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REVIEW AND UPDATES OF THE RISK ASSESSMENT FOR ADVANCED TEST REACTOR OPERATIONS FOR OPERATING EVENTS AND EXPERIENCE

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ABSTRACT

Annual or bi-annual reviews of the operating history of the Advanced Test Reactor (ATR) at the Idaho National Engineering Laboratory (INEL) have been conducted for the purpose of reviewing and updating the ATR probabilistic safety assessment (PSA) for operating events and operating experience since the first compilation of plant-specific experience data for the ATR PSA which included data for operation from initial power operation in 1969 through 1988. This technical paper briefly discusses the means and some results of these periodic reviews of operating experience and their influence on the ATR PSA.

I. INTRODUCTION

A. Brief Description of the ATR

The Advanced Test Reactor (ATR) is a Department of Energy (DOE), high neutron flux materials irradiation test reactor located at the large Idaho National Engineering Laboratory (INEL) reservation in Eastern Idaho. The ATR is designed with nine flux traps within the reactor core in which in-core irradiation facilities are located for reactor or other materials irradiation effects experiments (Figure 1). Upsets or failures within these experiment facilities within the reactor core are an additional potential source of events which could cause reactor upsets or accidents with a potential for core fuel damage. Therefore, these experiment facilities along with the reactor systems are important to the risk assessment for the ATR.

B. Brief Description of the ATR PSA

A full scope (Levels I, II, and III) probabilistic safety assessment (PSA) which includes external events (natural phenomena hazards) and shutdown operations has been performed for the ATR. The Level I PSA without external events was completed in 1989, the Level I PSA was updated with external events in 1991, and a major update and an addition for Levels II and III of the PSA and including shutdown operations was completed in 1994.

C. Introduction to the Topic

The ATR PSA is a living and applied risk assessment rather than just a historical safety basis document. Therefore, it is necessary to review how the operation influences the risk assessment and what changes to the risk assessment should be made to appropriately and adequately represent the facility operation risk as well as is reasonably possible. To do this, it is planned to perform an update to the PSA whenever major facility modifications or operational changes necessitate an update in order that the PSA would continue to adequately represent the facility risk, and in order for the PSA to adequately define the important risk contributors and to provide the results and models necessary for application of the PSA to operations. Such an update is currently underway for the ATR PSA. But, it is also important to assess the relevant facility operations experience from month-to-month and year-to-year operation for how that experience influences the estimated risk and for changes that should be made to the risk models to better represent the current facility risk. This type of a review is also important to determine whether the risk continues to be controlled or managed such that the facility operations safety basis is adequately maintained. This technical paper briefly discusses the means and the results of these periodic reviews of operating experience and their influence on the estimation of the facility risk defined by the PSA.

II. OPERATING EXPERIENCE REVIEWS

Annual or bi-annual reviews of the operating history of the ATR at the INEL have been conducted for the purpose of reviewing and updating the ATR PSA for operating events and operating experience since the first compilation of plant-specific experience data for the ATR PSA which included data for operation from initial power operation in 1969 through 1988. This operating history data compilation consists of the following reviews.

- Review of ATR occurrence reports and other event reports for events significant to the PSA. An initial review for significance occurs as soon as event details are known, but incorporation of the event into the full risk review and a possible PSA update is not done until the annual or bi-annual review occurs.
- Review of operating occurrences at other nuclear reactors for their potential applicability and significance to the ATR PSA. (These reviews are also initially conducted for significance to the ATR as the operating occurrence reports are received.)
- A detailed review of operating reports and operating logs. (Maintenance records at the ATR are not yet capable of providing useful information for the PSA review. Therefore, the operating logs are used to obtain useful maintenance and outage data.)

