ADVANCED TEST REACTOR PROBABILISTIC RISK ASSESSMENT METHODOLOGY AND RESULTS SUMMARY

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Published January 1992

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**Prepared for the
U.S. Department of Energy
Office of Energy Research
Under DOE Field Office, Idaho
Contract No. DE-AC07-76ID01570**

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EXECUTIVE SUMMARY

The Advanced Test Reactor (ATR) probabilistic risk assessment (PRA) Level 1 report documents a comprehensive and state-of-the-art study to establish and reduce the risk associated with operation of the ATR, expressed as a mean frequency of fuel damage. The ATR Level 1 PRA effort is unique and outstanding because of its consistent and state-of-the-art treatment of all facets of the risk study, its comprehensive and cost-effective risk reduction effort while the risk baseline was being established, and its thorough and comprehensive documentation.

The PRA includes many improvements to the state-of-the-art, including the following: establishment of a comprehensive generic data base for component failures, treatment of initiating event frequencies given significant plant improvements in recent years, performance of efficient identification and screening of fire and flood events using code-assisted vital area analysis, identification and treatment of significant seismic-fire-flood-wind interactions, and modeling of large loss-of-coolant accidents (LOCAs) and experiment loop ruptures leading to direct damage of the ATR core.

The recently completed Level 1 analysis results are generally lower than the results for the other U.S. Department of Energy reactor facility PRAs that include both internal and external events. This is because of some unique characteristics of the ATR that are beneficial to safety (such as strong negative coolant temperature reactivity feedback; large coolant inventory; and an uninsulated, low temperature and moderate pressure, all stainless steel coolant system) combined with refined risk estimates that take advantage of those characteristics and the aggressive ATR risk reduction actions. The total ATR Level 1 mean fuel damage frequency is as follows:

Internal initiating events	2.4E-5/yr
External initiating events	2.5E-5/yr
Total Level 1 PRA	4.9E-5/yr

Table i provides estimated mean fuel damage frequencies for both the internal and external initiating event groups. The low total fuel damage

frequency for ATR with no overly-dominating sequences indicates that the recommended risk reduction efforts and risk estimate refinements are successful. Emphasis should now be placed on maintenance of this low risk profile and application of the results for effective ATR risk management.

The principal objectives of the ATR PRA program have been, since the inception of the program, to apply the results and insights from the PRA to improve the operation and safety posture of the ATR. The PRA has been used to evaluate and prioritize potential facility or operational changes and questions or concerns about the facility and its operation. The comprehensive ATR PRA models and results have already been applied to eliminate some previously planned facility upgrades that were shown to have a negligible effect on reducing fuel damage risk and to define a more cost-effective approach for other projects that would still meet risk reduction objectives. It is expected that risk management applications of the ATR PRA will result in cost savings to ATR (in addition to avoided costs of a severe accident) that will more than cover the costs associated with development of the PRA.

This document provides an overview of the techniques used to perform the analysis and a summary of the overall results of the PRA. Event trees, fault trees, and cutsets are not provided in this document. Also, specific identification of the equipment failures involved in the event tree sequences have been removed in order to make the document available for broader distribution.

Table 1. Level 1 fuel damage frequencies by initiator

Initiating Event	Mean Fuel Damage Frequency/Yr	Percent Contribution
Internal Events		
Loss of commercial power (LCP)	9.5E-06	19.3
Loss of diesel power (LDP)	6.7E-06	13.6
Loop reactivity (high range) (RLH)	3.7E-06	7.4
Small LOCA (S12)	1.5E-06	3.1
Reactor shutdown (RSD)	6.7E-07	1.3
Direct-damage large LOCA (DLC)	5.0E-07	1.0
Loss of diesel-commercial power (LDC)	3.6E-07	0.7
Fast reactivity ramp (high range) (RFH)	3.3E-07	0.7
Low PCS pressure (LPP)	2.6E-07	0.5
Low PCS flow (LPF)	2.5E-07	0.5
High coolant pressure (HCP)	1.7E-07	0.3
Others	<1.0E-07	0.5
Internal Event Total	2.4E-05	49.0
External Events		
Seismic	1.9E-05	38.1
Volcanic activity	2.8E-06	5.7
Extreme wind	1.4E-06	2.8
Internal flooding	1.1E-06	2.2
Internal fire	1.0E-06	2.0
Others	<1.0E-07	0.2
External Event Total	2.5E-05	51.0
Level 1 Total	4.9E-05	100.0

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ACRONYMS

ACNFS	Advisory Committee on Nuclear Facility Safety
AHT	high temperature loop overheating
AHTL	Advanced High Temperature Loop
ATR	Advanced Test Reactor
ATWS	anticipated transient without scram
CLOFA	complete loss of flow accident
DLC	direct-damage LOCA
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor II
FCV	flow control valve
FIS	firewater injection system
FMEA	failure modes and effects analysis
HCP	high PCS pressure
HCR	human cognitive reliability
HDW	high-pressure demineralized water
HRA	human reliability analysis
IE	initiating event
INEL	Idaho National Engineering Laboratory
IPT	in-pile tube
ISF	interfacing system LOCA (LDW)
ISL	interfacing system LOCA (FIS)
LAC	loss of all ac power
LCP	loss of commercial power
LDC	loss of diesel-commercial bus

ACRONYMS (CONTINUED)

LDP	loss of diesel power
LDW	LDW purge
LDW	low-pressure demineralized water
LFS	local faults
LHS	loss of heat sink
LIA	loss of instrument air
LLC	large LOCA
LOCA	loss-of-coolant accident
LOF	loss of flow
LPF	low PCS flow
LPP	low PCS pressure
MCC	motor control center
NRC	Nuclear Regulatory Commission
NS	neck shim
OSCC	outer shim control cylinder
PCS	primary coolant system
PDS	plant damage state
PPS	plant protective system
PRA	probabilistic risk assessment
PWR	pressurized water reactor
QPS	quadrant power setback
RFH	reactivity insertion (fast ramp)
RIL	large reactivity insertion
RLH	reactivity insertion (loop ramp)

ACRONYMS (CONTINUED)

RLP	lobe power imbalance
RPF	very large reactivity insertion
RSD	normal/spurious reactor shutdown
RSH	reactivity insertion (slow ramp)
THERP	Technique for Human Error Rate Prediction
TM	test and maintenance
TRA	Test Reactor Area
UCNI	unclassified controlled nuclear information
UCW	utility cooling water
UOR	unusual occurrence report
VVS	vessel vent system

ADVANCED TEST REACTOR PROBABILISTIC RISK ASSESSMENT METHODOLOGY AND RESULTS SUMMARY

1. RISK ASSESSMENT METHODOLOGY

The Advanced Test Reactor (ATR) Level 1 probabilistic risk assessment (PRA) report¹ documents a comprehensive and state-of-the-art analysis to establish and reduce the fuel damage risk associated with operation of the ATR. A fuel damage event at ATR would severely disrupt Department of Energy (DOE) nuclear fuels testing (for the Navy, the New Production Reactor, and perhaps future space nuclear reactors), in addition to dealing a severe blow to the reputation of the Idaho National Engineering Laboratory (INEL). Therefore, establishing an accurate risk baseline and cost-effective risk management are important goals.

The three-and-one-half-year Level 1 PRA effort, costing approximately \$2,500,000, has been an outstanding success. Not only was the risk baseline established in a comprehensive and credible manner, but also the fuel damage frequency was reduced by approximately a factor of 20 to its present level of $4.9\text{E-}5/\text{yr}$ through a concerted and focused effort to consider plant-specific characteristics and to identify cost-effective plant design and procedures changes to reduce risk. The present fuel damage frequency of $4.9\text{E-}5/\text{yr}$ is one of the lowest among all DOE and commercial nuclear reactors. The efforts at ATR now can be focused on appropriate risk management to maintain this low level of risk and completing the Levels 2 and 3 portions of the PRA.

The ATR PRA established the risk of ATR operation by examining all significant upset events that would require shutdown in order to protect the reactor core. Such upset events are termed initiating events and include losses of electrical power, primary or secondary coolant abnormalities, reactivity transients, experiment loop problems, and a wide variety of external events (fires, seismic events, floods, high winds, volcanism, and

others). Given such initiators, ATR is designed to respond in a variety of diverse ways to reach a stable shutdown with no fuel damage. The PRA identified and modeled such diverse shutdown paths (through the use of event trees) and quantified the probabilities of combinations of safety system failures and human errors (using fault trees) resulting in fuel damage. Specific accident analyses were used to define success and failure paths for sequences involving complex accident phenomenon. Results are expressed as a set of fuel damage sequences, each involving an initiating event and combinations of safety system failures and human errors, and associated frequencies. The total mean fuel damage frequency for ATR is the sum of the individual mean fuel damage sequence frequencies.

The ATR PRA Level 1 effort is unique and outstanding because of its consistent and state-of-the-art treatment of all facets of the study, its comprehensive and cost-effective risk reduction effort while the risk baseline was being established, and its thorough and comprehensible documentation. The PRA includes many improvements to the state of the art, including the following: establishment of a comprehensive generic data base for component failures, treatment of initiating event frequencies given significant plant improvements in recent years, performance of efficient identification and screening of fire and flood events using code-assisted vital area analysis, identification and treatment of significant seismic-fire-flood-wind interactions, and modeling of large loss-of-coolant accidents (LOCAs) and experiment loop ruptures leading to direct damage of the ATR core. With regard to the comprehensive and cost-effective plant upgrade effort, the Advisory Committee on Nuclear Facility Safety (ACNFS) PRA Subcommittee had the following comment concerning the ATR effort: "...[a comprehensive identification of plant upgrades is] sort of everybody's wish list, but in fact ... what it looks like, which is interesting, is that a number of these improvements have either been completed or are in the process of being implemented."² The resulting report and risk reduction effort have set a standard for other DOE and commercial nuclear reactor PRAs.

Throughout the ATR PRA study, quality has been of great concern. During the early phases of the study, when the plant models were being constructed,

several review committees involving both ATR and PRA personnel were formed to ensure that the models accurately reflect the plant as it exists. After the plant models were constructed, the internal review process was streamlined to include independent reviews by PRA and ATR personnel. Also, throughout the effort, an external peer review was conducted by Mr. P. R. Davis of PRD Consulting. Mr. Davis has also been involved in reviews of most other DOE reactor PRAs. Several external review committees conducted periodic reviews. These committees included the PRA Subcommittee of the ACNFS, the EG&G Idaho Nuclear Safety Review Group, and the ATR Reactor Safety Analysis Program External Review Board. All groups included internationally-recognized PRA experts. Finally, an earlier version of the PRA was reviewed in detail by PRA experts at Brookhaven National Laboratory. Comments, suggestions, and criticisms from all of these review efforts were considered and where appropriate have been incorporated into the final report issued in September 1991.

The methodologies used in the study are outlined in this section, following the structure outlined in Figure 1.

1.1 PROJECT INTEGRATION

The Level 1 (fuel damage risk) ATR PRA was performed during the period from February 1987 through September 1991. The study includes external events (seismic, flooding, fire, and others), but does not include Level 2 and 3 analyses (severe accident analysis, confinement response, and radionuclide release to and transport through the atmosphere). These extensions to the PRA are in progress and will be incorporated into this report as they are completed. The Level 1 PRA includes a detailed analysis of the ATR as it existed in May 1989. Plant risk was analyzed for both high and low (start-up and shutdown) power operation. In addition, risk from the fuel storage canal was considered.

Figure 1. ATR PRA task flowchart.

1.2 PLANT DESCRIPTION

ATR is located at the Test Reactor Area (TRA) within the boundaries of the Idaho National Engineering Laboratory (INEL). ATR, in operation since 1968, is a 250-MW_{th} nuclear reactor designed to study the effects of intense irradiation on samples of reactor materials. A comparison of ATR to a typical commercial pressurized water reactor (PWR) is presented in Table 1. ATR is significantly different from a PWR, having a smaller core, higher power density, lower primary coolant system pressure and temperature, greater ratio of coolant weight to power, and a confinement structure rather than a containment building. The confinement structure is designed to be a barrier to radionuclide release to the atmosphere, similar to a containment building for a PWR; however, the confinement structure has a higher leakage rate and lower overpressurization structural failure limit. Also, the confinement does not have to be vented during a severe accident because of the low pressure and low power of ATR.

1.3 SYSTEMS ANALYSIS

1.3.1 Plant Familiarization

The ATR PRA is based generally on the plant design and procedures existing as of May 1989. However, several plant modifications planned for 1991-1993 have been incorporated into the models. These modifications include diesel pit sealing and additional drains and several seismic upgrades.

1.3.2 Initiating Event Identification and Quantification

Initiating events (IEs) are defined as anomalous events that require a plant shutdown and challenge the plant safety systems. Given an IE, subsequent failures in safety systems could result in fuel damage and radionuclide release from the core. (For ATR, several very low frequency initiators exist, which may result in initial fuel damage even with success of the safety systems. These initiators are termed direct-damage initiators.)

Table 1. Comparison of ATR and PWR characteristics

Reactor Operating Conditions	ATR	PWR
Power (MW_{th})	250	2000-4000
Operating pressure (psig)	355	≈ 2250
Inlet temperature ($^{\circ}F$)	125	≈ 550
Outlet temperature ($^{\circ}F$)	170	≈ 600
Primary coolant flow rate (gpm)	<48,000	$\approx 300,000$
Primary coolant weight (lb)	600,000	$\approx 450,000$
Primary coolant weight/thermal power (lb/MW)	2,400	170
Decay heat (MW)	13 @ 10 s 1.3 @ 1 day	135 @ 10 s 19 @ 1 day
<u>Fuel</u>		
Total uranium weight (lb)	89	180,000
Enrichment	93% U-235	2-4% U-235
Configuration	48-in. long Al plates attached to side plates	Zirc rods containing stacked pellets
Matrix	UAl _x	UO ₂
Fuel temperature ($^{\circ}F$)	430	2000-3000
Fission product inventory	<60 days operation @ 250 MW	10 times ATR

The methodologies used to identify the ATR IEs included both the comprehensive engineering evaluation and master logic diagram approaches.³ For the comprehensive engineering evaluation, three subtasks were performed: review of ATR safety documents and unusual occurrence reports (UORs), failure modes and effects analyses (FMEAs) of major plant systems, and structured input from PRA-independent ATR safety analysts. The resulting list of ATR IEs is presented in Table 2. The IEs are grouped into 24 categories, based on similarities in plant response. Most of the IE categories were quantified using ATR experience, as indicated in Table 3. A Bayesian update of a noninformative prior was used to account for the ATR data. All of the IE category frequencies are listed in Table 4. Loss of all ac power, loss of the diesel-commercial bus, and loss of diesel power IE categories were evaluated by requantifying the appropriate system event trees. The interfacing loss-of-coolant accidents (LOCAs) were quantified based on the ATR piping and valve arrangements. For the large LOCA and large reactivity insertion (experiment loop rupture) events, the frequencies are based on analyses using ATR design and operating conditions.

Estimating frequencies for large LOCAs and experiment loop ruptures is a difficult task. No such events have occurred in U.S. commercial nuclear power plants or U.S. Department of Energy (DOE) Class A reactors. Therefore, the approach taken for the ATR PRA was to estimate such frequencies by several philosophically different approaches and combine the results to obtain a suggested frequency and distribution for each event. The philosophically different approaches included the following: statistical estimation, based on historical data; phenomenological analysis of dominant failure mechanisms using the probabilistic fracture mechanics code PC-PRAISE; and expert judgment elicitation.

Table 2. ATR initiating event categories

<u>Initiating Event Category</u>	<u>Initiating Event(s)</u>
1. Normal/spurious reactor shutdown (RSD)	Spurious shutdowns and normal plant shutdowns
2. Low Primary Coolant System (PCS) flow (LPF)	<p>Butterfly valve closed against stop by operator, control, or valve failure</p> <p>Two PCS pumps tripped by 4160 Vac undervoltage relay failure</p> <p>Combinations of two PCS pumps or two PCS pump valve failures</p> <p>Flange or sliding joint leakage in outlet pipe in vessel (core flow bypass)</p> <p>Outlet pipe ruptures within the vessel</p> <p>One emergency pump fails or is turned off (momentary loss of flow) in depressurized operation</p> <p>Both emergency pumps fail (long-term failure, depressurized operation)</p> <p>Butterfly valve slams closed when the shaft fails or the valve is closed and the stop fails</p> <p>Emergency pump discharge valve fails closed</p> <p>Emergency pump suction valve fails closed during depressurized operations</p> <p>Loss of ventilation to primary pump motors</p> <p>Core flow bypass when in depressurized (open head) operation</p>
3. Low PCS pressure (LPP)	<p>Pressure control valve PCV-1-1 opens (operator or controller failure)</p> <p>Flow control valve FCV-1-8 closes (operator or controller failure)</p> <p>Excessive surge tank vent into the degassing tank</p> <p>Pressurizing pumps fail</p>

Table 2. (continued)

<u>Initiating Event Category</u>	<u>Initiating Event(s)</u>
3. Low PCS pressure (LPP) (continued)	<p>Flow control valve FCV-1-8 fails closed</p> <p>Bypass demineralizer purge valve opened wide</p> <p>Loss of low-pressure demineralized water (LDW) makeup</p> <p>Degassing tank failure or LDW failure leading to a loss of makeup</p>
4. High PCS pressure (HCP)	<p>Pressure control valve PCV-1-1 closes (operator or controller failure)</p> <p>Flow control valve FCV-1-8 opens (operator or controller failure)</p> <p>Second pressurizing pump is started</p> <p>Butterfly valve, pump discharge valve, or check valve slams closed (water hammer)</p> <p>Emergency pump loop check valve fails open or fails to close on primary pump startup (local overpressure)</p> <p>Extreme diesel overspeed (>15%) causes significant pressurizing pump motor overspeed and high pressurizing flow</p> <p>High primary pump discharge pressure due to extreme commercial power overfrequency</p> <p>Startup of a primary coolant pump against a closed discharge valve with high pump suction pressure (local overpressure)</p> <p>Primary flow control (butterfly) valve slams closed when shaft fails or valve is closed and stop fails</p> <p>Hydrogen explosion in the degassing tank after depressurization</p>

Table 2. (continued)

<u>Initiating Event Category</u>	<u>Initiating Event(s)</u>
5. Loss of heat sink (LHS)	<p>Secondary system pump M-4 variable frequency drive failure</p> <p>Failure of all secondary coolant pumps (long-term)</p> <p>Failure of cooling tower fans (long-term)</p> <p>Loss of cooling tower basin water to drains, or makeup failure</p> <p>Cooling tower failure (icing, or flow bypassed to the basin)</p> <p>Secondary flow control valve fails closed</p> <p>High instrument air pressure and pressure regulator failure, or other secondary flow control failures</p> <p>Secondary coolant system rupture upstream of heat exchangers</p>
6. Loss of commercial power (LCP)	<p>Loss of commercial power or severe commercial power undervoltage and underfrequency</p> <p>Momentary complete loss of flow accident (CLOFA) due to loss of commercial, diesel, and utility dc power</p> <p>Long-term CLOFA due to a long-term loss of commercial power and failure of both emergency pumps, or long-term loss of all power sources</p> <p>Momentary CLOFA due to loss of commercial power and emergency flow looping through slow closing primary coolant pump check valves</p> <p>Long-term CLOFA due to a loss of commercial power and emergency flow looping through stuck-open PCS pump check valve</p>
7. Loss of all ac power (LAC)	Extended diesel and commercial power loss
8. Loss of diesel-commercial bus (LDC)	Loss of diesel-commercial power bus

Table 2. (continued)

<u>Initiating Event Category</u>	<u>Initiating Event(s)</u>
9. Loss of diesel power (LDP)	A special case of extended loss of commercial, diesel, and utility dc power
10. Loss of instrument air (LIA)	Loss of instrument air
11. Small LOCA (0.5-in. to to 5.3-in. diameter break) (S12)	<p>Open valve outside radiographic limits or at the radiographic boundary, including an open relief valve</p> <p>Rupture outside radiographic limits (<3-in. diameter)</p> <p>Multiple drain valves opened</p> <p>Surge tank vent rupture</p> <p>Rupture/crack in PCS piping inside radiographic limits</p> <p>Two relief valves stick open (one relief valve sticking open is covered by the first event)</p> <p>2-in. firewater line rupture at the bottom head</p> <p>Pressure tube penetration or bottom head closure plug failure</p> <p>Dummy in-pile tube (IPT) lifting</p> <p>Tube failures in the PCS heat exchangers</p>
12. Large LOCA (>5.3-in. diameter)(LLC)	<p>Rupture/crack in PCS piping inside radiographic limits</p> <p>Vessel flange closure failure (nozzle trench or top dome)</p>
13. Direct-damage large LOCA (>5.3-in. diameter) (DLC)	Rupture in PCS piping resulting in direct damage to the core
14. Interfacing system LOCA (LDW line)(ISL)	LDW line valves fail open

Table 2. (continued)

Initiating Event Category	Initiating Event(s)
15. Interfacing system LOCA [firewater injection system (FIS) line] (ISF)	Firewater valves open and one check valve fails (this event automatically implies a failure of reactor building confinement via PCS water injection into the firewater system)
16. Reactivity insertion (fast ramp)(RFH) rods) from $10^{-10} N_F$	<p>Withdrawal of all rods [outer shim control cylinders (OSCCs), neck shims (NSs), and safety</p> <p>Withdrawal of all OSCCs and NSs and regulating rod cock from $10^{-3} N_F$</p> <p>Movement of one reflector block toward the core with limiting anticipated reactivity excursion</p> <p>Limiting anticipated ramp plus the limiting anticipated step (i.e., reflector movement) (0.25\$ step) plus the 0.5-in. blowdown</p> <p>Withdrawal of all OSCCs and NSs from $10^{-5} N_F$ (coupled with a regulating rod cock from $10^{-3} N_F$)</p> <p>Regulating rod cock and withdrawal of OSCCs from $10^{-3} N_F$</p> <p>Regulating rod cock (ramp and ramp-step changes)</p> <p>Movement of two reflector pieces toward the core</p> <p>Perched fuel drop of two fuel elements from within measurement accuracy (1-in.)</p> <p>Firewater injection during depressurized operation</p> <p>Gas release into the core</p>
17. Reactivity insertion (slow ramp)(RSH)	<p>Withdrawal of all OSCCs and one NS from N_L</p> <p>Withdrawal of all OSCCs from $10^{-10} N_F$</p> <p>Withdrawal of all OSCCs and NSs from $10^{-5} N_F$</p> <p>Withdrawal of all OSCCs and NSs from $10^{-10} N_F$ (startup accident)</p>

Table 2. (continued)

