UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 14 1983

MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Robert B. Minogue, Director
Office of Nuclear Regulatory Research

SUBJECT: INTERIM RESEARCH INFORMATION LETTER 136 FOR THE ACCIDENT SEQUENCE PRECURSOR PROGRAM

A. Background and History of the Precursor Program

The Precursor Program has been ongoing since 1979. It was instituted because of the need to assess, in a disciplined manner, operating experience, and in part because of the findings of the Risk Assessment Review Group (Lewis Committee) in their review of WASH-1400, the Reactor Safety Study. The Lewis Committee report (NUREG/CR-0400, September 1978) indicated that reactor plant operational data should be evaluated by the kind of analysis contained in WASH-1400. This RIL summarizes the status and findings of the precursor program to date. Additional information is found in the attachment to this RIL.

At the start of the program the objective was purposely left vague to allow experimentation and evolution of useable techniques. The objective was generally to examine nuclear plant operational experience data and assess plant safety as it is reflected by operational experience. The operational experiences of interest were challenges to plant safety systems (initiating events), total failure of a plant safety system or function, or partial failures of multiple plant safety systems.

In early 1981 the first evaluation of operational data for precursors was assembled into a draft report and circulated within NRC and selectively outside of NRC for comment. This was followed by a more formal draft in early 1982 which was similarly distributed for comment.

Comments on the above two drafts were, where pertinent, factored into the 1969-1979 Precursor Report (NUREG/CR-2497), a status report which was widely distributed in July 1982 in order to obtain broader review. The report was controversial in that it indicated a comparatively high (greater than 10^{-3} per reactor year) industry averaged severe core damage estimate for the eleven year interval evaluated.

Substantial individual and peer group reviews of the report were begun in the summer of 1982. The most extensive of these was the review undertaken by the Institute for Nuclear Power Operations (INPO) in the summer of 1982. INPO assembled a team from within INPO, EPRI, the Nuclear Utilities, and private consultants to scrutinize and evaluate the precursor report. They produced their own report (INPO 82-025) in September 1982 which reassessed the important sequences of the precursor report. The INPO report concluded that the severe core damage frequency of the precursor report was overestimated by a factor of thirty. The INPO precursor analysis differed from the ORNL work in that the cumulative reactor years of operation extended into 1982 without including precursor events in the 1979-1982 period, and the expected effect of fixes resulting from TMI and other past accidents of importance were incorporated in their estimate of accident occurrence frequency. The INPO report also noted that the precursor report functional level event trees did not adequately reflect actual equipment, and that not enough credit was given to the ability of operators to innovate and cope with accident situations and equipment failures. Furthermore the INPO report noted that some of the ORNL selected precursor failures were not pertinent, i.e., that ORNL was too conservative in their selection criteria for certain failure events.

During the fall and winter of 1982-1983, numerous other comments were received on the precursor report, including a Science Applications Incorporated assessment, and comments from the System Reliability Service (UKAEA), NRC offices, and the nuclear utilities. Concurrently during this period, the second precursor report was in preparation by ORNL assessing the 1980-1981 operating experience.

While many of the reviews of the precursor report were comprehensive, they mainly addressed questions of methodology or technique. Other comments were focused on critiquing individual precursor sequences. What was lacking from the study, reviews, and critiques was an assessment of what the real insights were; i.e., what did the list of 169 precursor sequences indicate about reactor safety. In an attempt to correct this shortcoming, ORNL has been asked to further assess the meaning or implications of related groups (and subgroups) of precursors to the safety of reactors in the final version of their second precursor report. Likewise, precursor methodology work being performed at the University of Maryland has been redirected to emphasize interpretation of the insights gained from the precursor study.

In the winter of 1983, EPRI sponsored a precursor studies workshop. At the workshop, problems with precursor analysis were broadly discussed, including a discussion of what can or should be done in such studies. An initial draft of the second precursor report (1980-1981 data) was distributed by ORNL to the workshop participants.

The ACRS held two PRA subcommittee meetings in spring 1983 in which various problems or methodological aspects of the precursor program were discussed. The ACRS full committee later issued a letter in which they generally endorsed the precursor program efforts. This letter is summarized in the attachment to this RIL.

In late spring of this year the draft second precursor report (unquantified version) was sent to INPO, and to the utilities affected, by ORNL with a request for comments. Over 80 percent of the utilities responded with useful comments on plant configuration or additional information on circumstances surrounding the identified precursor incidents.

We are now in the process of assessing the draft second precursor report and are actively seeking comments on that report. Listed below is a summary assessment of what has been found in the precursor program to date. The attachment to this RIL provides additional background information on the precursor program.

B. Summary of RES Conclusions at this time

The research done to date in the ASP Program and the intense peer review of it in the past year have led us, as the sponsors of the work, to form the following conclusions. The bulk of our comment is on the 1969-1979 data set. Some tentative conclusions based on a "quick look" at the 1980-1981 data set are included.

There has been intense evaluation of the objectives of the ASP program and we conclude that there are and should be three principal objectives for this work. They are treated here in the order of their priority and utility with the most important treated first. The objectives are:

a. From operational events identify significant or important sequences that, more likely than others, could have led to severe core damage.

b. Search operational events for the elements or precursors of severe core damage accident sequences which are not predicted or poorly predicted in current probabilistic risk analyses (PRA).

c. Analyze operational events to estimate the frequencies and trends of system failures, function failures, and overall frequency of severe core damage as an alternate data source to compare to frequencies estimated in PRAs. That is, test the validity of the PRA results.

B.1 The First Objective - Identification of Important Sequences

The first of these objectives is the most reliable use of ASP analysis. NUREG/CR-2497 analyzed 19,400 LERs and listed 169 as significant. We have listed the 52 most significant precursors in Table 1 attached. Although the peer review process raised substantive questions about the absolute values of frequency estimates associated with these events, they still stand as a reasonably defensible set of the fifty or so most safety significant events of that set of 19,400 LERs. Our contractors have been asked to evaluate this experience base for its safety significance. Nevertheless, we recommend that NRR and I&E analyze these events for their regulatory significance. Our own review of these events indicates that substantial regulatory attention has been given or is being given to the principal problems identified by study of the events. Certainly, the two most significant, the TMI-2 accident and the Browns Ferry fire have generated many real improvements in plant safety. The third, the Rancho Seco light bulb incident, has led to improved power supplies for nonnuclear instrumentation and has figured in consideration of pressurized thermal shock. Also, the likelihood of a number of other accident sequences would have been affected by requirements implemented since TMI, such as improvements in auxiliary feedwater systems.

The 1980-1981 operational data set consisted of 8200 LERs (or LER abstracts), only about 40 percent as much data as the 1969-1979 set; and the draft report now under review identifies 56 events which can be considered significant. While the 1969-1979 precursor evaluation concluded that only the events with a conditional probability for core damage greater than 10^{-3} contributed significantly to risk, a similar rationale is not statistically adequate for the 1980-1981 data because there are only eight sequences of greater than 10^{-3} probability for this period. Eight sequences (refer to Table 2) are not sufficient to provide much confidence for comparison or trending with the 52 dominant sequences of the 1969-1979 report. Notwithstanding this limitation, we have made a rather simplistic comparison (refer to Table 3) of the fractions of core damage attributable to various scenarios for the 1969-1979 time period versus the 1980-1981 time period. The intent of this table is to try to show any trends that may be apparent in the operational data for the two time periods assessed. Although the relevance of Table 3 is certainly open to question, particularly for BWRs, it does seem to indicate that the fraction of severe core damage attributable to Loss of Feedwater events is decreasing.

It may be of interest to note that the Browns Ferry failure to scram (LER accession number 163405 on 6/28/80) did not make the list of dominant sequences in the 1980-1981 report. That sequence probability of severe core damage was calculated to be approximately 3 E-4. Generally, the reason why this sequence has a somewhat low probability is that the fault exposure (time since prior successful scram) was comparatively short. Yet, it is still a precursor of considerable importance as it relates to scram system reliability. This precursor event and its quantification will be examined further in the final report.

