April 5, 1990

NOTE FOR: Richard C. Robinson
Probabilistic Risk Assessment Branch
Division of Systems Research
Office of Nuclear Regulatory Research

FROM: Marie A. Pohida
Risk Applications Branch
Division of Radiation Protection and Emergency Preparedness
Office of Nuclear Reactor Regulation

SUBJECT: LOW POWER AND SHUTDOWN ACCIDENT FREQUENCIES PROGRAM SENIOR CONSULTING GROUP (SCG) MEETING, MARCH 14 AND 15, 1990

Enclosed are my comments regarding the above referenced meeting that are to be incorporated in the SCG group comments. If you should have any questions please contact me at X21063.

cc. RBarrett
AEl-Bassioni

Comments

1. To compute component maintenance unavailability, the time between placing the safety tags on and off the equipment must be used. This time interval may be much greater than the actual equipment repair time, so beware of using the plant maintenance log books.

2. Using NPRDS to compute maintenance unavailabilities should be performed with caution due to the inconsistency of utility reporting and the special nature of the shutdown operations.

3. Daily plant events reportable under 10CFR50.72 ("50.72 events") should be reviewed to evaluate recent abnormal occurrences at shutdown. Short term losses of shutdown cooling are frequently reported as 50.72 events. The AEOD Core Damage Precursor Studies should also be consulted as a "sanity check".

4. A phased approach to this Shutdown PRA is preferred because of the complexity of the problem. This first phase should concentrate on finding accident scenarios and identifying significant human errors. Sensitivity studies should be used to identify those human actions that should/may require refined models.

5. To compute or verify the time duration of each shutdown phase within each mode, the data base from which the AEOD "Grey Book" is compiled can be used. The time duration of these shutdown phases is presented in this data base. We suspect that this information may be plant specific. This reference can be used to assess the variability of phase duration between Surry and Grand Gulf as compared to other plants.

6. Because human errors will constitute a large portion of the core damage scenarios, Event Sequence Diagrams (ESDs) would be a useful tool to present the individual accident sequences. ESDs would be especially useful for the reactor operators who do not have PRA experience to be able to provide input during the HRA process.

SUMMARY OF GRAND GULF FULL POWER PRA

DONNIE W. WHITEHEAD
SANDIA NATIONAL LABORATORIES

LOW POWER AND SHUTDOWN ACCIDENT FREQUENCIES
PROGRAM SENIOR CONSULTING GROUP MEETING

MARCH 14 AND 15, 1990

ALBUQUERQUE, NEW MEXICO

A/17

INITIATING EVENTS

T1:	LOSS OF OFFSITE POWER TRANSIENT	0.11
T2:	TRANSIENTS WITH LOSS OF POWER CONVERSION SYSTEM (PCS)	1.62
T3A:	TRANSIENTS WITH PCS INITIALLY AVAILABLE	4.51
T3B:	TRANSIENTS INVOLVING LOSS OF FEEDWATER (LOFW) BUT WITH THE STEAM SIDE OF THE PCS INITIALLY AVAILABLE	0.76
T3C:	TRANSIENTS CAUSED BY AN INADVERTENT OPEN RELIEF VALVE (IORV) ON THE REACTOR VESSEL	0.14
TIAS:	TRANSIENT CAUSED BY LOSS OF INSTRUMENT AIR	8.1E-4

SYSTEMS MODELED

HPCS	-	HIGH PRESSURE CORE SPRAY
RCIC	-	REACTOR CORE ISOLATION COOLING
CRD	-	CONTROL ROD DRIVE
SLC	-	STANDBY LIQUID CONTROL
ADS	-	AUTOMATIC DEPRESSURIZATION
CDS	-	CONDENSATE
LPCS	-	LOW PRESSURE CORE SPRAY
LPCI	-	LOW PRESSURE COOLANT INJECTION
SSW X-TIE	-	STANDBY SERVICE WATER CROSSTIE

SYSTEMS MODELED (CONCLUDED)

EVS - EMERGENCY VENTILATING SYSTEM

IAS - INSTRUMENT AIR SYSTEM

RPS - REACTOR PROTECTION SYSTEM

SPMU - SUPPRESSION POOL MAKEUP

SBGT - STANDBY GAS TREATMENT

CI - CONTAINMENT ISOLATION

H_2I - HYDROGEN IGNITERS

EVENT TREES

- LARGE LOSS OF COOLANT ACCIDENT (LOCA)
- INTERMEDIATE LOCA
- SMALL LOCA
- SMALL-SMALL LOCA
- LOSS OF OFFSITE POWER (LOSP)
- STATION BLACKOUT
- LOSS OF POWER CONVERSION SYSTEM

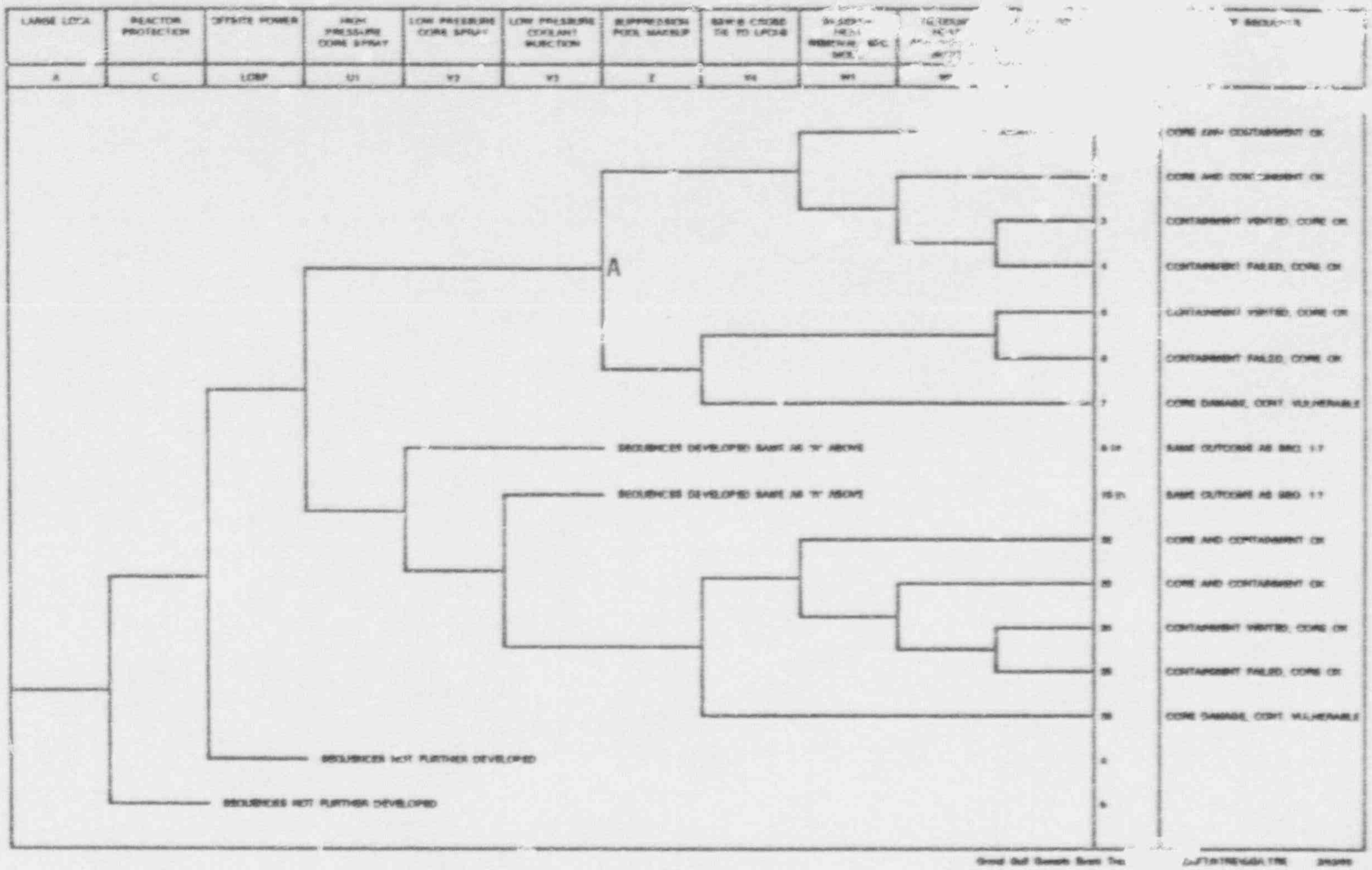


Figure 4.4-1. Large LOCA Event Tree.

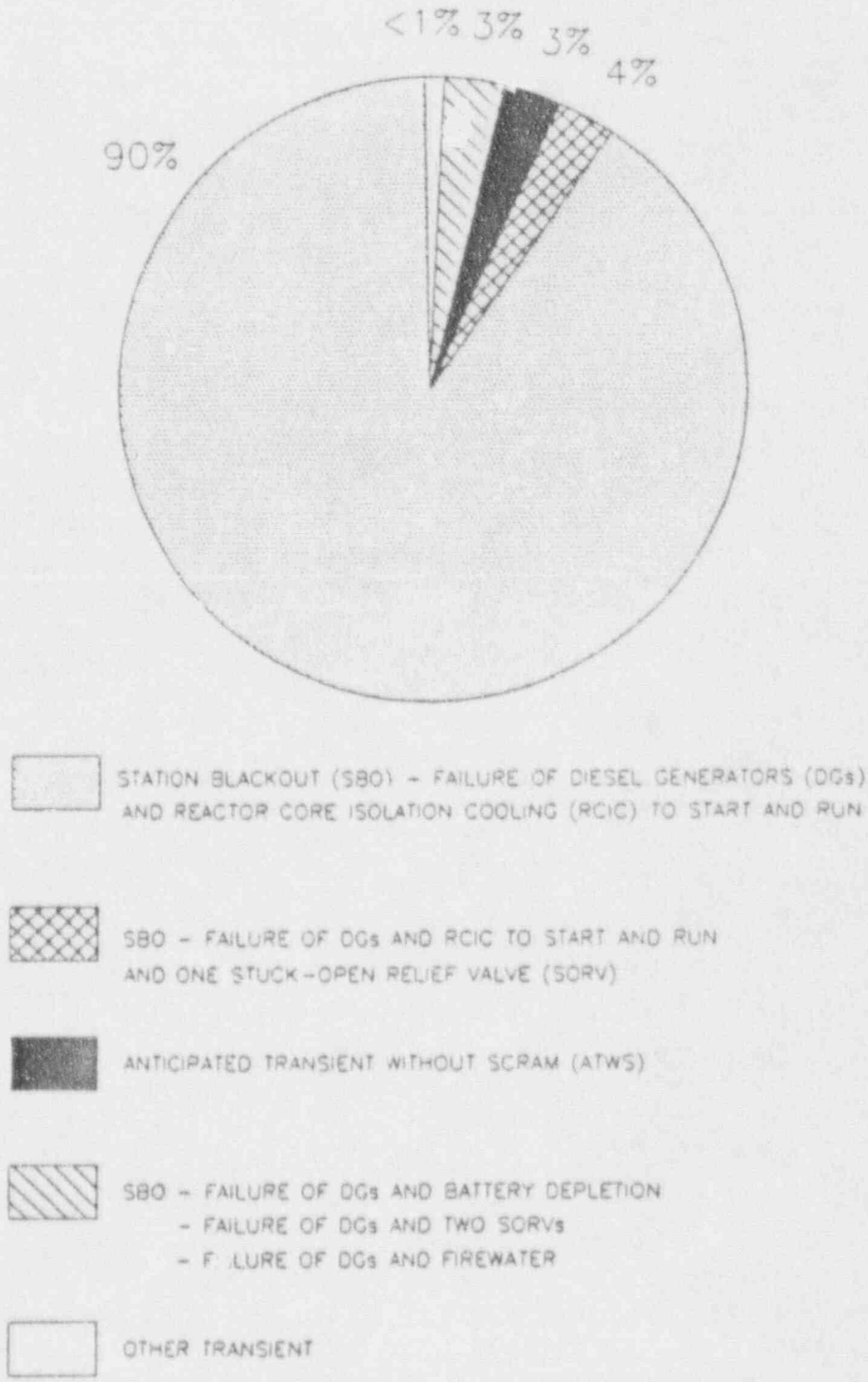


Figure 1-1. Grand Gulf Core Damage Frequency Types.

