INDEPENDENT REVIEW OF NUREG/CR-2497
"PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE
ACCIDENTS: 1969-1979 A STATUS REPORT"

SCIENCE APPLICATIONS, INC.

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1. INTRODUCTION AND SUMMARY

A. Object

In response to Science Applications, Inc. (SAI) Corporate concern about comments that were received on NUREG/CR-2497 (Ref. 1), an independent SAI review team of experienced risk assessment personnel was formed, with a charter to review the findings of the Accident Sequence Precursor (ASP) study and their bases in fact. The intention of this SAI review is to clarify the information in NUREG/CR-2497 in the light of relevant comments on that document by government, industry and other organizations. In addition, this SAI review is intended to highlight the ASP study strengths and limitations and to clarify its relevancy to risk assessment. This report is the product of the independent SAI review team and as such, should be considered as the consensus opinion of that team.

B. Scope

This SAI review analyzes NUREG/CR-2497 and comments made by the external technical community. Most of the issues examined in this report were also raised by other reviewers of NUREG/CR-2497. This SAI review is not exhaustive in nature, but rather, is in sufficient depth to support the recommendations and conclusions made herein.

C. Historical Background

The accident sequence precursor (ASP) study reported in NUREG/CR-2497 was initiated in 1979 by the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory on behalf of the

Division of Risk Analysis of the Nuclear Regulatory Commission (NRC). Under subcontract to NOAC, SAI performed much of the ASP study reported in NUREG/CR-2497 and is continuing to provide technical support for this ongoing NRC program. NUREG/CR-2497 was published in June 1982 after an extensive review, which solicited comments from various government and industry groups. After it was published, many additional reviewers offered comments and raised some concerns over the results documented in NUREG/CR-2497. In recognition of the importance of the ASP study, and the significance of some of the concerns raised over the study results, SAI established in November 1982 a group to perform an independent review of NUREG/CR-2497.

D. Key Results Summarized

The key results of this SAI review are summarized briefly below and are discussed in more detail in Sections 5 and 6.

1. Licensee Event Report (LER) Selection and Screening Process

   a. The ASP study was successful in providing a reasonable methodology for extracting a small set of potentially significant events that have occurred at nuclear power plants from the very large LER data base. This methodology screens LERs based on subjective measures of their significance.

   b. Other LER selection and screening methods could be developed and would likely yield different results. Exhaustive comparison of LER screening methods or development of an "ultimate" screening system is of questionable value, however, because of the incomplete and continually changing nature of the LER data base.

2. Initiating Event and System Failure Probabilities

a. The ASP initiating event and system failure probabilities derived from LER data represent average values for the aggregate light water reactor population (i.e., PWRs and BWRs are combined) during the period from 1969 to 1979. Because they are generic values, these data are not representative of any plant in particular; however, they are used consistently within the ASP study and should be acceptable for ranking of precursors.

   Corrective actions in response to LERs have the effect of altering the relevant data base that should be used to compute initiating event and system failure probabilities. Therefore, probabilities computed for some past period (e.g., 1969 to 1979) may have little relevance to future periods for specific equipment or systems which have undergone significant design or operations changes since that time.

   Caution should be exercised in comparing ASP data to probabilistic risk assessment (PRA) data or in using the ASP data in PRAs.

b. For the precursors judged to be significant in the ASP study, it is recommended that additional plant-specific information be considered to ensure the accurate representation of the precursor sequence.

c. A consistent methodology should be used for updating the ASP initiating event and system failure probabilities as new time periods are included in the ASP study and old data becomes inapplicable because of corrective actions that have been taken (the existing ASP methodology may contain these features).

3. Quantification and Ranking of Precursor Sequences

a. An important contribution of the ASP study is its demonstration that the potentially significant precursors identified by the prior screening process can be ranked based upon simplified, plant-specific analyses from which semi-quantitative indicators of precursor significance can be derived. Together, the screening and ranking process accounted for the majority of the work reported in NUREG/CR-2497.

b. The statistical approach, level of modeling detail, and use of generic data in the ASP study to quantify individual precursor events appears to be reasonable for ranking purposes (e.g., the ASP ranking results did not contradict expectations that events such as TMI, Browns Ferry and Rancho Seco would be among the more significant events).

c. In the ASP study, results called "conditional probabilities of severe core damage given the precursor" were calculated for specific LER events using generic data generated from the LERs. These results are measures or "indicators" which relate to the likelihood of severe core damage, but should not necessarily be construed as accurate estimates of the conditional probability of severe core damage for specific LER events. This is an historical indicator that may not apply to the same plant after changes have been made in design, operating practices or operator training.

d. Because of differences in design, operation, and training among plants, an ASP indicator calculated for a precursor at one plant may not have the same numerical value (and hence significance) at another plant. It appears reasonable, however, to use ASP screening and ranking techniques to identify potential precursors and then have individual utilities evaluate the importance of the precursor on their plants.

4. Estimate of Average Severe Core Damage Frequency per Reactor-Year

   a. In the ASP study, an industry-wide "indicator" called "estimated frequency of severe core damage per reactor-year for the years 1969 to 1979" was derived by combining the conditional probabilities of severe core damage for the relevant precursor events. The calculation of this industry-wide indicator represented only a small portion of the overall ASP effort, however, it has contributed significantly to the controversy over NUREG/CR-2497, largely because of comparisons made between the ASP indicator and PRA core melt frequencies.

      The ASP estimate of severe core damage frequency should be treated as a semi-quantitative indicator of the past performance of the nuclear industry. Because of limitations associated with generic data and modeling detail, this indicator should not be treated with confidence as an absolute estimate of severe core damage frequency per reactor-year. In contrast, the PRA process leads to an analytical estimate of severe core damage frequency for a specific plant. Therefore, comparison of a specific estimate of core melt frequency from a PRA with the ASP indicator of severe core damage frequency has no meaning of itself. However, comparison of numerical estimates and the bases for differences and agreements between such estimates may provide insights into PRA techniques and completeness.

   b. The ASP industry-wide indicator of severe core damage frequency is a measure of past performance; it should not be used for future inferences. This point is clearly made in NUREG/CR-2497. However, similarly derived indicators for different time periods (e.g., post-1979) could be compared with the ASP industry-wide indicator for the 1969 to 1979 period to bring to light trends in the industry.

5. Trend Analysis

Trend analysis was performed in the ASP study without reaching substantive conclusions. Information from a new ASP study period (e.g., post-1979) should be plotted along with information from the 1969 to 1979 period. Such a plot should show system-by-system effects of corrective actions and improvements made since the original ASP study period. This feature could provide a means for measuring, at least in qualitative terms, the contribution to safety of post-TMI efforts by the NRC and the nuclear industry.

6. Uncertainties

Uncertainties were not defined for the numerical estimates made in the ASP study. To the extent practical, a consistent approach for addressing uncertainties in system failure and initiating event probability and the indicator of severe core damage frequency should be developed in future studies of this type.

7. Application of ASP Results

The following are potential qualitative or semi-qualitative applications of the results of the ASP study.

a. The ASP study provides a mechanism for identifying potentially significant weaknesses in system/equipment design, operating practices and operator training. As part of, or separate from the ASP study, there should be a mechanism to evaluate, on a plant-specific basis, the actual significance of precursors that were identified as potentially significant in the ASP study. One approach is to submit for review at each plant all the precursors identified from the ASP initial screening process (e.g., 529 out of 19,400 LERs). The review of all 529 precursors would prevent an oversight that may occur in the case of an LER that is judged to be not significant at one plant while it might be more significant at another plant with different design features or operating practices. The level-of-effort involved in this type of review is not as great as it may appear because the 529 precursors would not apply to all plants.

b. The ASP ranking of precursors in conjunction with additional plant specific analysis should be useful for allocating resources and prioritizing corrective actions in response to significant LERs.

c. The ASP study provides input to plant-specific PRAs in the following areas:

   - Identification of sequences not previously included in PRAs

   - Identification of overlooked component/system dependencies

   - Better perception of the importance of some component failure modes

- Insight into person-machine interfaces and their roles in accident response.
  - operator error
  - operator recovery actions

d. In the future, results of the ASP study for time periods beyond 1979 will become available. With this new information, it should be possible to make meaningful comparisons with the original ASP results for the 1969 to 1979 period. In particular, expanded time-on-test plots may show new trends in initiating event and system failure probabilities. In addition, the indicator called "estimated frequency of severe core damage per reactor-year" can now be compared with a similarly derived value and a crude trend may be seen in this measure of the aggregate nuclear power industry. It may be possible to relate some of these changes to post-TMI NRC regulatory activities and thereby gain some indication of the effects of the reactor licensing process. It may be of interest to subdivide ASP data by reactor type or by NSSS vendor so that a check can be made for other meaningful trends.

2. GOALS OF NUREG/CR-2497

NUREG/CR-2497 (Ref. 1) describes work done in the general area of interpreting reactor operational data to help assess the composition and the totality of the risks from nuclear power plants. The report focuses on the 1969 to 1979 time period and includes a description and ranking of accident precursors to severe core damage. Summing and weighing the probabilities of these precursors leads to an estimated frequency of severe core damage per reactor-year for the period from 1969 to 1979.

A similar study is now in progress for the period from 1980 onward. When completed, NUREG/CR-2497 and the new study will permit comparisons between the pre- and post-Three Mile Island (TMI) periods.

3. GENERAL NATURE OF REVIEWERS CRITIQUE OF NUREG/CR-2497

Many organizations have reviewed NUREG/CR-2479 and have offered comments dealing with the following aspects of the study:

- LER Selection process
  - use of incomplete information from LERs
  - alternative ranking schemes

- Plant, system and operator modeling
  - use of functionally-based generic event trees
  - judgements regarding availability of mitigating systems
  - judgements regarding ability of plant operators to repair inoperable systems

- Data analysis and inferences
  - estimated system failure and initiating event probabilities
  - small sample set size
  - uncertainty
  - statistical methods
  - interpretation of numerical results of the study
  - trend analysis

The Institute of Nuclear Power Operations (INPO) has presented a very detailed review of NUREG/CR-2479. Significant issues addressed in the INPO review (Ref. 2) and in reviews by others are discussed further in Section 5 of this report.

4. SYNOPSIS OF NUREG/CR-2479

A. Introduction

NUREG/CR-2497 (Ref. 1) is a status report prepared by the Accident Sequence Precursor (ASP) study group covering work completed on LERs submitted during the 11-year period from 1969 through 1979. It contains material related to: (1) the development and application of selection criteria for identifying those reactor events that are potential severe core damage precursors, (2) the application of these criteria as a filter to the eleven year LER historical data base, and (3) the analysis of the selected events for their relative ranking from a risk viewpoint. For the time period from 1969 and 1979, the report integrates the data to arrive at an estimated severe core damage frequency range of from 1.7×10^{-3} to 4.5×10^{-3} per reactor-year.

B. Report Approach and Key Features

1. LER Selection and Ranking Criteria

Approximately 19,400 LER abstracts were examined, and specific LERs were chosen for further detailed review if any of the following criteria were met (see Figure 4-1):

a. any failure to function of a system that should have functioned as a consequence of an off-normal event or accident,

b. any instance where two or more failures occurred,

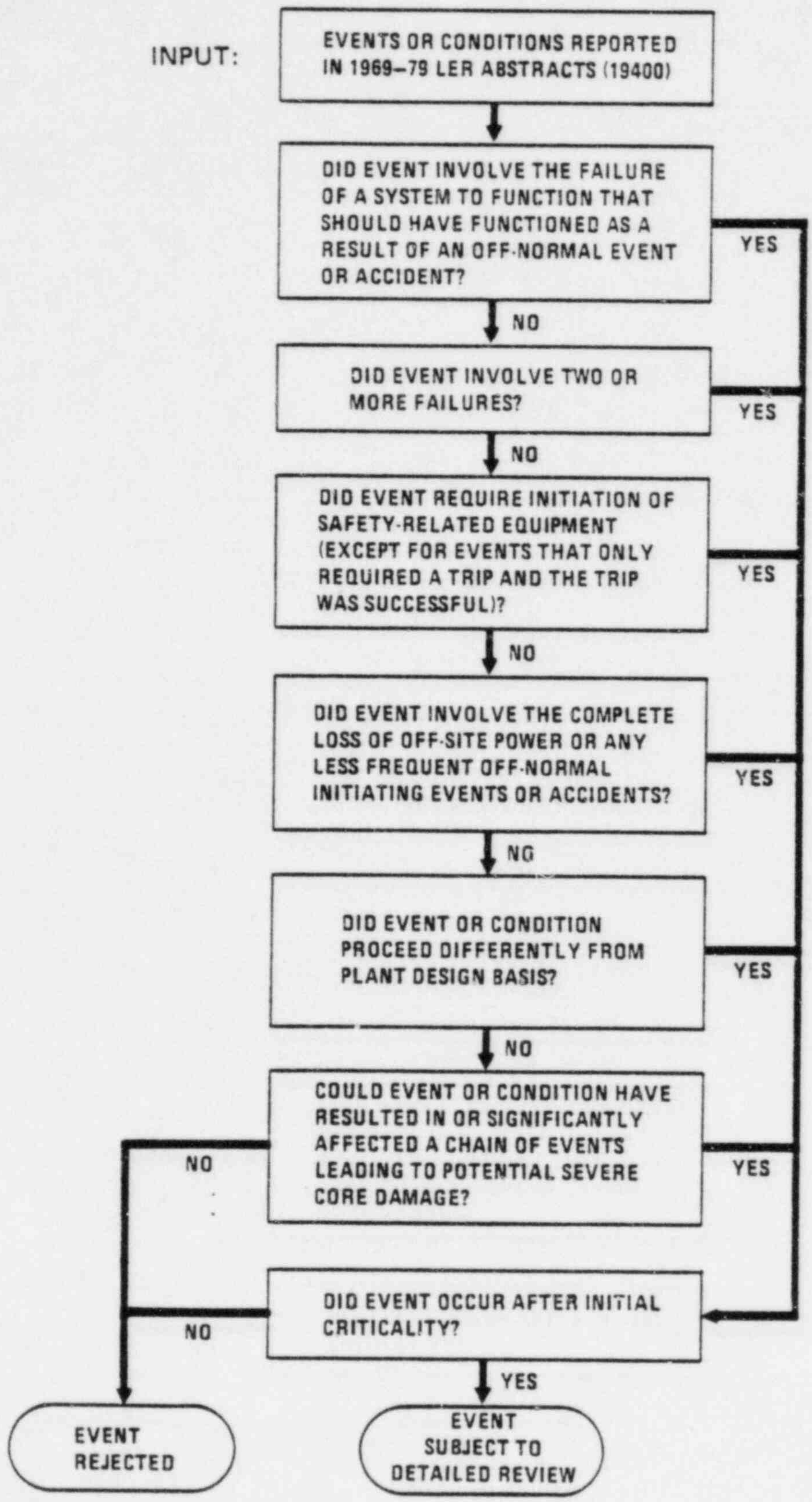


Figure 4-1. Initial LER Screening and Selection Process.

