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INTERIM RELIABILITY EVALUATION PROGRAM: ANALYSIS OF THE BROWNS FERRY, UNIT 1, NUCLEAR PLANT

APPENDIX C—SEQUENCE QUANTIFICATION

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FOREWORD

This report describes a risk study of the Browns Ferry, Unit 1, nuclear plant. The study is one of four such studies sponsored by the NRC Office of Research, Division of Risk Assessment, as part of its Interim Reliability Evaluation Program (IREP), Phase II. Other studies include evaluations of Arkansas One, Unit 1, by Sandia National Laboratories; Calvert Cliffs, Unit 1, by Science Applications, Inc.; and Millstone, Unit 1, by Science Applications, Inc. EG&G Idaho, Inc. was assisted by Energy Inc., Seattle, in its evaluation of the Browns Ferry, Unit 1, plant. Battelle-Columbus Laboratories provided information regarding the fission product releases that result from risk-significant accident scenarios. Sandia National Laboratories has overall project management responsibility for the IREP studies. It also has responsibility for the development of uniform probabilistic risk assessment procedures for use on future studies by the nuclear industry.

This report is contained in four volumes: a main report and three appendixes. The main report provides a summary of the engineering insights acquired in doing the study and a discussion regarding the accident sequences that dominate the risks of Browns Ferry, Unit 1. It also describes the study methods and their limitations, the Browns Ferry plant and its systems, the identification of accidents, the contributors to those accidents, and the estimating of accident occurrence probabilities. Appendix A provides supporting material for the identification of accidents and the development of logic models, or event trees, that describe the Browns Ferry accidents. Appendix B provides a description of Browns Ferry, Unit 1, plant systems and the failure evaluation of those systems as they apply to accidents at Browns Ferry. Appendix C generally describes the methods used to estimate accident sequence frequency values.

Numerous acronyms are used in the study report. For each volume of the report, these acronyms are defined in a listing immediately following the table of contents.

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NOMENCLATURE

$\bar{A}$	The complement of A (a success event if A is a failure event). ($\bar{A}$ may also be used to mean "unavailability.")
A	Alarm
AC	Alternating current
ACC	Accumulator
ADS	Automatic depressurization system
AH	Alarm-high
AO	Air operator
APRM	Average power range monitor
AT	Anticipated transient
ATWS	Anticipated transient without scram
BF1	Browns Ferry, Unit 1, nuclear plant
BI	Break isolation
BWR	Boiling water reactor
CAD	Containment atmosphere dilution
CCW	Condenser circulating water
CD	Complete dependence
CE	Conductivity element
CIS	Containment isolation system
Clg	Cooling
COND	Main condenser
CR-3	Crystal River, Unit 3, nuclear plant IREP study
CRD	Control rod drive
CRDH	Control rod drive hydraulic
CRDHS	Control rod drive hydraulic system
CRW	Clean rad waste
CS	Core spray
CS&T	Condensate storage and transfer
CSCS	Core standby cooling system
CSS	Core spray system
CST	Condensate storage tank
CV	Control valve
D	Demand
DC	Direct current
DEP	Depressurization
DG	Diesel generator
DHR	Decay heat removal
Diff	Different
DPI	Differential pressure indicator
DPIS	Differential pressure indicating switch
DPS	Differential pressure switch
DPT	Differential pressure transmitter
EAC	Equipment area cooling
ECCS	Emergency core cooling system
ECI	Emergency coolant injection
EECW	Emergency equipment cooling water
EHC	Electro-hydraulic control

EMI	Electrical Maintenance Instruction
EOI	Equipment Operating Instructions
EPRI	Electric Power Research Institute
EPS	Electrical power system
ESFAS	Engineered safety features actuation system
F(•)	Frequency of initiator in parentheses
FCV	Flow control valve
FE	Flow element
FI	Flow indicator
FIC	Flow indicating controller
FLS	Front-line system
FMEA	Failure mode effects analysis
FR	Flow recorder
FS	Flow switch
FSAR	Final Safety Analysis Report
FT	Flow transmitter
FWC	Feedwater control
FWCS	Feedwater control system
G	Green
GOI	General Operating Instructions
H	High
H/L	High/low
HCU	Hydraulic control unit
HCV	Hand control valve
HEP	Human error probability
HPCI	High pressure coolant injection
HPCS	High pressure core spray
HPI	High pressure injection
HS	Handswitch
HSS	High speed stop
HVAC	Heating, ventilation, and airconditioning
HX	Heat exchanger
I&C	Instrumentation and control
I&E	Inspection and enforcement
IMI	Instrument Maintenance Instruction
INJ	Injection
IREP	Interim Reliability Evaluation Program
IRM	Intermediate range monitor
L	Low
LA	Level alarm
LD	Low dependence
LER	Licensee Event Report
LIC	Level indicating controller
LIS	Level indicating switch
LL	Low-low
LOCA	Loss of coolant accident
LOSP	Loss of offsite power
LPCI	Low pressure coolant injection
LPI	Low pressure injection

LS	Limit switch
LSS	Low speed stop
LT	Level transmitter
M	Motor (operated valve)
MCR	Main control room
MD	Moderate dependence
MGU	Master governor unit
MMG	Motor generator
MMI	Mechanical Maintenance Instruction
MO	Motor operated
MOV	Motor-operated valve
MSC	Manual speed control
MSI	Main steam isolation
MSIV	Main steam isolation valve
MSL	Main steam line
NA; N/A	Not applicable
NC	Normally closed
NMS	Neutron monitoring system
NO	Normally open
OI	Operating Instructions
OL	Overload
OP	Overpressure protection
OP(C)	Overpressure protection (relief valves closed)
OP(O)	Overpressure protection (relief valves open)
PA	Pressure alarm
PB	Pipe break
PCIS	Primary containment isolation system
PCS	Power conversion system
PCV	Pressure control valve
PG	IREP Procedure Guide
PI	Pressure indicator
PORV	Power-operated relief valve
PRA	Probabilistic risk assessment
PS	Pressure switch
PSCWT	Pressure suppression chamber water transfer
PT	Pressure transmitter
PWR	Pressurized water reactor
Q(•)	Unavailability of system in parentheses
QA	Quality assurance
R	Red
RBCCW	Reactor building component cooling water
RBEDT	Reactor building equipment drain tank
RCB	Reactor coolant boundary
RCIC	Reactor core isolation cooling
RCS	Reactor coolant system
RCW	Raw cooling water
RCWS	Raw cooling water system
Recirc	Recirculation

RFP	Reactor feed pump
RFPT	Reactor feed pump turbine
RFWPT	Reactor feedwater pump turbine
RHR	Residual heat removal
RHRSW	Residual heat removal service water
RMOV	Reactor motor-operated valve
RMS	Remote manual switch
RPS	Reactor protection system
RPT	Recirculation pump trip
RS	Reactor subcriticality; reactor shutdown; reactor scram
RV(C)	Relief valve (closed)
RV(O)	Relief valve (open)
RWCU	Reactor water cleanup
RX	Reactor
S/D	Shutdown
S/RV	Safety relief valve
S/V	Safety valve
SBCS	Standby coolant supply
SBGT	Standby gas treatment
SCI	Short-term containment integrity
SD-BD	Shutdown board
SDV	Scram discharge volume
SIV	Scram instrument volume
SJAE	Steam jet air ejector
SLCS	Standby liquid control system
SORV	Stuck-open relief valve
SRM	Source range monitor
TA	Temperature alarm
TCV	Turbine control valve
TD	Time delay
TDC	Time delay contact
TDPU	Time delay pickup
TE	Temperature element
TIP	Traversing in-core probe
TMI	Three Mile Island
TR	Temperature recorder
Trans	Transient
TS	Technical Specifications; torque switch
TVA	Tennessee Valley Authority
UV	Undervoltage
V	Volts
VB	Vacuum breaker
VO	Valve open
VS	Vapor suppression
VSS	Vapor suppression system
VWI	Vessel water inventory
ε	An insignificant quantity, generally less than 10^{-8}

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ANALYSIS OF THE BROWNS FERRY, UNIT 1, NUCLEAR PLANT

APPENDIX C--SEQUENCE QUANTIFICATION

1. APPROACH

The purpose of Appendix C is to describe the method used to quantify the accident sequences defined in Appendix A that result in a core melt. The basic approach to calculate a sequence frequency is to multiply the probabilities associated with the various events depicted in the accident sequence. That is, the frequency of the sequence is equal to the frequency of the initiating event multiplied times the probability of system (or systems) failure. Sequence quantification was based on the systemic event trees for LOCAs and transients. For each system, the unavailability was calculated from the fault trees using the Reliability Analysis System (RAS) computer code.[1] System minimal cut sets of order five or more and having a probability value less than 10^{-8} were truncated.

Dependencies were incorporated in the risk analysis at various stages. During event tree formulation, functional dependencies between the various accident mitigating systems were depicted in the accident sequence construction. The fault trees for the front-line systems were constructed considering potential interface dependencies such as human error, test and maintenance, and support systems. Finally, in the event tree quantification, system fault trees were reduced by Boolean techniques using the COMCAN code[2] to pinpoint any further common dependencies between systems.

The potential for recovery was considered for those sequences where the dominant contributors to sequence frequency were recoverable.

1.1 System Unavailabilities

Each front-line and support system fault tree was evaluated using the RAS computer code. The RAS code resolves the fault tree into its minimal cut sets and evaluates the system unavailability based on the failure data associated with the basic events. RAS calculates a time dependent unavailability that was specified for this analysis as 8 hours. The analysis considered stable hot shutdown as successful core cooling. After discussion with TVA personnel,[3] the time limit of 8 hours was chosen as a reasonable limit for reaching stable hot shutdown conditions.

The RAS code calculates component unavailabilities based on the failure rate data. Failure rates may be entered either as demand rates or hourly rates.

1.1.1 Demand Failure Probabilities

Standby safety systems are characterized by having many components that are required to change state when the system is demanded. The RAS computer code treats the demand failure probabilities for these components as a constant unavailability. That is, these values are unaffected by the length of time the system is required to operate (8-hour mission time as noted above). However, it is possible that these components could have been tested, found to be failed, and undergoing repair at the time the accident or transient places the demand for the component. This unscheduled maintenance contribution to the unavailability of the component (and therefore, system) is negligible compared with demand failures when the component repair time is small compared to the testing interval. This is shown as follows:

Q = (unavailability on demand) + (unavailability due to unscheduled maintenance)

$$= Q_D + \frac{Q_D T_R}{T} = Q_D(1 + T_R/T),$$

where

Q_D = demand unavailability

T = testing interval

T_R = repair time.

If $T_R << T$, then

$$Q = Q_D (1 + 0)$$

$$= Q_D.$$

A review of Browns Ferry component data shows that this is the case. Typically, T_R is less than 3 days and T is 1 month. Thus, the unscheduled maintenance contribution at Browns Ferry is negligible for demand failures.

1.1.2 Time-Dependent Failure Probabilities

Some of the failure modes considered for components of standby safety systems are characterized as "fails to continue to run/operate given the initial demand on the component was successful." The RAS computer code calculates a time-dependent unreliability (λT_m) for these component failure modes based on the mission time; that is, the component is required to work for the length of the mission (T_m). Since the operability of the component is known at essentially mission time zero (given that the demand was successful), it is not necessary to consider any unavailability contribution due to unscheduled maintenance.

However, for components with failure rates (λ) given per hour, where it is not known at mission time zero if the component is in a failed state, it is necessary to determine the component unavailability manually based on the testing interval and repair time and then to enter this point estimate as a constant unavailability to the RAS code.

The unavailability for these components depends primarily on the time to detect faults, which depends on their testing interval (T). Assuming that the accident or transient is equally likely to occur at any time during the testing interval results in an average value for the component unavailability of $\lambda T/2$. The time to detect failures was based on the testing frequencies in the surveillance procedures associated with the components of the system being modeled. For example, if the surveillance procedures for a certain system required a system flow check be performed once per month, then this test frequency was used to determine the time to detect failures associated with the pump in that system.

In addition to this average unavailability over the testing interval, it is necessary to account for the component unavailability due to unscheduled maintenance. That is, for components with hourly failure rates where it is not known at mission time zero if it is in a failed state, a modification to the unavailability must be made to account for the fact that the component may be undergoing repair at mission time zero due to a fault that occurred within the span of mission time zero minus the component repair time.

Correction to the unavailability is made in the following manner. The probability of the component entering a failed state during any testing interval is estimated as λT, where T is the testing interval. The maintenance unavailability during any testing interval equals the probability of being in a down state times the fraction of the interval during which the component is in repair. This fraction is the repair time T_R divided by the testing interval T. Thus, the unavailability due to unscheduled maintenance is $(\lambda T)(T_R/T)$, which equals λT_R. The unavailability for these components is modified by adding this factor to the previously calculated fault detection based value. This combined value is entered into the RAS code as a constant unavailability.

Q = (average unavailability over testing interval)
+ (unavailability due to unscheduled maintenance)

$$= \lambda T/2 + \lambda T_R$$

$$= \lambda(T/2 + T_R).$$

Repair times were taken from Table III 5-2 of WASH-1400[4] for pumps, valves, diesels, and instrumentation. Electrical components (other than diesels) were assumed to have the same repair rate as that shown for instrumentation (7 hours). Table C-1 summarizes the treatment of component unavailabilities used for system quantification.

1.2 Treatment of Commonalities

Wherever possible, the RAS code was used to evaluate the unavailability of combinations of systems. However, due to the complexity of the individual system fault trees, computer core space, and processing time limitations, this method was not viable for some system combinations and the following approach was used.

1.2.1 Use of COMCAN

The unavailability of two systems in an AND logic configuration is equal to the product of the individual unavailabilities if the cut sets for

TABLE C-1. COMPONENT UNAVAILABILITIES

Failure Mode	Unavailability without Unscheduled Maintenance	Unscheduled Maintenance	Final Unavailability
Fails to start/operate when required	Q_D	$Q_D T_R/T$	Q_D*
Fails to continue to run/operate, given start	λT_m	None	λT_m
Fails to run/operate, successful start not given	$\lambda T/2$	λT_R	$\lambda(T/2 + T_R)$

Q_D = demand unavailability
T_R = repair time
T = testing interval
T_m = mission time

* $T_R << T$.

the two systems are independent. If dependent cut sets exist, then the unavailability of the combined systems equals the product of the independent cut sets plus the value of the dependent sets. Equations (C1) and (C2) apply.

$$Q(A \cap B) = Q(A)Q(B), \text{ if A and B are independent} \tag{C1}$$

$$Q(A \cap B) = Q(A_I)Q(B_I) + Q(D), \text{ if A and B are dependent,} \tag{C2}$$

since

$$Q(A) = Q(A_I) + Q(D)$$

$$Q(B) = Q(B_I) + Q(D),$$

where

$Q(A_I)$ = unavailability of cut sets in A independent of B

$Q(B_I)$ = unavailability of cut sets in B independent of A

$Q(D)$ = unavailability of dependent cut sets in both.

To calculate the unavailability of the dependent cut sets for the two systems, the COMCAN code was used to identify the commonalities between the systems. The COMCAN code does not quantify the commonalities, but identifies those combinations of similar events which can cause both systems to fail.

The fault tree models for the systems evaluated on COMCAN were modified so that the first three characters in the eight-character code for each support system were the same (SUP). This allowed COMCAN to identify all combinations of support systems that could cause both systems to fail. COMCAN also identified, based on the first three characters in the eight-digit code, all combinations of similar basic events (i.e., those with the same three characters) that could cause both systems to fail. These cut sets were then evaluated manually or using RAS.

The unavailability of the potentially dependent cut sets identified by COMCAN were compared to the unavailability of the systems assuming independence. If the unavailability of these common sets was at least two orders of magnitude less than the unavailability of both systems assuming independence, then the unavailability assuming independence was used. If the unavailability of the common sets was less than two orders of magnitude smaller than the unavailability assuming independence, the sum of the two was used to represent the unavailability of the two system combination. Equations (C3) through (C5) apply.

$$Q(A \cap B) = Q(A_I)Q(B_I) + Q(D). \tag{C3}$$

If $Q(D) << Q(A)Q(B)$, then

$$Q(A \cap B) = Q(A_I)Q(B_I) \simeq Q(A)Q(B). \tag{C4}$$

If $Q(D) \geq \frac{Q(A)Q(B)}{100}$, then

$$Q(A \cap B) = Q(A_I)Q(B_I) + Q(D), \text{ or}$$

$$\simeq Q(A)Q(B) + Q(D). \tag{C5}$$

Note that the product of $Q(A)Q(B)$ is always greater than or equal to $Q(A_I)Q(B_I)$. No attempt was made to determine $Q(A_I)$ or $Q(B_I)$. Instead, the minimal conservatism caused by using $Q(A)$ and $Q(B)$ in the equations for finding $Q(A \cap B)$ was not significant enough to justify the time and expense of calculating $Q(A_I)$ or $Q(B_I)$ (by removing $Q(D)$ from the fault trees for A and B and recalculating their unavailabilities using RAS).

1.2.2 Treatment of Commonalities Not Found Using COMCAN

A potential problem involved with using COMCAN in this manner to calculate the unavailability of two systems is that only those cut sets that are combinations of events with the same first three characters in their code identifier are identified. Although COMCAN has other uses, this analysis used the code only in one mode of operation as described before. Therefore, if a cut set X exists in System A and a cut set XY exists in System B where the first three characters of the basic event Y are different from those in X, COMCAN will not identify either cut set as a potential common candidate. Although X and XY are not the same cut set and are thus only partially dependent, the dependence that comes from X having an effect on both systems must be accounted. Equations (C6) through (C9) apply to this situation.

Let

$$Q(A) = Q(A_I) + COM(A \cap B) + Q(X), \text{ and} \tag{C6}$$

$$Q(B) = Q(B_I) + COM(A \cap B) + Q(XY), \tag{C7}$$

where

$Q(A_I)$ and $Q(B_I)$ are defined as before

$COM(A \cap B)$ = unavailability of commonalities found using COMCAN

$Q(X)$ = unavailability of a cut set in A

$Q(XY)$ = unavailability of a cut set in B;

then

$$Q(A \cap B) = Q(A_I)Q(B_I) + COM(A \cap B) + Q(XY), \tag{C8}$$

but using COMCAN as previously described only produces

$$Q(A \cap B) = Q(A)Q(B) + COM(A \cap B). \tag{C9}$$

The problem of identifying all the X,XY combinations possible between two systems at this point is impractical. Therefore, a numerical bounding analysis was used to determine whether or not such a combination, if it existed, would have a value large enough to cause a significant nonconservative error in the evaluation of $Q(A \cap B)$ previously discussed.

In order for the X,XY combination to be important, the unavailability Q(XY) must be on the same order of magnitude as $Q(A \cap B)$ previously calculated. The RAS code lists cut set probabilities in descending order of magnitude such that the unavailability of the first cut set listed is always greater than or equal to that of the second cut set listed. Equations (C10) and (C11) apply.

From RAS,

$$Q(A) = Q(A_1) + Q(A_2) + \ldots + Q(A_n), \tag{C10}$$

where

$$Q(A_1) \geq Q(A_2) \geq \ldots \geq Q(A_n). \tag{C11}$$

Therefore, the highest valued cut set for Systems A and B is readily identified and hereafter identified as A_T or B_T. Thus, if the X,XY combination exists, $Q(X) \leq Q(A_T)$ and $Q(XY) \leq Q(B_T)$.

Assuming that the X,XY combination exists and that the unavailability Q(XY) is greater than or equal to one tenth of the value of $Q(A \cap B)$ calculated previously, a lower bound on the value of Y can be determined. Equations (C12) through (C19) apply.

If

$$Q(XY) \geq \frac{Q(A \cap B)}{10}, \text{ and} \tag{C12}$$

$$Q\ (A_T) \geq Q(X), \tag{C13}$$

then

$$Q(A_T Y) \geq Q(XY). \tag{C14}$$

Therefore,

$$Q(A_T Y) \geq Q(XY) \geq \frac{Q(A \cap B)}{10}, \text{ or} \tag{C15}$$

$$Q(A_T Y) \geq \frac{Q(A \cap B)}{10}. \tag{C16}$$

But

$$Q(A_T Y) = Q(A_T)Q(Y) \tag{C17}$$

since they are independent.

Thus,

$$Q(A_T)Q(Y) \geq \frac{Q(A \cap B)}{10}, \text{ and} \tag{C18}$$

$$Q(Y) \geq \frac{Q(A \cap B)}{10 \cdot Q(A_T)}. \tag{C19}$$

Using this lower bound for Q(Y), the basic event list for System B is searched to see if any basic events exist in B with unavailabilities greater than Q(Y). If not, then there can not exist any X,XY combinations whose unavailability Q(XY) is significant compared to $Q(A \cap B)$.

If there are basic events in B that have unavailabilities greater than Q(Y), a list of these events is made. Then the cut set list of System B is examined. Each cut set containing a basic event from the list is examined to see if its comembers appear as cut sets of A. If not, then no X,XY combinations of significance exist. If the comembers are cut sets of A by themselves, then the value of that XY type cut set is added to the previously calculated value of $Q(A \cap B)$. The process is then repeated to determine if any X,XY combinations exist where X is in System B and XY is in A.

It should be noted that this method will detect dependencies not found using COMCAN that are within one order of magnitude of the value of $Q(A \cap B)$. Dependencies with values less than $Q(A \cap B)/10$ may not be located. The choice for $Q(A \cap B)/10$ as a bounding value is an arbitrary one based on engineering judgement and familiarity with the systems. The bounding value could be chosen as $Q(A \cap B)/100$ or $Q(A \cap B)/1000$ if desired. The latter choices merely expand the scope of the manual search of the system cut set lists. The method described above is conservative, however, because: $Q(A_T)$ is an upper bound for the unavailability of potential cut sets X of A completely contained in cut sets of B; and an examination of the Browns Ferry cut set lists, for cases where this issue applies, shows that $Q(A_T)$ is always several orders of magnitude greater than the highest-valued unavailability of a cut set in A containing any basic events that are also in B.

1.3 Treatment of Complement or Success Sets

It is necessary in sequence quantification to account for the effect of success of one system in an AND combination with failure of another. Since the RAS code does not deal with complement or success sets, they were treated in the following manner. If the systems are totally independent then Equation (C20) applies.

$$Q(\bar{A} \cap B) = Q(\bar{A})Q(B), \tag{C20}$$

where $\bar{A}$ designates success for A.

If there are common cut sets between A and B, then Equations (C21) and (C22) apply.

$$Q(\bar{A} \cap B) = Q(\bar{A})Q(B_I) \tag{C21}$$

$$Q(\bar{A} \cap B) = Q(\bar{A})\ [Q(B) - COM(A \cap B)], \tag{C22}$$

where

$COM(A \cap B)$ = value of common cut sets of A and B

$Q(B_I)$ = value of cut sets of B independent of A.

A screening tool used in the sequence quantification determines when potential commonalities of significance may exist. If $Q(B) >> Q(A)$, then even if all of A is assumed to be common with B, Q(B) is still essentially equal to $Q(B_I)$. If $Q(B) \leq Q(A)$, then a COMCAN search is used to identify the potential commonalities. If the unavailability of the commonalities $COM(A \cap B)$ is much less than Q(B), then again Q(B) is essentially equal to $Q(B_I)$. Otherwise, $Q(B_I)$ is calculated by subtracting $COM(A \cap B)$ from Q(B). Also, since $Q(\bar{A})$ equals $1 - Q(A)$ and Q(A) is usually small, $Q(\bar{A}) \approx 1$ in most cases. Therefore, Equation (C23) applies.

$$Q(\bar{A} \cap B) \approx Q(B) - COM(A \cap B). \tag{C23}$$

Therefore, success sets, or complements, are accounted for by either recognizing the nonsignificant potential impact or by evaluating the known commonalities for significance and including their effect where appropriate.

1.4 Treatment of Initiator Effects on Mitigating Systems

Some of the LOCA initiators have the potential to render LOCA mitigation systems partially or completely inoperable. To account for this possibility in the sequence calculations, the following procedure was used.

If a LOCA initiator could disable a mitigating system, the length of piping for the mitigating system susceptible to that LOCA was calculated using TVA supplied isometric drawings. Then, the total length of piping susceptible to that initiator was calculated. It was assumed that for a particular break size, the LOCA was equally likely to occur at any point on the piping susceptible to the LOCA. The unavailability of the mitigating systems was calculated considering the initiator affecting the system and then without considering the effect of the initiator. Therefore, the sequence frequency is the sum of two terms. The first term is the product of the probability of a break occurring in a location that affects the mitigating systems and the unavailabilities of those systems. The second term is the product of the probability of the break occurring in a location that does not affect the mitigating systems and the unavailability of those systems under that condition. The example calculation in Section 2 provides an example of this method.

For transient initiators, Section 2.3.2 of Appendix A describes the methodology for identifying potential transient initiators that would affect the unavailabilities of the mitigating systems. Only the loss of offsite power (LOSP) event was significant in this regard. A separate event tree exists for this particular initiator.

1.5 Treatment of Potential Logic Loops

A potential problem in the quantification of $T_pR_BR_A$, as well as the other sequences involving loss of offsite power ($T_pQR_BR_A$ and $T_pKR_BR_A$), is the presence of loop dependencies. That is, the EECW system requires electrical power in order to function. Given a loss of offsite power, this power must come from the diesel generators. However, the diesel generators need EECW or they will eventually fail.