These information and data sources are reviewed to look for:

- Reactor scrams and the cause of the scrams (normal shutdown, spurious, forced due to a problem, automatic or manual, etc.)
- Reactor safety system challenges
- Potential precursor events or changes in initiating event frequencies
- Failures of risk-significant equipment (RSE)
- Maintenance outages of RSE
- RSE manipulations (pump starts and stops, valve manipulations, which systems or trains are in operation and which are on standby, etc.)
- RSE run times
- Experiment problems, failures, or outages and whether or not they resulted in a reactor protective action or upset
- Electric power systems upsets and outages or upsets or outages of other support systems (water, air)
- Operation with off-normal equipment configurations or conditions
- Off-normal events or behavior which were not significant enough to result in a plant upset or formal occurrence reporting but which may still be useful data for the plant PSA
- Significant external events (high wind, flooding, range fires, etc.) or internal fire or flooding events

The data from the review are organized according to the accident initiator groups used for the PSA so that the analysis of the data can be directly related to the risk assessment to determine the effect on the estimated risk and safety significance. The data is analyzed to determine and report upon the following items.

- Changes in elements of the PSA (initiating event frequencies, equipment failure probabilities, dominant sequence frequencies, etc.)
- Observed trends in the data or risk trends, explanations for the trends or changes, other observations potentially useful to operations or maintenance, and any recommendations based on the observations and data analysis

III. REACTOR RISK STATUS REPORT

The Reactor Risk Status Report reports on the results and conclusions of analyses of data and information regarding the probabilistic risk of severe accidents for the ATR. The report provides information on:

- The overall risk levels, Level I and Level III including uncertainties

- How the plant risk is changed from previous years, trends in the risk and reasons for the changes (such as systems or operations upgrades or degradations)

- A comparison of the ATR risk to other nuclear reactors and to DOE and Nuclear Regulatory Commission guidelines

- Updates and analysis for equipment plant-specific data used for the PSA

- Whether there are any precursor events or trends that are a potential future risk concern or whether there are indications of significant changes in equipment failure probabilities or aging effects

The review and update also evaluates the risk significance of major plant modifications or operational changes and incorporates the results of any specific risk assessments performed since the last PSA update.

The results of the review for the plant risk status are communicated to plant operations management by an informal risk summary report. In addition, periodic updates of the PSA will be made when the accumulative effects of plant changes or experience make the prior PSA along with the results of subsequent specific risk assessments and risk reviews difficult to use and interpret.

IV. PSA IMPACT OF OPERATING EXPERIENCE

A. Initiating Event Frequencies

Potential initiating events or precursor events are defined from the annual/biannual operating history review and the ATR PSA initiating event frequencies are updated for the additional data. Table 1 displays the event data in five year increments and the resulting initiating event frequencies. Two initiating event frequency columns are shown in Table 1; one column for the initiating event frequencies used for the ATR PSA from data taken through 1988, and updated initiating event frequencies for data taken through 1994. These latest frequencies are used in the current ATR PSA update. The initiating event frequencies are defined using a Bayesian update of a noninformative prior[1] since most of the event data is sparse with one or zero failures and to assure that the potential for a future event is appropriately considered in the initiating event frequencies. Some of the initiating event frequencies cannot be based on plant-specific history because they are for rare events which are never expected to occur during the plant lifetime.

The collection of additional years of data generally results, for most events, in lower plant-specific initiating event frequencies for those systems that continue to function with no more than one failure. The 1994 updated initiating event frequencies are reduced by about 24 % from the 1988-based ATR PSA frequencies for the failure free or very low failure systems. However, some initiating event frequencies are seen to be increased by the accumulation of additional experience data, such as for the loss of commercial (off-site) power, loss of diesel generator power (ATR operates with a continuously operating diesel generator powering operationally important loads and experiment loads), and potential loop experiment initiated reactivity addition events. None of the loop experiment failures actually resulted in a reactivity addition, but they caused a reactor protective trip. This is true for most of the listed events except for the loss of electrical power events.

The experience data is significantly affected by facility changes and changes in operation. Reactor operation has transitioned from high power but short operating cycles to lower power and longer operating cycles which affects the number of scrams and shutdowns. The reactor scram and shutdown frequency has also been significantly affected by the installation of a new Plant Protection System in 1978 and a state-of-the-art control room in 1986. The addition of a third, automatic-start-and-load diesel generator and overhaul of the two heavy duty diesel generators affects the reliability of the diesel generator power system. The recent installation of a state-of-the-art instrumentation and control system for the loop experiments is expected to significantly lower the frequency for loop experiment failure events. This trend is already seen in 1994 data.