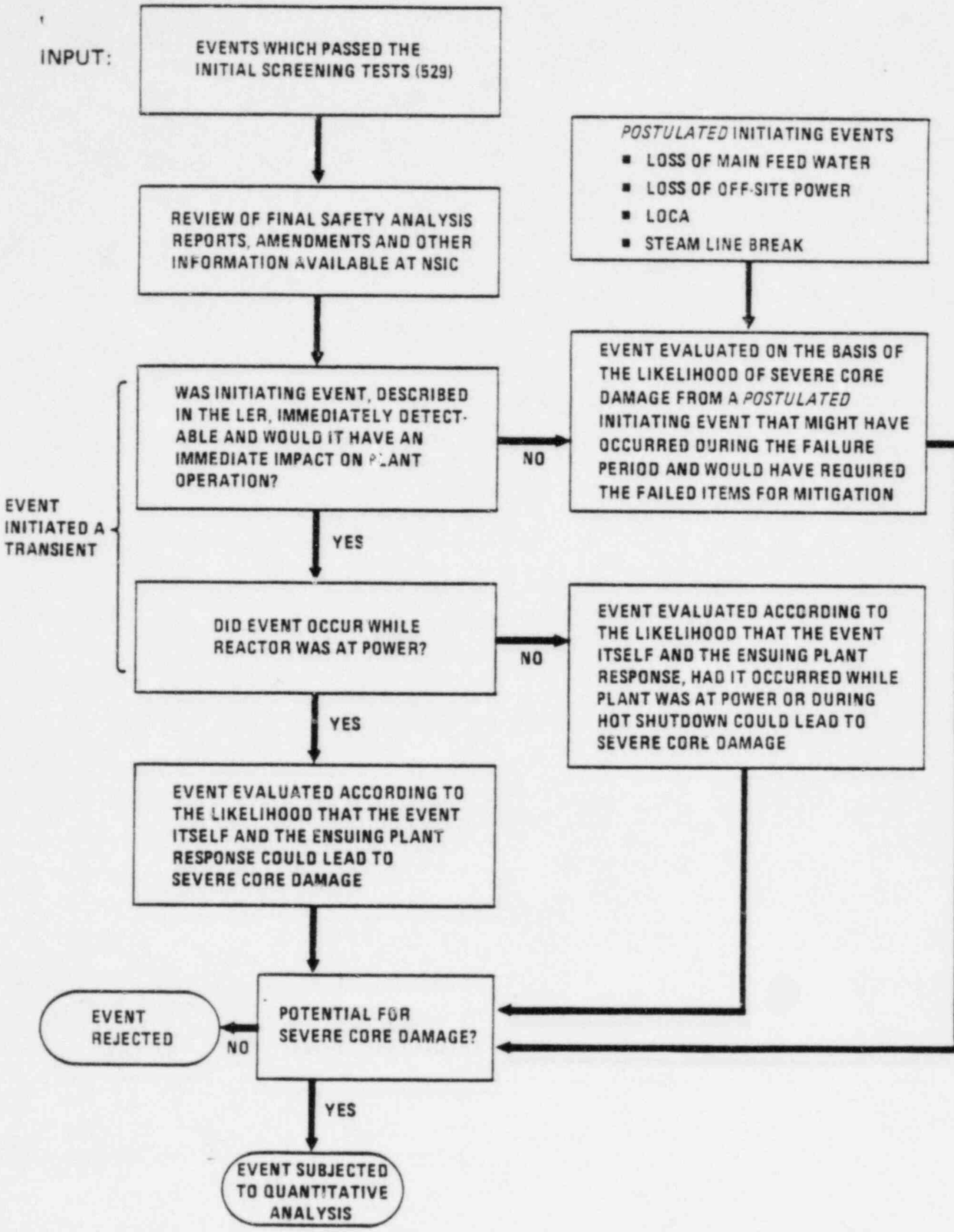


Figure 4-2. Detailed LER Review Process.

c. all events that resulted in or required initiation of safety-related equipment (except events that only required trip and when trip was successful).

d. all complete losses of offsite power or any less frequent off-normal initiating events or accidents,

e. any event or operating condition that was not enveloped by or proceeded differently from the plant design bases,

f. any other event that, based on the reviewer's experience, could have resulted in or significantly affected a chain of events leading to potential severe core damage.

Out of the 19,400 LERs, a total of 529 LERs were selected for detailed review. The detailed review of each LER was done as follows (see Figure 4-2):

- If the event was immediately detectable, it was evaluated according to the likelihood that the event and the plant response would lead to severe core damage during power operation.

- If the event did not impact plant operation, an initiator was postulated and the event was evaluated based upon the likelihood of severe core damage from the postulated initiating event and the failed items found in the LER. Postulated initiators included: (1) loss of main feedwater, (2) loss of offsite power, (3) loss-of-coolant-accident, and (4) main steam line break.

This detailed LER review reduced the number of sequences considered from 529 to 169. For each of the 169 events, an analysis of the sequence was performed. As part of each sequence analysis, the following four items were produced:

- A tabular summary description and data on the precursor.
- An event tree of the actual occurrence of the precursor.
- A standard event tree representation of the precursor.
- A tabular summary categorization of the precursor.

An example of each of these items for incident 101444 (Browns Ferry fire) is given on six pages at the end of this section (reproduced from Ref. 1).

2. Sequence Quantification

Initiating event and system failure probabilities were determined from the LER data base directly when possible. A summary of the ASP data and a comparison with WASH-1400 (Ref. 3) data is shown in Table 4-1 (from Ref. 1). The ASP data agrees within a factor of 10 with most of the WASH-1400 data listed in Table 4-1.

When routine testing found a system to be in a failed state during normal operation, the ASP study assigned a subjective estimate of the probability of restoring the system to operation. This estimate was assigned as follows:

- If the failure could be easily corrected from the control room in a short period of time, the probability of such repair was assumed to be 0.9.

- If the failure could be repaired at the failure location, the probability of system repair was assumed to be 0.5.

- If short term repair at either location was not possible, the failure rate was as determined by the data.

Standardized PWR and BWR functional event trees were constructed for the four postulated initiating events. Each LER

Table 4-1. Initiating Event Frequencies and Demand Failure Probabilities Determined using Precursor Information Compared with Values Determined in the Reactor Safety Study (from Ref. 1).

Event	Frequency or failure probability	
	ASP value	*Reactor Safety Study* value
Loss of offsite power (combined PWR and BWR) (≥30 min), per year	0.041	0.04[a]
PWR loss of offsite power (≥30 min), per year	0.048	
BWR loss of offsite power (≥30 min), per year	0.030	
PWR small LOCA, per year	$8.3 \times 10^{-\text{[illegible]}}$	$10^{-\text{[illegible]}}$ [b]
BWR small LOCA, per year	$2.1 \times 10^{-\text{[illegible]}}$	$10^{-\text{[illegible]}}$ [c]
PWR AFW failure, per demand	$1.1 \times 10^{-\text{[illegible]}}$	$3.7 \times 10^{-\text{[illegible]}}$ ($7 \times 10^{-\text{[illegible]}}$ to $3 \times 10^{-\text{[illegible]}}$)[d]
PWR HPI failure, per demand	$1.3 \times 10^{-\text{[illegible]}}$	$8.6 \times 10^{-\text{[illegible]}}$ ($4.4 \times 10^{-\text{[illegible]}}$ to $2.7 \times 10^{-\text{[illegible]}}$)[e]
PWR long-term core cooling (sump recirculation) failure, per demand	$1.2 \times 10^{-\text{[illegible]}}$	$1.3 \times 10^{-\text{[illegible]}}$ ($4.4 \times 10^{-\text{[illegible]}}$ to $3.1 \times 10^{-\text{[illegible]}}$)[f]
PWR emergency power failure, per demand	$1.8 \times 10^{-\text{[illegible]}}$	$1 \times 10^{-\text{[illegible]}}$ [g]
PWR steam generator isolation failure, per demand	$1.2 \times 10^{-\text{[illegible]}}$	
PWR HPI for steam line break mitigation (concentrated boric acid injection) failure, per demand	$2.8 \times 10^{-\text{[illegible]}}$	
BWR RCIC and HPCI failure, per demand	$3.9 \times 10^{-\text{[illegible]}}$	$7.8 \times 10^{-\text{[illegible]}}$ [h]
BWR ADS failure, per demand	$2.7 \times 10^{-\text{[illegible]}}$	$5 \times 10^{-\text{[illegible]}}$ ($3.3 \times 10^{-\text{[illegible]}}$ to $7.5 \times 10^{-\text{[illegible]}}$)[i]
BWR emergency power failure, per demand	$5.0 \times 10^{-\text{[illegible]}}$	$1 \times 10^{-\text{[illegible]}}$ [j]
BWR HPCI failure, per demand	$5.7 \times 10^{-\text{[illegible]}}$	$9.8 \times 10^{-\text{[illegible]}}$ ($6.8 \times 10^{-\text{[illegible]}}$ to $1.4 \times 10^{-\text{[illegible]}}$)[k]
BWR reactor vessel isolation failure, per demand	$3.0 \times 10^{-\text{[illegible]}}$	

[a] Ref. 1, p. I-85/86, footnote 3.

[b] Ref. 1, p. 63.

[c] Ref. 1, Sect. 5.3.4.1, p. 64.

[d] Ref. 1, Table II 5-8.

[e] Ref. 1, p. II-144.

[f] Ref. 1, p. II-176.

[g] Ref. 1, p. II-90.

[h] Ref. 1, p. 56.

[i] Ref. 1, p. II-405.

[j] Ref. 1, p. II-355.

[k] The *Reactor Safety Study* failure probabilities include a test and maintenance contribution that would not be included in numbers derived from testing. The nontest and maintenance failure probability is $1.3 \times 10^{-\text{[illegible]}}$/D (median) (Ref. 1, p. II-395).

sequence was then fit into the applicable standard event tree where possible - unique event trees were required in a few cases - and conditional probabilities for severe core damage, given the precursor event, were estimated. The 26 LERs with the highest conditional probabilities of severe core damage are listed in Table 4-2 (abstracted from NUREG/CR-2497). Summing the probabilities of precursors associated with actual initiating events and accounting in an approximate manner for multiple counting of events leads to an estimated frequency of severe core damage of 1.7×10^{-3} to 4.5×10^{-3} per reactor-year during the period from 1969 to 1979.

3. Trends Examined

Time line plots were constructed, but no distinctive conclusions or trends could be confirmed. Similarly, time-phased failures were examined without any significant trends found. It was found that approximately 1/3 of the significant LERs contained human errors as part of the sequence reported.

4. Appendices

The bulk of the report is in the form of appendices that contain the supporting information, details of each selected presursor, and details on the evaluations of the 52 significant precursor events.

Table 4-2. Conditional Core Melt Probabilities Calculated for Dominant Precursor Events.

ACCESS	DATE		ACTUAL OCCURRENCE	PLANT	PRECURSOR PROBABILITY
153164	790328	LOFW	LOSS OF FEEDWATER & OPEN PORV	TMI 2	1.0
101444	750322	LOFW	CABLE TRAY FIRE CAUSED EXTENSIVE DAMAGE	BRN FERRY 1	0.4
138830	780320	LOFW	FAILURE OF NNI & STEAM GENERATOR DRYOUT	RANCHO SECO	0.25
90421	740407	LOFW	INOPERABLE AFW PUMPS DURING PLANT SHUTDOWN	PT BEACH 1	0.025
91 76	740508	LOFW	FAILURE OF 3 AUX FWDTR PMPS TO START AT TEST	TKY POINT 3	.025
108078	751105	LOFW	INOPERABLE AFW PUMPS DURING PLANT STARTUP	KEWAUNEE	.025
133706	771221	LOFW	AUX FEEDWATER PUMPS INOPERABLE DURING TEST	DVS-BESSE 1	.025
149450	790502	LOFW	LOSS OF FEEDWATER FLOW	OYSTER CREEK	.016
63129	710324	LOOP	LOSS OF OFFSITE POWER	LACROSSE	.016
88451	740119	LOOP	LOSS OF OFFISTE POWER	HAD NECK	.013
128906	770831	LOFW	LOSS OF NO-BREAK-POWER AND FEEDWATER CONTROL	COOPER	.013
137305	780325	LOFW	LOW-LOW WATER LEVEL IN ONE STEAM GENERATOR	FARLEY 1	.013
149961	790603	LOFW	HPCI FAILS TO START GIVEN LOFW	HATCH 1	.010
61434	700717	LOOP	LOSS OF OFFSITE POWER	HUMBOLT BAY	.0079
61565	710902	LOOP	LOOP AND FAILURE OF A DIESEL GENERATOR TO LOAD	PALISADES	.0063
137918	780423	MSLB	MULTIPLE STUCK-OPEN RELIEF VALVES	TMI 2	.0063
137543	780413	LOOP	LOSS OF OFFSITE POWER WHILE SHUTDOWN	CAL CLIFFS 1	.0050
139565	780514	LOOP	LOSS OF OFFSITE POWER DURING REFUELING	ST LUCIE 1	.0050
140335	780728	LOOP	LOOP AND DIESEL GENERATOR FAILURE	BVR VALLEY 1	.0050
142462	781127	LOCA	LOSS OF VITAL INST. BUS-REACTOR TRIP	SALEM 1	.0050
85566	731119	LOOP	LOSS OF A.C. POWER CAUSE HPCI/RCIC TO FAIL	BRN FERRY 1	.0032
85738	731119	LOFW	RCIC/HPCI FAILS DURING TESTING	BRN FERRY 1	.0032
152187	790903	LOOP	SWITCHYARD LOCKOUT DUE TO CABLE DROP AT STORM	ST LUCIE 1	.0032
153810	791120	LOFW	RCIC TURBINE TRIP WITH HPCI UNAVAILABLE	BRUNSWICK 1	.0032
103002	750429	LOFW	MULTIPLE VALVE FAILURES & RCIC INOPERABLE	BRUNSWICK 1	.0025
103077	750501	LOCA	RCP SEAL FAILURE	ROBINSON 2	.0025

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 101444

Date: March 22, 1975

Title: Cable Tray Fire Causes Extensive Damage at Browns Ferry

The failure sequence was:

1. A fire started in cable spreading room (at approximately 12:20). About 20 minutes later anomalous behavior of the controls and instruments, especially in items associated with ECCS, were observed by the operator.
2. The operator scrammed the reactor at 12:51 pm.
3. Electrical boards supplying power to systems used to cool the reactor were lost, (12:54).
4. The MSIV automatically closed. This resulted in the following (1:00 pm):
   (a) The steam generated by the decay heat could not be dumped to the condenser.
   (b) The steam supply to the feedwater pump turbine was lost thus eliminating the remaining source of high pressure coolant injection.
5. Operable pressure relief valves were manually opened to depressurize the reactor to the point where the condensate booster pumps could be used to supply water, (1:20).
6. These valves later failed closed (6:00 pm), then the condensate booster pumps could no longer supply water. Only the control rod drive pumps were left operable.
7. The control of the relief valves was restored (9:50 pm) and the condensate booster pumps again supplied water.
8. Normal shutdown cooling was established at 9:10 am the next morning.

Corrective action:

1. The fire was extinguished.
2. The cables were repaired (1½ years).
3. Extensive changes were made to the cable spreading room, electrical penetrations, and insulating materials.

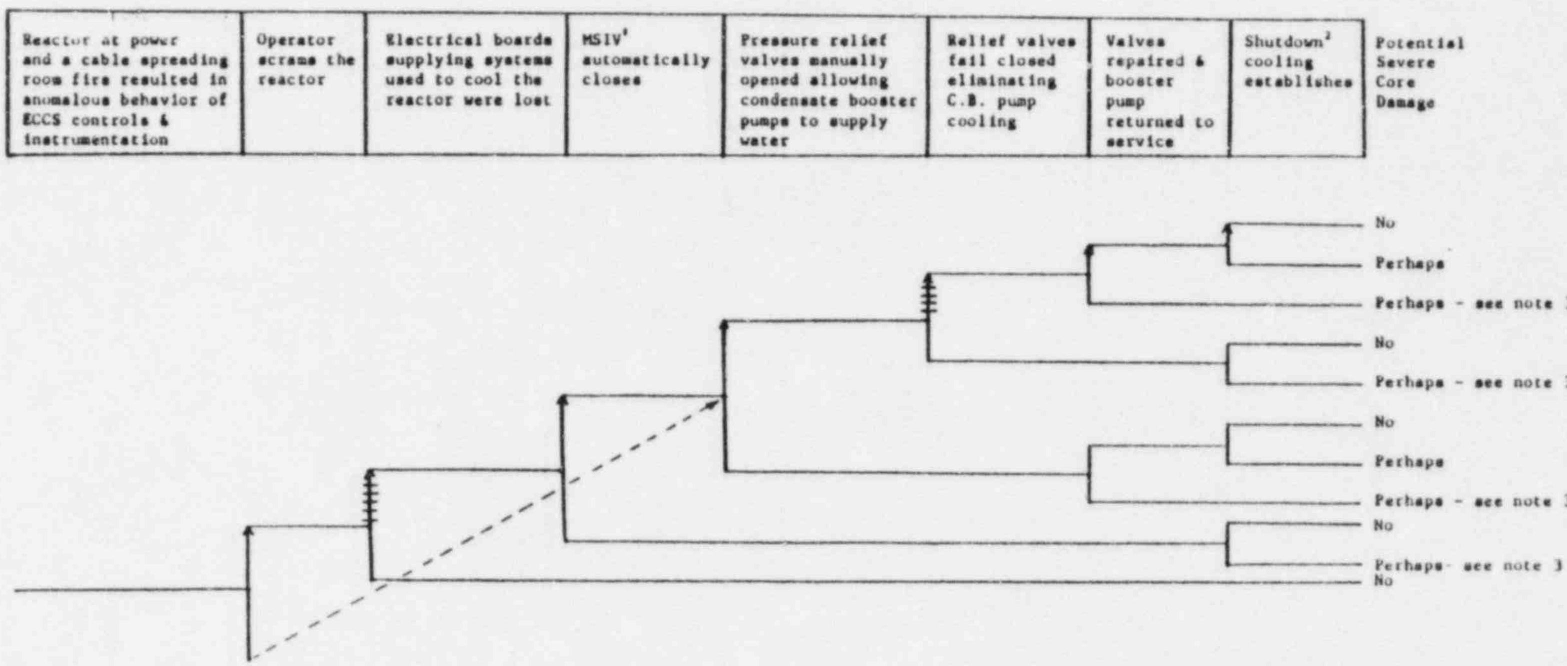
Design purpose of failed system or component:

Electric cables carry electric power and control signals to the various loads throughout the plant.