This problem was resolved by recognizing three important considerations. First, the diesels can operate for some finite time without rated flow from the EECW system. Second, the diesels that supply EECW are not all the same as those supplying power to other mitigating systems. Figure C-1 shows the power dependencies between the RHR, RHRSW, and EECW systems. Third, EECW represents a common mode failure not only of the diesel generators but all AC powered mitigating systems.

Therefore, quantification of sequences involving loss of offsite power requires a special process. First, the unavailability of EECW is calculated assuming that diesel failures are not caused by loss of EECW. In other words, EECW does not cause its own failure. Then, the mitigating systems' unavailabilities are calculated assuming successful EECW operation. This

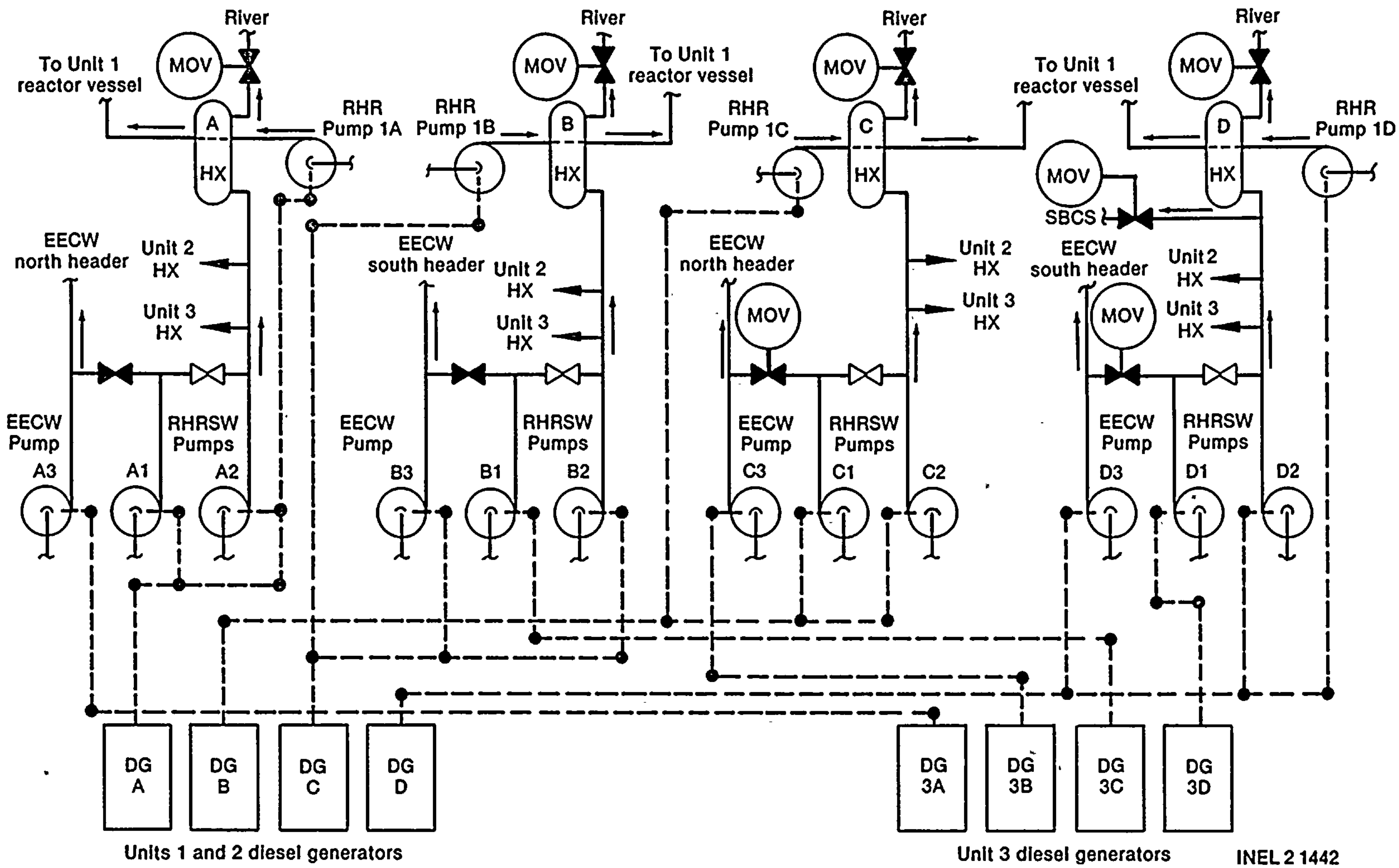


Figure C-1. RHR/RHRSW/EECW system power dependencies.

value added to the EECW unavailability represents the total system unavailability. That is, the system unavailability can be dichotomized into the unavailability due to EECW faults and the unavailability assuming EECW works. When considering the case where two or more AC systems must fail, the EECW unavailability is treated as a common mode failure of both AC systems. Thus, for a general sequence, the frequency is given by the following equations.

$$\begin{aligned} F(\text{seq}) &= F(\text{LOSP})[Q(\text{DC powered systems}) \cap Q(\text{AC powered systems})] \\ &= F(\text{LOSP})Q(\text{DC systems})[Q(\text{AC systems given EECW works}) \cup Q(\text{EECW})] \\ &= F(\text{LOSP})Q(\text{DC systems})Q(\text{AC systems given EECW works}) \\ &\quad + F(\text{LOSP})Q(\text{DC systems})Q(\text{EECW}). \end{aligned}$$

In general, the unavailability of EECW was about an order of magnitude higher than the unavailability of the combinations of AC powered systems. Therefore, these sequences tended to be dominated, at least in part, by EECW faults.

2. EXAMPLE CALCULATION

The intermediate steam break was chosen for this example calculation since its sequence quantification requires the use of all the methods described previously. All of the intermediate steam break sequences are evaluated in this example. Figure C-2 is the systemic event tree for the intermediate steam break with the system and sequence values filled in.

2.1 Initiator Frequency

The intermediate steam break frequency was determined to be 2.1×10^{-4} per reactor-year. Since 70% of all piping susceptible to intermediate breaks is steam piping, the remaining 30% of the piping would cause an intermediate liquid break if it ruptured. The WASH-1400 frequency of 3×10^{-4} per reactor-year was used as the frequency of all intermediate breaks. Assuming that an intermediate break was equally likely to occur at any point in the piping susceptible to intermediate breaks, the frequency of intermediate steam breaks would be 70% of 3×10^{-4}, or 2.1×10^{-4} per reactor-year.

2.2 Example System Unavailabilities

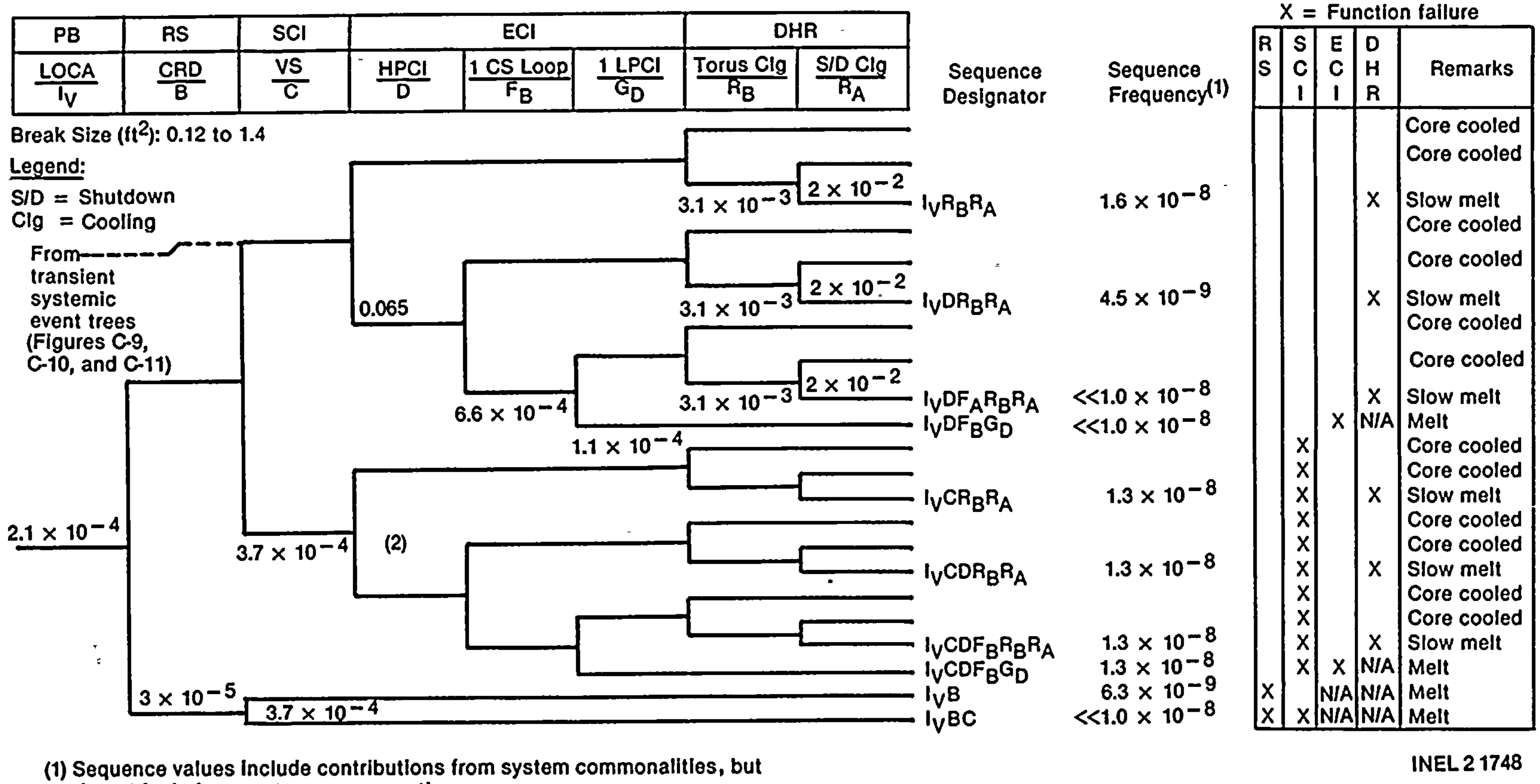
2.2.1 Reactor Subcriticality

The unavailability for the reactor subcriticality function was taken from NUREG-0460[5] to be 3×10^{-5} per demand. A qualitative model was developed (in Appendix B, Section 2.9.4) for the control rod drive mechanism but was not quantified since insufficient data exists for estimating the occurrence rate for common mode failures identified in the model.

2.2.2 Vapor Suppression System

Short-term containment integrity (SCI) is maintained and containment overpressurization is prevented by directing the steam from the break through the downcomer piping to a position below the water level of the torus. This action condenses the steam formed and provides a "scrubbing" effect to trap radioactivity from the steam in the torus water rather than allowing it to remain airborne. Failure of the vapor suppression system will result in a containment rupture and a release of radioactivity. The amount of radioactivity released depends on the performance of the ECI and DHR systems.

Bypass leakage from the drywell to the airspace of the wetwell could pressurize the wetwell airspace to the same pressure as the drywell, preventing the pressure differential required to force the steam through the downcomer piping into the pool of water in the suppression chamber (torus). Therefore, the quenching and scrubbing features will not be accomplished and overpressurization will result. The RAS code was used to evaluate the vapor suppression system unavailability from the fault tree given in Appendix B, Section 2.8.4, and the value 3.7×10^{-4} was obtained. Dominat cut sets are listed in Table B-53 of Appendix B.



(1) Sequence values include contributions from system commonalities, but do not include operator recovery actions.
(2) Conditional probability of torus failure gives VS failure ~ 0.162.

Figure C-2. LOCA systemic event tree for intermediate steam break (I_V), with system and sequence values filled in.

2.2.3 Emergency Cooling During Injection

There are three systems available to mitigate the intermediate liquid break. These are the HPCI, core spray, and LPCI systems. The designations for these systems for the systemic event trees are D, F_B, and G_D, respectively. Table C-2 lists each system, its failure criteria, the unavailability for each calculated by the RAS code, and Appendix B tables for lists of dominant cut sets. Operation of any of these three systems is sufficient to perform the ECI function.

TABLE C-2. INTERMEDIATE STEAM BREAK ECI CRITERIA

System Designation	Failure Criteria	Unavailability	Appendix B Dominant Cut Set Table Number
D	HPCI fails to inject rated flow to core	6.5×10^{-2}	B-33
F_B	Less than one of two core spray loops delivers rated flow to core	6.6×10^{-4}	B-47
G_D	Less than one of four LPCI pumps delivers rated flow to core	1.1×10^{-4}	B-17 B-18

2.2.4 Decay Heat Removal

There is only one system that performs the DHR function, the RHR system. However there are two modes of RHR used to mitigate the intermediate liquid break: torus cooling (R_B) and shutdown cooling (R_A). Either torus cooling or shutdown cooling must operate to remove decay heat from the reactor to prevent containment overpressurization and eventual core melt.

Table C-3 summarizes the failure criteria for each mode of RHR operation and presents Appendix B dominant contributor tables, and the unavailability from the RAS code for each. It is noteworthy that, although R_A only requires one of four pumps and heat exchangers where R_B requires two of four, the unavailability of R_A is significantly higher than that of R_B. This is because shutdown cooling (R_A) uses a single suction line with three MOVs, whereas torus cooling (R_B) uses a double suction line with no valves required to change position.

TABLE C-3. DECAY HEAT REMOVAL FAILURE CRITERIA

System Designation	Failure Criteria	Unavailability	Appendix B Dominant Cut Set Table Number
R_A	Less than one of four pump and heat exchanger combinations recirculating reactor coolant	2.0×10^{-2}	B-19 B-20
R_B	Less than two of four pump and heat exchanger combinations recirculating torus water	3.1×10^{-3}	B-21

2.3 Sequence Calculations

There are 10 core melt sequences on the intermediate steam break systemic event tree. Six sequences involve failure of the DHR function, two involve failure of the reactor subcriticality function, and two involve failure of the ECI function. The sequences are designated by the initiating event letter code, I_V, and the system(s) failure code associated with the particular sequence.

2.3.1 Sequence $I_V R_B R_A$

In this sequence the reactor subcriticality, vapor suppression, and HPCI systems perform satisfactorily, but the torus cooling and shutdown cooling modes of the RHR system fail. The unavailability of torus cooling and shutdown cooling for this sequence is 7.6×10^{-5} as shown below.

$$Q(R_B R_A) = Q(\bar{B} \cap \bar{C} \cap \bar{D} \cap R_B \cap R_A)$$

$$= Q(R_B \cap R_A) - COM[R_B \cap R_A \cap (B \cup C \cup D)]$$

$$= Q(R_B \cap R_A) - COM(R_B \cap R_A \cap B) - COM(R_B \cap R_A \cap C) - COM(R_B \cap R_A \cap D)$$

$$= Q(R_B \cap R_A) - 0 - 0 - 0$$

$$= Q(R_B)\, Q(R_A) + COM(R_B \cap R_A)$$

$$= (3.1 \times 10^{-3})(2.0 \times 10^{-2}) + 1.4 \times 10^{-5}$$

$$= 7.6 \times 10^{-5}.$$

The term $COM[(R_B \cap R_A \cap (B \cup C \cup D)]$ accounts for success of the systems preceeding $R_B \cap R_A$ precluding some failure modes of $R_B \cap R_A$. In this case, they are negligible. The term $COM(R_B \cap R_A)$ accounts for commonalities between R_B and R_A. These commonalities were identified using the methods of Section 1.2 above. Three dominant cut set tables in Appendix B (Tables B-19 through B-21) apply to the RHR system (two tables for two loops of R_A and one table for R_B). However, cut sets that simultaneously fail all three systems cannot be readily identified from these tables. Rather, a case-by-case examination of potential commonalities flagged by COMCAN runs was required. The results showed that commonalities between R_B and R_A are primarily due to minimum-flow bypass valve faults.

Since the initiator has no effect on torus cooling or shutdown cooling, the sequence frequency is equal to the product of the initiator frequency and the systems unavailability:

$$P(I_V R_B R_A) = F(I_V)\ Q(R_B R_A)$$

$$= (2.1 \times 10^{-4})(7.6 \times 10^{-5})$$

$$= 1.6 \times 10^{-8}.$$

2.3.2 Sequence $I_V D R_B R_A$

This sequence is similar to the previous one, but in this sequence the HPCI system fails. The core spray system operates to replace the lost reactor coolant. Subsequently, torus cooling and shutdown cooling fail. The unavailability for HPCI, torus cooling and shutdown cooling is 4.9×10^{-6} as shown below.

$$Q(DR_B R_A) = Q\ (\bar{B} \cap \bar{C} \cap D \cap \bar{F}_B \cap R_B \cap R_A)$$

$$= Q(D \cap R_B \cap R_A) - COM[D \cap R_B \cap R_A \cap (B \cup C \cup F_B)]$$

$$= Q(D \cap R_B \cap R_A) - 0$$

$$= Q(D)\ Q(R_B \cap R_A) + COM\ (D \cap R_B \cap R_A)$$

$$= (0.065)(7.6 \times 10^{-5}) + 0$$

$$= 4.9 \times 10^{-6}.$$

The term $COM[(D \cap R_B \cap R_A \cap (B \cup C \cup F_B)]$ accounts for the success of the reactor subcriticality, vapor suppression, or core spray systems precluding some failure modes of $D \cap R_B \cap R_A$. In this case, they are negligible. The term $COM(D \cap R_B \cap R_A)$ accounts for commonalities between the HPCI system and $R_B \cap R_A$. These are also negligible.

Unlike the previous sequence, the initiator can affect the mitigating systems for this sequence since 23.2% of the piping susceptible to intermediate steam breaks is HPCI piping. The first term in the equation below, $0.232\ F(I_V)Q(R_BR_A)$, represents the frequency when the break is on the HPCI line. Note that D does not appear in this term since it is assumed that the break disables HPCI. The term $0.768F(I_V)Q(DR_BR_A)$ represents the sequence frequency for intermediate steam breaks that do not affect HPCI operability.

$$P(I_V DR_B R_A) = 0.232\ F(I_V)\ Q(R_B R_A) + 0.768\ F(I_V)\ Q(DR_B R_A)$$

$$= (0.232)(2.1 \times 10^{-4})(7.6 \times 10^{-5}) + (0.768)(2.1 \times 10^{-4})(4.9 \times 10^{-6})$$

$$= 3.7 \times 10^{-9} + 8.0 \times 10^{-10}$$

$$= 4.5 \times 10^{-9}.$$

2.3.3 Sequence $I_V DF_B R_B R_A$

After successful operation of the reactor subcriticality and vapor suppression systems, both the HPCI and core spray systems fail. The LPCI mode of RHR functions properly but torus cooling and shutdown cooling fail. The unavailability for the mitigating systems for this sequence is negligible as shown below.

$$Q(DF_B R_B R_A) = Q(\bar{B} \cap \bar{C} \cap D \cap F_B \cap \bar{G}_D \cap R_B \cap R_A)$$

$$= Q(D \cap F_B \cap R_B \cap R_A) - COM[D \cap F_B \cap R_B \cap R_A \cap (B \cup C \cup G_D)]$$

$$= Q(D \cap F_B \cap R_B \cap R_A) - COM(D \cap F_B \cap R_B \cap R_A \cap B)$$

$$- COM(D \cap F_B \cap R_B \cap R_A \cap C)$$

$$- COM\ (D \cap F_B \cap R_B \cap R_A \cap G_D)$$

$$= Q(D \cap F_B \cap R_B \cap R_A) - COM(D \cap F_B \cap R_B \cap R_A \cap G_D) - 0 - 0$$

$$= Q(D)\ Q(F_B)\ Q(R_B \cap R_A) + COM(D \cap F_B \cap R_B \cap R_A)$$

$$- COM(D \cap F_B \cap R_B \cap R_A \cap G_D)$$

$$= Q(D)\ Q(F_B)\ Q(R_B \cap R_A) + 0 - COM(D \cap F_B \cap R_B \cap R_A \cap G_D).$$

The term $COM[D \cap F_B \cap R_B \cap R_A \cap (B \cup C \cup G_D)]$ accounts for success of the reactor subcriticality, vapor suppression, or LPCI mode of RHR precluding some failure modes of $D \cap F_B \cap R_B \cap R_A$. In this case the contribution from reactor subcriticality or vapor suppression is negligible.

However, the success of the LPCI mode precludes the dominant contributors to failure of $R_B \cap R_A$. Therefore, the value of $Q(DF_BR_BR_A)$ is much less than the 3.9×10^{-9} value obtained by ignoring the success of the LPCI mode.

Since the initiator frequency for this sequence is 2.1×10^{-4} and the unavailability $Q(DF_BR_BR_A)$ must be less than 3.9×10^{-9}, the sequence frequency will be much less than 1.0×10^{-8}. As discussed later, 1.0×10^{-8} was chosen as the initial screening value for determining candidate dominant sequences. Since it is obvious that this sequence frequency will be less than 1.0×10^{-8}, no further quantification is necessary.

2.3.4 Sequence $I_VDF_BG_D$

Following successful reactor subcriticality and vapor suppression system operation, none of the ECI systems operate to restore reactor vessel water level. The unavailability for the mitigating systems for this sequence is 7.2×10^{-9}, as shown below.

$$Q(DF_BG_D) = Q(\bar{B} \cap \bar{C} \cap D \cap F_B \cap G_D)$$

$$= Q(D \cap F_B \cap G_D) - COM[D \cap F_B \cap G_D \cap (B \cup C)]$$

$$= Q(D \cap F_B \cap G_D) - 0$$

$$= Q(D)\ Q(F_B \cap G_D) + COM(D \cap F_B \cap G_D)$$

$$= Q(D)\ Q(F_B \cap G_D) + 0$$

$$= Q(D)\ [Q(F_B)\ Q(G_D) + COM(F_B \cap G_D)]$$

$$= (0.065)(7.3 \times 10^{-8} + 3.4 \times 10^{-8}$$

$$= 7.2 \times 10^{-9}.$$

The term $COM[D \cap F_B \cap G_D \cap (B \cup C)]$ accounts for success of either reactor subcriticality or vapor suppression precluding some failure modes of $D \cap F_B \cap G_D$. In this case they are negligible. The term $COM(D \cap F_B \cap G_D)$ accounts for commonalities between the HPCI system and the combination of $F_B \cap G_D$. The term $COM(F_B \cap G_D)$ accounts for commonalities between $F_B \cap G_D$ that are primarily due to combinations of electric power faults.

In this sequence also, the initiator can effect the mitigating systems since 23.2% of piping susceptible to intermediate steam breaks is HPCI piping and 3.8% is core spray piping. The term $0.232\ Q(F_BG_D)$ accounts for that percentage of breaks that disables HPCI. Therefore, D does not appear in this term. The next term, $0.038\ Q(DF_B'G_D)$, accounts for those breaks that disable one core spray loop. The term F_B' represents the unavailability of the remaining loop; its probability is less than that of F_B since there are no longer two loops available in this case. The last term, 0.730 $Q(DF_BG_D)$, accounts for breaks not occurring on any of the mitigating systems. As with the previous sequence, this sequence is designated as having a frequency less than 1×10^{-8}.

$$P(I_VDF_BG_D) = F(I_V)[0.232\ Q(F_BG_D) + 0.038\ Q(DF_B'G_D) + 0.730\ Q(DF_BG_D)]$$

$$= (2.1 \times 10^{-4})[(0.232)(1.1 \times 10^{-7}) + (0.038)(0.065)(5.7 \times 10^{-6})$$

$$+ 0.730\ (7.2 \times 10^{-9})]$$

$$= (2.1 \times 10^{-4})(2.6 \times 10^{-8} + 1.4 \times 10^{-8} + 5.3 \times 10^{-9})$$

$$= (2.1 \times 10^{-4})(4.5 \times 10^{-8})$$

$$< 1 \times 10^{-8}.$$

2.3.5 Sequences $I_VCR_BR_A$, $I_VCDR_BR_A$, $I_VCDF_BR_BR_A$, and $I_VCDF_BG_D$

These sequences are identical to the four sequences just discussed except that the vapor suppression system fails to operate properly and overpressurization of the containment occurs. Overpressurization causes the containment to rupture. This could impact the ability of the ECI and DHR functions if the rupture occurs below the water line of the torus. Assuming that the break is equally likely to occur anywhere on the primary containment boundary, the probability of the break occurring below the torus water line is equal to the ratio of surface area of the containment below the water line to the total surface area (about 0.162). Therefore, the unavailability of the ECI and DHR systems given vapor suppression system failure is 0.162 + 0.838 (the unavailability for those systems from the vapor suppression system success sequences). In each case, the dominant contributor to ECI or DHR failure is where the rupture occurs below the torus water line. Thus, the unavailability of the mitigating systems for these sequences are equal and have a value of 6.0×10^{-5} as shown below. The designator X in this case represents the combination of any of the four previous ECI or DHR systems.

$$Q(CX) = Q(\bar{B} \cap C \cap X)$$

$$= Q(C \cap X) - COM(B \cap C \cap X)$$

$$= Q(C \cap X) - 0$$

$$= Q(C)(0.162 + 0.838\ Q(X))$$

$$= Q(C)(0.162)$$

$$= 6.0 \times 10^{-5}.$$

Therefore, the sequence frequency for each of these sequences is equal. The value of the frequency is the product of the initiator frequency and the systems unavailability.

$$P(I_V CX) = F(I_V)Q(CX)$$

$$= (2.1 \times 10^{-4})(6.0 \times 10^{-5})$$

$$= 1.3 \times 10^{-8}.$$

2.3.6 Sequences $I_V B$ and $I_V BC$

In both of these sequences, an intermediate steam break is followed by a failure to scram. While both sequences result in a core melt, they are treated as distinct sequences since the operability of the vapor suppression system can effect the magnitude of the radionuclide release by "scrubbing" some of the fission products prior to containment failure. The unavailabilities and sequence frequencies are given below.

$$Q(B) = Q(B \cap \bar{C})$$

$$= Q(B) - COM(B \cap C)$$

$$= 3.0 \times 10^{-5} - 0$$

$$= 3.0 \times 10^{-5}$$

$$Q(BC) = Q(B \cap C)$$

$$= Q(B)\ Q(C) + COM(B \cap C)$$

$$= (3.0 \times 10^{-5})(3.7 \times 10^{-4}) + 0$$

$$= 1.1 \times 10^{-8}$$

$$P(I_V B) = F(I_V)\ Q(B)$$

$$= (2.1 \times 10^{-4})(3.0 \times 10^{-5})$$

$$= 6.3 \times 10^{-9}$$

$$P(I_V BC) = F(I_V)\ Q(BC)$$

$$= (2.1 \times 10^{-4})(1.1 \times 10^{-8})$$

$$< 1.0 \times 10^{-8}.$$

3. FAILURE DATA

3.1 Component Failure Data

Each of the failure events identified in the various fault trees and described by the eight-character event-naming code was assigned failure data so that fault tree quantification based on these events could be accomplished.