MOST SIGNIFICANT RISK REDUCTION EVENTS

- FAILURE TO RECOVER OFFSITE POWER
- FAILURE OF THE RCIC SYSTEM TURBINE DRIVEN PUMP TO RUN
- FAILURE TO RECOVER THE DIESEL GENERATORS (DGs)
- FAILURE OF THE DGs TO START
- COMMON CAUSE FAILURE OF THE BATTERIES
- COMMON CAUSE FAILURE TO START OF TWO DGs

DOMINANT COMPONENT FAILURE DATA UNCERTAINTIES CONTRIBUTING TO THE UNCERTAINTY IN THE CORE DAMAGE FREQUENCY

- FAILURE OF THE DGs TO RUN
- FAILURE OF THE DGs TO START
- COMMON CAUSE FAILURE OF THE DGs

METHODOLOGY FOR BWR LOW POWER AND SHUTDOWN PROGRAM

DONNIE W. WHITEHEAD
SANDIA NATIONAL LABORATORIES

LOW POWER AND SHUTDOWN ACCIDENT FREQUENCIES
PROGRAM SENIOR CONSULTING GROUP MEETING

MARCH 14 AND 15, 1990

ALBUQUERQUE, NEW MEXICO

1. OBJECTIVES

- TO ASSESS THE FREQUENCIES OF SEVERE ACCIDENTS INITIATED DURING PLANT OPERATIONAL MODES OTHER THAN FULL POWER OPERATION FOR A COMMERCIAL BWR - GRAND GULF UNIT 1 (STUDIED IN NUREG-1150)

- TO COMPARE RESULTS OF THIS STUDY:

  - ESTIMATED CORE DAMAGE FREQUENCIES
  - IMPORTANT ACCIDENT SEQUENCES
  - OTHER QUALITATIVE AND QUANTITATIVE RESULTS

  WITH THOSE ACCIDENTS INITIATED DURING FULL POWER OPERATION (AS ASSESSED IN NUREG-1150)

- TO DEMONSTRATE METHODOLOGIES FOR EVALUATING FREQUENCIES OF ACCIDENTS INITIATED DURING PLANT OPERATING MODES OTHER THAN FULL POWER

3. TECHNICAL APPROACH CONSISTS OF 13 STEPS

1. IDENTIFICATION OF PLANT OPERATIONAL MODES (POMs)
2. DETERMINE APPLICABLE INITIATING EVENTS (IEs) FOR EACH POM
3. DETERMINE APPLICABLE SYSTEMS AND SUCCESS CRITERIA FOR EACH POM AND IE
4. DEVELOP EVENT TREES FOR EACH INITIATING EVENT GROUP
5. CONSTRUCT SYSTEM FAULT TREES
6. DEVELOP NON-FULL-POWER DATA BASE (a big effort)
7. CONDUCT DEPENDENT FAILURE AND HUMAN RELIABILITY ANALYSES

1. IDENTIFICATION OF PLANT OPERATIONAL MODES (POMs)

- MODE 1 - POWER OPERATION (LOW POWER < 15%)
- MODE 2 - STARTUP
- MODE 3 - HOT SHUTDOWN
- MODE 4 - COLD SHUTDOWN
- MODE 5 - REFUELING

3. DETERMINE APPLICABLE SYSTEMS AND SUCCESS CRITERIA FOR EACH POM

- SYSTEM OPERABILITY DEPENDENT ON INITIAL CONDITIONS OF PLANT (E.G., RCS PRESSURE/TEMPERATURE, AVAILABILITY OF STEAM, ETC.)

- CONSIDER SAFETY SYST_MS AS WELL AS NONSAFETY SYSTEMS

- THERMAL HYDRAULIC CONSIDERATIONS (don't anticipate having to do very many recalculations) [illegible] model they have available in LTS.

5. CONSTRUCT SYSTEM FAULT TREES

- MODIFY EXISTING FAULT TREES (FTs) BASED ON LOW POWER/SHUTDOWN SUCCESS CRITERIA

- CONSTRUCT NEW FTs FOR SYSTEMS NOT MODELED IN GRAND GULF ANALYSIS (expect to be using IRRAS 3.5 — tutorial class will be given by INEL

- FTs SHOULD INCLUDE ALL MAJOR COMPONENTS

- FAILURE MODES AND EVENTS SHOULD NOT BE GROUPED

  - EXTERNAL EVENTS INTERFACE
  - IMPORTANCE CALCULATIONS
  - DEPENDENCIES AND CORRELATIONS

- probably MORE DETAIL SHOULD INCREASE DEFENSIBILITY AND SCRUTABILITY

7. CONDUCT DEPENDENT FAILURE AND HUMAN RELIABILITY ANALYSES

- DEPENDENT FAILURE ANALYSIS EXAMINES 3 CATEGORIES OF DEPENDENT FAILURES
  - DIRECT FUNCTIONAL DEPENDENCIES
    - INCORPORATED IN THE FAULT TREE MODELS AND INCLUDE SUCH THINGS AS EFFECTS OF INITIATING EVENTS, SUPPORT SYSTEM DEPENDENCIES, AND SHARED-EQUIPMENT DEPENDENCIES
  - COMMON CAUSE FAILURES
    - LIMITED TO WITHIN SYSTEM AND "LIKE" COMPONENTS
    - BETA FACTOR APPROACH

8. QUANTIFY ACCIDENT SEQUENCES

- SOLVE SYSTEM FAULT TREES USING IRRAS (OR OTHER SOFTWARE)
- COMBINE SYSTEM FAULT TREES AS PERSCRIBED BY THE ACCIDENT SEQUENCE EVENT TREES
- SOLVE ACCIDENT SEQUENCES BY GENERATING CUT SETS AND POINT ESTIMATES (SCREENING QUANTIFICATION)
- TURNCATE SEQUENCES BELOW 10^{-8}
- IDENTIFY AND ADD RECOVERY ACTIONS TO CUT SETS
- REEVALUATE PROBABILITIES FOR DOMINANT FAILURE EVENTS
- RECALCULATE POINT ESTIMATES FOR THE SURVIVING ACCIDENT SEQUENCES (FINAL QUANTIFICATION)

10. PERFORM A FIRE ANALYSIS

- IDENTIFY FIRE ZONES
- DETERMINE INITIATING EVENT FREQUENCIES FOR FIRES
- SCREEN FIRE ZONES
- VERIFY IMPORTANT FIRE ZONES
- DETERMINE RESPONSE OF EQUIPMENT IN REMAINING FIRE ZONES
- FINALIZE FIRE SEQUENCES

12. CONDUCT AN UNCERTAINTY ANALYSIS

- LHS/TEMAC USED TO CALCULATE UNCERTAINTIES
  - FLEXIBLE UNCERTAINTY DISTRIBUTIONS
  - CORRELATION OF EVENTS POSSIBLE
  - GENERATES IMPORTANCE CALCULATIONS
- UNCERTAINTY CALCULATIONS FOR:
  - EACH ACCIDENT SEQUENCE SURVIVING SCREENING
  - EACH PLANT DAMAGE STATE
  - TOTAL CORE MELT MODEL

4. SUMMARY

- METHODOLOGIES USED WILL BE CONSISTENT WITH CURRENT LEVEL 1 TECHONOLOGY
- PROVIDE ESITIMATE OF CORE DAMAGE FREQUENCY AS A RESULT OF INTERNAL EVENTS FOR MODES OF OPERATION OTHER THAN FULL POWER
- CAN BE COMPARED WITH NUREG-4550

PLANT OPERATIONAL MODES CHARACTERIZATION

BWR LOW POWER AND SHUTDOWN ACCIDENT FREQUENCIES PROJECT

TANIA M. HAKE

SANDIA NATIONAL LABORATORIES

SENIOR CONSULTING GROUP MEETING

MARCH 14-15, 1990

They are still working on completing the chart

GRAND GULF OPERATING MODES/CONDITIONS

PARAMETER	1. POWER OPERATION	2. STARTUP	3. HOT SHUTDOWN	4. COLD SHUTDOWN	5. REFUELING
MODE SWITCH POSITION	RUN	STARTUP/HOT STANDBY	SHUTDOWN	SHUTDOWN	SHUTDOWN OR REFUEL
AVERAGE REACTOR COOLANT TEMPERATURE	ANY TEMPERATURE (< 560 F)	ANY TEMPERATURE (< 560 F)	T > 200 F	T ≤ 200 F	T ≤ 140 F
STEAM DOME PRESSURE	p < 1064.7 psig (Trip Setpoint) p < 1045 psig (except transients) p > 800 psig (MSL pressure)	p < 1064.7 psig (Trip Setpoint) p < 1045 psig (except transients)			ATMOSPHERIC
CRITICALITY					
% RATED THERMAL POWER	(Switch to Run between about 10 - 15 % power)	P < 15%	0%	0%	0%
OTHER				- Refueling Interlocks apply.	- Refueling Interlocks apply.

TECHNICAL SPECIFICATION OPERABILITY REQUIREMENTS

- CHARACTERIZE POMs IN TERMS OF SYSTEM OPERABILITY

- CONSIDERATION OF SAFETY SYSTEMS AS WELL AS "NON-SAFETY" SYSTEMS FOR WHICH SAFETY CREDIT MAY BE GIVEN

- DEVELOPMENT OF TECHNICAL SPECIFICATION SYSTEM OPERABILITY REQUIREMENTS TABLE, TO INCLUDE:

    - SYSTEMS INCLUDED IN FULL-POWER STUDY

    - SYSTEMS NOT fully modeled ~~CONSIDERED~~ IN FULL-POWER STUDY BUT WHICH MAY BE RELEVANT FOR THE SHUTDOWN MODES

- DETERMINE TIME DURING WHICH PLANT IS OUTSIDE TECHNICAL SPECIFICATIONS SO THAT THIS TABLE CAN BE USED TO CHARACTERIZE ACTUAL SYSTEM AVAILABILITY

GRAND GULF TECHNICAL SPECIFICATION SYSTEM OPERABILITY REQUIREMENTS

TABLE OF CONTENTS, cont'd

PAGE

SYSTEMS NOT fully modeled ~~INCLUDED~~ IN FULL-POWER STUDY

PRELIMINARY SYSTEM AND SUCCESS CRITERIA LIST

DONNIE W. WHITEHEAD
SANDIA NATIONAL LABORATORIES

LOW POWER AND SHUTDOWN ACCIDENT FREQUENCIES
PROGRAM SENIOR CONSULTING GROUP MEETING

MARCH 14 AND 15, 1990

ALBUQUERQUE, NEW MEXICO

PRELIMINARY SYSTEM LIST
(CONCLUDED)

SYSTEM	MODES 1	2	3	4	5
SSW	X	X	X	X	X
EVS/EHV	X	X	X	X	X
IAS	X	?	?	?	?
RPS	X	X	?	?	?
SPMU	X	X	X	?	?
SBGT	X	X	X	X	X
CI	X	X	X	X	X
H_2I	X	X	?	?	?
ADHRS	?	?	?	X	X
RWCU	?	X	X	X	X
SFPCCU	?	?	?	X	X
CCW	NA	?	?	?	?
TBCW	NA	?	?	?	?
PSW	NA	?	?	?	?
Circ W	NA	?	?	?	?