Unavailability of system per WASH 1400:* —

Unavailability of component per WASH 1400:* —

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.



NSIC 101444 – Actual Occurrence of Cable Tray Fire at Browns Ferry 1

Notes

1. Closure of the MSIV had two immediate consequences.

   (a) The steam generated by decay heat could not be passed to the main condenser, but instead remained in the reactor.

   (b) The steam supply to the feedwater pump turbine was lost thus eliminating the remaining source of high pressure cooling water.

   The MSIV is designed to fail closed.

2. The MSIV closed shortly after 1:00 pm. Shutdown cooling wasn't restored until 4:30 am the next morning. Without adequate cooling over this period of time the core would have been severely damaged.

3. See the attached sheet. This information was obtained from *Nuclear Safety*, Vol. 17, No. 5, September—October 1976, p. 607.

Note 3

AVAILABILITY OF COOLING WATER

Following a reactor shutdown, radioactive material still present in the reactor fuel continues to generate a significant amount of heat, called decay heat, that must be removed to prevent fuel damage. There are several methods of removing this decay heat (Figs. 11 and 12).

1. Steam can be passed to the main condensers and water returned to the reactor to keep the fuel covered at all times.

2. The main-steam-line isolation valves can be closed to allow the reactor temperature and pressure to increase, thus causing relief valves to open and close automatically to maintain a relatively constant pressure. The steam discharge from the relief valves passes through pipes to a large pool of water called a suppression pool. For this type of operation, the reactor water level is automatically maintained above the fuel by means of high-pressure water makeup systems.

3. Relief valves can be opened by remote control to discharge steam to the suppression pool in order to reduce reactor pressure to a low value, thus permitting the use of low-pressure water makeup systems to maintain the water level above the fuel.

4. Decay heat can also be removed when the reactor is at low pressure by pumping the water directly through the residual-heat-removal heat exchangers.

After the reactors were shut down, supplying cooling-water makeup on Unit 1 was complicated because the fire in the electrical cables had caused a number of pieces of equipment to lose some or all of their capabilities. There was adequate high-pressure water makeup available on Unit 2 at all times to keep the fuel covered. The means of heat removal throughout the course of the incident was not critical because relief valves to transfer the decay heat to the suppression pool were available.

The fire damaged the control arrangements for the main-steam-line isolation valves in Unit 1, and the valves closed and could not be reopened. The decay heat was removed for a time using automatic operation of the relief valves with the reactor remaining at high pressure. However, the fire had also affected the two primary high-pressure water makeup systems provided for maintaining water level in an emergency. Therefore the operator chose to depressurize the reactor by remote control of the relief valves and to use the low-pressure water makeup systems that were still available.

It should be pointed out that, with the reactor at high pressure, there were other alternatives for obtaining makeup water to the reactor. A few examples of other alternatives are listed below.

1. The Unit 2 control-rod-drive (CRD) pump and a shared spare CRD pump could have been used in addition to the CRD pump on Unit 1.

2. The standby liquid-control pumps could have been made available by performing a manual valve alignment, actuating two valves, and manually restoring power to the pumps.

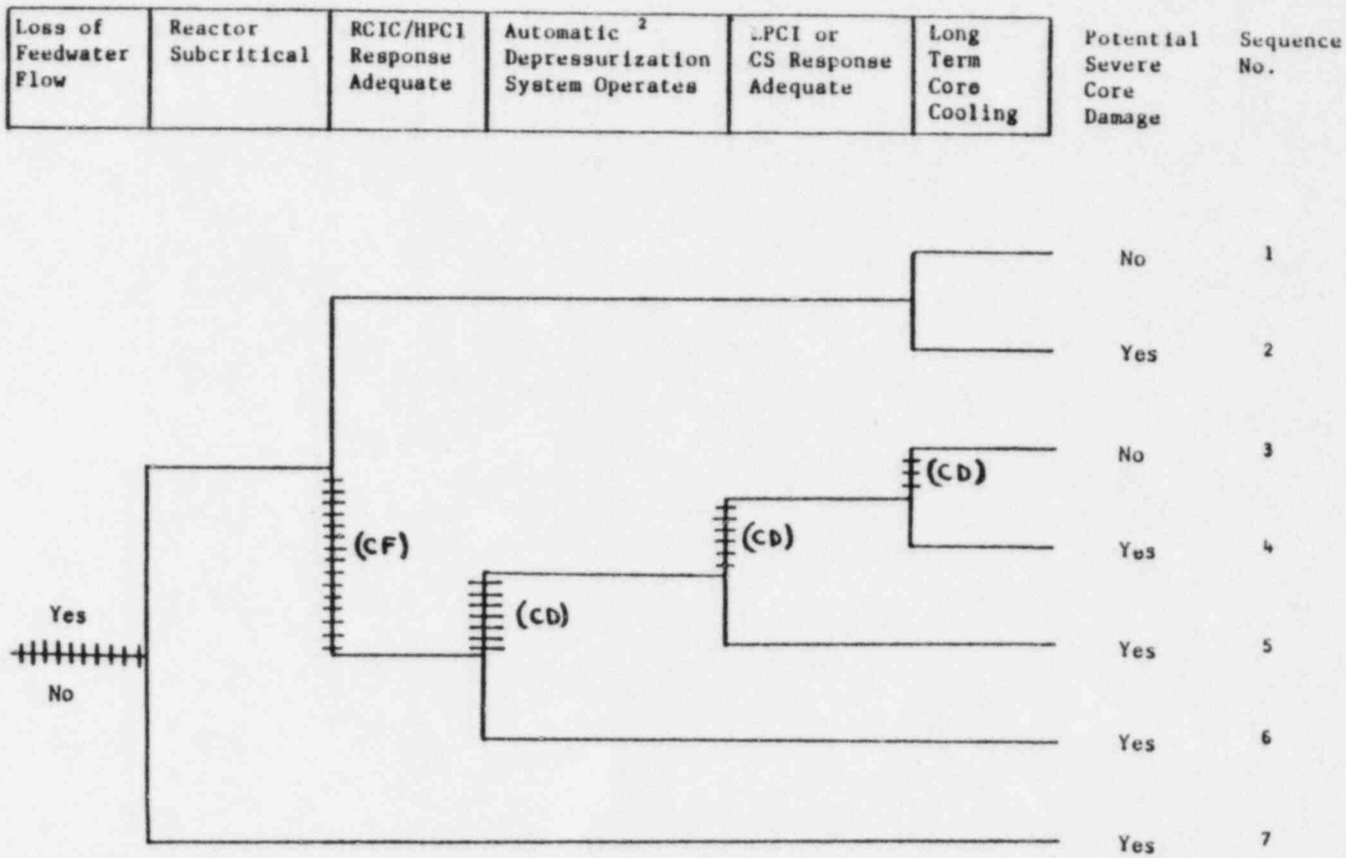
3. The reactor core-isolation cooling system could have been made available by installing a special short piece of pipe that was stored nearby.

During the period of depressurization, the water level in the core dropped but never fell below 1.22 m above the fuel. (Normal level is 5.08 m; the 1.22-m level is still 0.76 m above the level at which additional emergency cooling systems are actuated.) Once the reactor pressure was reduced below 2.4 MPa, a condensate booster pump and a condensate pump provided an adequate source of makeup water, and the normal water level was obtained. However, when stable low-pressure operation was attained, the operability of the relief valves being used to maintain low pressure was lost as a result of the loss of control air. The relief valves closed, pressure increased, and the availability of the low-pressure water makeup systems was lost. After about 3½ hr, operability of the relief valves was reestablished and low-pressure operation restored.

With low-pressure operation now secured, adequate makeup water could be supplied by one of the condensate pumps. In addition, two additional condensate booster pumps and two additional condensate pumps were available to the operator.

Another alternative would have been to use a nonstandard system configuration and manual valve alignment. Two residual-heat-removal pumps in Unit 2 could have been aligned to the Unit 1 reactor through a crosstie pipe, and, as an additional backup, river water could have been used from either of two available service-water pumps.

The point is that adequate cooling-water makeup was provided throughout the incident, and additional alternatives could have been used to provide makeup water with the reactor at either high or low pressure.



NSIC 101444 — Sequence of Interest of Cable Tray Fire at Browns Ferry 1

[1] Nonstandard techniques could have been used to make RCIC operable.

[2] The depressurization was manually initiated.

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 101444

DATE OF LER: April 1, 1975

DATE OF EVENT: March 22, 1975

SYSTEM INVOLVED: electrical distribution

COMPONENT INVOLVED: electrical cable

CAUSE: fire, human error

SEQUENCE OF INTEREST: loss of feedwater

ACTUAL OCCURRENCE: cable tray fire causes extensive damage

REACTOR NAME: Browns Ferry 1

DOCKET NUMBER: 50-259

REACTOR TYPE: BWR

DESIGN ELECTRICAL RATING: 1065 MWe

REACTOR AGE: 1.59 yr

VENDOR: GE

ARCHITECT-ENGINEERS: TVA

OPERATORS: TVA

LOCATION: 10 miles NW of Decatur, Alabama

DURATION: N/A

PLANT OPERATING CONDITION: 100% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; (d) ________________

DISCOVERY METHOD: operational event

COMMENT: —

5. DETAILED REVIEW AND CRITIQUE OF NUREG/CR-2497

The major thrust of this section is to support the key results previously stated in Section 1 as well as to provide an overview of the major topics addressed by reviewers. Where practical, the approach taken in the Accident Sequence Precursor (ASP) study is compared with comments and recommendations made by reviewers. Because of the depth of comments and documentation available in the INPO review (Ref. 2), many of the comparisons in this section are between the ASP study and the INPO review.

5.1 LER SELECTION AND SCREENING PROCESS

A. Discussion

The ASP study used a two-phase LER screening and selection process to review a large licensee event data base and to identify a relatively small set of precursors of severe core damage. The basic steps in the ASP screening process were described in Section 4 and are shown in the flowcharts in Figures 4-1 and 4-2. As shown in Figure 4-1, the initial screening phase involved the review of about 19,400 LER abstracts and resulted in the identification of 529 events for further detailed review. In the second phase of the selection process, shown in Figure 4-2, 169 precursors of severe core damage were identified. It is noted in NUREG/CR-2497 that this selection process might lead to some significant events being missed and conversely to the selection of some inappropriate or marginally significant events. The subsequent ranking process evaluates their significance.

Other methods of screening the LERs using alternative criteria could be developed. In fact, INPO applied the screening criteria from their Significant Events Evaluation and Information Network (SEE-IN) program to the 52 top-ranked events identified in the ASP study and classified 38 of these events as being "significant" and 14 as "not significant." In many cases, INPO used information beyond that which was available in the LERs and claimed that this more detailed information was necessary to resolve the significance of the events. A comparison of the ASP and INPO screening criteria is shown in Table 5-1 (from Ref. 2). As can be seen in this table, the SEE-IN and ASP screening criteria have some basic similarities. INPO classifies LERs as significant, conditionally significant or not significant. Classification of an LER as "significant" in the SEE-IN program means that "remedial measures are likely to reduce risk and improve availability." The purpose of the SEE-IN screening is therefore somewhat different than the

ASP screening which is intended to identify LERs that are precursors of severe core damage. To date, about 2.5 percent of all events screened by INPO have been categorized as significant, (Ref. 4). This is comparable to the percentage of significant events identified in the ASP study (e.g., 529/19,400 = 2.7 percent).

One reviewer of NUREG/CR-2497 proposed a selection and ranking process whereby the number of remaining operable mitigating systems (or "barriers") was considered as the primary selection criterion. The highest ranked LERs would be those in which no additional systems were available to prevent severe core damage in an event sequence of interest. Other LERs would be ranked lower based on the availability of one, two or more additional mitigating systems or functions. This selection method would likely define a set of significant LERs that is different than the sets identified in the ASP study and the INPO review.

It probably is not possible to define an "absolute" system for selection and ranking of LERs. The selection criteria are not derived mathematically, and therefore, subjective judgements are made in establishing a set of selection criteria. An individual criterion may also require that additional subjective judgements be made during actual screening of LERs. This subjectivity makes it difficult to get absolute agreement even on the application of a single set of selection criteria. In their review, for example, INPO noted that 8 of the 52 most highly ranked LERs did not appear to be significant when measured by INPO in terms of the ASP selection criteria. Had INPO applied the screening criteria to a wider set of events, it is possible that additional significant events might have been found.

What then is the value of the ASP, or any other, LER selection criteria? Their primary importance is in their ability to extract from a very large LER data base a small, manageable set of potentially significant events to be analyzed in more detail. The ASP

Table 5-1. NUREG/CR-2497 Screening Criteria and INPO Screening Guide.

NUREG CRITERIA		INPO SEE-IN SCREENING GUIDE*
1.	any failure of a system to function that should have functioned as a consequence of an off-normal event or accident	S1, S2, S3, S4, C1, C2, C4
2.	any instance where two or more failures occurred	S1, S2, S3, C2
3.	all events that resulted in or required initiation of safety-related equipment (except events that only require trip and when trip was successful)	C6
4.	all complete losses of off-site power and any less frequent off-normal initiating events or accidents	S1, S6, S7, S8, C3, C5
5.	any event or operating condition that was not enveloped by or proceeded differently than the plant design basis	S5, S6, S8
6.	any other event that, based on the reviewer's experience, could have resulted in, or significantly affected a chain of events leading to potential severe core damage	S7

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*Definitions of the INPO Screening Guide categories are shown on the chart below:

Table 5-1. (continued).

Definition of INPO Screening Guide Categories

S1. Two or more failures occur in redundant systems during the same event.

S2. Two or more failures due to a common cause occur during the same event.

S3. Three or more failures occur that would have easily escaped detection by testing or examination.

S4. Component failures occur that would have easily escaped detection by testing or examination.

S5. An event proceeds in a way significantly different than would be expected.

S6. An event or operating condition occurs that is not enveloped by the plant design bases.

S7. An event occurs that could have been a greater threat to plant safety with different plant conditions, the advent of another credible occurrence, or a different progression of occurrences.

S8. Administrative, procedural, or operational errors are committed that result from a fundamental misunderstanding of plant performance or safety requirements.

C1. A single failure occurs in a non-redundant system.

C2. Two apparently unrelated failures occur during the same event.

C3. A problem results in an off-site radiation release or personnel exposure.

C4. A design or manufacturing deficiency is identified as the cause of a failure or potential failure.

C5. A problem results in a long outage or major equipment damage.

C6. An ESF actuation occurs during an event.

C7. A particular occurrence is recognized as having a significant recurrence rate.

Note: The division of the INPO criteria into two categories, "significant" (S1 to S8) and "conditionally significant" (C1 to C7) is used for ease of screening. Events in the latter category are ultimately upgraded to "significant" or downgraded to "not significant" following additional evaluation. Events that are unique to a plant design and have no generic applicability are usually categorized as "not significant."

study was successful in developing a reasonable methodology for this purpose. Admittedly, some potentially significant events may be overlooked but this should not be considered as a fatal flaw in a screening or selection process that is being applied to an incomplete data base.

B. Conclusions

The LER selection and screening process used in the ASP study is a reasonable approach for selecting from a large data base a relatively small set of potentially significant events. The process is subjective, but this is an almost unavoidable aspect of a screening process of this type.

Other screening criteria could, of course, be developed and would likely yield results that are different from those of the ASP study, but not substantially so.

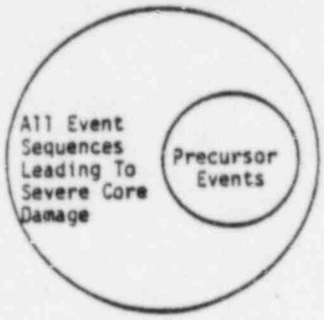
5.2 COMPLETENESS OF THE LER DATA BASE

A. Discussion

The LER data base contains information on events that actually have occurred at licensed nuclear power plants. In spite of the size of the LER data base, in terms of the numbers of events, the ASP study identified only a relatively small set of potentially significant events, and these constitute a sparce set of data from which initiating event and system failure probabilities were calculated. The calculated data are then used in the ASP study to make inferences regarding: (1) the conditional probability of severe core damage of a precursor event, and (2) the estimated frequency of severe core damage per reactor-year for the years 1969 to 1979. The LER data base must, however, be considered as an incomplete data base for identifying precursors of severe core damage because of the following considerations:

- The LER data represents the experience gained from only a limited reactor operational history (e.g., 432 reactor-years). As noted by INPO (Ref. 2) the total commercial reactor operational history in the free world was much greater at the time the ASP study was published (e.g., about 1500 reactor-years). The LER data base only includes information on U.S. reactors. Adding relevant data on non-U.S. reactors could have an effect on the results of the ASP study but should only be done in a manner consistent with the treatment of U.S. data.