<u>Initiating Event Category</u>	<u>Initiating Event(s)</u>
17. Reactivity insertion (slow ramp)(RSH) (continued)	<p>Recriticality after xenon decay (failure to insert shims after safety rod insertion following a restart attempt)</p> <p>Cold water addition by opening a primary pump discharge valve</p> <p>Reflector movement towards the core</p> <p>Perched fuel drop from within measurement accuracy (1-in.)</p> <p>Cold water addition due to inadvertent primary pump startup</p> <p>Cold water addition due to high secondary system flow with cold secondary water</p> <p>Failure of the variable frequency drive for the secondary cooling system with a failure to enable the temperature control valve</p>
18. Reactivity insertion (loop ramp)(RLH)	<p>Experiment loop blowdown due to a 0.5-in. line break</p> <p>Experiment loop flow coastdown</p> <p>Experiment loop blowdown due to a 1-in. diameter break</p> <p>Loss of loop temperature control due to a heat exchanger failure or line heaters sticking on</p> <p>Experiment loop voiding due to rupture disk or relief valve failure, or due to an open and accessible valve</p> <p>Experiment loop voiding due to opening of a normally inaccessible valve</p> <p>Experiment movement or shroud movement (drive experiment)</p> <p>Simultaneous flow coastdown and/or loss of temperature control in all ATR loops</p>

Table 2. (continued)

<u>Initiating Event Category</u>	<u>Initiating Event(s)</u>
18. Reactivity insertion (loop ramp) (RLH) (continued)	<p>Voiding in all IPTs due to a simultaneous flow coastdown in all loops</p> <p>Driven test loop blowdown with a holder failure</p> <p>Cold loop draining due to open and accessible valve (depressurized operation)</p> <p>Loop heaters on inadvertently (depressurized operation)</p> <p>Cold loop draining due to an open normally inaccessible valve (depressurized operation)</p> <p>Simultaneous movement of two powered axial locator mechanism (PALM) tests</p> <p>High-pressure induced leak</p> <p>Experiment loop heaters accidentally on</p> <p>High-pressure demineralized water (HDW) coolant high-temperature events</p> <p>Partial loss of HDW flow to the IPT cubicles</p> <p>Total loss of HDW flow to the IPT cubicles</p> <p>Partial loss of utility cooling water (UCW)</p> <p>Total loss of UCW flow</p> <p>PALM movement damage to the fuel</p> <p>Fuel experiment support structural damage and test movement</p> <p>Experiment damage and movement due to a high differential pressure</p>
19. Lobe power imbalance (RLP)	<p>Mispositioning of NSs</p> <p>Mispositioning of OSCCs</p> <p>Mispositioned safety rods</p>

Table 2. (continued)

Initiating Event Category	Initiating Event(s)
19. Lobe power imbalance (RLP)(continued)	Lobe power balance shift due to shim or experiment loop or operator failure with operator compensation Slow power increase or lobe power balance shift due to shim or experiment loop or operator failure
20. Large reactivity insertion (RIL)	Design basis accident: flux trap voiding due to in-pile tube and gas jacket rupture Experiment loop blowdowns due to a full out-of-pile pipe break coupled with holder failure Loop blowdown due to a large out-of-pile pipe leak Loop blowdown in a standard IPT with a driven experiment Perched fuel drop from a height greater than the measurement accuracy
21. High temperature loop overheating (AHT)	Loss of loop temperature control in the advanced high-temperature loop (AHTL) due to heat exchanger failure or line heaters stuck on without a loop scram (potential loop pressure boundary failure)
22. Very large reactivity insertion (RPF)	Perched fuel drop from a height significantly greater than the measurement accuracy
23. Local faults (LFS)	Fuel channel blockage from small beryllium pieces or material dropped into the vessel during shutdown Fuel channel blockage due to sight glass or other inventoried items left in the vessel Fuel channel blockage due to structural failures of small components above the core or inlet flow screen failure When depressurized, gas leakage into the fuel element channels (heat transfer degradation) When pressurized, gas leakage into the fuel element channels (heat transfer degradation)

Table 2. (continued)

Initiating Event Category	Initiating Event(s)
23. Local faults (LFS) (continued)	<p>Excessive oxide film buildup (coolant chemistry control failure)</p> <p>Fuel element failures affecting a few elements in the core (poor fabrication control)</p> <p>Fuel element failure due to handling damage</p> <p>Unexpected high power peaking (loading or analysis error)</p> <p>Bypass demineralizer failure</p>
24. Storage canal event (CAN)	<p>Handling and shipping damage</p> <p>Fueled experiment dropped on the floor</p> <p>Significant fuel meltdown of one entire fuel element (or partial melting of more than one element) due to crushing or other accidents</p> <p>Handling errors</p> <p>Storage errors or failures</p> <p>Cask drop</p> <p>Storage grid tips over</p> <p>Canal drain valve opened and cover plate off of drain</p> <p>Canal drain line rupture and cover plate off of drain</p> <p>Canal rupture</p> <p>Fueled experiment dropped on the floor</p> <p>Fuel transfer cask dropped more than one foot above the canal floor</p> <p>Fuel transfer cask dropped in restricted canal areas</p> <p>Fuel channel blockage</p>

Table 3. ATR scram and shutdown history

Initiating Event Category	ATR Scrams/Shutdowns					Data/Period Chosen		Bayesian Updated Frequency Distribution Mean/Yr Error Factor	
	1969-73	1974-78	1979-83	1984-88	Total	Events	Years Period		
1. RSD	253	219	151	109	732	109	5	21.9	<2.0
2. LPF	0	0	0	1	1	1	20	0.075	3.3
3. LPP	1	1	0	0	2	0	10	0.050	8.4
4. HCP	0	2	0	0	2	0	10	0.050	8.4
5. LHS	0	0	0	0	0	0	20	0.025	8.4
6. LCP	5	11	16	4	36	4	5	0.90	2.0
7. LAC	0	0	0	0	0	0	20	0.025 ^c	8.4
8. LDC	1	0	0	0	1	1	20	0.075 ^c	3.3
9. LDP	- ^a	- ^a	- ^a	0 ^b	0 ^b	0	2	0.25 ^c	8.4
10. LIA	0	0	0	0	0	0	20	0.025	8.4
11. S12	0	0	0	0	0	0	20	0.025	8.4
12. LLC	0	0	0	0	0	0	20	0.025 ^d	8.4
13. DLC	0	0	0	0	0	0	20	0.025 ^d	8.4
14. ISL	0	0	0	0	0	0	20	0.025 ^e	8.4
15. ISF	0	0	0	0	0	0	20	0.025 ^e	8.4
16. RFH	4	2	2	0	8	0	5	0.10	8.4
17. RSH	0	1	1	0	2	0	5	0.10	8.4
18. RLH	10	16	25	5	56	5	5	1.1	<2.0
19. RLP	0	1	2	0	3	0	5	0.10	8.4
20. RIL	0	0	0	0	0	0	20	0.025 ^d	8.4
21. AHT	2	4	3	0	9	0	5	0.10 ^f	8.4
22. RPF	0	0	0	0	0	0	20	0.025 ^d	8.4
23. LFS	0	0	0	0	0	0	20	0.025	8.4
24. CAN	0	0	0	0	0	0	20	0.025 ^d	8.4

a. Automatic start diesel (M6) added in 1986. LDP involves loss of normally-operating diesel and failure of M6 to start or run.

b. 1987 and 1988 only.

Table 3. (continued)

- c. This frequency was not used. Instead, the event was quantified using a detailed system fault tree.
- d. This frequency was not used. A separate analysis was performed to estimate the frequency (Appendix S).
- e. This frequency was not used. The ATR-specific configuration was identified and quantified.
- f. This frequency applies to all (9) experiment loops. For the AHTL, the result was divided by 9 to obtain 0.011/yr.

Notes:

For categories with 0 or 1 event, the entire 20-yr period was chosen. For categories with more than one event, either the last 5 or 10 yr were chosen (5 yr if a significant decreasing or increasing trend was observed).

For the Bayesian update with a noninformative prior, the mean is $(2n+1)/(2T)$, where n is the number of events and T is the time period (yr).

The error factor is defined as 95th percentile/50th. A lognormal distribution was assumed.

Table 4. ATR PRA initiating event category frequencies

Initiating Event Category	Identifier	Mean Frequency/Yr	Error Factor ^a	Source
1. Normal/spurious reactor shutdown	RSD	21.9	2.0 ^b	ATR history
2. Low PCS flow	LPF	0.075	3.3	ATR history
3. Low PCS pressure	LPP	0.050	8.4	ATR history
4. High PCS pressure	HCP	0.050	8.4	ATR history
5. Loss of heat sink	LHS	0.025	8.4	ATR history
6. Loss of commercial power	LCP	0.90	2.0	ATR history
7. Loss of all ac power	LAC	1.1E-3	10.0 ^b	ATR fault tree quantification
8. Loss of diesel-commercial bus	LDC	0.075	3.3	ATR fault tree quantification
9. Loss of diesel power	LDP	0.37	3.0 ^b	ATR fault tree quantification
10. Loss of instrument air	LIA	0.025	8.4	ATR history
11. Small LOCA	S12	0.025	8.4	ATR history
12. Large LOCA	LLC	4.5E-6	30.0 ^b	ATR-specific evaluation
13. Direct-damage large LOCA	DLC	5.0E-7	30.0 ^b	ATR-specific evaluation
14. Interfacing system LOCA (LDW)	ISL	3.3E-8	10.0 ^b	ATR piping configuration
15. Interfacing system LOCA (FIS)	ISF	1.5E-8	10.0 ^b	ATR piping configuration
16. Reactivity insertion (fast ramp)	RFH	0.10	8.4	ATR history
17. Reactivity insertion (slow ramp)	RSH	0.10	8.4	ATR history
18. Reactivity insertion (loop ramp)	RLH	1.1	2.0 ^b	ATR history
19. Lobe power imbalance	RLP	0.10	8.4	ATR history
20. Large reactivity insertion	RIL	5.0E-8	30.0 ^b	ATR-specific evaluation
21. High temperature loop overheating	AHT	1.1E-2	8.4	ATR history
22. Very large reactivity insertion	RPF	3.0E-7	10.0 ^b	ATR configuration and procedures

Table 4. (continued)

Initiating Event Category	Identifier	Mean Frequency/yr	Error Factor ^a	Source
23. Local faults	LFS	0.025	8.4	ATR history
24. Storage canal draining ^c	CAN	4.2E-7	10.0 ^b	ATR piping configuration
		Total = 24.9/yr		

a. A lognormal distribution was assumed. The error factor is the 95th percentile/50th.
b. Estimated.
c. This event concerns irradiated fuel in the storage canal. All of the other IE categories involve fuel in the core (and while ATR is at power).

For the direct-damage large LOCA, the individual methodology mean frequency estimates are listed below:

- Statistical estimation
 - Integral 3.3E-5/yr
 - Basic elemental 7.4E-6/yr
 - Thomas elemental 1.0E-5/yr
- Phenomenological analysis
 - PC-PRAISE 3.0E-11/yr
- Expert judgment elicitation 7.4E-6/yr.

These five estimates were combined using a weighted geometric averaging technique, with several weighting schemes postulated. Based on the results of the geometric averaging, the direct-damage large LOCA (DLC) mean frequency is estimated to be 5.0E-7/yr, with an upper bound of 1.8E-6/yr (95th percentile), a lower bound of 2.0E-9/yr (5th percentile), and an error factor of 30.

For the direct-damage experiment loop rupture (RIL), the individual mean frequency estimates for the pressure tube are listed below:

- Statistical estimation
 - Basic elemental 6.7E-6/yr
 - Thomas elemental 2.7E-6/yr
- Phenomenological analysis
 - PC-PRAISE 2.9E-8/yr.

Again, the results were combined using the weighted geometric averaging technique, with several weighting schemes. Also, a factor of 0.1 was applied to the result to account for the gas envelope tube also having to rupture. Based on the results, the direct-damage experiment loop rupture (RIL) mean frequency is estimated to be 5.0E-8/yr, with an upper bound of 1.8E-7/yr (95th

percentile), a lower bound of $2.0\text{E-}10/\text{yr}$ (5th percentile), and an error factor of 30.

Results from the comprehensive analysis in Appendix S from Reference 1 can also be used to quantify the nondirect-damage large LOCA initiator, LLC. Most of the frequency estimation techniques used for DLC indicated that the large LOCA frequency for the entire PCS should be approximately ten times higher than that for DLC. Therefore, LLC, which represents the portion of the PCS not resulting in direct damage, has a frequency of $4.5\text{E-}6/\text{yr}$. An error factor of 30 was assumed.

Finally, the very large reactivity insertion event (RPF) involves dropping a fuel element into the ATR core. The fuel element must be perched or unseated a significant height above its seated position after the ATR core has become critical or is closely approaching criticality. The frequency for RPF was estimated based on a detailed analysis of ATR procedures and history.

The total yearly frequency for ATR IEs is $24.9/\text{yr}$. Approximately 12 of these are normal shutdowns as experiments are completed.

1.3.3 Event Tree Development

Event trees (ETs) were developed for all of the IE categories discussed previously. The ETs were used to graphically depict significant accident scenarios, starting with initiators and branching out into various combinations of safety system successes and failures. The various paths through the ETs are termed accident sequences. Outcomes involve either no significant fuel damage (a successfully mitigated sequence) or fuel damage. Additionally, the fuel damage cases were divided into those involving only limited initial fuel damage (resulting from the direct-damage IEs) and those involving core-wide damage (as a result of long-term core uncover). Success criteria for the safety systems modeled in the ETs were based primarily on conservative safety analyses. (In most cases, it is believed that these conservatisms did not significantly affect the fuel damage frequency results.) Future best-estimate severe accident analyses conducted as part of the ongoing

Level 2 analysis may indicate areas where more realistic success criteria may be applied.

There are 24 ETs developed for the ATR PRA. These event trees are not highly detailed in terms of the number of top events. Much of the detail is included in the system fault trees. This methodology is often termed the "small event tree and large fault tree" approach and results in a relatively small number of accident sequences but large and detailed fault trees. The ATR ETs include 230 sequences leading to fuel damage. All of the fuel damage sequences were assigned to plant damage states (PDSs). These PDSs characterize the sequences in terms of phenomena important to follow-on containment analysis (Level 2 PRA). (The Level 2 work will be incorporated into this report when it is completed.) A description of the PDSs is presented in Table 5.

Table 5. Plant damage state definitions

Plant Damage State	Description
P1	Transients with failure of PCS recirculation, successful PCS depressurization, but failure of low-pressure coolant injection systems. Low-pressure boil-off with eventual core uncover and core-wide fuel damage.
P2	Transients with failure of PCS recirculation and failure to depressurize the PCS. High-pressure boil-off with eventual core uncover and core-wide fuel damage.
P3	LOCAs with failure of low-pressure coolant injection systems. Core uncover and core-wide fuel damage.
P3M	Direct-damage large LOCA, resulting in initial fuel damage (10 - 30% of the core), even with successful scram and coolant injection.
P4	Sequences involving failure to shut down the reactor, termed anticipated transient without scram (ATWS) sequences. Potential fuel damage even with the core covered.
P5	Very large reactivity insertion event potentially resulting in initial core-wide fuel damage and reactor vessel failure.
P5M	Direct-damage large reactivity insertion events, resulting in initial fuel damage even with successful scram.
P6	Interfacing system LOCAs with failure of low-pressure coolant injection systems.
P7	Storage canal draining with failure to provide makeup. Uncovery of irradiated fuel stored in canal and eventual fuel damage.

1.3.4 Fault Tree Development

Many of the systems modeled in the event tree top events were developed into detailed fault trees. The fault trees were used to develop and depict the logical interrelationships of the basic events (e.g., pump, valve, and electrical faults), which lead to system failure. Fault trees covering 12 systems were developed.

1.3.5 Dependent Failure Analysis

The PRA Procedures Guide³ lists nine different types of dependent failures that should be considered in a comprehensive PRA. The ATR PRA has accounted for all of the different types including those resulting from spatial dependencies covered by the external event analysis. In addition to explicit modeling of known dependent failure mechanisms, parametric modeling of dependent failures of similar components was also performed. This involved identifying similar components (valves, pumps, and diesel generators) within systems and adding parametric dependent failure events to the fault trees. Quantification of these parametric dependent failure events was performed using the multiple Greek letter methodology.⁴ Beta factors (multiplication factors applied to single failure events to account for dependent failure of two similar components) used in the ATR PRA were obtained from Reference 4, except for diesel generators, where plant-specific events indicated a value of 0.20 was appropriate. For the gamma factor (the multiplication factor applied to the beta factor to account for a third component dependent failure), a value of 0.5 was used. A list of the parametric similar component dependent failure events is shown in Table 6.

1.3.6 Human Reliability Analysis

The ATR PRA human reliability analysis (HRA) task was divided into three subtasks: screening analysis of human errors before an accident (miscalibration and failure to restore components following test or maintenance), screening analysis of operator errors during an accident, and refined analysis of dominant human errors. Human errors causing an IE were

Table 6. Parametric dependent failure events

Dependent Event	Description	Mean	Beta	Gamma	Reference Event
DGP-DGN-CR-DIESL	3 diesel generators fail to run	2.1E-04	0.20	0.5	DGP-DGN-FR-00M42
DGP-DGN-CS-DIESL	2 diesel generators fail to start	6.8E-03	0.20	NA	DGP-DGN-FS-000M6
DGP-MDP-CR-62732	2 diesel transfer pumps fail to run	7.2E-05	0.10	NA	DGP-MDP-FR-06273
DGP-MDP-CS-62732	2 diesel transfer pumps fail to start	3.0E-04	0.10	NA	DGP-MDP-FS-06273
ESF-LOG-CF-OESF1	2:3 division logic units	3.0E-05	0.10	NA	ESF-LOG-F0-0ESF1
ESF-MDP-CF-OPPIC	Pressurizing pump trip circuits	5.0E-05	0.10	NA	DGP-BKR-CC-0E9D3
FIS-AOV-CA-004F4	4 FIS injection valves	4.3E-05	0.10	0.5	FIS-AOV-CC-V1615
FIS-AOV-CA-P1516	2 FIS upper injection valves	8.6E-05	0.10	NA	FIS-AOV-CC-V1615
FIS-AOV-CA-P7A7B	2 FIS lower injection valves	7.0E-05	0.10	NA	FIS-AOV-CC-LCV7A
FIS-CKV-CA-004F4	4 FIS lower injection check valves	1.5E-06	0.06	0.5	FIS-CKV-CC-CKB77
FIS-DDP-CR-FWPMP	2 FIS pumps fail to run (619-10 & 633-1)	1.2E-02	0.10	NA	FIS-DDP-FR-06331
FIS-DDP-CS-FWPMP	2 FIS pumps fail to start (619-10 & 633-1)	1.0E-03	0.10	NA	FIS-DDP-FS-61910
FIS-LOG-CF-FWACT	4 logic units	3.0E-05	0.10	NA	FIS-LOG-F0-B23MA
FIS-MDP-CR-RWPMP	3 raw water pumps fail to run	3.6E-05	0.10	0.5	FIS-MDP-FR-P6191
FIS-MDP-CS-RAWAT	3 raw water pumps fail to start	3.0E-04	0.10	NA	FIS-MDP-FS-P6191
IAS-MDC-CR-0M678	3 compressors fail to run (M-6, 7, & 8)	1.2E-04	0.10	0.5	IAS-MDC-FR-000M6
IAS-MDC-CS-00M78	2 compressors fail to start (M-7 & M-8)	5.0E-04	0.10	NA	IAS-MDC-FS-000M7
IAS-MDC-CS-0M678	3 compressors fail to start (M-6, 7, & 8)	2.5E-04	0.10	0.5	IAS-MDC-FS-000M6
LDW-CKV-CA-DEMIN	3 flush check valves (DW-11, 13, & 14)	1.5E-06	0.06	0.5	LDW-CKV-CC-ODW11
LDW-MDP-CR-PUMPS	4 LDW pumps fail to run (18, 19, 20, & 21)	3.6E-05	0.10	0.5	LDW-MDP-FR-60820
LDW-MDP-CS-PUMPS	3 LDW pumps fail to start (18, 19, & 20)	1.5E-04	0.10	0.5	LDW-MDP-FS-60818
MKP-CKV-CA-CK269	2 discharge valves fail to open (26 & 29)	3.0E-06	0.06	NA	MKP-CKV-CC-CKB26
MKP-MDP-CR-M1415	2 pumps fail to run (M-14 & M-15)	7.2E-05	0.10	NA	MKP-MDP-FR-00M14
MKP-MDP-CS-M1415	2 pumps fail to start (M-14 & M-15)	5.7E-04	0.10	NA	MKP-MDP-FS-00M14
MKP-SRV-CA-712KC	2 safety relief valves (SRVs) fail to close (SF-A-71 & 72)	2.1E-04	0.07	NA	MKP-SRV-00-00S71
MKP-SRV-CA-712KO	2 SRVs fail to open (SF-A-71 & 72)	2.1E-04	0.07	NA	MKP-SRV-CC-00S71
MKP-SRV-CF-SF756	2 SRVs fail to remain closed (75 & 76)	2.1E-07	0.07	NA	MKP-SRV-CO-SFA75
PCS-CKV-CA-A1720	2 check valves fail to open (A-17 & 20)	3.0E-06	0.06	NA	PCS-CKV-CC-00A17
PCS-FAN-CR-53A3B	3 PCS pump cooling fans fail to run	4.7E-05	0.13	0.5	PCS-FAN-FR-HVS50
PCS-FAN-CS-53A3B	3 PCS pump cooling fans fail to start	3.3E-04	0.13	0.5	PCS-FAN-FS-HVS50
PCS-MDP-CR-01011	2 PCS pump trains fail to run (M-10 & M-11)	7.2E-05	0.10	NA	PCS-MDP-FR-00M10

Table 6. (continued)