SUCCESS CRITERIA MODE 2

- "FULL POWER INITIATORS" WILL HAVE THE SAME SUCCESS CRITERIA
- LOCA INITIATORS ("MAINTENANCE INDUCED", "RECOVERABLE DIVERSION", AND "PIPING OR COMPONENT FAILURE IN OPERATING SYSTEM") WILL HAVE THE SAME SUCCESS CRITERIA AS THE APPROPRIATE SIZED "FULL POWER LOCA" EXCEPT THE (....) SYSTEMS MAY BE FAILED AND NO LONGER CONSIDERED IN THE SUCCESS CRITERIA LIST
- DECAY HEAT REMOVAL (DHR) INITIATORS WILL HAVE THE SAME TYPE OF SUCCESS CRITERIA AS "FULL POWER TRANSIENT INITIATORS" EXCEPT THE SYSTEM INVOLVED IN THE DHR INITIATOR MAY BE FAILED AND NO LONGER CONSIDERED IN THE SUCCESS CRITERIA LIST

INITIATING EVENT IDENTIFICATION AND FREQUENCIES

BWR LOW POWER AND SHUTDOWN ACCIDENT FREQUENCIES PROJECT

TANIA M. HAKE

SANDIA NATIONAL LABORATORIES

SENIOR CONSULTING GROUP MEETING

MARCH 14-15, 1990

FIVE GROUPS OF INITIATING EVENTS CONSIDERED:

1. TRANSIENT EVENTS

2. LOSS OF COOLANT ACCIDENT EVENTS

3. DECAY HEAT REMOVAL CHALLENGE INITIATORS

4. SPECIAL EVENTS

5. HAZARDS EVENTS

INITIATING EVENTS

2. LOCAs

	LOCA CATEGORY	MODE: LOW POWER	STARTUP	HOT SHUTDOWN	COLD SHUTDOWN	REFUELING
A	LARGE LOCA	X	X	X	X	X
S1	INTERMEDIATE LOCA	X	X	X	X	X
S2	SMALL LOCA	X	X	X	X	X
S3	SMALL-SMALL LOCA	X	X	X	X	X
V	INTERFACING SYSTEM LOCA (FAILURE OF HIGH-TO-LOW PRESSURE INTERFACE)	N/A	TBD	X*	X*	X*
R	VESSEL RUPTURE	N/A	N/A	N/A	N/A	N/A
H	DIVERSION OF VESSEL INVENTORY THROUGH CONNECTED SYSTEM:					
	RHRS		X	X	X	X
	HPCS		X	X	X	X
	LPCS		X	X	X	X
J	LOCA IN OPERATING CONNECTED SYSTEM:					
	RHRS		TBD	X	X	X
	RWCU		X	X	X	X
K	MAINTENANCE-INDUCED LOCA		X	X	X	X

X* = EVENTS CONSIDERED UNDER ANOTHER LOCA EVENT CATEGORY ("H", "J", OR "K")

POSSIBLE SCREENING CRITERIA FOR MAINTENANCE-INDUCED LOCAs:

- "DOUBLE FAILURE" REQUIRED FOR CONSIDERATION
- MUST REVIEW MAINTENANCE PROCEDURES IN DETAIL TO IDENTIFY PROBLEMS AND SCREEN SEGMENTS OF SYSTEMS
- SCREEN ON MULTIPLE DETECTION CAPABILITY
- SCREEN ON NUMBER OF POSSIBILITIES TO MITIGATE SEQUENCE

INITIATING EVENTS

4. SPECIAL EVENTS

	SPECIAL EVENTS	MODE LOW POWER	STARTUP	HOT SHUTDOWN	COLD SHUTDOWN	REFUELING
T4	CRITICALITY EVENTS:					
	ROD WITHDRAWAL ERROR	X	X	X	X	X
	REFUELING ACCIDENT					X
T5	LOSS OF ANY SERVICE WATER SYSTEM:					
	COMPONENT COOLING WATER		X	X	X	X
	STANDBY SERVICE WATER		X	X	X	X
	TBCW		X	X	X	X
	PLANT SERVICE WATER		X	X	X	X
	CIRCULATING WATER SYS		X	X	X	
TIAS	LOSS OF INSTRUMENT AIR SYSTEM	T2	X	X	X	X
TORV	INADVERTANT OPEN RELIEF VALVE (SHUTDOWN)	(T3C)	(T3C)	X		
TIOP	INADVERTANT OVERPRESSURIZATION EVENT		TBD	TBD	TBD	

AREAS REQUIRING FURTHER INVESTIGATION

- STEPS IN STARTUP OR SHUTDOWN PROCEDURES WHICH COULD SERVE AS INITIATING EVENTS
- WHETHER VESSEL RUPTURE FREQUENCY IS SIGNIFICANTLY INCREASED FOR THE STARTUP MODE
- COLD OVERPRESSURIZATION OF VESSEL IN SHUTDOWN
- EXTENT OF MAINTENANCE ACTIVITIES IN STARTUP
  - POSSIBILITY FOR MAINTENANCE-INDUCED TRANSIENTS
  - VARIOUS LOCAs INVOLVING INTERFACING SYSTEMS
- IS FRACTION OF TIME RHRS OPERATES IN STARTUP SIGNIFICANT?
  - LOCA IN OPERATING RHRS DURING STARTUP
  - LOSS OF RHR-SDC FOR DHR IN STARTUP
- HOW OFTEN RCIC IS USED FOR NORMAL SHUTDOWN EVOLUTION
  - LOSS OF RCIC AS DHR SYSTEM (STARTUP/HOT SHUTDOWN)
- WHETHER TRANSIENT EVENTS CAUSING SCRAM ON HIGH NEUTRON FLUX ARE APPLICABLE OR HAVE SAME EFFECT AT LOW POWER ("T3A")

(CONTINUED)

LOSP FREQUENCY CALCULATION

BACKGROUND

- THREE CAUSES FOR LOSP EVENTS:
  1. PLANT-CENTERED (SWITCHYARD)
  2. GRID-RELATED
  3. SEVERE WEATHER

- EVENT CATEGORIES:

  IA: NO OFF-SITE POWER AVAILABLE FOR LONGER THAN 30 MIN. (WITH UNIT TRIP)

  IB: NO OFF-SITE POWER AVAILABLE FOR LESS THAN 30 MIN. (WITH UNIT TRIP)

  II: LOSS OF BACKUP OFF-SITE POWER, BUT UNIT REMAINS CONNECTED TO NORMAL OFF-SITE POWER (NO UNIT TRIP)

  III: LOSS OF NORMAL OFF-SITE POWER, BUT BACKUP OFF-SITE POWER AVAILABLE THROUGH SWITCHING (NO UNIT TRIP)

  IV: LOSP DURING A COLD SHUTDOWN BECAUSE OF SPECIAL MAINTENANCE CONDITIONS UNIQUE TO SHUTDOWN

LOSP APPROACH FOR LOW POWER/SHUTDOWN

<u>LOW POWER/STARTUP</u>:

- USE ALL EVENTS INCLUDED FOR 1150

  (PLUS ADDITIONAL EVENTS OCCURRING IN LATTER HALF OF 1987)

- APPLY ALL EVENTS OVER PLANT CALENDAR YEARS

- USE SAME COMPUTER PROGRAM AS IN 1150 TO GENERATE LHS SAMPLES

- THE FREQUENCY FOR EACH MODE IS FOUND BY MULTIPLYING THE OVERALL FREQUENCY BY THE FRACTION OF TIME SPENT IN A PARTICULAR MODE.

LOSP FREQUENCY EQUATIONS, MODES 1 THROUGH 5:

$$LOSP_I = F_I * \{(PC + G\&W)/(SITE\ YRS)\} \qquad I = 1,2$$

$$LOSP_I = F_I * \{(PC + G\&W)/(SITE\ YRS) + (CAT\ IV)/(T_{SD})\} \qquad I = 3,4,5$$

WHERE:

$LOSP_I$	=	FREQUENCY OF LOSP DURING MODE I [/YR]
F_I	=	FRACTION OF TIME SPENT IN MODE I
PC	=	NUMBER OF PLANT-CENTERED EVENTS OCCURRING THRU 1987
G&W	=	NUMBER OF GRID & WEATHER EVENTS OCCURRING THRU 1987
SITE YRS	=	CUMULATIVE PLANT CALENDAR YEARS
CAT IV	=	NUMBER OF CATEGORY IV EVENTS OCCURRING THRU 1987
T_{SD}	=	PLANT SHUTDOWN YEARS CALCULATED BY CORRECTING THE SITE YEARS BY THE COMPLEMENT OF THE PLANT AVAILABILITY FACTOR

LOSP EVENTS VERSUS MODE (All Years Through 1987)

NOTE: 1st column contains single-unit site events; 2nd column contains multiple unit site events (counted as 0.5 in totals)

Format: plant;date (see attached plant abbreviations) e.g. Brunswick Unit 1 in 04/83 = BR1;0483

>15% POWER		<15% POWER		STARTU[illegible]BY		HOT SHUTDOWN		COLD SHUTDOWN		REFUELING		UNKNOWN MODE		# CATEGORY IV	
BV;0778	AN1;0978	FA;0977		DB;1177	AN2;0978	RB;0186	MI1;0985	FA;1083	BR2;0375	MO;0481	SO1;1180		PB1;0474	FT;0379	BR1;0483
DB;1079	CC1;0787			HN;0884	PV1;1085		MI2;0985	FV;0583	IP2;0777		TP3;0585		PB2;0474	FT;1078	IP3;1184
DR;1165	CC2;0787							GI;0371	OC1;0174					FC;0375	MC1;0987
FC;0276	DR1;0885							HN;C376	PB1;0271					HN;0884	MC2;0987
FC;0877	DR2;0885							MK;1173	PI1;0780					MO;0684	PB2;1084
GI;1073	IP2;0772							PA;0971	SL1;0578					OY;11*3	OC2;0585
HN;0468	IP2;0680							PL;1186	TP4;0577					PA;0184	SA1;0684
HN;0769	MC1;0884							PL;1187						SO;0673	TP3;0485
HN;0772	MI2;0776							YR;1165						SO-0480	
HN;0174	MI1;0876													YR;0584	
OY;0973	MI2;0876														
PA;0977	OC2;0174														
PA;0787	PV1;1085														
PL;0577	PI2;0780														
PL;0278	OC1;1177														
RO;0186	OC2;1177														
	OC2;0682														
	SO1;1185														
	SL1;0577														
	SN1;0784														
	SN2;0784														
	TP3;0473														
	TP3;0473														
	TP3;0474														
	TP4;0474														
	TP3;0674														
	TP4;0674														
	TP3;0577														
	TP3;0284														
	TP4;0284														
	TP3;0284														
	TP4;0284														
	TP4;0585														
********	********	********	********	********	********	********	********	********	********	********	********	********	********	****************	
TOTAL=16	TTL=16.5	TOTAL=1	TOTAL=0	TOTAL=2	TOTAL=1	TOTAL=1	TOTAL=1	TOTAL=9	TOTL=3.5	TOTAL=1	TOTAL=1	TOTAL=0	TOTAL=1	TOTAL = 17 CAT IV	
	(TOT/2)		(TOT/2)		(TOT/2)		(TOT/2)		(TOT/2)		(TOT/2)		(TOT/2)		

TOTAL NON-CAT.IV EVENTS CONSIDERED: 53 (+ 1 UNKNOWN MODE EVENT)

PERCENTAGE (>15% + <15% + STARTUP/HOT STBY) = 36.5/53 = 68.8%

PERCENTAGE (HOT S/D + COLD S/D + REFUEL) = 16.5/53 = 30.5%

FURTHER LOSP WORK REQUIRED:

- CALCULATION OF RECOVERY TIME--CATEGORY IV EVENTS
- ADD ONE CATEGORY IV EVENT MISSED (NOT INCLUDED IN NSAC-118)
- EXTEND SCOPE THROUGH 1988??