- Precursor events identified in the ASP study involve significant sequences or partial sequences that actually have occurred. There are other unrelated severe core damage sequences that are predicted probabilistically in PRAs but have not yet occurred at any nuclear power plant. There also may be other significant sequences that have not yet occurred and are not yet included in PRA-type plant models (e.g., these sequences have been overlooked and therefore are not predictable probabilistically). These unrelated sequences cannot be identified from the LER data base. As shown below, precursor events constitute a subset of all event sequences that could lead to severe core damage.



- The precursor events are rare events. Statistical inferences made from such sparce data will have large uncertainties.

- LER reporting requirements may not capture all information relevant to identifying precursors of severe core damage that have actually occurred. INPO noted that LERs are "...inherently incomplete and sometimes inaccurate" (Ref. 2).

  - Events that do not involve safety systems are not reported. The probability of some initiating events such as loss of feedwater therefore may be underestimated.

  - The significance of some support system failures (e.g., instrument air, equipment room ventilation) may be overlooked, particularly when the impact on front-line systems is not straight forward. Potential sources of common mode failure may therefore not be detected.

  - Some safety system failures may not be reportable if they occur during plant operating modes when the system in question is not required to be operable by the plant

Technical Specifications. Variations in plant Technical Specifications, particularly before the advent of the NRC standard Technical Specifications, introduce nonuniformities in LER reporting requirements.

- LER reporting requirements have changed during the 1969 to 1979 period.

B. Conclusions

The LER data base is incomplete. It does not include all precursors that have actually occurred. In addition, it does not include any severe core damage sequences that may be possible but have not occurred. These considerations impose limitations on the statistical inferences that can be drawn from the numerical results of the ASP study. Since the degree of incompleteness was not quantified, and since uncertainties were not estimated, numerical estimates of past and future event probabilities based upon the ASP information must be treated with great care. The LER data base is, however, an important source of operational data which has been used in the ASP study to provide insight into severe core damage precursors.

5.3 STANDARD EVENT TREES

A. Discussion

Event trees and fault trees are tools that can be used together to model the fault logic of a nuclear power plant in response to specified initiating events. The event tree describes the response of the plant to an initiating event, and the fault trees provide detailed modeling of the systems related to each function in the event tree.

There is a trade-off that is possible between the level of detail of an event tree and the associated fault trees. Fault trees and event trees are similar logical models, and therefore, one can be made more detailed while the other is simplified (to a point) without affecting the validity of the plant model. In NUREG/CR-2497, simple event trees with broadly defined generic functions are used to describe PWR and BWR response to the following initiating events:

- Loss of main feedwater
- Loss of offsite power
- LOCA (small LOCA only for PWR)
- Main steam line break

The mitigating functions included in the PWR and BWR standard event trees are summarized in Tables 5-2 and 5-3, respectively. An individual function in these event trees may correspond to a single system or to several systems which have interrelated success criteria.

As an example, consider Table 5-4 which lists BWR plants and the systems available at each to perform the high and low pressure coolant injection, shutdown cooling and containment cooling functions (adapted from Ref. 5). In a BWR, adequate core cooling

Table 5-2. Summary of Mitigating Functions Included in Standard PWR Event Trees in NUREG/CR-2497.

Functions in Standard PWR Event Trees	Initiating Event: Loss of Main Feedwater	Loss of Offsite Power	Small LOCA	Steam Line Break
Reactor Trip	✓		✓	✓
Steam Generator Isolation				✓
Turbine Generator Runs Back and Assumes House Loads		✓		
Emergency Power		✓		
Auxiliary Feedwater and Secondary Heat Removal	✓	✓	✓	✓
PORV Demanded	✓	✓		
PORV Opened Due To Continued High Pressure Injection				✓
PORV or PORV Isolation Valve Closure	✓	✓		✓
High Pressure Injection	✓	✓	✓	✓
Long-Term Core Cooling	✓	✓		✓
Low-Pressure Recirculation and LPR/HPI Cross Connect			✓	

* PORV = Power-operated relief valve
LPR/HPI = Low pressure recirculation/high pressure injection

Table 5-3. Summary of Mitigating Functions Included in Standard BWR Event Trees in NUREG/CR-2497.

			Initiating Events			
			Loss of Feed-water	Loss of Offsite Power	LOCA	Main Steam Line Break
Functions in Standard BWR Event Trees	Reactor Subcritical	Reactor Scram			✓	✓
		Reactor Scram or SBLCS or Rods Manually Driven In	✓	✓		
	Reactor Vessel Isolated					✓
	Diesel Start and Load			✓		
	RCIC/HPCI Response Adequate		✓	✓	✓	✓
	Automatic Depressurization System Operates		✓	✓	✓(+)	✓
	LPCI or CS Response Adequate		✓	✓	✓	✓
	Long-Term Core Cooling		✓	✓	✓	✓

* SBLCS = Standby Liquid Control System
RCIC = Reactor Core Isolation Cooling System
HPCI = High Pressure Coolant Injection System
CS = Core Spray System
\+ Depressurization system only required for small LOCA.

Table 5-4. Summary of BWR High and Low Pressure Coolant Injection, Shutdown Cooling and Containment Cooling Capabilities

	BWR Type	Systems for Core Coolant Injection at High Pressure(1)					Systems for Core Coolant Injection at Low Pressure and Other Functions(1)				Systems Performing Only a Shutdown Cooling Function(1)		Systems for Containment Spray and Suppression Pool Cooling(1)	
		RCIC	HPCI	HPCS	FWCI(2)	CRDHS(2)	LPCS	LPCI	RHR-M1	RHR-M2	RHR-S	Isolation Condenser	CS-S	CS-M
Dresden 1	1	-	(4)	-	-	X	X	-	-	-	X	X	X	-
Humboldt Bay	1	-	-	-	X	X	X	X	-	-	X	X	X	-
Big Rock Point	1	-	-	-	-	X	(5)	-	-	-	X	X	-	(7)
Oyster Creek	2	-	-	-	-	X	X	-	-	-	X	X	X	-
Nine Mile Point	2	-	-	-	X	X	X	-	-	-	X	X	X	-
Millstone 1	3	-	-	-	X	X	X	X	-	-	X	X	-	(8)
Dresden 2 & 3	3	-	X	-	-	X	X	X	-	-	X	X	-	(8)
Pilgrim	3	X	X	-	-	X	X	-	X	-	-	-	-	(9)
Monticello	3	X	X	-	-	X	X	-	X	-	-	-	-	(9)
Quad Cities 1 & 2	3	X	X	-	-	X	X	-	X	-	-	-	-	(9)
Hatch 1 & 2	4	X	X	-	-	X	X	-	X	-	-	-	-	(9)
Browns Ferry 1, 2 & 3	4	X	X	-	-	X	X	-	X	-	-	-	-	(9)
Vermont Yankee	4	X	X	-	-	X	X	-	X	-	-	-	-	(9)
Peach Bottom 2 & 3	4	X	X	-	-	X	X	-	X	-	-	-	-	(9)
Cooper	4	(3)	X	-	-	X	X	-	-	X	-	-	-	(9)
Duane Arnold	4	(3)	X	-	-	X	X	-	-	X	-	-	-	(9)
Fitzpatrick	4	(3)	X	-	-	X	X	-	-	X	-	-	-	(9)
Brunswick 1 & 2	4	(3)	X	-	-	X	X	-	-	X	-	-	-	(9)
Shoreham	4	(3)	X	-	-	X	X	-	-	X	-	-	-	(9)
Fermi 2	4	(3)	X	-	-	X	X	-	-	X	-	-	-	(9)
Susquehanna 1 & 2	4	(3)	X	-	-	X	X	-	-	X	-	-	-	(9)
LaSalle 1 & 2	5	(3)	-	X	-	X	(6)	-	-	X	-	-	-	(9)
Zimmer	5	(3)	-	X	-	X	(6)	-	-	X	-	-	-	(9)
Hanford 2	5	(3)	-	X	-	X	(6)	-	-	X	-	-	-	(9)
Grand Gulf 1 & 2	6	(3)	-	X	-	X	(6)	-	--	X	-	-	-	(9)
Other BWR/5 & /6		(3)	-	X	-	X	(6)	-	-	X	-	-	-	(9)

Table 5-4. Summary of BWR High and Low Pressure Coolant Injection, Shutdown Cooling and Containment Cooling Capabilities (continued)

Notes:

(1) RCIC = reactor core isolation cooling system
HPCI = high pressure coolant injection system
HPCS = high pressure core spray system
FWCI = feedwater coolant injection system
CRDHS = control rod drive hydraulic system
LPCS = low pressure core spray system
LPCI = low pressure coolant injection system (single mode system)
RHR-M1 = multi-mode RHR system performing LPCI, shutdown cooling, suppression pool cooling and containment spray functions
RHR-M2 = same as RHR-M1 plus steam condensing operation with RCIC
RHR-S = single-mode residual heat removal system
CS-M = containment spray, which is an operating mode of some other multi-mode system
CS-S = single-mode containment spray system

(2) Non-engineered safety feature system except Millstone 1 FWCI

(3) Injection plus steam-condensing modes of operation

(4) Being installed

(5) Also performs suppression pool cooling function

(6) Only a single 100% capacity LPCS pump. Other plants typically have two 100% capacity LPCS trains

(7) Containment/suppression pool cooling is an operating mode of LPCS system

(8) Containment/suppression pool cooling is an operating mode of the LPCI system

(9) Containment/suppression pool cooling is an operating mode of the multi-mode RHR system (RHR-M1 or RHR-M2)

at low pressure usually can be provided by either a low pressure core spray (LPCS) system or by a low pressure coolant injection (LPCI) system. In many BWR plants, the LPCI function is performed by a specific alignment of a multi-mode residual heat removal (RHR) system. In Table 5-4, it can be seen that the Browns Ferry plants have reactor core isolation cooling (RCIC), high pressure coolant injection (HPCI), and control rod drive hydraulic (CRDHS) systems for performing the high pressure coolant injection function. Low pressure coolant injection is performed by an LPCS and a multimode RHR system. The varied success criteria for LPCS and LPCI systems are summarized in Table 5-5 (from Ref. 6). At Browns Ferry, adequate low pressure core cooling could be provided by two of the four LPCS pumps, or by two of the four LPCI pumps. Combinations of LPCS and LPCI pumps may also provide adequate core cooling, but are not listed in Table 5-5.

To accurately describe system interrelationships of the type described above, detailed plant-specific models are required. With detailed models, the fault trees can be quantified using component-level failure data (e.g., pump, diesel generator demand and running failure probabilities). The ASP study used an alternative approach to quantify the event trees using initiating event probabilities and system (rather than component) failure probabilities that were calculated from LER data. This approach results in a loss of resolution of the detailed plant response that may be traced in a detailed plant-specific model. However, such detailed analysis appears to be inconsistent with the goal of the ASP study to screen and rank a relatively large number of LERs.

To fit an actual event into a standard event tree, some or all of the following steps were taken in the ASP study.

- As in PRAs, engineering judgements were made regarding the mitigating systems for which credit would be taken. In some cases, credit may not be taken for some potentially available systems. The point-of-view of the analyst can affect the

Table 5-5. Summary of BWR Low Pressure Core Spray (LPCS) and Low Pressure Coolant Injection (LPCI) System Characteristics.[1]

BWR Plant	BWR Type	LPCS[2]	LPCI[2]
Oyster Creek	2	(8x) 3400 @ 110 (4/8)	-
Nine Mile Point	2	(8x) 1700 @ 113 (2/8)	-
Millstone 1	3	(2x) 3600 @ 90 (1/2)	(4x) 2500 @ 165 (2/4)
Dresden 2 & 3	3	(2x) 4500 @ 90 (1/2)	(4x) 2675 @ 200 (2/4)
Pilgrim	3	(2x) 3600 @ 120 (1/2)	(4x) 4800 @ 20 (3/4)
Monticello	3	(2x) 4500 @ 145 (1/2)	(4x) 4000 @ 20 (2/4)
Quad Cities 1 & 2	3	(2x) 4500 @ 90 (1/2)	(4x) 4830 @ 20 (2/4)
Hatch 1 & 2	4	(2x) 4625 @ 132 (1/2)	(4x) 7700 @ 20 (2/4)
Browns Ferry 1, 2 & 3	4	(4x) 3125 @ 122 (2/4)	(4x) 9999 @ 20 (2/4)
Vermont Yankee	4	(2x) 3000 @ 120 (1/2)	(4x) 7200 @ 20 (2/4)
Peach Bottom 2 & 3	4	(4x) 3125 @ 122 (2/4)	(4x) 10000 @ 20 (2/4)
Cooper	4	(2x) 4500 @ 115 (1/2)	(4x) 7700 @ 20 (1/4)
Duane Arnold	4	(2x) 3020 @ 113 (1/2)	(4x) 4800 @ 20 (3/4)
Fitzpatrick	4	(2x) 4625 @ 120 (1/2)	(4x) 7710 @ 290 (3/4)
Brunswick 1 & 2	4	(2x) 4725 @ 100 (1/2)	(4x) 4100 @ 246 (1/2)
Shoreham	4	(2x) 4725 @ 121 (1/2)	(4x) 7700 @ 20 (2/4)
LaSalle 1 & 2	5	(1x) 6350 @ 122 (1/1)	(3x) 7450 @ 20 (2/3)
Zimmer	5	(1x) 4625 @ 119 (1/1)	(3x) 5050 @ 20 (2/3)
Grand Gulf 1 & 2	6	(1x) 7000 @ 122 (1/1)	(3x) 7450 @ 20 (2/3)
Perry 1 & 2	6	(1x) 6000 @ 122 (1/1)	(3x) 7100 @ 20 (2/3)

Notes:

(1) Abstracted from NUREG/CR-2069, "Summary Report on A Survey of Light Water Reactor Safety Systems," with modifications and additions from other safety analysis reports.

(2) System characteristics are listed as follows: the number in the first set of parentheses indicates the number of pumps (e.g., "4x" means four pumps), the fraction in the second set of parentheses is the system success criteria (e.g., "2/4" means that two of four pumps are required for the system safety function to be successful). The data between the two sets of parentheses includes the capacity of each pump (in gallons per minute) at a specified discharge head (in psid). The code "(4x) 7700 @ 20 (2/4)" is therefore read as follows: four pumps rated at 7700 gpm at 20 psid, two-of-four pumps are required.

choice of systems to be included in the event model, and contributes to a systematic bias of the results of the study.

- A simplified event tree was developed for the actual LER event.

- This simplified event tree was translated into the appropriate standard event tree.

The standard event tree representation of the LER may not be entirely accurate because of the loss of resolution associated with generic modeling and because subjective engineering decisions may overestimate or underestimate the mitigating effects of plant systems. To illustrate this point, the Browns Ferry fire (event 10144 in the ASP study) is reviewed below.

In the ASP evaluation of the Browns Ferry fire, the high pressure coolant injection function was assumed to be failed with a probability of 1.0 as a consequence of the fire. As shown in Table 5-4, high pressure coolant injection at Browns Ferry can be provided by the RCIC, HPCI, and CRD systems and additionally by the Standby Liquid Control (SLC) system. The RCIC, HPCI, and SLC systems were inoperable due to effects of the fire. Throughout the fire, one control rod drive pump was available and was used as a source of high pressure makeup. One CRD pump can provide about 100 gpm makeup at an RCS pressure of 1080 psig. It has been calculated that a makeup rate of about 218 gpm at 40 minutes after shutdown would have been adequate to prevent the Browns Ferry core from ever being uncovered (Ref. 7). The following options for increasing the high pressure coolant injection capability using existing features of the CRD hydraulic system were available to the plant operators, but were not used:

- Increase the output of the operating CRD pump to about 225 gpm by manually valving-in a pump test bypass line which provides an additional flow path. This realignment could have been accomplished with a single valve. Adequate core coolant inventory could then have been maintained and depressurization would be unnecessary.

- Align the shared, spare CRD pump and run it in parallel with the operating pump. This makeup capability would not be enough to match the rate of coolant boiloff at 40 minutes after reactor shutdown, but would buy extra time (about 40 to 50 minutes) for repair activities to restore other systems to operation.