In general, the recommended data base provided by NRC (Table C-4) was utilized to obtain this failure data. WASH-1400 was the major source of the tabular data. However, in some instances, the WASH-1400 data is supplemented by data found in the various LER Data Summary NUREGs. For the Browns Ferry study, the generic WASH-1400 data was applied where appropriate.

Occasionally, a failure rate that corresponded directly to a specific component failure mode could not be determined from the data in Table C-4. In these cases, other methods were used to determine an acceptable failure rate for the component in question. Table C-5 lists these failure modes and the corresponding failure rates that were used in the BF1 study. Most of these additional failure modes considered could be related to a similar failure mode category in the WASH-1400 data, with three exceptions:

1. Rupture disk leakage/rupture failure rates were estimated by using plant-specific data supplied by TVA.

2. No data source was available for the probability of heat exchanger or strainer plugging; an estimate of 1.0×10^{-6} per hour was used for these modes.

3. Since many of the motor-operated valves (MOV) and pump control circuits were similar in design, generic probability values were derived for output failure of typical MOV and pump motor control circuits. These values varied depending on whether the circuit was tested or demanded on a monthly or quarterly basis. The auto-initiation logic placing the "demand" on the control circuit was explicitly modeled in every case. The analysis of the generic control circuits can be found in Section 5 of Appendix B.

The repair times for components was taken from Table III 5-2 of WASH-1400, Summary of Major Maintenance Act Duration, for pumps, valves, diesels, and instrumentation. Electrical components (other than diesels) were assumed to have the same repair rate as that shown for instrumentation (7 hours).

3.2 Human Error Rates

Human errors of omission were included where appropriate in the fault tree models for errors involving test and maintenance, and those involving errors in response to an accident situation. Surveillance and maintenance instructions were reviewed to identify potential human errors during testing or maintenance and are discussed in Appendix B on a system-by-system basis. Emergency operating instructions were reviewed with regard to potential

TABLE C-4. IREP DATA TABLE 3A AND 3B

Mechanical Components
(from WASH-1400, Table III 4-1)

Component and Failure Mode	Failure Rate Type	Assessed Range		Median	Error Factor
Pumps (includes driver)					
Motor and turbine driven (generic class)					
Failure to start on demand	D[a]	3E-4	3E-3	1E-3	3
Failure to run, given start (normal environments)	O	3E-6	3E-4	3E-5	10
Failure to run, given start (extreme, post accident environments inside containment)	O	1E-4	1E-2	1E-3	10
Failure to run, given start (postaccident, after environmental recovery)	O	3E-5	3E-3	3E-4	10
Turbine driven pumps					
Failure to start on demand (failure rates shown are in addition to WASH-1400 values)	D	1E-3	1E-2	3E-3	3
Failure to run, given start (failure rates shown are in addition to WASH-1400 values)	O	1E-5	1E-4	3E-5	3
Valves					
Motor operated					
Failure to operate (includes driver)	D[b]	3E-4	3E-3	1E-3	3
Failure to remain open (plug)	D[c]	3E-5	3E-4	1E-4	3
Failure to remain open (plug)	S	1E-7	1E-6	3E-7	3
Rupture	S	1E-9	1E-7	1E-8	10
Solenoid operated					
Failure to operate	D[b]	3E-4	3E-3	1E-3	3
Failure to remain open (plug)	D	3E-5	3E-4	1E-4	3
Rupture	S	1E-9	1E-7	1E-8	10

TABLE C-4. (continued)

Mechanical Components (from WASH-1400, Table III 4-1)					
Component and Failure Mode	Failure Rate Type	Assessed Range		Median	Error Factor
Valves (continued)					
Air-fluid operated					
Failure to operate	D[b]	1E-4	1E-3	3E-4	3
Failure to remain open (plug)	D	3E-5	3E-4	1E-4	3
Failure to remain open (plug)	S	1E-7	1E-6	3E-7	3
Rupture	S	1E-9	1E-7	1E-8	10
Check valves					
Failure to open	D	3E-5	3E-4	1E-4	3
Internal leak (severe)	D	1E-7	1E-6	3E-7	3
Rupture	S	1E-9	1E-7	1E-8	10
Vacuum valve					
Failure to operate	D	1E-5	1E-4	3E-5	3
Manual valve					
Failure to operate (failure rates shown are in addition to WASH-1400 values)	D	3E-5	3E-4	1E-4	3
Failure to remain open (plug)	D	3E-5	3E-4	1E-4	3
Rupture	S	1E-9	1E-7	1E-8	10
Primary safety valves (PWR)					
Fail to open (failure rates shown are a revision of WASH-1400 values)	D	1E-3	1E-2	3E-3	3

TABLE C-4. (continued)

Mechanical Components (from WASH-1400, Table III 4-1)					
Component and Failure Mode	Failure Rate Type	Assessed Range		Median	Error Factor
Valves (continued)					
Primary safety valves (PWR) (continued)					
Premature open (failure rates shown are a revision of WASH-1400 values)	S	1E-6	1E-5	3E-6	3
Failure to reclose (given valve opened) (failure rates shown are a revision of WASH-1400 values)	D[d]	3E-3	3E-2	1E-2	3
Primary safety valves (BWR)					
Fail to open (failure rates shown are a revision of WASH-1400 values)	D	3E-3	3E-2	1E-2	3
Premature open (failure rates shown are a revision of WASH-1400 values)	S	1E-6	1E-5	3E-6	3
Fail to reclose (given valve opened) (failure rates shown are a revision of WASH-1400 values)	D	1E-3	1E-2	3E-3	3
Test valves, flow meters, orifices					
Failure to remain open (plug)	D	1E-4	1E-3	3E-4	3
Rupture	S	1E-9	1E-7	1E-8	10
Pipes					
Pipes $\leq$3-in. diameter (per section)					
Rupture/plug	S + D	3E-11	3E-8	1E-9	30
Pipe >3-in. diameter (per section)					
Rupture/plug	S + D	3E-12	3E-9	1E-10	30

TABLE C-4. (continued)

Component and Failure Mode	Failure Rate Type	Assessed Range		Median	Error Factor
Mechanical Components (from WASH-1400, Table III 4-1)					
Clutch, mechanical					
Failure to operate	D[b]	1E-4	1E-3	3E-4	3
Scram rods (single)					
Failure to insert	D	3E-5	3E-4	1E-4	3
Electrical Components (from WASH-1400, Table III 4-2)					
Clutch, electrical					
Failure to operate	D[a]	1E-4	1E-3	3E-4	3
Premature disengagement	O	1E-7	1E-5	1E-6	10
Motors, electric					
Failure to start	D[a]	1E-4	1E-3	3E-4	3
Failure to run, given start (normal environment)	O	3E-6	3E-5	1E-5	3
Failure to run, given start (extreme environment)	O	1E-4	1E-2	1E-3	10
Relays					
Failure to energize	D[a]	3E-5	3E-4	1E-4	3
Failure of NO contacts to close, given energized	O	1E-7	1E-6	3E-7	3
Failure of NC contacts by opening, given not energized	O	3E-8	3E-7	1E-7	3

TABLE C-4. (continued)

Electrical Components (from WASH-1400, Table III 4-2)					
Component and Failure Mode	Failure Rate Type	Assessed Range		Median	Error Factor
Short across NO/NC contact	O	1E-9	1E-7	1E-8	10
Coil open	O	1E-8	1E-6	1E-7	10
Coil short to power	O	1E-9	1E-7	1E-8	10
Circuit breakers					
Failure to transfer	D[a]	3E-4	3E-3	1E-3	3
Premature transfer	O	3E-7	3E-6	1E-6	3
Switches					
Limit					
Failure to operate	D	1E-4	1E-3	3E-4	3
Torque					
Failure to operate	D	3E-5	3E-4	1E-4	3
Pressure					
Failure to operate	D	3E-5	3E-4	1E-4	3
Manual					
Failure to transfer	D	3E-6	3E-5	1E-5	3

TABLE C-4. (continued)

Electrical Components
(from WASH-1400, Table III 4-2)

Component and Failure Mode	Failure Rate Type	Assessed Range		Median	Error Factor
Switch contacts					
Failure of NO contacts to close given switch operation	D	1E-8	1E-6	1E-7	10
Failure of NC contacts by opening, given no switch operation	D	3E-9	3E-7	3E-8	10
Short across NO/NC contact	D	1E-9	1E-7	1E-8	10
Battery power system (wet cell)					
Failure to provide proper output	S	1E-6	1E-5	3E-6	3
Transformers					
Open circuit primary or secondary	O	3E-7	3E-6	1E-6	3
Short primary to secondary	O	3E-7	3E-6	1E-6	3
Solid state devices, high power applications (diodes, transitors, etc.)					
Fails to function	O	3E-7	3E-5	3E-6	10
Fails shorted	O	1E-7	1E-5	1E-6	10
Solid state devices, low power applications					
Fails to function	O	1E-7	1E-5	1E-6	10
Fails shorted	O	1E-8	1E-6	1E-7	10

TABLE C-4. (continued)

Electrical Components (from WASH-1400, Table III 4-2)					
Component and Failure Mode	Failure Rate Type	Assessed Range		Median	Error Factor
Diesels (complete plant)					
Failure to start	D	1E-2	1E-1	3E-2	3
Failure to run, emergency conditions, given start	O	3E-4	3E-2	3E-3	10
Diesels (engine only)					
Failure to run, emergency conditions, given start	O	3E-5	3E-3	3E-4	10
Instrumentation--general (includes transmitter, amplifier, and output devices)					
Failure to operate	O	1E-7	1E-5	1E-6	10
Shift in calibration	O	3E-6	3E-4	3E-5	10
Fuses					
Failure to open	D	3E-6	3E-5	1E-5	3
Premature open	O	3E-7	3E-6	1E-6	3
Wires (typical circuits, several joints)					
Open circuit	O	1E-6	1E-5	3E-6	3
Short to ground	O	3E-8	3E-6	3E-7	10
Short to power	O	1E-9	1E-7	1E-8	10

TABLE C-4. (continued)

Terminal boards					
Open connection	O	1E-8	1E-6	1E-7	10
Short to adjacent circuit	O	1E-9	1E-7	1E-8	10

a. Demand probabilities are based on the presence of proper input control signals. For turbine driven pumps, the effect of failures of valves, sensors, and other auxiliary hardware may result in significantly higher overall failure rates for turbine driven pump systems.

b. Demand probabilities are based on presence of proper input control signals.

c. Plug probabilities are given in demand probability, and per hour rates, since phenomena are generally time dependent; but plugged condition may only be detected upon a demand of the system.

d. These rates are based on LERs for Babcock & Wilcox pressurizer PORV failure to reseat, given the valve has opened.

Abbreviations:

D = Demand failure rate (failures per demand)

O = Operating failure rate (failures per hour of operation)

S = Standby failure rate (failures per hour of standby)

S + D = Standby or operating failure rate (failures per hour).

TABLE C-5. COMPONENT DATA NOT AVAILABLE IN TABLE C-4

Component	Failure Mode	Unavailability Calculation	Remarks
Stop check valve	Does not open	1×10^{-4}	Used check valve rate
Governor control valve	Does not operate	3×10^{-4}	Used data for air/ fluid operated valves
Rupture disks	Leakage/ rupture	2.1×10^{-2}	$\lambda = 5.7 \times 10^{-5}$/hr (based on information from TVA)
Time-delay relays	Premature close	1×10^{-4}	Used relay failure-to-energize rate
Heat exchanger	Plugged	1×10^{-6}	Engineering judgement
Strainer	Plugged	1×10^{-6}	Engineering judgement
MOV control circuit	No output	3.2×10^{-3} (8.8×10^{-3})	Generic rate based on monthly testing (quarterly)
Pump control circuit	No output	2.9×10^{-3} (8.4×10^{-3})	Generic rate based on monthly testing (quarterly)

accident scenarios to determine the required human interactions with mitigating systems in response to the accidents. Section 4.2 of Appendix A describes in more detail these operator response errors.

Initial screening guidelines suggested that human error events in the models be assigned a probability value of 0.1. This proved to be too conservative and tended to mask significant hardware contributions to system unavailability. Thus, initial screening values were refined on a case-by-case basis using engineering judgement.

For those systems where the reduced human error rates still made a significant contribution to the probability of failure, an explicit human error model was developed based on the procedures found in the Sandia publication, NUREG/CR-1278.[6] It was especially important to create these models for human error events that affected multiple systems. For example, miscalibration of reactor vessel level switches could result in failure of the core standby cooling systems to be auto-initiated when required. These human error models can be found in Section 4 of Appendix B.

3.3 Recovery Factors

For the candidate dominant accident sequences, the potential for recovery was considered in the final sequence frequency. To determine recoverability, the dominant contributors to the sequence frequency were examined to determine answers to several questions:

1. Are the failure modes of the dominant contributors ones that allow for recovery? For example, an initiation fault may be recoverable by having the operator manually starting a system/component, whereas a mechanical failure of a valve may not be recoverable.

2. How much time is available to take the recovery action?

3. What must be done to repair the fault, and where must the action be taken? The only faults considered recoverable when the time available was less than 2 hours were those where simple action by the operator, such as throwing a switch or pushing a button in the control room, would correct the fault. Local faults recoverable from outside the control room where the recovery time available was more than 2 hours were also considered.

Recoverable faults were requantified by multiplying the fault unavailability by the probability of nonrecovery factor. Table C-6 summarizes these factors.

TABLE C-6. NONRECOVERY FACTORS

Time Available to Recover	Probability of Nonrecovery Factor
Less than 5 min	1.00
5 to 10 min	0.25
10 to 20 min	0.10
20 to 30 min	0.05
30 to 60 min	0.03
More than 60 min	0.01
More than 2 hr (outside control room)	0.01

4. CANDIDATE DOMINANT SEQUENCES

4.1 Introduction

Figures C-3 through C-13 are the systemic event trees. As an initial screening tool, only sequences with frequencies greater than 1.0×10^{-8} were considered significant. From the systemic event trees listed above, Table C-7 lists by initiating event those sequence frequencies greater than 10^{-8}. Table C-8 lists these systemic event tree sequence frequencies in decreasing order of magnitude. Of these sequences, those with frequencies greater than 1.0×10^{-6} were chosen as the candidate dominant sequences. Table C-9 lists these sequences along with the sequence initiator, the systemic event tree sequence designator, the initial sequence frequency, and the final sequence frequency after recovery has been considered.

In the following section, the quantification for each candidate dominant sequence is discussed. Each candidate sequence is identified by a letter designator representing the initiator (see Table C-10) and a group of letters corresponding to the systems that fail for the sequence. The sequences are also described by a written description that includes the initiator and the system(s) that must fail to cause the sequence to occur. Each candidate sequence is discussed in terms of what happens, what its initial frequency is, what the dominant contributors are, and what, if any, recovery actions are possible. Where availability of data permits, the sequence frequencies were refined to take into account recovery actions.

For clarification, three tables are provided to assist the reader in understanding how the values for the sequence frequencies were determined. Table C-11 lists the various front-line systems and their corresponding designators and unavailabilities for various accident conditions. Appendix B gives the dominant cut sets for each system.

Table C-12 lists the unavailabilities for important combinations of systems. The table lists the independent, commonality, and net unavailabilities for these combinations. Appendix B contains the dominant cut sets for each system but does not show implicitly the source of commonalities between systems. Table C-13 lists the system combinations of Table C-12 that have significant commonalities and briefly describes the major contributions to each.

4.2 Sequence Evaluation

There are 11 candidate dominant accident sequences. Six of these sequences involve failure of the DHR function to remove decay heat, three involve failure to inject water, and two involve failure to scram. All of the candidate dominant sequences involve transient initiators.

4.2.1 Loss of Offsite Power with DHR Failure ($T_P R_B R_A$)

For this sequence, the LOSP transient results in a reactor scram and the reactor vessel is isolated from the steam system by the main steam isolation valves (MSIVs). The primary relief valves lift to relieve reactor vessel pressure and reclose when pressure falls below the valve setpoint.

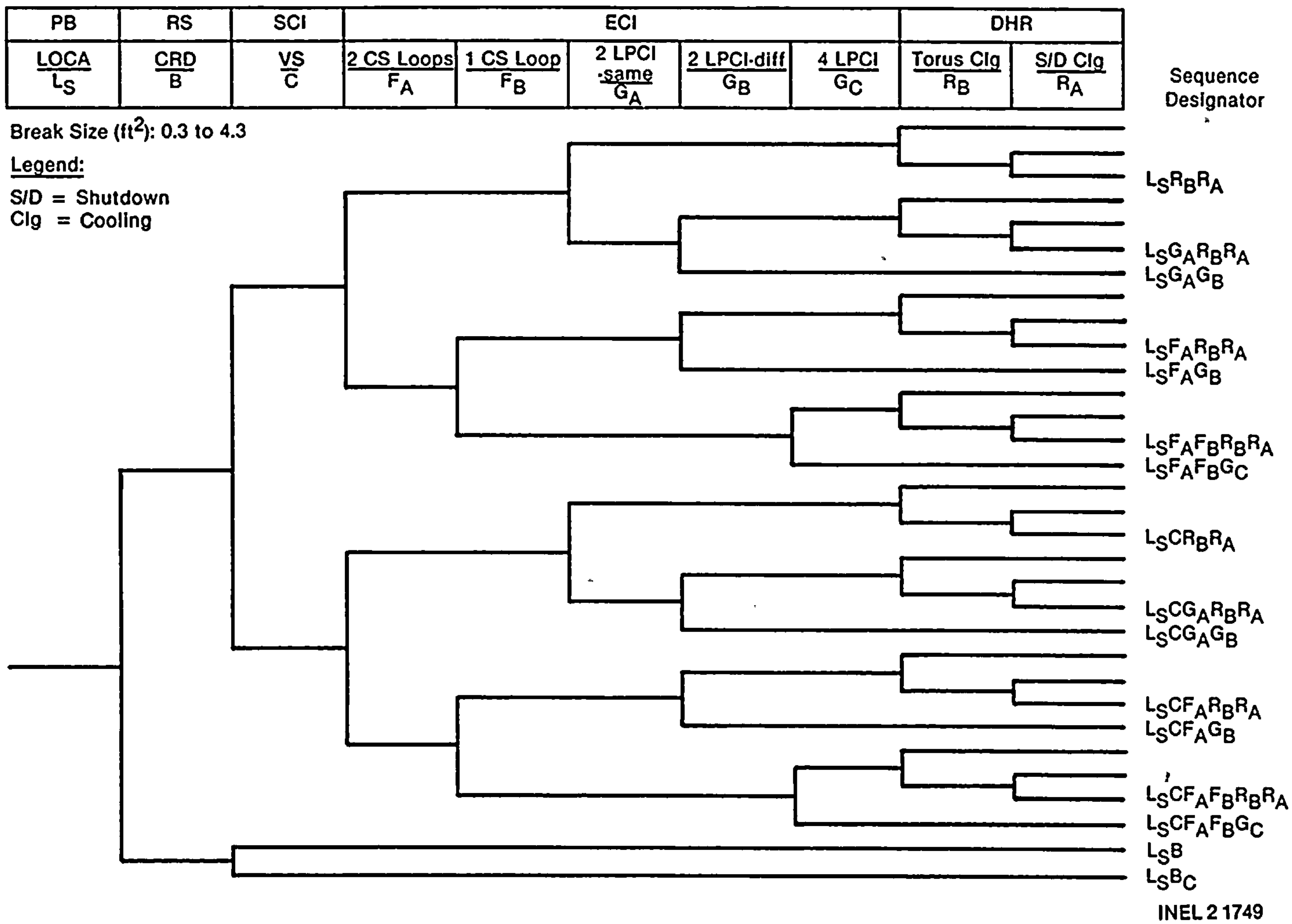


Figure C-3. LOCA systemic event tree for large liquid break, suction-side of recirculation pumps (L_S).

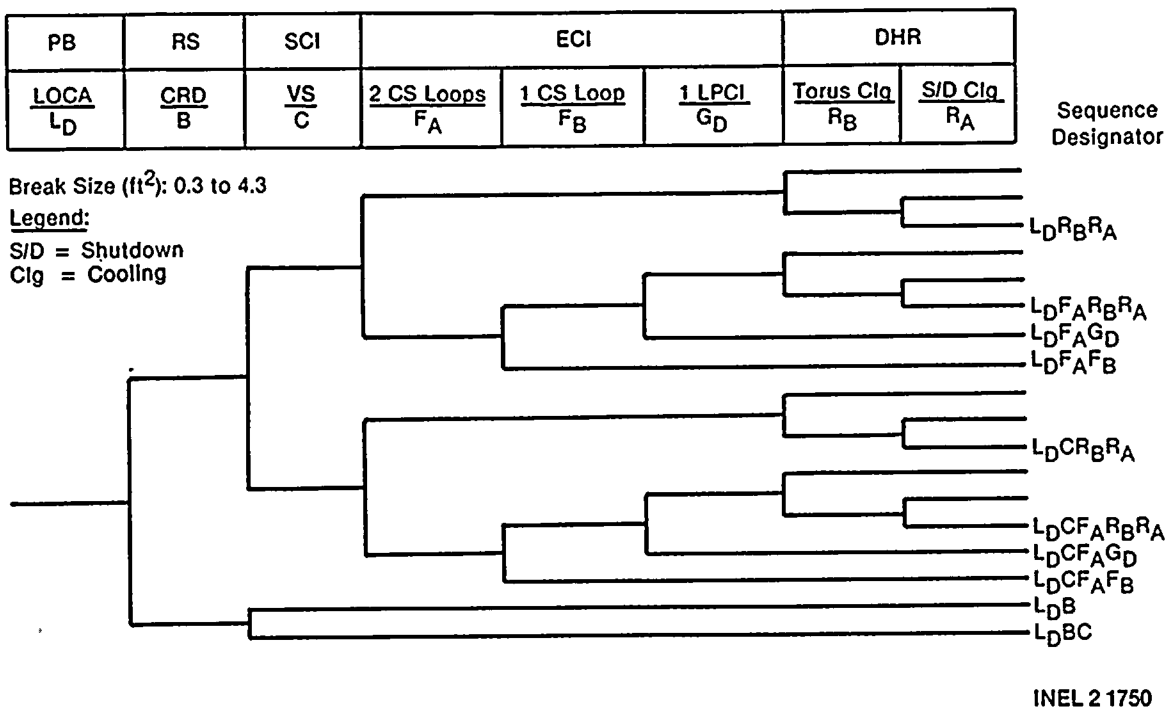


Figure C-4. LOCA systemic event tree for large liquid break, discharge-side of recirculation pumps (L_D).

