CONF-861211--8 BNL-NUREG-38408

ANALYSIS OF THE RESULTS OF THE MIDLAND PRA*

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BNL-NUREG--38408

TI86 014763

ABSTRACT

This paper presents the results of a limited review of the Midland PRA (MPRA), aimed at facilitating regulatory analyses and at providing insights into safety related plant failures. In particular, stress was laid on the root causes of accident sequences, particularly their failure modes, viz., hardware, human, maintenance, test, and repair.

Because this information was difficult to extract in this detailed and sophisticated PRA (and in certain others) a special algorithm was developed to display the leading sequences contributing to core damage and/or to public risk in terms of the above generic failure modes. This was done in a hierarchical fashion to allow tracing the important accident sequences to the systems failures.

The weighted core damage frequency (CDF) values of the sequences considered then provided a (quantitative) ranked importance listing of these failure modes, led by hardware failures (in 78%) and human factors (in 37%), etc. Multiple maintenance situations (reflecting on test and maintenance specifications) were also evaluated, but played only a small role ($\leq$3%).

Similar tabulations were also made of the (weighted) importance of the support systems (such as the electric power system, the component cooling water systems, etc.) and of the role of the major systems/functions (e.g., high pressure injection, etc.) contributions to these accident sequences.

Finally, the role of certain particular operational and plant features (e.g., loss of offsite power initiators, RCP seal failure, bunkering, etc.) were briefly examined.

The methods displayed seem applicable to an important class of extant and projected PRAs.

*This work was done under the auspices of the U.S. Nuclear Regulatory Commission. Views in this paper do not necessarily represent those of the U.S. Nuclear Regulatory Commission.

INTRODUCTION

Although the Midland Nuclear Power Plant (MNP) has been abandoned, an extensive and sophisticated Probabilistic Risk Assessment[1] (the MPRA) was carried out on the plant prior to its completion. In addition to paying careful attention to the various plant technicalities, the MPRA was also distinguished by its use of relatively advanced and often improved methods of analysis. By the same tokens, however, (viz., its extensiveness and intensiveness) the methodology and results of the MPRA were not easily accessible to outsiders concerned with general insights. The NRC requested BNL to carry out a reduced scope review of the MPRA in order to make its output more readily available to and usable by such parties.

Since the NRC was mainly interested in Midland's susceptibility to core damage, the review focused on those initiating events and accident sequences which contributed significantly to the overall core damage (leading accident sequences) frequency (CDF). The work reported here summarizes the BNL findings. It consists of the following parts:

- a very brief description of the method used for the accident sequence (AS) (event scenario) development in the MPRA (Section 2);
- the main characteristics of the leading accident sequences listed in the MPRA (Section 3);
- the approach used to determine the failure mode contributions (hardware, human, maintenance, test, common cause) to the CDF (these latter are not addressed explicitly among the results of the MPRA), a summary of the importance of various support systems in the leading ASs, and of the role of major system/function features (section 4); and
- a discussion and some brief concluding remarks.

ACCIDENT SEQUENCE DESCRIPTION

The methodology applied in the MPRA is an extended version of the approach developed by Pickard, Lowe, and Garrick, Inc. for performing PRAs starting with the Zion PRA[2] in 1979. It involves as new features the introduction of Event Sequence Diagrams

(ESDS) and the application what one may call a Modularized Event Tree (MET) technique. This allows the construction of large, complete, comprehensive and yet tractable event trees. The use of conditional support system/frontline system states enables the trees and the corresponding event sequences to be decomposed into support system/frontline system (main tree) and containment system (subtree) sequences.

More specifically, the event sequences are characterized by four factors:

- the initiating event frequencies (I),
- the support system state probabilities (SSSs), conditional on the initiating events, but independent of the frontline event trees,
- the logical union of appropriate conditional split fraction (CSFs), which are composed of two parts - the main tree CSFs and the subtree CSF's. (The CSFs are probabilities conditional on the initiator, on the SSS and the position, in which they occur in the trees.)

The event sequences result either in a success state or a plant damage state representing the state of the reactor core and the failed frontline containment systems.