Dependent Event	Description	Mean	Beta	Gamma	Reference Event
PCS-MDP-CR-M6789	3 PCS pumps fail to run (M-6, 7 & 8)	3.6E-05	0.10	0.5	PCS-MDP-FR-00M06
PCS-MDP-CS-01011	2 PCS pump trains fail to start (M-10 & M-11)	2.5E-04	0.10	NA	PCS-MDP-FS-00M10
PCS-MDP-CS-M6789	3 PCS pumps fail to start (M-6, 7 & 8)	1.5E-04	0.10	0.5	PCS-MDP-FS-00M06
PCS-MOV-CA-MOVCC	3 PCS motor-operated valves fail to open	1.2E-04	0.08	0.5	PCS-MOV-CC-000A3
PCS-MOV-CA-MOV00	2 PCS motor-operated valves fail to close	2.4E-04	0.08	NA	PCS-MOV-00-000A3
PPS-ACT-CF-LFS00	Loop low flow scram channels	3.0E-05	0.10	NA	PPS-LOG-F0-CLLCA
PPS-ACT-CF-LPS00	Loop low pressure scram channels	3.0E-05	0.10	NA	PPS-LOG-F0-CLLCA
PPS-ACT-CF-SCR00	Manual scram signal trains	3.0E-05	0.10	NA	PPS-LOG-F0-516AS
PPS-CRM-CF-3R0DS	3:5 or 4:6 rods fail to insert	6.5E-07	0.10	0.5	PPS-CRM-F0-SRSRS
PPS-CRM-CF-4R0DS	2:5 or 3:6 rods fail to insert	1.3E-06	0.10	NA	PPS-CRM-F0-SRSRS
PPS-DPT-CF-LDP00	Low diff. pressure sensor trains	7.2E-06	0.10	NA	PCS-PST-FC-0500A
PPS-DTT-CF-DTS00	Quad 1 diff temperature sensor trains	3.0E-07	0.10	NA	PPS-TST-FC-T505A
PPS-LOG-CF-OH1PL	High inlet pressure 2:3 logics	3.0E-05	0.10	NA	PPS-LOG-F0-VH1P1
PPS-LOG-CF-OH1TL	High inlet temperature 2:3 logics	3.0E-05	0.10	NA	PPS-LOG-F0-VH1T1
PPS-LOG-CF-OH1LL	High neutron level 2:3 logics	3.0E-05	0.10	NA	PPS-LOG-F0-0HNL1
PPS-LOG-CF-OH0PL	High outlet pressure 2:3 logics	3.0E-05	0.10	NA	PPS-LOG-F0-VH0P1
PPS-LOG-CF-OH0TL	High outlet temperature 2:3 logics	3.0E-05	0.10	NA	PPS-LOG-F0-VH0T1
PPS-LOG-CF-0LDPL	Low diff pressure 2:3 logics	3.0E-05	0.10	NA	PPS-LOG-F0-VLDP1
PPS-LOG-CF-0LIPL	Low inlet pressure 2:3 logics	3.0E-05	0.10	NA	PPS-LOG-F0-VL1P1
PPS-LOG-CF-0LOPL	Low outlet pressure 2:3 logics	3.0E-05	0.10	NA	PPS-LOG-F0-VLOP1
PPS-LOG-CF-0WRNL	Wide range neutron 2:3 logics	3.0E-05	0.10	NA	PPS-LOG-F0-HWNL1
PPS-LOG-CF-L20F3	Loop 2:3 logic modules	3.0E-05	0.10	NA	PPS-LOG-F0-LSCR1
PPS-LOG-CF-MSCR1	Manual scram 2:3 logics	3.0E-05	0.10	NA	PPS-LOG-F0-MSCR1
PPS-LOG-CF-Q1DTL	Quad 1 diff temperature 2:3 logics	3.0E-05	0.10	NA	PPS-LOG-F0-Q1DT1
PPS-LOG-CF-RCCCS	Rod clutch coil controllers	2.6E-06	0.10	0.5	PPS-LOG-F0-RCCA1
PPS-NID-CF-HNL00	High neutron level sensor trains	5.0E-08	0.10	NA	PPS-NID-FC-0510A
PPS-NID-CF-WRN00	Wide range neutron level sensor trains	5.0E-08	0.10	NA	PPS-NID-FC-0512A
PPS-PST-CF-HIP00	High inlet pressure sensor trains	7.2E-06	0.10	NA	PCS-PST-FC-0502A
PPS-PST-CF-HOP00	High outlet pressure sensor trains	7.2E-06	0.10	NA	PCS-PST-FC-0501A
PPS-PST-CF-LIP00	Low inlet pressure sensor trains	7.2E-06	0.10	NA	PCS-PST-FC-0502A

Table 6. (continued)

<u>Dependent Event</u>	<u>Description</u>	<u>Mean</u>	<u>Beta</u>	<u>Gamma</u>	<u>Reference Event</u>
PPS-PST-CF-LOP00	Low outlet pressure sensor trains	7.2E-06	0.10	NA	PCS-PST-FC-0501A
PPS-SYS-CF-SCR00	Loss of scram system	1.5E-05	0.10	NA	See Table 3.7-3
PPS-TST-CF-HIT00	High inlet temperature sensor trains	3.0E-07	0.10	NA	PPS-TST-FC-T509A
PPS-TST-CF-HOT00	High outlet temperature sensor trains	3.0E-07	0.10	NA	PPS-TST-FC-T517A
RRS-NSR-CF-00000	Many neck shim failures	1.0E-20	0.10	NA	RRS-NSR-FC-00000
RRS-OSC-CF-00000	Many outer shim control cylinder failures	1.5E-06	0.10	0.5	RRS-OSC-FC-00512
RRS-SGR-CF-G1234	Rod group insert bus failures	1.0E-09	0.10	NA	RRS-SGR-ST-G1BUS
VVS-AOV-CA-02312	2 vessel vent valves (GB-23-1 & -2)	6.3E-05	0.10	NA	VVS-AOV-CC-GB231

considered to be covered in the IE category frequencies obtained from ATR experience. The screening human error probabilities were obtained using the Accident Sequence Evaluation Program Nominal HRA technique.^{5,6} This technique included discussions with operators, but did not involve detailed walk-throughs and talk-throughs. Based on an interim accident sequence quantification, 12 dominant human errors were chosen for refinement. The refined HRA included a state-of-the-art combination of the technique for human error rate prediction (THERP)⁷ and human cognitive reliability (HCR).⁸ This combination of models has been recommended as potentially more accurate than either individual method.^{9,10} This combined technique involved detailed walk-throughs and talk-throughs with operators of events in the context of the accident sequences in question. Post-accident human errors modeled in the system fault trees are listed in Table 7. Post-accident human errors not developed into fault trees are listed in Table 8. Finally, the human error events analyzed using the refined technique are listed in Table 9. No recovery actions were applied to accident sequence cutsets because very few of the cutsets were recoverable within the time available.

1.3.7 Data Development

Data development involves the quantification of the basic events in the fault trees and any event tree top events not developed into fault trees. A comprehensive generic data base was developed, based mainly on the Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR).¹⁰ This data base was then supplemented with ATR-specific data on test and maintenance (TM) outages, selected pumps and valves, diesel generators, and scram system components. The TM outage events modeled in the fault trees are listed in Table 10, along with the limited ATR data obtained. ATR-specific component failure data are listed in Table 11. The failure rates were obtained using a Bayesian update with the generic value as the prior and the ATR data as the evidence.

Table 7. Operator failure to act events in fault trees (screening)

Identifier	Mean		Description
	Unavailability		
BBP-BCH-XF-0E105	8.0E-2		Operator fails to align battery charger 670-E-105 to diesel bus 670-E-9
CDP-SYS-XF-STPWR ^a	8.0E-2		Operator failure to realign after recovery of site power
DGP-BKR-XF-00017	2.5E-1		Operator failure to allow commercial power backup from bus 670-E-2 to diesel bus 670-E-3 by closing breaker 17
DGP-DGN-XF-00M43	8.0E-2		Operator fails to start and align diesel generator 670-M-43
DGP-MDP-XF-06272 ^a	8.0E-2		Operator fails to manually start diesel oil transfer pump 627-2 or 627-3
DGP-SYS-XF-01OAP	8.0E-2		Operator fails to detect and correct loss of air pressure in any starting air tank
FIS-MDP-XF-RAWAT	5.0E-2		Operator fails to actuate raw water pumps 619-1, -6, and -7
FIS-XSW-XF-FIACF	2.7E-3		Operator fails to attempt manual firewater injection
IAS-MDC-XF-0M678 ^a	2.9E-3		Operator fails to start any compressor
LDW-SYS-XF-PURGE	8.0E-2		Operator fails to initiate LDW purge
MKP-SYS-XF-CEMXF	8.0E-2		Operator failure to restore makeup flow from the pressurizing system when primary coolant circulation has failed (closure of control valve PCV-1-1 is required).
MKP-SYS-XF-HPTRX	1.3E-1		Operator failure to reset pressurizing pump trip and restore makeup from the pressurizing system
MKP-SYS-XF-MCRXF	8.0E-2		Operator failure to restore makeup flow from the pressurizing system when primary coolant circulation has not failed

Table 7. (continued)

Identifier	Mean		Description
	Unavailability		
PCS-MDP-XF-00M10	2.7E-3		Operator fails to restart emergency PCS pump M-10 after loss of all ac power
PCS-MDP-XF-00M11	2.7E-3		Operator fails to start emergency PCS pump M-11
RRS-XSW-XF-G1SS2	2.7E-5		Rod group 1 control switch SS-2 not lined up for automatic transfer
PPS-XSW-XF-OPSCR	2.5E-3		Operator fails to actuate a manual scram
RRS-XSW-XF-G1XSW	2.7E-5		Rod group 1 exercise switch operator error
RRS-XSW-XF-G2SS2	2.7E-5		Rod group 2 control switch SS-2 not lined up for automatic transfer
RRS-XSW-XF-G2XSW	2.7E-5		Rod group 2 exercise switch operator error
RRS-XSW-XF-G3SS2	2.7E-5		Rod group 3 control switch SS-2 not lined up for automatic transfer
RRS-XSW-XF-G3XSW	2.7E-5		Rod group 3 exercise switch operator error
RRS-XSW-XF-G4SS2	2.7E-5		Rod group 4 control switch SS-2 not lined up for automatic transfer
RRS-XSW-XF-G4XSW	2.7E-5		Rod group 4 exercise switch operator error
VVS-SYS-XF-ACTUA ^b	8.0E-2		Operator fails to initiate vessel vent procedure

a. These operator actions were developed for fault tree modeling that took place during 1991; thus, they are not included in Appendix B of Reference 1.

b. This event has been replaced by three events modeled in the depressurization fault tree: VDF-SYS-XF-VPID, VDF-SYS-XF-0VDP2, and VDF-SYS-XF-VDPIIM.

Table 8. Operator failure to act events in event trees (screening)

Identifier	Mean		Description
	Unavailability		
CFA	8.0E-2		Failure to provide firewater to the storage canal
CLD	3.2E-2		Failure to detect inadvertent draining of the storage canal
DCP	8.0E-2		Failure to restore the diesel-commercial bus, given recovery of offsite or diesel power
HTD ^a	8.0E-2		Loop operator failure to notice and correct coolant overheating in the high-temperature loop. The basic screening value (median = 0.05, Table B-6 in Appendix B ¹) without recovery was used for this event. Assuming an error factor of 5, the mean value is 0.08.
RCD ^a	2.5E-1		Reactor operator failure to detect and stop regulating rod control system from compensating for reactivity insertions from coolant overheating in high-temperature loop. Estimates for this failure probability range from 0.1 to 1.0. A mean value of 0.25 was chosen, with an error factor of 5.
PVG ^a	1.0E-1		Failure to detect a grossly perched fuel element during visual inspection prior to closing up the reactor vessel for startup. A grossly perched fuel element must be at least 15 in. above its fully seated position. Such a gross perch is readily noticeable. A mean value of 0.1 was chosen, with an error factor of 10.

a. These operator actions were developed for event tree modeling that took place during 1991; thus, they are not included in Appendix B of Reference 1.

Table 9. Operator failure to act events (refined)

Screening Identifier	Refined Identifier	Mean Failure Screening	Probability Refined	Error Factor	Description
BBP-BCH-XF-0E105	BBP-BCH-XF-0E105	8.0E-2	5.87E-2	10	Operator fails to align battery charger 670-E-105 to diesel bus 670-E-9
FIS-XSW-XF-FIAC	FIS-XSW-XF-FIAC	2.7E-3	1.87E-3	10	Operator fails to attempt manual firewater injection
LDW-SYS-XF-PURGE	LDW-SYS-XF-PURGE	8.0E-2	4.11E-2	10	Operator fails to initiate LDW purge
MKP-SYS-XF-CEMXF	MKP-SYS-XF-CEMXF	8.0E-2	3.73E-2	10	Operator failure to restore makeup flow from the pressurizing system when primary coolant circulation has failed (closure of control valve PCV-1-1 is required)
MKP-SYS-XF-HPTRX	MKP-SYS-XF-HPTRX	1.3E-1	8.30E-3	10	Operator failure to reset pressurizing pump trip and restore makeup from the pressurizing system
-	PCS-MOV-XF-000A6 ^b	-	8.03E-3	10	Operator Fails to Close Motor Operated Valve A-6 Upon Shutdown of PCS Pump M-7
-	PCS-MOV-XF-000A9 ^b	-	8.03E-3	10	Operator Fails to Close Motor Operated Valve A-9 Upon Shutdown of PCS Pump M-8
-	PCS-MOV-XF-000A12 ^b	-	8.03E-3	10	Operator Fails to Close Motor-Operated Valve A-12 Upon Shutdown of PCS Pump M-9
PFR	PCS-MOP-XF-PFR01	1.3E-1	8.50E-2	10	Operator Fails to Restart one of the PCS Pumps Given Both Emergency Pumps Fail
VVS-SYS-XF-ACTUA ^a	VDF-SYS-XF-0VDP2	8.0E-2	1.50E-4	10	Operator Failure to Actuate VVS (VDP2)
VVS-SYS-XF-ACTUA ^a	VDF-SYS-XF-VDP1D	8.0E-2	1.00E-4	10	Operator Failure to Diagnose the Need For Action (VDP1D)
VVS-SYS-XF-ACTUA ^a	VDF-SYS-XF-VDP1M	8.0E-2	3.72E-2	10	Operator Failure to Open Manual Valve GT-39 (VDP1 action)

a. These operator actions were developed for fault tree modeling that took place during 1991; thus, they are not included in the screening analysis.

b. This event has been replaced by three events modeled in the depressurization fault tree: VDF-SYS-XF-VDP1D, VDF-SYS-XF-0VDP2, and VDF-SYS-XF-VDP1M.

Table 10. ATR test and maintenance data

System	Event Identifier	Component		Description
		TM Outage During 1987-88 (h) ^a	TM Unavailability ^b	
DGP	DGP-DGN-TM-00M42	802	3.6E-2 ^c	Diesel generator 670-M-42 unavailable due to test or maintenance
DGP	DGP-DGN-TM-00M43	163	3.6E-2 ^c	Diesel generator 670-M-43 unavailable due to test or maintenance
DGP	DGP-DGN-TM-00M06	146	1.1E-2 ^c	Diesel generator 670-M-06 unavailable due to test or maintenance
DGP	DGP-MDC-TM-00M66	?	1.0E-2 ^d	High-pressure air compressor 670-M-66 unavailable due to test or maintenance
FIS	FIS-DDP-TM-06331	?	1.0E-2 ^d	Diesel firewater pump 633-1 unavailable due to test or maintenance
FIS	FIS-DDP-TM-61910	?	1.0E-2 ^d	Diesel firewater pump 619-10 unavailable due to test or maintenance
FIS	FIS-MDP-TM-61912	?	1.0E-2 ^d	Firewater pump 619-12 unavailable due to test or maintenance
FIS	FIS-MDP-TM-P6191	?	1.0E-2 ^d	Raw water pump 619-1 unavailable due to test or maintenance
FIS	FIS-MDP-TM-P6196	?	1.0E-2 ^d	Raw water pump 619-6 unavailable due to test or maintenance
FIS	FIS-MDP-TM-P6197	?	1.0E-2 ^d	Raw water pump 619-7 unavailable due to test or maintenance
LDW	LDW-MDP-TM-60818	?	1.0E-2 ^d	Flush pump 608-18 unavailable due to test or maintenance
LDW	LDW-MDP-TM-60819	?	1.0E-2 ^d	Flush pump 608-19 unavailable due to test or maintenance
LDW	LDW-MDP-TM-60821	?	1.0E-2 ^d	Deminerallized water pump 608-21 unavailable due to test or maintenance
MKP	MKP-MDP-TM-00M15	?	1.0E-2 ^d	Pressurizing pump m-15 unavailable due to test or maintenance

- a. Only outages while ATR was at power were counted.
b. Unavailability is outage hours/operating hours. For 1987-88, the operating hours were 13,360.
c. An average of both diesel generators (M-42 and M-43) was used.
d. A generic estimate was used because no plant-specific data were available.

Table 11. ATR component failure data

System	Event Identifier	Failures/ Demands or Hours ^a	Component Failure Rates			Description
			Generic ^b	Plant-Specific ^c	Bayesian Update ^d	
PCS	PCS-MDP-FS-00M10 -00M11	0/40 6/54	3.0E-3	0.0E-0	2.5E-3 (5)	Emergency PCS pump M-10 fails to start
			3.0E-3	0.0E-0	2.4E-3 (5)	Emergency PCS pump M-11 fails to start
MKP	MKP-MDP-FS-00M14 -00M15	1/40 0/38	3.0E-3	1.3E-2 ^e	5.7E-3 (3.1) ^e	Pressurizing pump M-14 fails to start
			3.0E-3	1.3E-2 ^e	5.7E-3 (3.1) ^e	Pressurizing pump M-15 fails to start
PCS	PCS-CKV-00-00DA5 -00DA8 -00A11	0/2296 ^{f,g}	1.0E-3	0.0E-0	2.1E-4 (5) ^e	PCS pump M-7 discharge check valve A-5 fails to close
					2.1E-4 (5) ^e	PCS pump M-8 discharge check valve A-8 fails to close
					2.1E-4 (5) ^e	PCS pump M-9 discharge check valve A-11 fails to close
MKP	MKP-AOV-00-0FCV8	0/70 ^h	1.0E-3	0.0E-0	7.0E-4 (10)	Pressurizing flow control valve FCV-1-8 fails to close
					7.6E-4 (10) ^j	Pressure control valve PCV-1-1 fails to close
MKP	MKP-AOV-00-0CV11 -CC-0CV11	0/53 ^{i,j}	1.0E-3	0.0E-0	7.6E-4 (10) ^j	Pressure control valve PCV-1-1 fails to open
					7.6E-4 (10)	Pressurizing valve LCV3A fails to open
FIS	FIS-AOV-CC-LCV7A -LCV7B	0/70 ^{k,l}	1.0E-3	0.0E-0	7.0E-4 (10) ^o	FIS injection valve LCV7A fails to open
					7.0E-4 (10) ^o	FIS injection valve LCV7B fails to open
FIS	FIS-AOV-CC-V1615 -V1616	0/26 ^{k,l}	1.0E-3	0.0E-0	8.6E-4 (10) ^o	FIS injection valve V1615 fails to open
					8.6E-4 (10) ^o	FIS injection valve V1616 fails to open
VWS	VWS-AOV-CC-GB231 -GB232	0/96 ^{k,l}	1.0E-3	0.0E-0	6.3E-4 (10) ^o	WS vent valve GB231 fails to open
					6.3E-4 (10) ^o	WS vent valve GB232 fails to open
PPS	PPS-CRM-FO-0SRSE -0SRSM -0SRSE -0SRSM -0SRSM -0SRSM	0/7336 ^{f,g}	3.0E-5	0.0E-0	1.3E-5 (10) ^o	Safety rod SRSE fails to insert
					1.3E-5 (10) ^o	Safety rod SRSE fails to insert
					1.3E-5 (10) ^o	Safety rod SRSE fails to insert
					1.3E-5 (10) ^o	Safety rod SRSE fails to insert
					1.3E-5 (10) ^o	Safety rod SRSE fails to insert
PPS	PPS-LOG-FO-RCCA1 -RCCA2 -RCCA3 -RCCA4 -RCCA5 -RCCA6	0/2600 ^{f,g}	3.0E-4	0.0E-0	5.2E-5 (10) ^o	RCCC module A1 fails to open its circuit
					5.2E-5 (10) ^o	RCCC module A2 fails to open its circuit
					5.2E-5 (10) ^o	RCCC module A3 fails to open its circuit
					5.2E-5 (10) ^o	RCCC module A4 fails to open its circuit
					5.2E-5 (10) ^o	RCCC module A5 fails to open its circuit
					5.2E-5 (10) ^o	RCCC module A6 fails to open its circuit

Table 11. (continued)

System	Event Identifier	Failures/ Demands or Hours ^a	Component Failure Rates			Description
			Generic ^b	Plant-Specific ^c	Bayesian Update ^d	
PPS	scram signal	0/547 ^e	3.0E-4	0.0E-0	1.5E-4 (10)	Entire scram signal, used as base for PPS-SYS-CF-SCR00 (dependent failure event, Section 3.5)
DGP	DGP-DGN-FS-00M42	6/40	1.0E-2	1.1E-1 ^f	6.3E-2 (1.7) ^g	Diesel generator 670-M-42 fails to start
	-00M43	2/35	1.0E-2	1.1E-1 ^f	6.3E-2 (1.7) ^g	Diesel generator 670-M-43 fails to start
	-00M46	8/191 ^h	1.0E-2	4.2E-2	3.4E-2 (1.7)	Diesel generator 674-M-6 fails to start
DGP	DGP-DGN-FR-00M42	0/6790h	5.0E-3/h	7.5E-5/h ⁱ	8.7E-5/h(3.8) ^j	Diesel generator 670-M-42 fails to run
	-00M43	1/6570h	5.0E-3/h	7.5E-5/h ⁱ	8.7E-5/h(3.8) ^j	Diesel generator 670-M-43 fails to run
	-00M46	2/423h ^k	5.0E-3/h	4.7E-3/h	4.7E-3/h(2.7)	Diesel generator 674-M-6 fails to run
DCP	DCP-BAC-ST-0E015	1/20yr	not used	8.2E-6/h	1.2E-5/h(3.3) ^l	480 Vac diesel-commercial bus 670-E-15 short/open circuit
CDP	CDP-BAC-LP-STPUR	4/5yr	not used	9.1E-5/h	1.0E-4/h(2.0) ^m	132 kV ac site loop bus loss of power

a. Obtained from the process control room logs for 1987 and 1988, unless otherwise indicated.

b. Obtained from the generic data base, Table 3.7-1.

c. Obtained from failures/demands or failures/hours.

d. Bayesian update using beta (demand-related) or Poisson (time-related) prior.

e. Average value obtained by combining the data from similar components listed.

f. Combined data from similar components listed.

g. Obtained from 1969 through 1988.

h. Obtained from 1981 through 1988.

i. Obtained from 1984 through 1988 (valve changed in 1983).

j. Failure modes not distinguished.

k. Obtained from 1986 through 1988 (system installed in 1986).

l. Obtained from 1988 through 1990 (system installed in 1988).

m. Obtained from 1987, 1988, and June 1989 through June 1990.

n. Obtained from the initiating event (LPC) frequency from Section 3.2. Because only those events that occurred while ATR was at power were considered, the conversion from per year to per hour was accomplished by dividing by (8760)(0.7) = 6132 h, where 0.7 is the ATR availability factor.

o. Obtained from the initiating event (LCP) frequency from Section 3.2. Because all events were considered, as long as whatever caused the power outage could have occurred while ATR is at power, the conversion from per year to per hour was accomplished by dividing by 8760 h.

1.3.8 System Quantification

Results of the individual system quantifications are listed in Table 12. These results were obtained with all support systems included. In general, a mission time of 24 hours was assumed. System unavailabilities from Table 12 cannot be multiplied together to obtain sequence frequencies (the next section) because of system interdependencies and system dependencies upon IEs.

1.3.9 Accident Sequence Quantification

Internal event accident sequences leading to fuel damage as indicated in the ETs were quantified by using the technique of fault tree linking.³ The system fault trees were edited using CAFTA^{11,12} on a microcomputer. Accident sequence quantification and cutset manipulation were also performed using CAFTA. An accident sequence cutset truncation value of $1.0\text{E}-10/\text{yr}$ was used.