  (1987 USED IN ORDER TO BE COMPATIBLE WITH NUREG-1150 RESULTS)

TRANSIENT EVENTS

- BASED ON FREQUENCIES AND DISTRIBUTIONS USED FOR 1150.
  ALL FREQUENCIES ARE FOUND BY SUMMING THE FREQUENCIES FOR THE INDIVIDUAL EPRI EVENTS COMPRISING A TRANSIENT CATEGORY.
- ALL FREQUENCIES ARE TO BE CORRECTED BY THE FRACTION OF TIME SPENT IN A PARTICULAR MODE.

LOSS OF COOLANT ACCIDENTS

- MAINTENANCE-INDUCED LOCAs AND DIVERSION OF VESSEL INVENTORY THROUGH HPCS AND LPCS REQUIRE FURTHER INVESTIGATION BEFORE SCREENING FREQUENCIES CAN BE ASSIGNED.

- SEVERAL FREQUENCIES HAVE BEEN CALCULATED USING A BINOMIAL COMPUTER TO ESTIMATE THE MEDIAN AND ERROR FACTOR, GIVEN X NUMBER OF EVENTS OVER A CERTAIN AMOUNT OF TIME. A LOGNORMAL DISTRIBUTION HAS BEEN ASSIGNED FOR ALL EVENTS IN ORDER TO CALCULATE A MEAN.

DECAY HEAT REMOVAL CHALLENGE INITIATORS

- SOME OF THE DHR CHALLENGE EVENTS HAVE BORROWED EPRI TRANSIENT EVENTS FROM THE 1150 TRANSIENT CATEGORIES. IN THESE CASES, THE UPPER BOUND ESTIMATE REPORTED IN NUREG/CR-3862 FOR THE EPRI EVENT HAS BEEN USED, AND AN ERROR FACTOR OF FIVE ASSIGNED.

- FURTHER INFORMATION IS REQUIRED BEFORE SCREENING FREQUENCIES FOR EVENTS E1C, ISOLATION FROM OPERATING RWCU, E2C, LOSS OF OPERATING RWCU, AND E2E, LOSS OF OPERATING RCIC SYSTEM CAN BE ASSIGNED.

SPECIAL INITIATORS

- THE SCREENING FREQUENCY FOR EVENT T4β, REFUELING ACCIDENT, IS NOT MEANT TO BE CORRECTED BY ANY FACTORS, BECAUSE THIS EVENT HAS BEEN CALCULATED USING DATA UNIQUE TO REFUELING.

- SCREENING FREQUENCIES FOR ALL THE T5 EVENTS, LOSS OF ANY SERVICE WATER SYSTEM, ARE TO BE DETERMINED VIA FAULT TREES FOR THE VARIOUS SYSTEMS.

- THE INADVERTANT OVERPRESSURIZATION EVENT, TIOP, REQUIRES FURTHER INVESTIGATION BEFORE SCREENING FREQUENCIES CAN BE ASSIGNED TO THIS EVENT.

PRELIMINARY EVENT TREES

DONNIE W. WHITEHEAD
SANDIA NATIONAL LABORATORIES

LOW POWER AND SHUTDOWN ACCIDENT FREQUENCIES
PROGRAM SENIOR CONSULTING GROUP MEETING

MARCH 14 AND 15, 1990

ALBUQUERQUE, NEW MEXICO

EVENT TREES FOR MODE 2

EVENT TREE	DESCRIPTION
LOSP	SAME AS MODE 1
LOCAs	
PRIMARY SYSTEM	SAME AS MODE 1
MAINTENANCE INDUCED	DEPENDING ON BREAK SIZE, SIMILAR TO PRIMARY SYSTEM LOCAs "MAINTAINED" SYSTEM MAY BE FAILED
RECOVERABLE DIVERSION	DEPENDING ON BREAK SIZE, SIMILAR TO PRIMARY SYSTEM LOCAs "RECOVERABLE" SYSTEM MAY BE FAILED
PIPING OR COMPONENT FAILURE IN OPERATING CONNECTED SYSTEM	DEPENDING ON BREAK SIZE, SIMILAR TO PRIMARY SYSTEM LOCAs "FAILURE" SYSTEM MAY BE FAILED

EVENT TREES FOR MODE 2
(CONTINUED)

EVENT TREE	DESCRIPTION
FAILURE OF OPERATING RCIC	REACTOR SUBCRITICAL, RECOVER FAILURE, RCS OVERPRESSURE PROTECTION, CORE COOLING, CONTAINMENT OVERPRESSURE PROTECTION
RWCU	REACTOR SUBCRITICAL, RECOVER FAILURE, RCS OVERPRESSURE PROTECTION, CORE COOLING, CONTAINMENT OVERPRESSURE PROTECTION
INABILITY TO ESTABLISH SDC	REACTOR SUBCRITICAL, ESTABLISH SDC, RCS OVERPRESSURE PROTECTION, CORE COOLING, CONTAINMENT OVERPRESSURE PROTECTION

HUMAN RELIABILITY ANALYSIS METHODOLOGY FOR GRAND GULF SHUTDOWN STUDY

MARCH 15, 1990

TERESA T. SYPE

DIVISION 6412

PROCEDURES TO ENSURE EFFICIENT USE OF CHOSEN HRA METHODOLOGY

- TALENT
  - BEING NEGOTIATED WITH LIVERMORE AND NRC
- SHARP
- TEAM

NUCLARR DATA BASE

- INEL
- USEFUL TOOL FOR REFERENCE PURPOSES

HUMAN FACTORS

- SYSTEMS ANALYST IDENTIFIES HUMAN ACTION AND HRA BEGINS
- TWO CATEGORIES OF HUMAN ACTIONS

  1. PRE-ACCIDENT

     - RESTORATION OF COMPONENTS AFTER TEST OR MAINTENANCE ACTIVITIES

  2. POST-ACCIDENT

     - RESTORATION OF A FAILED FUNCTION

HRA METHODOLOGY
ASEP THERP

CHAPTER 7 OF NUREG/CR-4550, METHODOLOGY DOCUMENT

- ALLOWS DIRECT COMPARISON TO FULL-POWER STUDY
- ACCEPTABLE TO HRA COMMUNITY
- COST-EFFECTIVE
- EASY TO USE
- PRODUCE NEEDED/REQUIRED HERs
- DOCUMENTED/TRACEABLE

SIMULATOR

- UTILITY PARTICIPATION
- SUPPLY REACTOR OPERATORS
- USE OF SIMULATOR

PRIOR TO PLANT VISIT

1. PROCEDURES
   - ADMINISTRATIVE
   - SURVEILLANCE
   - CALIBRATION
   - TEST AND MAINTENANCE
   - EOPs

   SYSTEM DESCRIPTIONS

   PLANT LAYOUT DRAWINGS

   TECHNICAL SPECIFICATIONS

PRIOR TO PLANT VISIT (CONT.)

2. ARRANGE

   - ORIENTATION TOUR OF PLANT (CONTROL ROOM)
   - PHOTOGRAPHS
   - TALK-THROUGHS
   - WALK-THROUGHS
   - OPERATORS
   - SIMULATOR

DURING PLANT VISIT

- TASK ANALYSIS DATA SHEET DEVELOPED AS DETAILED RECORD KEEPING TOOL
- STUDY HOW TASK INTENDED TO BE CARRIED OUT AND ADMINISTRATIVE CONTROL EMPLOYED
- INTERVIEW AND OBSERVE SUBJECT MATTER EXPERTS
- SAMPLE OF REAL TASKS
    - TALK-THROUGHS
    - WALK-THROUGHS
    - TIMING-STOPWATCH
    - CHECKLISTS
    - PROCEDURES
- ADMINISTRATIVE CONTROL
    - HANDLING AND TYPE OF TAGS - RESTORATION
    - SIMULATOR
- CLOSE UP PHOTOGRAPHS TO READ DISPLAYS AND LABELS

SURRY SHUTDOWN STUDY APPROACH AND PROGRESS REPORT

PRESENTED BY: T.L. CHU

DEPARTMENT OF NUCLEAR ENERGY

BROOKHAVEN NATIONAL LABORATORY

PRESENTED TO: QA COMMITTEE

MARCH 14-15, 1990

A/13

A PWR AT SHUTDOWN

- 235 LOSS OF RHR EVENTS AT U.S. PWRs (Up to 1986)
- W SCENARIO MAY LEAD TO CORE DAMAGE IN 30 MINUTES
- FEW TECHNICAL SPECIFICATION REQUIREMENTS
  - SAFETY INJECTION SYSTEM, DIESEL GENERATORS, CONTAINMENT INTEGRITY
- HIGHER MAINTENANCE UNAVAILABILITY
  - 4kv ESSENTIAL BUS
- REACTOR COOLANT SYSTEM MAY BE PARTIALLY DRAINED
  - STEAM GENERATOR, TURBINE-DRIVEN AUXILIARY FEED PUMP

LOSS OF RESIDUAL HEAT REMOVAL SYSTEM AT DIABLO CANYON, UNIT 2

APRIL 10, 1987

- ONE WEEK AFTER SHUTDOWN
- MID-LOOP OPERATION
- EQUIPMENT HATCH OPEN
- INADVERTENT DRAINING OF RCS
- BOTH RHR PUMPS BECAME VAPOR BOUND
- RESTORATION OF VESSEL LEVEL WAS DELAYED
- RCS WAS BOILING
- RHR RESTORED IN 89 MINUTES

SUMMARY OF EXISTING PRA STUDIES

STUDY	SCOPE	CORE DAMAGE FREQUENCY
NSAC-84 (Zion)	LOSS OF RHR LOCA LOW TEMPERATURE OVERPRES- SURIZATION	$1.8x10^{-5}$
NUREG/CR-5015 (Generic/Zion)	LOSS OF RHR LOSS OF OFFSITE POWER LOCA	$5.22x10^{-5}$
SEABROOK	MODE 4, 5, 6 TRANSIENTS LOCAs EXTERNAL EVENTS	$4.5x10^{-5}$
W LARGE BREAK LOCA	LARGE BREAK LOCA IN MODES 3 AND 4	SMALLER THAN THAT FOR MODE 1
FRENCH TECH. SPEC. STUDY	LOSS OF RHR LOCA	$4x10^{-5}$
SURRY SHUTDOWN STUDY	NON-FULL POWER FIRE, FLOOD	

SCOPE OF SURRY SHUTDOWN STUDY

- LOW POWER AND SHUTDOWN
- FUEL IN THE CORE
- CHARACTERIZATION OF PLANT OPERATIONAL MODES
- LEVEL 1 INTERNAL EVENT PRA WITH FIRES AND FLOODS
- LEVELS 2 AND 3 WILL BE PERFORMED LATER

IDENTIFICATION AND CHARACTERIZATION OF PLANT OPERATIONAL MODES

- DEFINITION OF PLANT OPERATIONAL MODES (POM)
- DETERMINATION OF APPLICABLE INITIATING EVENTS (IEs)
- ESTABLISH APPLICABLE SYSTEMS AND SUCCESS CRITERIA FOR EACH POM AND IE
- DEVELOPMENT OF NON-FULL-POWER DATA BASE

DURATIONS AND CHARACTERIZATION OF PHASES
OF THREE TYPES OF OUTAGES
(Used in NUREG/CR-5015)

OUTAGES	PHASE	START* (HR)	END* (HR)	DURATION (HR)	PLANT CONDITIONS
REFUELING	1	54	167	113	RCS COOLING DOWN, RCS FILLED
REFUELING	2	167	587	420	RCS DRAINED, SG EDDY CURRENT TEST
REFUELING	3	587	1087	500	REFUELING CAVITY FILLED, FUEL SHUFFLING, VESSEL HEAD OFF
REFUELING	4	1087	1996	909	RCS FILLED, MAINTENANCE
DRAINED MAINTENANCE	1	21	83	62	RCS COOLING DOWN, RCS FILLED
DRAINED MAINTENANCE	2	83	179	96	RCS DRAINED, MAINTENANCE
DRAINED MAINTENANCE	3	179	982	803	RCS FILLED, MAINTENANCE
NONDRAINED MAINTENANCE	1	21	146	125	RCS FILLED, MAINTENANCE

*TIME AFTER SHUTDOWN

DEFINITION AND CHARACTERIZATION OF POMs

- POMs INCLUDE OUTAGE TYPES AND PHASES WITHIN EACH OUTAGE TYPE

  REFUELING, DRAINED MAINTENANCE, NON-DRAINED MAINTENANCE, MID-LOOP OPERATIONS, PLANT OPERATIONAL MODES AS DEFINED IN TECH. SPEC.