The complete quantitative event scenarios then take the form (I), viz

I_i[initiating event frequency]*

SSS(I_i,α-k)[appropriate SSS frequency]*

$CSF_1^m(I_i,\alpha\text{-}k)*CSF_2^m(I_i,\alpha\text{-}k)*$...[logical union of main tree CSFs]*

$CSF_1^s(I_i,\alpha\text{-}k)*CSF_2^s(I_i,\alpha\text{-}k)*$...[logical union of corresponding subtree CSFs]

In this symbolic equation, the SSSs, as well as the CSFs are indexed as "alpha-states," α-1, α-2, ... α-9. Their definition and quantification depend on whether, zero, one or two trains of the HPI/LPI and AFWS fail due to support system failures.

GENERAL CHARACTERISTICS OF THE LEADING SEQUENCES

From 47 internal and external initiators the MPRA identified 12 which resulted in the 103 leading sequences with CDF contributions greater than 10^{-9}/year. It is to be noted, that only one of these "leading" initiators involved an "external" event, viz. fire. The MPRA lists 48 plant damage states, of which only a few are found to be important for core damage and even fewer for risk. The total CDF was found to be 3.104×10^{-4}/year.

As a first task in the review BNL performed a statistical analysis of the leading sequences. The statistical and other features of the leading sequences have intrinsic interest and provide gross rankings of contributions to CDF in terms of initiators and PDSs. They also provide a logical and plausible basis for selecting a smaller representative set of sequences for some of the more detailed analyses described below.

Figure 1 depicts the cumulative number of sequences as a function of their contribution to the CDF, i.e., the ordinate is the number of sequences which contribute (individually) more than the CDF given in the abscissa (which follows a logarithmic scale). Two aspects are noteworthy here:

(i) The graph shows an approximate linearity decade by decade (following the first few points), with an occasional shift. This is a specialized example of Benford's Law[3] which states that (empirically) in any large body of physical data, the proportion of data with first significant digit n or less is approximately $\log_{10}(n+1)$. The reason for this lies in the complexity of the nuclear plant and the PRA plant model which result in the logarithm of the sequence CDFs being approximately *uniformly distributed* under most conditions. (A number of other PRAs which were examined (but are not discussed further here) also follow particular versions of Benfords Law.)

(ii) The graph shows a change in slope after about 46 sequences (which account for 84% of the total CDF), the number of sequences per decade (of CDF) changing from 40 to 20. This suggests a convenient restriction for the number of sequences to be considered. [The limited scope of this review required such a restriction, subject, of course to the need for maintaining representative data. It was not, however, necessary to use the restricted set for all the considerations, as indicated below.] The further discussion below shows that with some slight modification this is indeed the case.

A more careful study of the 46 leading sequences cited above shows that they cover 11 of the 12 initiators of interest in this work (see Table 1a). However, the initiator (VS), inadvertent opening of DHR valve is not include, though it involves a sequence important for risk, though not for core damage. In addition, there are three small core damage sequences (caused by fire in the auxiliary building, initiator F-3, and by steam generator tube rupture (TR), respectively), which have significant risk consequences, but which are not part of the leading 46 CDF contributors. It was therefore decided to augment the considerations by including these four additional sequences plus one additional loss of service water sequence, because of the potential importance of this initiator.

Thus, the selected set of sequences discussed below (51 in number)

- cover more than 84% of the CDF,
- includes all the initiators of interest, and
- includes all significant risk (i.e., fatality) contributors.

The overall situation is summarized in Table 2.

As was outlined above, the MPRA plant model and its implementation inthe PRA relates a set of initiators (Table 1a) to a set of plant damage states (Table 1b). This relationship is quantitatively summarized in Table 3 which shows the core damage frequencies (CDF) as a function of the initiators and the plant damage states (and also indicates the presence of early and late fatalities). Table 1a also lists the mean frequencies of the initiators.

Table 3 shows that severe core damage results from scenarios initiated by the loss of offsite power 42% of the time, 16% of the time from reactor trip, 13% of the time from total loss of component cooling water, 6% of the time from auxiliary building fires, 5% of the time from very small LORIs or turbine trips (30%), and the rest of the time from scenarios of very low frequency. The scenarios of very low frequency are initiated by steam generator tube rupture, small LORI, loss of main feedwater, excessive main feedwater, and the total loss of service water.