The results of the accident sequence quantification are presented in Tables 13 through 16. The total fuel damage frequency for ATR resulting from the internal events Level 1 analysis is $2.4\text{E}-5/\text{yr}$. A summary of the dominant sequences (those sequences with frequencies greater than or equal to $1.0\text{E}-8/\text{yr}$) is presented in Table 13. The top 15 sequences compose over 97% of the sum of the frequencies of the dominant sequences. The rest of the dominant sequences each account for less than 0.5% of the total of the frequencies of the dominant sequences. The top 15 sequences are briefly described below. (The descriptions have been shortened to remove discussion of specific equipment failures.)

- **Sequence LDP9** -- This sequence represents over 27% of the sum of the frequencies of the dominant sequences. The initiator, loss of diesel power, results in an experiment loop caused reactivity ramp due to a low flow condition in all loops.

Table 12. System failure probabilities^a

<u>Fault Tree Top Event</u>	<u>System</u>	<u>Probability</u>
BBP23001	Utility PCC 250 Vdc bus 670-E-23	2.4E-6
BBPU7001	Instrument uninterruptible power supply (UPS) ac panel 670-E-117	7.1E-5
CDPCC001	Building 609 commercial/diesel 480 Vac Bus 609-MCC-1	1.3E-4
CDPE1001	4160 Vac commercial bus 670-E-1	2.9E-4
CDPE2001	4160 Vac commercial bus 670-E-2	2.9E-4
CDPE5001	480 Vac commercial bus 670-E-5	3.2E-4
CDPE7001	480 Vac commercial bus 670-E-7	3.2E-4
DCP15001	480 Vac diesel/commercial bus 670-E-15	3.6E-4
DCPTR001	Failure of automatic transfer to bus 670-E-7 upon loss of diesel bus	3.6E-4
DGPDS001	Failure of power from all three diesel generators	2.7E-3
DGPDS102	Failure of diesel generator 670-M-42	1.1E-2
DGPDS201	Failure of diesel generator 670-M-43	2.2E-1
DGPDS202	Failure of diesel generator 674-M-6	2.5E-1
DGPE3001	Failure of 4160 Vac diesel bus 670-E-3	7.4E-4
ESFSP001	Failure of engineered safety feature (ESF) pressurizing pump trip	1.2E-3
FISNF001	Failure of firewater injection	4.3E-5
IASFC001	Failure of instrument air	4.3E-4
LDWPF001	Failure of LDW purge	4.4E-2

Table 12. (continued)

<u>Fault Tree Top Event</u>	<u>System</u>	<u>Probability</u>
MKPCM001	Failure of core emergency makeup	4.1E-2
MKPCV020	Failure of either safety relief valve to close	6.2E-3
MKPEV010	Failure of both safety relief valves to open	6.2E-3
MKPHP001	Failure of high-pressure plant protective system (PPS) trip (HPT) reset and inventory control	1.0E-2
MKPMR001	Makeup control not restored	8.1E-2
MKPNC040	Failure of FCV-1-8 to close	1.2E-2
MKPNF030	Failure of PCV-1-1 to close	1.2E-2
MKPRV001	Failure of either safety relief valve to open	2.2E-4
PCSEF001	Failure of emergency flow given ac power recovery	7.8E-4
PCSEF050	Failure of backup emergency flow pump M-11	4.8E-3
PCSEF110	Failure of emergency flow pump M-10, given ac power recovery	8.1E-3
PCSEF201	Failure of emergency flow pump M-10	1.6E-3
PCSPF001	Failure of PCS pump restart	8.6E-2
PPS3R001	Failure to insert at least 3 safety rods into the core	6.5E-7
PPSCP001	Failure to shut down reactor on loss of commercial power	1.8E-5
PPSDP001	Failure to shut down reactor on loss of diesel power	1.8E-5

Table 12. (continued)

<u>Fault Tree Top Event</u>	<u>System</u>	<u>Probability</u>
PPSDT001	Failure to shut down reactor on high quadrant differential temperature	1.8E-5
PPSFR001	Failure to shut down reactor on fast ramp from high range	1.8E-5
PPSHPO01	Failure to shut down reactor on high PCS pressure	1.8E-5
PPSHS001	Failure to shut down reactor on loss of heat sink	1.8E-5
PPSLF001	Failure to shut down reactor on low flow	1.8E-5
PPSLL001	Failure to shut down reactor on low range loop ramp	1.8E-5
PPSLP001	Failure to shut down reactor on low PCS pressure	1.8E-5
PPSMS001	Failure to shut down reactor by manual scram	2.8E-3
PPSSR001	Failure to shut down reactor on slow ramp	2.2E-5
RRSFI001	Failure to initiate reverse on high Log/N period	1.3E-5
RSSHPO01	Failure to initiate setback on high quadrant power	1.4E-3
RSSHT001	Failure to initiate setback on high vessel inlet temperature	1.4E-3
VVSDP001	Failure to initiate vessel venting	1.1E-4

a. The system failure probabilities are minimal cutset upper bound approximations.

Table 13. Summary of dominant internal event accident sequences

<u>Sequence Name</u>	<u>Frequency (/yr)</u>	<u>Percent</u>	<u>Sequence Name</u>	<u>Frequency (/yr)</u>	<u>Percent</u>
LDP9	6.680E-06	27.52%	LPP12	1.645E-07	0.68%
LCP20	4.484E-06	18.48%	LDC6	1.049E-07	0.43%
RLH10	3.621E-06	14.92%	LPP3	9.304E-08	0.38%
LCP17	2.962E-06	12.20%	LIA19	8.226E-08	0.34%
LCP15	2.099E-06	8.65%	RSD7	6.367E-08	0.26%
S12-2	1.082E-06	4.46%	RIL1	5.000E-08	0.21%
RSD8	6.043E-07	2.49%	LDP8	3.704E-08	0.15%
DLC1	5.000E-07	2.06%	CAN2	3.360E-08	0.14%
S12-3	4.562E-07	1.88%	RPF2	3.000E-08	0.12%
RFH10	3.291E-07	1.36%	RLH8	2.365E-08	0.10%
LDC15	2.468E-07	1.02%	AHT6	2.264E-08	0.09%
LPF10	2.468E-07	1.02%	AHT4	1.606E-08	0.07%
HCP19	1.645E-07	0.68%	CAN3	1.344E-08	0.06%
			TOTAL	2.427E-05/yr	

Table 14. Summary of dominant internal event sequences by plant damage state

<u>PDS</u>	<u>Sequence</u>	<u>Frequency (/yr)</u>	<u>Percent</u>
P1	LPP3	9.304E-08 (55.0%)	
	RSD7	6.367E-08 (37.6%)	
	TOTAL	1.692E-07	0.7%
P2	LCP15	2.099E-06 (72.6%)	
	RSD8	6.043E-07 (20.9%)	
	LDC6	1.049E-07 (3.6%)	
	LDP8	3.704E-08 (1.3%)	
	RLH8	2.365E-08 (0.9%)	
	TOTAL	2.891E-06	11.9%
P3	S12-2	1.082E-06 (100%)	
	TOTAL	1.082E-06	4.5%
P3M	DLC1	5.000E-07 (100%)	
	TOTAL	5.000E-07	2.1%
P4	LDP9	6.680E-06 (34.3%)	
	LCP20	4.484E-06 (23.1%)	
	RLH10	3.621E-06 (18.6%)	
	LCP17	2.962E-06 (15.2%)	
	S12-3	4.562E-07 (2.3%)	
	RFH10	3.291E-07 (1.7%)	
	LDC15	2.468E-07 (1.3%)	
	LPF10	2.468E-07 (1.3%)	
	HCP19	1.645E-07 (0.8%)	
	LPP12	1.645E-07 (0.8%)	
	LIA19	8.226E-08 (0.4%)	
	AHT4	1.606E-08 (0.1%)	
	TOTAL	1.947E-05	81.2%
P5	RPF2	3.000E-08 (100%)	
	TOTAL	3.000E-08	0.1%

Table 14. (continued)

<u>PDS</u>	<u>Sequence</u>	<u>Frequency (/yr)</u>	<u>Percent</u>
P5M	RIL1	5.000E-08 (68.8%)	
	AHT6	2.264E-08 (31.2%)	
	TOTAL	7.264E-08	0.3%
P6	None $\geq 1.0\text{E-}10$		
	TOTAL	0	0%
P7	CAN2	3.360E-08 (71.4%)	
	CAN3	1.344E-08 (28.6%)	
	TOTAL	4.704E-08	0.2%

Table 15. Summary of dominant internal event sequences by initiator

Initiator		Number of Sequences >1.0E-10	Frequency (/yr)	Percent
LCP	Loss of commercial power	5	9.549E-06	39.3%
LDP	Loss of ATR diesel power	4	6.722E-06	27.7%
RLH	Loop ramp from high range	5	3.658E-06	15.1%
S12	Small LOCA	2	1.538E-06	6.3%
RSD	Reactor shutdown	2	6.679E-07	2.8%
DLC	Direct-damage large LOCA	1	5.000E-07	2.1%
LDC	Loss of diesel-commercial power	5	3.621E-07	1.5%
RFH	Fast ramp from high range	4	3.305E-07	1.4%
LPP	Low PCS pressure	5	2.606E-07	1.1%
LPF	Low PCS flow	4	2.477E-07	1.0%
HCP	High coolant pressure	4	1.656E-07	0.7%
LIA	Loss of instrument air	2	8.263E-08	0.3%
RIL	Large reactivity insertion	1	5.000E-08	0.2%
CAN	Storage canal draining	2	4.704E-08	0.2%
AHT	High temperature loop over-heating	2	3.870E-08	0.2%
RPF	Very large reactivity insertion	1	3.000E-08	0.1%
LAC	Loss of all ac power	5	1.160E-08	0.1%
RLP	Lobe power imbalance	2	2.640E-09	0.0%
RSH	Slow ramp from high range	1	7.495E-10	0.0%
LLC	Large LOCA	1	6.300E-10	0.0%
LHS	Loss of heat sink	-	-	-
ISF	Interfacing system LOCA (FIS)	-	-	-
ISL	Interfacing system LOCA (LDW)	-	-	-
TOTAL		58	2.427E-05	100%

Table 16. Summary of dominant internal event sequences by type

<u>Sequence Type</u>	<u>Sequence</u>	<u>Frequency (/yr)</u>	<u>Percent</u>
ATWS	LDP9	6.680E-06	(34.3%)
	LCP20	4.484E-06	(23.1%)
	RLH10	3.621E-06	(18.6%)
	LCP17	2.962E-06	(15.2%)
	S12-3	4.562E-07	(2.3%)
	RFH10	3.291E-07	(1.7%)
	LDC15	2.468E-07	(1.3%)
	LPF10	2.468E-07	(1.3%)
	HCP19	1.645E-07	(0.8%)
	LPP12	1.645E-07	(0.8%)
	LIA19	8.226E-08	(0.4%)
	AHT4	1.606E-08	(0.1%)
	TOTAL	1.947E-05	80.2%
LOSS OF FLOW (pressurized)	LCP15	2.099E-06	(73.2%)
	RSD8	6.043E-07	(21.1%)
	LDC6	1.049E-07	(3.7%)
	LDP8	3.704E-08	(1.3%)
	RLH8	2.365E-08	(0.8%)
	TOTAL	2.891E-06	11.9%
LOCA	S12-2	1.082E-06	(100%)
	TOTAL	1.082E-06	4.5%
DIRECT DAMAGE	DLC1	5.000E-07	(83.0%)
	RIL1	5.000E-08	(8.3%)
	RPF2	3.000E-08	(5.0%)
	AHT6	2.264E-08	(3.8%)
	TOTAL	6.026E-07	2.5%
LOSS OF INVENTORY CONTROL	LPP3	9.304E-08	(100%)
	TOTAL	9.304E-08	0.4%

Table 16. (continued)

<u>Sequence Type</u>	<u>Sequence</u>	<u>Frequency (/yr)</u>	<u>Percent</u>
LOSS OF FLOW (depressurized)	RSD7	6.367E-08 (100%)	
	TOTAL	6.367E-08	0.3%
CANAL DRAIN	CAN2	3.360E-08 (71.4%)	
	CAN3	1.344E-08 (28.6%)	
	TOTAL	4.704E-08	0.2%

This sequence results in PDS P4, an ATWS potentially causing core fuel damage.

- **Sequence LCP20** -- This sequence represents over 18% of the sum of the frequencies of the dominant sequences. The initiator is loss of commercial power. This sequence results in PDS P4, an ATWS potentially causing core fuel damage.
- **Sequence RLH10** -- This sequence represents almost 15% of the sum of the frequencies of the dominant sequences. It is initiated by an experiment loop upset resulting in a reactivity ramp from high range. This sequence results in PDS P4, an ATWS potentially causing core fuel damage.
- **Sequence LCP17** -- This sequence represents over 12% of the sum of the frequencies of the dominant sequences. The initiator is loss of commercial power resulting in low primary and secondary flow. This sequence results in PDS P4, an ATWS potentially causing core fuel damage.
- **Sequence LCP15** -- This sequence represents about 8.6% of the sum of the frequencies of the dominant sequences. The sequence is initiated by loss of commercial power resulting in low primary and secondary flow. This sequence results in PDS P2, transients with failure of PCS recirculation and failure to depressurize the PCS.
- **Sequence S12-2** -- This sequence represents about 4.5% of the sum of the frequencies of the dominant sequences. The sequence is initiated by a small LOCA. This sequence results in PDS P3.
- **Sequence RSD8** -- This sequence represents about 2.5% of the sum of the frequencies of the dominant sequences. The sequence is initiated by an anticipated reactor shutdown. This sequence results

in PDS P2, transients with failure of PCS recirculation and failure to depressurize the PCS.

- **Sequence DLC1** -- This sequence represents over 2% of the sum of the frequencies of the dominant sequences. The initiator is a direct-damage large LOCA resulting in PDS P3M.
- **Sequence S12-3** -- This sequence represents about 1.9% of the sum of the frequencies of the dominant sequences. The initiator is a small LOCA. This sequence results in PDS P4, an ATWS potentially causing core fuel damage.
- **Sequence RFH10** -- This sequence represents about 1.4% of the sum of the frequencies of the dominant sequences. The initiator is a fast reactivity ramp from high range. This sequence results in PDS P4, an ATWS potentially causing core fuel damage.
- **Sequence LDC15** -- This sequence represents about 1% of the sum of the frequencies of the dominant sequences. The initiator is a loss of the diesel-commercial power bus. This sequence results in PDS P4, an ATWS potentially causing core fuel damage.
- **Sequence LPF10** -- This sequence represents about 1% of the sum of the frequencies of the dominant sequences. The initiator is low PCS flow (without failure of commercial power). This sequence results in PDS P4, an ATWS potentially causing core fuel damage.
- **Sequence HCP19** -- This sequence represents almost 0.7% of the sum of the frequencies of the dominant sequences. The initiator is high coolant pressure. This sequence results in PDS P4, an ATWS potentially causing core fuel damage.
- **Sequence LPP12** -- This sequence represents about 0.7% of the sum of the frequencies of the dominant sequences. The initiator is low PCS

pressure. This sequence results in PDS P4, an ATWS potentially causing core fuel damage.

- **Sequence LDC6** -- This sequence represents about 0.4% of the sum of the frequencies of the dominant sequences. The sequence is initiated by the loss of the diesel-commercial power bus. This sequence results in PDS P2, transients with failure of PCS recirculation and failure to depressurize the PCS.

The results of the sequence quantification process by plant damage state are presented in Table 14. The results of the sequence quantification process grouped by initiator are presented in Table 15. Finally, results grouped by functional failure are presented in Table 16. The dominant sequences were divided into 7 sequence types:

- **ATWS** Failure to reduce power by scram, reverse, or setback
- **LOF (pressurized)** Loss of flow with the PCS pressurized. If a sequence could belong either to this category or to the LOSS OF INVENTORY CONTROL category, this category was given precedence.
- **LOF (depressurized)** Loss of flow with PCS depressurized. If a sequence could belong either to this category or to the LOSS OF INVENTORY CONTROL category, this category was given precedence.
- **LOCA** Loss-of-coolant accidents without inventory makeup
- **Direct damage** Sequences for which no safety function could have prevented core damage (i.e., in some sequences of this type, coolant flow and volume are maintained). Most sequences of this type result in moderate initial core damage.
- **Loss of inventory control** Sequences in which flow may be provided, but no source of inventory makeup is available. If a sequence could belong to either this category or a LOF category, LOF was given precedence.
- **Canal** Canal drainage events.

1.3.10 External and Spatially-Dependent Internal Events Analysis

The external event analysis covers risk to the ATR core and the storage canal fuel while ATR is at power. The ATR external event analysis is a comprehensive, state-of-the-art analysis that examined an exhaustive list of 22 external events applicable to the ATR. This list was generated from a review of information concerning ATR site characteristics contained in Chapter 2 of the ATR Safety Analysis Reports Collection and from Table 10-1 of the PRA Procedures Guide³. Screening guidelines were applied to eliminate events from further consideration and to consolidate others. The list of external events was reduced to 10 events for further in-depth analyses listed below:

- Internal fire
- Internal flooding
- Seismic
- External flooding
- Volcanic activity
- Onsite chemical release
- Industrial facility accident
- Transportation accident
- Extreme wind
- Aircraft impact.

A comprehensive search for seismic, fire, flood, and wind interaction was performed. A detailed human reliability analysis specific to flood-, fire-, and seismic-related conditions was performed. Many of these features of the study are not found in recent external event analyses.

The total mean frequency of fuel damage (including both core and fuel canal) at ATR from the external events was estimated to be $2.5E-5/\text{yr}$. The contribution from each external event initiator is presented in Table 17. The seismic events were the most dominating, with 75% of the total fuel damage frequency from external events, followed by volcanism

Table 17. Summary of dominant external event initiators

<u>External Event</u>	<u>Mean Fuel Damage Frequency/yr</u>
Seismic	1.9E-5
Volcanic activity	2.8E-6
Extreme wind	1.2E-6
Internal flooding	1.1E-6
Internal fire	1.0E-6
Aircraft impact	1.5E-7
External flooding	1.3E-7
Onsite chemical release	< 1.0E-7
Industrial facility accident	< 1.0E-7
Transportation accident	< 1.0E-7
Total	2.5E-5

with 11%, extreme wind with almost 5%, and internal flooding and internal fire, each with approximately 4%. The contribution for volcanism, however, is based on a screening analysis. A more detailed assessment will be performed when additional INEL data for this class of event are available. The total contribution to the core damage risk from the remaining external events was approximately 1%.

The mean frequency of ATR canal fuel damage from seismic events was estimated to be $9.1\text{E-}7/\text{yr}$. Risk to canal fuel from all other external events was found to be less than $1.0\text{E-}7/\text{yr}$.

Table 18 provides a summary of external event fuel damage sequences with mean frequencies above $1.0\text{E-}8/\text{yr}$. These external events analysis results are based on the completion of recommended facility external events risk reduction upgrades that have been committed to and are to be completed by 1993.

A brief description of the methodologies used in the dominant external event analyses is provided below.

Seismic Analysis--The ATR seismic risk analysis is a comprehensive, state-of-the-art study that included a comprehensive search for seismic, and seismically-induced fire and flood interactions. A detailed human reliability analysis was performed specific to seismic-related conditions. Human error rates were modeled as a function of instrumentation available, and the instrumentation at ATR was judged to be able to withstand an earthquake of 0.24 g. Therefore, moderate operator stresses were assumed for earthquake magnitudes below 0.24 g; above this magnitude severe operator stresses were assumed. Seismically-induced direct-damage and non-direct-damage loss-of-coolant-accident (LOCA) events were explicitly modeled based on the location of the break. Many of these features of the study are not found in recent seismic analyses. For this study, a family of seismic hazard curves generated for the EBR-II reactor PRA (located at the INEL) was used. If more detailed seismic hazard curves are developed for the INEL as part of the New Production Reactor project, then the seismic risk will be reanalyzed using the newer hazard curves.

Table 18. Summary of external event fuel damage sequences

Number	Sequence	Mean Frequency/yr	Plant Damage State	Description
1.	EQE6	1.3E-5	P1	Seismic event results in failure of coolant flow systems and low-pressure injection systems. This leads to core uncover and core-wide fuel damage. (Loss of commercial power is assumed due to seismic event.)
2.	EQE7	4.0E-6	P2	Seismic event causes the loss of coolant flow. Subsequent failures lead to core uncover and core-wide fuel damage. (Loss of commercial power is assumed due to seismic event.)
3.	VOLCAN	2.8E-6	P2*	Basaltic volcanism causes lava flow that reaches the ATR reactor building. Core-wide fuel damage was assumed for this event.
4.	EQE14	9.4E-7	P4	Seismic event with failure to scram (ATWS) potentially causing core fuel damage. (Loss of commercial power is assumed due to seismic event.)
5.	EQC4	8.0E-7	P7	Seismic event causes the loss of storage canal integrity that results in canal drainage. (No simultaneous reactor LOCA.) Failure of both low-pressure injection systems leads to fuel uncover in the storage canal. (Loss of commercial power is assumed due to seismic event.)
6.	W6M3A	7.6E-7	P2	Internal flooding caused by inadvertent actuation of wet pipe sprinkler system in the process control room, subsequent random failures, and failure to depressurize lead to core uncover and core-wide fuel damage.
7.	WND7	7.0E-7	P2	Extreme wind event causes the loss of coolant flow. Subsequent failure to depressurize leads to core uncover and core-wide fuel damage. (Loss of commercial power is assumed due to extreme wind.)
8.	WND6	4.5E-7	P1	Extreme wind event with subsequent failure of low-pressure injection systems leads to core uncover and core-wide fuel damage. (Loss of commercial power is assumed due to extreme wind.)
9.	FIR2F26	3.4E-7	P2	Fire in the switchgear room disables safety equipment. Subsequent random failures and failure to depressurize lead to core uncover and core-wide fuel damage.
10.	FIRATWS	2.3E-7	P4	Internal fire event with failure to scram (ATWS) potentially causing core fuel damage.
11.	W7ALL	2.3E-7	P4	Internal flooding event with failure to scram (ATWS) potentially causing core fuel damage.

Table 18. (continued)

Number	Sequence	Mean Frequency/yr	Plant Damage State	Description
12.	WND8	2.1E-7	P4	Extreme wind event with failure to scram (ATWS) potentially causing core fuel damage. (Loss of commercial power is assumed due to extreme wind.)
13.	FIRIM36	2.1E-7	P2	Fire in the process control room disables safety equipment. Subsequent failure to depressurize leads to core uncover and core-wide fuel damage.
14.	EQC5	1.1E-7	P7	Seismic event causes loss of storage canal integrity resulting in drainage of the canal. Operator fails to recognize the loss of water from the canal and uncover of fuel leads to fuel damage. (Loss of commercial power is assumed due to seismic event.)
15.	EXF19	8.5E-8	P2	External flooding causes a station blackout due to flooding of the diesel generator area. (Operator is not aware of the dam failure until a loss of commercial power occurs.) Subsequent failure to depressurize leads to core uncover and core-wide fuel damage.
16.	FIR5F26	6.4E-8	P2	Fire in the switchgear room disables safety equipment. Random failure of emergency recirculation and depressurization lead to core uncover and core-wide fuel damage.
17.	EXF11	4.3E-8	P2	External flooding causes a station blackout due to flooding of the diesel generator area. (Operator is notified of the dam failure, but reactor is not scrammed until a loss of commercial power occurs.) Subsequent failure to depressurize leads to core uncover and core-wide fuel damage.
18.	FIR1V16	3.8E-8	P2	Fire in the nozzle trench disables safety equipment. Random failure of emergency recirculation and depressurization lead to core uncover and core-wide fuel damage.
19.	FIR2F46	3.3E-8	P2	Fire in the pump room disables safety equipment. Subsequent failure to depressurize leads to core uncover and core-wide fuel damage.
20.	W6W7A	3.0E-8	P2	Internal flooding caused by leakage of wet pipe sprinkler system in the reactor main floor area. Random failure of emergency recirculation and depressurization lead to core uncover and core-wide fuel damage.