- Cross-connect Unit 1 and Unit 2 CRD hydraulic systems and use all three CRD pumps (1 per unit and 1 shared) to supply makeup to Unit 1. This system alignment would have provided an adequate coolant makeup capability.

It appears, therefore, that an adequate high pressure coolant injection capability could have been provided by the CRD pumps, which were available throughout the Browns Ferry fire.

The INPO review (Ref. 2) models the loss of high pressure injection (HPI) function using the fault tree in Figure 5-1 which was taken from WASH-1400 (Ref. 3) This model includes the CRD system as well as repair activities that were in progress for restoring the RCIC and/or the SLC systems. If restored, the RCIC system would have provided an adequate coolant makeup capability. The SLC system would have provided an additional 56 gpm of makeup per pump.

An event tree illustrating an even greater range of system options related to the single event "RCIC/HPCI Response Adequate" in the ASP standard event tree is shown in Figure 5-2. This event tree was developed based on the description of the Browns Ferry fire in References 7 and 8, and includes some system options not found in the fault tree in Figure 5-1.

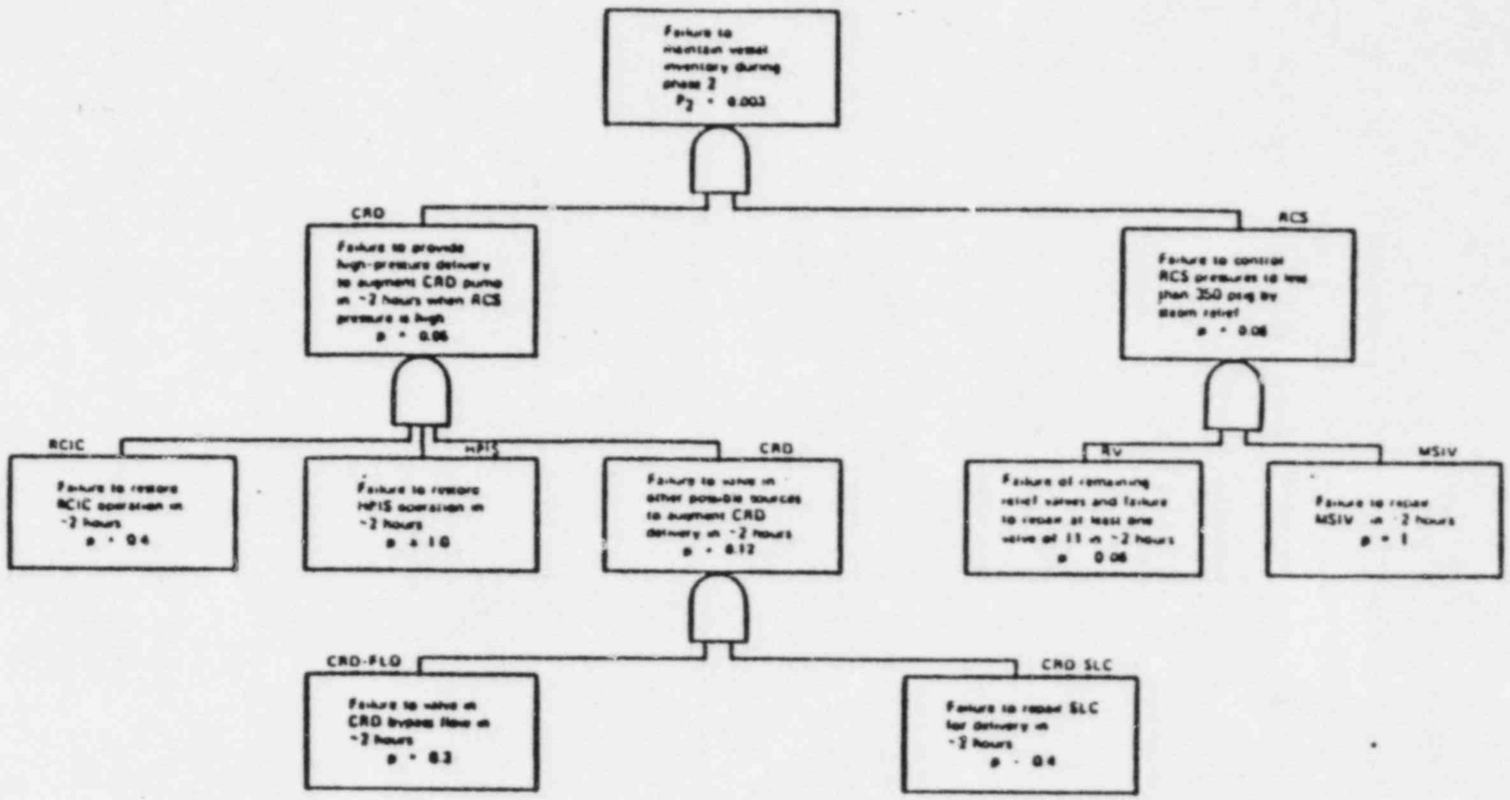


Figure 5-1. Failure Possibilities for the Browns Ferry Fire (from Refs. 2 and 6).

have been accomplished with a single valve. Adequate core coolant inventory could then have been maintained and depressurization would be unnecessary.

- Align the shared, spare CRD pump and run it in parallel with the operating pump. This makeup capability would not be enough to match the rate of coolant boiloff at 40 minutes after reactor shutdown, but would buy extra time (about 40 to 50 minutes) for repair activities to restore other systems to operation.

- Cross-connect Unit 1 and Unit 2 CRD hydraulic systems and use all three CRD pumps (1 per unit and 1 shared) to supply makeup to Unit 1. This system alignment would have provided an adequate coolant makeup capability.

It appears, therefore, that an adequate high pressure coolant injection capability could have been provided by the CRD pumps, which were available throughout the Browns Ferry fire.

The INPO review (Ref. 2) models the loss of high pressure injection (HPI) function using the fault tree in Figure 5-1 which was taken from WASH-1400 (Ref. 3) This model includes the CRD system as well as repair activities that were in progress for restoring the RCIC and/or the SLC systems. If restored, the RCIC system would have provided an adequate coolant makeup capability. The SLC system would have provided an additional 56 gpm of makeup per pump.

An event tree illustrating an even greater range of system options related to the single event "RCIC/HPCI Response Adequate" in the ASP standard event tree is shown in Figure 5-2. This event tree was developed based on the description of the Browns Ferry fire in References 7 and 8, and includes some system options not found in the fault tree in Figure 5-1.

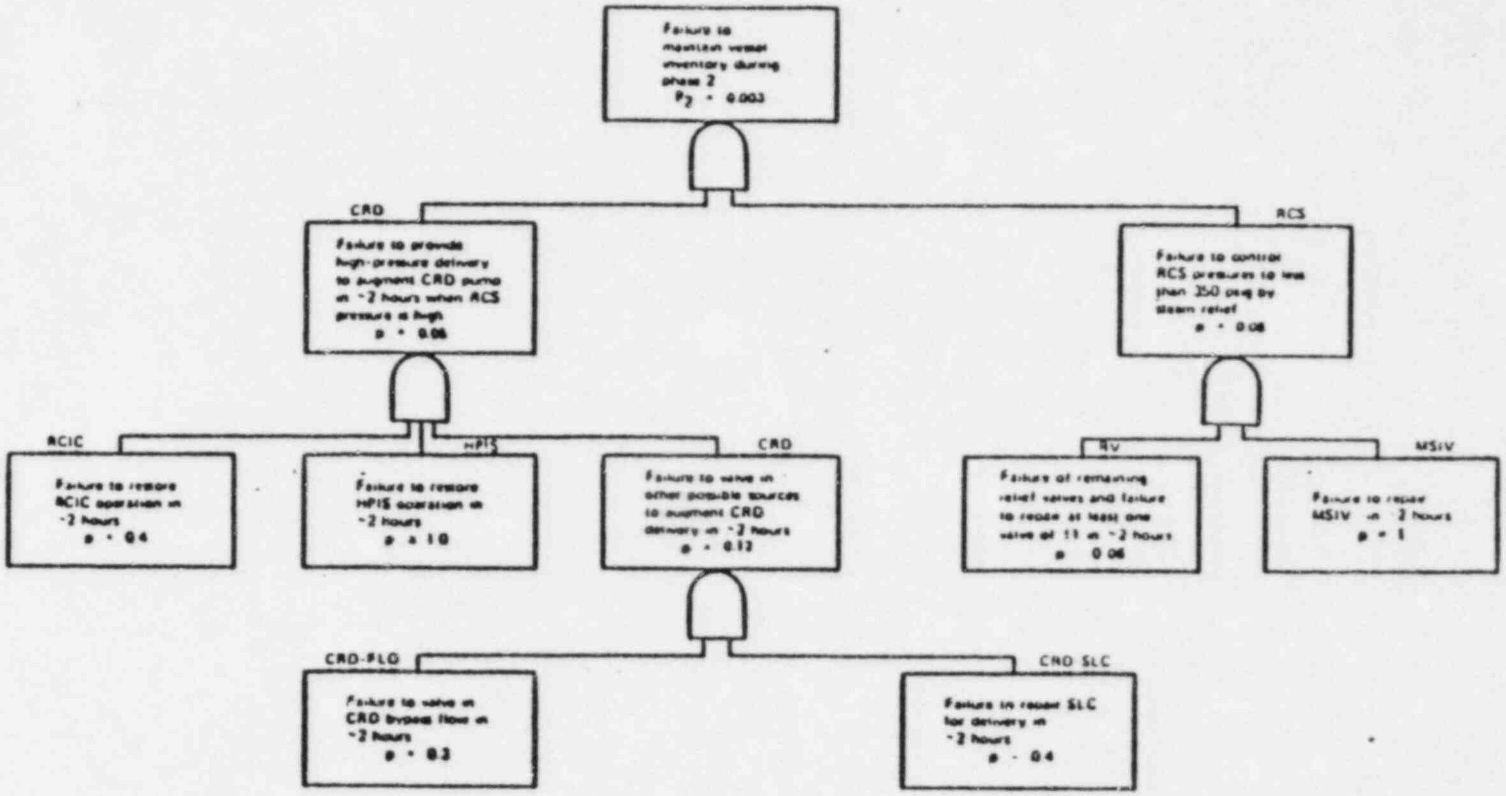


Figure 5-1. Failure Possibilities for the Browns Ferry Fire (from Refs. 2 and 6).

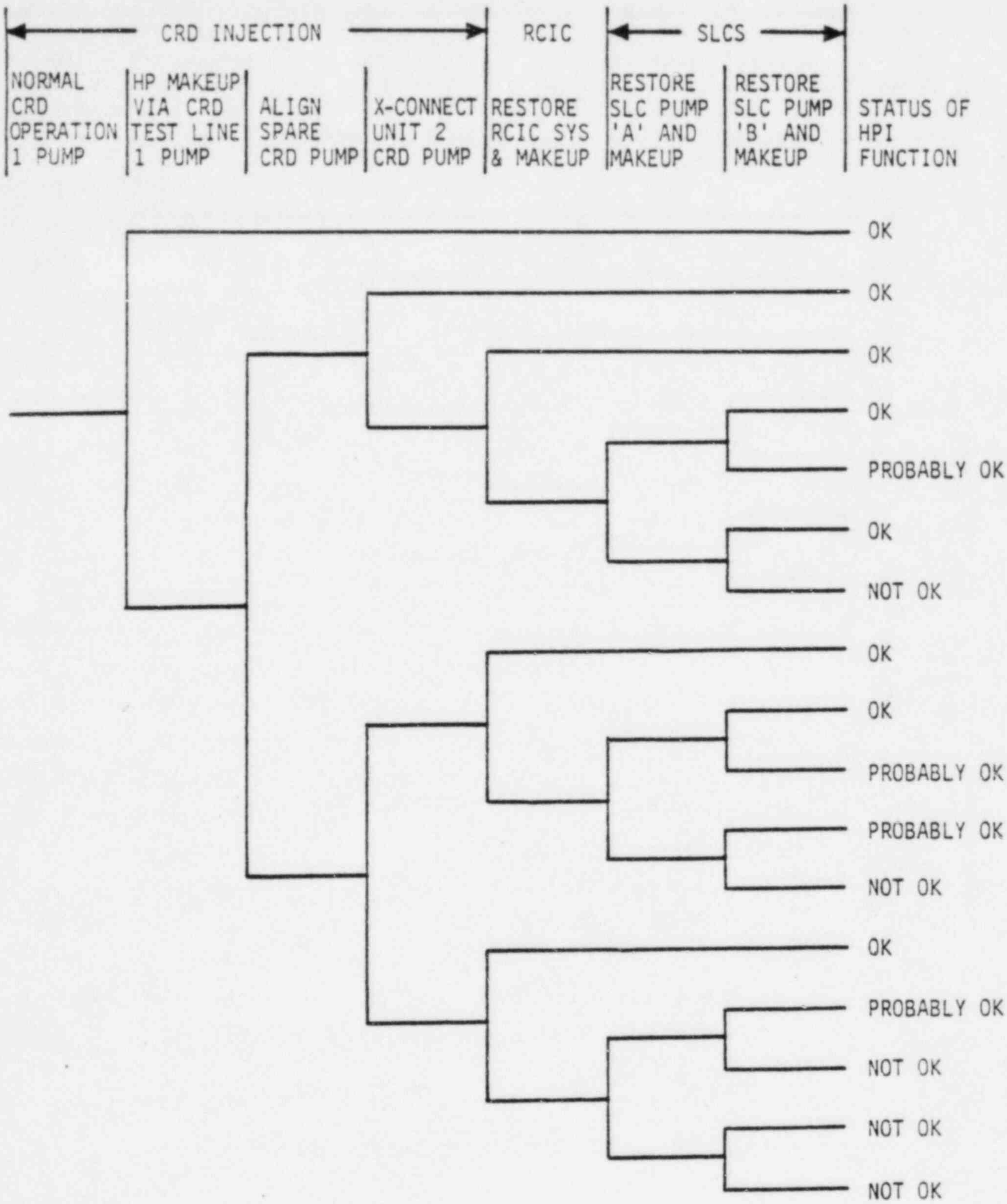


Figure 5-2. Example Event Tree for High Pressure Injection Function at Browns Ferry.

Thus, we have seen three levels of modeling detail for the same function in a standard event tree. A very simple, conservative assumption of high pressure injection (HPI) unavailability is used in the ASP study. The INPO review references the WASH-1400 fault tree shown in Figure 5-1 to justify a lower probability of HPI unavailability. An even more detailed representation of possible options for restoring the HPI function is shown in Figure 5-2, and this model could be used to calculate yet another failure probability for the HPI function. The important point is that the numerical estimates of function failure probability are affected by the depth of the supporting system model, and, of course, by the data used to quantify the model.

The RCS depressurization capability was lost during the Browns Ferry fire as a result of loss of control air and depletion of the air supply in the accumulators for the ADS valves. A solenoid valve had failed closed in the common air supply line serving the ADS valves. Manual actions were necessary to bypass the failed valve and reestablish control air. It has been noted that the bypass action took 3.5 hours to accomplish (Ref. 7). The loss of depressurization function was not modeled in detail in either the ASP study or the INPO review. A relatively simple fault tree representation for loss of the depressurization function is shown in Figure 5-3. Each event in this fault tree could, of course, be modeled in much greater detail.