- PARAMETERS USED IN DEFINITION

  REACTIVITY, COOLANT TEMPERATURE, RCS PRESSURE, VESSEL LEVEL, TIME AFTER SHUTDOWN, DURATION

- CHARACTERIZATION OF POMs

  FREQUENCY, PLANT CONFIGURATION, SYSTEM AVAILABILITY, SHUTDOWN ACTIVITY, TIME TO CORE UNCOVERY, MAINTENANCE UNAVAILABILITY, RCS INTEGRITY, CONTAINMENT INTEGRITY

- APPROACH (INFORMATION NEEDED)

  PLANT VISIT, REVIEW OF EVOLUTION OF PLANT PAST OUTAGES (LOG BOOKS), REVIEW OF PROCEDURES, THERMAL HYDRAULIC ANALYSIS, DISCUSSION WITH PLANT PERSONNEL, USE OF EXPERIENCE FROM EXISTING SHUTDOWN STUDIES

Surry Plant Operational modes

		Time	Power	Vessel Level	K	T F	P psig	Surry T.S. Mode	Standard. T.S. Mode
	Low Power Operation		15%		1.0	543	2235	6,5	1,2
Reactor Trip		0							
	Cooldown w. S.G.		decay heat	full	<1.0			4,3	3
RHR Start		54				350	450		
	Cooldown w. RHR	(Zion)						3	4
		11.67/22.3				200			
	Cooldown w. RHR	(Surry)						2	5
		118				140			
	Drained down to midloop	(Zion)				(150	Surry)	2,1	5,6
		74.8/84.6		Midloop					
	Midloop Operation I							1,2	5,6
	Fill for Refueling							1	6
				Cavity filled					
	Refueling							-	6
	Drain after Refueling							1	6
				Midloop					
	Midloop Operation II							1,2	5,6
	Fill RCS							1,2	5,6
				RCS Filled					
	RCS Heatup							1,2	5,6
RHR Stopped						350	450		
	Startup w. S. G.				<1.0			3,4	3
		2860				543	2235		
	Low Power Operation				=1.0			6	1,2
	Full Power Operation								

SURRY TECHNICAL SPECIFICATIONS

- OPERATIONAL MODE DEFINITIONS DO NOT FOLLOW STANDARD TECH. SPECS.

- BOUNDARIES OF DIFFERENT REQUIREMENTS
  - CRITICALITY
  - Tavg = 350°F and P = 450 psig
  - Tavg = 200°F (Cold Shutdown)
  - REFUELING OPERATIONS

- REQUIREMENT ON OTHER UNIT
  - AUXILIARY FEEDWATER
  - CHARGING PUMP

W STANDARD TECH. SPEC. REVISION 3

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350 F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350 F
3. HOT STANDBY	< 0.99	0	$\geq$ 359 F
4. HOT SHUTDOWN	< 0.99	0	350 F > T_{AVG} > 200 F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200 F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140 F

*Excluding decay heat.
**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

SURRY TECH. SPEC.

1. REFUELING SHUTDOWN CONDITION (6)

WHEN THE REACTOR IS SUBCRITICAL BY AT LEAST 5% $\Delta k/k$ AND T_{AVG} IS $\leq 140°F$ AND FUEL IS SCHEDULED TO BE MOVED TO OR FROM THE REACTOR CORE.

2. COLD SHUTDOWN CONDITION (5)

WHEN THE REACTOR IS SUBCRITICAL BY AT LEAST 1% $\Delta k/k$ AND T_{AVG} IS $\leq 200°F$.

3. INTERMEDIATE SHUTDOWN CONDITION

WHEN THE REACTOR IS SUBCRITICAL BY AN AMOUNT GREATER THAN OR EQUAL TO 1.77% $\Delta k/k$ AND $200°F < T_{AVG} < 547°F$.

4. HOT SHUTDOWN CONDITION (3)

WHEN THE REACTOR IS SUBCRITICAL BY AN AMOUNT GREATER THAN OR EQUAL TO 1.77% $\Delta k/k$ AND T_{AVG} IS $\geq 547°F$.

5. REACTOR CRITICAL

WHEN THE NEUTRON CHAIN REACTION IS SELF-SUSTAINING AND $k_{eff} = 1.0$.

SURRY TECH. SPEC. (Continued)

6. POWER OPERATION (1)

WHEN THE REACTOR IS CRITICAL AND THE NEUTRON FLUX POWER RANGE INSTRUMENTATION INDICATES GREATER THAN 2% OF RATED POWER.

7. REFUELING OPERATION

ANY OPERATION INVOLVING MOVEMENT OF CORE COMPONENTS WHEN THE VESSEL HEAD IS UNBOLTED OR REMOVED.

SURRY TECH. SPEC. FOR Tavg > 350°F

- AT LEAST TWO REACTOR COOLANT LOOPS SHALL BE OPERABLE WITH AT LEAST ONE IN OPERATION
- AT LEAST ONE OF THE TWO SGs IN NON-ISOLATED LOOP SHALL BE OPERABLE
- CONTAINMENT INTEGRITY (Required Unless Tavg < 200°F)
- TWO DIESEL GENERATORS (Seven Days AOT Before Cold Shutdown)
- TWO MOTOR-DRIVEN AUXILIARY FEEDWATER PUMPS (Three Day AOT Before Hot Shutdown)
- TWO CCW PUMPS (One Unit Operation)
  THREE CCW PUMPS (Two Unit Operation)
- ONE CHARGING PUMP COOLING SYSTEM SHALL BE OPERATING. THE SPARE SUBSYSTEM SHALL BE OPERABLE WITH 24 HOUR AOT.
- 96,000 GALLON IN CONDENSATE STORAGE TANK
- TWO DIESEL GENERATORS, TWO EMERGENCY BUSES (4160 AND 480)

SURRY TECH. SPEC. FOR Tavg < 350°F

- A MINIMUM OF TWO NON-ISOLATED LOOPS, CONSISTING OF ANY COMBINATION OF REACTOR COOLANT LOOPS OR RHR LOOPS SHALL BE OPERABLE, EXCEPT IN MODE 7, ONE LOOP OF RHR MUST BE OPERABLE AND CAN BE REMOVED ONE HOUR PER EIGHT HOUR PERIOD.
- ONE PRESSURIZER SAFETY VALVE SHALL BE OPERABLE WHEN THE HEAD IS ON THE VESSEL.
- TWO PORVs WITH SETTING OF < = 435 psig OR A BUBBLE IN THE PRESSURIZER FOR LESS THAN 72 HOURS OR THE RCS VENTED WITH ONE OPEN PORV.
- A MAXIMUM OF ONE CHARGING PUMP SHALL BE OPERABLE.
- ONE FLOW PATH FOR BORIC ACID INJECTION.
- O ACCUMULATOR
- BORON CONCENTRATION IN ISOLATED LOOPS.
- TWO TRAINS OF THE MAIN CONTRO' ROOM AND EMERGENCY SWITCH ROOM VENTILATION AND AIR CONDITIONING SYSTEM (REQUIRED IF ABOVE COLD SHUTDOWN).
- CONTAINMENT INTEGRITY (Tavg > 200°F)

SURRY TECH. SPEC. FOR REFUELING OPERATION

- CONTAINMENT INTEGRITY

- ONE RHR PUMP AND HEAT EXCHANGER SHALL BE OPERABLE. IT CAN BE REMOVED ONE HOUR PER EIGHT HOUR PERIOD.

- NO MOVEMENT OF IRRADIATED FUEL IN THE REACTOR CORE BEFORE 100 HOURS AFTER REACTOR TRIP

INITIATING EVENT ANALYSIS

IDENTIFICATION OF INITIATING EVENTS

SEARCH OF COMPUTERIZED DATA BASES - SCSS, NPE, NPRDS

REVIEW OF SHUTDOWN DATA REPORTS - NSAC-52, AEOD/C503

REVIEW OF EXISTING SHUTDOWN STUDY - NSAC-84, SEABROOK, FRENCH STUDY (NEED HELP TO GET REPORT), NUREG/CR-5015

REVIEW OF PROCEDURES USED AT SHUTDOWN - OPERATING, TEST, MAINTENANCE, EMERGENCY, OFF-NORMAL, ABNORMAL

REVIEW OF IEs FOR POWER OPERATIONS FOR APPLICABILITY

REVIEW OF GENERIC LETTERS, INFORMATION NOTICES, BULLETINS (TO ADDRESS NRC CONCERNS)

REVIEW OTHER STUDIES THAT IDENTIFIED SCENARIOS AT SHUTDOWN:

- NUREG/CR-4999 LTOP
- NUREG/CR-4982 SPENT FUEL POOL ACCIDENT, FUEL TRANSFER CASK DROP
- NUREG/CR-5368 REACTIVITY ACCIDENTS
- WESTINGHOUSE THERMAL HYDRAULIC ANALYSIS ON MID-LOOP OPERATIONS
- WESTINGHOUSE PRA OF LOCAs IN MODES 3 AND 4 (NEED REPORT)

IDENTIFIED SCENARIOS AT SHUTDOWN

SCENARIO	STUDY	CORE DAMAGE FREQUENCY
LTOP	NUREG/CR-4999 NUREG/CR-4550, VOL.3 NSAC-84	$2.6x10^{6}$ (INDIAN POINT) $<10^{8}$ (SURRY) NOT ASSESSED (VERY LOW)
REFUELING CAVITY SEAL FAILURE	NUREG/CR-4982	10^{6} (GINNA)
FUEL TRANSFER CASK DROP	NUREG/CR-4982	$3.1x10^{5}$
BORON DILUTION	CE SYSTEM 80 PRA	$<10^{6}$
OTHER REACTIVITY ACCIDENTS	NUREG/CR-5368	$<10^{7}$

REFUELING CAVITY SEAL FAILURE

- MAY LEAD TO DRAINING OF THE SPENT FUEL POOL
- MAY DRAIN THE VESSEL LEVEL TO VESSEL FLANGE
- SURRY HAS REDUNDANT SEALS AND A BARRIER IN THE SPENT FUEL POOL THAT PRELUDE EXPOSING SPENT FUEL DUE TO SEAL FAILURE
- SURRY HAD A SEAL FAILURE WITH LOSS OF THOUSANDS OF GALLONS OF WATER IN THE CAVITY
- CORRECTIVE ACTIONS WERE TAKEN

FRONTLINE SYSTEMS

- ACCUMULATORS
- CHARGING AND HIGH PRESSURE INJECTION/ RECIRCULATION
- LOW PRESSURE INJECTION/RECIRCULATION
- INSIDE SPRAY RECIRCULATION
- OUTSIDE SPRAY RECIRCULATION
- POWER CONVERSION SYSTEM
- AUXILIARY FEEDWATER SYSTEM
- PRIMARY PRESSURE RELIEF SYSTEM
- REACTOR PROTECTION SYSTEM
- RESIDUAL HEAT REMOVAL SYSTEM
- BORIC ACID TRANSFER PUMPS
- REFUELING WATER STORAGE TANK
- REFUELING CAVITY SEAL

SURRY DESIGN FEATURES

- RHR SYSTEM INSIDE THE CONTAINMENT
- NO AUTO-CLOSURE INTERLOCK ON RHR SUCTION VALVES
- NO RHR SUCTION RELIEVE VALVE
- RHR PUMPS RECEIVE POWER FROM STUB BUSES THAT WILL BE SHREDDED UPON LOSS OF OFFSITE POWER
- LOOP ISOLATION VALVES

COMPONENT FAILURE DATA

- HARDWARE FAILURE DATA-USE THAT OF NUREG/CR-4550, VOLUME 3, REVISION 1

- COLLECTION OF PLANT-SPECIFIC DATA FOR MAINTENANCE

  UNAVAILABILITY

  NSAC-84 DATA ONLY APPLIES TO ZION. THERE IS A PROBLEM INTERPRETING IT.