The risk and safety significance of internal fire and flooding events and high wind events is dependent on event specifics and location. Therefore, these observed event frequencies in Table 1 cannot be directly applied to the PSA. They are evaluated based on the specifics of the event severity, observed consequences, and location. Internal flood events which were of potential risk significance involved flooding in the pit in which the primary diesel generators are located. The concern with flooding in this area is the potential for a loss of the diesel generator power bus. Therefore, these events are evaluated for their effect on the risk for loss of electric power sequences. However, a risk-reduction upgrade performed in response to the original analysis for internal fire and flood events, a relocation of the risk-significant Utility battery-backed power system away from other switchgear and from its prior exposed location under the diesel pit, has limited the significance of these diesel pit flooding events.

B. Plant-Specific Failure and Usage Data

Table 2 displays data that is collected and tracked for the usage and failure events for risk-significant equipment (RSE). Also shown in Table 2 are the electric power failure events translated into failure rates useful for the PSA. RSE that have a long and reliable history of operation will generally have a failure rate or failures on demand better than the published nuclear reactor/facility generic data. Additional years of good operation results in additional reduction of the failure data compared to the generic data. When using actual facility experience, good equipment and operations will not be penalized by use of the "industry average" data. However, the operating experience data is Bayesian updated as for the initiating event data. The generic failure data, from Reference 2, was assumed to have a lognormal distribution for the updates.

When the data identifies RSE that is not performing as well as it should, then that equipment can be recommended for improvement. Such equipment is usually recognized beforehand to be a potential problem, but the identification of the risk significance of the observed failure data can be helpful for deciding upon an appropriate action or the timing of an upgrade. For example, the ATR pressure control valve was found to have a high number of failures compared to the industry data. The valve was replaced and in the 5 years since its replacement its performance is seen, from Table 2, to be better than for the generic data. The data also identifies that the ATR diesel generators do not operate nearly as well as those used for nuclear power plants. But the ATR diesel generators also do not have the same risk significance as nuclear power plant emergency diesel generators. However, the risk significance of the operation of the ATR diesel generator power system has been recently reviewed for potential changes or improvements such as the use of uninterruptible power systems for the most risk and safety significant loads.

The ATR PSA also utilized plant-specific outage times for RSE determined from operating experience. This data has been more difficult to obtain because of a lack of specific equipment outage maintenance records. Therefore, RSE outages are identified from operating logs and outage durations are estimated from work order duration and interviews with maintenance personnel. Use of work order duration for equipment outage duration is conservative but is used when no other data or method of estimation is available.

Most ATR RSE outages occur during shutdown periods (reactor outages) since the ATR operates with 10 to as many as 18 outages a year. Therefore, most RSE outage times were not significant for the power operation PSA but were more important for the shutdown operations PSA. Therefore, most of the RSE outage data used for the PSA is recent and was not collected until the shutdown operations PSA was performed in 1992-93. It is desirable that future data would better define the RSE outage time to reduce the current conservatism in the RSE outage durations.

C. Plant-Specific Experience Data Insights

A primary purpose and importance of the continuing review and evaluation of ATR operating experience is to base the facility risk assessment on actual facility design and operation rather than some idealized systems or operations or on the experience of some other loosely related facilities. As seen in the Tables, the actual facility experience can be worse than the industry norm but, in most cases, it is significantly better. Another important purpose of the review and evaluation is to observe and to be able to report upon the significance of changes in the systems risk. This knowledge can provide a focus for further evaluation and potential suggestions for improvements.

Application of the updated experience-based initiating event frequencies (Table 1) to the ATR PSA will reduce the mean fuel damage frequency for the listed events by 20 percent. However, other events for which experience data does not define the initiating event frequency dominate the risk, especially potential earthquakes. Therefore, the total mean fuel damage frequency is only reduced by 3 percent when the updated experience-based initiating event frequencies are applied.