B.2 The Second Objective - PRA Completeness

The second objective of the ASP Program is to identify elements or precursors of severe core damage sequences which are not predicted or are poorly predicted in PRAs. Turning again to Table 1 for the 52 most significant precursors for the 1969-1979 period, one finds several important sequences for this objective. Much has been written about whether the TMI-2 sequence was predicted in the Reactor Safety Study. It should be noted that the functional and systemic characteristics of the accident sequence are readily predictable by PRA but that the unique or specific accident progression involving operator error of commission is not modeled well in PRAs. That topic is the subject of research both for ways to reduce it and for ways to model or predict it. Another precursor for which PRA identification and prediction capability has been questioned is the Rancho Seco overcooling transient. It could arguably be said that this incident represents a new generic class of accident sequences not previously treated by PRA. The pressurized thermal shock research program seeks to remedy that situation, at least in part.

Examination of the top eight precursors for the 1980-1981 period shows that they can be categorized as falling within one or another of the generic accident sequence classes which have been previously identified in PRAs. However, these precursors do include some unique and unusual failures and interactions. The ability to identify such unique characteristics of dominant accident sequences is generally determined by the resources applied to searching out such potentialities (i.e., modeling completeness) and the availability of data to quantify unique sequences of events. The limit of resolution in PRA is generally believed to be at the system and component failure rate level, with a more limited capability to identify the likelihood of specific individual component failure modes.

Our conclusion on this matter is that most precursors are totally rational scenarios and are acknowledged to have been predictable as to the accident sequence, but it is not reasonable to expect to identify all possible actual failure modes or to predict the likelihood of all specific individual component failure modes.

B.3 The Third Objective - Frequency and Trends

The third objective of the ASP Program is to obtain what some call validating data for PRA predictions. We must emphasize that because of the inherent paucity of the data base, the LERs alone, frequencies estimated by the ASP program could be less reliable than those obtained by careful estimate in PRA studies. This concern needs to be and will be analyzed further. For example, the failure of the AFW system is derived in NUREG/CR-2497 from just eight events (see Table C.1). In addition to the basic paucity of the data base there is the problem of extracting frequencies of different elements ranging from initiating events on through the accident sequence to the frequency of severe core damage. The farther along the event tree that the element is, the more will it be the product of analytical extrapolation rather than the product of observation--and therefore the less reliable will be an estimate of sequence occurrence. It is for this reason that the least reliable of the ASP derived frequencies is the frequency of severe core damage, which is the best known and most controversial of the results of the ASP analyses.

One result of the NUREG/CR-2497 analysis was an industry average estimated severe core damage (point estimate) frequency of 1.7 to 4.5 x 10^{-3}/year. After consideration of the peer reviews, the contractors, ORNL/SAI, have revised this to approximately 1.3 to 3.5 x 10^{-3}/year. An industry average from PRAs might be inferred by averaging the results of the PRAs published so far (see W. J. Dircks memo on Safety Goals, sent to the Commissioners on January 5, 1983) giving an average estimated frequency of severe core damage of about 3 x 10^{-4}/year. Recognizing the uncertainties in both the ASP extrapolations and the PRA predictions, and excluding TMI, Browns Ferry, and Rancho Seco from the ASP data base, these results are not inconsistent. We assume the fixes to prevent recurrence of these particular classes of events have been effective, and will significantly reduce the likelihood of future occurrence. It is important to note that both averages are subject to obsolescence. The ASP results based on the 1969-1979 period are from a time when consideration and treatment of severe accident sequences was not emphasized. Thus it can be argued that the 1969-1979 period represents the pre-TMI data base. Many of the lessons learned from TMI have been implemented in the years since, although the degree of implementation and the timing varies from plant to plant. Therefore, one can expect, at least in theory, that trends will show the results of these changes. The inherent paucity of the data base for the ASP will make it very difficult to identify such trends with much confidence at this time. However, the preliminary results of the 1980-1981 analysis indicates a core melt frequency for that period of approximately 2.5 E-4 per reactor year, which might cautiously be interpreted as indicating a gross downward trend.

The published PRA predictions are in the main obsolete. Many plant owners have taken steps to suppress dominant sequences identified in the PRA. New considerations (e.g., seismic risk or modelling of cooling by feed-and-bleed) could either increase or decrease the predicted frequencies of severe core damage. One must look with caution at the ASP derived frequencies for reliable and useful insights. Table 5.1 of NUREG/CR-2497 compares system unavailabilities and initiating event frequencies derived from ASP with WASH-1400 predictions. It indicates a fairly good comparison. We believe that the correlated (common cause/mode) failures and previously mentioned subtle interactions which are not now capable of being modeled by PRAs are what cause significant differences in predictions, as well as the other inherent uncertainties in PRA.

C. Tentative Conclusions

Our preliminary conclusion is that the industry wide expected core melt frequency probably has decreased from the period 1969-1979 to the period 1980-1981, but the amount of decrease cannot be quantified with much confidence. This is not surprising in light of the reactor safety activities which have taken place since the TMI accident. The 1980-1981 precursor analyses do not indicate new accident sequences, or changes in sequence occurrence frequency which would require prompt remedy. However, we should emphasize the need for caution regarding this conclusion due to the preliminary nature the 1980-1981 precursor report and the methodological limitations of this work, which at this time should be viewed much in the nature of a "quick look" report.

We suggest that NRR and I&E review the precursors to determine that the scope of their safety programs and regulatory activities cover the areas of plant design and operation found to be most important. As has been indicated, we are analyzing the precursor data further and are actively seeking critiques of the draft 1980-1981 report. We will keep you informed as progress is made in this program.

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cc w/encl:
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Table 1. 1969-1979 Precursor Sequences ($P_r \geq 10^{-3}$)

