

Idaho National Laboratory Advanced Test Reactor Probabilistic Risk Assessment (PRA)

September 2012





History of PRA for the ATR

- > PRA studies began in the late 1980s
- ▶ 1989, ATR PRA published as a summary report
- ▶ 1991, ATR PRA full report
- ▶ 1994 and 2004 various model changes
- ▶ 2011, Consolidation, update and improvement of previous PRA work
- > 2012/2013, PRA risk monitor implementation





Purpose/Goal of the ATR PRA

- The PRA supports the ATR Updated Final Safety Analysis Report (UFSAR)
- The PRA provides sufficient information regarding either core or fuel damage (CDF or FDF) to enable ATR personnel to make risk informed decisions
- Improved performance in facility operation, testing, maintenance, training, and emergency procedures
- Ensure cost-effective approaches and the setting of priorities for plant upgrades and modifications, especially for risk reduction/system improvements
- Evaluate multiple overlapping contingent controls and equipment outages





PRA Applications

- Assess increases (or decreases) in risk as the plant changes due to equipment failures or maintenance activities (e.g., Risk Monitor)
 - Train Work Week Managers, Operations, and Engineering to use for evaluating work weeks, daily operations, and planning activities performed during operations and shutdown modes.
- Assistance in categorizing Structures, Systems, and Components (e.g. Safety Class, Safety Related)
- Changes to licensing basis (SAR, TSRs) such as completion times
- ▶ Inservice inspection and testing



ATR PRA Modules



- ▶ Power Operations (Includes Power Operations greater than ~3MW)
- ► Shutdown and Fuel Handling (Includes operating states less than ~3MW)
- ▶ Internal Flood
- Internal Fire
- Seismic
- ATR Confinement



Power Operations Module Safety Workshop Power Operations Module Working Together to Enhance Nuclear Safety

- ▶ 40 initiating events (e.g., cask drop, small LOCA)
- > 51 system functional criteria (e.g., forced flow for 30 minutes, vessel venting)
- ▶ 86 fault trees (e.g., core emergence makeup, secondary heat removal)
- ▶ 2680 basic events (e.g., cooling pump fails to run, emergency pump fails to start, operator fails to actuate valve)
- > 24 ATR systems modeled (e.g., deep wells, plant protection system)
- Meets ASME/ANS Standard RA-Sa-2009 capability category II criteria (All 6 modules)
- Independently reviewed by highly experienced PRA experts from the commercial power industry (All 6 modules)
- Forms the basis for all other ATR PRA Modules



Shutdown and Fuel Handling Nuclear Safety Workshop Working Together to Enhance Nuclear Safety Module

- Replicated Power Operations Module 6 times and modified each to specifically represent each plant operating state.
- Plant Operating States (POSs) modeled (original POSs 5-7 subsumed in other POSs)
 - POS 1, Transition From Pressurized with EFIS in Auto to Depressurized with EFIS in manual
 - POS 2, Depressurized Shutdown, Vessel is Vented, Fuel in the Core
 - POS 3, Depressurized Shutdown, Actively Transferring Fuel Into or Out of the Reactor
 - POS 4, Reactor Defueled
 - POS 8, Transition From Depressurized with EFIS in Manual to Pressurized with EFIS in Auto
 - POS 9, Low Power Operation, Startup and Transition to Power Operations, PCS >100 psig, Automatic EFIS
 - POS 10, Power Operations Separate Module
- Constructed module such that 1 flag (logic switch) can be set and then solve any individual POS





Internal Flood Module

- ▶ 296 initiating events (e.g., fire protection pipe flood, gland seal spray in pump motor room, demineralized water spray in second basement)
- > System functional criteria of power operations module
- Modified power operations module fault trees to consider flood and spray damage
- Calculations to determine time to flood critical equipment depending on the piping system flow and location of the assumed break or spray
- Consideration of penetrations (e.g., ventilation ducts, cable trays, drain gutters, door jam space, stair wells)



Internal Fire Module



- ▶ 150 initiating events (screened many more)
- System functional criteria of power operations module
- Modified power operations module fault trees to consider damage caused by fire (e.g., transient fire, cable tray, running motor, high energy arc faults) and possible fire protection actuation.
- Fires modeled via CFAST considering zones of influence and smoke layers resulting in time to reach combustion of overhead components and fire sprinkler actuation





Seismic Module

- > Site specific seismic hazard curve.
- Specialized event tree for the unique nature of seismic events
- ▶ Modified power operations module fault trees to consider damage caused by seismic events (e.g., both random faults and seismic damage ~ 300 plant specific seismic fragilities are considered)
- Sensitivity studies for the site hazard curve and acceleration specific variations