Table 18. (continued)

Number	Sequence	Mean Frequency/yr	Plant Damage State	Description
21.	W6F2BR	2.8E-8	P2	Internal flooding caused by UCW piping rupture along the north wall of the switchgear room disables safety equipment. Power losses result in loss of core emergency recirculation. Subsequent failure to depressurize leads to core uncover and core-wide fuel damage.
22.	FIR7AB6	2.6E-8	P2	Fire in the east hallway (in the first basement) disables safety equipment. Random failure of emergency recirculation and depressurization lead to core uncover and core-wide fuel damage.
23.	W6F2CL	2.1E-8	P2	Internal flooding caused by UCW piping leakage disables safety equipment. Power losses result in loss of core emergency recirculation. Subsequent failure to depressurize leads to core uncover and core-wide fuel damage.
24.	FIR6F26	1.5E-8	P2	Fire in the switchgear room in S.E. corner disables safety equipment. Random failure of emergency recirculation and depressurization lead to core uncover and core-wide fuel damage.
25.	FIR8F26	1.5E-8	P2	Fire in the switchgear room in N.E. corner disables safety equipment. Random failure of emergency recirculation and depressurization lead to core uncover and core-wide fuel damage.
26.	FIR9F26	1.5E-8	P2	Fire in the switchgear room in N.E. corner disables safety equipment. Random failure of emergency recirculation and depressurization lead to core uncover and core-wide fuel damage.
27.	FIR3F46	1.4E-8	P2	Fire in the pump room fire disables safety equipment. Subsequent failure to depressurize leads to core uncover and core-wide fuel damage.
28.	FIR1F46	1.2E-8	P2	Fire in the pump room fire disables safety equipment. Subsequent failure to depressurize leads to core uncover and core-wide fuel damage.
Total		2.5E-5/yr		

a. Plant damage state P2 was assumed for this scenario.

The seismic risk analysis was performed in two parts. The first involved the development of preliminary fuel damage results and identification of potential vulnerabilities of the facility. The second phase involved reevaluation and requantification of the seismic risk based on the implementation of equipment upgrades identified during the analysis.

EQE Engineering Inc. (EQE) performed the seismic fragility evaluation of structures and equipment. Two extensive plant walkdowns were performed by EQE engineers. Two event trees were developed: one for fuel damage in the core, and one for fuel damage in the storage canal. The most likely early consequence of a significant seismic event at INEL is a loss of commercial power. Therefore, the event tree for the core was based on the loss-of-commercial-power event in the internal events analysis. The canal event tree was based on a loss-of-canal-integrity (crack or rupture) event utilizing the event tree in the internal events analysis. Loss of component(s) due to seismically-initiated flood or fire was modeled explicitly at the fault tree level. EG&G Idaho developed the Boolean equations for seismically-induced sequences that could result in fuel damage. EQE quantified these sequences using their proprietary computer code EQESRA. The final results provided five fuel damage sequences with frequencies greater than $1.0\text{E-}8/\text{yr}$.

Internal Fire Analysis--A comprehensive state-of-the-art internal fire analysis was performed that included comprehensive facility walkdowns, a vital area analysis using a location-transformation technique, a fire-specific human reliability analysis, and COMPBRN¹³ analyses of fire propagation for selected areas. A screening analysis eliminated all but the most important scenarios from further consideration. Refined analyses of the remaining internal fire scenarios resulted in 12 fuel damage sequences with frequencies greater than $1.0\text{E-}8/\text{yr}$.

Internal Flooding Analysis--The ATR internal flooding analysis is a comprehensive, state-of-the-art study. Vital areas of the plant were identified by both comprehensive plant walkdowns and a vital area analysis as in the fire analysis. Previous internal flooding events at U. S. commercial

nuclear power plants were examined to use as a guide to the type of events that might occur. A search was also performed to identify internal flooding events that had occurred at ATR. Leakage and rupture events, and also inadvertant actuation of wet pipe and preaction sprinkler systems were postulated and analyzed. Refined analyses of the unscreened internal flooding scenarios resulted in five fuel damage sequences with frequencies greater than $1.0\text{E}-8/\text{yr}$.

Volcanic Activity Analysis--Basaltic volcanism was determined to be predominant for ATR. The frequency of lava reaching ATR was determined by combining the frequency of episodes of basaltic activity (one every 18,000 years) with the conditional probability of lava reaching ATR (0.05, given that ATR is 12 miles from the centerline of the volcanic rift zone). Both values were determined from historical data at the INEL. Fuel damage was assumed to occur if lava reached ATR. No credit was taken for potential recovery actions because basaltic lava flows can move faster than 40 miles per hour.

Extreme Wind Analysis--A comprehensive ATR extreme wind analysis was performed using state-of-the-art techniques. Site-specific wind hazard curves previously developed for the INEL by Lawrence Livermore National Laboratory were used. Site-specific wind fragility curves were generated by EQE for important structures and components. A comprehensive search for wind and flood interactions was performed. Human error rates developed for severe operator stress in the seismic analysis were used. Quantification of wind risk resulted in three fuel damage sequences with frequencies greater than $1.0\text{E}-8/\text{yr}$.

1.3.11 Uncertainty, Sensitivity, and Importance Analysis

An uncertainty analysis of the ATR PRA results was performed analyzing what are termed "data uncertainties." The UNCERT^{14,15} code was used to propagate basic event and initiator error factors through the plant models using a Monte Carlo simulation. The Level 1 uncertainty results are presented

by plant damage state in Table 19 and by sequence in Table 20. The total fuel damage frequency has an error factor (95th percentile/50th) of 5.

Modeling uncertainties were analyzed with sensitivity analyses. The sensitivity analyses results for internal events are presented in Section 3.11.2.2 and for external events in the corresponding external event discussion in Section 3.10.

The results of the importance analysis are presented in Table 21. Four importance rankings are listed: Fussell-Vesely, Birnbaum, risk reduction ratio, and risk achievement ratio.

1.4 CONTAINMENT ANALYSIS

The Level 2 (containment) analysis is ongoing and will be summarized in an updated version of Reference 1 when it is completed.

1.5 CONSEQUENCE ANALYSIS

The Level 3 (consequence) analysis is ongoing and will be summarized in an updated version of Reference 1 when it is completed.

1.6 ASSURANCE OF TECHNICAL ADEQUACY

Because PRAs deal with low frequency events, the technical adequacy or quality of the results cannot be judged based on real-life experience. Instead, the quality of a PRA is normally judged by the methods used and the applications of such methods. Areas of concern with respect to technical adequacy include the study team, the technical work, and the documentation. Essentially two separate study teams were used to complete the ATR Level 1 PRA. The combined teams provided excellent coverage in terms of both ATR-specific knowledge and overall PRA knowledge. Detailed guidelines were used to guide each PRA task, based on various PRA procedures guides. Close cooperation and actual involvement in the PRA from ATR personnel ensured that

Table 19. Level 1 uncertainty results by plant damage state

	Mean	5th	Median	95th
P1	1.37E-05	1.31E-07	2.01E-06	4.47E-05
P2	1.56E-05	1.34E-06	6.67E-06	4.66E-05
P3	8.87E-07	8.36E-09	1.84E-07	3.38E-06
P3M	4.79E-07	2.05E-09	5.59E-08	1.93E-06
P4	2.21E-05	2.12E-06	1.08E-05	6.96E-05
P5	2.93E-08	1.54E-10	4.50E-09	1.14E-07
P5M	7.96E-08	5.74E-10	1.20E-08	2.07E-07
P6	No sequences			
P7	8.21E-07	8.89E-09	1.29E-07	2.86E-06
ALL SEQUENCES	5.14E-05	9.56E-06	2.98E-05	1.44E-04

Table 20. Level 1 sequence results of uncertainty analysis

	<u>Mean</u>	<u>5th</u>	<u>Median</u>	<u>95th</u>
INTERNAL EVENTS				
LDP9	5.84E-06	1.03E-07	1.43E-06	2.52E-05
LCP20	4.78E-06	2.66E-07	1.91E-06	1.67E-05
RLH10	3.54E-06	2.12E-07	1.51E-06	1.25E-05
LCP17	3.17E-06	2.07E-07	1.32E-06	1.09E-05
LCP15	3.75E-06	8.16E-08	8.14E-07	1.03E-05
S12-2	1.04E-06	7.51E-09	1.89E-07	4.21E-06
RSD8	6.12E-07	3.33E-09	7.04E-08	2.51E-06
DLC1	4.79E-07	2.05E-09	5.60E-08	1.93E-06
S12-3	4.67E-07	6.75E-09	9.54E-08	1.72E-06
RFH10	3.68E-07	4.57E-09	6.89E-08	1.36E-06
LDC15	2.59E-07	1.06E-08	9.64E-08	8.96E-07
LPF10	2.66E-07	1.16E-08	9.47E-08	9.49E-07
HCP19	1.84E-07	2.10E-09	3.58E-08	6.69E-07
LPP12	1.91E-07	2.03E-09	3.50E-08	5.97E-07
LDC6	2.12E-07	3.85E-09	3.98E-08	5.88E-07
LPP3	1.00E-07	1.63E-10	6.42E-09	2.83E-07
LIA19	9.12E-08	1.00E-09	1.72E-08	3.41E-07
RSD7	7.32E-08	1.29E-10	4.64E-09	2.28E-07
RIL1	4.81E-08	1.88E-10	5.79E-09	1.78E-07
LDP8	4.25E-08	5.25E-11	2.25E-09	1.26E-07
CAN2	3.37E-08	1.80E-10	5.13E-09	1.37E-07
RPF2	2.98E-08	1.54E-10	4.29E-09	1.15E-07
RLH8	3.16E-08	8.62E-11	2.51E-09	9.95E-08
AHT6	1.75E-08	1.10E-11	7.64E-10	6.36E-08
AHT4	1.52E-08	6.24E-11	1.95E-09	5.47E-08
CAN3	1.35E-08	6.91E-11	1.97E-09	5.32E-08
<i>ALL INTERNAL EVENT SEQUENCES</i>	<i>2.87E-05</i>	<i>3.83E-06</i>	<i>1.47E-05</i>	<i>7.85E-05</i>
EXTERNAL EVENTS				
Seismic				
EQE6	1.30E-05	5.09E-08	2.45E-06	7.00E-05
EQE7	3.97E-06	6.15E-08	1.07E-06	1.96E-05
EQE14	9.43E-07	4.41E-11	2.03E-08	2.10E-06
EQC4	8.02E-07	3.24E-09	9.03E-08	4.15E-06
EQC5	1.09E-07	3.86E-12	1.66E-09	2.17E-07

Table 20. (continued)

	Mean	5th	Median	95th
Internal Flood				
W6M3A	2.98E-06	1.28E-08	2.12E-07	3.78E-06
W7ALL	2.23E-07	2.16E-09	4.35E-08	9.48E-07
W6F2BR	5.61E-08	4.54E-10	7.18E-09	1.20E-07
W6F2CL	2.99E-08	7.84E-11	2.48E-09	8.79E-08
W6M7A	2.89E-08	2.66E-10	5.30E-09	1.09E-07
Volcanic Activity				
VOLCAN	2.68E-06	1.15E-08	3.13E-07	1.08E-05
Extreme Wind				
WND7	7.00E-07	2.10E-11	3.60E-08	2.30E-06
WND6	4.50E-07	9.60E-11	3.90E-08	1.90E-06
WND8	2.10E-07	3.70E-13	1.40E-09	4.10E-07
Internal fire				
FIR1M36	2.63E-07	5.59E-10	2.21E-08	7.95E-07
FIR2F26	2.57E-07	9.42E-10	3.00E-08	1.02E-06
FIRATWS	2.40E-07	2.07E-09	4.39E-08	8.33E-07
FIR8F26	4.30E-08	1.08E-11	6.84E-10	4.86E-08
FIR5F26	4.22E-08	1.74E-11	1.55E-09	1.32E-07
FIR1V16	4.17E-08	4.20E-11	1.75E-09	9.79E-08
FIR2F46	3.64E-08	4.37E-11	1.90E-09	1.28E-07
FIR7AB6	2.58E-08	2.61E-11	1.17E-09	7.09E-08
FIR9F26	2.10E-08	1.21E-11	6.71E-10	4.08E-08
FIR3F46	1.68E-08	2.39E-11	8.90E-10	4.69E-08
FIR1F46	1.54E-08	3.27E-11	1.19E-09	5.11E-08
FIR6F26	1.29E-08	5.19E-11	1.74E-09	4.97E-08
ALL EXTERNAL EVENT SEQUENCES	2.32E-05	2.29E-06	1.05E-05	8.19E-05
ALL LEVEL 1 SEQUENCES	5.14E-05	9.56E-06	2.98E-05	1.44E-04

Table 21. Results of Level 1 importance analysis

Event Name	Probability	F-V ^a	Birm ^b	RRR ^c	RAR ^d	Description
EOE6	1.30E-05	2.63E-01	1.00E+00	1.36E+00	2.03E+04	Seismic sequence EQE6
PPS-SYS-CF-SCR00	1.50E-05	1.96E-01	5.24E-01	1.24E+00	1.06E+04	Common cause loss of scram system
LCP	9.00E-01	1.93E-01	1.08E-05	1.24E+00	1.02E+00	Initiating event: loss of commercial power
PPS-LOG-CF-RCCCS	2.60E-06	1.66E-01	1.00E+00	1.20E+00	2.03E+04	Common cause failure of RCCCs to release
LDP	3.66E-01	1.36E-01	1.84E-05	1.16E+00	1.24E+00	Initiating event: loss of diesel power
EQE7	4.00E-06	8.11E-02	1.00E+00	1.09E+00	2.03E+04	Seismic sequence EQE7
RLH	1.10E+00	7.41E-02	3.33E-06	1.08E+00	9.93E-01	Initiating event: reactivity insertion (loop ramp)
DGP-DGN-FR-000M6	1.07E-01	5.93E-02	2.73E-05	1.06E+00	1.49E+00	Diesel generator 674-M-6 fails to run
VOLCAN	2.80E-06	5.67E-02	1.00E+00	1.06E+00	2.03E+04	Volcanic activity
CPR	2.00E-02	4.28E-02	1.05E-04	1.04E+00	3.09E+00	Failure to recover commercial power
PPS-CRM-CF-3R0DS	6.50E-07	4.15E-02	1.00E+00	1.04E+00	2.03E+04	Common cause failure that prevents having at least three rods fully inserted
S12	2.50E-02	3.12E-02	6.15E-05	1.03E+00	2.22E+00	Initiating event: small LOCA
VDF-SYS-XF-OVDP2	1.50E-04	2.92E-02	9.57E-03	1.03E+00	1.95E+02	Operator failure to actuate VWS (VDP2)
FIS-AOV-CA-004F4	4.30E-05	2.52E-02	2.88E-02	1.03E+00	5.84E+02	Common cause loss of both FIS injection paths
VDF-SYS-XF-VDP1D	1.00E-04	1.96E-02	9.64E-03	1.02E+00	1.96E+02	Operator failure to diagnose the need for action (VDP1D)
EQE14	9.40E-07	1.90E-02	1.00E+00	1.02E+00	2.03E+04	Seismic sequence EQE14
DGP-DGN-FS-000M6	3.40E-02	1.88E-02	2.72E-05	1.02E+00	1.53E+00	Diesel generator 674-M-6 fails to start
VWS-REG-FC-0E14C	7.20E-03	1.86E-02	1.28E-04	1.02E+00	3.57E+00	32-VDC-to-24-VDC voltage regulator for panel E14C fails
DGP-SDP-FS-0M6RA	3.00E-02	1.65E-02	2.72E-05	1.02E+00	1.53E+00	Diesel generator 674-M-6 attached cooling pump fails to start
DGP-SDP-FS-M6FOP	3.00E-02	1.65E-02	2.72E-05	1.02E+00	1.53E+00	Diesel generator 674-M-6 engine-driven fuel oil pump fails to start
EQC4	8.00E-07	1.62E-02	1.00E+00	1.02E+00	2.03E+04	Seismic sequence EQC4
CDP-BAC-LP-STPWR	2.40E-03	1.61E-02	3.30E-04	1.02E+00	7.67E+00	132KV ac site loop bus loss of power
VWS-REG-FC-0E14B	7.20E-03	1.58E-02	1.08E-04	1.02E+00	3.18E+00	32VDC-to-24-VDC voltage regulator for panel E14B fails
WM3A	1.90E-03	1.54E-02	4.00E-04	1.02E+00	9.09E+00	Flood initiator: PCR inadvertent sprinkler actuation
WMD7	7.00E-07	1.42E-02	1.00E+00	1.01E+00	2.03E+04	Wind sequence 7
RSD	2.19E-01	1.35E-02	3.05E-08	1.01E+00	9.87E-01	Initiating event: normal/spurious reactor shutdown
CDP-SYS-XF-STPWR	8.00E-02	1.30E-02	8.03E-06	1.01E+00	1.15E+00	Operator failure to realign after recovery of site power
VWS-AOV-CA-02312	6.30E-05	1.22E-02	9.51E-03	1.01E+00	1.94E+02	Common cause failure of vessel vent valves 68-23-1 and 68-23-2 to open
DLC	5.00E-07	1.01E-02	1.00E+00	1.01E+00	2.03E+04	Initiating event: direct-damage LOCA
WMD6	4.50E-07	9.12E-03	1.00E+00	1.01E+00	2.03E+04	Wind sequence 6
FIE-F2	5.01E-03	9.03E-03	8.90E-05	1.01E+00	2.79E+00	Fire initiating event: switchgear room
LDC	7.50E-02	7.34E-03	4.83E-06	1.01E+00	1.09E+00	Initiating event: loss of diesel-commercial bus
GF-F2-2	6.09E-02	6.82E-03	5.61E-06	1.01E+00	1.11E+00	Fire geometric factor: Switchgear room factor 2
NS-F2-2	1.00E+00	6.82E-03	3.37E-07	1.01E+00	1.00E+00	Nonsuppression probability
SF-F2-2	1.00E-01	6.82E-03	3.37E-06	1.01E+00	1.06E+00	Fire severity: switchgear room factor 2
RFH	1.00E-01	6.70E-03	3.31E-06	1.01E+00	1.06E+00	Initiating event: reactivity insertion (fast ramp)
DGP-DGN-TM-000M6	1.10E-02	6.03E-03	2.71E-05	1.01E+00	1.54E+00	Diesel generator 674-M-6 unavailable due to TM
DGP-ICC-FO-0M6RA	1.00E-02	5.48E-03	2.71E-05	1.01E+00	1.54E+00	Diesel generator 674-M-6 cooling system engine-driven pump instrument and control circuit (ICC) failure
DGP-ICC-FO-M6FOP	1.00E-02	5.48E-03	2.71E-05	1.01E+00	1.54E+00	Diesel generator 674-M-6 fuel oil supply engine-driven pump ICC failure
LPP	5.00E-02	5.28E-03	5.21E-06	1.01E+00	1.10E+00	Initiating event: low PCS pressure

Table 21. (continued)