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	SC	RATE	T	V	AE	OPR	CRITXX
153164	790328	LOFW	LOSS OF FEEDWATER & OPEN PORV	TMI 2	320	CJ	VALVEX	E	O	Y	Y	365	00	906	P	B	BR	MEC	780328
101444	750322	LOFW	CABLE TRAY FIRE CAUSED EXTENSIVE DAMAGE	BRN.FERRY1	259	SF	ELECON	E	O	Y	Y	582	04	1065	B	G	UX	TVA	730817
138830	780320	LOFW	FAILURE OF NNI & STEAM GENERATOR DRYOUT	RANCHOSECO	312	IE	INSTRU	E	O	Y	Y	1281	06	918	P	B	BX	SMU	740916
90421	740407	LOFW	INOPERABLE AFW PUMPS DURING PLANT SHUTDOWN	PT.BEACH 1	266	SF	FILTER	D	O	N	N	1252	16	497	P	W	BX	WMP	701102
91676	740508	LOFW	FAILURE OF 3 AUX FWDTR PMPS TO START AT TEST	TKY.POINT3	250	SF	PUMPXX	E	T	Y	N	565	16	693	P	W	BX	FPL	721020
108078	751105	LOFW	INOPERABLE AFW PUMPS DURING PLANT STARTUP	KEWAUNEE	305	SF	FILTER	C	O	N	N	608	16	535	P	W	FP	WPS	740307
133706	771211	LOFW	AUX. FEEDWATER PUMPS INOPERABLE DURING TEST	DVS-BESSE1	346	SF	PUMPXX	G	T	N	N	121	16	906	P	B	BX	TEC	770812
149450	790502	LOFW	LOSS OF FEEDWATER FLOW	OYSTER CRK	219	CJ	VALVEX	E	O	Y	Y	3651	16	650	B	G	BR	JCP	690503
63129	710324	LOOP	LOSS OF OFFSITE POWER	LACROSSE	409	EA	CKTBRK	E	O	N	Y	1352	18	50	B	A	SL	DLP	670711
88451	740119	LOOP	LOSS OF OFFSITE POWER	HAD.NECK	213	EA	INSTRU	E	O	N	Y	2401	18	575	P	W	SW	CYA	670624
128906	770831	LOFW	LOSS OF NO-BREAK-POWER AND FEEDWATER CONTROL	COOPER	298	SF	GENERA	E	O	N	Y	1287	19	778	B	G	BR	NPP	740221
137305	780325	LOFW	LOW-LOW WATER LEVEL IN ONE STEAM GENERATOR	FARLEY 1	348	SH	VALVEX	E	O	Y	N	228	19	829	P	W	BX	APC	770809
149961	790603	LOFW	HPCI FAILS TO START GIVEN LOFW	HATCH 1	366	SF	PUMPXX	E	O	Y	Y	334	19	784	B	G	SS	GPC	780704
116212	760720	LOOP	APPARENT LOOP & FAILURE OF SAFETY RELATED COMP	MILLSTONE2	336	EB	RELAYX	E	O	Y	Y	277	20	870	P	C	BX	NNE	751017
61434	700717	LOOP	LOSS OF OFFSITE POWER	HUMBDLTBAY	133	EA	INSTRU	E	O	Y	Y	2708	21	65	B	G	BX	PGE	630216
61565	710902	LOOP	LOOP AND FAILURE OF A DIESEL GENERATOR TO LOAD	PALISADES	255	EA	RELAYX	U	O	N	Y	101	22	805	P	C	BX	CPC	710524
137918	780423	MSLB	MULTIPLE STUCK-OPEN RELIEF VALVES	TMI 2	320	HB	VALVEX	E	O	N	Y	26	22	906	P	B	BR	MEC	780328
137543	780413	LOOP	LOSS OF OFFSITE POWER WHILE SHUTDOWN	CALCLIFFS1	317	EA	CKTBRK	G	O	N	Y	1284	23	845	P	C	BX	BGE	741007
139565	780514	LOOP	LOSS OF OFFSITE POWER DURING REFUELING	ST.LUCIE 1	335	EA	CKTBRK	H	O	Y	Y	752	23	802	P	C	EX	FPL	760422
140335	780728	LOOP	LOOP AND DIESEL GENERATOR FAILURE	BVRVALLEY1	334	EB	TRANSF	E	O	N	Y	809	23	852	P	W	SW	DLC	760510
142462	781127	LOCA	LOSS OF VITAL INST. BUS-REACTOR TRIP	SALEM 1	272	EB	GENERA	E	O	Y	Y	716	23	1090	P	W	UX	PEG	761211
85566	731119	LOOP	LOSS OF A.C. POWER CAUSE HPCI/RCIC TO FAIL	BRN.FERRY1	259	SF	RELAYX	B	T	N	Y	65	25	1065	B	G	UX	TVA	730817
85738	731119	LOFW	RCIC/HPCI FAILS DURING TESTING	BRN.FERRY1	259	SF	VALVEX	B	T	N	Y	65	25	1065	B	G	UX	TVA	730817
152187	790903	LOOP	SWITCHYARD LOCKOUT DUE TO CABLE DROP AT STORM	ST.LUCIE 1	335	EE	ENGINE	G	O	N	Y	1229	25	802	P	C	EX	FPL	760422
153810	791120	LOFW	RCIC TURBINE TRIP WITH HPCI UNAVAILABLE	BRUNSWICK1	325	CE	MECFUN	E	O	N	Y	1139	25	821	B	G	UE	CPL	761008
103002	750429	LOFW	MULTIPLE VALVE FAILURES & RCIC INOPERABLE	BRUNSWICK2	324	CC	VALVEX	E	O	Y	N	40	26	821	B	G	UE	CPL	750320
103077	750501	LOCA	RCP SEAL FAILURE	ROBINSON 2	261	CB	PUMPXX	E	O	N	Y	1684	26	700	P	W	EX	CPL	700920
66996	711010	LOCA	TRANSIENT AND BLOWDOWN	MILLSTONE1	245	CC	VALVEX	E	O	N	Y	349	27	660	B	G	EX	NNE	701026
128569	770715	LOCA	SAFETY RELIEF VALVE FAILS TO RESET	BRUNSWICK2	324	CA	VALVEX	E	O	N	Y	281	27	821	B	G	UE	CPL	750320
152563	791002	SGTR	STEAM GENERATOR TUBE BREAK	PRAIRIEIS1	282	HB	HTEXCH	E	O	N	Y	2131	27	530	P	W	PX	NSP	731201
36147	690715	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	HAD.NECK	213	EB	CKTBRK	E	O	Y	Y	752	28	575	P	W	SW	CYA	670724
59484	710112	LOCA	SUMP ISO. VALVES CLOSED	PT.BEACH 1	266	SF	VALVEX	E	T	N	N	71	28	497	P	W	BX	WMP	701102
65757	710205	LOOP	LOSS OF OFFSITE POWER	PT.BEACH 1	266	EA	CKTBRK	G	O	N	Y	95	28	497	P	W	BX	WMP	701102
85370	731021	LOOP	LOOP,EXCESSIVE RCS COOLDOWN,SFTY INJ.&INS BUS	GINNA	244	EA	ELECON	E	O	N	Y	1443	28	470	P	W	GX	RGE	691108
115870	760706	LOCA	IMPROPER INSULATION	VT.YANKEE	271	SF	VALVEX	G	T	N	N	1565	28	514	B	G	EX	VYC	720324
116780	760810	LOOP	GAS TURBINE FAILS DURING PLANT TRIP	MILLSTONE1	245	SF	ENGINE	E	O	N	Y	2153	28	660	B	G	EX	NNE	701026
120443	761101	LOFW	TWO ELECTROMATIC RV FAILURES	QUAD-CTES2	265	SF	VALVEX	C	T	N	N	1650	28	789	B	G	SL	CWE	720426
124222	770408	LOFW	SIX MAIN STEAM RELIEF VALVES FAIL TO LIFT	D.ARNOLD	331	SF	VALVEX	G	T	N	N	1110	28	538	B	G	BX	IEL	740323
63144	710308	LOOP	FAILURE OF BOTH DIESEL GENER. DURING TESTING	ROBINSON 2	261	EE	ENGINE	E	T	N	N	169	29	700	P	W	EX	CPL	700920
81523	730618	LOFW	FAILURE OF AFW PUMPS TO AUTO-START	TKY.POINT4	251	SF	INSTRU	B	O	Y	N	7	29	693	P	W	BX	FPL	730611
115875	760626	LOOP	LOSS OF OFFSITE POWER DURING REFUELING	HAD.NECK	213	EB	RELAYX	H	O	Y	Y	3169	29	575	P	W	SW	CYA	670724
123150	770303	MSLB	50% OPEN ATMOSPHERIC DUMP VALVES	CRYSTALRV3	302	IF	GENERA	E	O	N	Y	48	29	825	P	B	GX	FPC	770114
127384	770712	LOCA	DUMMY SIGNALS INSTALLED ON INSTRUMENTENTATION	ZION 2	304	IA	INSTRU	G	T	Y	Y	1296	29	1040	P	W	SL	CWE	731224
130788	770924	LOCA	STUCK OPEN PORV	DVS-BESSE1	346	CB	VALVEX	E	O	N	Y	43	29	906	P	B	BX	TEC	770812
132927	771129	LOOP	LOSS OF OFFSITE POWER	DVS-BESSE1	346	EB	XXXXXX	E	O	Y	Y	109	29	906	P	B	BX	TEC	770812
148764	790118	MSLB	STUCK OPEN STEAM DUMP VALVES	BVRVALLEY1	334	HB	VALVEX	E	O	Y	Y	983	29	852	P	W	SW	DLC	760510
89205	731130	LRTR	FAILED RCP SHAFT AT 92% POWER	SURRY 1	280	CB	PUMPXX	E	O	N	Y	517	30	822	P	W	SW	VEP	720701
106616	750913	LOOP	LOSS OF OFFSITE PWR & RELF.VALV STKS OPN	PILGRIM 1	293	EB	CKTBRK	D	O	N	Y	1184	30	655	B	G	BX	BEC	720616
120293	761111	LOOP	PLANT SERVICE WATER STRAINERS PLUGGED	HATCH 1	321	WA	FILTER	E	O	N	N	791	30	786	B	G	SS	GPC	740912
141097	780329	LOCA	STUCK OPEN PORV	TMI 2	320	EB	GENERA	B	O	N	Y	1	30	906	P	B	BR	MEC	780328
146744	790112	LOFW	LOSS OF VITAL BUS WHILE AT POWER	DVS-BESSE1	346	EB	GENERA	E	O	Y	N	518	30	906	P	B	BX	TEC	770812
150882	790606	LOOP	BOTH DG'S TRIPPED DURING TEST	CRYSTALRV3	302	EE	ENGINE	G	T	Y	N	873	30	825	P	B	GX	FPC	770114