Once the depressurization capability was restored, the following pumps were available and capable of performing the low pressure injection (LPI) function:

- Condensate and condensate booster pumps
- Two of four LPCI pumps
- Two of four LPCS pumps
- RHR service water pumps

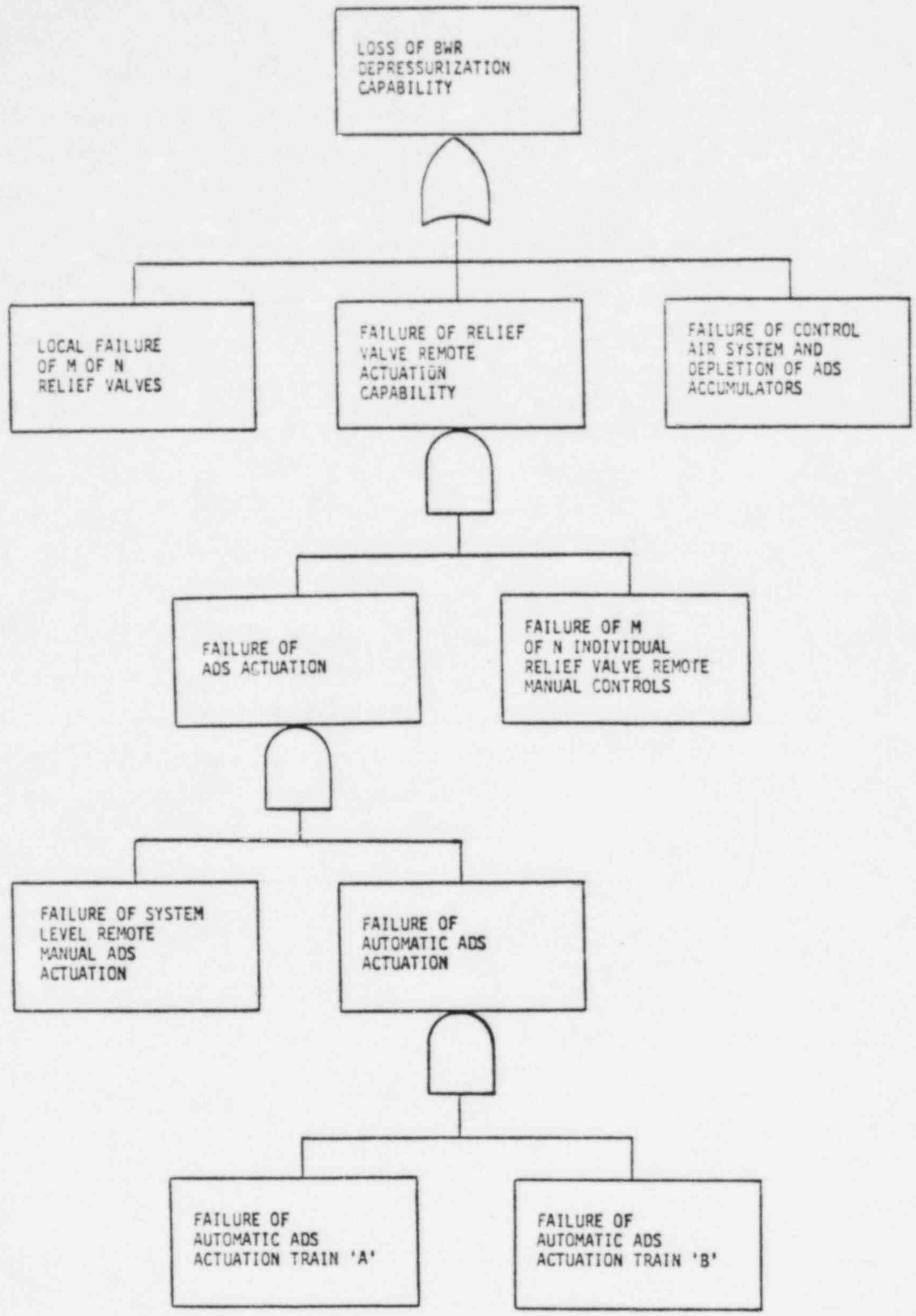


Figure 5-3. Example Fault Tree for Loss of Depressurization Capability at Browns Ferry.

The condensate and booster pumps depend on offsite power, but the other pumps can be supplied from the diesels. A detailed event tree or fault tree of loss of the LPI function could be developed, and would provide the basis for an estimate of the probability of loss of the LPI function (within the limitations imposed by the available data). A detailed model was not used in either the ASP study or the INPO review.

B. Conclusions

Use of standard functional event trees without supporting plant-specific models inherently limits the detail with which response of a plant can be modeled, and hence the accuracy of numerical results derived from the model for that plant. Ranking of significant LERs does not require a high degree of precision in the probability estimates. The consistent application of standard event trees and associated engineering judgements may, in fact, aid ranking LERs. Numerical results derived from the simple models used for ranking LERs should not, however, be considered as comparable to the numerical results from more sophisticated analyses such as WASH-1400 or plant-specific PRAs.

5.4 IMPLICIT ASSUMPTIONS ASSOCIATED WITH STATISTICAL ANALYSIS

A. Discussion

The statistical analysis is based on the following assumptions, some of which may not be entirely valid. Other aspects of the statistical analysis are discussed in Sections 5.5 to 5.7.

1. For the purpose of determining initiating event and system failure probabilities, precursors are treated as if they are each applicable to the entire population of reactors in the U.S. This assumption is not true in general. Failures and/or events may be specific to a reactor type, a vendor, or to an individual plant. Because the experience is aggregated, probability measures for events that affect only a subset of plants will tend to be underestimated in comparison to events that affect a majority of the operating reactor population. For example, auxiliary feedwater system failure probability was calculated with respect to the total years of reactor operating experience rather than just PWR experience. One approach to minimizing this problem is to sort the LERs into groups of similar plants (e.g., sort by reactor type or NSSS vendor). This approach should make it easier to identify vendor-specific problems and to make comparisons using precursor data more meaningful, however, this reduces the sample size and aggrivates the problems associated with small samples.

2. Plant designs and components are assumed not to change with time. Clearly, modifications have taken place in light of the observed failures. In a sense, the reporting of failures and/or conditions together with the subsequent corrective or preventive actions represents a very dynamic evolutionary process which plays an important role in the continuing improvement of nuclear safety.

3. An important assumption that has been made is that the precursor events occur randomly in time in accordance with a Poisson process. The following are the assumptions associated with a Poisson process:

   a. The number of events that occur during non-overlapping time intervals are independent random variables,

   b. The pro ability that an event will occur in a small interval of time, say (t, t + delta t), depends upon the length of the interval and not on the location of the interval in time,

   c. The probability that an event will occur in the interval (t, t + delta t) becomes proportional to the length of the interval, as delta t gets very small,

   d. The probability that two or more events occur in the interval (t, t + delta t) is negligible.

   It is noted that the ASP study did investigate the validity of the Poisson postulates. Although conclusive evidence for validity of these assumptions was not found, no evidence of their invalidity was identified.

   The above assumption that the precursor events represent the Poisson process gives rise to an exponential (constant failure rate) distribution of the time between successive events. Such an assumption is typically made in reliability/availability modeling in order to ensure that the analysis is mathematically tractable. This does not imply that such an assumption is valid. In fact, it is contrary to experience with mechanical systems subject to infant mortality and/or wearout mechanisms. The average failure rate will tend to be an overestimate in "mature" plants and may be an underestimate in the newer and older plants.

4. It is assumed that statistical estimates for event probabilities can be combined with subjective probability estimates to yield an estimate for the combined event.

B. Conclusions

The assumptions discussed in this section appear to be acceptable for a ranking of events, but the acceptability of these assumptions as a basis for making estimates of actual severe core damage probability is subject to question.

5.5 DATA FOR QUANTIFICATION OF THE STANDARD EVENT TREES

A. Discussion

1. Initiating Event and System Failure Probabilities

As discussed previously, initiating event and system failure probabilities were calculated from data for the 169 significant LERs identified in the ASP study. The system failure probabilities were based largely on demand failures observed during testing, and the numerical results derived in this manner did not include the following potential contributors to equipment unavailability, both of which would tend to increase the failure probabilities derived from LER data:

- Maintenance (unless actual demand occurred during the maintenance period)
- Failure to run for extended periods of time.

Initiating event and system demand failure probabilities used in NUREG/CR-2497 and a comparison with WASH-1400 (Ref. 3) values are shown in Table 4-1. The ASP values are point estimates and do not include uncertainties, whereas the WASH-1400 numbers are median values, usually stated with uncertainties.

INPO noted that the ASP failure rate data were derived in many instances from events in which: (1) the total function of a system was not lost, or (2) failure was discovered as part of post-maintenance testing (Ref. 2). Because of these effects, it was stated that the generic failure probabilities for certain key mitigating systems "appear to be very conservative" (Ref. 2). The PWR auxiliary feedwater (AFW) system and the BWR depressurization system were two systems for which INPO presented alternative system reliability estimates, as listed in Table 5-6. As can be seen in this table, the NUREG and INPO estimates differ by approximately one order of magnitude, and both are stated without confidence intervals.

The design details of particular systems vary from plant to plant and these design differences can have a significant impact on estimated and actual system reliability. This point was demonstrated in NUREG-0611 (Ref. 9) for PWR auxiliary feedwater (AFW) systems. The results of the NRC reliability evaluation, which took into account the plant-specific details of AFW system design are summarized in Figure 5-4 (from Ref. 9). Results in terms of relative AFW system estimated reliability for many Westinghouse plants are depicted in this figure. The reliability evaluations considered the following three transient conditions for which AFW system operation would be required:

- Loss of main feedwater (LMFW)
- LMFW plus loss of offsite power (LOOP)
- LMFW plus loss of all AC power

Subjectively, Figure 5-4 illustrates that AFW system reliability estimates range from "low" to "high". On a more quantitative basis, this range depicts differences of as much as two orders of magnitude in the estimated reliability of AFW systems. The difference between the NUREG and INPO reliability estimate for a "generic" AFW system seems, therefore, relatively small considering the range of plant-specific AFW system reliability estimates shown in Figure 5-4. It is interesting to note that more than half of the LERs used in the ASP study to calculate AFW and secondary heat removal failure probability occurred at Westinghouse plants that were ranked in NUREG-0611 as having rather

Table 5-6. Comparison of System Failure Probabilities.

Mitigating System	NUREG*	INPO*	WASH-1400**
PWR Auxiliary Feedwater System	1.1×10^{-3}	2.15×10^{-4}	3.7×10^{-5}
BWR Depressurization System	2.7×10^{-2}	3.0×10^{-3}	5×10^{-3}

* Point estimate demand failure probability from LER data.

** Median value demand failure probability with following uncertainty bounds:

- AFW: 7×10^{-6} to 3×10^{-4}
- ADS: 3.3×10^{-3} to 7.5×10^{-3}

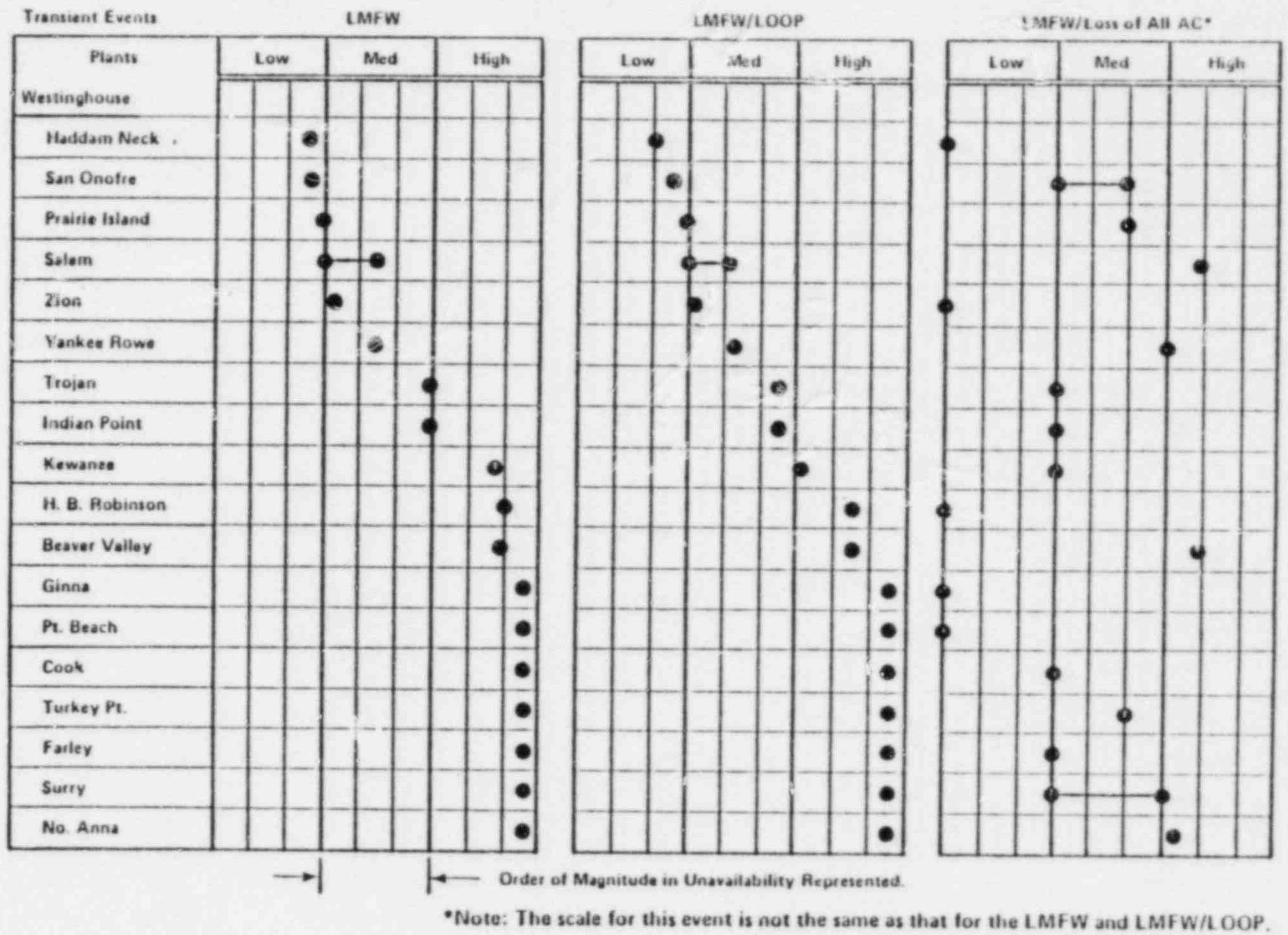


Figure 5-4. Reliability Characterizations for AFWS Designs in Plants Using the Westinghouse NSSS.

reliable AFW systems (see Table 5-7 and compare with plants listed in Figure 5-4).

Similar plant-specific reliability estimates are not readily available for the BWR depressurization system or for other mitigating systems. It is reasonable, however, to expect that plant-specific reliability estimates for these other systems would also be distributed over a significant range. The ASP point estimates of system failure probability do not provide a measure of the uncertainty or variability in the value and this information is important to the meaningful interpretation of the ASP measures of severe core damage probability.

A further point of interest in Figure 5-4 is that system reliability can be improved significantly (e.g., by up to an order of magnitude) when dominant system failure modes are eliminated or reduced. Once improvements are made, the plant no longer "looks" the same as it did before. The effects of such improvements are excluded from the ASP study, but should result in lower system failure probability and/or initiating event probability in the future. The 11-year period covered by the ASP study is long enough that any improvements made as a result of events early in the study period will adversely affect the applicability of the failure data in the latter part of the study period, and certainly beyond 1979.

Some precursor sequences involve systems that were determined to be unavailable by testing during normal operations. In the ASP study, it was assumed that the average period of time a system was unavailable could be estimated by one-half of the normal test interval.

Table 5-7. ASP Data for Estimation of AFW and Secondary Heat Removal Function Failure (from NUREG/CR-2497).

Initiating event/function under consideration	Event description: NSIC accession number	Event date	Plant	Plant age (d)	Description	Severity	Total number of events × severity
						PWR loss of main feedwater	
Initiating event	Initiating event frequency for a PWR LOFW was determined based on a review of main-feedwater-related events reported in NUREG-0611 (pp. II-15 through II-27). Eighty LOFW initiating events occurred over a three-calendar-year period. Approximately 50% appeared to be rectifiable in the short term. Westinghouse LOFW events were used for this estimate because the majority of PWR operating hours are contributed by Westinghouse plants.						40
Reactor trip failure	Demand failure probability for failure to trip was assumed to be equal to value calculated in WASH-1400 (p. II-97).						
AFW and secondary heat removal failure given trip success	81523	June 18, 1973	Turkey Point 4	7	Failure of pumps to auto-start due to failure to install fuses	0.1	6.1
	90421	Apr. 7, 1974	Point Beach 1	1252	Failure to deliver flow due to clogged suction stainers	1.0	
	91676	May 8, 1974	Turkey Point 3	565	Failure of pumps to start due to overtightened packings and controller malfunction	1.0	
	108078	Nov. 5, 1975	Kewaunee	608	Failure to deliver flow due to clogged suction strainers	1.0	
	133706	Dec. 11, 1977	Davis-Besse 1	121	Loss of AFW pump control due to mechanical binding and blown control power fuses	1.0	
	137305	Mar. 25, 1978	Farley	228	Failure of turbine-driven pumps to start plus open bypass valves	0.5	
	138830	Mar. 20, 1978	Rancho Seco	1281	Failure of AFW to deliver flow due to NNI failure	1.0	
	153164	Mar. 28, 1979	TMI-2	365	Failure of AFW to deliver flow due to closed valves	0.5	
AFW and secondary heat removal failure given failure to trip	Demand failure probability for failure of AFW and secondary heat removal given failure to trip assumed to be 10 times failure probability calculated for AFW and secondary heat removal given trip.						