Event Name	Probability	F-V*	Birr ^b	RRR ^c	RAR ^d	Description
LPF	7.50E-02	5.02E-03	3.30E-06	1.01E+00	1.06E+00	Initiating event: low PCS flow
FIE-ATWS	7.00E-02	4.67E-03	3.29E-06	1.00E+00	1.08E+00	Fire initiating event: ATWS
W7ALL	7.00E-02	4.67E-03	3.29E-06	1.00E+00	1.06E+00	Flood initiator: failure to scram given an internal flooding event
DGP-BKR-XM-074FN	8.00E-03	4.37E-03	2.70E-05	1.00E+00	1.54E+00	Breaker on 674-E-1 feeding 674-M-6 radiator fan 4-8 miscalibrated
WND8	2.10E-07	4.26E-03	1.00E+00	1.00E+00	2.03E+04	Wind sequence 8
FIE-M3	5.14E-03	4.17E-03	4.00E-05	1.00E+00	1.81E+00	Fire initiating event: PCR
GF-CR	2.00E-01	4.17E-03	1.03E-06	1.00E+00	1.02E+00	Fire geometric factor: PCR M-3
NS-CR	1.00E+00	4.17E-03	2.06E-07	1.00E+00	1.00E+00	Nonsuppression probability
SF-CR	5.00E-01	4.17E-03	4.11E-07	1.00E+00	1.00E+00	Fire severity: PCR
HCP	5.00E-02	3.36E-03	3.31E-06	1.00E+00	1.06E+00	Initiating event: high PCS pressure
CDP-SYS-FC-STPMR	2.00E-02	3.03E-03	7.49E-06	1.00E+00	1.15E+00	Failure to recover site power within 4 hours
WVS-SYS-FC-NITR1	1.56E-03	3.03E-03	9.60E-05	1.00E+00	2.94E+00	Loss of nitrogen gas backup to valve GB-23-1
DGP-MDP-XF-06272	8.00E-02	2.99E-03	1.84E-06	1.00E+00	1.03E+00	Operator fails to manually start diesel oil transfer pump 627-2 or 627-3
DGP-FAN-FS-074H8	5.00E-03	2.72E-03	2.68E-05	1.00E+00	1.54E+00	Diesel generator 674-M-6 radiator M-8 fan fails to start
DGP-FAN-FS-0M6FO	5.00E-03	2.72E-03	2.68E-05	1.00E+00	1.54E+00	Diesel generator 674-M-6 injector cooling fuel oil cooler fan fails to start
WVS-ICC-FO-GB231	1.00E-03	2.43E-03	1.20E-04	1.00E+00	3.43E+00	Actuation circuitry for valve GB-23-1 fails
EQC5	1.10E-07	2.23E-03	1.00E+00	1.00E+00	2.03E+04	Seismic sequence EQC5
WVS-SYS-FC-NITR2	1.56E-03	2.09E-03	6.61E-05	1.00E+00	2.34E+00	Loss of nitrogen gas backup to valve GB-23-2
PPS-XSM-XF-OPSCR	2.70E-03	2.06E-03	3.77E-05	1.00E+00	1.76E+00	Operator fails to actuate a manual scram
WVS-ICC-FO-GB232	1.00E-03	1.97E-03	9.72E-05	1.00E+00	2.97E+00	Actuation circuitry for valve GB-23-2 fails
LDW-SYS-XF-PURGE	4.11E-02	1.84E-03	2.21E-06	1.00E+00	1.04E+00	Operator fails to initiate LDW purge
LIA	2.50E-02	1.67E-03	3.31E-06	1.00E+00	1.07E+00	Initiating event: loss of instrument air
DGP-BDD-CC-00674	3.00E-03	1.60E-03	2.64E-05	1.00E+00	1.53E+00	Diesel generator 674-M-6 combustion air intake damper fails to open
DGP-MDP-FS-OFTW6	3.00E-03	1.60E-03	2.64E-05	1.00E+00	1.53E+00	Diesel generator 674-M-6 fuel oil pump for filling daytank 674-M-9 FTS
WVS-AOV-CC-GB231	6.30E-04	1.49E-03	1.17E-04	1.00E+00	3.37E+00	Vessel vent valve GB-23-1 fails to open
PCS-ORF-PG-0503A	3.00E-04	1.45E-03	2.38E-04	1.00E+00	5.82E+00	Emergency PCS pumps combined discharge flow element plugged
GF-F2-5	7.50E-02	1.29E-03	8.46E-07	1.00E+00	1.02E+00	Fire geometric factor: switchgear room factor 5
NS-F2-5	1.00E+00	1.29E-03	6.34E-08	1.00E+00	1.00E+00	Nonsuppression probability
SF-F2-5	1.00E+00	1.29E-03	6.34E-08	1.00E+00	1.00E+00	Fire severity: switchgear room factor 5
DGP-SDP-FR-0M6RA	2.40E-03	1.28E-03	2.64E-05	1.00E+00	1.53E+00	Diesel generator 674-M-6 attached cooling pump fails to run
DGP-SDP-FR-M6FOP	2.40E-03	1.28E-03	2.64E-05	1.00E+00	1.53E+00	Diesel generator 674-M-6 engine-driven fuel oil pump fails to run
FIE-F4	3.71E-03	1.19E-03	1.58E-05	1.00E+00	1.32E+00	Fire initiating event: pump motor room
WVS-AOV-CC-GB232	6.30E-04	1.19E-03	9.29E-05	1.00E+00	2.88E+00	Vessel vent valve GB-23-2 fails to open
WVS-SOV-CC-V2310	5.00E-04	1.16E-03	1.15E-04	1.00E+00	3.33E+00	Solenoid valve SV-23-10 controlling air to WVS valve GB-23-1 fails to open
PCS-MDP-FS-00M11	2.40E-03	1.12E-03	2.31E-05	1.00E+00	1.47E+00	Emergency PCS pump M-11 fails to start
RIL	5.00E-08	1.01E-03	1.00E+00	1.00E+00	2.03E+04	Initiating event: large reactivity insertion
CAN	4.20E-07	9.53E-04	1.09E-01	1.00E+00	2.22E+03	Initiating event: storage canal draining
WVS-SOV-CC-V2311	5.00E-04	9.20E-04	9.08E-05	1.00E+00	2.84E+00	Solenoid valve SV-23-11 controlling air to WVS valve GB-23-2 fails to open
AHT	1.10E-02	7.84E-04	3.52E-06	1.00E+00	1.07E+00	Initiating event: loss of coolant temperature control
HTD	8.00E-02	7.84E-04	4.84E-07	1.00E+00	1.01E+00	Loop operator diagnosis and mitigation
FIE-V1	1.98E-03	7.78E-04	1.94E-05	1.00E+00	1.39E+00	Fire initiating event: nozzle trench area

Table 21. (continued)

Event Name	Probability	F-V*	Birn*	RRR ^c	RAR ^d	Description
GF-V1	5.00E-01	7.78E-04	7.67E-08	1.00E+00	1.00E+00	Fire geometric factor: nozzle trench
NS-V1	1.00E+00	7.78E-04	3.84E-08	1.00E+00	1.00E+00	Nonsuppression probability
SF-V1	5.00E-01	7.78E-04	7.67E-08	1.00E+00	1.00E+00	Fire severity: nozzle trench
CFA	8.00E-02	6.81E-04	4.20E-07	1.00E+00	1.01E+00	Operator fails to initiate firewater addition to canal
GF-F4-2	8.00E-02	6.61E-04	4.08E-07	1.00E+00	1.01E+00	Fire geometric factor: pump motor room factor 2
NS-F4-2	1.00E+00	6.61E-04	3.26E-08	1.00E+00	1.00E+00	Nonsuppression probability
SF-F4-2	1.00E+00	6.61E-04	3.26E-08	1.00E+00	1.00E+00	Fire severity: pump motor room factor 2
PVG	1.00E-01	6.08E-04	3.00E-07	1.00E+00	1.01E+00	No gross undetected perch
RPF	3.00E-07	6.08E-04	1.00E-01	1.00E+00	2.03E+03	Initiating event: very large reactivity insertion
V6M7A	7.50E-03	6.04E-04	3.97E-06	1.00E+00	1.08E+00	Flood initiator: leakage of sprinkler system in M-7 main reactor floor
V6F2BR	7.00E-05	5.61E-04	3.95E-04	1.00E+00	9.01E+00	Flood initiator: switchgear room M. wall UCV rupture
FIE-7AB	6.82E-03	5.26E-04	3.81E-06	1.00E+00	1.08E+00	Fire initiating event: 7A and 7B hall
GF-F7	5.00E-01	5.26E-04	5.19E-08	1.00E+00	1.00E+00	Fire geometric factor: east first basement hall
NS-F7	1.00E+00	5.26E-04	2.60E-08	1.00E+00	1.00E+00	Nonsuppression probability
SF-F7	1.00E-01	5.26E-04	2.60E-07	1.00E+00	1.00E+00	Fire severity: East first basement hall
DGP-ICC-F0-00674	1.00E-03	5.19E-04	2.56E-05	1.00E+00	1.52E+00	Diesel air supply damper ICC failure to operate
DGP-ICC-F0-074M8	1.00E-03	5.19E-04	2.56E-05	1.00E+00	1.52E+00	Diesel generator 674-M-6 cooling system fan ICC failure
DGP-ICC-F0-0FTM6	1.00E-03	5.19E-04	2.56E-05	1.00E+00	1.52E+00	Diesel generator 674-M-6 fuel oil motor-driven transfer pump ICC failure
DGP-ICC-F0-0M6CF	1.00E-03	5.19E-04	2.56E-05	1.00E+00	1.52E+00	Diesel generator 674-M-6 fuel oil injector cooling fan ICC failure
VDF-SYS-XF-VDP1M	3.72E-02	4.70E-04	6.23E-07	1.00E+00	1.01E+00	Operator failure to open manual valve GT-39 (VDP1 action)
RCD	2.50E-01	4.59E-04	9.05E-08	1.00E+00	1.00E+00	Reactor operator stops reactivity compensation
TUB	1.00E-01	4.59E-04	2.26E-07	1.00E+00	1.00E+00	AHTL gas envelope tube holds
PCS-ICC-F0-00M11	1.00E-03	4.58E-04	2.26E-05	1.00E+00	1.46E+00	Emergency PCS pump M-11 ICC failure
V6F2CL	6.20E-04	4.27E-04	3.40E-05	1.00E+00	1.69E+00	Flood initiator: switchgear room UCV leakage above UPS
DGP-FAN-FR-074M8	7.20E-04	3.69E-04	2.53E-05	1.00E+00	1.51E+00	Diesel generator 674-M-6 radiator M-8 fan fails to run
DGP-FAN-FR-0M6F0	7.20E-04	3.69E-04	2.53E-05	1.00E+00	1.51E+00	Diesel generator 674-M-6 injector cooling fuel oil cooler fan fails to run
DGP-MDP-FR-0FTM6	7.20E-04	3.69E-04	2.53E-05	1.00E+00	1.51E+00	Diesel generator 674-M-6 fuel oil pump for filling daytank 674-M-9 fails
PCS-MDP-FR-00M11	7.20E-04	3.29E-04	2.26E-05	1.00E+00	1.46E+00	Emergency PCS pump M-11 fails to continue to run
PCS-MDP-CR-01011	7.20E-05	3.25E-04	2.23E-04	1.00E+00	5.52E+00	Common cause failures of emergency PCS pump trains to run
DGP-SYS-XF-0LOAP	8.00E-02	3.24E-04	2.00E-07	1.00E+00	1.00E+00	Operator fails to detect and correct loss of air pressure in any tank
DGP-DGN-CS-DIESL	6.80E-03	3.17E-04	2.30E-06	1.00E+00	1.05E+00	Common cause failure of diesels 670-M-6 and 670-M-43 to start
GF-F2-6	7.00E-02	3.09E-04	2.18E-07	1.00E+00	1.00E+00	Fire geometric factor: switchgear room S.E.
NS-F2-6	1.00E+00	3.09E-04	1.53E-08	1.00E+00	1.00E+00	Nonsuppression probability
SF-F2-6	1.00E-01	3.09E-04	1.53E-07	1.00E+00	1.00E+00	Fire severity: switchgear room S.E.
GF-F2-8	9.00E-02	3.08E-04	1.69E-07	1.00E+00	1.00E+00	Fire geometric factor: switchgear room M.E. F2-8
GF-F2-9	9.00E-02	3.08E-04	1.69E-07	1.00E+00	1.00E+00	Fire geometric factor: switchgear room N.E. F2-9
NS-F2-8	1.00E+00	3.08E-04	1.52E-08	1.00E+00	1.00E+00	Nonsuppression probability
NS-F2-9	1.00E+00	3.08E-04	1.52E-08	1.00E+00	1.00E+00	Nonsuppression probability
SF-F2-8	1.00E+00	3.08E-04	1.52E-08	1.00E+00	1.00E+00	Fire severity: switchgear room M.E. F2-8
SF-F2-9	1.00E+00	3.08E-04	1.52E-08	1.00E+00	1.00E+00	Fire severity: switchgear room N.E. F2-9

Table 21. (continued)

Event Name	Probability	F-V ^a	Birn ^b	RRR ^c	RAR ^d	Description
GF-F4-3	3.50E-01	2.89E-04	4.08E-08	1.00E+00	1.00E+00	Fire geometric factor: pump motor room factor 3
NS-F4-3	1.00E+00	2.89E-04	1.43E-08	1.00E+00	1.00E+00	Nonsuppression probability
SF-F4-3	1.00E-01	2.89E-04	1.43E-07	1.00E+00	1.00E+00	Fire severity: pump motor room factor 3
CLD	3.20E-02	2.72E-04	4.20E-07	1.00E+00	1.01E+00	Operator fails to detect canal drainage
DGP-BKR-00-06052	5.00E-04	2.53E-04	2.49E-05	1.00E+00	1.50E+00	Diesel generator 674-M-6 output breaker in building 674 fails to close
DGP-BKR-00-60902	5.00E-04	2.53E-04	2.49E-05	1.00E+00	1.50E+00	Breaker 52-2 between transformer 4160/2.4/1.39kV transformer and Building 609
DGP-SOV-CC-01461	5.00E-04	2.53E-04	2.49E-05	1.00E+00	1.50E+00	Diesel generator 674-M-6 starting solenoid valve 50-14-61 fails to open
GF-F4-1	8.00E-02	2.38E-04	1.47E-07	1.00E+00	1.00E+00	Fire geometric factor: pump motor room factor 1
NS-F4-1	1.00E+00	2.38E-04	1.17E-08	1.00E+00	1.00E+00	Nonsuppression probability
SF-F4-1	1.00E-01	2.38E-04	1.17E-07	1.00E+00	1.00E+00	Fire severity: pump motor room factor 1
LAC	1.10E-03	2.35E-04	1.05E-05	1.00E+00	1.21E+00	Initiating event: loss of all ac power
PCS-MDP-XF-PFR01	8.50E-02	2.33E-04	1.35E-07	1.00E+00	1.00E+00	Operator fails to restart one of the PCS pumps when both emergency pumps fail
DCP-BAC-ST-OE015	2.90E-04	2.00E-04	3.41E-05	1.00E+00	1.69E+00	480V ac ciesel/commercial bus 670-E-15 short/open circuit
DGP-MDC-FS-00M65	5.00E-03	1.97E-04	1.94E-06	1.00E+00	1.04E+00	High-pressure air compressor 670-M-65 fails to start
DGP-MDC-CS-62732	3.00E-04	1.46E-04	2.40E-05	1.00E+00	1.49E+00	Common cause failure of diesel fuel oil transfer pumps 627-3 & -2 to start
DGP-RLY-F0-00674	3.00E-04	1.46E-04	2.40E-05	1.00E+00	1.49E+00	Diesel generator 674-M-6 run relay fails to operate
PPS-LOG-F0-CLLCA	3.00E-04	1.34E-04	2.20E-05	1.00E+00	1.45E+00	Loop scram contact-to logic level controller A fails to transmit a scram signal
PPS-LOG-F0-CLLCC	3.00E-04	1.34E-04	2.20E-05	1.00E+00	1.45E+00	Loop scram contact-to logic level controller C fails to transmit a scram signal
PPS-RLY-F0-08U51	3.00E-04	1.34E-04	2.20E-05	1.00E+00	1.45E+00	Loop power reduction bus 1 scram relay fails to operate
DGP-MDP-FS-06273	3.00E-03	1.19E-04	1.96E-06	1.00E+00	1.04E+00	Diesel fuel oil transfer pump 627-3 fails to start
PCS-MDP-FR-00M10	7.20E-04	1.15E-04	7.88E-06	1.00E+00	1.16E+00	Emergency PCS pump M-10 fails to continue to run
IAS-MDC-XF-0M678	2.90E-03	1.11E-04	1.89E-06	1.00E+00	1.04E+00	Operator fails to start any compressor
DGP-DGN-CR-DIESL	2.10E-04	1.01E-04	2.36E-05	1.00E+00	1.48E+00	Common cause failure of diesels 670-M-6, 670-M-42, and 670-M-43 to run
MKP-SYS-XF-CENXF	3.70E-02	9.52E-05	1.27E-07	1.00E+00	1.00E+00	Operator failure to restore makeup flow from the pressurizing system
DGP-MDC-FR-00M65	2.40E-03	9.18E-05	1.89E-06	1.00E+00	1.04E+00	High-pressure air compressor 670-M-65 fails to run
FIS-DDP-FR-06331	1.13E-01	8.27E-05	3.61E-08	1.00E+00	1.00E+00	Diesel firewater pump 633-1 fails to continue to run
FIS-XVM-XF-00FW3	5.00E-02	7.71E-05	7.61E-08	1.00E+00	1.00E+00	Firewater pump 619-12 discharge valve FW-3 fails due to failure to restore
DGP-BKR-CO-0E326	1.60E-04	7.47E-05	2.30E-05	1.00E+00	1.47E+00	Breaker 52-26 between diesel M-6 and Building 609 2.4kV bus
FIS-DDP-FR-61910	1.13E-01	6.84E-05	2.98E-08	1.00E+00	1.00E+00	Diesel firewater pump 619-10 fails to continue to run
FIS-DDP-CR-FWPMF	1.20E-02	6.22E-05	2.56E-07	1.00E+00	1.01E+00	Common cause failure of firewater pumps 619-10 & 633-1 to run
FIS-CKV-CC-00L12	5.00E-05	5.61E-05	5.54E-05	1.00E+00	2.12E+00	Raw water check valve L-12-CV fails to open
FIS-CKV-CC-00L13	5.00E-05	5.61E-05	5.54E-05	1.00E+00	2.12E+00	Raw water check valve L-13-CV fails to open
DGP-STR-PG-00M69	1.20E-04	5.45E-05	2.24E-05	1.00E+00	1.45E+00	Diesel fuel oil from tank farm to tank ATR-776 filter fails due to plugging
DGP-STR-PG-02712	1.20E-04	5.45E-05	2.24E-05	1.00E+00	1.45E+00	Diesel fuel oil filter 627-12 in diesel fuel oil system fails due to plugging
RLP	1.00E-01	5.35E-05	2.64E-08	1.00E+00	1.00E+00	Initiating event: lobe power imbalance
MKP-SYS-XF-HPTRX	8.30E-03	5.22E-05	3.10E-07	1.00E+00	1.01E+00	Operator failure to restore makeup flow from the pressurizing system
LDM-CKV-00-CVB15	1.00E-03	4.36E-05	2.15E-06	1.00E+00	1.04E+00	Emergency pumps combined suction check valve CV-B-15 fails to close
DCP-BKR-CO-0M6FC	1.00E-04	4.35E-05	2.15E-05	1.00E+00	1.44E+00	Breaker on 674-E-3 feeding fuel oil cooling fan spuriously opens
DCP-BKR-CO-0M6FP	1.00E-04	4.35E-05	2.15E-05	1.00E+00	1.44E+00	Breaker on 674-E-3 feeding fuel oil transfer pump spuriously opens
BBP-BKR-CO-00C82	1.00E-04	4.02E-05	1.99E-05	1.00E+00	1.40E+00	250VDC 670-E-23 breaker CB2 feeding emergency pump 670-M-11 spuriously opens
DGP-RLY-F0-0027X	3.00E-04	3.92E-05	6.44E-06	1.00E+00	1.13E+00	Diesel bus 27X (undervoltage) relay failure to operate

Table 21. (continued)

Event Name	Probability	F-V*	Birm ^b	RRR ^c	RAR ^d	Description
RSS-CRM-EV-0AUI0	1.00E+00	3.83E-05	1.89E-09	1.00E+00	1.00E+00	Probability that the regulating rod is in automatic
DGP-AOV-CC-0LCV5	1.00E-03	3.30E-05	1.63E-06	1.00E+00	1.03E+00	ATR-776 daytank level control valve LCV-5 fails to open
DGP-ICC-F0-000M6	1.00E-03	3.30E-05	1.63E-06	1.00E+00	1.03E+00	Diesel Generator 674-M-6 control circuitry fails to function
DGP-ICC-F0-000M5	1.00E-03	3.30E-05	1.63E-06	1.00E+00	1.03E+00	High-pressure air compressor 670-M-65 ICC failure
DGP-ICC-F0-06273	1.00E-03	3.30E-05	1.63E-06	1.00E+00	1.03E+00	Diesel fuel oil pump 627-3 ICC failure
DGP-ICC-F0-0LCV5	1.00E-03	3.30E-05	1.63E-06	1.00E+00	1.03E+00	Diesel generator 670-M-42 fuel oil supply level control valve LCV-5 ICC failure
FIS-TNK-FC-ORV01	7.20E-05	3.27E-05	2.24E-05	1.00E+00	1.45E+00	Overhead storage tank RW01 low level indicator fails
FIS-LOG-CF-FWACT	3.00E-05	3.08E-05	5.06E-05	1.00E+00	2.03E+00	Common cause failure of 4 logic modules to function
FIS-XSW-XF-FIACI	1.87E-03	3.08E-05	8.12E-07	1.00E+00	1.02E+00	Operator fails to attempt manual firewater injection
DGP-MDP-CR-62732	7.20E-05	2.97E-05	2.03E-05	1.00E+00	1.41E+00	Common cause failure of diesel fuel oil transfer pumps 627-3 & -2 to run
DGP-MDP-FR-06273	7.20E-04	2.37E-05	1.63E-06	1.00E+00	1.03E+00	Diesel fuel oil transfer pump 627-3 fails to run
DGP-SDP-FR-02F05	2.40E-03	2.29E-05	4.72E-07	1.00E+00	1.01E+00	Diesel generator 670-M-42 engine-driven fuel oil pump fails to run
DGP-SDP-FR-02JWS	2.40E-03	2.29E-05	4.72E-07	1.00E+00	1.01E+00	Diesel generator 670-M-42 engine-driven jacket water circ pump fails to run
DGP-SDP-FR-02L05	2.40E-03	2.29E-05	4.72E-07	1.00E+00	1.01E+00	Diesel generator 670-M-42 engine-driven lube oil pump fails to run
LDW-XVM-CC-GTD39	5.00E-04	2.18E-05	2.15E-06	1.00E+00	1.04E+00	Upper vessel drain valve GT-D-39 used for LDW purge fails to open
LDW-XVM-CC-GTE20	5.00E-04	2.18E-05	2.15E-06	1.00E+00	1.04E+00	LDW purge valve GT-E-20 fails to open
FIS-AOV-CA-P7A7B	7.00E-05	2.01E-05	1.41E-05	1.00E+00	1.29E+00	Common cause failure of valve paths LCV-7A and LCV-7B to open
LDW-CKV-CC-CVB15	5.00E-05	2.01E-05	1.99E-05	1.00E+00	1.40E+00	Emergency pumps combined suction check valve CV-B-15 fails to open
PCS-CKV-CC-00A20	5.00E-05	2.01E-05	1.99E-05	1.00E+00	1.40E+00	Emergency PCS pump M-11 discharge check valve A-20 fails to open
DGP-CKV-CC-01224	5.00E-05	1.88E-05	1.85E-05	1.00E+00	1.38E+00	Diesel generator 674-M-6 check valve CK-12-24 on supply from daytank ATR-776
PPS-ACT-CF-SCRO0	3.00E-05	1.82E-05	3.00E-05	1.00E+00	1.61E+00	Common cause failure of manual scram signal trains
PPS-LOG-CF-MSCL	3.00E-05	1.82E-05	3.00E-05	1.00E+00	1.61E+00	Common cause failures of manual scram 2:3 logics
BBP-BCH-XF-0E105	5.87E-02	1.68E-05	1.41E-08	1.00E+00	1.00E+00	Operator fails to align battery charger 670-E-105 to diesel bus 670-E-9
DGP-DGN-FR-00M42	2.09E-03	1.61E-05	3.82E-07	1.00E+00	1.01E+00	Diesel generator 670-M-42 fails to run
RSH	1.00E-01	1.52E-05	7.49E-09	1.00E+00	1.00E+00	Initiating event: reactivity insertion (slow ramp)
FIS-ORF-PG-01108	3.00E-04	1.47E-05	2.42E-06	1.00E+00	1.05E+00	Flow element FE-1-108 upstream of LCV-1-615 and -616 fails due to plugging
FIS-XVM-XT-00FW6	5.00E-02	1.43E-05	1.41E-08	1.00E+00	1.00E+00	Firewater pump 619-10 discharge FW-6 fails due to failure to restore after TM
PPS-ACT-CF-LFS00	3.00E-05	1.34E-05	2.20E-05	1.00E+00	1.45E+00	Common cause failures of loop low flow scram channel
PPS-LOG-CF-L20F3	3.00E-05	1.34E-05	2.20E-05	1.00E+00	1.45E+00	Common cause failure of loop 2:3 logic modules
DGP-BKR-00-00025	5.00E-04	1.31E-05	1.30E-06	1.00E+00	1.03E+00	Diesel generator 674-M-6 feeder breaker 25 to 670-E-3 fails to close
DGP-SOV-F0-0LFF5	5.00E-04	1.31E-05	1.30E-06	1.00E+00	1.03E+00	Solenoid valve LFV-5 fails to open inlet valve LCV-5 to the fuel oil daytank
LLC	4.50E-06	1.28E-05	1.40E-04	1.00E+00	3.84E+00	Initiating event: large LOCA
FIS-MDP-TM-61912	1.00E-02	1.26E-05	6.20E-08	1.00E+00	1.00E+00	Firewater pump 619-12 unavailable due to maintenance
PPS-CON-XT-ASCRH	2.70E-05	1.20E-05	2.20E-05	1.00E+00	1.45E+00	Loop scram channel A test block fails or is removed
PPS-CON-XT-CSCRM	2.70E-05	1.20E-05	2.20E-05	1.00E+00	1.45E+00	Loop scram channel C test block fails or is removed
RSS-TCH-FC-RRTAC	3.60E-04	1.09E-05	1.50E-06	1.00E+00	1.03E+00	Regulating rod tachometer failure
DCP-BKR-CO-015B4	1.00E-04	1.05E-05	5.20E-06	1.00E+00	1.11E+00	Breaker B4 on bus 670-E-15 for emergency PCS pump M-10 spuriously opens
BBP-TSW-ST-0MT51	2.40E-05	9.23E-06	1.90E-05	1.00E+00	1.38E+00	MTS1 switchgear control power bus transfer switch short/open circuit
RSS-RLY-F0-0RSB1	3.00E-04	9.12E-06	1.50E-06	1.00E+00	1.03E+00	Setback relay RSB-1 fails to transmit setback signal to the regulating rod
RSS-RLY-F0-Q1171	3.00E-04	9.12E-06	1.50E-06	1.00E+00	1.03E+00	Quadrant 1 water power relay R-17-1 fails to close its contacts
RSS-RLY-F0-RS834	3.00E-04	9.12E-06	1.50E-06	1.00E+00	1.03E+00	Contact 2 on fast setback relay RSB-3 or RSB-4 fails open - fails to operate