Table 2. 1980-1981 Precursor Sequences ($P_r \geqslant 10^{-3}$)

SC	ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	I	E	AGEX	RATE	T	V	AE	OPR	CRITXX
18	166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WB	HTEXCH	D	T	N	N	1654	821	B	G	DE	CPL	761008
22	164617	810102	LOOP	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	WA	PIPEXX	E	O	N	N	1904	870	P	C	DX	NNE	751017
24	158860	800419	LOOP	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	ZZZZZZ	H	O	N	Y	918	906	P	B	BX	TEC	770812
24	160846	800226	LOOP	24 VDC TO NON-NUCLEAR INSTRUMENTATION LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	825	P	B	GX	FPC	770114
25	158233	800611	RCPT	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	802	P	C	EX	FPL	760422
25	167611	810211	LOCA	ALL CORE COOLING PATHS LOST	SEQUOYAH 1	327	CB	PUMPXX	X	O	Y	N	221	1148	P	W	UX	TVA	800705
28	171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	906	P	B	BX	TEC	770812
29	163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILURE	ARKANSAS 1	313	CB	PUMPXX	E	O	N	Y	2104	850	P	B	BX	APL	740806

Table 3. Approximate Precentages of Potential Severe Core Damage from Various Precursor Types or Initiators

	Sequence of Initiator Type	1969-1979*	1980-1981
PWR	LOCA	5%	15%
	LOOP	30%	70%
	LOFW	60%	0%
	Other	5%	15%
BWR	LOCA	5%	95%
	LOOP	30%	0%
	LOFW	65%	5%
	Other	0%	0%

*1969-1979 percentages exclude TMI, Browns Ferry Fire, and Rancho Seco NNI incident contributions.

Attachment to Interim Research Information Letter 136 For
The Precursor Program

Detailed Discussion of Accident Sequence Precursor Program

I. Introduction

This interim RIL transmits a summary of what has been found from the precursor program thus far. The bulk of the comment and discussion in this RIL concerns the 1969-1979 data set. Some preliminary discussion is included on the 1980-1981 data set. Much of the work that has been done to date on the precursor program is contained in NUREG/CR-2497, "Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report," published June, 1982 (Ref. 1). This RIL summarizes the substance of what is contained in that report, plus what has been indicated by the various reviews, the EPRI precursor workshop, and the ACRS meetings. In light of the controversy surrounding the 1969-1979 precursor report, this RIL includes discussion on the applicability of the technique, the shortcomings and problems with the method, the validity and meaning of the results, and what is planned to corroborate or substantiate the analysis results.

The Accident Sequence Precursor program was initiated in part because of conclusions contained in the Risk Assessment Review Group Report, NUREG/CR-0400, dated September, 1978 (Ref. 2). This report stated "...unidentified event sequences significant to risk might contribute...a small increment...[to the overall risk.]" The report recommended: "It is important, in our view, that potentially significant (accident) sequences, and precursors, as they appear, be subjected to the kind of analysis contained in WASH-1400...."

The precursor program utilizes nuclear plant operating experience information contained in the Licensee Event Reports (LERs) to estimate the susceptability of the U.S. commercial reactor population to severe core damage for a specified time period, and highlights those accidents or incidents which contribute most to this susceptibility. The precursor program is perhaps the most ambitious program yet undertaken to attempt to provide an alternate method of estimating nuclear power plant risk using operational experience directly in the estimation. In the context of this program, a precursor is considered to be an incident or failure affecting or degrading the plant safety systems or an incident or accident which requires a response from the plant safety systems. The precursor program is sponsored by the NRC and is performed by ORNL and its subcontractor SAI.

The purpose of the precursor program is to develop and apply a formalized, systematic methodology for the evaluation of nuclear plant operational experience data to identify potentially serious accidents or incidents and estimate their occurrence frequency. The methodology uses event tree modeling of plant systems to assess possible plant response to the reported operational experience data. With future methodological development, the precursor program is intended to provide an alternate method, corollary to PRAs, for gaining insights regarding nuclear plant risk using operational experience data more directly in the assessment of functional performance. Thus the precursor methodology is viewed as complementary to the nuclear plant probabilistic risk assessments (PRAs) currently being performed by industry and the NRC. It can provide a check on mathematically modeled nuclear plant probabilistic risk assessments by:

a. Indicating quantitatively which types of accidents or incidents operational experience data shows are most likely at the various NSSS reactor types, allowing a comparison and check on the PRA-derived "dominant" accidents.

b. Indicating significant accident or incident sequences which may be overlooked or missed by PRA analysis. This may be particularly so for sequences driven by common cause failures, or by human "errors of commission."

c. Providing an indication of the trends of system failures, function failures, and the overall industry-wide nuclear plant severe core damage frequency (likelihood) by estimating the frequency over specified periods of time, based on the experience data from the time period.

The operational experience data utilized in the precursor program is contained in the LERs and their abstracts which report incidents requiring safety system operation, or failures or degradation of safety systems. The models used in determining this estimate are generic event trees which are specialized in their quantification to the specific plant being evaluated. The modeling effort required includes determining what plant equipment can or may respond to incidents or accidents and quantifying the likelihood of these systems being unavailable or failing if called upon. The task of determining how and what systems may respond is found mainly from Safety Analysis Reports (SARs). The likelihood of plant safety systems being unavailable or failed is estimated from the precursor (LER) data itself or, where necessary due to lack of data, from generic judgments based on the results of PRAs.

The major limitations of the precursor program are:

a. It is limited mainly to using readily available information such as Safety Analysis Reports. In this regard, for the first report there was insufficient interaction with the licensees to obtain plant-specific information on plant configuration and procedures.

b. It is "reportive" insofar as it indicates estimated severe core damage potential for past periods of time. To be "predictive" the method must be improved to account for "learning" and corrective actions taken because of past incidents or accidents.

c. The models used are generic event trees and generic data intermixed with plant specific data.

d. Other methodology problems of concern include double counting, success/failure criteria, rectification assumptions, and screening criteria.

Approximately 19,400 LER abstracts concerning events that occurred during the 1969-1979 time period were examined for potential accident sequence precursors. This initial examination yielded over 500 LER abstracts (approximately 3%) as potential precursors. LER abstracts (events) were chosen as potential precursors if any of the following criteria were met:

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

2. any instance where two or more failures occurred,

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

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

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

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.

The 500 plus potential precursors were then subjected to an information search for more background. This included a detailed review and in-depth evaluation of the actual LER and supporting information, including a review of the accident sequence (if there was one) as described in the LER. It also included a review of the design of systems in the reactor plant reporting the LER to determine the impact of the failure on the operation of these systems. In addition, it included a review of the plant accident analyses to determine the extent to which affected systems would be required to function for different off-normal and accident conditions. The end result of this second level of much more detailed review was to reduce the 500 plus potential precursors to 169 events which were considered to be precursors. For this study, LER events were selected as precursors if they met one of the following criteria.

1. The event involved the failure of at least one function required to mitigate an initiating event of interest.

2. The event involved the degradation of more than one function required to mitigate an initiating event of interest.

3. The event involved an expected or unusual actual initiating event (e.g., a total loss of offsite power, a stuck-open primary relief valve, or another infrequent event).

For the second precursor report, which assesses 1980-1981 operational data, a similar procedure was followed to that outlined above for the 1969-1979 data. The 1980-1981 operational data set consisted of 8200 LER abstracts, which after the analysis process indicated above reduced down to 56 precursor events of significance. These 56 events are comparable in significance to the 169 events found in the 1969-1979 data set. As a point of possible interest, the proportion of LERs submitted and LERs found significant when compared to reactor years of operation is reasonably consistent for the two reporting periods analyzed.