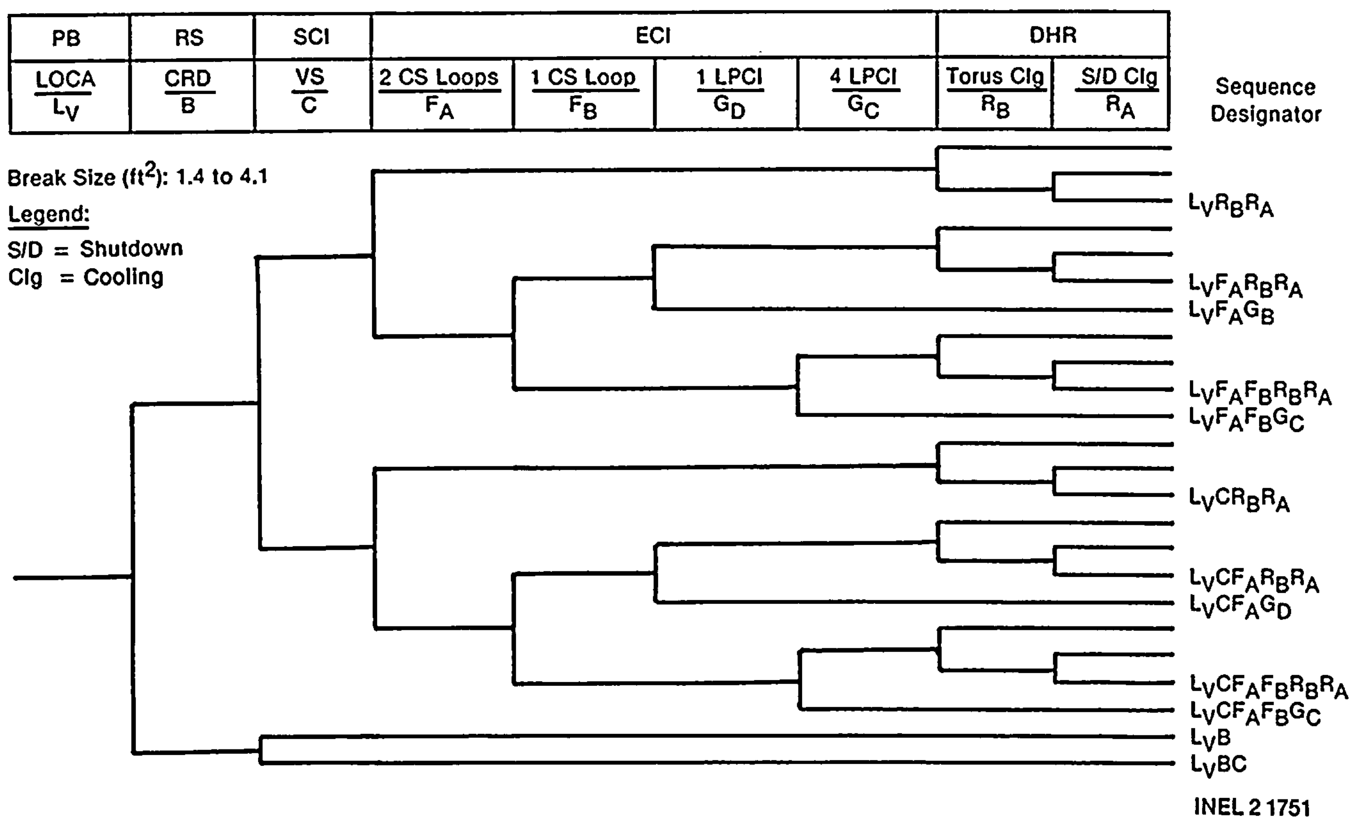


Figure C-5. LOCA systemic event tree for large steam break (L_V).

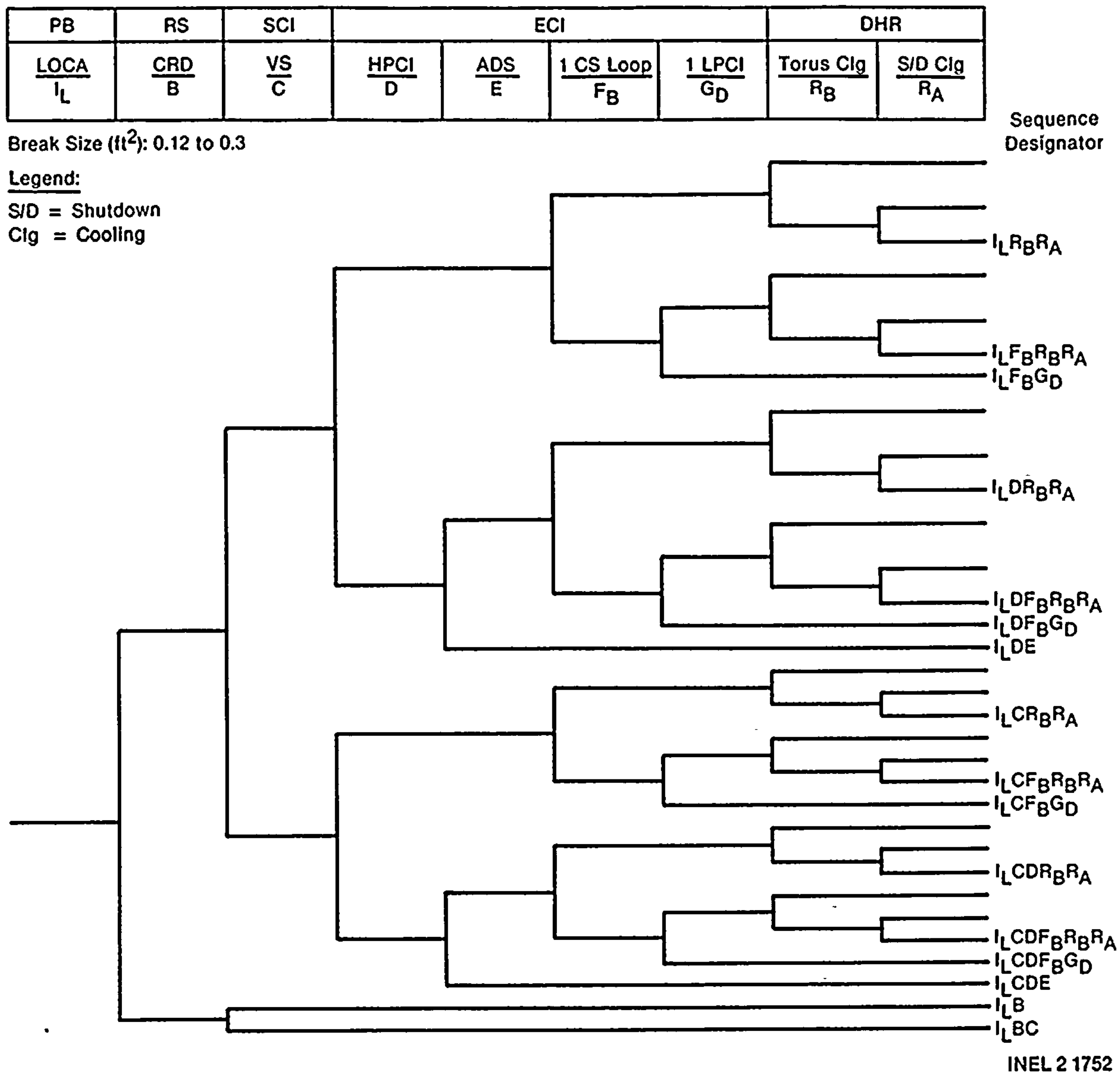


Figure C-6. LOCA systemic event tree for intermediate liquid break (I_L).

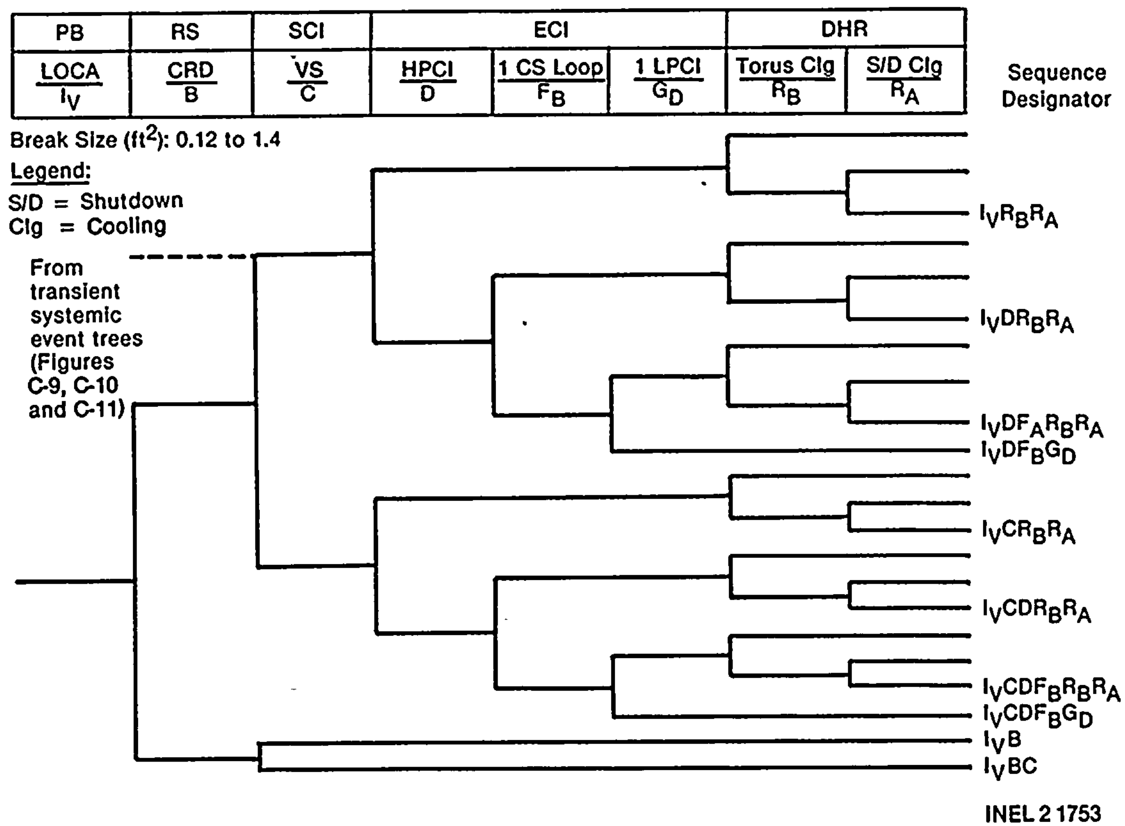


Figure C-7. LOCA systemic event tree for intermediate steam break (I_V).

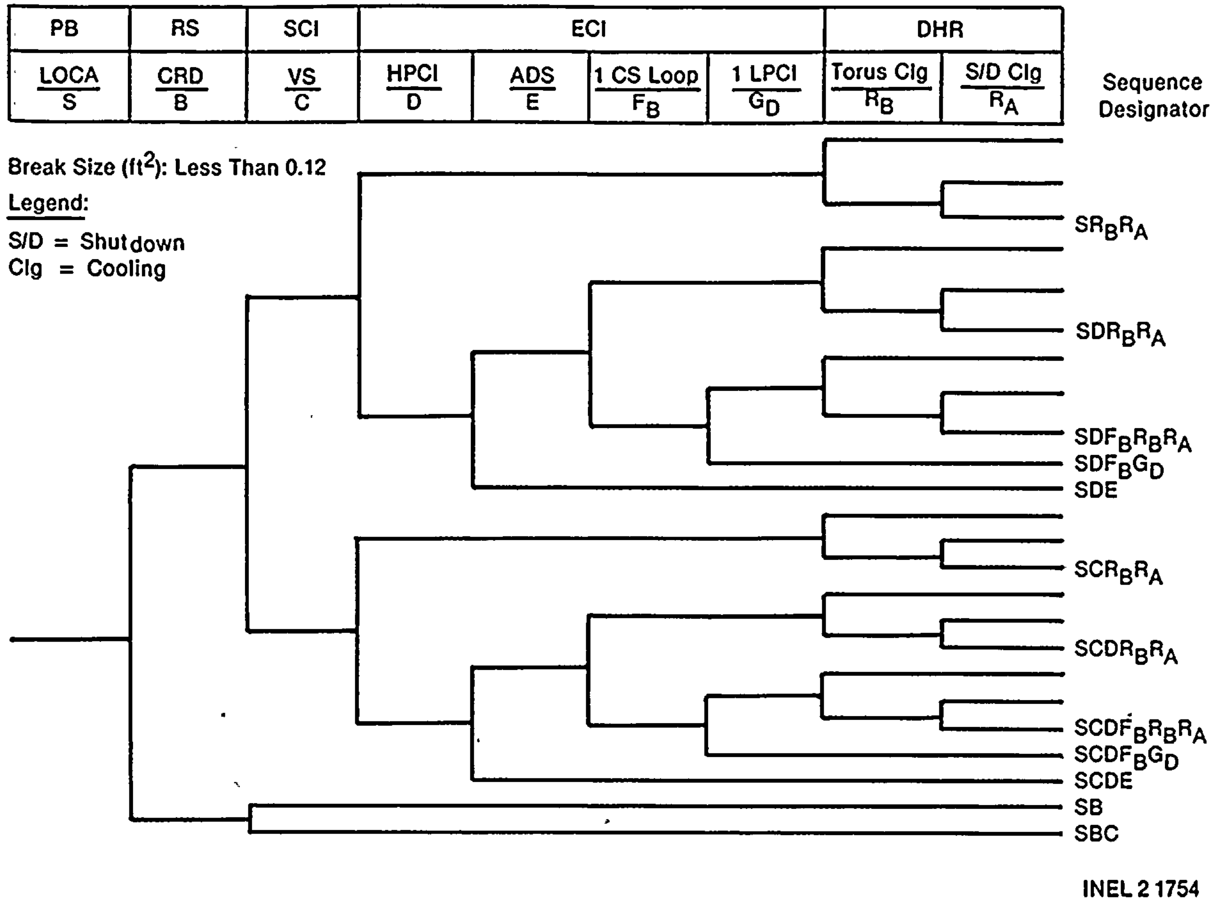


Figure C-8. LOCA systemic event tree for small liquid or steam break (S).

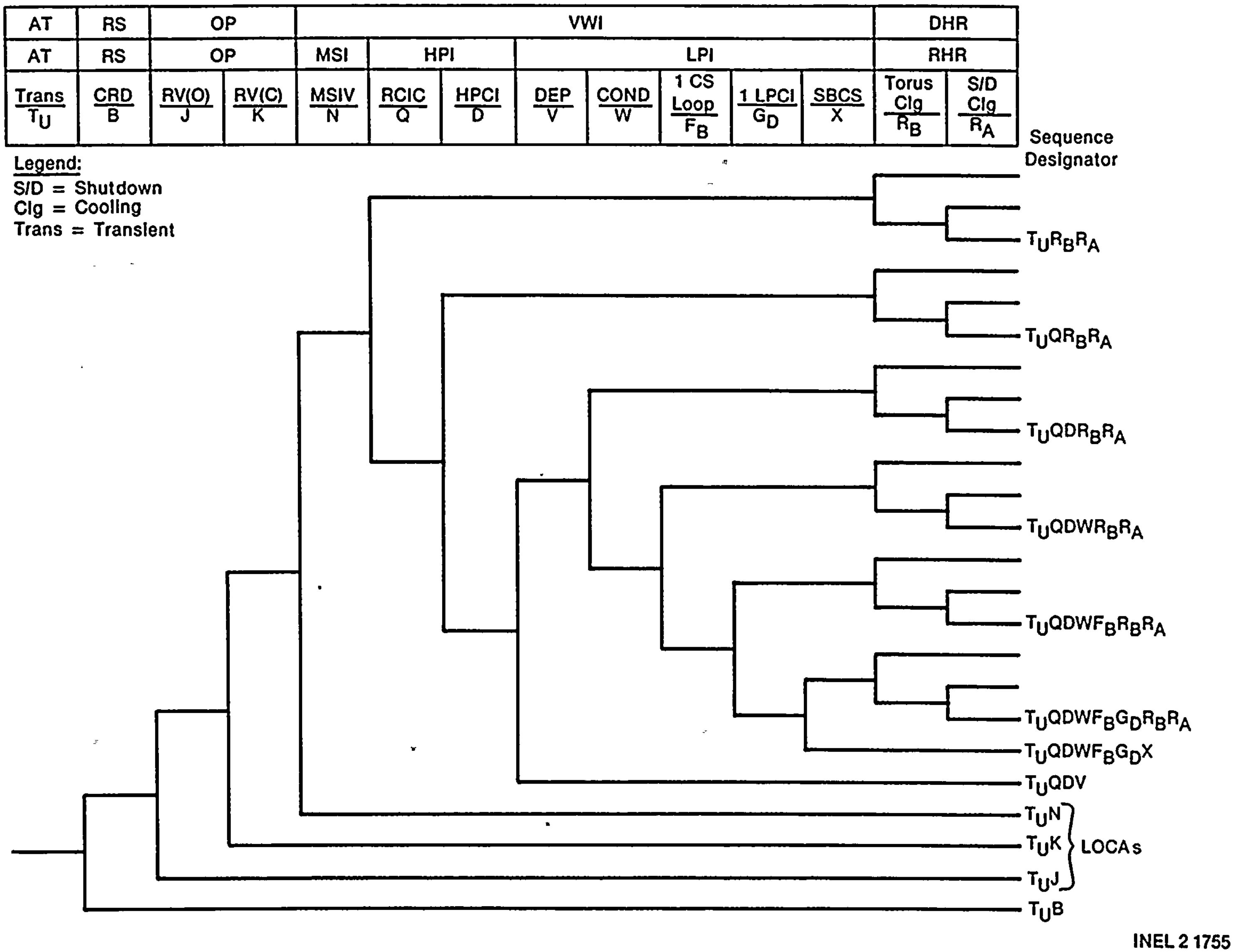


Figure C-9. Transient systemic event tree where PCS is unavailable (T_U).

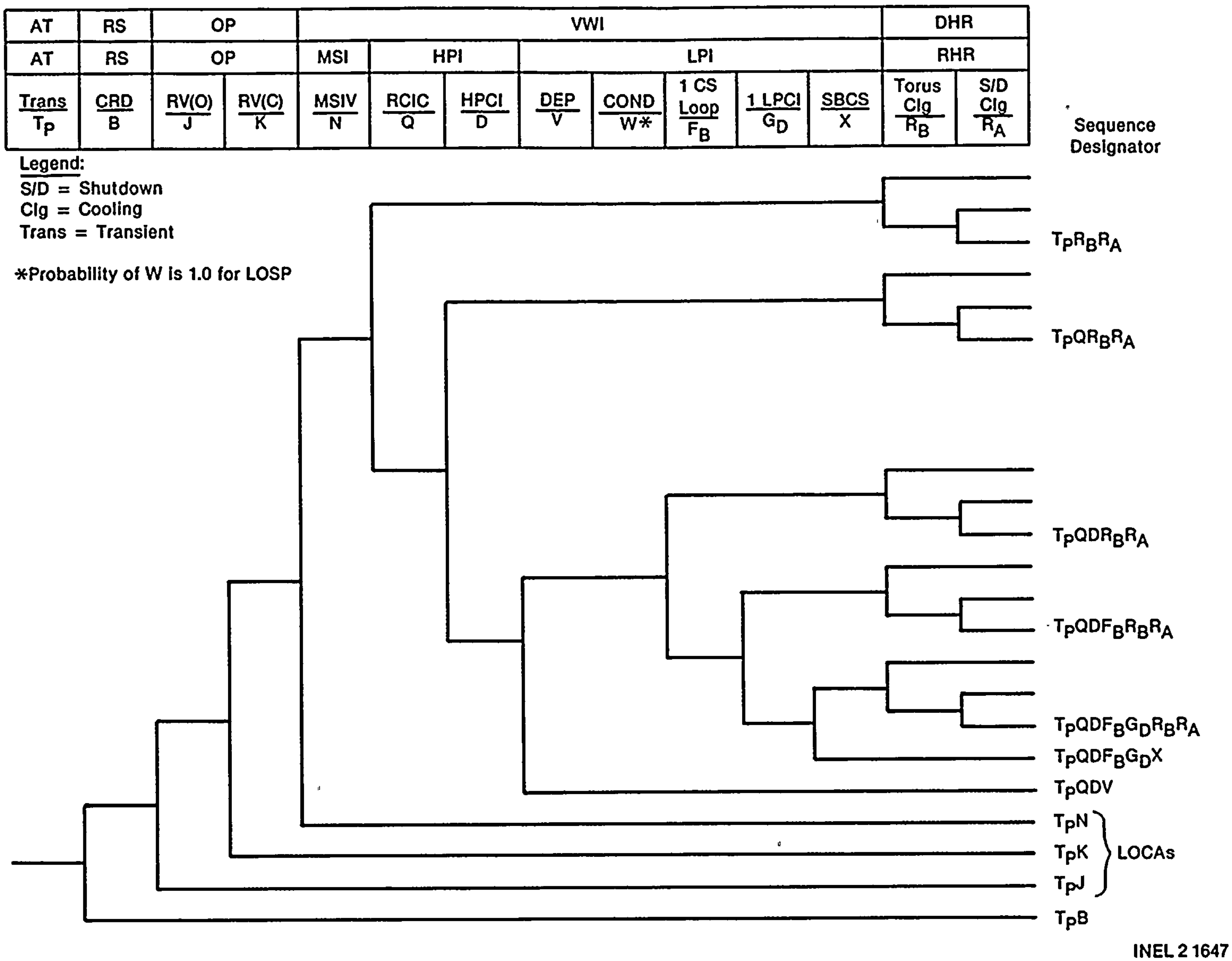


Figure C-10. LOSP-induced transient systemic event tree (PCS unavailable) (T_P).

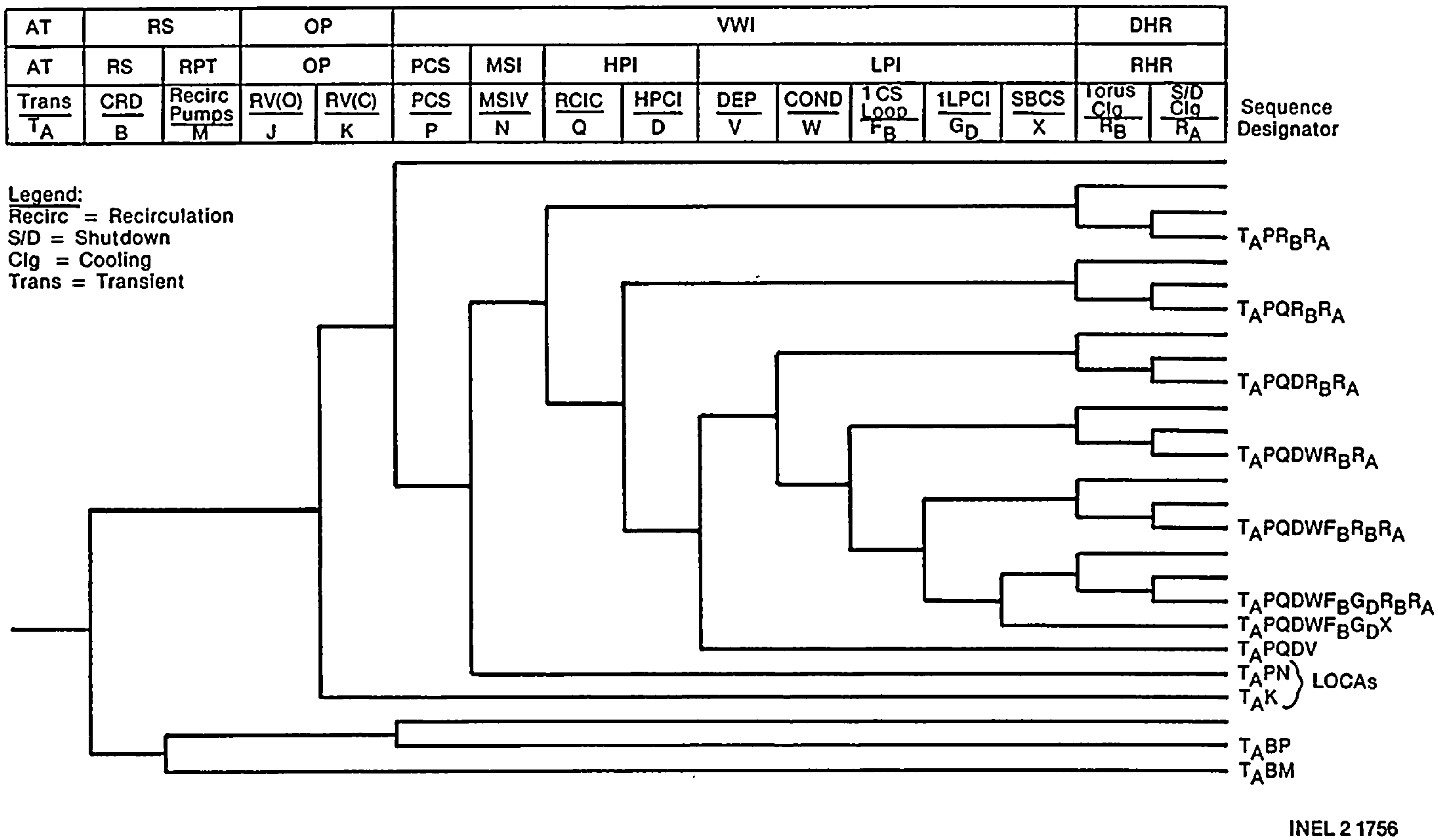


Figure C-11. Transient systemic event tree where PCS is available (T_A).