Table 3 displays a clear correlation between the PDSs contributing the most to the CDF and the dominant scenarios. The leading PDS states (7B, 8A, 7D, and 7A with 25%, 21%, 20%, and 17% contributions to the CDF are all late states when the pressure is high in the RCS with or without water in the reactor vessel cavity, while containment spray and fans may or may not work. These PDS states dominate the leading scenarios. The table also reveals that the top contributors to the CDF dominate the late fatalities. (Early fatalities are caused by largely different initiators.)

ACCIDENT SEQUENCE FAILURE CHARACTERISTICS

In this section consideration is given to more detailed aspects of the accident sequence failure characteristics. These include

(i) the importance of the various failure modes, to wit, hardware (random) failures (W), maintenance connected failures (M), human (operator) failures (H), test induced failures (T), and common cause failures (C), as well as the incidence of multiple maintenance (MM) connected failures;

(ii) the importance of the support systems, viz the component cooling water system (CW), the emergency core cooling actuation system (EC), the engineered safety features actuation of system (ES), the electric power system (EP), the safeguards chilled water system (SC), and the service water system (SW); and

(iii) the role of the major system/function failures (i.e., the split fractions or top events) in the leading sequences of the various initiators included in the MPRA.

Items (i) and (ii) require additional more detailed calculations, and are restricted to the more limited (but still representative) set of accident sequences selected in Section 3. Their generation is described ibelow, and the summarized results are given in Tables 4 and 5.

Items (iii) may be derived by direct inspection of the form (I) (see Section 2) of all the 103 listed sequences, and its results are summarized in Table 6.

The approach for extracting the information for items (i) and (ii) utilizes the fact that in the systems analysis part of the MPRA the conditional split fractions or their component parts and operations (actions) are described in greater detail involving the various failure modes W, M, H, T, C cited above. The accident sequence failure distributions and the corresponding cut sets are then obtained by mapping the dominant sequences to the "space" of the failure modes W, M, H, T, C and solving the transformed equations for the failure mode dependent split fractions using fault tree analysis type algorithms (viz the SETS code). The sequences then appear in the form (II)

$$\sum_{x_1^m}..\sum_{x_2^m}..\sum_{x_1^s}..\sum_{x_2^s}..\sum_{x_1^0}..\sum_{x_2^0}..I_i*SSS(I_i,\alpha-k)*CSF_1^m(I_i,\alpha-k,x_1^m)*CSF_2^m(I_i,\alpha-k,x_2^m)*...$$

$$...CSF_1^s(I_i,\alpha-k,x_1^s)*CSF_2^s(I_i,\alpha-k,x_2^s)*...$$

where the summations over x_1^m, x_2^m...x_1^s, x_2^s... denote summations over all the applicable failure modes for the CSFs' in question (and in some cases, x_1^0, x_2^0...for the relevant support systems).

For example, a sequence which appears in simplified form (I) as

$$(\alpha-5)*\overline{AF}*\overline{SR}$$

where $\overline{AF}$ denotes a (particular) AFWS (auxiliary feedwater system) failure and $\overline{SR}$ denotes a (particular) sump recirculation failure might transform to five terms

$$\overline{ES}(W)*\overline{AF}(W)*\overline{SR}(W) + \overline{ES}(W)*\overline{AF}(W)*\overline{SR}(H) +$$

$$\overline{ES}(W)*\overline{AF}(M)*\overline{SR}(M) + \overline{ES}(W)*\overline{AF}(M)*\overline{SR}(H) +$$

$$\overline{ES}(M)*\overline{AF}(M)*\overline{SR}(H)$$

where the terms $\overline{ES}(W)$, $\overline{ES}(M)$ refer to failure modes of the ESFs (engineered safety featured system).

Using this breakdown the contributions of the failure modes to the 51 dominant sequences (described earlier may be calculated, and the results, summed and appropriately weighted are given for the various initiators in Table 4.

In addition, the importance of cut sets with multiple maintenance components (MM) is included, e.g., the third term in the above example. These numbers provide an upper estimate to one element of conservatism in the calculations. The Plant Technical Specifications generally prohibit multiple maintenance operations, except (possibly) in a single train. No distinction is made here between single and multiple train maintenance actions because the overall figures for the contributions are very small.