II. Results and Findings

The first Accident Sequence Precursor report documents 169 operational events which took place in the U.S. during the interval 1969 thru 1979 inclusive, and which the authors considered to be precursors to severe core damage accidents. These events were identified based on a methodical and detailed review of the Licensee Event Report data base for that time period. Each precursor event involving system failure was assessed to account for the possibility that the failure could have been corrected to restore the system to operational status in sufficient time. One of three weighting factors (0.1, 0.5, or 1.0) was applied to each failure to account for the likelihood of failure rectification and system restoration. The resulting modified failure was called the effective failure.

The effective number of system failures and initiating events were used in the report, along with estimated demands (or plant years for initiating events) to produce a generic estimate for certain failure on demand probabilitites and initiating event frequencies. The system failure probabilities and initiating event frequencies developed from the precursor events were used in conjunction with event trees to estimate the probability of subsequent severe core damage during the precursor event. These calculations involved the use of functional event trees, which facilitated the consideration of the different systems found on different plants and the consideration of observed support system failures without specific inclusion of support systems in the event trees. The functional event trees used were, for the most part, trees to represent:

LOFW - Loss of Feedwater
LOOP - Loss of Offsite Power
LOCA - Loss of Coolant Accident
MSLB - Main Steam Line Break

The calculated probabilities of postulated severe core damage were used to rank the precursors in order of severity. Such a ranking was necessary to eliminate events which met the selection criteria but were clearly not important for subsequent trends analysis (e.g., a total system unavailability for a short period of time due to a maintenance error). Trend analyses were performed using the more significant events to indicate the instantaneous failure or occurrence rates of important systems and initiating events with time.

The first precursor report estimates the frequency of severe core damage to be 1.7E-3 to 4.5E-3 per reactor year for the overall reactor population for the period 1969-1979 inclusive. It must be emphasized that the results of this NUREG are applicable only to the 1969-1979 time period. That period was fairly stable in regard to constancy of plant regulation; i.e., there were no incidents or accidents mandating drastic changes to the reactor plant or plant operations until TMI-2. In essence, it is the "quiet" period which is analyzed in the first precursor report.

One of the most substantive results of the first report is the documentation of 169 ranked severe core damage precursors as shown in Table 4.2 of the report. These 169 precursor events have as yet only been evaluated or interpreted qualitatively for their meaning and implications to reactor safety. More work is planned to infer meaning utilizing insights gained through critiques of prior precursor work, and through statistical and engineering insights gained by exercising the ORNL Precursor Sequence Quantification code on sequence variables, functions, or assumptions.

Additional miscellaneous findings in the first report are:

1. Many of the initiating event frequencies and function failure-on-demand probabilities developed from operational event information agree reasonably well (within a factor of 10) with the Reactor Safety Study (Ref. 3) median results.

2. A variation in the rate of occurrence of significant precursors per plant as a function of plant age cannot be justified by the results of this study.

3. Differences do not appear to exist in the number of significant precursors observed between plant types and among reactor vendors, architect-engineers, and plant power ratings.

4. Approximately 38% of all significant precursors involved human error.

5. While the TMI-2, Browns Ferry, and Rancho Seco events are certainly significant, even if these events were excluded the core damage estimates would still be in the high 10^{-4}/RY frequency range.

The second precursor report has these tentative findings. These findings should be considered tentative in light of the fact that the report has not yet undergone peer review.

a. Comparison of this study with 1969-1979 data indicates that the number of precursors per plant year are comparable, but that probabilities associated with the sequences are somewhat lower. This is attributed to:

- a slight decrease in some system unavailabilities
- partial credit given for feed and bleed
- auxiliary feedwater system improvements

b. The combined industry averaged severe core damage frequency for the years 1980-1981 is estimated to have been approximately 2.5 E-4 per reactor year.

III. Evaluation

The Accident Sequence Precursor (ASP) program can, with further development, provide a corollary alternative to Probabilistic Risk Assessments (PRAs) for estimating nuclear power plant risk. It uses a higher level (i.e., system level) of operational experience failure information to more directly obtain the estimates. The results of the first precursor report are applicable only to the 1969-1979 time period and should not be projected to estimate risk for the post TMI-2 period. These results should not be projected because the means of coping with accidents or incidents, in terms of plant design, procedures, and operator training were changed significantly post TMI-2 from what was assumed in the report. Such changes include, for example, provision or formalization of a feed-and-bleed procedure, the lessened A.C. dependence of auxiliary feedwater, and improvements in training and equipment to better measure coolant level or conditions within the reactor. These items which were omitted or given only limited credit in the subject precursor report will have to be accounted for in future precursor reports covering the time period after the TMI-2 accident.

One of the major areas of controversy with the precursor method concerns the use of the probabilities of individual severe core damage sequences to estimate the averaged frequency of severe core damage for the reactor population over the period of interest. The approach is not new (see, for example, Apostolakis and Mosleh, Ref. 4) but the resulting estimate in the first report is higher than is calculated in some PRAs by factors of 10 to 100. It should be emphasized that due to inherent uncertainties in both approaches, the Precursor and PRA frequencies are not necessarily inconsistent. The detailed comparison of the contributors to the two sets of core damage frequencies is only now beginning and will continue for some time. This interim RIL gives some initial insights, and the next RIL will make a more complete comparison.

Part of the precursor program desired end products have been met with the issuance of the first precursor report, NUREG/CR-2497. As was noted in the introduction to this RIL, one of the desired end products of the precursor program is to provide a means for checking or benchmarking nuclear plant PRAs, particularly for confirming the type and frequency of previous accident sequence evaluations and to identify any new or unique accident sequences which were missed by past PRA analyses. Secondly, the program may provide an indication of trends of overall industry-averaged nuclear power plant severe core damage frequency (likelihood). The methodology needs additional development and more detailed modeling to allow a meaningful estimation of a specific plants' level of risk from precursor information. Regarding the first desired end product, that is to provide a means to allow benchmarking of PRA, the program has partially done this through Table 4.2 of the first report. This Table provides a listing by severity of the 169 precursor events evaluated by this study. The Table indicates that LOFW and LOOP type sequences together contribute the largest fraction of the severe core damage frequency irrespective of NSSS type. The Table also indicates that LOCA and MSLB sequences together contribute a small fraction of the severe core damage frequency irrespective of NSSS type. These results confirm the value of reliable AFW, HPI, and electric power as is also indicated by several PRAs.

The precursor program results shown in the first precursor report correspond somewhat to PRA results. PRAs generally seem to indicate that transients (including LOOP) contribute a large fraction of core melt frequency. However PRAs for PWRs also seem to indicate that LOCA (or subsequent failure of long term decay heat removal) contribute a large fraction of core melt risk whereas the ASP indicates it contributes a small fraction of risk. An activity that will be undertaken in future ASP studies will be to include a finer breakdown of plant design details and the susceptibility to certain precursors as opposed to the present broad assignment of ASP to all LWRs or a specific NSSS types. This will involve use of accident delineation models being developed in the Accident Sequence Evaluation Program (ASEP) and the use of a greater level of plant design detail.

A cursory evaluation of Table 5.4 of the second precursor (draft) report for 1980-1981 data, as previously noted in the cover memo to this RIL is not so conclusive, partcularly for BWR events. As an example, there is only one dominant (greater than 10^{-3} probabiliyt) BWR sequence in the 1980-1981 data set. Therefore any meaningful inferences will require a more indepth evaluation of all the 1980-1981 sequences.

It is certainly not clear how one should interpret the severe core damage frequency estimates derived in this study. One can argue that these numbers represent the average from a population of dissimilar reactors. As such the averaged number could be meaningless in regard to specific individual plants if it totally masks the wide range in risk possible. The reactor core damage frequencies might vary among reactors, from being very low to very high, and the very high frequency reactors could be dominating this averaged number. The question of what meaning this number has thus depends on how much variation there is in the risk from individual nuclear plants. Because there is considerable variation in design and operation among various NSSS types, and further variation in balance of plant due to different A&Es, considerable risk variation could exist.