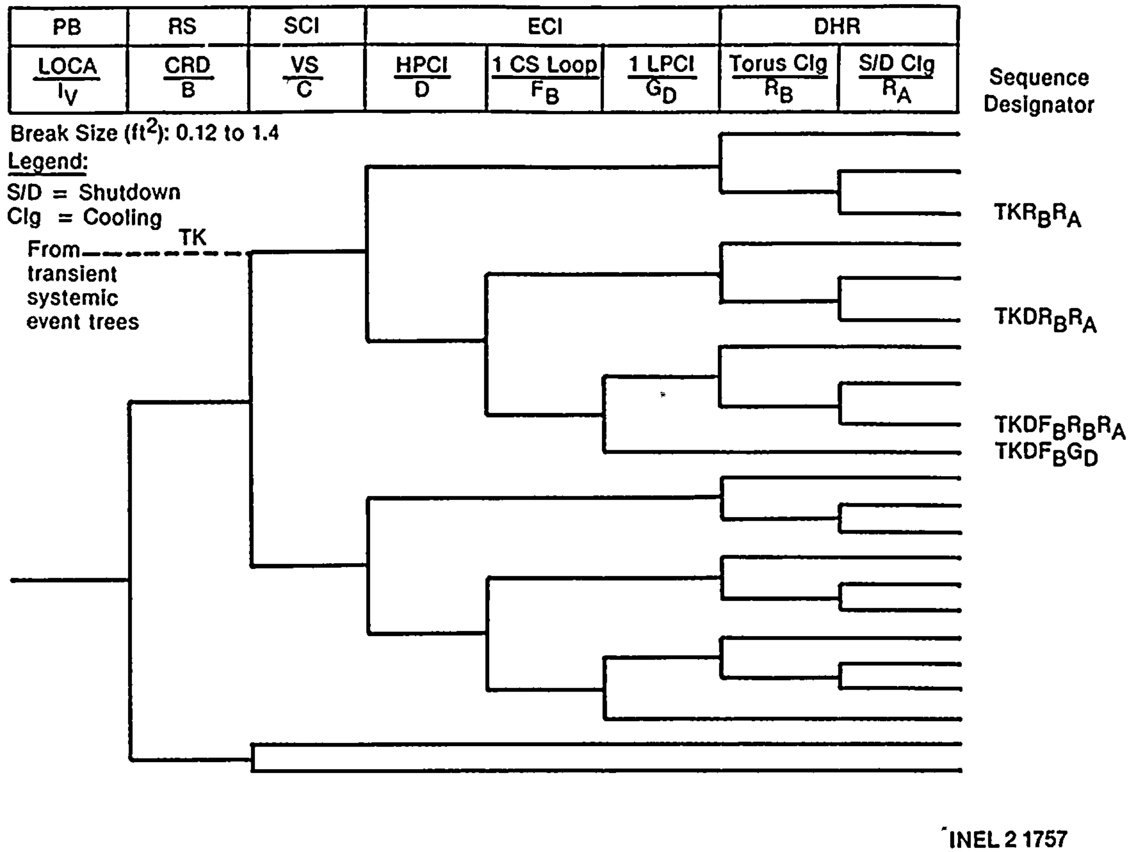


Figure C-12. Transient-induced SORV LOCA systemic event tree (intermediate steam break) (TK).

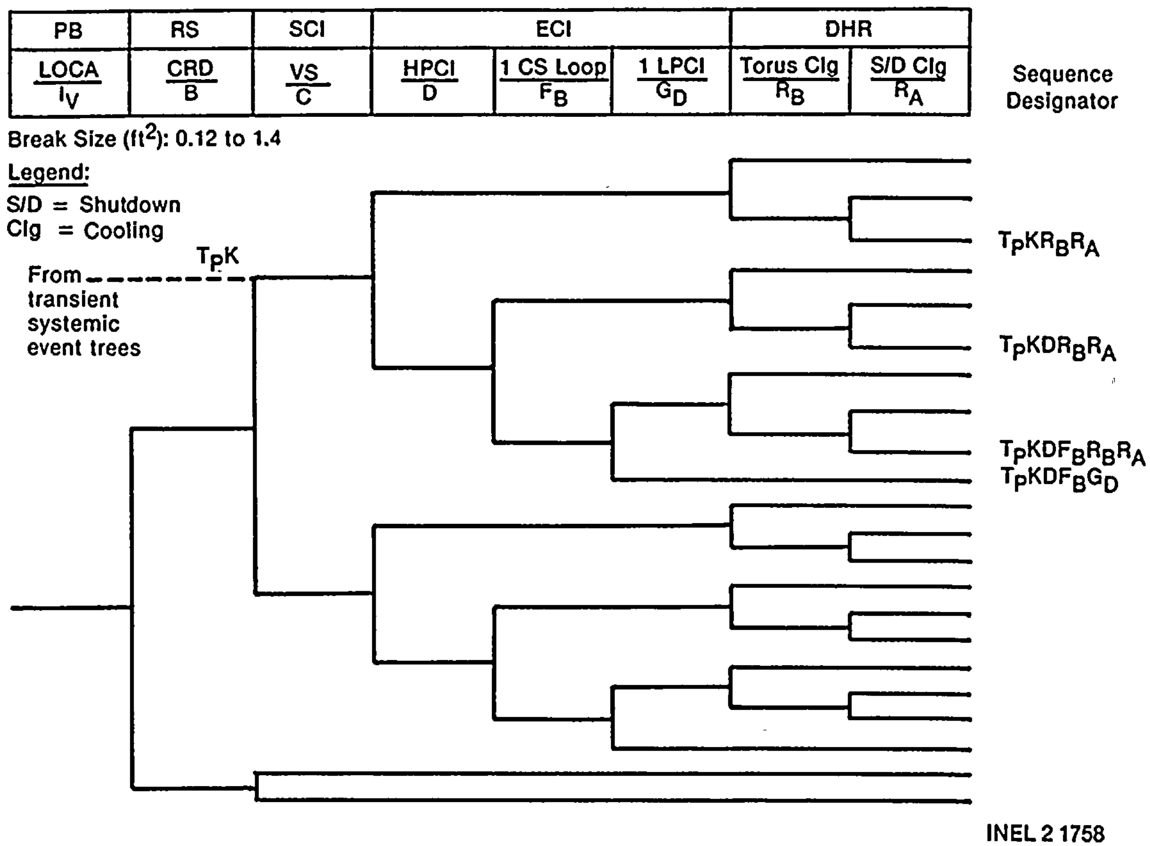


Figure C-13. LOSP-induced SORV LOCA systemic event tree (intermediate steam break) (TpK).

TABLE C-7. SEQUENCE FREQUENCIES GREATER THAN 10^{-8} BY INITIATOR

Sequence Designator	Sequence Frequency	Sequence Designator	Sequence Frequency
$L_SR_BR_A$	1.7×10^{-8}	$T_UR_BR_A$	1.3×10^{-4}
$L_SF_AG_B$	1.1×10^{-8}	$T_UQR_BR_A$	5.5×10^{-6}
		$T_UQDR_BR_A$	2.4×10^{-7}
$L_DR_BR_A$	6.3×10^{-8}	$T_UQDWF_BG_DX$	4.1×10^{-8}
$L_DF_AG_D$	2.0×10^{-8}	T_UQDV	9.2×10^{-6}
$L_DF_AF_B$	2.6×10^{-8}	T_UB	5.1×10^{-5}
$I_LR_BR_A$	1.4×10^{-7}	$T_PR_BR_A$	1.5×10^{-3}
		$T_PQR_BR_A$	6.2×10^{-5}
$I_VR_BR_A$	1.6×10^{-8}	$T_PQDR_BR_A$	1.7×10^{-8}
$I_VCR_BR_A$	1.3×10^{-8}	$T_PQDF_BG_DX$	1.2×10^{-6}
$I_VCDR_BR_A$	1.3×10^{-8}	T_PQDV	1.7×10^{-7}
$I_VCDF_BR_BR_A$	1.3×10^{-8}	T_PB	9.0×10^{-7}
$I_VCDF_BG_D$	1.3×10^{-8}		
		$T_APR_BR_A$	8.9×10^{-7}
SR_BR_A	5.3×10^{-7}	$T_APQR_BR_A$	3.7×10^{-8}
SDR_BR_A	1.2×10^{-7}	$T_APQDWF_BG_DX$	2.8×10^{-8}
SCR_BR_A	6.0×10^{-8}	T_APQDV	6.3×10^{-8}
$SCDR_BR_A$	6.0×10^{-8}	T_ABP	3.5×10^{-7}
$SCDF_BR_BR_A$	6.0×10^{-8}	T_ABM	3.7×10^{-6}
$SCDF_BG_D$	6.0×10^{-8}		
SCDE	6.0×10^{-8}	TKR_BR_A	1.2×10^{-5}
SB	3.0×10^{-8}	$TKDR_BR_A$	7.8×10^{-7}
		$TKDF_BG_D$	3.9×10^{-7}
		$T_PKR_BR_A$	8.3×10^{-5}
		$T_PKDR_BR_A$	3.3×10^{-8}
		$T_PKDF_BG_D$	2.5×10^{-6}

TABLE C-8. SYSTEMIC SEQUENCE FREQUENCIES IN DRCREASING ORDER OF MAGNITUDE

Sequence Designator	Sequence Frequency	Sequence Designator	Sequence Frequency
$T_PR_BR_A$	1.5×10^{-3}	$L_DR_BR_A$	6.3×10^{-8}
$T_UR_BR_A$	1.3×10^{-4}	SCR_BR_A	6.0×10^{-8}
$T_PKR_BR_A$	8.3×10^{-5}	$SCDR_BR_A$	6.0×10^{-8}
$T_PQR_BR_A$	6.2×10^{-5}	$SCDF_BR_BR_A$	6.0×10^{-8}
T_UB	5.1×10^{-5}	$SCDF_BG_D$	6.0×10^{-8}
TKR_BR_A	1.2×10^{-5}	SCDE	6.0×10^{-8}
T_UQDV	9.2×10^{-6}	$T_UQDWF_BG_DX$	4.1×10^{-8}
$T_UQR_BR_A$	5.5×10^{-6}	$T_APQR_BR_A$	3.7×10^{-8}
T_ABM	3.7×10^{-6}	$T_PKDR_BR_A$	3.3×10^{-8}
$T_PKDF_BG_D$	2.5×10^{-6}	SB	3.0×10^{-8}
$T_PQDF_BG_DX$	1.2×10^{-6}	$T_APQDWF_BG_DX$	2.8×10^{-8}
T_PB	9.0×10^{-7}	$L_DF_AF_B$	2.6×10^{-8}
$T_APR_BR_A$	8.9×10^{-7}	$L_DF_AG_D$	2.0×10^{-8}
$TKDR_BR_A$	7.8×10^{-7}	$T_PQDR_BR_A$	1.7×10^{-8}
SR_BR_A	5.3×10^{-7}	$L_SR_BR_A$	1.7×10^{-8}
$TKDF_BG_D$	3.9×10^{-7}	$I_VR_BR_A$	1.6×10^{-8}
T_ABP	3.5×10^{-7}	$I_VCR_BR_A$	1.3×10^{-8}
$T_UQDR_BR_A$	2.4×10^{-7}	$I_VCDR_BR_A$	1.3×10^{-8}
T_PQDV	1.7×10^{-7}	$I_VCDF_BR_BR_A$	1.3×10^{-8}
$I_LR_BR_A$	1.4×10^{-7}	$I_VCDF_BG_D$	1.3×10^{-8}
SDR_BR_A	1.2×10^{-7}	$L_SF_AG_B$	1.1×10^{-8}
T_APQDV	6.3×10^{-8}		

TABLE C-9. CANDIDATE DOMINANT SEQUENCES

Sequence Initiator	Sequence Designator	Sequence Frequence Initial	Sequence Frequence Final
Transient-induced LOCAs	TKR_BR_A	1.2×10^{-5}	9.3×10^{-6}
LOSP-induced LOCAs	$T_PKR_BR_A$	8.3×10^{-5}	1.6×10^{-6}
	$T_PKDF_BG_D$	2.5×10^{-6}	8.7×10^{-8}
PCS unavailable	$T_UR_BR_A$	1.3×10^{-4}	9.7×10^{-5}
	$T_UQR_BR_A$	5.5×10^{-6}	4.1×10^{-6}
	T_UB	5.1×10^{-5}	5.1×10^{-5}
	T_UQDV	9.2×10^{-6}	5.5×10^{-7}
PCS available	T_ABM	3.7×10^{-6}	3.7×10^{-6}
LOSP	$T_PR_BR_A$	1.5×10^{-3}	2.8×10^{-5}
	$T_PQR_BR_A$	6.2×10^{-5}	1.2×10^{-6}
	$T_PQDF_BG_DX$	1.2×10^{-6}	3.6×10^{-8}

TABLE C-10. INITIATOR DESIGNATORS

Designator	Initiator	Frequency (per reactor-year)
L_S	Large suction break	9.9×10^{-6}
L_D	Large discharge break	3.9×10^{-5}
L_V	Large steam break	5.2×10^{-5}
I_L	Intermediate liquid break	9.0×10^{-5}
I_V	Intermediate steam break	2.1×10^{-4}
S	Small liquid or steam break	1.0×10^{-3}
T_U	Transients where PCS is unavailable	1.70
T_P	Loss of offsite power transient	3.0×10^{-2}
T_A	Transients where PCS is available	1.68
TK	Transient induced SORV	1.63×10^{-1}*
TpK	Loss of offsite power-induced SORV	1.7×10^{-3}*

* Two additional initiators are defined in this table. In each case they represent transient-induced SORVs. The designator TK is used to represent the combined frequency for a SORV from both the PCS available and PCS unavailable transient event trees. The designator K represents a system that is described for both of these cases in Table C-11. TpK represents the frequency for a SORV from the PCS unavailable transient event tree for only the special case where LOSP was the initiator. Each of these initiators transfer to the intermediate steam break LOCA systemic event tree at the ECI systems branch point. The LOSP transient-induced SORV was treated independently from the PCS unavailable category due to the important dependencies of the mitigating systems on emergency onsite AC power.

TABLE C-11. FRONT-LINE SYSTEMS UNAVAILABILITIES

Designator	System	Special Conditions	Unavailability
B	Control rod drive	--	3.0×10^{-5}
C	Vapor suppression	--	3.7×10^{-4}
D	HPCI	LOCA initiator	6.5×10^{-2}
		Transient initiator	4.4×10^{-2}
E	ADS	--	3.2×10^{-4}
F_A	Core spray (two core spray loops)	Normal power	5.2×10^{-2}

TABLE C-11. (continued)

Designator	System	Special Conditions	Unavailability
F_B	Core spray (one core spray loop)	Normal power	6.6×10^{-4}
		LOSP	9.6×10^{-4}
		Steam break on core spray pipe	2.6×10^{-2}
G_A	RHR (LPCI mode) (two LPCI pumps in same loop)	--	6.6×10^{-4}
G_B	RHR (LPCI mode) (two LPCI pumps, one in each loop)	--	2.1×10^{-2}
G_C	RHR (LPCI mode) (four LPCI pumps)	--	5.0×10^{-2}
G_D	RHR (LPCI mode) (one LPCI pump)	Normal power	1.1×10^{-4}
		LOSP	2.7×10^{-4}
		Break on recirculation discharge	1.0×10^{-2}
J	Relief valves (opening)	--	7.2×10^{-9}
K	Relief valves (closing)	Transients without PCS	5.7×10^{-2}
		Transients with PCS	3.9×10^{-2}
M	Recirculation pumps	--	8.7×10^{-3}
N	Main steam isolation valve	--	4.4×10^{-7}
P	Power conversion system	--	7.0×10^{-3}
Q	RCIC	--	4.2×10^{-2}
R_A	RHR (shutdown cooling)	Normal power	2.0×10^{-2}
		LOSP	4.2×10^{-2}
		Break on recirculation discharge	3.1×10^{-2}
R_B	RHR (torus cooling)	Normal power	3.1×10^{-3}
		LOSP	7.2×10^{-3}
V	Manual depressurization	--	3.0×10^{-3}
W	Condensate pumps	--	7×10^{-3}
X	RHR (SBCS mode)	Normal power	4.2×10^{-2}
		LOSP	4.6×10^{-2}

TABLE C-12. SYSTEM COMBINATIONS OF IMPORTANCE

System Combination	Special Conditions	Unavailability		
		Independent	Common	Net
$R_A \cap R_B$	Normal power	6.2×10^{-5}	1.4×10^{-5}	7.6×10^{-5}
	LOSP	3.0×10^{-4}	4.9×10^{-2}	4.9×10^{-2}
	Break on recirculation discharge	1.6×10^{-3}	ε	1.6×10^{-3}
$G_A \cap G_B$	--	1.4×10^{-5}	4.2×10^{-4}	4.3×10^{-4}
$F_A \cap G_B$	--	1.1×10^{-3}	4.6×10^{-6}	1.1×10^{-3}
$F_B \cap G_C$	--	3.3×10^{-5}	2.5×10^{-6}	3.6×10^{-5}
$F_A \cap G_D$	Break on recirculation discharge	5.2×10^{-4}	2.4×10^{-6}	5.2×10^{-4}
	No break	5.7×10^{-6}	1.9×10^{-8}	5.7×10^{-6}
$D \cap E$	--	2.3×10^{-6}	ε	2.3×10^{-6}
$F_B \cap G_D$	Break on recirculation discharge	6.6×10^{-6}	2.3×10^{-6}	8.9×10^{-6}
	No break	7.3×10^{-8}	3.4×10^{-8}	1.1×10^{-7}
	LOSP	1.5×10^{-3}	2.1×10^{-2}	2.2×10^{-2}
$D \cap F_B \cap G_D$	No break	7.2×10^{-9}	2.4×10^{-6}	2.4×10^{-6}
	Break on recirculation discharge	5.8×10^{-7}	ε	5.8×10^{-7}
	Break on core spray pipe	1.9×10^{-7}	ε	1.9×10^{-7}
	LOSP	1.5×10^{-3}	ε	1.5×10^{-3}
$Q \cap D$	Transients	1.8×10^{-3}	2.4×10^{-6}	1.8×10^{-3}
$Q \cap D \cap F_B \cap G_D$	Transients	2.0×10^{-10}	2.4×10^{-6}	2.4×10^{-6}
$Q \cap D \cap V$	Transients	5.4×10^{-6}	ε	5.4×10^{-6}
$Q \cap D \cap F_B \cap G_D \cap W$	Transients	1.7×10^{-8}	ε	1.7×10^{-8}
$P \cap W$	Transients	4.9×10^{-6}	7.0×10^{-3}	7.0×10^{-3}
$P \cap Q \cap D \cap V$	Transients	3.8×10^{-8}	ε	3.8×10^{-8}
$P \cap Q \cap D \cap F_B \cap G_D$	Transients	1.7×10^{-8}	ε	1.7×10^{-8}

TABLE C-12. (continued)

System Combination	Special Conditions	Unavailability		
		Independent	Common	Net
$\cap W \cap X$				
$F_B \cap G_D \cap X$	LOSP	1.5×10^{-3}	2.1×10^{-2}	2.2×10^{-2}
$Q \cap D \cap F_B \cap G_D \cap X$	LOSP	3.8×10^{-5}	2.4×10^{-6}	4.0×10^{-5}
$B \cap M$	Transients	2.6×10^{-7}	1.9×10^{-6}	2.2×10^{-6}

TABLE C-13. COMMONALITIES OF IMPORTANCE

System Combination	Special Conditions	Commonalities Unavailable	Remarks
$R_A \cap R_B$	Normal power	1.4×10^{-5}	Minimum-flow bypass valves
	LOSP	4.9×10^{-2}	Diesel generator and EECW faults
$G_A \cap G_B$	--	4.2×10^{-4}	Minimum-flow bypass valves and loop discharge valves
$F_A \cap G_B$	--	4.6×10^{-6}	Electric power faults
$F_B \cap G_C$	--	2.5×10^{-6}	Electric power faults
$F_A \cap G_D$	Break on recirculation discharge	2.4×10^{-6}	Electric power faults
	No break	1.9×10^{-8}	Electric power faults
$F_B \cap G_D$	Break on recirculation discharge	2.3×10^{-6}	Electric power faults
	No break	3.4×10^{-8}	Electric power faults

TABLE C-13. (continued)

System Combination	Special Conditions	Commonalities Unavailable	Remarks
	LOSP	2.1×10^{-2}	Primarily EECW faults
$D \cap F_B \cap G_D$ $Q \cap D$ $Q \cap D \cap F_B \cap G_D$ $Q \cap D \cap F_B \cap G_D \cap X$	Transients Transients Transients LOSP	2.4×10^{-6}	Maintenance error to level switches
$P \cap W$	Transients	7×10^{-3}	Assumed that PCS failure causes condensate pump failure
$F_B \cap G_D \cap X$	LOSP	2.1×10^{-2}	Primarily EECW faults
$B \cap M$	Transients	1.9×10^{-6}	Reactor protection system common mode failures

The operator maintains normal reactor vessel water level using RCIC, a system that will automatically initiate on low reactor vessel level. Following successful coolant injection, the torus cooling (R_B) and shutdown cooling (R_A) systems fail. A sustained loss of these systems will result in the inability to provide makeup water to the reactor to replace the inventory lost due to boil off caused by decay heat. A core melt will eventually occur. The initial value for this sequence is 1.5×10^{-3} per reactor-year based on an initiating frequency of 3×10^{-2} per reactor-year and an unavailability of 4.9×10^{-2} for the combination of R_B and R_A. Figure C-14 is the systemic event tree for the LOSP transient.

The RHR system in either shutdown cooling or torus cooling mode removes the reactor decay heat. The 4.9×10^{-2} unavailability for R_B and R_A is comprised of 2.9×10^{-2} due to failures of R_B and R_A independent of EECW faults and 2.0×10^{-2} due to EECW faults. The 2.9×10^{-2} unavailability is dominated by combinations of electric power system unavailabilities due primarily to diesel generator faults. The remaining 2.0×10^{-2} contribution to DHR failure comes from the unavailability of the EECW system to provide its required cooling given a LOSP. If the EECW fails, all diesel generators will eventually fail and the RHR system will be unavailable. The major contributor to the EECW unavailability is combinations of two or more diesel generators failing to start. These diesels are not necessarily the same as those that fail R_B and R_A directly. Section 1.5 details the procedure for handling this type of potential logic

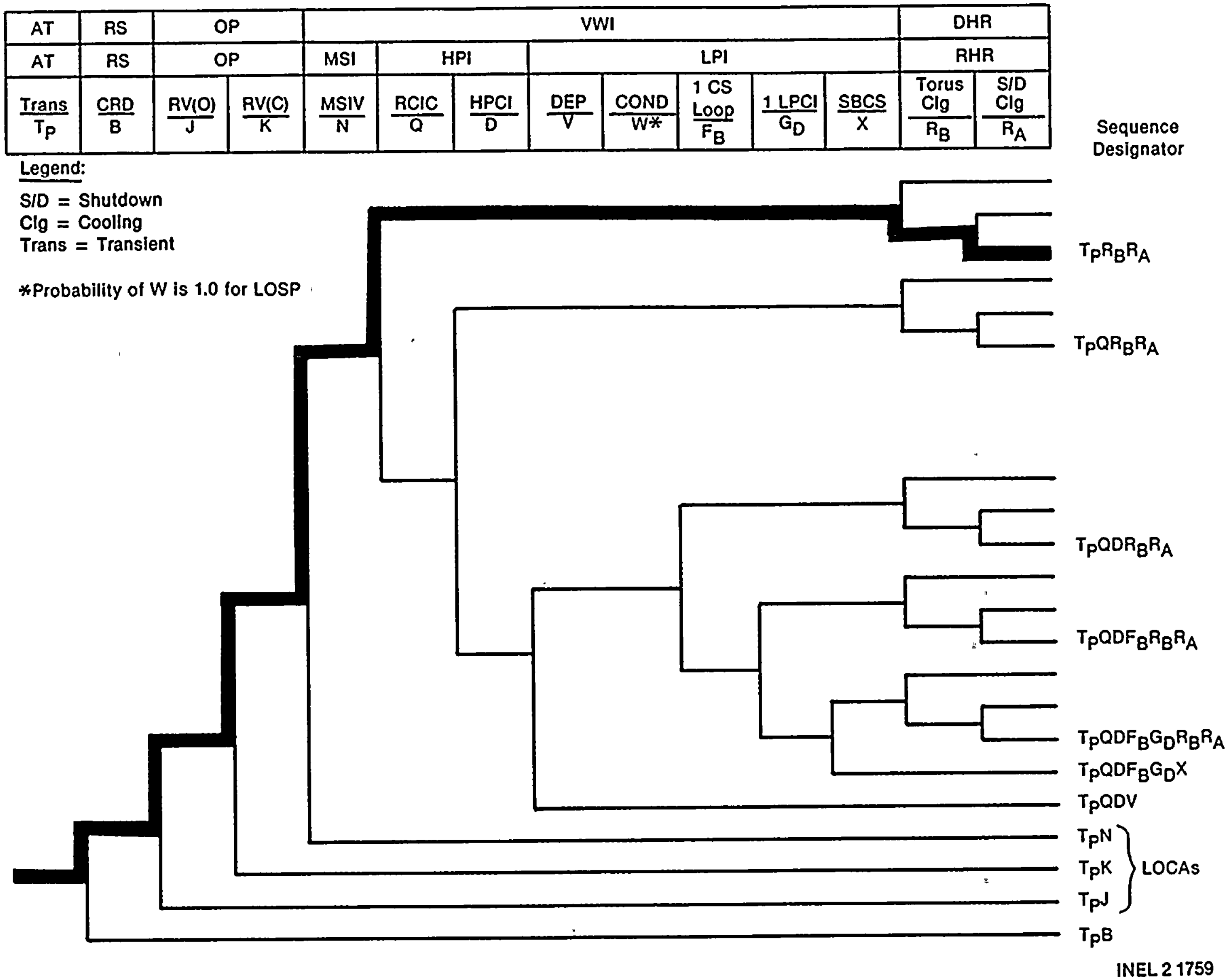


Figure C-14. Systemic event tree showing the $T_PR_BR_A$ sequence.

loop. Essentially, the R_B and R_A unavailability is split into two portions, with the unavailability assuming EECW works added to the unavailability of EECW. Each of the candidate dominant sequences involving loss of offsite power is treated similarly. Figure C-15 is a sequence evaluation diagram showing the dominant contributors to the unavailability of R_B and R_A.