The MPRA tabulates the contribution of the various support systems (listed above) to the support system state failure frequencies (based on the GO-code calculations). It is relatively straightforward to apply these results to the sequence definitions (I). The summed (and weighted) results for the various initiators (and overall) are given in Table 5.

Tables 4, 5, and 6 display the most important failure modes, the most important support systems, and the major systems and/or function failures contributing most to the CDF contribution from the leading accident sequences of the most significant initiating events. In all cases the individual sequence results have been weighted by the relative sequence contribution to the total CDF before summation for the initiator totals, and similarly for the overall summations.

Table 6 requires an additional comment. The summation of the importances of the various systems for a given initiator is a measure of the mean complexity of its sequences. This sum, divided by 100 (since the importances are given as percentages here) gives the mean number of major systems in the accident sequences generated by the corresponding initiator. (Also, since a systems involves a number of split fractions, the summation for a given system may sometimes exceed 100%.)

DISCUSSION AND CONCLUSION

Using the detailed sequence descriptions of the MPRA together with the further analysis sketched here, it is possible to identify many important features of features of the MNP and its behavior. Only a few of these are touched on below.

(i) Of primary importance is the loss of offsite power (LOOP) initiator, (AC) which contributes 1.32-4/yr, i.e., 40% of the overall CDF. 69% of this is due to station blackout (SBO) scenarios, involving failure of the two diesels to start or run. Underlying many of these SBO scenarios is the failure to recover soon enough, with the associated battery depletion, and consequent failure of the steam driven pump. A number of other SBO scenarios involve reactor coolant pump (RCP) seal failure (SF) (>9.7-6/yr CDF contribution).

(ii) Other important scenarios involving RCPSF are loss of component cooling water (LCCW) with a CDF contribution of 3.97-5 and loss of service water (LSW) with a contribution of 3.72-6. Thus, RCPSF contributes to about 17% of the overall CDF.

RCPSF plays an important role in various accident sequences because

- it leads to a loss of reactor coolant (LOCA),
- it may inhibit suitable mitigating actions, and
- the events that give rise to it may defeat appropriate mitigating actions (for the LOCA).

(iii) "Bunkering" at the Midland Plant is not to be regarded in its strict definition, but rather as a standard method of constructional separation. Estimates made of the maximum CDF increase (via cooling requirements) possibly associated with such separation and compared with the fire induced CDF to which it is related seem to justify the separation.

The MPRA provides a comprehensive and methodologically advanced overview of the Midland Nuclear Plant using the modularized event tree approach which places the conceptual structure of the analysis and the plant on a rational basis. However, while such techniques and their results are powerful and flexible, and readily adaptable and extendable by the original analysts, they are by no means directly usable by outside reviewers, particularly if they do not have ready access to the (otherwise comprehensive and illuminating) "GO" methodology and algorithms.

Nevertheless, the detailed sequence descriptions and CSF definitions provided, together with the type of analysis carried in this review provide the basis for a deeper and expanded insight into the MNP, and for that matter, other nuclear plants.

REFERENCES

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2. Pickard, Lowe, and Garrick, Inc., Westinghouse Electric Corporation, and Fanske & Associates, Inc., Zion Probabilistic Safety Study (prepared for Commonwealth Edison Company), 1981.
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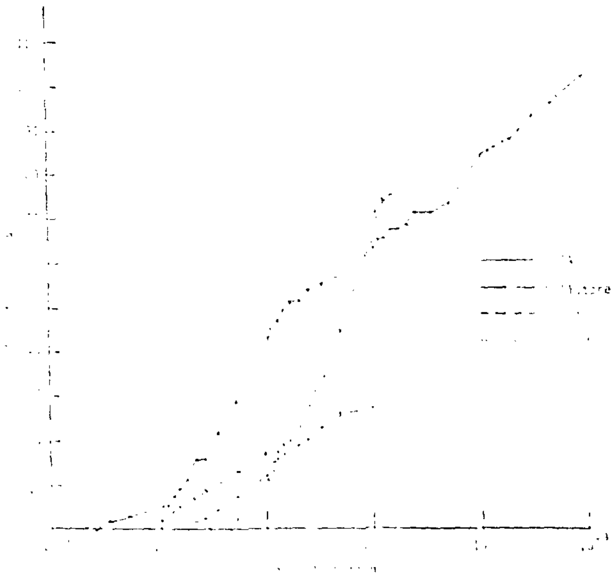

Figure [illegible]. Cumulative number of sequences as a function of their contribution to CDF.