2. Severity Factors

As discussed in Section 4, three different probabilities of not restoring a failed system are defined in the ASP study, depending on the perceived severity of the system failure. Assignment of a "severity factor" is a subjective process as can be seen in the analysis of the Browns Ferry fire in the ASP study and the INPO review.

- Probability of loss of high presssure coolant injection capability (ASP = 1.0, INPO = 0.05).
- Probability of loss of RCS depressurization capability (ASP = 0.25, INPO = 0.06).
- Probability of loss of low pressure coolant injection capability (ASP = 0.1, INPO = 0.01).

The differences between the ASP and NUREG values provides a measure of the systematic bias between the two groups.

B. Conclusions

ASP initiating event and system failure probabilities are point estimates derived from a relatively small sample set. The estimates appear to be conservative, however, uncertainties are not stated, and may be quite large. If possible, the ASP data should be refined by a more detailed review of events contributing to failure data so that partial failures are not counted as complete function failures. This improvement may be warranted because of the sparce nature of precursor data and the signficant adverse impact that each false data point can have on estimates of system failure and initiating event probability. This improvement must be tempered, however, in the light of the incomplete nature of the LER data base.

Severity factors for estimating the probability of restoring a failed system are determined, based on subjective engineering judgements. The point of view of the analyst can be a significant factor affecting the value of severity factors. The uncertainty in subjective assessments in the ASP study can be large, therefore, it does not seem necessary to provide finer graduations for the severity factor values that can be assigned.

Once system design or operational changes have been modified and previously dominant system failure modes eliminated or reduced, there can be significant changes in system failure probability and/or initiating event probability. Where such changes have been made, the ASP system failure data will have only limited applicability outside the time frame of the ASP study.

5.6 QUANTIFICATION OF STANDARD EVENT TREES

A. Discussion

The events considered significant among the 169 precursors were identified by a ranking method based on "a measure of the probability of severe core damage associated with each precursor". This probability measure is "an estimate of the chance of severe core damage, given the conditions of the precursor exist". A total of 52 precursors with probability measures of $\geq 10^{-3}$ were selected as significant.

The statistical methods for determining individual precursor probabilities are straight forward and are based on the generic event trees and the initiating event frequency and system failure probability data as well as the "severity factors" discussed previously. In the ASP study, the LER data base represents a body of available experience and the ASP staff is the source of one consensus expert opinion on severe core damage precursors. Relevant "formal procedures" instituted in the ASP study include the following:

- Generic initiating event and system failure probabilities,
- Assumption of Poisson process,
- Limited range of "severity factors" to define the probability that a failed system will not be restored to operation,
- Assumption that system outage time equals one-half of the testing interval (for systems discovered to be inoperable during testing).

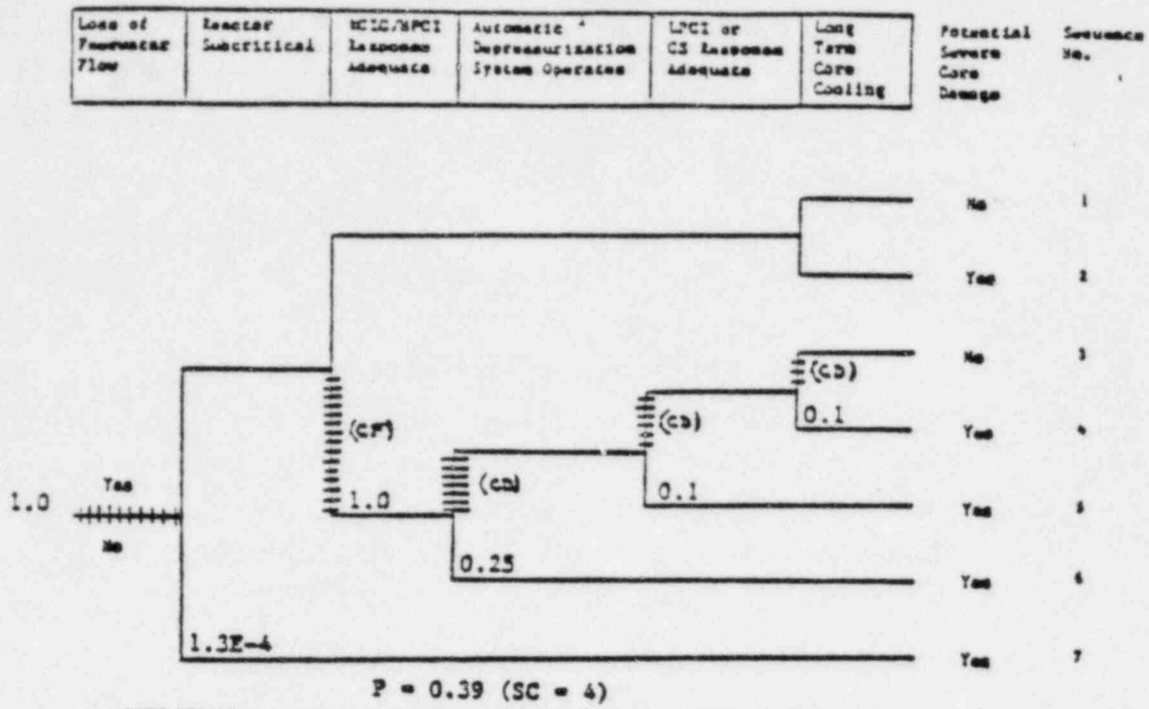
The INPO review team is a second source of expert opinion. The present body of knowledge of this group and their "formal procedures" are somewhat different than that of the ASP study group. These factors introduce a systematic bias that virtually ensures that the ASP and INPO results will be consistently different. A summary of the ASP and INPO conditional probabilities of severe core damage is shown in Table 5-8. It is instructive to note that the INPO values (with only one exception) are all lower probabilities. As can be seen in Figure 5-5 (Browns Ferry fire, from Ref. 2), the ASP and INPO assessments both are based on comparable event trees, but signficantly different data are used in quantification of the event tree. Systematic bias of the ASP and INPO groups is believed to be a significant contributor to the differences between their respective estimates. While methodologies exist to perhaps resolve these differences (Ref. 10), it is beyond scope of this review to do so.

B. Conclusions

The quantitative techniques used in the ASP study lead to a reasonable ranking of precursor events. However, alternative quantification schemes (e.g., INPO) may provide an equally valid, though different ranking. The estimated probability determined from a particular LER can be considered as a measure of the "distance" to core damage that existed at the time of the event. Any changes in plant configuration may affect the applicability of the ASP study estimates.

The ASP results cannot readily be applied from plant to plant, because significant differences exist in design, operating procedures, training, maintenance practices and plant age. For this reason, all 529 precursors identified from the initial screening should be evaluated, and possibly ranked, on a plant-by-plant basis. It is possible that a precursor judged to be not significant at one plant may be more significant at another plant because of differences in design, operating practices or training.

NUREG ASSESSMENT

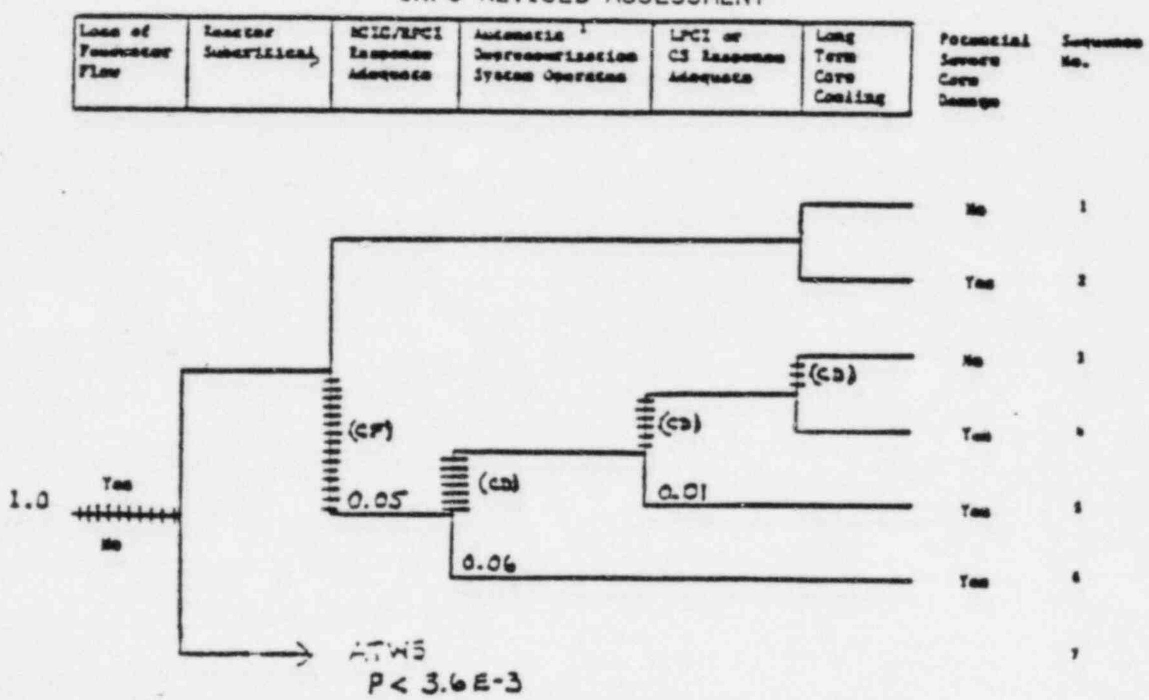


NSIC 101444 — Sequence of Interest of Cable Tray Fire at Browns Ferry 1

[1] Nonstandard techniques could have been used to make RCIC operable.

[2] The depressurization was manually initiated.

INPO REVISED ASSESSMENT



NSIC 101444 — Sequence of Interest of Cable Tray Fire at Browns Ferry 1

[1] Nonstandard techniques could b e make RCIC operable.

A-14

[2] The depressurization was manu

Figure 5-5. Comparison of ASP and INPO Assessments Of the Browns Ferry Fire.

Table 5-8. Comparison of NUREG and INPO Ranking of Fifty-Two LER Events*

NUREG Ranking	NUREG Conditional Probability	INPO Ranking	INPO Conditional Probability
1	1.0	1	
2	$3.9x10^{-1}$	2	$<3.6x10^{-3}$
3	$2.5x10^{-1}$	16	$<3.6x10^{-4}$
4	$2.5x10^{-2}$	13	$<4.9x10^{-4}$
5	$2.5x10^{-2}$	18	$<2.7x10^{-4}$
6	$2.3x10^{-2}$	6	$<1.2x10^{-3}$
7	$2.5x10^{-2}$	19	$<2.7x10^{-4}$
8	$2.77x10^{-2}$	28	$<1.4x10^{-4}$
9	$1.8x10^{-2}$	42 (NS)	$<1.6x10^{-5}$
10	$1.44x10^{-2}$	8 (NS)	$<1.0x10^{-3}$
11	$1.4x10^{-2}$	4	$<1.6x10^{-3}$
12	$1.2x10^{-2}$	36	$<5.3x10^{-5}$
13	$1.38x10^{-2}$	3	$<1.8x10^{-3}$
14	$1.0x10^{-2}$	12	$<5.0x10^{-4}$
15	$8.8x10^{-3}$	14	$<4.0x10^{-4}$
16	$6.4x10^{-3}$	- (NS)	NR
17	$6.1x10^{-3}$	37	$<4.0x10^{-5}$
18	$4.8x10^{-3}$	33 (NS)	$<8.1x10^{-5}$
19	$4.3x10^{-3}$	- (NS)	NR
20	$5.6x10^{-3}$	7	$<1.2x10^{-3}$
21	$5.5x10^{-3}$	40	$<2.2x10^{-5}$
22	$3.1x10^{-3}$	35	$<6.4x10^{-5}$
23	$3.4x10^{-3}$	38	$<3.6x10^{-5}$
24	$2.8x10^{-3}$	34 (NS)	$<8.1x10^{-5}$
25	$2.9x10^{-3}$	17 (NS)	$<3.6x10^{-4}$
26	$2.4x10^{-3}$	29	$<1.4x10^{-4}$
27	$2.5x10^{-3}$	5	$<1.3x10^{-3}$

Table 5-8. Comparison of NUREG and INPO Ranking of Fifty-Two LER Events* (Continued).

NUREG Ranking	NUREG Conditional Probability	INPO Ranking	INPO Conditional Probability
28	1.8×10^{-3}	24	$<1.7 \times 10^{-4}$
29	1.8×10^{-3}	25 (NS)	$<1.7 \times 10^{-4}$
30	1.8×10^{-3}	39	$<2.6 \times 10^{-5}$
31	1.45×10^{-3}	9	$<1.0 \times 10^{-3}$
32	1.7×10^{-3}	10	$<8.7 \times 10^{-4}$
33	1.45×10^{-3}	21	$<2.0 \times 10^{-4}$
34	1.45×10^{-3}	22 (NS)	$<2.0 \times 10^{-4}$
35	1.6×10^{-3}	31 (NS)	$<1.1 \times 10^{-4}$
36	1.6×10^{-3}	23	$<2.0 \times 10^{-4}$
37	1.6×10^{-3}	45	$<6.3 \times 10^{-6}$
38	1.6×10^{-3}	-	NR
39	1.3×10^{-3}	26	$<1.6 \times 10^{-4}$
40	1.2×10^{-3}	30	$<1.2 \times 10^{-4}$
41	1.1×10^{-3}	11 (NS)	$<7.5 \times 10^{-4}$
42	1.26×10^{-3}	43 (NS)	$<1.0 \times 10^{-5}$
43	1.4×10^{-3}	27	$<1.5 \times 10^{-4}$
44	1.25×10^{-3}	2	$<1.0 \times 10^{-2}$
45	1.28×10^{-3}	15	$<3.9 \times 10^{-4}$
46	1.3×10^{-3}	44	$<1.0 \times 10^{-5}$
47	9.12×10^{-4}	41	$<2.0 \times 10^{-5}$
48	9.3×10^{-4}	46 (NS)	$<3.1 \times 10^{-6}$
49	9.1×10^{-4}	47	$<2.2 \times 10^{-7}$
50	9.4×10^{-4}	32	$<9.8 \times 10^{-5}$
51	9.9×10^{-4}	20	$<2.2 \times 10^{-4}$
52	9.8×10^{-4}	- (NS)	NR

* Based on data in INPO "Review of NRC Report: Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report, NUREG/CR-2497"

NR = Not Reevaluated, NS = Not a Significant LER (INPO)

5.7 ESTIMATE OF AN OVERALL FREQUENCY OF SEVERE CORE DAMAGE

A. Discussion

As stated in NUREG/CR-2497, "The probabilities of severe core damage associated with the precursors were also used to estimate the frequency of severe core damage per reactor-year for the years 1969-1979. This point estimate is between 1.7×10^{-3} and 4.5×10^{-3} per reactor year....". These results were compared in the ASP study with the estimates from PRAs as shown in Figure 5-6 (Figure 1, from Ref. 1). Based on discussions in previous sections, comparison of a specific estimate of core melt frequency from a PRA with the ASP indicator of severe core damage frequency has no real meaning of itself. Therefore, Figure 5-6 can be misleading, and it has contributed significantly to the controversy over NUREG/CR-2497.

In spite of this, trend information can be derived by consistent application of the ASP methodology to different periods of time. Other methodologies, if consistently applied, could also serve this purpose. When viewed from this perspective, the ASP approach of summing the estimated severe core damage frequencies of relevant precursors is an acceptable technique for deriving a numerical "indicator" for use in future trend analysis. Comparison of such consistently derived numerical indicators among themselves should provide qualitative or semi-qualitative insight on plant-specific and industry-wide performance. Some further issues are discussed below.