Should this sequence occur, the RCIC system providing the VWI function can continue to do so for approximately 6 to 8 hours without RHR operation. This estimate is based on the time it takes to deplete the condensate storage tank and to heat the torus water to a temperature that prevents the RCIC system from pumping the water, assuming no containment backpressure.[7] With containment backpressure considered, operation of RCIC can continue for approximately 24 hours before containment failure occurs, followed by an inability to pump the torus water back to the core.

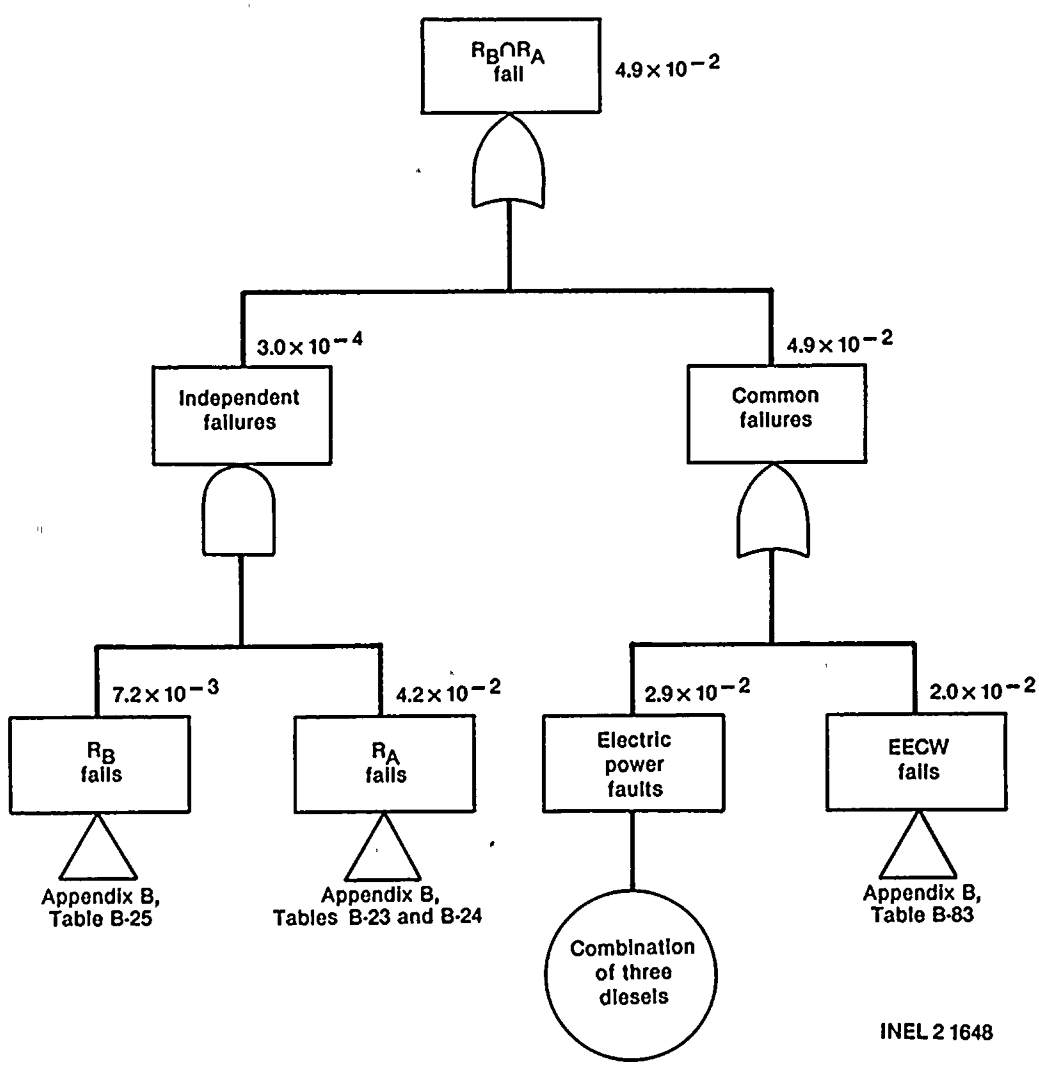


Figure C-15. Dominant contributors to the unavailability of torus cooling and shutdown cooling given LOSP.

There are several viable recovery considerations available to the operators during this time period. One is a restoration of offsite power. If offsite power is restored within the 6 to 8 hour period, the unavailability of the DHR function changes from 2.9×10^{-2} to 7.6×10^{-5}. Another recovery consideration is the restoration of EECW. The success criteria used in this analysis requires three of the four EECW pumps to operate to provide cooling to all of the EECW loads. Two of four pumps will provide up to 91% of rated flow and would provide the operator with some grace period to restore the lost pumps or valve in spare pumps from the RHRSW system. The operator could also isolate flow to nonessential loads supplied by EECW so that the flow of two pumps would provide sufficient cooling. The operator actions to restore EECW fall within the recovery guidelines as discussed previously in Section 3.3. That is, for the time period considered, there is only a 10^{-2} probability that the operator will not take corrective action during this time.

From the WASH-1400 data (Figure III 6-4), approximately 97% of all offsite power outages can be repaired in 6 to 8 hours. Using the WASH-1400 restoration figure plus the recovery factor for providing the EECW with sufficient pumping capability, the probability for R_B and R_A failure is given by:

$$Q(R_BR_A) = (0.97)[\text{probability of } R_BR_A \text{ failure with LOSP recovered}]$$

$$+ (0.03)[\text{probability of } R_BR_A \text{ failure with LOSP not recovered}]$$

$$= (0.97)(7.6 \times 10^{-5}) + (0.03)[R_BR_A \text{ failure + EECW failure}]$$

$$= (0.97)(7.6 \times 10^{-5}) + (0.03)[(2.9 \times 10^{-2}) + (2.0 \times 10^{-2})(0.01)]$$

$$= 7.4 \times 10^{-5} + (0.03)\ (2.9 \times 10^{-2})$$

$$= 9.4 \times 10^{-4}$$

$$P(T_PR_BR_A) = F(\text{LOSP})\ Q(R_B \cap R_A)$$

$$= (3 \times 10^{-2})(9.4 \times 10^{-4})$$

$$= 2.8 \times 10^{-5} \text{ per reactor-year.}$$

4.2.2 Loss of Offsite Power with RCIC and DHR Failure ($T_PQR_BR_A$)

This sequence is essentially identical to sequence $T_PR_BR_A$ except that the RCIC system fails but the HPCI system operates to maintain reactor water level. Subsequently, the torus cooling and shutdown cooling modes of RHR fail to remove decay heat. Sustained failure of these two modes will result in torus water heating to the point that the HPCI system can no longer pump water to the core. A core melt would then occur. This sequence is highlighted on the systematic event tree Figure C-16. Its initial value

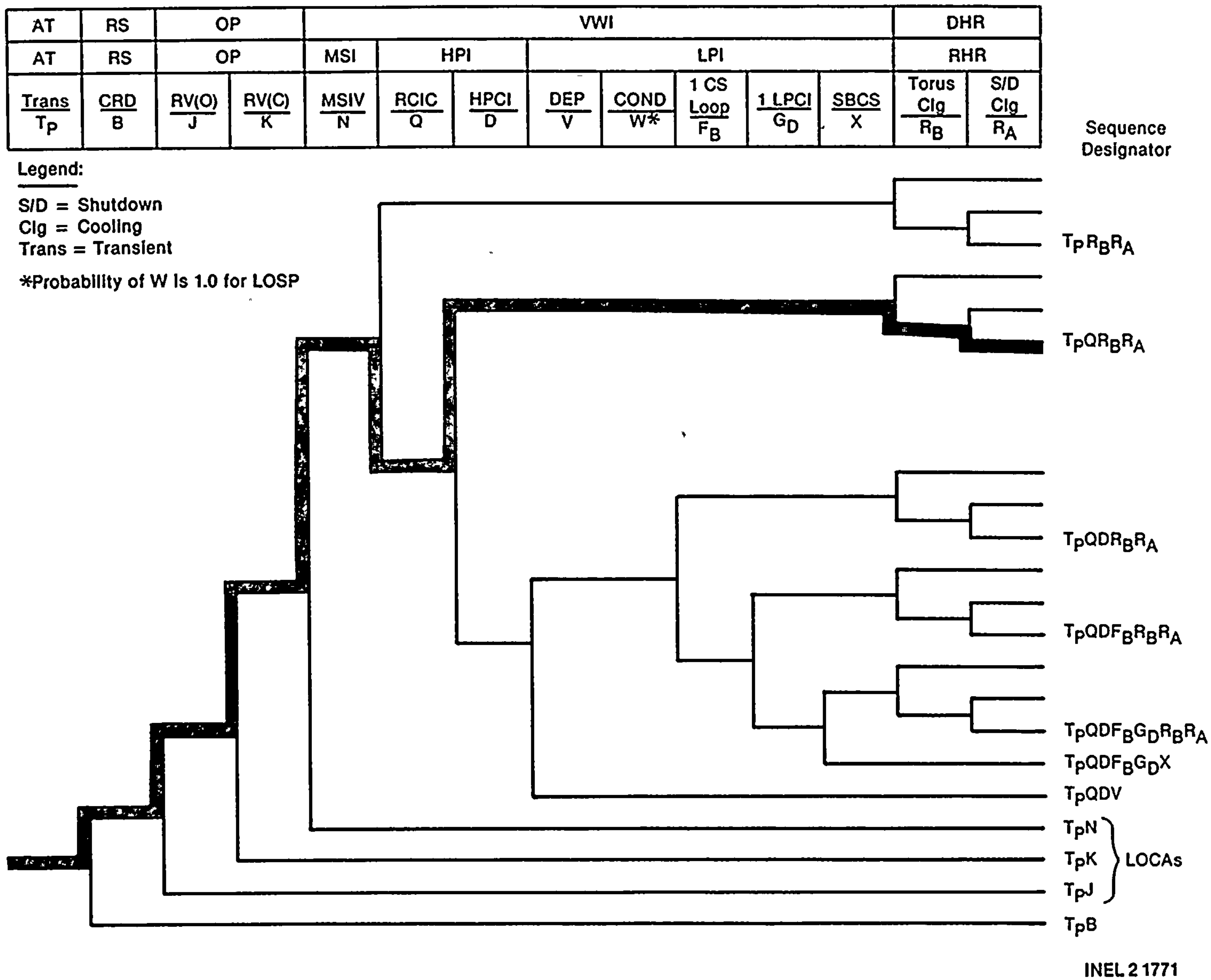


Figure C-16. Systemic event tree showing the $T_PQR_BR_A$ sequence.

is 6.2×10^{-5}, based on a 3×10^{-2} per year probability for the LOSP initiator and 2.1×10^{-3} for the unavailability of $Q \cap R_B \cap R_A$. Figure C-17 is a sequence evaluation diagram showing the dominant contributors to the unavailability of $Q \cap R_B \cap R_A$.

The dominant contributors to torus cooling and shutdown cooling failure for this sequence are the same as for sequence $T_P R_B R_A$. Therefore, the recovery factors for these two systems are the same. RCIC is essentially unaffected by the LOSP. Its dominant contributors are rupture disk and control circuit faults, which are not recoverable under the guidelines. Therefore, no credit is taken for recovery of the RCIC system. The unavailability of the mitigating systems becomes 3.9×10^{-5}. The final sequence value then is 1.2×10^{-6}, as shown below.

$Q(QR_BR_A)$ = $Q(Q)\ Q(R_BR_A$ considering recovery)

= $Q(Q)\ Q(R_BR_A$ from sequence $T_PR_BR_A)$

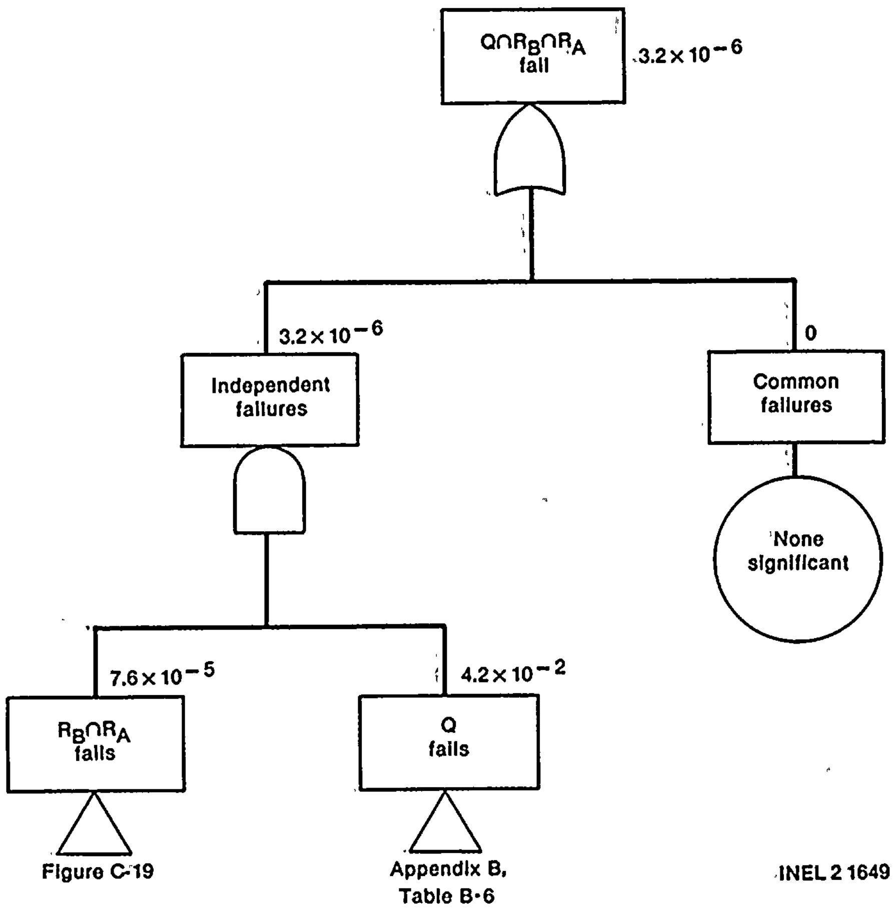


Figure C-17. Dominant contributors to the unavailability of RCIC, torus cooling, and shutdown cooling given LOSP.

$= (0.042)(9.4 \times 10^{-4})$

$= 3.9 \times 10^{-5}$

$P(T_P Q R_B R_A) = F(LOSP)\ Q(QR_B R_A)$

$= (3 \times 10^{-2})(3.9 \times 10^{-5})$

$= 1.2 \times 10^{-6}$ per reactor-year.

Q in parenthesis (Q) represents the RCIC system code.)

4.2.3 Transients Where PCS is Unavailable and DHR Fails ($T_U R_B R_A$)

For this sequence, the RS, OP, MSI, and RCIC systems succeed and the long term decay heat removal of torus cooling and shutdown cooling fails. This sequence is similar to the previously-discussed sequence ($T_P R_B R_A$) except that offsite power remains available. The initial screening value for the frequency of this sequence is 1.3×10^{-4} per reactor-year, based on an initiating frequency of 1.70 per reactor-year and an unavailability of 7.6×10^{-5} for the combination of R_B and R_A. The sequence is outlined on the systemic event tree, Figure C-18.

In this sequence, the RHR system in shutdown cooling or torus cooling mode provides the decay heat removal. Both modes must be inoperable to fail the function. The unavailability for both systems is 7.6×10^{-5}. Control circuit faults for the suction and discharge motor-operated valves and the minimum-flow bypass valves dominates this unavailability. It was assumed during the fault tree analyses of the core spray and RHR systems that minimum flow bypass valves failing to close could divert sufficient flow in a given loop to cause failure of that coolant path. Section 6 provides a sensitivity analysis of this assumption. Figure C-19 is a sequence evaluation diagram showing the dominant contributors to the unavailability of $R_B \cap R_A$.

For these transients, even though the PCS was originally lost due to MSIV closure, the potential exists to recover the main condenser as a heat sink. This depends on the cause of the transient. For example, if the transient were initiated by a fault in the feed pumps that was not immediately repairable, then the PCS could not be used. If the transient was due to faulty automatic level control, the operator could manually control level with the feed pumps after reopening the MSIVs. However, there is inadequate data available on which to base a probability of PCS recovery.

Recovery of the RHR system due to the dominant faults (control circuit faults) would involve either manual operation of the affected valves or bypass/repair of the faulted control circuit. In either case, the control room operator would have to recognize the cause of the valve's failure to operate and dispatch personnel to operate/repair the valve. Given that the RCIC system has been successful, the operator would have at least 6 to

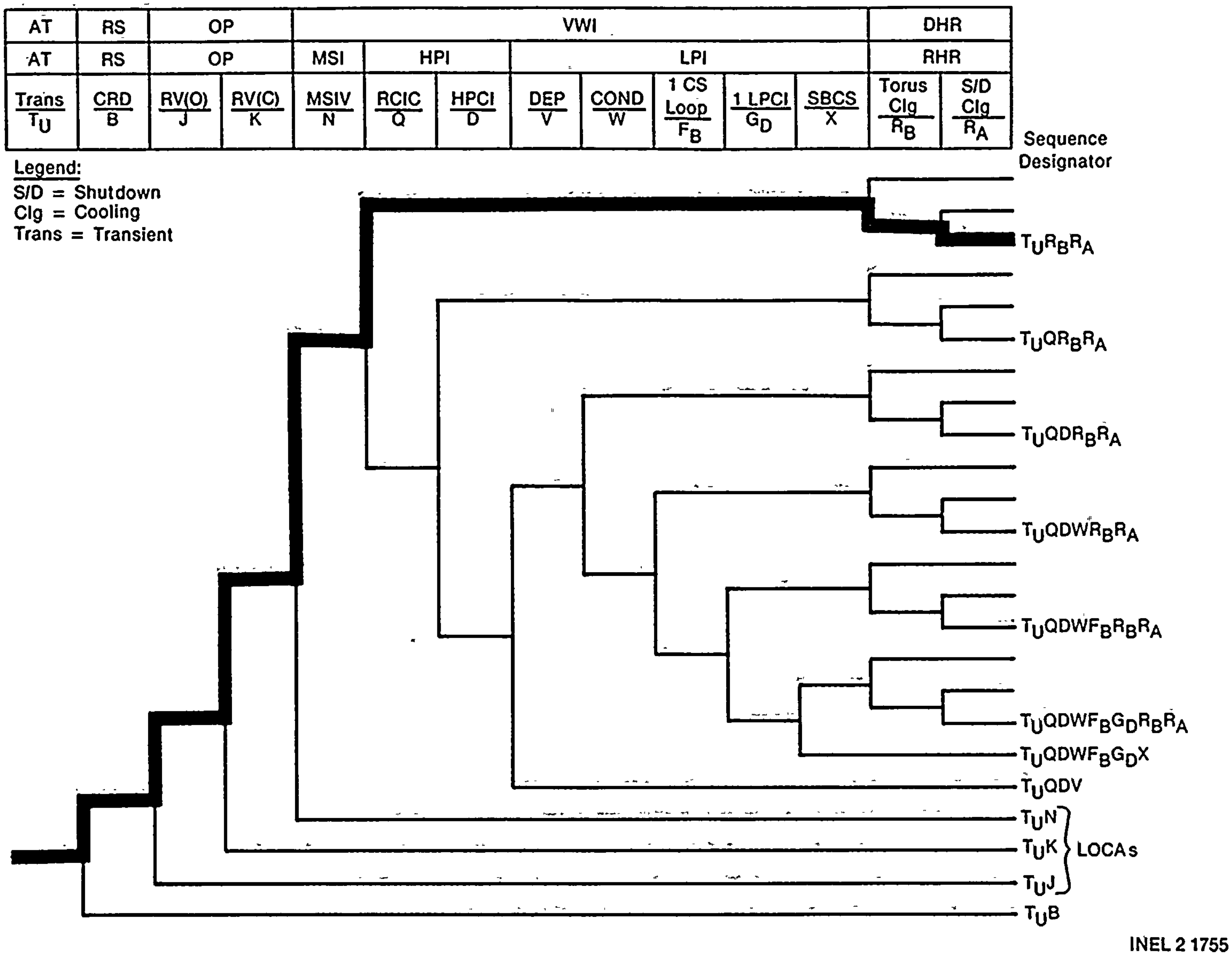


Figure C-18. Systemic event tree showing the $T_UR_BR_A$ sequence.

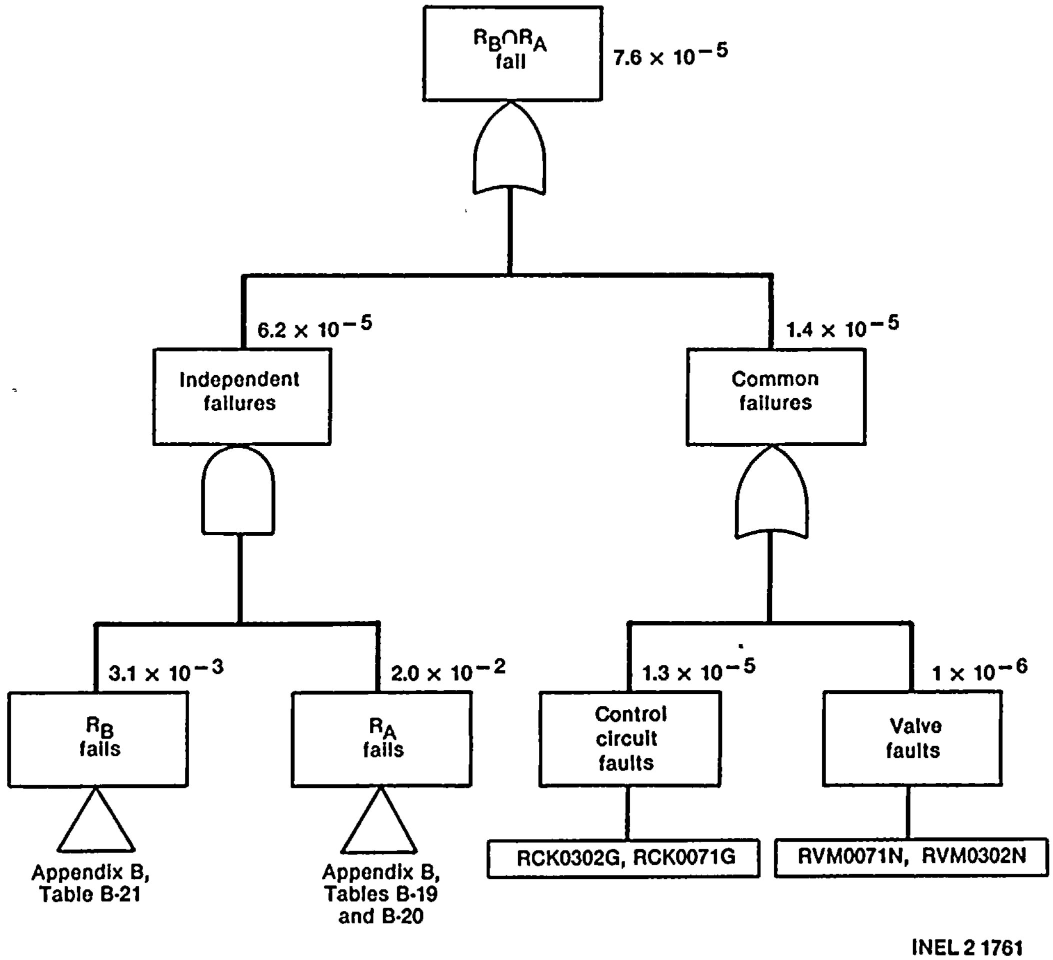


Figure C-19. Dominant contributors to the unavailability of RHR systems following a transient which disables the PCS (normal power available).