Most of the RSE failure data in Table 2 are not significantly affected by the collection of additional data since the 1991 (Revision 1) version of the ATR PSA was completed. However, review of the equipment failure data and operating occurrence reports for the ATR reveals an increase in age-related failures, not unexpected for a facility that is over 20 years old. This is especially apparent in Table 2 for the primary pump check valves and the pressurizing pumps. The consequence of the failures is an increase in the failure probability for this equipment of about a factor of 10. However, because of the multiple failures required for a fuel damage sequence involving these components, the PSA is not significantly affected.

Other age-related failures of less significant equipment are also noted in the experience reviews. Some failures have occurred in switchgear for loop experiment pumps which effects the loop failure category of initiating event frequencies (event RLH in Table 1). Age-related failures have also occurred in confinement door seal systems and control rod drives which do not significantly affect the PSA results.

The annual or bi-annual data collection and review has been found to be too infrequent to provide risk-based analysis for some equipment wear-out before action has to be taken to correct the problem. For example, the primary, heavy duty diesel generators experienced a significant increase in failures in 1992 as they approached component end-of-life, but this was not seen in the PSA data until the collection and review of the 1992 data in 1993. Prior to 1992, increased maintenance had been able to maintain their reliability, therefore no significant change was seen in the failure event data. So, the significant increase in failures occurred within one year, and the machines were overhauled in that same year. However, the PSA was still used to define the risk significance of operation with the higher probabilities of failure and to define the preferred option for providing power while continuing to operate during the sequential overhaul of the diesel engines. Because of the reduced reliability of the primary diesel generators, it was determined that a portable diesel generator needed to be brought on site to provide power with the good auto-start diesel generator and the other primary but questionable diesel generator as backups.

ACKNOWLEDGEMENTS

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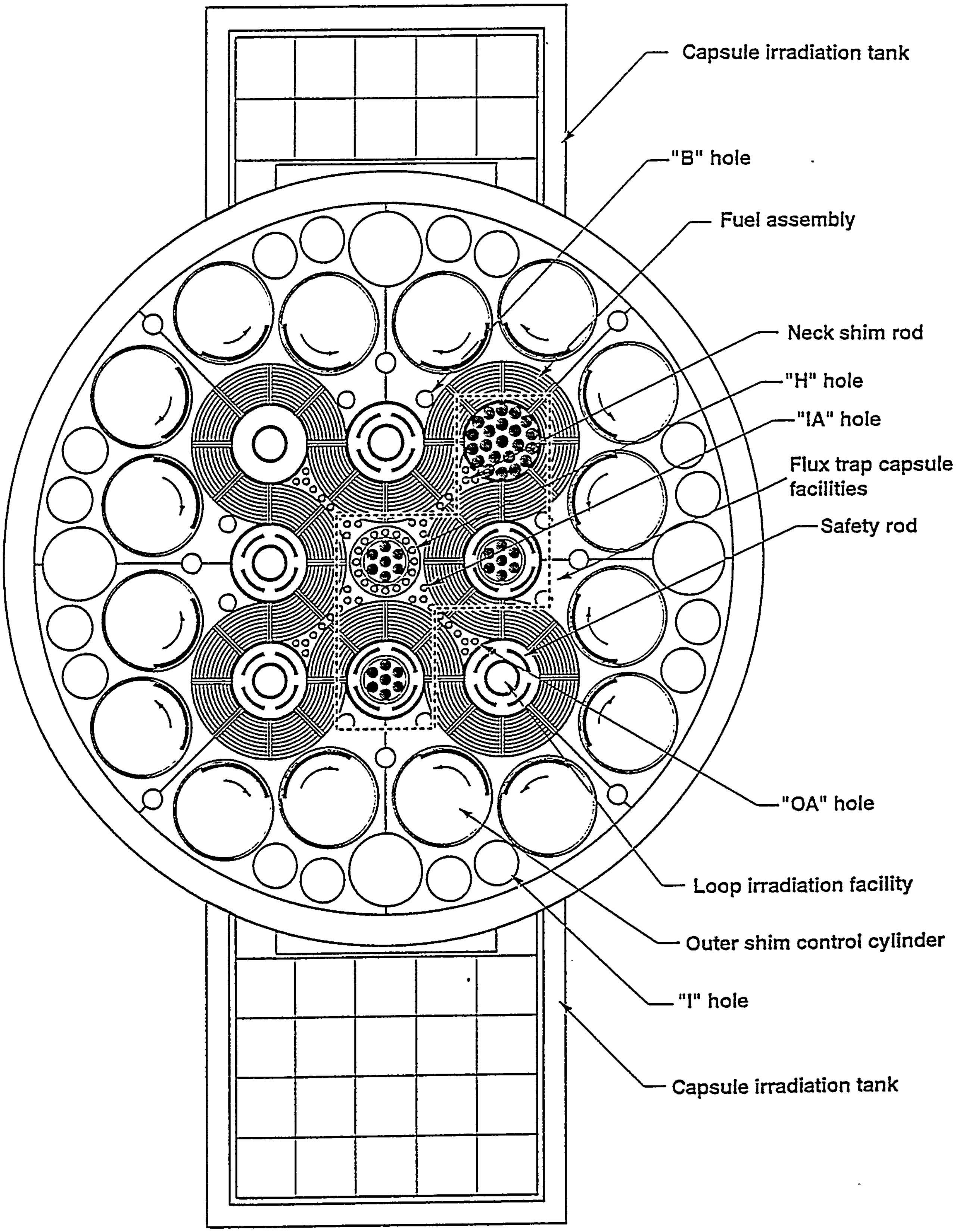