1. Frequency of Severe Core Damage Per Reactor-Year

   In NUREG/CR-2497, a severe core damage frequency per reactor-year was estimated by taking the estimated conditional probabilities of severe core damage given the precursors, multiplying by the precursor frequency estimates and summing these

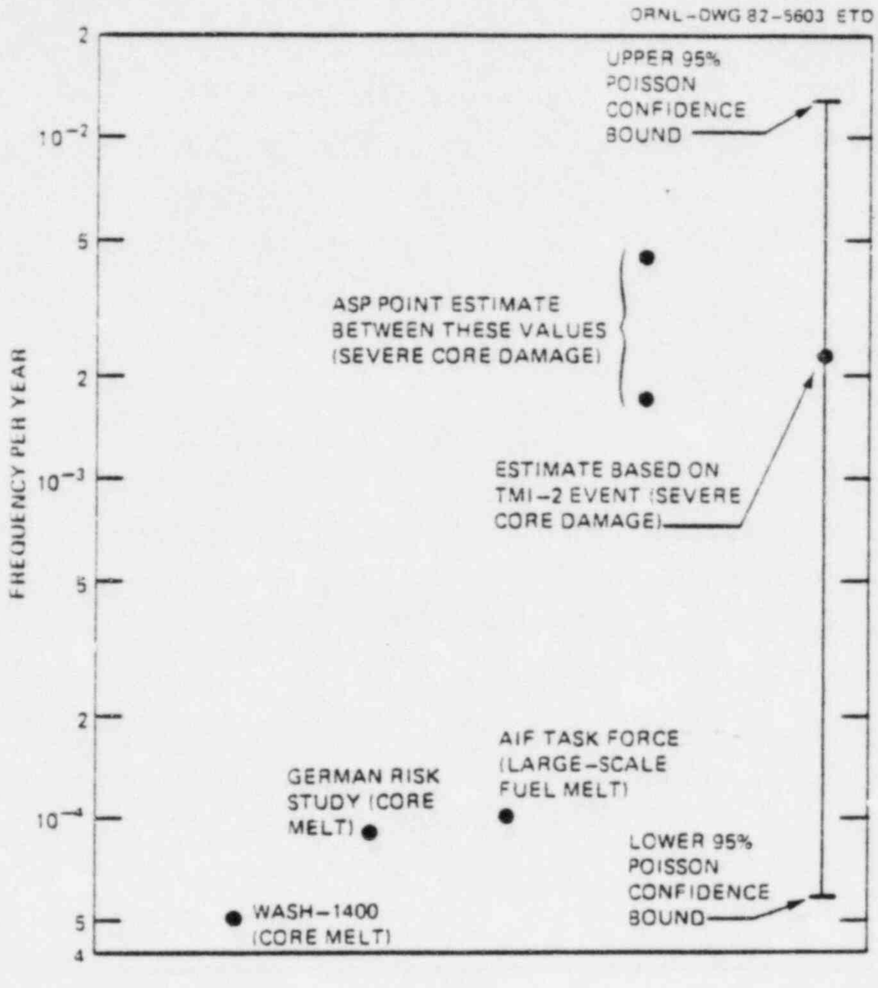


Figure 5-6. Comparison of ASP Results With Other Core Damage Estimates (From NUREG/CR-2497).

results for all precursors. The validity of this technique was questioned by INPO due to the adding together of actual failures with the probability of failure for those sequences which were actual successes.

An estimate of severe core damage frequency per reactor-year for the 11-year ASP study period can simply be 1/432 = 2×10^{-3} per reactor year (e.g., TMI was the only observed case of severe core damage in 432 reactor-years). This is an observed frequency of severe core damage which assumes that the TMI event was not an outlier. Another estimate, and the one used in the ASP study, adds the additional information gained from the LERs. This estimate attempts to define an underlying frequency of severe core damage and is a statistical method that has been previously documented (Ref. 10). This approach also assumes that the TMI event was not an outlier. As a result of adding additional probability contributions from precursors which did not actually result in severe core damage but did have a significant likelihood of occurrence, the underlying frequency estimate is greater than observed frequency estimate by a small amount. These estimates apply to the period of the ASP study (1969 to 1979) and would have some predictive value if all relevant factors remained unaltered. As discussed previously, this is not the case because of corrective actions taken in response to the LERs and the consequent impact on the ASP data base. The ASP estimate of severe core damage frequency per reactor-year is therefore not predictive. In fact, the limitations in the data and depth of system modeling suggest that this estimate should be treated more as a figure-of-merit (e.g., only an indicator) rather than an actual estimate of severe core damage frequency.

2. Treatment of TMI as a Precursor

In NUREG/CR-2497 the entire TMI event was treated as a precursor and this event was included in the estimation of severe core damage frequency. The inclusion of this event automatically makes the frequency estimate at least as large as 1/432 since the TMI event led to severe core damage. An alternative approach would be to treat only a part of the TMI sequence as a precursor and treat the remainder as probabilistic system failures rather than as given failures. This treatment of precusors is not as conservative as the current approach in NUREG/CR-2497.

B. Conclusions

Comparison of specific estimates of core melt frequency from a PRA with the ASP indicator of severe core damage frequency has no real meaning of itself. By including the illustration shown in Figure 5-6, the ASP study implied that such comparisons could be made. Comparison of the ASP indicators of severe core damage frequency derived for different time periods should provide insight on plant-specific and industry-wide performance.

5.8 TREND ANALYSIS

A. Discussion

An analysis of failure trends is presented in NUREG/CR-2497 in the form of time-on-test plots which were used to determine instantaneous failure rates for selected initiating events and function failures. The time-on-test plot is an x-y plot of fraction of total observation time (e.g., 432 reactor-years of operation from 1969 to 1979) against fraction of total failures. The y-axis can be related to calendar dates however the relationship is non-linear. A straight line at a 45^{o} angle represents the predicted failures when there is a constant instantaneous failure rate.

The results of the trend analysis provides interesting insight, however, these results for small samples can be very misleading. One reason is that many different factors may be affecting the data base for the trend analysis. The fact that new plant data is mixed with old plant data can obscure trends dealing with infant mortality or wearout of systems. The results as they are currently applied would be good for seeing the impact of major design or procedure changes on failure rate. For this reason, new data from the post-1979 period should be plotted along with the 1969 to 1979 ASP data.

Given enough data to make reasonable inferences, there are five generalized situations that the data can reflect from trend analysis results:

- Gradually decreasing failure rate
- Step decrease in failure rate
- Gradually increasing failure rate
- Step increase in failure rate
- Constant failure rate

Example trend curves for each of these situations are shown in Figure 5-7. Each curve includes the reference straight line for constant failure rate.

All these curves can have calendar date as well as plant life time considerations blended. The differences between these two effects can mask information from the curves.

B. Conclusions

Trend analysis procedures used in NUREG/CR-2497 are valid for data based on calendar periods such as the period from 1969 to 1979. There is a paradox, however, in that the trend plots are capable of showing failure rate changes as a result of plant, operational or training improvements for which credit is not taken in the ASP study. In addition, trend data may be very difficult to interpret however because: (1) paucity of data may yield unreliable trends, and (2) mixture of data from new and old plants may "smear" trends associated with infant mortality or wearout.

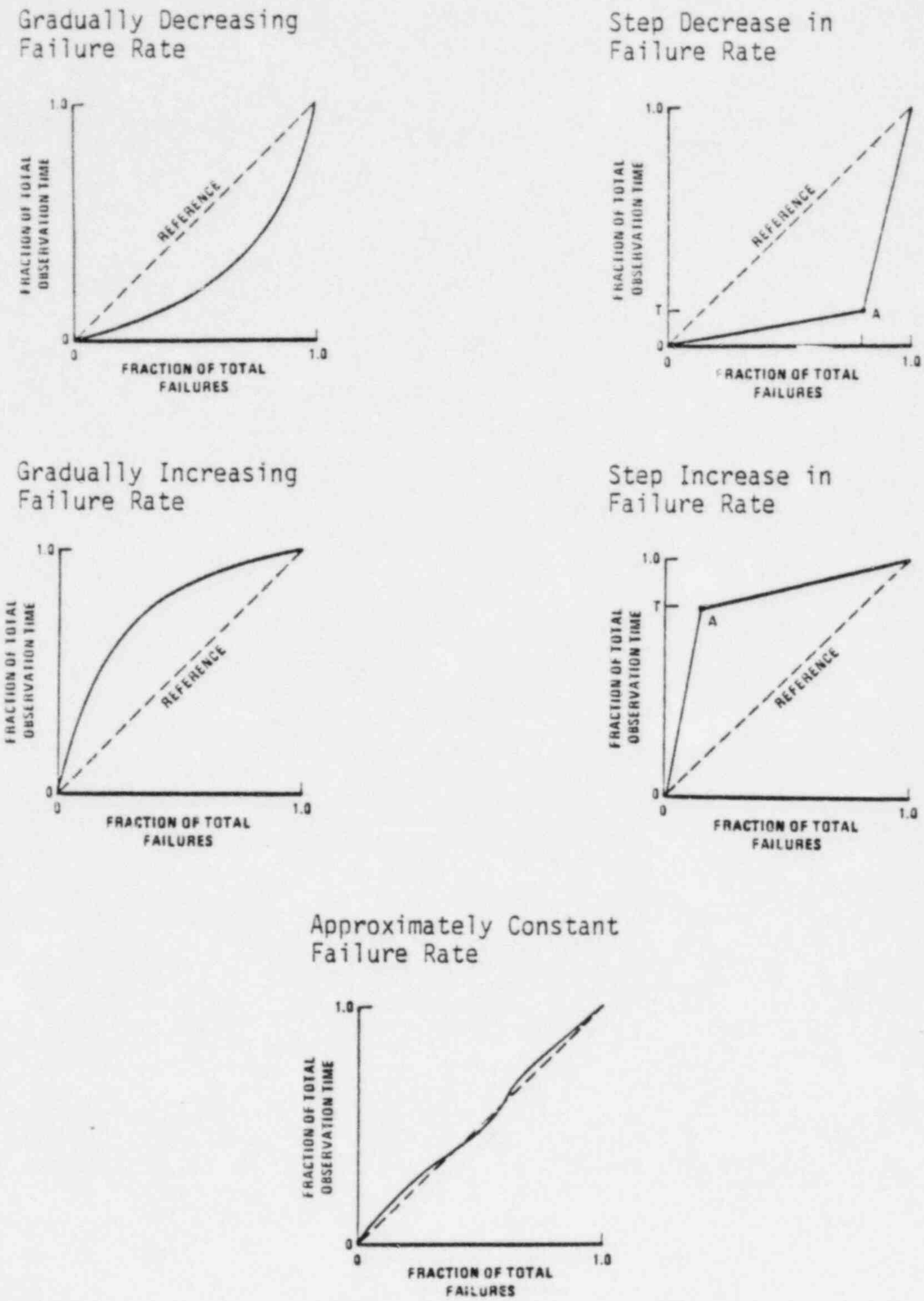


Figure 5-7. Example Trend Plots.

6. VALUE OF PRECURSOR STUDY

In the preceding section, various aspects of the ASP study were reviewed and critiqued. The general conclusions that are drawn in Section 5 are that: (a) the ASP screening and ranking process is reasonable, (b) the LER data base and the ASP plant models have significant limitations related to quantitative analysis, (c) the absolute values calculated in the ASP study are indicators only, and should not be treated with confidence as accurate estimates of the conditional probability of severe core damage given the precursor, or severe core damage frequency per reactor-year, (d) the ASP indicators described in (c), above, should not be casually compared with the results of plant-specific analyses such as PRAs. These conclusions do not detract from many useful applications of the results of the ASP study. Several potential applications are discussed in this section.

A. Identify and Rank Potential Weaknesses in Plant Design, Operating Practices or Training.

In the ASP study, precursor event sequences were ranked based on a measure of their potential significance as contributors to risk. The ranking establishes a relative order of importance of LERs and thereby aids in focusing attention on the most significant events that have occurred at nuclear power plants. Through a more detailed and consistent review of the significant LERs, it is reasonable to expect that weaknesses in plant design, operating practices or training will be uncovered. In fact, INPO noted many plant changes that have already been made in response to significant LERs identified in the ASP study (see Table 6-1 from Ref. 2). Some important insights that are highlighted in Table 6-1 are the following.

Table 6-1. Plant Changes Relative to Significant Precursor Events (From INPO, Ref. 2).

NSIC Accession Number	Actual Occurrence	Plant	Relevant Changes
153164	Loss of feedwater & open PORV	TMI-2	Post-TMI changes (made to Unit 1 and all plants of similar design)
101444	Cable tray fire caused extensive damage	Browns Ferry 1	Fire protection design and procedure changes
138830	Failure of NNI & steam generator dryout	Rancho Seco	More reliable non-nuclear instrumentation, post-TMI changes
90421	Inoperable AFW pumps during plant shutdown (clogged strainers)	Point Beach 1	Strainers removed
91676	Failure of three AFW pumps to start at test (packing overtightened)	Turkey Point 3	Procedure requires post-maintenance testing
108078	Inoperable AFW pumps during startup (clogged strainers)	Kewaunee	Fine mesh strainers removed
133706	AFW pumps inoperable during test (pumps started; loss of speed control)	Davis-Besse 1	Control System modified (this is not a precursor event by ORNL criteria; no loss of function occurred)
149450	Loss of feedwater flow	Oyster Creek	Procedures and Tech Specs. preclude shutting of all recirculation loop valves, switch demarcation, training emphasis
63129	Loss of off-site power	LaCrosse	Installed additional service water pumps (gasoline-powered), additional diesel generator
88451	Loss of off-site power	Haddam Neck	Turbine-driven AFW pump made AC-independent

- Better perception of the significance of some component failure modes (e.g., auxiliary feedwater system failure due to strainer plugging at Point Beach 1 and Kewannee).

- Identification of component/support system requirements that should be revised (e.g., AC-dependent turbine-driven AFW pump at Haddam Neck).

- Identification of procedural training, and Technical Specification deficiencies (e.g., related to loss of feedwater at Oyster Creek).

As part of, or separate from, the ASP study, there should be a mechanism to evaluate, on a plant-specific basis, the actual significance of precursors that were identified as potentially significant in the ASP study. One approach is to submit for review at each plant all the precursors identified from the ASP initial screening process (e.g., 529 out of 19,400 LERs). The review of all 529 precursors would prevent an oversignt that may occur in the case of an LER that is judged to be not significant at one plant while it might be more significant at another plant with different design features or operating practices. The level-of-effort involved in this type of review is not as great as it may appear because the 529 precursors would not apply to all plants.

B. Improving Probabilistic Risk Assessments

As stated previously, the numerical estimates made in the ASP study should be treated only as "indicators" that are not directly comparable to the results of plant-specific analyses. In spite of this, the ASP study is capable of providing insight into the completeness of PRAs and the adequacy of data used for quantifying PRAs. Some examples are discussed below.

1. Identification of Sequences That Were Not Previously Included in PRAs.

It is difficult to be confident that a PRA is complete in the sense that it includes all relevant event sequences. Ideally, one would expect to find all of the ASP significant precursor sequences in plant-specific PRAs for related plants. If a significant ASP precursor sequence cannot be identified in a plant-specific analysis it may indicate an oversight, and a revision of the plant model used in the PRA may be necessary. It is instructive to note that Oyster Creek event 149450 was not included in the unpublished PRA for that plant. It is also possible that the ASP study may illustrate the importance of some operator interfaces and system/support system dependencies that are often not modeled in detail in current plant-specific studies. The system dependencies may include systems such as equipment room cooling and ventilation and control air systems which are not required in the short-term following an initiating event, but which may be required for long-term maintenance of a safe shutdown condition.

2. Identification of Suspect Data

The ASP system failure probabilities are derived from generic sparce operating experience and should not be expected to be the same as system failure probabilities derived from plant-specific analytical studies. The ASP data may, however, serve as a useful indicator by identifying systems such as the PWR auxiliary feedwater system and the BWR depressurization system for which there is significant variance between ASP and other estimates of the failure probability. Resolution of this variance could lead to refinement of the ASP data base and reevaluation of component failure rates or system models used in probabilistic studies.

C. Improved Understanding of the Role of the Operator

In the ASP study, it was noted that operator error was involved in 38 percent of the significant precursor sequences. The operator also had a role in the recovery of the plant during some sequences. Further evaluation of the ASP results may provide a better understanding of the potential impact that an operator can have on accident response, and the need for plant, operational or training changes related to the operator.

D. Insight into Nuclear Industry Safety Trends

The time-on-test trend plots establish baseline trends for selected initiating events and system failures. By adding data for other time periods (e.g., post-1979) as it becomes available, changes in these trends may be observed. These changes could provide a qualitative measure of the impact on reactor safety of the post-TMI NRC regulatory process. In addition, it may be possible to identify new trends requiring regulatory attention.

7. REFERENCES

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see RPACIW004

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February 18, 1983

Mr. G. R. Quittschreiber, Chief
Project Review Branch No. 2
U.S. Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards
Washington, D.C. 20555

Dear Mr. Quittschreiber:

As a follow-up to your telephone conversation with Mr. E. L. Zebroski of our Institute, I am enclosing five (5) copies of the INPO report, INPO 82-025 "Review of NRC Report: Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report NUREG/CR-2497." I understand that these reports are to be used for an ACRS Subcommittee that will review the ORNL report.

Sincerely,

Angelina S Howard

Angelina S. Howard
Director
Communications Division

ASH/la
Enclosures

cc: E. L. Zebroski

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X-REG-6.2