Table 1a
Nomenclature and Frequency of Dominant Initiating Events in MPRA

Acronym	Initiating Event	Frequency*
[illegible]	Loss of offsite power	[illegible]
[illegible]	Reactor trip	[illegible]
[illegible]	Loss of component cooling water	[illegible]
[illegible]	Fire in auxiliary building	[illegible]
[illegible]	Very small LOCA	[illegible]
[illegible]	Turbine trip	[illegible]
[illegible]	Steam generator tube rupture	[illegible]
[illegible]	Small LOCA	[illegible]
[illegible]	Excessive main feedwater	[illegible]
[illegible]	Loss of main feedwater	[illegible]
[illegible]	Loss of service water	[illegible]
[illegible]	Inadvertent opening of [illegible] isolation valve	7.5E-7

*All event frequencies are in event per reactor year.

Table 1b
Definition of Dominant Plant Damage States in MPRA

[illegible]	Description
[illegible]	Late melt, at high pressure, no water in RV cavity, and no sprays
[illegible]	Late melt, at high pressure, water in RV cavity
[illegible]	Late melt, at high pressure, no water in RV cavity, no containment heat removal, and no sprays
[illegible]	Late melt, at high pressure, no water in RV cavity
[illegible]	Late melt, at high pressure, water in RV cavity, no sprays
[illegible]	Early melt, at high pressure, no water in RV cavity, no sprays
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]

[illegible]

Table 2
Number of Sequences of Certain Type
(Grouped by Initiator)

Initiator	CDF Contribution	No. of Sequences [illegible]	No. of Sequences [illegible]	[illegible]	[illegible]
AC	1.22E-4	25	[illegible]	2	12
RT	4.37-5	24	11	2	13
LC	3.97-5	2	2	1	2
F-3	1.35-5	5	3	2*	3
TT	1.71-5	13	5		5
VL	1.35-5	13	2	1	[illegible]
SL	[illegible]	2	2		2
TR	7.35-6	22	3	2*	5
EF	[illegible]	1	1		
FW	6.55-6	2	1		1
LS	3.77-6	2	1		2
VS	[illegible]	2	[illegible]	1	1
		[illegible]	3[illegible]		3[illegible]

*[illegible]

Table 3
Core Damage Frequencies as Function of Initiators and Plant
Damage States (PDS) and the Presence of Early (E) and Late (L) Fatalities

Initiators	Totals	PDS [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	Totals	[illegible] (25%)	[illegible] (21%)	[illegible] (23%)	[illegible] (17%)	[illegible] (12%)	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
AC	1.22E-4 (42%)	[illegible] L	[illegible]	[illegible] E	[illegible]	[illegible] L	--	[illegible]	[illegible]	--	--	[illegible]	--	--
RT	4.37-5 (15%)	[illegible]	[illegible] L	[illegible]	[illegible]	[illegible] L	--	--	[illegible]	--	--	[illegible]	--	[illegible]
LC	3.97-5 (13%)	[illegible] L	--	--	--	--	--	--	--	--	--	--	--	--
F-3	[illegible] (5%)	[illegible] L	--	--	[illegible]	--	--	--	[illegible]	--	--	[illegible] E	--	--
TT	[illegible] (5%)	[illegible]	[illegible]	--	[illegible]	[illegible]	--	--	[illegible]	--	--	[illegible]	--	[illegible]
VL	[illegible] (5%)	--	[illegible]	--	--	--	[illegible]	--	--	--	--	--	--	--
SL	[illegible]	--	[illegible]	--	--	--	--	--	--	--	--	--	--	--
TR	[illegible]	--	--	--	--	--	[illegible]	[illegible]	--	[illegible]	[illegible]	--	--	--
EF	[illegible] (2%)	--	[illegible]	--	--	--	--	--	--	--	--	--	--	--
FW	[illegible] (2%)	--	--	--	--	[illegible]	--	--	--	--	--	[illegible]	--	[illegible]
LS	[illegible] (1%)	--	--	--	[illegible]	--	--	--	--	--	--	--	--	[illegible]
VS	[illegible] (<1%)	--	--	--	--	--	--	--	--	--	--	--	[illegible] E	--

(Note that the totals do not quite match because the data in the table were, of necessity, obtained from the restricted number (103) of sequences cited in HRA.)