Table 21. (continued)

Event Name	Probability	F-V ^a	Birm ^b	RRR ^c	RAR ^d	Description
CDP-RLY-FQ-0093C	3.00E-04	6.57E-06	1.08E-06	1.00E+00	1.02E+00	Starting relay for diesel fuel oil transfer pump 627-3 fails to operate
CDP-TFM-ST-009T1	2.40E-05	6.57E-06	1.35E-05	1.00E+00	1.27E+00	Transformer 609-I-1 short/open circuit
DCP-TFM-ST-074E2	2.40E-05	6.57E-06	1.35E-05	1.00E+00	1.27E+00	480-208/120VAC transformer 674-E-2 short/open circuit
DGP-TFM-ST-070E8	2.40E-05	6.57E-06	1.35E-05	1.00E+00	1.27E+00	DXF70E8D 4160/480V transformer 670-E-8 short/open circuit
DGP-TFM-ST-60902	2.40E-05	6.57E-06	1.35E-05	1.00E+00	1.27E+00	Transformer 4160/2.4/1.39kV between diesel M-6 and building 609 2.4kV bus
FIS-XVM-XT-GT184	2.70E-05	6.57E-06	1.20E-05	1.00E+00	1.24E+00	Lower FIS manual valve GT-1-84 upstream of LCV-7A and -7B fails due to failure
IAS-MDC-CS-0M678	2.50E-04	5.47E-06	1.08E-06	1.00E+00	1.02E+00	Common cause failure Of M-6,7,8 to start
PCS-CKV-CC-00A17	5.00E-05	5.27E-06	5.20E-06	1.00E+00	1.11E+00	Emergency PCS pump M-10 discharge check A-17 fails to open
IAS-PIP-RU-0M678	2.40E-04	5.25E-06	1.08E-06	1.00E+00	1.02E+00	Instrument air supply line rupture
CDP-TFM-ST-074E2	2.40E-05	4.33E-06	8.90E-06	1.00E+00	1.18E+00	ATR incoming line transformer 774-E-2 short/open circuit
CDP-TSW-ST-074E4	2.40E-05	4.33E-06	8.90E-06	1.00E+00	1.18E+00	ATR transformer 774-E-2 secondary side disconnect 774-E-4 short/open circuit
DGP-BKR-CO-00025	1.60E-04	3.50E-06	1.08E-06	1.00E+00	1.02E+00	Diesel generator 674-M-6 feeder breaker 25 to 670-E-3 spuriously opens
CDP-REC-FC-0E116	2.40E-04	3.21E-06	6.60E-07	1.00E+00	1.01E+00	Utility UPS 670-E-116 rectifier fails to function
MKP-SYS-XF-MCRXF	8.00E-02	3.15E-06	1.94E-09	1.00E+00	1.00E+00	Operator failure to restore makeup flow from the pressurizing system
MKP-ADV-CO-01C3A	7.20E-05	3.14E-06	2.15E-06	1.00E+00	1.04E+00	Degassing tank level control valve LCV-3A spuriously opens
FIS-ADV-CA-P1516	8.60E-05	3.05E-06	1.75E-06	1.00E+00	1.04E+00	Common cause failure of valve paths LCV-CC-1-615 and -616 to open
IAS-MDC-CR-0M678	1.20E-04	2.63E-06	1.08E-06	1.00E+00	1.02E+00	Common cause failure Of M-6,7,8 To run
DGP-MDP-FS-05272	3.00E-03	2.46E-06	4.05E-08	1.00E+00	1.00E+00	Diesel fuel oil standby transfer pump 627-2 fails to start
LDW-MDP-CR-PUMPS	3.60E-05	2.35E-06	3.22E-06	1.00E+00	1.07E+00	Common cause loss of LDW pumps 18, 19, 20, & 21 to run
CDP-BKR-CO-0093C	1.00E-04	2.19E-06	1.08E-06	1.00E+00	1.02E+00	Breaker 3C on 480V AC 609-MCC-1 feeding diesel oil pump 627-3 spuriously opens
DGP-BKR-CO-00M65	1.00E-04	2.19E-06	1.08E-06	1.00E+00	1.02E+00	Breaker A3 feeding air compressor 670-M-65 spuriously opens
LDW-CKV-CC-CV819	5.00E-05	2.18E-06	2.15E-06	1.00E+00	1.04E+00	LDW flush check valve CV-B-19 fails to open
MKP-ADV-CC-LCV3A	7.60E-04	2.04E-06	1.33E-07	1.00E+00	1.00E+00	Air-operated level control valve LCV-3A fails to open

a. The F-V (Fussell-Vesely) measure is the difference between the fuel damage frequency with the event at its nominal value and the fuel damage frequency with the event set to zero, normalized by the nominal fuel damage frequency. It indicates the fraction of the total fuel damage frequency that involves the event.

b. The Birnbaum measure is the difference between the fuel damage frequency with the event set to 1.0 and the fuel damage frequency with the event set to zero. It indicates the change in fuel damage frequency with respect to a change in the event probability.

c. The RRR (risk reduction ratio) measure is the ratio of the fuel damage frequency with the event at its nominal value to the fuel damage frequency with the event set to zero. It indicates how much the results would be reduced if the event were totally reliable.

d. The RAR (risk achievement ratio) measure is the ratio of the fuel damage frequency with the event set to one and the fuel damage frequency with the event at its nominal value. It indicates how much the results would go up if the event were totally unreliable (i.e., guaranteed to fail).

the PRA models accurately reflect the plant as of May 1989. Three types of formal reviews were used during the PRA: internal preliminary (all tasks as they were performed), internal quality assurance (as tasks were completed), and external peer review (completed tasks and documentation). Additionally, when the second PRA team took over the project, a very thorough informal review of models and methodologies was conducted. Finally, the PRA final report structure and contents were developed from experience gained in writing several similar reports. It is believed that the structure and detail make the ATR PRA comprehensible, scrutable, and reproducible.

2. RISK SUMMARY

2.1 RISK PROFILE

The total Level 1 fuel damage frequency of $4.9\text{E-}5/\text{yr}$ is the sum of the internal event mean frequency of $2.4\text{E-}5/\text{yr}$ and the external event mean frequency of $2.5\text{E-}5/\text{yr}$. Table 22 provides estimated mean fuel damage frequencies for both internal and external event initiators and the percentage contribution to the Level 1 PRA result. The internal events, external events, and Level 1 events fuel damage contributions are depicted in Figures 2, 3, and 4.

2.2 RISK INSIGHTS

The low total fuel damage frequency with no extremely dominant sequences indicates that risk reduction efforts were successful. Emphasis should now be placed on maintenance of this low risk profile.

The principal contributors to the reduction in fuel damage risk are listed in Table 23 and depicted in Figure 5. The risk reduction contributions are for the changes in mean fuel damage frequency estimated in Revision 0 of the Level 1 ATR PRA for internal events¹⁶ and from the results of the external events analysis before implementation of risk reduction upgrades. The external events risk without upgrades were analyzed in Reference 17 in the sensitivity cases in Sections 3.10.3 and 3.10.4. The mean fuel damage frequency from these prior analyses before risk reduction actions was

Internal events	$1.8\text{E-}4/\text{yr}$
External events	$8.7\text{E-}4/\text{yr}$

Total Level 1	$1.05\text{E-}3/\text{yr}$
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Table 22. Level 1 fuel damage frequencies by initiator

Initiating Event	Mean Fuel Damage Frequency/Yr	Percent Contribution
Internal Events		
Loss of commercial power (LCP)	9.5E-06	19.3
Loss of diesel power (LDP)	6.7E-06	13.6
Loop reactivity ramp (high range) (RLH)	3.7E-06	7.4
Small LOCA (S12)	1.5E-06	3.1
Reactor shutdown (RSD)	6.7E-07	1.3
Direct-damage large LOCA (DLC)	5.0E-07	1.0
Loss of diesel-commercial power (LDC)	3.6E-07	0.7
Fast reactivity ramp (high range) (RFH)	3.3E-07	0.7
Low PCS pressure (LPP)	2.6E-07	0.5
Low PCS flow (LPF)	2.5E-07	0.5
High coolant pressure (HCP)	1.7E-07	0.3
Others	<1.0E-07	0.5
Internal Event Total	2.4E-05	49.0
External Events		
Seismic	1.9E-05	38.1
Volcanic activity	2.8E-06	5.7
Extreme wind	1.4E-06	2.8
Internal flooding	1.1E-06	2.2
Internal fire	1.0E-06	2.0
Others	<1.0E-07	0.2
External Event Total	2.5E-05	51.0
Level 1 Total	4.9E-05	100.0

Figure 2. Internal events fuel damage contributors by initiator.

Figure 3. External events fuel damage contributors by initiator.

Figure 4. Level 1 fuel damage contributors by initiator.

As seen in Table 23 and Figure 5, the risk reduction actions being taken to eliminate the threat to important electrical switchgear from diesel pit fire or flood events, which can result in significant flooding of the diesel pit and propagation of the flood to the switchgear below the diesel pit, are the most important risk reduction actions. These actions are as follows:

- Relocate the utility battery-backed power system switchgear and panels, including the battery charger, away from the diesel pit and separated from other important switchgear.
- Seal the penetrations from the diesel pit to the switchgear room sufficient to prevent significant water propagation to the switchgear room from a flooded diesel pit, and provide a diesel pit draining capability.
- Protect the vessel vent system battery-backed power panels in the diesel generator area from fire sprinkler actuation.

Other important risk reduction actions that are being taken and which have contributed to the significant reduction in the ATR fuel damage frequency estimate are as follows^a:

- Provide a lateral seismic restraint for the PCS surge tank.
- Provide seismic restraints for the backup nitrogen bottles for the vessel vent system.
- Replace the plant protection system (PPS) backup power batteries that provide backup power for the vessel vent system, with more seismically robust batteries.

a. Letter from E. C. Anderson, Sr., EG&G Idaho, to D. E. Shadley, DOE-ID, "Transmittal of Recommendations for Advanced Test Reactor (ATR) Risk-Reduction Upgrades," ECA-12-91, February 28, 1991.

Table 23. Level 1 fuel damage risk reduction contributors

Risk Reduction Contributor	Fuel Damage Frequency (FDF) (yr ⁻¹)		Percent Reduction in Total FDF
	Without improvements	With improvements	
Internal Events			
ATR-specific direct-damage large LOCA frequency estimate	5.0E-5	5.0E-7	4.7
ATR-specific direct-damage experiment loop rupture frequency estimate	5.0E-5	7.3E-8	4.8
ATWS analyses demonstrating time for manual scram	4.2E-5	1.3E-5	2.8
Loss of flow procedures, human error and system modeling improvements	2.3E-5	2.9E-6	1.9
Loss of diesel power frequency estimate	1.2E-5	6.7E-6	0.5
Internal Events Total	1.8E-4	2.4E-5	14.8
External Events			
Facility upgrades for diesel pit fire and flood events	7.4E-4	2.7E-5	67.8
Surge tank seismic restraint	4.9E-5	<1.0E-7	4.7
Loss of flow procedures, human error and modeling improvements	3.85E-5	5.8E-6	3.1
Vessel vent N ₂ bottles seismic restraint	4.05E-5	1.8E-5	2.1
Risk estimate for volcanism	1.7E-5	2.8E-6	1.4
External Events Total	8.7E-4	2.5E-5	80.7

Figure 5. Level 1 fuel damage risk reduction contributors.

- Upgrade the seismic restraints for the starting batteries and diesel fuel day tanks for the diesel-driven fire pumps.
- Provide and/or upgrade the anchorage sufficient to withstand a design basis earthquake for the following power panels: 670-E-3, 670-E-9, 674-E-1, 674-E-3, 608 commercial/diesel MCC-1, 609 commercial diesel MCC-1, 609 commercial/diesel MCC-2, 2.4 kV commercial/diesel switchgear 609-3, and transformer 674-E-2.

The latter two recommendations provide for one percent or less reduction in the fuel damage frequency each, but they are not expensive upgrades and are advisable for ensuring electrical power availability and firewater injection system availability for seismic events. The third recommendation for the replacement of the PPS batteries will also provide for a one percent or less reduction in the fuel damage frequency, but it will be accomplished during the planned replacement near end-of-life for these batteries in about 1994.

The results of Revision 1 of the Level 1 ATR PRA assume that the above risk reduction upgrades have been accomplished.

The following risk maintenance insights from the Revision 1 Level 1 PRA are important for maintaining the current low estimate of the ATR fuel damage frequency.

The loss of commercial power, loss of diesel power, and experiment loop scram initiators are important contributors to the ATR risk profile. Efforts must be continued to limit these occurrences, as these initiators account for 82% of the internal events predicted fuel damage frequency. In addition, the loss of commercial power is also a contributor to risk as a failure combined with other initiating events; loss of commercial power failures of this type affect 7.5% of the Level 1 result.

Equipment failures that are dominant contributors to risk result from the scram system, the M-6 diesel generator, the firewater injection system, and vessel vent system. Their performance and availability must be

maintained. Dependent failures of the scram system contribute 40% of the Level 1 result. Failure of the M-6 diesel generator contributes more than 11% of the Level 1 result. Dependent failure of the firewater injection system contributes more than 2.5% of the Level 1 result. Failure of the vessel vent system contributes more than 5% of the Level 1 result. The contributions to the Level 1 result are underestimated here because the individual equipment contributions to seismic risk are difficult to define and are not included.

Human actions involving operator failure to depressurize the PCS are important contributors to risk and are the dominate human failures. These operator failures (to diagnose and act) dominate even though detailed human reliability analyses provide very low failure probabilities for the operator failures. These failures contribute more than 5% of the Level 1 result. Again, the contribution to the Level 1 result is underestimated because the contribution to seismic risk has not been included.

3. RISK CONTRIBUTORS

This section provides a description of the contributors to risk by accident sequence, accident sequence type, and basic event (including equipment failures and operator actions). The description of accident sequences provides a brief overview of the dominant sequences. The description of contributors by accident sequence type provides an overview of the dominant types of equipment failures and operator actions for each sequence type as well as the dominant sequences contributing to a particular sequence type. The prioritized list of basic events provides a concise list of prominent equipment and human actions upon which to focus attention.

3.1 CONTRIBUTORS BY SEQUENCE TYPE

The dominant sequences were divided into seven sequence types:

- ATWS Anticipated transients with failure to reduce power by scram, reverse, or setback.
- LOF (depressurized) Loss of flow with PCS depressurized.
- LOF (pressurized) Loss of flow with the PCS pressurized.
- LOCA Loss-of-coolant accidents without inventory makeup.

- CANAL Canal drainage events.
- DIRECT DAMAGE Sequences for which no safety function could have prevented core damage (i.e., in some sequences of this type, reactor shutdown occurs and coolant flow and volume are maintained). Most sequences of this type result in moderate local core damage. This category includes direct damage LOCA events, large reactivity insertion events from experiment loop rupture, very large reactivity insertion events from a perched fuel element drop, and high temperature loop overheating events.
- LOSS OF INVENTORY CONTROL Sequences in which coolant recirculation may be provided, but no source of inventory makeup is available.

The results of the sequence quantification process for both internal and external events grouped by sequence type are presented in Table 24. A description of the dominant sequences within these sequence types is provided below. (Expanded descriptions can be obtained from References 1 and 18.)

ATWS events. ATWS events contribute 43% to the total estimated fuel damage frequency. This type of event can occur for any initiating event requiring a scram. Therefore, high frequency initiating events generally yield high frequency ATWS events. The failure probability estimations of these dependent failures may be conservative; however, it is difficult to justify lower failure probabilities. Recent ATWS accident analyses indicate that credit may be taken for alternate shutdown methods for many of the ATWS sequences, which would reduce the dominance of this category of event.

Table 24. Summary of Level 1 fuel damage sequences by type

Sequence	Percent of Sequence Type	Percent of Level 1	Description
ATWS Sequences			
		43.1%	There are 16 ATWS sequences >1.0E-8/yr. The sequences described below account for 89% of the ATWS sequences.
LDP9	31.7%	13.7%	Loss of diesel power causes low flow in all loops and a reactivity ramp. Scram systems fail resulting in an ATWS potentially causing core fuel damage.
LCP20	21.3%	9.2%	Loss of commercial power and random failures result in an ATWS potentially causing core fuel damage.
RLH10	17.0%	7.4%	A loop reactivity ramp from high range and random failure of safety rod insertion result in an ATWS potentially causing core fuel damage.
LCP17	14.2%	6.1%	Loss of commercial power and random failure of safety rod insertion result in an ATWS potentially causing core fuel damage.
EQE14	4.5%	1.9%	Seismic event results in failure of the scram system (ATWS) potentially causing core fuel damage.
LOSS OF FLOW (depressurized)			
		27.6%	There are 3 loss of flow (depressurized) sequences >1.0E-8/yr. These sequences are described below.
EQE6	96.2%	26.5%	Seismic event results in failure of power to coolant flow systems and low-pressure injection systems. This leads to core uncovery and core-wide fuel damage.
WWD6	3.3%	0.9%	Extreme wind event with subsequent failure of low-pressure injection systems leads to core uncovery and core-wide fuel damage. (Loss of commercial power is assumed due to extreme wind.)
RSD7	0.5%	0.1%	Normal or spurious plant shutdown with random failure of low-pressure injection systems leads to core uncovery and core-wide fuel damage.

Table 24. (continued)

Percent of Sequence Type		Percent of Level 1		Description
LOSS OF FLOW (pressurized)		24.7%		There are 25 loss of flow (pressurized) sequences $>1.0E-8/\text{yr}$. The sequences described below account for 91% of these loss of flow sequences.
EQE7	33.0%	8.2%		Seismic event causes the loss of coolant flow. Subsequent failures lead to core uncover and core-wide fuel damage.
VOLCAN	23.1%	5.7%		Basaltic volcanism causes lava flow that reaches the ATR reactor building. Core-wide fuel damage was assumed for this event.
LCP15	17.3%	4.3%		A loss of commercial power occurs without recovery. Random failures and subsequent failure to depressurize leads to core uncover and core-wide fuel damage.
WGN3A	6.3%	1.6%		Internal flooding caused by inadvertent actuation of wet pipe sprinkler system in the process control room and subsequent additional failures lead to core uncover and core-wide fuel damage.
WND7	5.8%	1.4%		Extreme wind event causes the loss of coolant flow. Subsequent failure to depressurize leads to core uncover and core-wide fuel damage.
RSD8	5.0%	1.2%		Normal or spurious plant shutdown occurs. Random power failures and failure to depressurize leads to core uncover and core-wide fuel damage.
LOCA		2.2%		There is one LOCA sequence $>1.0E-8/\text{yr}$. This sequence is described below.
S12-2	100%	2.2%		A small LOCA occurs and subsequent failure to provide coolant results in core uncover and core-wide fuel damage.

Table 24. (continued)

Sequence	Percent of Sequence Type	Percent of Level 1	Description
CANAL DRAIN			
		2.0%	There are 4 canal draining sequences $>1.0E-8/yr$. The sequences described below account for 99% of the canal draining frequency.
EQC4	83.6%	1.6%	Seismic event causes the loss of storage canal integrity that results in canal drainage. (No simultaneous reactor LOCA.) Seismic failure of both low-pressure injection systems leads to fuel uncover in the storage canal.
EQC5	11.5%	0.2%	Seismic event causes loss of storage canal integrity, resulting in drainage of the canal. Operator fails to recognize the loss of water from the canal and uncover of fuel leads to fuel damage. (Loss of commercial power is assumed due to seismic event.)
CAN2	3.6%	0.1%	Storage canal draining event with failure to provide makeup results in uncover of irradiated fuel stored in the canal and eventual fuel damage.
DIRECT DAMAGE			
		1.2%	There are 4 direct damage sequences $>1.0E-8/yr$. Each of these sequences is described below.
DLC1	82.9%	1.0%	A direct-damage large LOCA results in initial fuel damage even with successful scram and coolant injection.
RIL1	8.3%	0.1%	A direct-damage large reactivity insertion event results in initial fuel damage even with successful scram.
RPF2	5.0%	0.1%	Very large reactivity insertion results in initial fuel damage and potential reactor vessel failure.
AHT6	3.8%	0.1%	A high-temperature loop overheating event results in initial fuel damage with successful scram.
LOSS OF INVENTORY CONTROL			
		0.2%	There is one loss of inventory control sequence $>1.0E-8/yr$.
LPP3	100%	0.2%	A low PCS pressure event with successful scram occurs but random failure of low-pressure injection systems leads to core uncover and core-wide fuel damage.

Loss of flow (depressurized) events. These events contribute 27.6% to the total estimated fuel damage frequency. Seismic sequence EQE6 dominates this category; extreme wind sequence WND6 and normal/spurious shutdown sequence RSD7 are minor contributors. In both the seismic and extreme wind sequences, commercial power and diesels M-42 and M-43 are assumed to be lost.

Loss of flow (pressurized) events. These events contribute 24.7% to the total estimated fuel damage frequency. Seismic sequence EQE7 dominates this category; basaltic volcanism and loss of commercial power initiators also contribute. The basaltic volcanism frequency is considered to be a conservative value used pending the outcome of New Production Reactor (NPR) detailed analysis of volcanism at the INEL.

LOCAs. The small LOCA event contributes 2.2% to the total estimated fuel damage frequency.