Figure 1. ATR Core Cross Section View

Table 1. ATR INITIATING EVENT FREQUENCY FROM EXPERIENCE DATA

Initiating Event Category	Initiating Event Description	ATR Events, Scrams, Shutdowns 1969-73	1974-78	1979-83	1984-88	1989-94	Rev. 1 PSA IEF/yr	Updated IEF for period chosen: Period	years	Events	Bayesian Updated IEF/yr
PSD	Planned shutdowns					51					
USD	Unplanned/spurious SD					25					
RSD	Reactor shutdowns	253	219	151	109	76	21.9	1989-94	6	76	12.8
LPF	Low flow	0	0	0	1	0	0.075	1969-94	26	1	0.058
LPP	Low pressure	1	1	0	0	0	0.05	1979-94	16	0	0.031
HCP	High pressure	0	2	0	0	0	0.05	1979-94	16	0	0.031
LHS	Loss of heat sink	0	0	0	0	0	0.025	1969-94	26	0	0.019
LCP	Lose commercial power	5	11	16	4	6	0.9	1984-94	11	10	0.95
LDC	Lose diesel-comm bus	1	0	0	0	0	0.075	1969-94	26	1	0.058
LDP	Lose diesel-gen power				0	6	0.25	1987-91	5	1	0.30
LIA	Lose instrument air	0	0	0	0	0	0.025	1969-94	26	0	0.019
S12	Small LOCA (at power)	0	0	0	0	0	0.025	1969-94	26	0	0.019
SSLC	Shutdown small LOCA						0.0074	1969-94	26	0	0.0057
LLC	Large LOCA	0	0	0	0	0	RA	1969-94	26	0	RA
ISL	LDW Interfacing LOCA	0	0	0	0	0	RA	1969-94	26	0	RA
ISF	FIS interfacing LOCA	0	0	0	0	0	RA	1969-94	26	0	RA
RFH	Fast reactivity addition	4	2	2	0	1	0.1	1984-94	11	1	0.14
RSH	Slow reactivity addition	0	1	1	0	0	0.1	1984-94	11	0	0.045
RLH	Loop initiated reactivity	10	16	25	5	9	1.1	1984-94	11	14	1.3
RLP	Lobe power unballance	0	1	2	0	0	0.1	1984-94	11	0	0.045
RIL	Large reactivity addition	0	0	0	0	0	RA	1969-94	26	0	RA
AHT	AHT loop overheating	2	4	3	0	0	0.1	1984-94	11	0	0.045
RPF	Perched fuel drop	0	0	0	0	0	RA	1969-94	26	0	RA
LFS	Local fuel failure	0	0	0	0	0	0.025	1969-94	26	0	0.019
CAN	Canal draining initiator	0	0	0	0	1	0.025	1969-94	26	1	0.058
EXT	External Events and internal flood/fire										
Flood	internal	1	0	3	0	6	See	1969-94	26	10	0.404
Fire	internal	0	0	1	1	0	EXT	1969-94	26	2	0.096
Wind	greater than 75 mph					1	note	1989-94	6	1	0.3