  APPROACH 1: REVIEW OF MAINTENANCE LOG BOOKS
  TIME CONSUMING
  MORE RIGOROUS
  USE OF NPRDS MAY HELP
  USE OF SURRY'S COMPUTERIZED
  MAINTENANCE DATA BASES

  APPROACH 2: DISCUSSION WITH PLANT PERSONNEL AND
  USE EXPERT'S JUDGMENT

NPRDS DATA BASE - MAINTENANCE UNAVAILABILITY

- DATE AND TIME FAILURE OCCURRED
- DATE AND TIME FAILURE ENDED
- EFFECT ON SYSTEM - LOSS OF REDUNDANCY
- EFFECT ON PLANT - LOSS OF REDUNDANCY
- DESCRIPTION OF PLANT CONDITION - SHUTDOWN ON 85% POWER
- DATA FROM 1/1/84 TO PRESENT
- PROPRIETARY - PLANT NAME CAN NOT BE USED
- ACCURACY IS QUESTIONABLE
- CAN BE USED TO IDENTIFY FAILURE EVENTS
- NEED PLANT DATA TO VERIFY DOWN TIME

LOSS OF RHR DATA BASE (PWRs)

A TOTAL OF 235 EVENTS (UP TO DECEMBER 1986)

<u>SOURCES</u>:

NSAC-52: RESIDUAL HEAT REMOVAL EXPERIENCE REVIEW AND SAFETY ANALYSIS, PRESSURIZED WATER REACTORS, JANUARY 1983.

AEOD-C503: DECAY HEAT REMOVAL PROBLEMS AT U.S. PRESSURIZED WATER REACTORS, DECEMBER 1985.

NUREG/CR-5015: IMPROVED RELIABILITY OF RESIDUAL HEAT REMOVAL IN PWRs AS RELATED TO RESOLUTION OF GENERIC ISSUE 99.

SEABROOK STATION PROBABILISTIC SAFETY STUDY, SHUTDOWN (MODES 4, 5 AND 6), MAY 1988.

LOSS OF RESIDUAL HEAT REMOVAL WHILE AT SHUTDOWN FOR PWRS

	PLANT NAME	DOCKET NO.	EVENT DATE	LER	INITIAL PLANT CONDITION	RECOVERY TIME	NSAC52 CAT.	GI 99 CAT.	SEABROOK CAT.	AEOD	CR5015 CAT.
1	Salem 1	5000272	09/02/76	76-004	Mode 6	19 min	A.1	A1			
2	Salem 1	5000272	09/20/76	76-005	Mode 5	30 min	A.1	A2			
3	Indian Point 3	5000286	09/30/76	76-3-36(A)	Mode 5	8 min	A.1	A3			
4	Davis Besse	5000346	05/14/77	77-006						AEOD-B	
5	Davis Besse	5000346	05/19/77	77-007						AEOD-C	
6	Trojan	5000344	05/21/77	77-016	Mode 5	55 min	A.2	C1			
7	Davis Besse	5000346	05/27/77	77-002						AEOD-C	
8	Davis Besse	5000346	05/28/77	77-003						AEOD-C	
9	Davis Besse	5000346	06/12/77	77-005						AEOD-C	
10	Davis Besse	5000346	07/22/77	77-009 (two events)						AEOD-C	
11	Crystal River 3	5000302	08/15/77	77-101	Mode 4		A.8	E34			
12	Three Mile Island 2	5000320	09/08/77		Mode 4 and 5		A.9	E40			
13	Ft. Calhoun	5000285	10/19/77	77-023	Mode 6		A.7	E26			
14	Palisades	5000255	01/08/78	78-003	Mode 5	45 min	A.5	E8			
15	Calvert Cliffs 1	5000317	01/24/78	78-004, 78-011	Mode 4 and 5		A.10	E44			
16	Trojan	5000344	03/20/78	78-010	Mode 5		A.1	A4			
17	Trojan	5000344	03/25/78	78-011	Mode 5	10 min	A.2	B1			
18	Trojan	5000344	03/25/78	78-011	Mode 5	10 min	A.2	B2			
19	Trojan	5000344	04/17/78	78-011	Mode 6		A.2	C2			
20	Crystal River 3	5000302	04/25/78	78-020	Cold shut.		A.6	E14			
21	Trojan	5000344	04/25/78	78-017	Mode 5	1 [illegible]	A.3	Da1			
22	Davis Besse 1	5000346	05/28/78	78-060	Mode 6	1.5 min	A.6	E15		AEOD-C	
23	Davis Besse 1	5000[illegible]	06/05/78	78-063	Mode 5		A.7	E27			
24	Davis Besse 1	5000346	06/15/78	78-067	Cold shut.	2 min (total of 3 events)	A.6	E16		AEOD-C	
25	Arkansas 2	5000368	08/16/78	78-001	Mode 5	2.8 hr	A.6	E17			
26	Crystal River 3	5000302	08/19/78	78-042	Mode 6	15 min	A.1	A5			
27	Beaver Valley 1	5000334	09/04/78	78-049		1 hr	A.2	C3			
28	Farley 1	5000348	09/18/78	78-069	Mode 5	7 min	A.1	A6			
29	Farley 1	5000348	09/18/78	78-069	Mode 5	3 min	A.1	A6			
30	Calvert Cliffs 2	5000318	09/24/78	78-033	Mode 5 or 6	23 min	A.4	E4			

LOSS OF OFFSITE POWER DATA BASE

A TOTAL OF 161 EVENTS

SOURCES:

NSAC-144: LOSS OF OFFSITE POWER AT U.S. NUCLEAR POWER PLANTS, ALL YEARS THROUGH 1988.

NUREG/CR-3992: COLLECTION OF COMPLETE AND PARTIAL LOSSES OF OFFSITE POWER AT NUCLEAR POWER PLANTS.

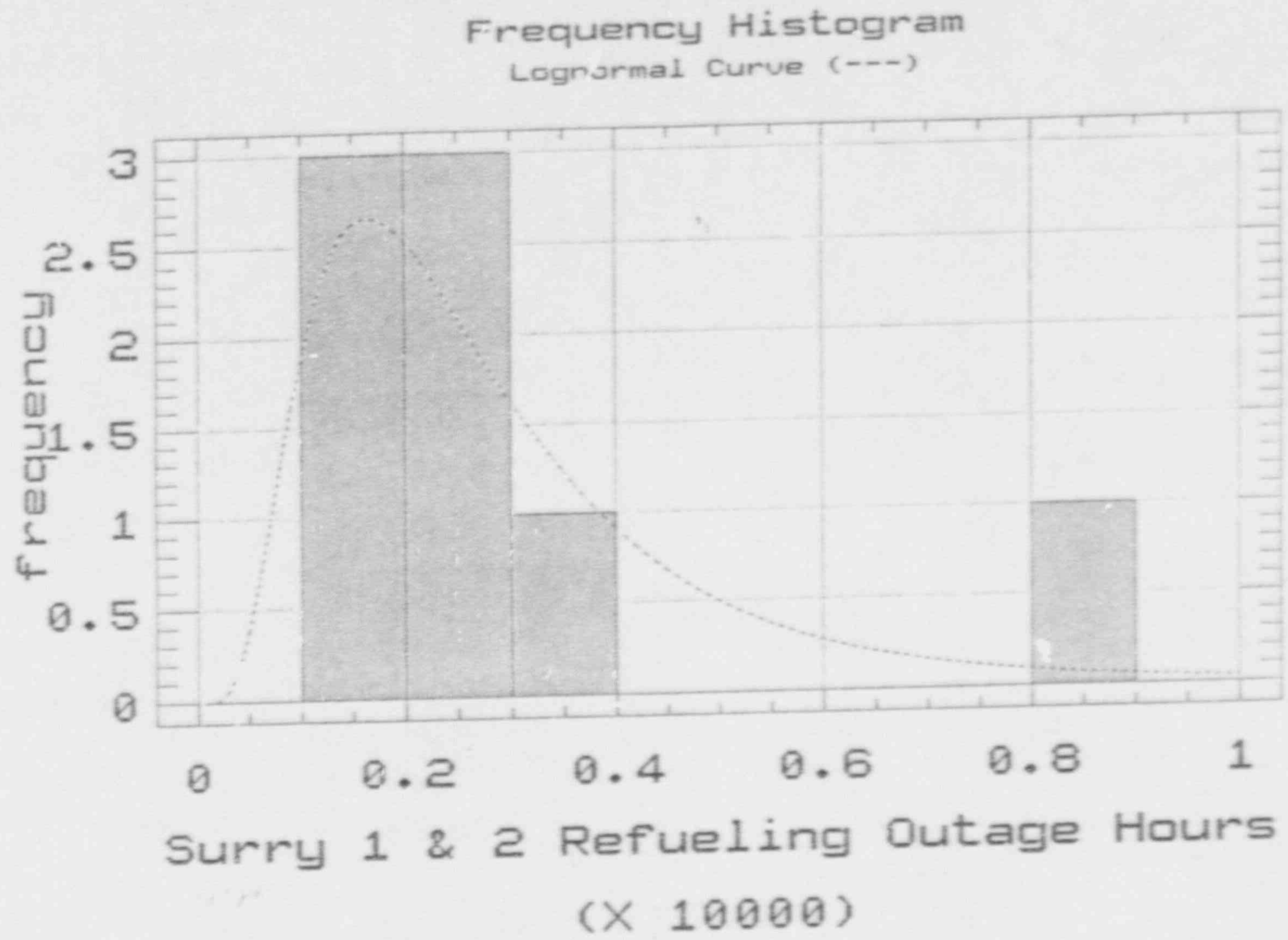
CATEGORY	NUMBER OF EVENTS
Ia	34
Ib	33
II	7
III	40
IV	18
V	3
OTHER	30

LOSS OF OFFSITE POWER CATEGORIES
(NUREG/CR-3992)