The various peer review and individual comment on the first precursor report ranged from comments or questions concerning the planned intent and use of the output from this program to comments and questions on the methodology or techniques used. Concerning the first type, many commentors indicated that the intent of the program was not made clear in the first precursor report, NUREG/CR-2497. They questioned whether the intent was to get a bottom line best estimate severe core damage risk number for the entire reactor population, or whether it was to gain "insights" from the operational data, or whether it was to complement PRAs at the sequence level and in some way help validate PRAs. (As is noted in the introduction, the program intent is really to do all of these.) Concerning the technical correctness of the report, the most extensive comments as of this date came from INPO. The INPO comments and our assessment of these comments, because they are extensive, are contained in Appendix A. The INPO and other reviewer comments are summarized below. Some of the review reports have not as of this date been formally published (e.g., the SRS report, Ref. 6). The substance of what was said, or submitted, as we interpret the comments are shown below.

INPO (Ref. 5)

The most extensive review of the precursor report was performed by INPO. INPO reevaluated the most probable sequences of the precursor report. They concluded the precursor report severe core damage probabilities could be too high by as much as a factor of 30. The main INPO contentions are that:

- The precursor report event trees do not include all systems which can help mitigate accidents including modifications which were made after the period covered by the precursor study.

- The calculations of system unavailability are too conservative and the failure events selected are not always pertinent.

- Not enough credit was given to the ability of operators to innovate and cope with failures.

A more thorough discussion of the INPO reevaluation is provided in Appendix A.

SRS (Ref. 6)

The SRS draft report indicates that the general methodology adopted by Oak Ridge seems appropriate. They feel that the mathematical methodology has some problems, but that it nevertheless does take some account of the dependencies within the incident data. Experience data sample sizes are sufficient to estimate the contribution to accident sequences and to provide some comment on the synthesis of the accident sequences. The precursor derived estimate of severe core damage accident frequency should be comparable with PRA in some average sense, but uncertainty analysis must be done to assess results. The SRS is one of the few reviewers which hasn't argued for more plant specific event trees. The SRS report indicates that even though they may be much more difficult to quantify, the more generalized event trees are more adaptable to evaluating operational incidents--such as an operator error of commission in which, for example, an operator inadvertently disables an operating safety system.

SAI (Ref. 7)

The SAI independent review indicated that the LER selection methodology was reasonable for screening LERs. It indicated that the precursor derived generic initiating event and system failure probabilities need to be periodically updated, since they will change with equipment modifications. It states that the calculated conditional probabilities of severe core damage should be considered as performance indicators only. Also that the estimated severe core damage per reactor year should be treated only as a semiquantitative indicator of past performance of the nuclear industry. This indicator can be used with subsequent data to show trends. The ASP results can be used for identifying weaknesses in plants, prioritizing corrective actions, and complementing or checking PRAs. It indicated that uncertainties for the numerical estimates should be included in future studies. The SAI review also notes a possible disparity between what seems to be indicated by operational data and by theoretical reliability studies of the AFW system. The SAI review (page 55, bottom) notes that "...more than half of the LERs used in the precursor study to calculate AFW and secondary heat removal failure probability occurred at Westinghouse plants that were ranked in NUREG-0611 as having rather reliable AFW systems...." (NUREG-0611 provided relative rankings of the auxiliary feedwater system's reliability for various PWR plants.) (Refer to Reference 8.)

AEOD (Ref. 9)

AEOD commented on the draft precursor report. They noted that since system unavailabilities at specific plants can vary by as much as an order of magnitude, the core damage measure calculated for each precursor may not be representative of the core damage calculated if plant specific data were used. They note that in regard to the average severe core damage per reactor year, that unless all reactor plants have equivalent risk, the averaging process, because of the magnitude of the numbers, allows the higher risk plants to dominate and this could account for the relatively elevated core damage number. By this argument the averaged severe core damage number is more typical of the worst plants rather than the "average".

NRR (Ref. 10)

NRR had several comments on the draft and on the final report. One NRR group questioned what they felt to be excessive conservatism in the derivation of safety system unavailabilities. They indicated that several reported component and system failures were inappropriate for inclusion, either because they felt they were Pre-Op failures, or associated with shutdown surveillance testing, or because they thought they were recoverable events. NRR also advised that future work should have early involvement of NRR and the licensee in order to assure the correct delineation of accident sequences.

NRR has proposed a method to partially deal with overcounting in the precursor methodology. The precursor report indicates that its calculated severe core damage frequency may be too high by as much as a factor of three. NRR contends (Reference 10, Appendix A, bottom of page 1) that the overcounting may be considerably higher than a factor of three. The NRR proposed method requires more rigorous definition and determination of the constituent parts of each precursor sequence. By the proposed method, each precursor sequence would be decomposed into an observed part and the potential subsequent system or operator failure (complementary part). One problem with the NRR method is that it requires prior event partitioning in order to be unbiased. ORNL is evaluating the proposed NRR method.

Duke Power (Ref. 11)

The Duke Power letter notes that remedial actions taken by plants to reduce the frequencies and/or consequences of precursor events need to be considered in the evaluation. It also notes that the precursor report compares its severe core damage calculations with WASH-1400 which calculated core melt and that there is approximately an order of magnitude difference between these two measures. The letter notes that LERs and FSARs are an insufficient source of information for assessing actual and potential safety impact of precursors, and indicates that additional information should have been used. Specific comments pertain to five precursor events at Oconee units 1 thru 3. These comments note evaluation deficiencies including incorrect assumptions of the duration of the event conditions, misjudgments of the relevance of the event and event conditions, and exclusion of other factors which should have been considered in the evaluation. It notes that because of the discrepancies, the calculated frequencies for the Oconee precursors were significantly overestimated.

Dairyland Power (Ref. 12)

The Dairyland Power letter relates to three Loss of Offsite Power events which occurred at the LaCrosse BWR and which were included in the precursor report. The letter notes that there are major errors in the analysis of these events and indicates that this is due to lack of understanding of the LaCrosse systems. Specifically, the letter indicates that erroneous precursor report assumptions were "...(1) the shutdown condensor and condensate pumps were the equivalent of RCIC and HPCI, and (2) operation of the shutdown condensor is dependent on the condensate pumps." The letter states that the compilation of errors made in the analyses of the Loss of Offsite Power sequences for LaCrosse totally invalidates those analyses.

ORNL (Ref. 13)

This Oak Ridge National Laboratory letter answers the F. Linder letter (Ref. 14) and acknowledges errors made in three precursor event trees delineating systems and system capabilities at the LaCrosse plant. Oak Ridge indicates that the error would not affect the quantification of two of the three precursors, but would reduce one of the three from significant to insignificant, i.e., from probability of core damage of 1.8E-2 down to approximately 1.2E-4.

ASA (Ref. 14)

A report from the ASA Ad Hoc Committee on Nuclear Regulatory Research which was evaluating the precursor methodology will not be received because the funding for the Committee has been terminated. We do expect to receive individual reports at some future date from three Ad Hoc committee members who had been studying the precursor methodology before the groups termination.

Univ. of MD (Ref. 15)

A review and assessment of the precursor report and its methodology is being performed by the University Research Foundation, College Park, Maryland under contract with Oak Ridge National Laboratories. This contracted review is expected to be completed in autumn 1983. Their draft report notes that the precursor report clarity and presentation could be considerably improved for ease of understanding. In a check of the precursor report calculations, it notes that no systematic calculational errors were revealed. It notes that in isolated instances the numerical results presented in the report could not be verified but the identified instances do not impact on the major findings of the precursor report. It also noted that the objectives of the report are not clearly stated. This review questions the validity of comparing precursor results with PRA results until uncertainty bounds are given for the precursor results.