8 hours to accomplish this recovery as discussed previously in Section 4.2.2. The recovery guidelines provide for a probability of nonrecovery for these faults of 0.01. The unavailability of R_B and R_A due to both independent and common control circuit faults is 1.9×10^{-5}. The remaining unavailability not subject to recovery is 5.7×10^{-5}. The final sequence value is 9.7×10^{-5}, as shown below.

$$Q(R_B R_A) = (0.01)\ (\text{recoverable faults}) + (\text{nonrecoverable faults})$$

$$= (0.01)(1.9 \times 10^{-5}) + 5.7 \times 10^{-5}$$

$$= 5.72 \times 10^{-5}$$

$$P(T_U R_B R_A) = F(T_U)\ Q(R_B \cap R_A) = (1.7)(5.7 \times 10^{-5})$$

$$= 9.7 \times 10^{-5} \text{ per reactor-year.}$$

4.2.4 Transients Where PCS is Unavailable and RCIC and DHR Fail ($T_UQR_BR_A$)

This sequence is essentially identical to sequence $T_UR_BR_A$ except that the RCIC system fails but the HPCI system operates to maintain reactor level. Subsequent failure of torus cooling and shutdown cooling will eventually lead to a core melt. This sequence is shown on the systemic event tree Figure C-20. Its initiator frequency is 1.7 per reactor-year and 5.5×10^{-6} per reactor-year is the screening sequence value. Figure C-21 is a sequence evaluation diagram showing the dominant contributors to the initial value of 3.2×10^{-6} for the unavailability of $Q \cap R_B \cap R_A$.

The dominant contributors for torus cooling and shutdown cooling failure are the same for this sequence as for sequence $T_UR_BR_A$. The recovery factors are also the same. The RCIC system dominant faults involve rupture disks and control circuits and are independent of the torus cooling and shutdown cooling faults. These faults are not recoverable under the guidelines, so no credit is taken for RCIC system recovery. Therefore, the unavailability of the mitigating systems considering recovery is 2.4×10^{-6}, as shown below.

$$Q(QR_BR_A) = Q(Q) \cdot Q(R_BR_A \text{ considering recovery})$$

$$= Q(Q)\ Q(R_BR_A \text{ for sequence } T_UR_BR_A)$$

$$= (0.042)(5.7 \times 10^{-5})$$

$$= 2.4 \times 10^{-6}$$

$$P(T_UQR_BR_A) = F(T_U)\ Q(QR_BR_A)$$

$$= (1.7)(2.4 \times 10^{-6})$$

$$= 4.1 \times 10^{-6} \text{ per reactor-year.}$$

4.2.5 LOSP-Induced Stuck Open Relief Valve (SORV) with DHR Failure ($T_PKR_BR_A$)

In this sequence, the LOSP causes a turbine trip without bypass, a reactor scram, main steam isolation, and opening of the relief valves. However, one or more of the relief valves fail to reclose after pressure has fallen below the relief valve setpoint. This is equivalent to an intermediate steam break with one exception. The steam from the relief valves does not go into the drywell. Rather, it goes to the torus water directly. After the HPCI system has succeeded in restoring reactor vessel level to normal, the torus cooling and shutdown cooling systems fail. The initial value for this sequence is 8.3×10^{-5}, based on an initiating frequency of 1.7×10^{-3} per year and 4.9×10^{-2} for the unavailability of $R_B \cap R_A$. The marked systemic event tree is Figure C-22.

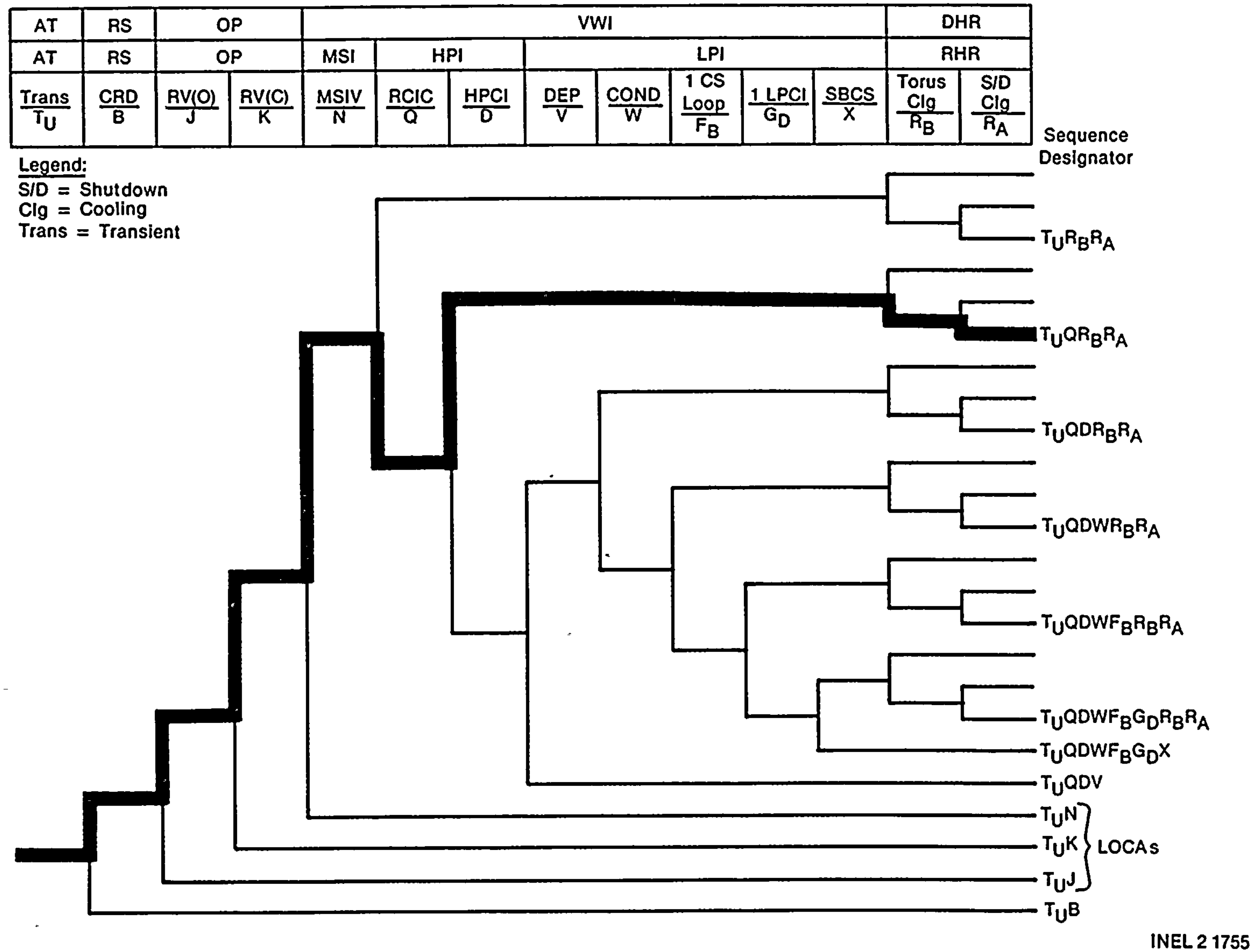


Figure C-20. Systemic event tree showing the $T_U Q R_B R_A$ sequence.

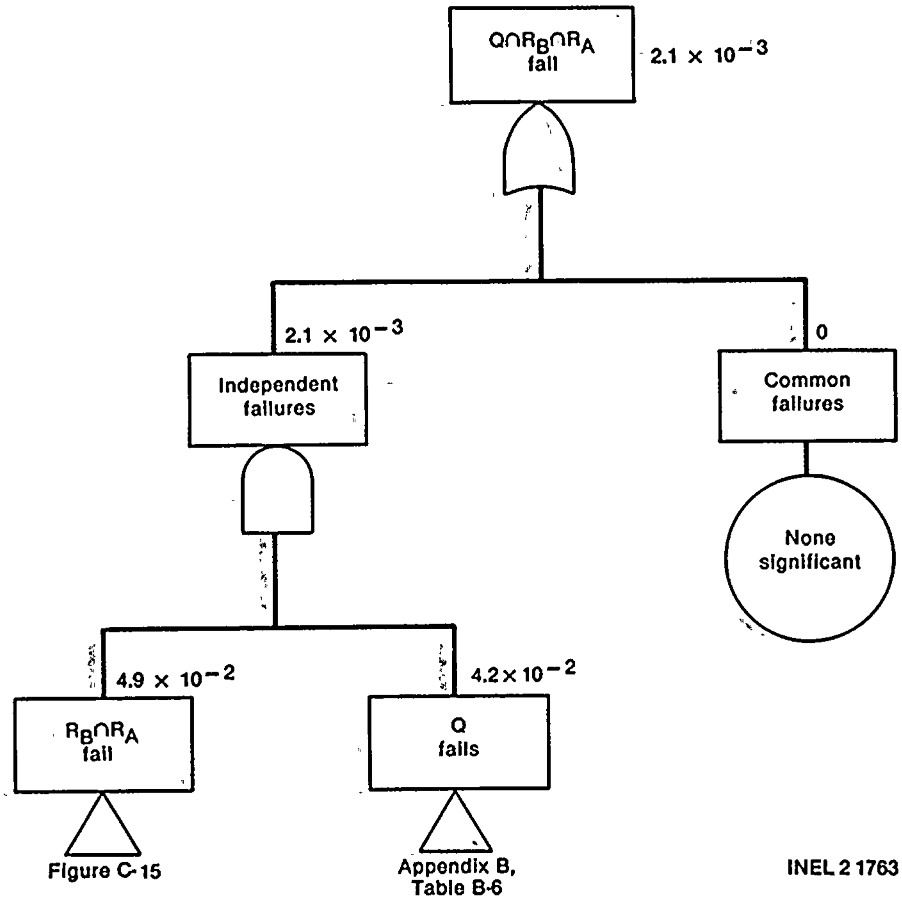


Figure C-21. Dominant contributors to the unavailability of the RCIC and RHR systems following a transient which disables the PCS (normal power available).

The RHR system in shutdown cooling or torus cooling mode provides the long term cooling. As was the case in the sequence $T_PR_BR_A$ of Section 4.2.1, the unavailability of the DHR function is 4.9×10^{-2}. The dominant contributor sequence evaluation diagram for that sequence applies to this sequence as well.

Preliminary phenomenological calculations being performed at INEL on BF1 as part of the Severe Accident Sequence Analysis (SASA) program[8] indicate that core temperatures will start to rise rapidly in as little as 30 min if the HPCI system does not function to replenish lost coolant inventory. Even with successful HPCI, torus water temperature will rise and eventually reach a temperature where the HPCI system will no longer have sufficient net positive suction head to maintain reactor level. The time available to recover is approximately the same as the LOSP with R_B and R failure. For this sequence, since HPCI is successful, it was assumed that at least 6 to 8 hours are available to restore offsite power. Using the WASH-1400 restoration figure and recovery factors based on 6 to 8 hours, the probability for R_B and R_A failure is the same as for sequence $T_PR_BR_A$ (9.4×10^{-4}). The final sequence value is then 1.6×10^{-6}.

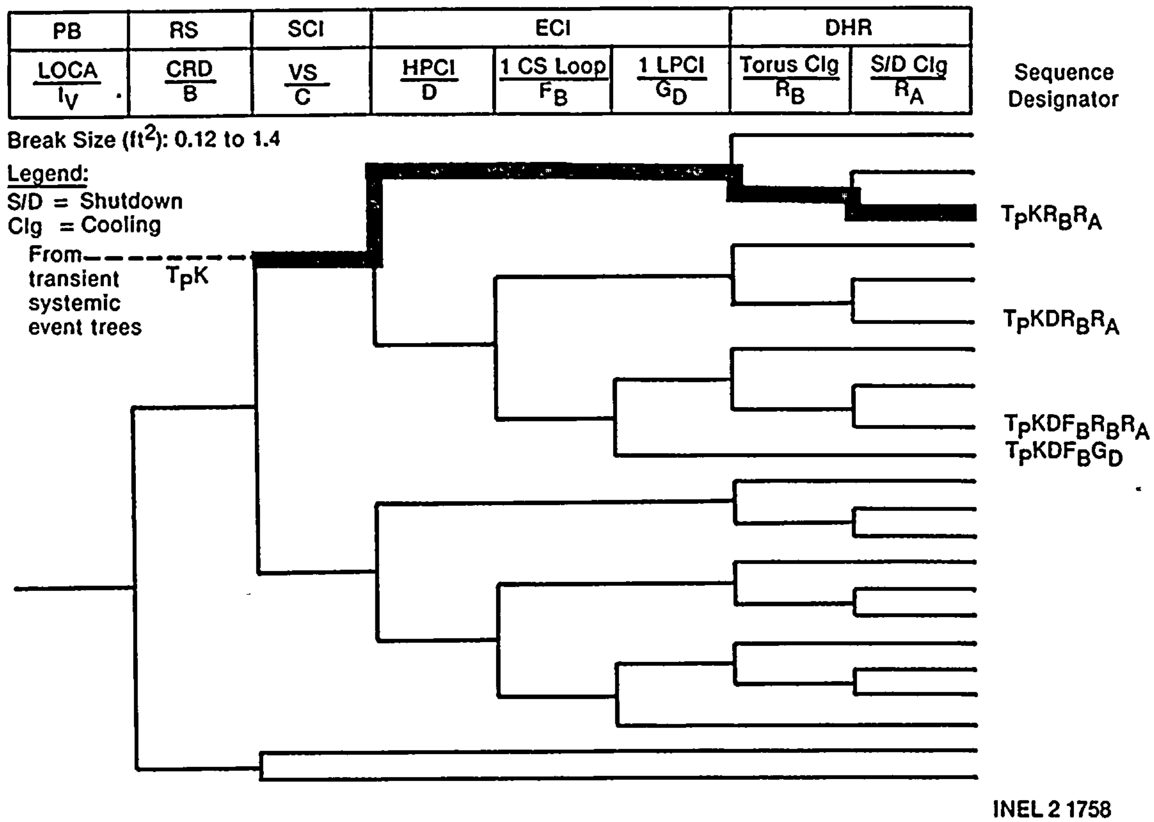


Figure C-22. Systemic event tree showing the $T_PKR_BR_A$ sequence.

$$P(T_PKR_BR_A) = F(T_PK)\ Q(R_BR_A)$$

$$= (1.7 \times 10^{-3})\ (9.4 \times 10^{-4})$$

$P(T_PKR_BR_A) = 1.6 \times 10^{-6}$ per reactor-year.

4.2.6 Transient-Induced SORV with DHR Failure (TKR_BR_A)

For this sequence, a transient causes a reactor scram and a number (depending on the initiator) of relief valves open. When pressure falls below the relief valve setpoint, one or more relief valves fail to close, and the reactor continues to blow down to the torus. When low water level is reached, the MSIVs shut and the HPCI system initiates. Following successful ECI by the HPCI system, the torus cooling and shutdown cooling systems fail. The initial value for this sequence is 1.2×10^{-5} per reactor-year, from a frequency of 1.63×10^{-1} per year for the initiator and an unavailability of 7.6×10^{-5} for the combination of $R_A \cap R_B$. The sequence is shown on the systemic event tree, Figure C-23.

This sequence is similar to the LOSP-induced SORV sequence of Section 4.2.5 except that the unavailability for the R_B and R_A combination is much lower because offsite power is available. However, the initiator frequency is approximately two orders of magnitude higher than for the LOSP-induced SORV sequences. The RHR system provides the long-term decay heat removal function in either the shutdown cooling or torus cooling mode. The unavailability for both modes is 7.6×10^{-5}. The major contributors to this unavailability are control circuit faults of the minimum-flow bypass valves and the suction and discharge path MOVs. The dominant contributor sequence evaluation diagram of Section 4.2.3 applies for this sequence also.

As with the sequence $T_UR_BR_A$ of Section 4.2.3, recoverability of the torus cooling and shutdown cooling systems is either by manual operation/repair of faulted RHR valve control circuits or by recovery of the PCS as a heat sink if possible. However, recovery of PCS is not easily quantifiable. Therefore, the final sequence probability for this sequence does not include a probability of recovery of PCS. The recoverability of R_B and R_A has been previously accounted for (5.7×10^{-5}). Therefore, the final sequence value is 9.3×10^{-6}, as shown below.

$$P(TKR_BR_A) = F(TK)\ Q(R_B \cap R_A)$$

$$= (0.163)(5.7 \times 10^{-5})$$

$$= 9.3 \times 10^{-6} \text{ per reactor-year.}$$

4.2.7 Transients without PCS with VWI Failure (T_UQDV)

A transient occurs that causes the reactor to scram and the MSIVs to close. After the relief valves open to relieve the increase in reactor pressure, all the valves reclose. This action repeats as reactor decay heat

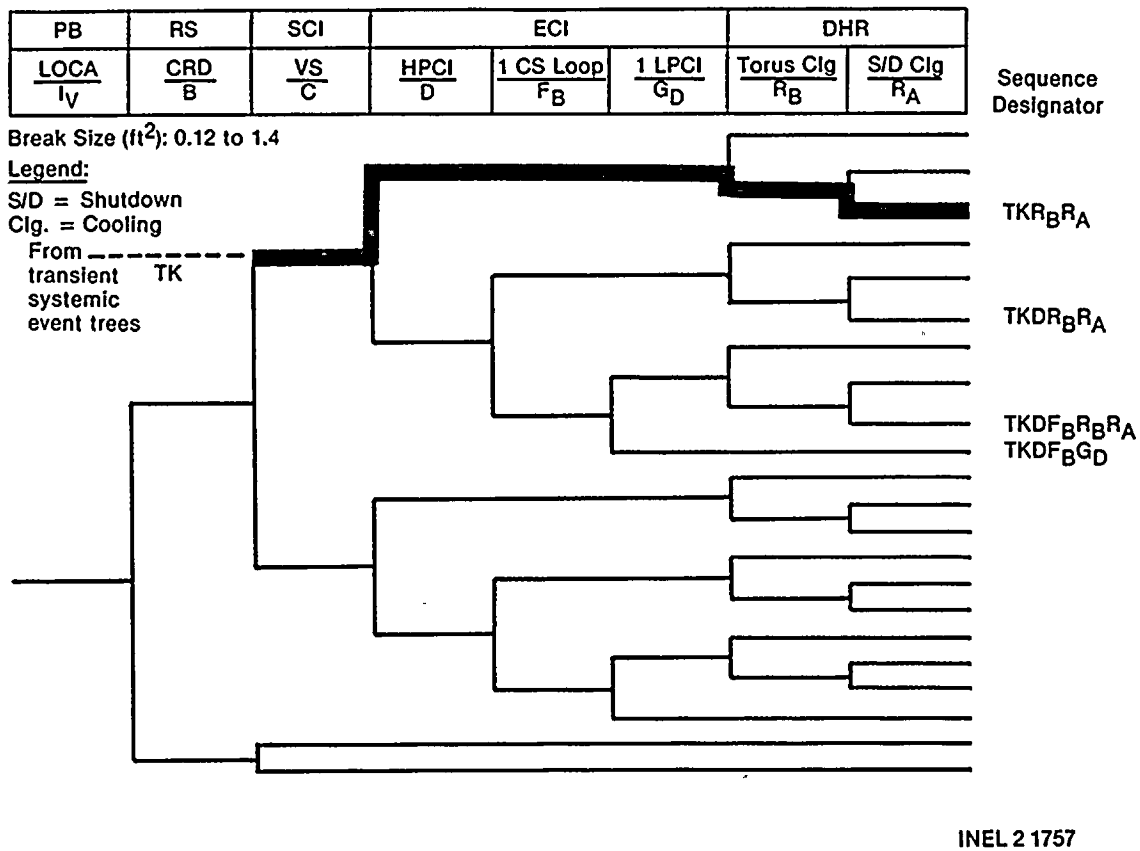


Figure C-23. Systemic event tree showing the TKR_BR_A sequence.

causes pressure to rise due to the lack of a heat sink. When low reactor level is reached, the HPCI and RCIC systems (D and Q, respectively) fail to operate to restore water level. As the water level continues to drop due to relief valve action, the operator fails to manually depressurize the reactor. If this condition persists, core melt occurs. The initial probability for this sequence is 2.8×10^{-5} per reactor-year, based on an initiating event frequency of 1.70 per year and 5.4×10^{-6} for the unavailability of $Q \cap D \cap V$. The systemic event tree showing this sequence is Figure C-24.

The unavailability of HPCI and RCIC combined is 1.8×10^{-3} and is dominated by rupture disk faults and control circuit faults of the MOVs in each system. The probability of failure to manually depressurize is dominated by failure of the operator to initiate depressurization, since only 4 of 13 valves are required to open for successful depressurization. Figure C-25 is a sequence evaluation diagram of the dominant contributors for this sequence. During the injection phase, there is little time available (30 to 40 min) for the operator to dispatch personnel to correct faults involving the HPCI and RCIC MOVs. If the operator fails to depressurize, then water level continues to decrease and the low pressure systems such as core spray, LPCI, and the condensate system cannot provide water to the reactor because reactor pressure is too high.

The probability of the operator failing to depressurize was originally taken to be 3×10^{-3}, based on the human error modeling guide of NUREG/CR-1278. This model, shown in Section 4.2 of Appendix B, does not include recovery because it was developed for initial screening purposes. However, recovery from an initial operator error in failing to depressurize is likely because of the heavy emphasis on depressurization given to operators during their training and the ease with which this action can be carried out. Since this recovery relates to operator error rather than actions directly mitigating the effect of hardware faults, the nonrecovery factors of Table C-6 are not applicable. Therefore, a more detailed operator action model was developed, considering not only the time frame for operator action but also the effect of additional operators in the control room. The new model, presented in Section 4.3 of Appendix B, shows that a consideration of recovery reduces the human error probability of failure to depressurize by a factor of 0.06. The final sequence frequency is then 5.5×10^{-7} per reactor-year, as shown below.

$$Q(QDV) = (0.06)\, Q(Q \cap D \cap V)$$

$$= (0.06)(5.4 \times 10^{-6})$$

$$= 3.2 \times 10^{-7}$$

$$P(T_U QDV) = F(T_U)\, Q(QDV)$$

$$= (1.7)(3.2 \times 10^{-7})$$

$$= 5.5 \times 10^{-7} \text{ per reactor-year.}$$

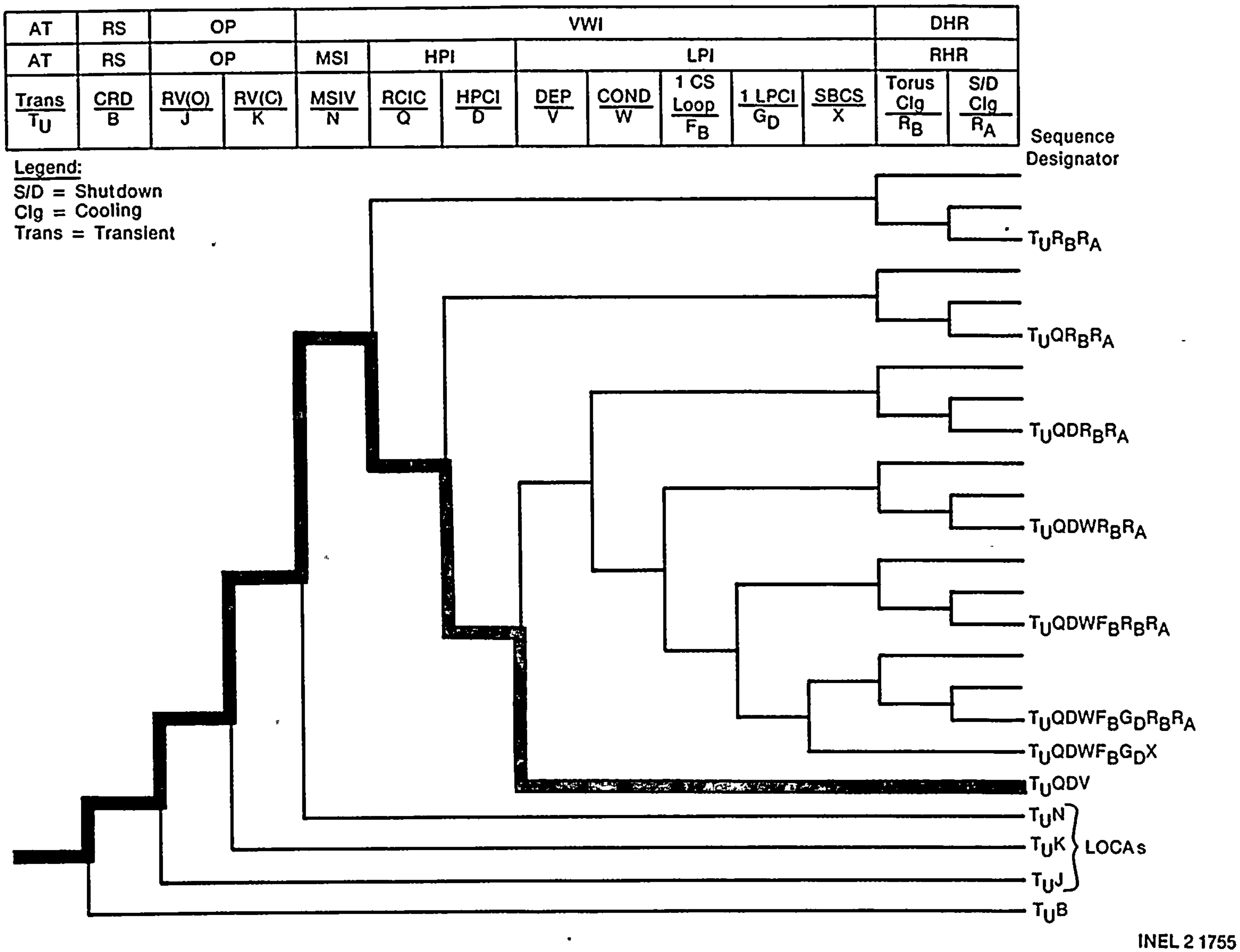


Figure C-24. Systemic event tree showing the T_UQDV sequence.