Category	NOTES:
all	Rev. 1 PSA initiating event frequencies based on data through 1988
any	Later experience period is chosen if improved operations or facility changes make a difference
	Modifications expected to influence IEFs:
	New Plant Protection System in 1978
	New reactor control room in 1986
	New process control system in 1993 and experiment control system in 1994
any	RA = rare event; cannot estimate IEF from limited plant experience
LDP	No count included until 1987 after installation of auto-backup diesel generator
LDP	1987-1991 believed typical for current operation after overhaul of diesel generators in 1992
RFH	All these events are regulating rod withdrawals (regulating rod cocking events)
SSLC	Shutdown small LOCA assumed to have frequency for S12 times fraction of operation in shutdown (0.3)
EXT	External and internal flood/fire event IEFs are event and location specific

Table 2. ATR RISK SIGNIFICANT COMPONENTS FAILURE DATA

Component	Failure mode	Data to 1988 for Rev. 1 ATR PRA				1994 Update		
		Failures/hr or demands	Failure rate	Generic value	Bayesian update	Failures/hr or demands	Failure rate	Bayesian update
Emergency pump M-10 (ac)	to start	0/40	0	3E-03	2.5E-03	1/231	4.3E-03	3.2E-03
Emergency pump M-11 (dc)	to start	0/54	0	3E-03	2.4E-03	0/273	0	1.8E-03
Pressurizing flow pumps	to start	1/78	1.3E-02	3E-03	5.7E-03	19/338	5.6E-02	4.6E-02
Primary pump check valves	to close	0/2296	0	1E-03	2.5E-04	8/2754	2.9E-03	2.5E-03
Pressurizing flow control valve	to close	0/70	0	1E-03	7.5E-04	0/98	0	7.2E-04
Pressure control valve	to close/open	0/53	0	1E-03	7.6E-04	0/90	0	7.3E-04
Makeup valve	to open	0/55	0	1E-03	7.6E-04	0/75	0	7.4E-04
Lower emergency injection valves	to open	0/70	0	1E-03	7.5E-04	0/150	0	6.7E-04
Upper emergency injection valves	to open	0/26	0	1E-03	8.0E-04	0/118	0	7.0E-04
Vessel vent valves	to open	0/96	0	1E-03	7.2E-04	0/156	0	6.7E-04
Safety rods	to insert	0/7336	0	3E-05	1.6E-05	0/8630	0	1.6E-05
Rod clutch coil controllers	to open	0/2600	0	3E-04	1.2E-04	0/3760	0	1.0E-04
PPS subsystem Scram signal	to actuate	0/547	0	3E-04	1.7E-04	0/680	0	1.6E-04
Diesel-generators M-42/M-43	to start	8/75	1.1E-01	1E-02	7.0E-02	23/363	6.3E-02	5.7E-02
Auto-start diesel-generator M-6	to start	8/191	4.2E-02	1E-02	3.3E-02	29/399	7.3E-02	6.7E-02
Diesel-generators M-42/M-43	to run	1/13360 hr	7.5E-5/h	5E-3/h	1.8E-04	1/29216 hr	3.4E-5/h	9.5E-05
Auto-start diesel-generator M-6	to run	2/423 hr	4.7E-3/h	5E-3/h	4.0E-03	2/980 hr	2.0E-3/h	2.0E-03
Diesel-commercial bus	short/open	1/20 yr	8.2E-6/h	NA	1.2E-5/h	1/26 yr	6.3E-6/h	9E-6/h
Commercial power	fails	4/5 yr	9.1E-5/h	NA	1.0E-4/h	10/11 yr	1.0E-4/h	1E-4/h

Component	NOTES:
All	Times over which the data was collected vary between components
All	Generic data assumed to have a lognormal distribution
Diesel-commercial bus	Considering failure only while operating at power (70 % or less of the year)