EPRI Workshop (Ref. 16)

The two day EPRI Precursor Workshop was divided up into five sessions plus a meeting summary session. The minutes of this workshop have not yet been published. The sessions dealt with the following issues:

1. Why perform precursor studies and what are reasonably deliverables from such studies?

2. What constitutes an acceptable LER selection and screening process for a precursor study?

3. What is the appropriate level of modeling detail needed to adequately utilize precursor information?

4. What quantification can be done on selected LERs?

5. How can precursor information be integrated into an industry-wide performance measure?

ACRS Meetings

The following ACRS meetings addressed and considered the precursor program and problems with the program.

2/9/83 ACRS Subcommittee Meeting on Reliability and Probabilistic Assessment. This three hour meeting presented and reviewed the precursor program, its methodology and intended results. Also included was a brief presentation of the known precursor problems and issues as of that date.

3/9/83 ACRS Subcommittee Meeting on Reliability and Probabilistic Assessment. This all day meeting was an expansion of the earlier meeting. In this meeting the results of the EPRI precursor workshop were reviewed, and INPO representatives presented a summary of the findings from their extensive review of the precursor report. The results and conclusions of NUREG/CR-2497 were discussed along with the status of LER review for the second (1980-1981) precursor report. NRR representatives briefly discussed NRC's plans for NREP, the use of plant specific PRAs, and the use of insights gained from precursor analysis in future risk assessment analysis.

4/16/83 ACRS Full Committee meeting. Because of overruns of other ACRS agenda items, the Precursor presentation and discussion was limited to an approximately half hour overview of the precursor methodology and results.

ACRS Letter (Ref. 18)

The ACRS provided their findings regarding the Accident Sequence Precursor Program and NUREG/CR-2497 in a letter to W. J. Dircks. They were generally supportive of the program effort and provided a good outside perspective of the program's limitations, its possible immediate and potential uses or applications, and suggestions for the methodological problems which should be addressed to make the program more effective. We have abstracted the following points from the ACRS letter:

- Concerning the possible factor of three overcounting acknowledged in the report--other methods of analyzing the data have been suggested but none appears to be unequivocally "the right one" and the actual degree of "overcounting" remains difficult to quantify.

- Significant differences exist between the precursor report sequence probability calculations and the re-calculations made by INPO. "...subjective judgment plays a considerable role in the ASP study, as it does in PRA in general, and large differences are likely to remain even if the same detailed event trees are employed by each group."

- Although desired by many, the money and resource costs of doing plant specific analysis would be far larger than presently expended in the ASP study. Even plant specific models would be forced in some instances to use generic data, and there is no universally accepted procedures for doing this. There appears to be merit in both plant-specific and generic interpretation of operation experience, when each is properly executed.

- Future efforts should include a careful evaluation of uncertainties and sensitivities.

- The analysis was an estimate of core damage likelihood for 1969-1979 and does not try to predict future risk.

- Future ASP program work should have better defined, albeit still flexible objectives.

- There is a class of information which the AEOD and SEE-IN programs may be treating only in part and which the ASP largely has not considered. This includes evaluating maintenance practices, adequacy of plant procedures, and effects of plant aging on safety, etc.

- The objectives of AEOD, SEE-IN and ASP should be coordinated to eliminate significant gaps in the combined effort.

- Strong interaction is needed between PRA and ASP.

- The importance of the qualitative aspects of the evaluation of operational data and experience must continue to be emphasized. Study efforts should examine the chain of events in important incidents in terms of root cause.

The ACRS letter also indicated that attention should be given to certain methodological issues. These are repeated below verbatim from the ACRS letter:

- Single failures should not be discarded automatically, as single failures of high frequency might be symptomatic of bad maintenance practices, or plant vulnerability to "external events" and other common causes or "bypass mechanisms." In particular, the screening process should be altered so that the ASP program includes single failures of on-line systems of special importance to safety.

- The ASP Study considered only precursors which could lead to core damage, so that experience with containment was not a factor. Such experience would, of course, become relevant were one to attempt to evaluate public risk. The ACRS recommends that the scope of future studies be extended to cover mitigation systems such as the containment.

- Sensitivity and uncertainty evaluation should be included as an integral part of the ASP program itself. A special effort should be made to identify the principal causes of uncertainty.

- Since there appears to be no single, agreed-upon way to perform the quantitative evaluation of risk, either in relative or in absolute terms, the evaluation should be attempted in several alternative ways and studies made to explain the key reasons for differences which may materialize. The issue of "blending" plant-specific and generic data is one area in which one could gain beneficial insight by pursuing such a "multifaceted" approach.

- An evaluation should be performed and published to make clear the reasons for the different results obtained for conditional severe core damage probability by the ASP Study and the INPO critique (INPO 82-025) for the 52 significant precursors.

- The handling of human error probability is highly subjective. It is potentially a large source of difference in the estimates developed. The matter should be specifically reviewed from several points of view, including: the usefulness of a common mode of treatment as one alternative in each such study to place similar studies on a comparable basis; the development of failure models more applicable to real occurrences; and the inclusion of errors of commission in the event trees.

Attachments:
1. Appendix A
2. Appendix B

Appendix A - Comments on INPO's Review of the Precursor Report

A major peer review of the precursor report was performed by INPO (reference report INPO 82-025). The INPO reviewers contend that credit was not properly given to the systems available to cope with accidents or incidents and that the sequence event trees were not properly quantified. Listed below is our assessment of this INPO review. Some of our comments on the INPO review are tentative insofar as the 1969-1979 precursors will be specifically reassessed in the next precursor report to address the concerns raised by INPO and other reviewers of the precursor report.

1. INPO continually perceives the precursor report as an estimator of future core damage frequency. The fact that this is not the case was clearly identified in the report (including the summary).

2. INPO concludes that the probabilities of subsequent severe core damage associated with the Browns Ferry fire and the Rancho Seco bus failure are low (this is significant in their factor of 30 difference, see below).

3. INPO has included in their reassessment additional credit for failure rectification and for the use of alternate mitigation paths.

4. INPO contends that the precursor report overestimates AFW failure probability by a factor of 5 and Automatic Depressurization System (ADS) failure probability by a factor of 9.

5. INPO does not consider ATWS or potential failures during long term core cooling.

One result of the above is an INPO conclusion that the precursor report estimates are a factor of 30 too high. This factor is derived, apparently, by excluding TMI-2 from consideration and then taking the ratio of the sum of the remaining significant precursor probability estimates. (At least that approach yields a factor of 30). However, we believe this factor of 30 is not applicable to the overall core damage frequency calculated in the report, for at least the following two reasons: (1) only probabilities associated with initiating events should be considered in calculating a core damage frequency (otherwise the number of initiating events are overestimated) and (2) since loss of main feedwater events are not reportable in the LER system, they would have to be accounted for differently (a classic event tree approach was used.)

From the precursor report, three core damage frequencies can be estimated: (1) using all precursor events involving initiators and also considering the contribution due to losses of feedwater, (2) excluding the TMI-2, Browns Ferry fire and Rancho Seco bus failure events but otherwise the same as the first case, and (3) by simply taking the initiating event frequencies and function failure probabilities developed prior to the subsequent severe core damage probability calculations and using them in a classic event tree approach (thus excluding actual failure combinations which existed during the historic events).

A calculation of severe core damage frequency based on a classic event tree approach and using the function failure probabilities and initiating event frequencies derived from the precursors resulted in an estimate of $\sim 4 \times 10^{-4}$/plant year for the 1969-1979 plant population. As a comparison, this approach was used to calculate PWR core damage frequency using both the ASP failure numbers and the INPO modified failure values. The resultant frequencies are $\sim 6.7 \times 10^{-4}$ (precursor report) and $\sim 1.5 \times 10^{-4}$ (INPO), a factor of about four different. The credit INPO took for the use of the condensate system for steam generator cooling following an unrecoverable loss of main feedwater would increase this factor somewhat, perhaps to ten. ORNL questions how much of this credit is legitimate in the pre-TMI period.

For comparison, the precursor report numbers and the INPO review numbers for the other cases mentioned above are assembled in the table below.