Table 4
Failure Mode Importance for the Sequences
of the Various Initiators

--	Failure Mode Importance (%)					
	H	M	R	T	C	[illegible]
AC Loss of Offsite Power (42%)	75	24	44	c	1	2
RT Reactor Trip (15%)	40	53	11	c	3	1
LC Loss of Component Cooling Water (13%)	92	7	--	1	--	--
F-3 Fire in Auxiliary Building (5%)	55	45	3	c	1	c
VL Very Small LOCA (5%)	12	5	100	--	1	--
TT Turbine trip (5%)	62	47	45	3	2	5
TR Steam generator tube rupture (2%)	81	23	--	--	13	2
SL Small LOCA (2%)	12	5	100	--	1	--
EF Excessive main feedwater (2%)	--	--	100	--	--	--
FW Loss of main feedwater (2%)	33 •	72	11	c	2	--
LS Loss of service water (1%)	100	--	--	--	--	--
VS Inadvertent opening of DH (<1%)	--	--	--	--	--	--
Total	74	27	37	<1	1	1

c - Less than 1%

Table 5
Support System Importance for the Sequences
of the Various Initiators

Initiator	[illegible]					
	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
AC Loss of Offsite Power (42%)	2	.	1	...		13
RT Reactor Trip (15%)	12	1	59	31	2	1
LC Loss of Component Cooling Water (13%)	-	-	57	41	-	-
F-3 Fire in Auxiliary Building (5%)	13	1	37	31	1	c
VL Very Small LOCA (5%)	12	3	-	-	2	.
TT Turbine Trip (5%)	2	.	43	23	.	-
TR Steam Generator Tube Rupture (2%)	53	22	1	.		-
SL Small LOCA (2%)	12	1	-	-	2	.
EF Excessive Main Feedwater (2%)	-	-	-	-	-	-
FW Loss of Main Feedwater (2%)	-	-	52	37	-	-
LS Loss of Service Water (1%)	-	-	-	-	-	-
VS Inadvertent Opening of DHR (<1%)	-	-	-	-	-	-
Total	4	[illegible]	21	50	.	4

Notes: c = Less than 1%.

In some cases the initiator itself entailed the failure of a support system or a frontline system.

In some cases there were no support system failures.

Table 6
Role of System/Function Failures in the Leading
Sequences of Various Initiators

Initiator	System/Function													
	HP	LP	V	MF	AF	EP	SR	DH	RP	MS	BW	I	OA	Total
AC	42	6	19	-	27	133	29	-	-	-	-	-	53	227
RT	60	7	71	59	91	3	22	20	-	13	-	-	42	245
LC	99	-	-	-	-	-	95	99	-	-	-	-	99	297
F3	99	-	-	28	95	-	60	90	-	-	-	-	90	434
TT	42	-	12	19	41	-	19	20	8	12	-	-	32	172
VL	-	-	-	-	-	-	-	105	-*	-	c	-	105	105
SL	-	-	-	-	-	-	-	14-	-	-	-	-	145	14b
TR	85	-	.	-	-	-	-	99	-	3	4	9	137	202
EF	51	-	51	-	-	-	-	102	-	-	-	-	153	204
FW	26	-	-	-	16	-	15	-	-	-	-	-	-	58
LS	88	-	-	-	-	-	-	-	-	-	-	-	-	63
VS	-	-	-	-	-	-	-	385		-	-	-	-	385
Totals	52	4	13	17	27	44	39	37	4	3	c	c	65	253

HP - High Pressure Injection
LP - Low Pressure Injection
V - Valves
MF - Main Feedwater
AF - Auxiliary Feedwater
EP - Electric Power
SR - Sump Recirculation (treated as a frontline system in the HRA)
DH - Decay Heat Removal
RP - Reactor Protection
MS - Main Steam
BW - Borated Water
I - Instrumentation
OA - Events involving operator action