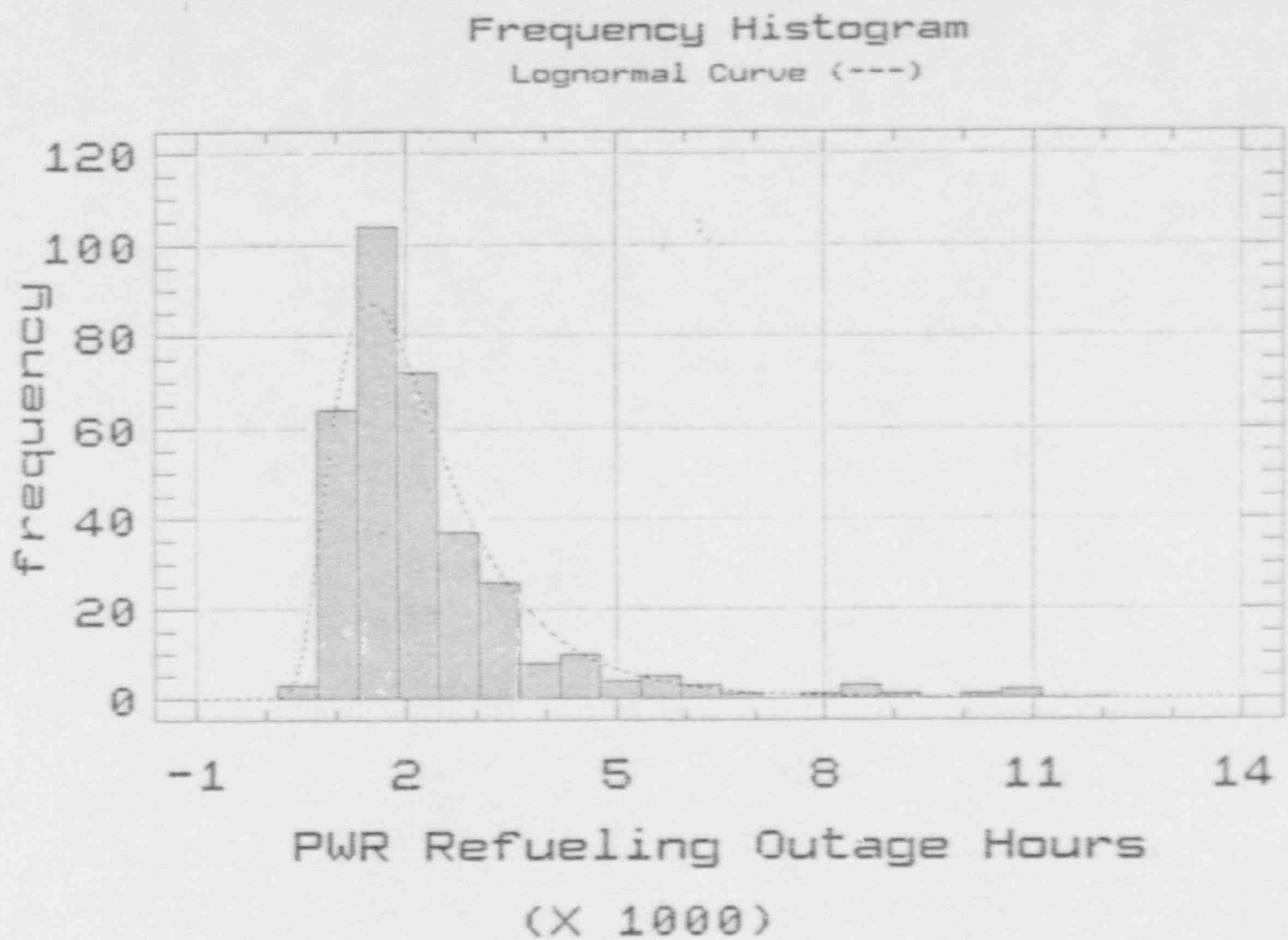
- CATEGORY I: FAILURE OF THE PREFERRED, ALTERNATE AND UNIT POWER SOURCES EITHER WHEN THE REACTOR WAS AT POWER OR WHEN SHUT DOWN, BUT IF THE PLANT WAS SHUT DOWN THE LOSS WOULD HAVE OCCURRED HAD THE PLANT BEEN OPERATING.
  - POWER WAS NOT RESTORED WITHIN 30 MINUTES.
  - POWER WAS RESTORED WITHIN 30 MINUTES.
- CATEGORY II: FAILURE OF THE PREFERRED AND ALTERNATE POWER SOURCES. THE UNIT DID NOT TRIP OR, IF NOT OPERATING, WOULD NOT HAVE TRIPPED HAD IT BEEN OPERATING.
- CATEGORY III: FAILURE OF THE PREFERRED AND UNIT OFFSITE POWER SOURCES. AN ALTERNATE OFFSITE POWER SOURCE WAS AVAILABLE BY MANUAL SWITCHING.
- CATEGORY IV: FAILURE OF ALL OFFSITE POWER DURING A COLD SHUTDOWN. THESE SPECIAL MAINTENANCE CONDITIONS DO NOT OCCUR DURING OR IMMEDIATELY FOLLOWING OPERATION.
- CATEGORY V: LOW VOLTAGE CONDITIONS ON THE TRANSMISSION SYSTEM. VOLTAGE WAS NEAR OR BELOW 90% OF NORMAL.

LOSSES OF OFF-SITE POWER AT U.S. NUCLEAR POWER PLANTS

	PLANT	DATE	RESTORATION TIME	SY 3992	SY 1150	CAT. 144	CAT. 3992	CAT. 1150	CAUSES OF FAILURE (FROM 1150)
1	Arkansas	02/22/75	Not reported	13	12	--	III	--	--
2	Arkansas	09/16/78	1:29	13	12	Ia	Ia	R, F	PC
3	Arkansas	04/07/80	#1-0:22; #2-0:52	13	12	III	III	--	--
4	Arkansas	06/24/80	--	13	12	III	III	--	--
5	Beaver Valley	07/28/78	0:17	13	12	Ib	Ib	R, F	PC
6	Big Rock Point	01/25/72	0:20		--	Ib	Ib	--	--
7	Braidwood	10/16/88	--			None			--
8	Browns Ferry	04/03/74	120 h for 161-kV line	11	--	III	III	--	--
9	Browns Ferry	03/01/80	0:06	11	--	II	II	--	--
10	Brunswick	04/26/83	0:17	13	12	IV	IV	--	--
11	Brunswick 2	03/26/75	0:04	13	12	Ib	Ib	R, F	PC
12	Byron	10/02/87	--			None			--
13	Calvert Cliffs	04/11/78	Not reporte	14	--	--	III	--	--
14	Calvert Cliffs	04/13/78	5:50	14	--	None	Ia	--	--
15	Calvert Cliffs	07/23/87	1:58	14		Ia			--
16	Connecticut Yankee	04/27/68	0:29	7	11	Ib	Ib	R, F	PC
17	Connecticut Yankee	07/15/69	0:09	7	11	Ib	Ib	R, F	PC
18	Connecticut Yankee	07/19/72	0:01	7	11	Ib	Ib	R, F	PC
19	Connecticut Yankee	01/19/74	0:20	7	11	Ib	Ib	R, F	PC
20	Connecticut Yankee	06/26/76	0:16	7	11	Ib	Ib	R, F	PC
21	Connecticut Yankee	08/01/84	0:10	7	11	Ib		R, F	PC
22	Connecticut Yankee	08/24/84	--	7	11	IV			--
23	Cook	02/17/75	Unit-3:58; preferred-73:06	11	--	III	III	--	--
24	Cook	09/01/77	Preferred-2:11	11	--	III	III	--	--
25	Cook	02/01/86	--	11		III			--
26	Cooper	02/21/76	0:29	11	--	III	III	--	--
27	Crystal River 3	06/16/81	Not reported	12	--	III	III	--	--
28	Crystal River 3	02/22/84	--	12		III			--
29	Crystal River 3	10/16/87	--	12		None			--
30	Davis Besse 1	11/29/77	11 s	13	11	III	III	R, F	PC



Frequency Histogram
Lognormal Curve (---)
frequency
0
0.5
1
1.5
2
2.5
3
0
0.2
0.4
0.6
0.8
1
Surry 1 & 2 Refueling Outage Hours
(X 10000)


Frequency Histogram
Lognormal Curve (---)
frequency
120
100
80
60
40
20
0
-1
2
5
8
11
14
PWR Refueling Outage Hours
(X 1000)

Enclosure

CLOSEOUT SUMMARY

TASK 1.4C

ANALYSIS OF RISK AT LOW POWER AND SHUTDOWN CONDITIONS

1. Task 1.4C: Analysis of Risk at Low Power and Shutdown Conditions.

2. Task Leader: Richard Robinson, Probabilistic Risk Analysis Branch, Division of Systems Research, Office of Nuclear Regulatory Research.

3. Issue:

Traditionally, probabilistic risk analyses of severe accidents in nuclear power plants (including those of the NRC staff's recent NUREG-1150 analysis) have considered the set of initiating events potentially occurring during full power operation. Some screening analyses of accident initiators during low power, shutdown, and other modes of plant operation other than full power have been performed. These suggested that risks during these modes were small relative to those occurring during full power operation. However, other studies (discussed later) and the Chernobyl accident, which occurred during low power testing exercises, suggested that low power and shutdown accident risks could be significant. The recent event at Vogtle Unit 1, while at cold shutdown, further emphasizes the need to systematically and comprehensively evaluate plant safety when operating at other than full power. As such, the analysis of the frequencies, consequences, and risks of these accidents was identified as one task (Task 1.4C) in the NRC staff s study of the implications of this accident to U.

A/14

S. commercial nuclear power plants.

4. Task Purpose:

The objectives of the task are:

- To assess the frequencies of severe accidents initiated during plant operational modes, other than full power operation, for a commercial pressurized water reactor (PWR) and a boiling water reactor (BWR);

- To combine accident frequencies with accident progression, source term, and offsite consequence analyses to yield estimates of severe accident risks from these plant operation modes in the studied PWR and BWR; and

- To compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results of this study with those of accidents initiated during full power operation (as assessed in NUREG-1150).

5. Scope:

As discussed above, the work performed under this task involves investigation of two operating commercial reactors, a PWR and a BWR, at plant operational modes (POM) other than full power operation. The current plan consists of a two-phased approach in order to provide an early analysis overview and to highlight any potential problem areas. Phase 1 is dedicated to producing preliminary PRA results, including internal fire and flooding analyses, for other related studies underway in the NRC. Phase 2 is to produce a final PRA analysis, guided by the Phase 1 results to proportionately concentrate the effort among the various operating modes, the dominant sequences, and pertinent data items according to their importance to core damage frequency and risk. The scope of this task does not include any seismic analysis, but does include the following (for both phases):

(i) Identification of Plant Operational Modes (POMs) and Parameters:

There are several plant operational modes besides full power operation characterized by parameters such as reactor criticality, reactor coolant system (RCS) pressure, RCS temperature, and percent thermal power. Such modes of operation are: low power, startup, hot standby, hot shutdown, cold shutdown, and refueling. Thus, this task is to define the plant modes of operation of interest as a foundation for performing risk analysis.

(ii) Determine the Applicable Initiating Events for Each POM:

The scope of this task is to determine a set of initiating events for each POM that potentially result in core damage, including those initiating events associated with and resulting from maintenance activities and plant modifications, as well as those associated with fires and floods internal to the plant.

(iii) Establish the Applicable Systems and Success Criteria for Each POM and Initiating Event:

The initial conditions of the plant, especially RCS pressure and temperature, and the availability of steam will affect the operability of engineered safety systems (and some systems not defined as "safety" but for which safety credit may be given). This task is to identify the applicable systems for each POM and initiating event with the corresponding success criteria so that system models can be constructed.