Confinement Module

- Initiating events derived from CDF and FDF power operations module results
- Considers timing, material, and component inactions during core melt progression
- Individual sequences resulted in 22 source terms
- Considers the specific initiating event regarding which systems may still be functional (e.g., firewater injection, building spray, power supplies) including whether the initiating event causes a confinement breach (e.g., drop events)
- Release progression throughout the building and evaluates confinement bypass (large early release fraction)
- Sensitivities studies for ventilation failures (e.g., dampers) and whether ventilation fans continue to run when they shouldn't



Core DamageFrequencies



ATR PRA Module	Point Estimate of CDF (1/yr)	Mean of CDF (1/yr)		
Power Operations	5.1E-06	5.5E-06		
Fuel Handling and Shutdown (in	radiated fuel in the vessel, in transit,	or stored in the canal)		
 Depressurized/vented 	2.1E-07	1.8E-07		
•Depressurized moving fuel	1.7E-07	1.6E-06		
•Reactor Defueled	6.6E-07	5.9E-07		
•Transition from depressurized to pressurized	2.5E-05	2.4E-05		
•Low power operation	2.1E-07	2.0E-07		
Internal Flood	8.4E-06	9.8E-06		
Internal Fire	3.0E-05	2.8E-05		
Seismic	4.1E-05	2.7E-03		
Level 2 (LERF)	1.1E-06	1.1E-06		



Dominant Full-Power Accident Sequences



Event Description	Frequency/year	% Total
Canal draining from non-cask drop	1.1 E-06	21.4%
	(1 in .9 million)	
Large LOCA	1.0 E-06	19.3%
	(1 in 1 million)	
Forklift load drop	9.0 E-07	17.5%
	(1 in 1.1 million)	
Loss of commercial power	5.1 E-07	10%
	(1 in 1.96 million)	





Insights

- There are no dominant sequence groups indicating mitigation systems are appropriate
- Environmental aspects of important components need to be evaluated to credit their potential safety function (e.g., fire water spray on switchgear and digital systems)
- Operating procedures and training emphasizing the importance of vessel venting and proper operation of firewater injection could be improved
- Replacing open cable trays with solid bottom cable trays above some buses could provide an effective thermal barrier
- Buildings housing support equipment are seismically weak and should be upgraded or equipment moved
- Upgrade unqualified primary piping (completed)
- Confinement release is dominated by load drop events and most large releases are due to stored fuel vs. the core



Recent Uses of the PRA



- Modeled secondary coolant system component replacement during operations to show insignificant change in risk
- Evaluated broken firewater valve to determine its importance in reactor startup
- Evaluated various configurations of running diesel generators to determine allowed outage time (completion time)
- Evaluated station blackout (similar to 10 CFR 50.63 and NRC Regulatory Guide 1.155)
- Determined risk significant components in support of system health program
- Ongoing evaluations of various design options for converting plant electrical systems to commercial power with diesel/battery backups