Canal drainage events. These events contribute 2.0% to the total estimated fuel damage frequency. Seismic sequence EQC4 dominates this category; seismic sequence EQC5 and internal event sequence CAN2 also contribute.

Direct damage events. These events contribute 1.2% to the total estimated fuel damage frequency. The most dominant direct damage event is a large LOCA. Other events are a large reactivity insertion due to experiment loop rupture, a very large reactivity insertion due to a perched fuel element drop, and a high temperature loop overheating event with several mitigating system failures.

Loss of inventory control events. Low PCS pressure (LPP) is the only initiating event causing a loss of inventory control event with a frequency above $1.0E-10/\text{yr}$, and this event contributes 0.2% to the total estimated fuel damage frequency.

3.2 DOMINANT EQUIPMENT FAILURES AND HUMAN ERRORS

The impact of equipment or human improvements, tabulated based on the Fussell-Vesely risk importance measure, is shown in Table 25. Equipment failure events are separated into various categories: internal and external initiating events, dependent failures, equipment failures (that are single point failures) and test and maintenance outages. Human error events are separated into two categories: post-accident operator errors and pre-accident human errors. Out of the 1300 basic events representing various equipment failures, operator actions, and initiating events, this table provides the dominant 70 basic events rated as either high or moderate impact events.

Table 25. Dominant equipment failures and human errors

Event Name	Impact*	Description
Internal Initiating Events		
LCP	High	Initiating event: loss of commercial power
LDP	High	Initiating event: loss of diesel power
RLH	High	Initiating event: reactivity insertion (loop ramp)
S12	High	Initiating event: small LOCA
RSD	High	Initiating event: normal/spurious reactor shutdown
DLC	High	Initiating event: direct-damage LOCA
LDC	Moderate	Initiating event: loss of diesel-commercial bus
RFH	Moderate	Initiating event: reactivity insertion (fast ramp)
LPP	Moderate	Initiating event: low PCS pressure
LPF	Moderate	Initiating event: low PCS flow
HCP	Moderate	Initiating event: high PCS pressure
LIA	Moderate	Initiating event: loss of instrument air
RIL	Moderate	Initiating event: large reactivity insertion
External Initiating Events		
Seismic Activity	High	Seismic sequences EQE6, EQE7, EQE14, EQC4 and EQC5
Volcanic Activity	High	Scoping volcanic sequence
W6M3A	High	Flood initiator: Process control room inadvertent sprinkler actuation
Extreme Wind	High	Wind sequences WND7, WND6, WND8
FIE-F2	Moderate	Fire initiator: switchgear room
FIE-ATWS	Moderate	Fire initiator: failure to scram given a fire event
W7ALL	Moderate	Flood initiator: failure to scram given an internal flooding event
FIE-M3	Moderate	Fire initiator: process control room
FIE-F4	Moderate	Fire initiating event: pump motor room
Dependent Failures		
PPS-SYS-CF-SCROO	High	Common cause loss of scram signal
PPS-LOG-CF-RCCCS	High	Common cause failure of RCCCs to release
PPS-CRM-CF-3RODS	High	Common cause failure that prevents having at least three rods fully inserted
FIS-AOV-CA-004F4	High	Common cause loss of both FIS injection paths (all four injection valves fail to open)
VVS-AOV-CA-02312	High	Common cause failure of vessel vent valves GB-23-1 and GB-23-2 to open
Equipment Failures		
DGP-DGN-FR-000M6	High	Diesel generator 674-M-6 fails to run
CPR	High	Failure to recover commercial power
DGP-DGN-FS-000M6	High	Diesel generator 674-M-6 fails to start
VVS-REG-FC-0E14C	High	32-24-Vdc voltage regulator for panel E14C fails
DGP-SDP-FS-0M6RA	High	Diesel generator 674-M-6 attached cooling pump fails to start
DGP-SDP-FS-M6FOP	High	Diesel generator 674-M-6 engine-driven fuel oil pump fails to start
CDP-BAC-LP-STPWR	High	132KV ac site loop bus loss of power
VVS-REG-FC-0E14B	High	32VDC-to-24-VDC voltage regulator for panel E14B fails
DGP-ICC-FO-0M6RA	Moderate	Diesel generator 674-M-6 cooling system engine-driven pump instrument and control circuit (ICC) failure
DGP-ICC-FO-M6FOP	Moderate	Diesel generator 674-M-6 fuel oil supply engine-driven pump ICC failure
CDP-SYS-FC-STPWR	Moderate	Failure to recover site power within 4 hours
VVS-SYS-FC-NITR1	Moderate	Loss of nitrogen gas backup to valve GB-23-1
DGP-FAN-FS-074M8	Moderate	Diesel generator 674-M-6 radiator M-8 fan fails to start
DGP-FAN-FS-0M6FO	Moderate	Diesel generator 674-M-6 injector cooling fuel oil cooler fan fails to start
VVS-ICC-FO-GB231	Moderate	Actuation circuitry for valve GB-23-1 fails

Table 25. (continued)

Event Name	Impact ^a	Description
Equipment Failures (continued)		
VVS-SYS-FC-NITR2	Moderate	Loss of nitrogen gas backup to valve GB-23-2
VVS-ICC-FO-GB232	Moderate	Actuation circuitry for valve GB-23-2 fails
DGP-BDD-CC-00674	Moderate	Diesel generator 674-M-6 combustion air intake damper fails to open
DGP-MDP-FS-0FTM6	Moderate	Diesel generator 674-M-6 fuel oil pump for filling daytank 674-M-9 FTS
VVS-AOV-CC-GB231	Moderate	Vessel vent valve GB-23-1 fails to open
PCS-ORF-PG-0503A	Moderate	Emergency PCS pumps combined discharge flow element plugged
DGP-SDP-FR-0M6RA	Moderate	Diesel generator 674-M-6 attached cooling pump fails to run
DGP-SDP-FR-M6FOP	Moderate	Diesel generator 674-M-6 engine-driven fuel oil pump fails to run
VVS-AOV-CC-GB232	Moderate	Vessel vent valve GB-23-2 fails to open
VVS-SOV-CC-V2310	Moderate	Solenoid valve SV-23-10 controlling air to VVS valve GB-23-1 fails to open
PCS-MDP-FS-00M11	Moderate	Emergency PCS pump M-11 fails to start
Internal Fire Hazard Variables		
GF-F2-2	Moderate	Fire geometric factor: Switchgear room near bus 670-E-15
NS-F2-2	Moderate	Nonsuppression probability: Switchgear room near bus 670-E-15
SF-F2-2	Moderate	Fire severity: Switchgear room near bus 670-E-15
GF-CR	Moderate	Fire geometric factor: Process control room
NS-CR	Moderate	Nonsuppression probability: Process control room
SF-CR	Moderate	Fire severity: Process control room
GF-F2-5	Moderate	Fire geometric factor: Switchgear room near E-1 and E-2
NS-F2-5	Moderate	Nonsuppression probability: Switchgear room near E-1 and E-2
SF-F2-5	Moderate	Fire severity: Switchgear room near E-1 and E-2
Test and Maintenance Outages		
DGP-DGN-TM-000M6	Moderate	Diesel generator 674-M-6 unavailable due to TM
Post-Accident Operator Errors		
VDF-SYS-XF-0VDP2	High	Operator failure to actuate VVS (VDP2)
VDF-SYS-XF-VDP1D	High	Operator failure to diagnose the need for depressurization (VDP1D)
CDP-SYS-XF-STPWR	High	Operator failure to realign after recovery of site power
DGP-MDP-XF-06272	Moderate	Operator fails to manually start diesel oil transfer pump 627-2 or 627-3
PPS-XSW-XF-OPSCR	Moderate	Operator fails to actuate a manual scram
LDW-SYS-XF-PURGE	Moderate	Operator fails to initiate LDW purge
Pre-accident Human Errors		
DGP-BKR-XM-074FN	Moderate	Breaker on 674-E-1 feeding 674-M-6 radiator fan 4-8 miscalibrated

a. Impact categories of high, moderate, and low are based on the Fussell-Vesely (F-V) importance measure that is defined as the difference between the fuel damage frequency with the event at its nominal value and the fuel damage frequency with the event set to zero, normalized by the nominal fuel damage frequency. It indicates the fraction of the total fuel damage frequency that involves the event. Impact categories are defined as follows: high impact has a F-V that is greater than or equal to 1.0E-2, moderate impact is less than 1.0E-2 but greater than 1.0E-3, and low impact (not included in the table) is less than 1.0E-3.

4. RISK COMPARISONS

4.1 COMPARISON TO PREVIOUS ATR PRA RESULTS

This section provides a comparison of Revision 1 of the ATR PRA completed in September 1991¹ to Revision 0¹⁶ completed in December 1989. Descriptions of the improvements made to the PRA are listed below. Some of the improvements decreased the fuel damage frequency, while others increased the fuel damage frequency.

- Large LOCA and experiment loop rupture direct damage frequency estimates were refined using a comprehensive program to characterize the actual conditions and locations needed for such events to actually result in direct damage to the ATR core. (See the preceding section.) The direct damage large LOCA (DLC) frequency was decreased from $5.0\text{E-}5/\text{yr}$ to $5.0\text{E-}7/\text{yr}$, and the experiment loop rupture direct damage frequency (RIL) value was decreased from $5.0\text{E-}5/\text{yr}$ to $5.0\text{E-}8/\text{yr}$.
- The human reliability analysis (HRA) was refined. The most significant change to the overall results was the decrease of the failure probability of PCS depressurization actions. (The basic event names for these actions are VDF-SYS-XF-VDP1D, VDF-SYS-XF-VDP1M, and VDF-SYS-XF-OVDP2.) The refinements used new estimates of allowable time for the operators to act based on RELAP/SCDAP models of ATR accident progression.
- ATWS event tree modeling was refined to account for the ability to manually scram the reactor for sequences where core damage analyses showed sufficient time was available as long as PCS circulation flows were maintained.
- An event tree for accident initiator AHT that models potential experiment in-pile loop rupture and direct core damage due to lack of adequate response to a slow overheating of the high temperature loop was added.

- A detailed fault tree was developed for restart of the PCS pumps that explicitly includes all power dependencies, ventilation requirements, and motor-operated valve manipulations.
- A detailed fault tree was developed for the instrument air system.
- A detailed fault tree of the equipment failures and operator failures affecting depressurization of the PCS was developed.
- Diesel fuel supply pump power supplies were added to the fault tree model.
- Commercial power recovery modeling to account for the likelihood of recovery within 4 hours was refined.
- PCS check valve modeling was improved to credit proceduralized operator actions to close motor-operated valves downstream of the PCS pump upon shutdown.
- More basic event unavailabilities were obtained using ATR-specific data. This addition provides a total of 43 basic event unavailabilities obtained using ATR-specific data. The additional basic events include the following:
 - Pressurizing valve LCV3A failure to open (MKP-AOV-CC-LCV3A)
 - FIS injection valve LCV-7A (and LCV-7B) failure to open (FIS-AOV-CC-LCV7A and FIS-AOV-CC-LCV7B)
 - FIS injection valve 615 (and 616) failure to open (FIS-AOV-CC-V1615 and FIS-AOV-CC-V1616)
 - Vessel vent valve failure to open (VVS-AOV-CC-GB231 and VVS-AOV-CC-GB232)
- The following plant-specific data were updated:
 - Pressurizing flow control valve FCV-1-8 failure to close (MKP-AOV-00-0FCV8)
 - Pressure control valve PCV-1-1 fails to close/open (MKP-AOV-00-0CV11 and MDP-AOV-CC-0CV11)

- Diesel generator M-6 fails to start (DGP-DGN-FS-000M6)
 - Diesel generator M-6 fails to run (DGP-DGN-FR-000M6)
- Dependent failure event generic "Beta" factors used to predict dependent failure probabilities were updated.

A comparison of the predicted frequencies by sequence type is provided in Figure 6. Reasons for the most significant differences are discussed below.

- Direct damage events decreased by 99.4% because of the new large LOCA and experiment loop rupture frequencies. [Note: fractional changes in frequency were obtained using $(Old - New)/Old$.]
- ATWS events decreased by 64% due to new ATWS event tree modeling.
- Loss of flow (pressurized) events decreased by 88% due to decreased operator error probabilities, refined check valve modeling, and refined commercial power recovery modeling. However, the diesel fuel supply modeling had a countering effect.

Figure 6. Comparison of ATR Revisions 0 and 1 results by sequence type

4.2 COMPARISON TO OTHER RISK ASSESSMENTS

It is useful to compare the ATR risk assessment results to the results of other PRAs, particularly of U.S. DOE reactor facilities. Table 26 provides the results of five other DOE risk assessments and five commercial power reactor PRAs. The ATR results of the Level 1 analyses including both internal and external analyses are approximately an order of magnitude below the three other risk assessments that provide a total Level 1 result. This is because of the aggressive ATR risk reduction program and refined risk estimates.

Outstanding characteristics of the ATR PRA in comparison to other PRAs include the following items.

- The ATR PRA included a very comprehensive external event analysis that included internal fire, internal flooding, seismic activity, external flooding, volcanic activity, accidental release of chemicals from onsite storage, industrial facility accident, transportation accident, aircraft impact, and extreme wind and tornado.
- The ATR PRA included a comprehensive search for potential interactions between seismic, flood, fire, and wind events. Such interactions can be extremely important contributors to risk, especially in older nuclear facilities. The ATR PRA comprehensively identified and quantified these interactions and is one of the first PRAs to do so.
- A novel approach was taken to provide the estimation of frequencies of large LOCA and experimental loop ruptures that could lead directly to core damage. This approach involved a comprehensive program to characterize the actual conditions and locations needed for such events to actually result in direct damage to the core (using a modified version of the RELAP/SCDAP code applicable to ATR) combined with a variety of sophisticated

Table 26. Mean fuel damage frequencies from various nuclear reactor facility PRAs

Facility	Mean Fuel Damage Frequencies (Yr ⁻¹) ^a							
	Internal		External					Total
	Total	Seismic	Fire	Wind/ Tornado	Internal flooding	External Flooding	Total	
ATR	2.4E-5	1.9E-5	1.0E-6	1.2E-6	1.1E-6	1.3E-7	2.5E-5	4.9E-5
US DOE Facilities								
EBR-II ^b	3.6E-6	-	4.4E-6	-	-	-	-	-
HFBR ^c	5.0E-4	-	-	-	-	-	-	-
HFIR ^d	3.1E-4	1.2E-4	1.0E-5	2.9E-4	-	-	4.3E-4	7.4E-4
N-Reactor ^e	6.4E-5	1.7E-4	1.9E-5	-	-	-	1.9E-4	2.5E-4
SRP K-Reactor ^f	2.1E-4	1.1E-4	1.4E-7	-	-	-	1.1E-4	3.1E-4
NRC NUREG-1150 PRAs^g								
Surry Unit 1	4.0E-5	1.2E-4	1.1E-5	-	-	-	1.3E-4	1.7E-4
Sequoyah Unit 1	5.7E-5	-	-	-	-	-	-	-
Peach Bottom Unit 2	4.5E-6	7.7E-5	1.9E-5	-	-	-	9.6E-5	1.0E-4
Grand Gulf Unit 1	4.0E-6	-	-	-	-	-	-	-
Zion Unit 1	3.4E-4	-	-	-	-	-	-	-

a. Mean fuel damage frequencies cited may be point estimate means or uncertainty distribution means. Point estimate means were cited when available.

b. D. J. Hill, W. A. Ragland, J. Roglans, "EBR-II PRA - Presentation to the ACNFS PRA Subcommittee," June 21, 1991.

c. M. A. Azarm et al., Level I Internal Event PRA for the High Flux Beam Reactor, Revision 1, July 1990.

d. D. H. Johnson et al., The High Flux Isotope Reactor Probabilistic Risk Assessment, ORNL/RRD/INT-36, Oak Ridge National Lab., 1988.

e. A. Camp et al., Level III PRA for N Reactor, WHC-MR-0045 (SAND89 -2102), April 1990.

f. C. N. Amos et al., Savannah River Site Reactor Safety Assessment, WSRC-RP-91-041-DRAFT, February 1991.

g. Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, 1990.

frequency estimations techniques. Five different estimation techniques were employed, with the final frequency estimates obtained from a combination of the individual technique estimates. The five methods included three techniques involving data from past performance of DOE and commercial reactor components, one involving expert judgement elicitation, and one involving ATR-specific probabilistic fracture mechanics analysis using a modified version of the PRAISE code. This comprehensive approach was needed because none of the individual techniques covers all issues and concerns. This comprehensive approach is different from similar efforts for the DOE Savannah River production reactor, High Flux Isotope Reactor (HFIR), and High Flux Beam Reactor (HFBR) PRAs, in which only one of the five techniques was used. The use of all five techniques in the ATR PRA results in more credible estimates.

5. RISK MANAGEMENT AND PRA APPLICATIONS

Completion of the current aggressive and comprehensive ATR risk reduction effort will enable ATR to shift from a more costly risk reduction phase to a less expensive risk maintenance phase. In addition to a reduction in the risk profile (the cost savings of which are significant but difficult to estimate), the comprehensive ATR PRA models and results have already been applied to eliminate some previously planned facility upgrades that were shown to have a negligible effect on reducing fuel damage risk and to define a more cost-effective approach for other projects that would still meet risk reduction objectives. It is expected that risk management applications of the ATR PRA will result in cost savings to ATR (neglecting avoided costs of a severe accident) that will more than cover the costs associated with development of the PRA.

The principal objectives of the ATR PRA program have been to apply the results and insights from the PRA to guide safety improvements at ATR. The PRA has been used to evaluate and prioritize potential facility or operational changes and questions or concerns about the facility and its operation. An ATR risk management report is issued twice a year that reports on specific PRA applications, event analyses, and event trends over the period. The applications are discussed below in four categories: operational safety, operational occurrence reviews, plant safety, and safety envelop upgrade program.

5.1 OPERATIONAL SAFETY

A principal application of the ATR PRA has been to provide input to the development of new symptom-based emergency procedures now being implemented at the ATR. The ATR PRA (Levels 1, 2 and 3) will continue to be utilized for new procedures development and for follow-on accident management system and procedures development. The ATR PRA defined important accident sequences, the important contributing component and human errors and their associated failure modes, and important system interactions or dependencies in the accident

sequences. These PRA results are important input for the development of symptom-based procedures. The ATR PRA included detailed human reliability analyses for the most significant human errors. These detailed human reliability analyses were used to help develop and define significant improvements in the operator actions for the new procedures.

5.2 OPERATIONAL OCCURRENCE REVIEWS

Operational occurrences and other ATR events are periodically reviewed for their risk significance and potential influence on the ATR PRA. Event data are collected, tracked, and periodically updated to define current event and component failure frequencies for the ATR and to detect any significant trends important to the ATR PRA risk estimate. Effects of past facility improvements show a trend over the last five years of data used for the ATR PRA, 1984 through 1988, of decreasing occurrences of several initiators (normal or spurious shutdown, loss of commercial or off-site power, and experiment loop upsets capable of resulting in a reactivity increase). If the first 15 years of ATR operation were used to estimate the frequency of these events, then the internal events fuel damage frequency would be almost double the current value and would not be representative of the current plant configuration. Planned replacements of the process and experiment control systems should provide further improvements. However, as the ATR facility ages, some event or component failure frequencies may be found to increase.

5.3 PLANT SAFETY

Suggestions for significant facility safety or risk upgrades have been developed and transmitted to management as the ATR PRA has developed and whenever significant improvements or risks have been identified in the process of the development of the ATR PRA. These suggestions have also been itemized for the ATR capital projects annual planning cycle and the PRA has been used to aid in prioritizing future capital projects. ATR risk-reducing upgrades that are being implemented and that contributed to the significant reduction

in fuel damage frequency for the Level 1 ATR PRA are listed in the "Risk Insights" section of this report.

5.4 SAFETY ENVELOP UPGRADE PROGRAM

The ATR PRA is being extensively used to provide guidance and input to the following ATR safety and operational upgrade projects:

- Upgraded (symptom-based) emergency procedures
- New process control system design
- New experiment control system design (human factors input)
- ATR simulators upgrade suggestions
- Accident monitoring instrumentation and safety parameter display system (SPDS)
- Emergency planning upgrades
- ATR aging evaluation and life extension program
- Equipment qualification
- ATR maintenance improvements
- ATR Updated Final Safety Analysis Report (UFSAR) program.

ATR PRA guidance and input to the UFSAR program consists of:

- Definition of ATR critical safety functions (these were also needed for symptom-based emergency procedures development and SPDS design)
- Definition of ATR "systems important to safety"
- Definition of Class 1E electrical systems
- Definition of seismic category I systems
- Definition of an inclusive accident list and the most important accident sequences for Chapter 15 analyses
- Accident modeling input and model development from the Level 2 PRA severe accident analyses.

The safety system definition input for the UFSAR is also useful for the aging and life extension program, equipment qualification, and maintenance improvement programs. Several studies have also been performed using the results of the ATR PRA to define risk-important components and systems and needed data collection for the aging and life extension and maintenance improvement programs.

6. REFERENCES

2. ACNFS meeting minutes, Washington, D. C., June 27, 1991.
3. PRA Procedures Guide, NUREG/CR-2300, January 1983.
4. A. Mosleh et al., Procedures for Treating Common Cause Failures in Safety and Reliability Studies, NUREG/CR-4780, January 1988.
5. A. D. Swain, Accident Sequence Evaluation Program Human Reliability Analysis Procedure, NUREG/CR-4772, February 1987.
6. A. D. Swain and H. E. Guttmann, Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278, 1983.
7. G. W. Hannaman, A. J. Spurgin, and Y. D. Lukic, Human Cognitive Reliability Model for PRA Analysis, NUS-4531 (Rev. 3), December 1984. (Draft)
8. L. N. Haney et al., Comparison and Application of Quantitative Human Reliability Analysis Methods for the Risk Methods Integration and Evaluation Program (RMIEP), NUREG/CR-4835, January 1989.
9. A. Poucet, "The European Benchmark Exercise on Human Reliability Analysis," Proceedings of the International Topical Meeting Probability, Reliability, and Safety Assessment PSA'89, Pittsburgh, Pennsylvania, April 2-7, 1989.
10. D. I. Gertman et al., Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR), NUREG/CR-4639, June 1989.
11. CAFTA, Version 2.1e, Science Applications International Corporation, Los Altos, California, (executable code dated 1/31/91).
12. Science Applications International Corporation, CAFTA User's Manual, Los Altos, California, November 1990 (with update sheets dated 2/91).
13. COMPBRN III Plus, Version dated May 23, 1990, G. Apostolakis, Nathan Sui, and Vincent Ho.
14. UNCERT, Version UNC386, Science Applications International Corporation, Los Altos, California, (executable code dated 5/16/91).
15. Science Applications International Corporation, UNCERT User's Manual, Los Altos, California, April 1991.

17. S. A. Eide et al., Advanced Test Reactor Probabilistic Risk Assessment External Events Analysis, PG-T-91-001, March 1991.
18. S. A. Atkinson, S. A. Eide, and T. A. Thatcher, Advanced Test Reactor Risk Summary, EGG-PRP-10025, December 1991.

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