(iv) Develop the Non-Full-Power Data Base:

In order to develop a data base for non-power operational modes, plant testing and maintenance practices, procedures, and logs will be examined under this task to characterize equipment and systems unavailabilities for the various POMs. Mean time duration (per year) will also be established for each of the POMs. Operating procedures will be reviewed to determine if and what systems may be bypassed during a given POM. Technical specifications will be reviewed to determine what relaxations will be in effect during the given POM.

(v) Analysis of Accident Frequencies:

Based on identifi d initiating events for each POM, accident frequency analysis will be carried out, encompassing (1) data analysis, (2) system analysis, (3) event tree analysis, (4) internal fire and flooding analysis, (5) dependent failure analysis, (6) human reliability analysis, (7) accident sequence quantification, (8) plant damage state analysis, and (9) uncertainty analysis.

(vi) Accident Progression and Containment Analysis:

The scope of this task includes examination of applicable technical specifications for each POM to identify containment status and systems availability and develop accident progression and containment event trees, and to carry out quantification and develop accident progression bins fur each plant damage state.

(vii) Source Term Analysis:

The next step in the risk calculation is the source term analysis. The results of the source term analysis are release fractions for groups of chemically similar radionuclides with associated energy content, time, and duration of release for each accident progression bin. Source terms for shutdown and lower power events will be corrected for reduced fission product and decay heat levels from the full power source terms.

(viii) Consequence Analysis:

The final step in the risk quantification will be the offsite consequence analysis for source terms defined in the previous step. The specific consequence measures may include early fatalities, latent cancer fatalities, population dose, etc.

After the performance of risk quantification for two plants, generic insights and specific recommendations, if necessary, will be developed to reduce estimates of frequencies and consequences for accidents which may occur during the low power or shutdown modes of operation.

6. Work Descriptions:

In this section, in addition to discussing some of the work performed directly for the task described in Section 5, some related studies performed by others are also identified. Findings from these studies are described in the next section.

Under this task, a study of a BWR plant is in progress at the Sandia National Laboratory (SNL) and a study of a PWR plant is in progress at the Brookhaven National Laboratory (BNL). For the selected BWR plant, some of the ongoing work includes review of the Final Safety Analysis Report (FSAR) and technical specifications to develop a matrix which describes each operating mode in terms of the selected parameters such as temperature, pressure, etc. Also, tabulated information on technical specification requirements versus operating modes is also being developed. Various documents, including safety analyses contained in the plant FSAR and licensee event reports, have been reviewed to define initiating events relevant to low power/shutdown modes. It is anticipated that the preliminary (Phase 1) accident frequency quantification for both plants will be completed by the middle of 1991 and a more comprehensive analysis (Phase 2) by the middle of 1992, to be followed by the final risk computations approximately six months later. Some of the other completed studies relevant to this task include the following:

(1) Seabrook Probabilistic Safety Study - Modes 4, 5, and 6 (Ref. 1);

(2) Brunswick Decay Heat Removal Probabilistic Safety Study (Ref. 2);

(3) Zion Nuclear Plant Residual Heat Removal PRA (Ref. 3);

(4) Improved Reliability of Residual Heat Removal Capability in PWRs as Related to Resolution of Generic Issue 99 (Refs. 4 and 5);

(5) Residual Heat Removal Experience Review and Safety Analysis - Pressurized Water Reactors (Ref. 6);

(6) Residual Heat Removal Experience Review and Safety Analysis - Boiling Water Reactors (Ref. 7);

(7) Reactivity Accidents - Reassessment of the Design-Basis Events (Ref. 8);

(8) Probabilistic Analysis of 900 MWe French PWR Shutdown Technical Specifications (Ref. 15); and

(9) PRAs of the French 900 MWe and 1300 MWe PWRs (Ref. 16).

7. <u>Findings:</u>

Before describing findings from studies listed in the previous section, some general discussion of the issues is useful. In most of the PRAs, it has been assumed that the level of risk associated with accidents initiated during full power operation, while small, is substantially greater than that associated with accidents during low power or shutdown. This assumption is supported by the fact that because of the lower decay heat levels and smaller radionuclide inventory during low power/shutdown modes there is generally more time available to recover from adverse situations during these modes of operation. However, there are other factors which might exacerbate the situation during accidents at low power/shutdown. Some of these factors are: (1) the fact that many of the automatic safety systems may have been disabled during these modes requiring greater operator intervention; (2) high equipment unavailability due to planned maintenance; (3) potential maintenance configurations requiring minimum RCS coolant inventory; (4) open containment penetrations and hatches; and (5) inadequacy of full power emergency procedures to address emergencies at low power/shutdown modes.

In addition to the above factors, certain experiences and events at operating reactors provide further impetus to study risk during low power/shutdown modes of operation (e.g., Refs. 4, 5, 6, 9, 10). One type of event is the Chernobyl type of event, that is, rapid insertion of reactivity causing accidents. Other types of events represent loss of decay heat removal functions, loss of coolant inventory, and inadvertent pressurization. To systematically examine these concerns, two NUREG-1150 (Ref. 11) plants are chosen under this task.

In Ref. 8, reporting on work performed in support of Task 2.1A, a study of accidents which result from large reactivity insertions was carried out for a PWR plant and a BWR plant. The potential reactivity accidents have been categorized in that study as follows:

PWR Events

1. Adding diluted accumulator water during refueling
2. Adding diluted RWST water during shutdown
3. LOCA with diluted ECCS water
4. LOCA with sump water diluted
5. LOCA/SGTR with secondary diluting primary
6. Inadvertent boron dilution at shutdown
7. Startup of reactor coolant pump after improper dilution
8. Beyond-design-basis rod ejection accidents
9. Thermal-hydraulic transients with positive MTC
10. Other beyond-design-basis events

BWR Events

1. Beyond-design-basis rod drop accident
2. Rod ejection accident
3. Beyond-design-basis overpressurization events
4. Flushing of boron during an ATWS
5. Operation in region of instability
6. Refueling accidents
7. Other beyond-design-basis events

Few of the above listed were identified requiring further analysis based on the estimated frequency of worst accident sequences. During the current task, these events will be examined for their applicability to operational modes of interest and further analysis.

Recently, a Level 3 PRA for the Seabrook station has been completed to evaluate the likelihood of severe core damage with various paths for offsite release for the plant in Mode 4 (hot shutdown), Mode 5 (cold shutdown), or Mode 6 (refueling) (Ref. 1). Radiological source terms and resultant public health consequences were also evaluated. Findings and conclusions from this study are reported as follows (Ref.1):

(i) With the benefit of relatively low cost modifications and administrative controls, the frequency of core damage during shutdown is small, but non-negligible, in comparison to power operation.

The improvements include:

Instrumentation and alarms to improve operator action and to foretell incipient loss of (RHR) system during the time when the RCS is drained to the hot leg midplane;

Procedures and training to cover the possible abnormal plant conditions and alternative cooling schemes; and

Administrative controls to minimize the time in the drained-down configuration, to assure that alternative cooling methods are available, and to assure control of containment integrity.

(ii) The following summarizes some quantitative conclusions of the study:

The mean core damage frequency when shutdown is less than during full power operation by about a factor of 6; and

The early fatality risk from shutdown is about an order of magnitude less than full power operation.

Results of two EPRI-sponsored studies to evaluate experiences with the RHR systems for both the PWR and BWR plants are summarized in Refs. 6 and 7, respectively. In Ref. 6, some 251 shutdown events from 1977 through 1981 for PWR plants were evaluated. About 100 of these events involved an actual loss or significant degradation in the RHRs while it was operating in a decay heat removal mode. The major safety implications of these events fall into three categories:

(1) Loss of reactor coolant inventory via the RHR;

(2) Inadvertent cold overpressurization of the RCS; and

(3) Loss of long-term decay heat removal capability via the RHR.

Similarly, Ref. 7 surveyed 480 BWR events involving the RHR, and 90 of those events involved an actual loss or significant degradation in the RHR. The safety implications are the same as those discussed for PWRs.

As a follow-on study to Ref. 6 findings, EPRI also sponsored a residual heat removal probabilistic study on the Zion (a PWR) plant. The study concluded, by comparison with the results of Zion Probabilistic Safety Study, that the annual frequency of fuel damage from events initiating during shutdown is less, by a factor of 5 to 20, than the frequency of core damage from transients initiated at power. However, the shutdown risk is highly dependent on operator error, and wider uncertainty exists for the shutdown model result than for the full power results. A similar probabilistic study was also performed for Brunswick Unit 1 (a BWR).

In support of the resolution of Generic Issue (GI) 99, which deals with loss of residual heat removal events in PWRs, BNL reanalyzed the Zion study (Ref. 4) by applying some modifications in the definition of outage phases and their duration and modeling of human cognitive errors. The estimated core damage frequency represented a non-trivial contribution to overall core damage frequency (at full power and shutdown).

Ref. 14 describes an inspection team report of a recent incident at Vogtle. The plant was operating in a mid-loop condition (reduced inventory) when a loss of offsite power occurred due to an accident in the switchyard. One of the onsite diesel generators was down for maintenance, and the other diesel failed to operate. Cooling for decay heat removal was lost for 36 minutes.

Ref. 15 is a conference paper that describes research by Electricite de France (EDF) that used PRA methodology to analyze the impact of technical specifications on core melt frequency of a French 900 MWe PWR during cold shutdown. By implementing changes in shutdown technical specifications in the area of scheduled unavailabilities of certain systems (mainly affecting RHR), the core melt frequency was reduced by a factor of four.

Ref. 16 summarizes the results of two PRAs: one done by EDF for a 900 MWe PWR, and the other by Institut de Protection et de Surete Nucleaire (IPSN) for a 1300 MWe PWR. Both PRAs investigated the importance of risk when the reactor is not at operating power (i.e., shut down with the RHR operating, or when refueling). These states accounted for over half (55%) of the total core melt frequency in the 1300-MWe plants, and almost one-third in the 900-MWe plants.

Findings of the studies discussed above will be reviewed in this task, and insights gained will be used to develop models and carry out analyses.

8. Conclusions

Work under this task is still in progress. However, as discussed in the previous section, some work has already been done to evaluate risk during non-full power modes. As a resolution of Generic Issue 99 and events occurring at operating plants, several information notices have been issued to licensees (e.g., Refs. 12 and 13). Administrative and procedural changes have been evaluated and implemented. Several hardware changes have also occurred.

9. Remarks

At the completion of this task, two NUREG/CR reports addressing risks at a PWR plant and at a BWR plant during the low power/shutdown modes are expected. These reports, in part, will deal with the risks which may result from the accidents initiated by the reactivity insertion (Chernobyl type accident). In addition they will also address risks initiated by system malfunctions, (Chernobyl could arguably have been initiated by human error) and other events. These studies may lead to recommendations regarding possible hardware, procedural, training, and staffing changes.

Following the establishemnt of NRC's research program in this area, a shutdown event (described in Section 7) occurred at the Vogtle plant. In response to this incident and growing concern in this area, the NRC Executive Director for Operations issued a June 21, 1990 memorandum regarding follow-up actions to the NRC inspection team's report on Vogtle (Ref. 14). Subsequently, a task plan was formulated to evaluate plant safety during shutdown operations to ensure that risk during all modes of operation is acceptably low. These evaluations will form the basis for: (1) any proposed changes to current technical specifications which govern shutdown operations, (2) changes in direction regarding the new standard technical specifications that are under developemnt by the staff, (3) recommendations to industry regarding emergency response procedures and outage management and control, and (4) modifications to the NRC inspection program. It is further planned to develop a working agreement with industry representatives to ensure cooperative efforts in addressing shutdown risk. Topics which will clearly involve significant interaction with industry groups include technical specifications, emergency operating procedures, and risk managemnt applied to shutdown activities. The major technical aspects are expected to be completed by approximately mid 1991.

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