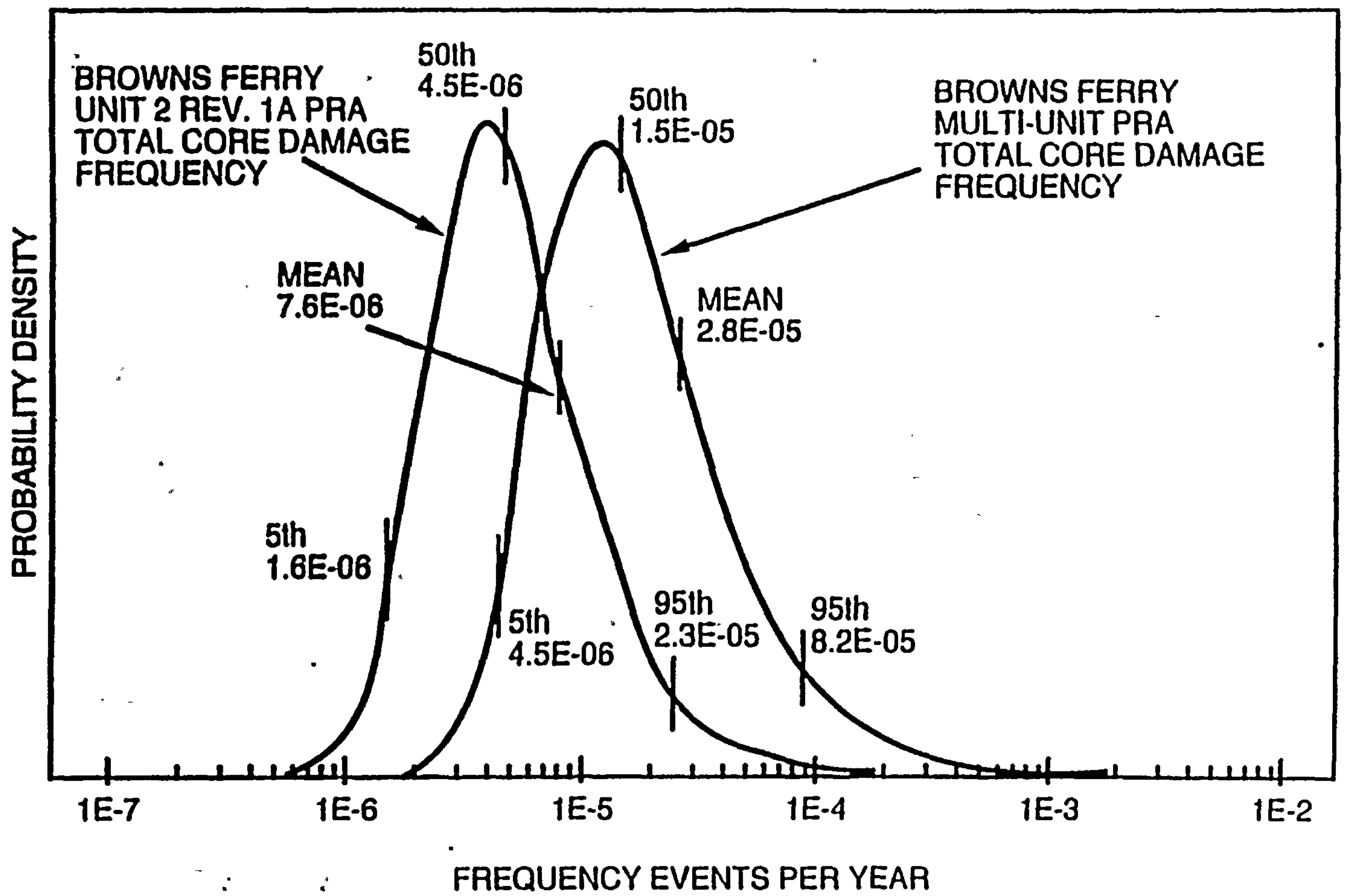

Figure 1-1. Total CDF for Browns Ferry Multi-Unit and Unit 2 Rev. 1A PRAs