Event	Precursor Report	INPO
TMI-2	1/432	.27*/432
Browns Ferry	.39/432	3.6E-3/432
Rancho Seco	.25/432	3.6E-4/432
Remaining Significant Precursor Associated with Initiating Events	$\sim 4 \times 10^{-4}$	$\sim 5 \times 10^{-5}$
LOFW Contribution	$\sim 4 \times 10^{-4}$	$\sim 7 \times 10^{-5}$**

Using the probability/frequency data listed above one may tabulate frequencies of severe core damage for the 1969-1979 events for each of the three cases and compare ASP and INPO results.

Core Damage Frequency	ASP	INPO
Case 1 (All initiators and LOFW)	4.5×10^{-3}	7.6×10^{-4}
Case 2 (Same as 1 except delete TMI-2, BF and RS)	8×10^{-4}	1.2×10^{-4}
Case 3 (Classic Event Trees, PWRs only)	6.7×10^{-4}	1.5×10^{-4}

The differences between the ASP and INPO results are approximately a factor of 4-7 in each case.

*INPO expands the period of observation for the TMI-2 event to ~1500 reactor years without including any additional operating experience. The value of .27 was used as the effective "impact" of TMI-2 within a 432 year period. (See the Hartung paper Ref. 17 for a listing of additional events within the 1500 reactor year period.)

**This was calculated using INPO derived AFW and Automatic Depressurization System (ADS) values and assuming no failure to trip or long term cooling failures.

The INPO response focused upon improvements which would result in lower failure frequencies than those calculated in the precursor study, if plant-specific event trees were used and assuming decreased repair times. The INPO study made no attempt to find additional precursors, despite contending that the data base for the study should be extended from the 432 U.S.-LWR reactor years on which the ASP study was based to some 1500 years of free world commercial reactor operation. Thus, although the INPO review pointed to some improvements that can be made in precursor-type studies, the suggestions were not followed through to their ultimate conclusions.

Appendix B - Planned Work for Next Report

ORNL has done considerable work on LER Accident Sequence Precursors, has published and widely distributed this work, and is now doing the more demanding time-changing LER analysis of post TMI-2 events. They have received substantial comment and feedback from reviewers of the precursor report, NUREG/CR-2497. EPRI has held a workshop to review and consider the problems and issues of the precursor methodology. Much useful information was obtained from the workshop and from various individuals and peer review groups. ORNL is assessing this feedback and will as a result of the feedback be in a better position to resolve some of the precursor problems. Certainly the more limited data presented by the 1980-1981 LERs along with the fact that this data comes from a reactor population whose equipment configuration, operating procedures, and regulatory requirements were changing functions of time during this interval requires changes in the methods and techniques of analysis. Whereas the LER operational experience data from the period 1969-1979 could reasonably be considered as somewhat constant, the same assumption is not adequate for the 1980-1981 LER data.

There are several things which may enhance ORNLs abilities to better address these problems. One of these is that ORNL has developed a means to encode all the event trees of those LERs of possible significance. In the last precursor report each event tree was quantified manually and all event sequences of less than 10^{-3} probability were discarded because they were felt to be noncontributory. That method of quantifying sequences on an individual "one-by-one" basis has the potential to overlook systems or parameters which may be important and which may only be found when the sequences are looked at as a group. The ORNL developed code is adaptive to using various importance measures as well as doing extensive parametric studies. It is intended that the next report will indicate the sensitivity of the calculations to the various parameter assumptions made and also indicate the expected ranges on the point estimate calculations.

ORNL has been requested to reevaluate the 1969-1979 (and later) sequences by computer. These computer studies may help resolve the dilemma of intermixing generic and plant specific data. Also planned for inclusion in this forthcoming report is a comparison of the precursor calculated severe core damage for this interval versus that for the 1969-1979 interval. A draft of the next report covering 1980-1981 LER experience has been distributed for review. It is expected a final version of the 1980-1981 precursor report will be distributed by January 31, 1984.

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RES Files
Subject File No. R-2912.01.04

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NUREG No. NUREG/CR-2497
Docket No.
Rulemaking No.
Other RIL 136
Return NRC-318 to RES, Yes ✓ No___ NRC PDR

OCT 14 1983

MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Robert B. Minogue, Director
Office of Nuclear Regulatory Research

SUBJECT: INTERIM RESEARCH INFORMATION LETTER 136 FOR THE ACCIDENT SEQUENCE PRECURSOR PROGRAM

A. Background and History of the Precursor Program

The Precursor Program has been ongoing since 1979. It was instituted because of the need to assess, in a disciplined manner, operating experience, and in part because of the findings of the Risk Assessment Review Group (Lewis Committee) in their review of WASH-1400, the Reactor Safety Study. The Lewis Committee report (NUREG/CR-0400, September 1978) indicated that reactor plant operational data should be evaluated by the kind of analysis contained in WASH-1400. This RIL summarizes the status and findings of the precursor program to date. Additional information is found in the attachment to this RIL.

At the start of the program the objective was purposely left vague to allow experimentation and evolution of useable techniques. The objective was generally to examine nuclear plant operational experience data and assess plant safety as it is reflected by operational experience. The operational experiences of interest were challenges to plant safety systems (initiating events), total failure of a plant safety system or function, or partial failures of multiple plant safety systems.

In early 1981 the first evaluation of operational data for precursors was assembled into a draft report and circulated within NRC and selectively outside of NRC for comment. This was followed by a more formal draft in early 1982 which was similarly distributed for comment.

Comments on the above two drafts were, where pertinent, factored into the 1969-1979 Precursor Report (NUREG/CR-2497), a status report which was widely distributed in July 1982 in order to obtain broader review. The report was controversial in that it indicated a comparatively high (greater than 10^{-3} per reactor year) industry averaged severe core damage estimate for the eleven year interval evaluated.

Substantial individual and peer group reviews of the report were begun in the summer of 1982. The most extensive of these was the review undertaken by the Institute for Nuclear Power Operations (INPO) in the summer of 1982. INPO assembled a team from within INPO, EPRI, the Nuclear Utilities, and private

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paucity of the data base for the ASP will make it very difficult to identify such trends with much confidence at this time. However, the preliminary results of the 1980-1981 analysis indicates a core melt frequency for that period of approximately 2.5 E-4 per reactor year, which might cautiously be interpreted as indicating a gross downward trend.

The published PRA predictions are in the main obsolete. Many plant owners have taken steps to suppress dominant sequences identified in the PRA. New considerations (e.g., seismic risk or modelling of cooling by feed-and-bleed) could either increase or decrease the predicted frequencies of severe core damage. One must look with caution at the ASP derived frequencies for reliable and useful insights. Table 5.1 of NUREG/CR-2497 compares system unavailabilities and initiating event frequencies derived from ASP with WASH-1400 predictions. It indicates a fairly good comparison. We believe that the correlated (common cause/mode) failures and previously mentioned subtle interactions which are not now capable of being modeled by PRAs are what cause significant differences in predictions, as well as the other inherent uncertainties in PRA.

C. Tentative Conclusions

Our preliminary conclusion is that the industry wide expected core melt frequency probably has decreased from the period 1969-1979 to the period 1980-1981, but the amount of decrease cannot be quantified with much confidence. This is not surprising in light of the reactor safety activities which have taken place since the TMI accident. The 1980-1981 precursor analyses do not indicate new accident sequences, or changes in sequence occurrence frequency which would require prompt remedy. However, we should emphasize the need for caution regarding this conclusion due to the preliminary nature the 1980-1981 precursor report and the methodological limitations of this work, which at this time should be viewed much in the nature of a "quick look" report.

We suggest that NRR and I&E review the precursors to determine that the scope of their safety programs and regulatory activities cover the areas of plant design and operation found to be most important. As has been indicated, we are analyzing the precursor data further and are actively seeking critiques of the draft 1980-1981 report. We will keep you informed as progress is made in this program.

Original signed by:
ROBERT B. MINOGUE

Robert B. Minogue, Director
Office of Nuclear Regulatory Research

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NOTE: This is in response to RES-002401.

cc w/encl:
R. C. DeYoung, IE
C. J. Heltemes, AEOD
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