ATR Risk Monitor Equipment Nuclear DOE+ NNSA + NRC + DNFSB + AEA

.Working Together to Enhance Nuclear Safety

Selection

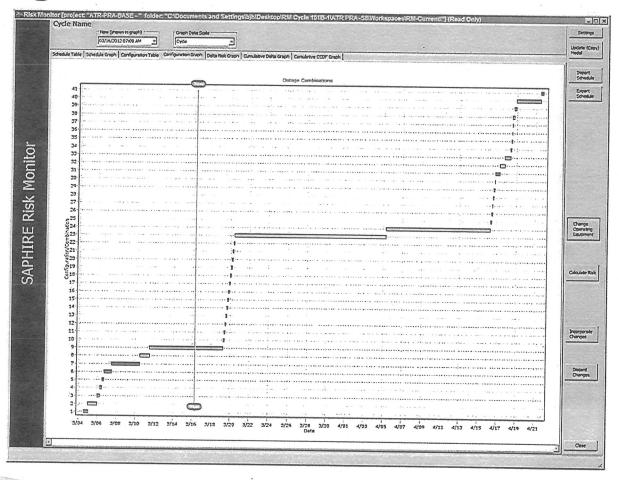
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lick on a system to view its a	ssociated components.	Click	the new state of a compo	nent.							
BBP - Battery Backed Power	HPA - High Pressure Air		Component	Component Desc	State (* indica	stes given current state)	Start Time End Time	Events	Change Set	C).	
CAN - Canal Structure and Systems	HVB - Realing and Verdiction		BBP-BAC-00000E82	Regulating red control distribution panel 120/2009/670-E-02	#Energized[]	CDeenergized		BBP-BAC-SS-00000E82-0000#B			
CCX - Control Complex	VS - Instrument and Plant Air	-	BBP-BAC-00000EB3	Regulating red control distribution panel 120/200y 670-E-83	#Energized[]	CDeenergized	www.martin	BBP-BAC-S3-00000E83-0000*B			
CDP - Comm or	LDW - Low Pressure	<u> </u>	BBP-BAC-00000IP1	Instrument channel #1 breaker panel IP-1	# Energized[7]	C Deenergized		BBP-BAC-FF-00000IP1-0000-B	tree! store in the second of all additional and	tem when of each of	
Comm/Diesel Power	Demin, Water	F	BBP-BAC-000000P2	Instrument channel #2 breaker panel #2-2	«Energized[*]	CDeenergized		BBP-BAC-FF-000001P2-0000=B			
CMU - Canal Makeup Water	PCS - Primary Coolant System	F	BBP-BAC-00670E05	208/120 V panel 670-E-85	#Energized[]	Deenergized		DDP-BAC-FF-00670EBS-0000=D			
DCP - Diesel/Commercial	PPS - Plant Protective System	F	BBP-BAC-00671E45	Breaker panel 671-E-45 (RPU power supply)	#Energized[7]	CDeenergized	\$ 100,000 to the state of the s	BBP-BAC-FF-00671E45-0000=B	ALMAN AND DESCRIPTION OF THE PARTY OF THE PA	***************	
DCS - Distributed Control		F	BBP-BAC-0670E115	Utility UPS panel 670-E-115	≪Energized[7]	C Deenergized		DBP-BAC-FF-0570E115-0000=B DBP-BAC-FF-0570E115-INIT=B			
System System	PWL - Experiment Loops	Г	BBP-BAC-0670E116	Upmy UPS panel 670-E-116	#Energized]	CDeenergized		DBP-BAC-FF-0570E116-0000=B			
DGP - Diesel Generators	RAW-Raw Water	Г	BBP-BAC-0670E117	Instrument UPS panel 670-E-117	#Energized[]	CDeenergized		BBP-BAC-FF-0870E117-0000=B BBP-BAC-FF-0870E117-INIT=B			
DWP - Deep Well Pumps	PMS - Reduction Monitoring	F	BBP-BAC-0670E445	DC3 power panel 670-E-446	# Enorghad(1)	CDeenergized		BBP-BAC-FF-0570E445-0000=B			
DWP - Deep (Yes Fumps	Cont	F	BBP-BAC-0570E455	UPS power panel 670-E-456	#Energized[*]	CDeenergized		BBP-BAC-FF-0570E456-0000=B			
EIS - Ernerg, Firewater Injection	RRS - Reactor Reverse System	F	BBP-BAT-0001CE30	Battery 1C-E-30 for LOCS UPS 670-E-63	#In service[7]	C003		BBP-BAT-LP-0001CE30-0000=B			
COG - Fire Copplication	RSS - Reader Shutdown	Г	DDP-DAT-00509E39	Battery 609-E-39	afin service[7]	roos		DBP-BAT-LP-00609E39-0000=B			
CO VIII GOMING MILITA	System	L	DDP-DAT-00570E50	Dattery bank 670-E-50	#In service[7]	C003		DDP-DAT-FF-00670E50-0000=D	***		
FWS - Firewater Supply	SC3 - Secondary Cooling	L	BBP-BAT-00670E59	Battery bank 670-E-50	#In service[7]	C003		GBP-BAT-FF-00670E50-0000=B			
7,7,7	, ,	F	DBP-BAT-OPPSE11A	PPS panel E-14A battery E-11A	will Selvice[.]			BBP-BAT-FF-0PPSE11A-0000=8			
GSW - Gland Seal Water	UCW - Utility Cooling Water	Г	BBP-BAT-OPPSE11B	PPS panel E-148 battery E-118	#In service[*]	C003		BBP-BAT-FF-0PP3E118-0000=B		700	
HDW - High Pressure Demin. Water	Q Search	Related Diegrams					•				



ATR Risk Monitor Cycle Configurations

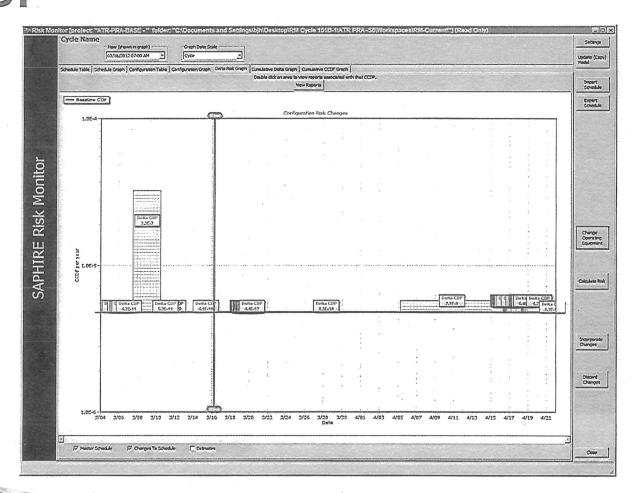


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ATR Risk Monitor Example CCDF





ATR Risk Monitor Example Cumulative CDP