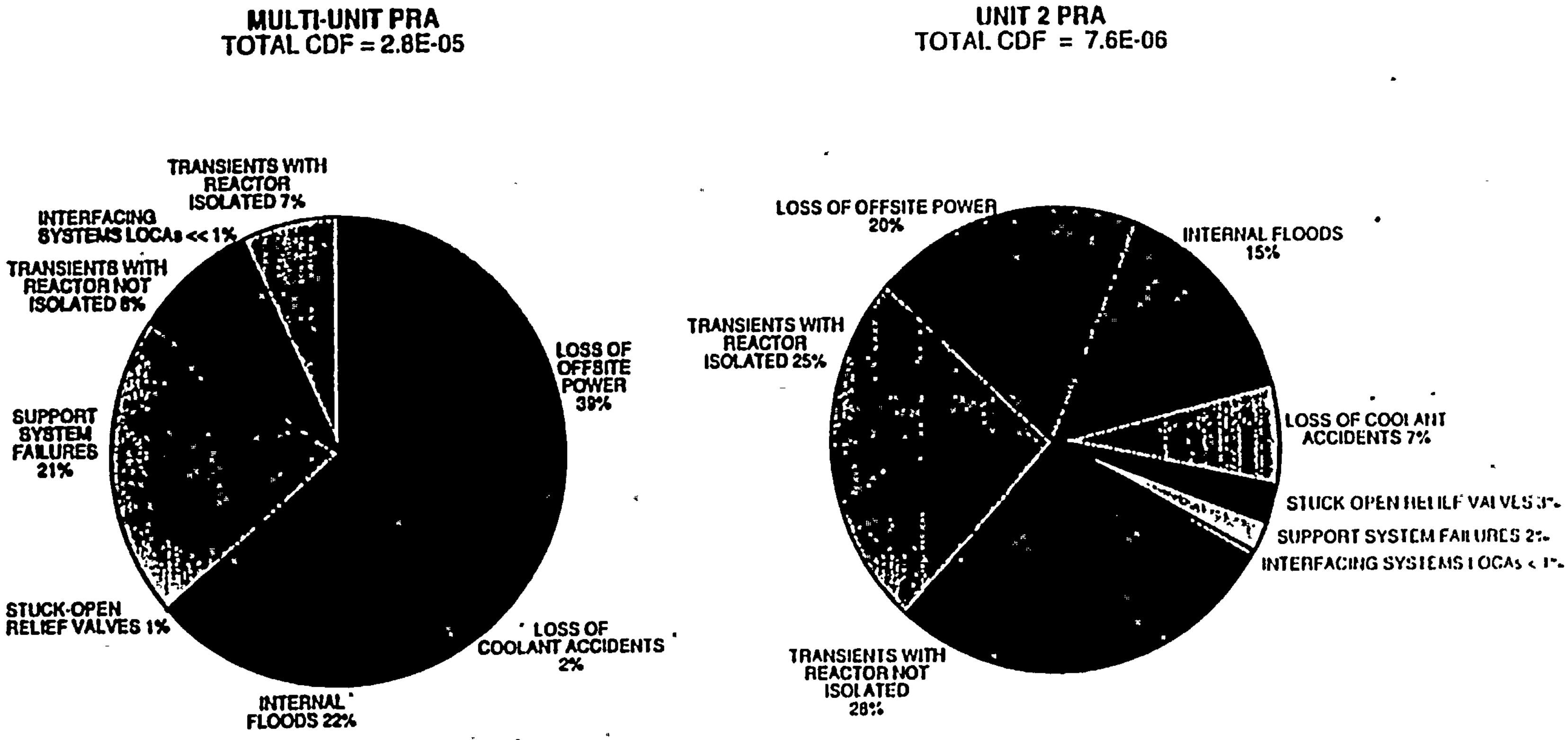

Figure 1-2. Browns Ferry CDF by Initiating Event Category