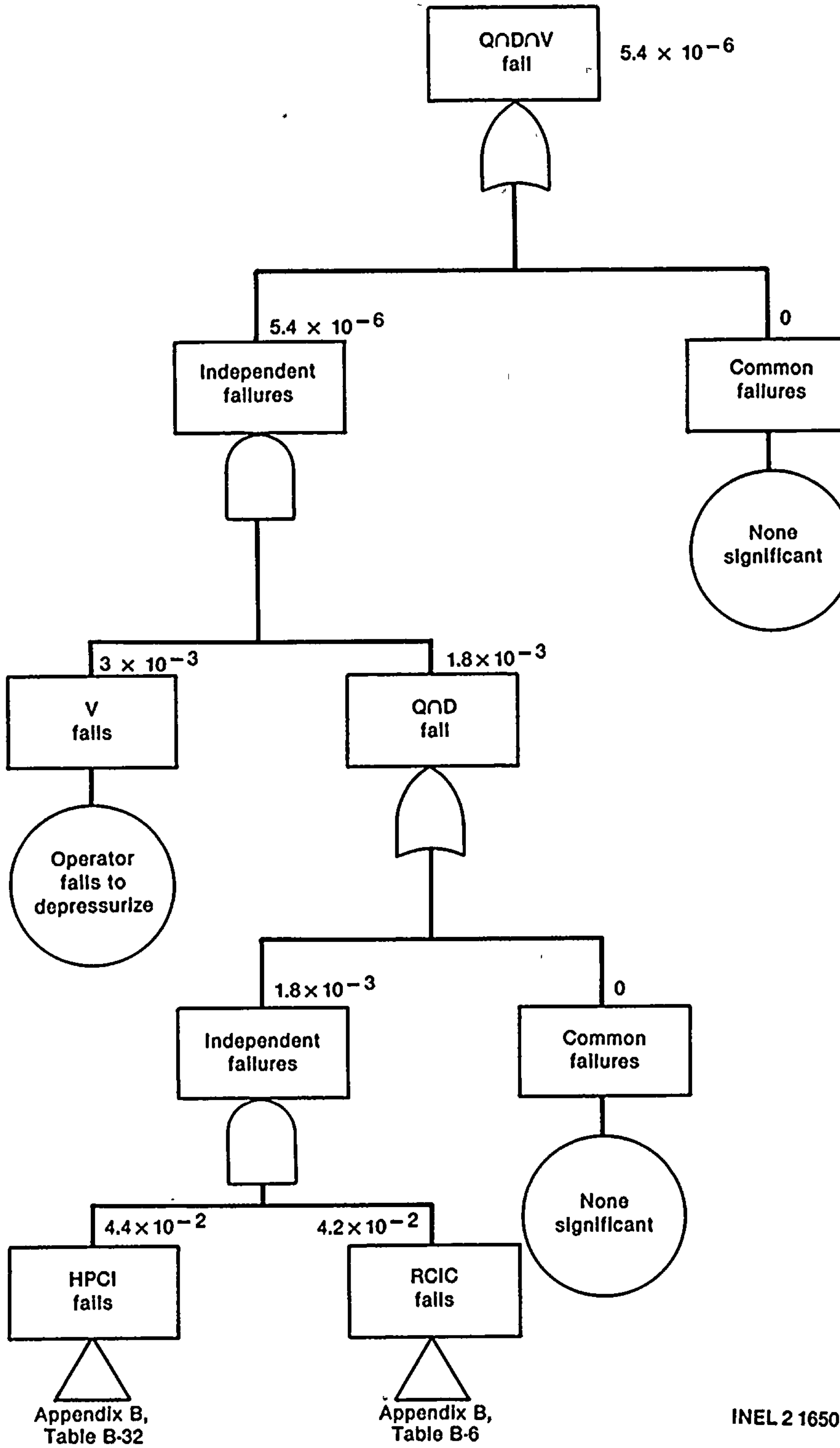


Figure C-25. Dominant contributors to the unavailability of RCIC, HPCI, and manual depressurization following a transient which disables the PCS (normal power available).

4.2.8 Loss of Offsite Power with VWI Failure ($T_PQDF_BG_DX$)

After a LOSP, a turbine trip without bypass and a reactor scram occur. The relief valves open to relieve the pressure increase caused by the loss of the heat sink and reclose when pressure falls below the relief valve setpoint. This cycle continues until a low reactor water level is reached. At this point, the HPCI and RCIC systems (D and Q) fail to operate to maintain reactor water level. As the water level continues to drop due to relief valve action, the operator successfully depressurizes the reactor, but the core spray, LPCI, and SBCS systems (F_B, G_D, and X, respectively) fail to restore water level and core melt occurs. The initial value for this sequence is 1.2×10^{-6} per reactor-year, based on 1.70 per reactor-year for the frequency of LOSP and 4×10^{-5} for the unavailability of $D \cap F_B \cap G_D \cap X$. This sequence is highlighted on Figure C-26, the systemic event tree for the LOSP transient.

The unavailability of the injection systems for this sequence, i.e., the unavailability of HPCI and RCIC, is 1.8×10^{-3} and is essentially unaffected by the loss of offsite power. The unavailability of core spray, LPCI, and SBCS is affected by the LOSP and is 1.5×10^{-3}. This number is primarily due to diesel generator faults. Additionally, failure of the EECW system to provide its required cooling will cause the loss of all diesels and, thus, AC power for the RHR and core spray pumps. The EECW unavailability is 2.0×10^{-2}. The EECW value is dominated by combinations of failure of two diesels to start. These are not necessarily the same diesels that cause core spray and LPCI failure. Figure C-27 is a sequence evaluation diagram showing the dominant contributors to the mitigating systems unavailability.

Should this sequence occur, with the injection systems failed there are approximately 30 to 40 min before boiloff reduces reactor coolant inventory to a point where core temperature begins to rapidly rise. There are several viable recovery considerations available to the operators during this time period. One is a restoration of offsite power. From WASH-1400 (Figure III 6-4), approximately 70% of all offsite power outages can be repaired in 30 to 40 min. If LOSP is restored, this sequence is essentially the same as the transients without PCS with VWI failure, sequence $T_UQDWF_BG_DX$.

Similarly, as with the LOSP with DHR failure sequence $T_PR_BR_A$ of Section 4.2.1, the EECW success criteria require three of four pumps to operate. If only two of four pumps operate, up to 91% of rated flow is available. Two additional RHRSW pumps are available for EECW service by opening (from the control room) one MOV for each pump.

Considering recovery, the unavailability of the injection systems becomes 1.2×10^{-6}. The final sequence frequency is then 3.6×10^{-8}, as shown below.

$$Q(QDF_BG_DX) = (0.70)(\text{injection systems failure with LOSP recovered})$$

$$+ (0.30)(\text{injection systems failure without LOSP recovered})$$

$$= (0.70)(\text{unavailability of sequence } T_UQDWF_BG_DX)$$

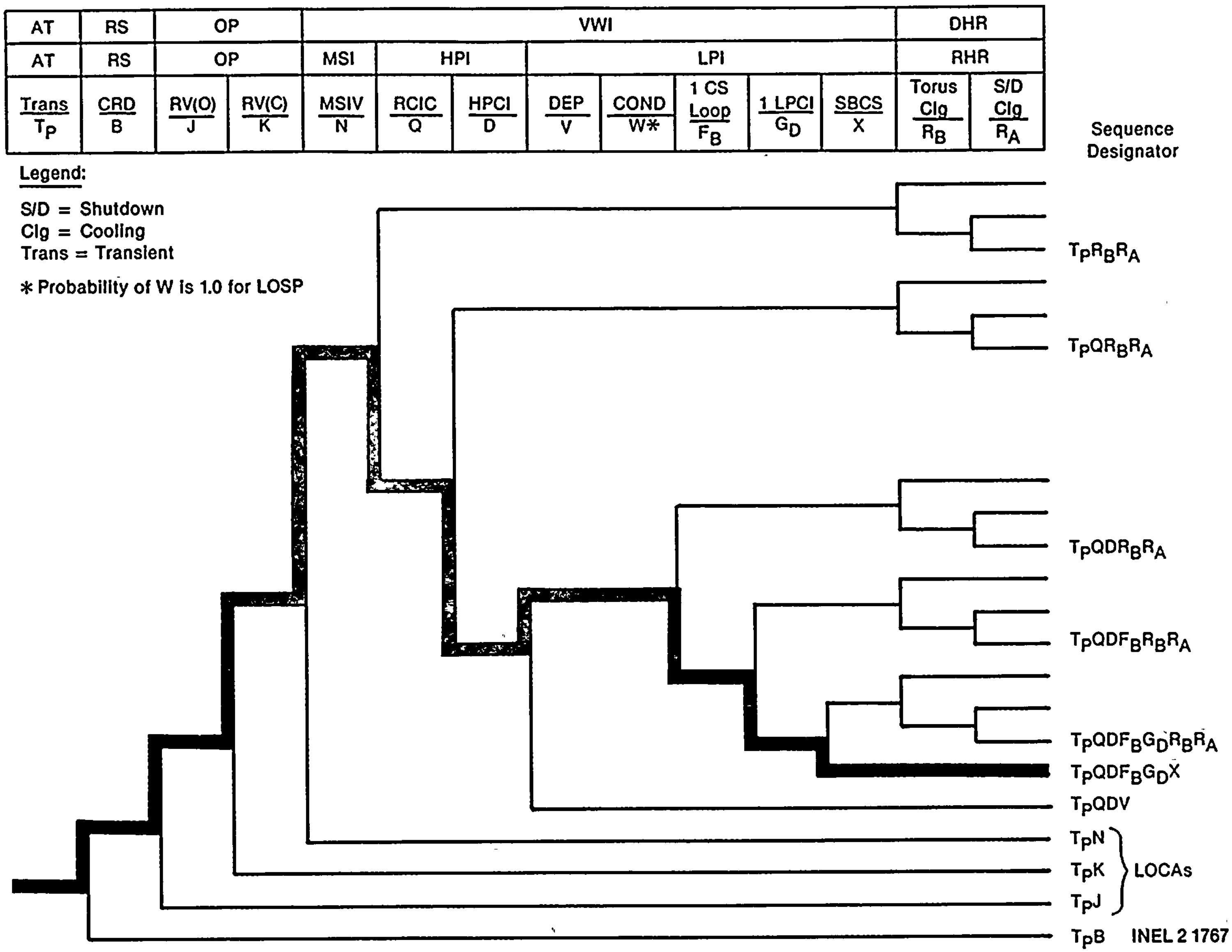


Figure C-26. Systemic event tree showing the $T_PQDF_BG_DX$ sequence.

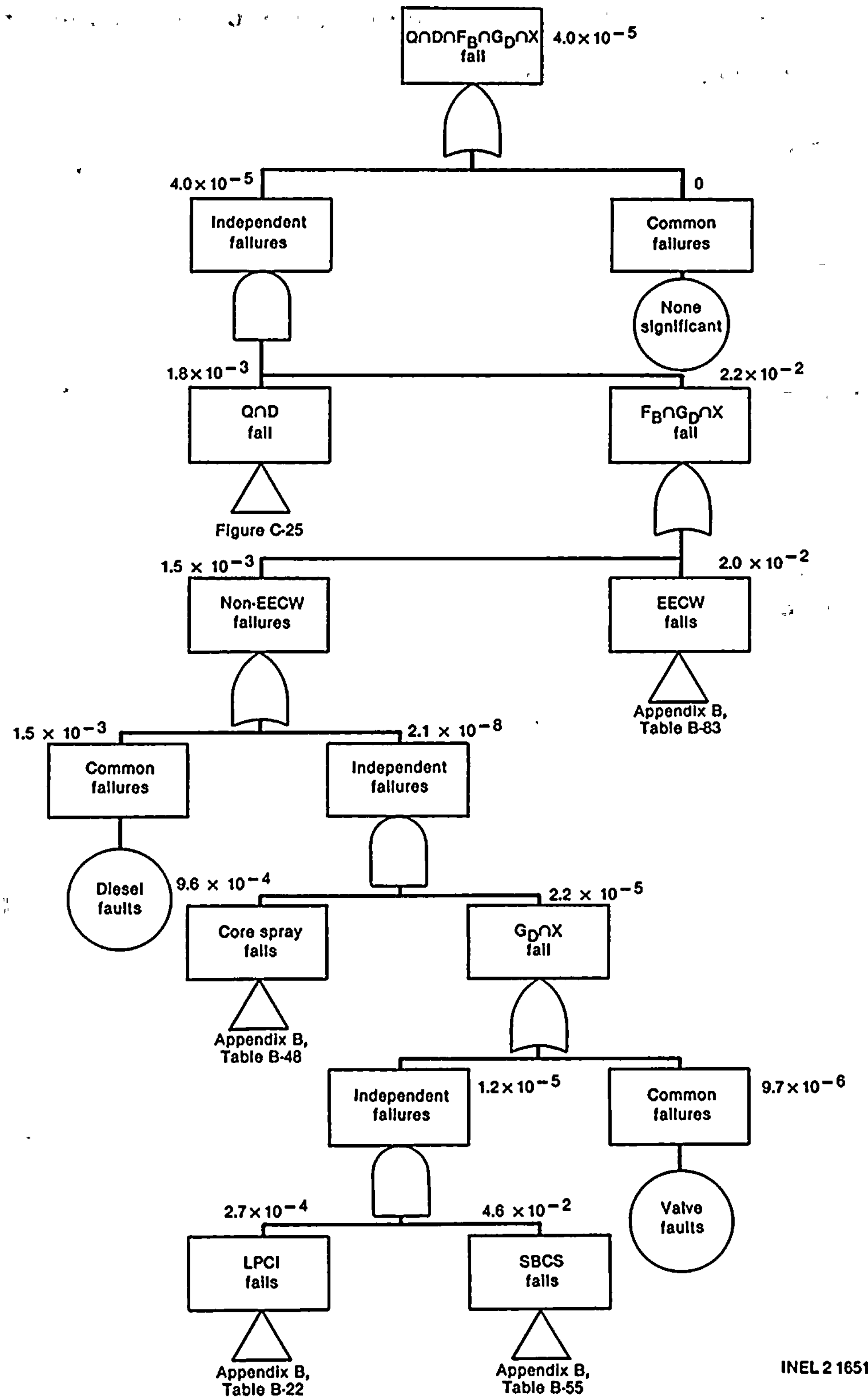


Figure C-27. Dominant contributors to the unavailability of RCIC, HPCI, LPCI, core spray, and SBCS given LOSP.

$$+ (0.30)(\text{failure of HPCI and RCIC})$$

$$\cdot [(\text{failure of LPCI, core spray, and SBCS}) + (\text{failure of EECW})$$

$$\cdot (\text{operator nonrecovery})]$$

$$= (0.70)(1.7 \times 10^{-8})$$

$$+ (0.30)(1.8 \times 10^{-3})[1.5 \times 10^{-3} + (2.0 \times 10^{-2})(0.03)]$$

$$= 1.2 \times 10 + 1.2 \times 10$$

$$= 1.2 \times 10^{-6}$$

$$P(T_P QDF_B G_D X) = F(T_P)\ Q(QDF_B G_B X_D)$$

$$= (3 \times 10^{-2})(1.2 \times 10^{-6})$$

$$= 3.6 \times 10^{-8} \text{ per reactor-year.}$$

4.2.9 LOPS-Induced SORV with ECI Failure ($T_PKDF_BG_D$)

A LOSP causes a reactor scram and turbine trip without bypass. After the relief valves open to relieve the pressure increase caused by the loss of the heat sink, one or more of the relief valves fail to reseat when pressure drops below the relief valve setpoint. When water level drops to the low level point, the MSIVs shut but the HPCI system does not operate to refill the reactor. Subsequently, as level and pressure drop, neither the core spray nor LPCI systems operate to fill the reactor and a core melt occurs. The initial value for this sequence is 2.5×10^{-6} per reactor-year based on an initiating frequency of 1.7×10^{-3} per reactor-year and an unavailability of 1.5×10^{-3} for the combination of D, F_B, and G_D. The sequence is shown on the systemic event tree, Figure C-28.

The unavailability for the injection systems for this sequence is based on failure of the HPCI, core spray, and LPCI systems to operate. The HPCI unavailability is essentially not affected by the LOSP. The unavailability of core spray and LPCI, however, is dominated by combinations of diesel generator faults. Furthermore, the EECW unavailability (2.0×10^{-2}) also contributes to the probability of core spray and LPCI failure. Figure C-29 is a sequence evaluation diagram showing the dominant contributors to the mitigating systems unavailability.

As mentioned before in other LOSP sequences, the EECW success criteria was three of four pumps operating. Since two of four pumps can provide at least 91% of rated flow and since two other RHRSW pumps are readily available to provide flow to the EECW header, EECW is subject to recovery considerations.

Approximately 30 min is available for recovery while the relief valve remains stuck open. As discussed in previous sequence descriptions,

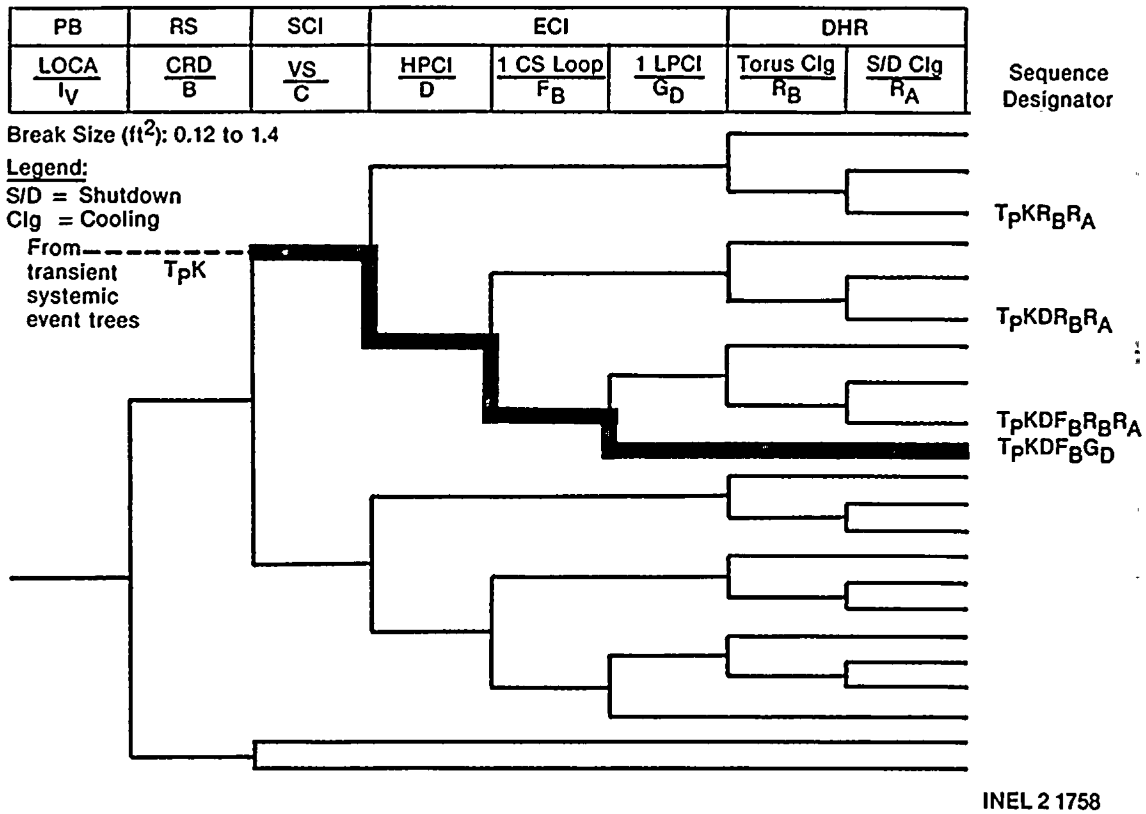


Figure C-28. Systemic event tree showing the $T_PKDF_BG_D$ sequence.

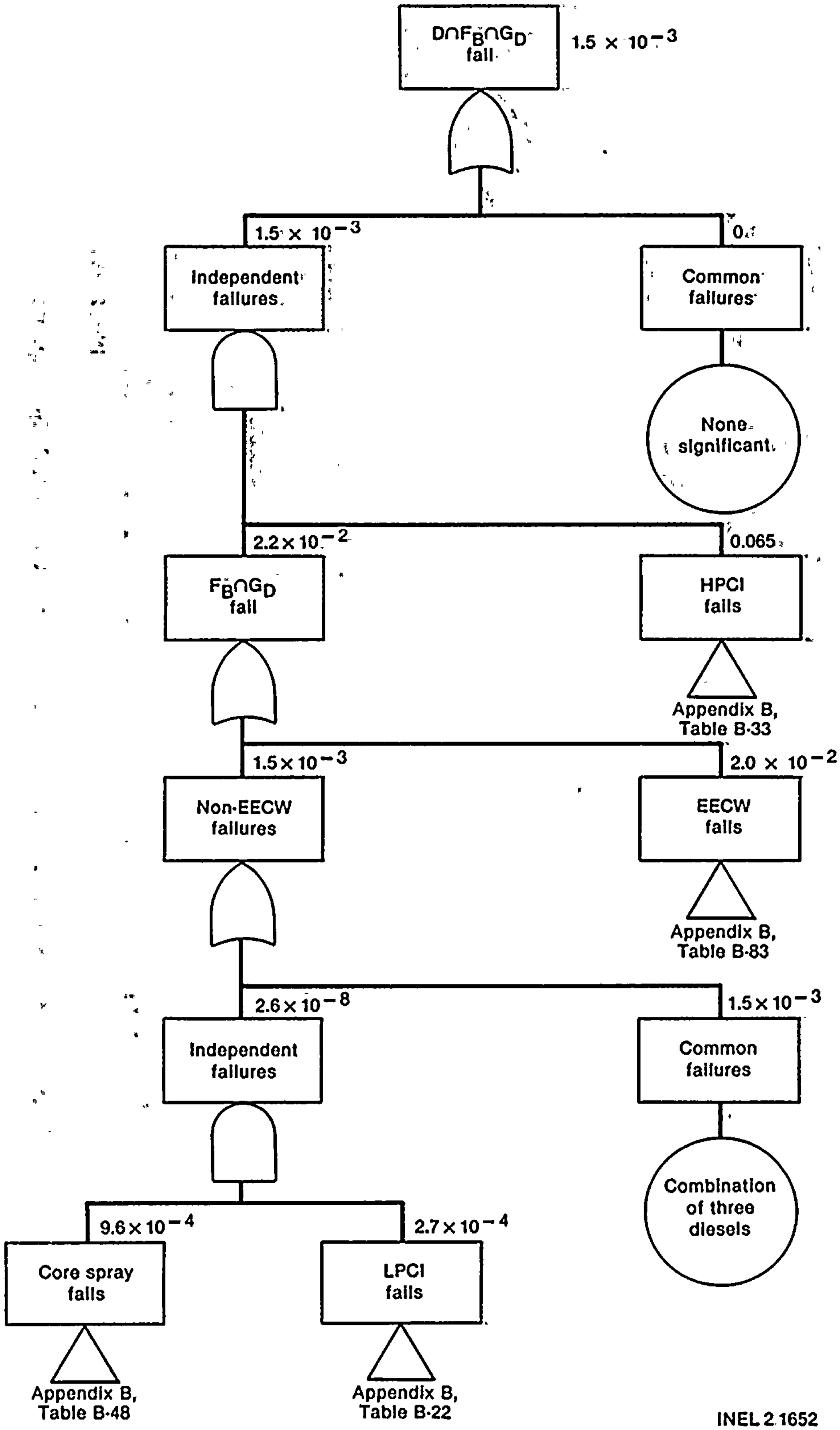


Figure C-29. Dominant contributors to the unavailability of HPCI, LPCI, and core spray, given LOSP and SORV.

approximately 70% of the offsite power outages can be repaired during this time. If offsite power is restored, this sequence becomes very similar to the transient-induced SORV sequence, $TKDF_BG_D$.

Considering the potential recovery of offsite power and the EECW system, the injection system unavailability becomes 5.1×10^{-5}. The final sequence frequency is then 8.7×10^{-8}, as shown below.

$$Q(F_BG_D) = (0.70)\ (\text{unavailability with LOSP recovered})$$

$$+ (0.30)\ (\text{unavailability with LOSP not recovered})$$

$$= (0.70)\ (\text{injection systems for sequence } TKDF_BG_D)$$

$$+ (0.30)(\text{failure of HPCI})[(\text{failure of core spray and LPCI to operate}) + (\text{failure of EECW})\ (\text{operator nonrecovery})]$$

$$= (0.70)(2.4 \times 10^{-6}) + (0.30)(6.5 \times 10^{-2})$$

$$\cdot [(1.5 \times 10^{-3}) + (2.0 \times 10^{-2})(0.05)]$$

$$Q(DF_BG_D) = 1.7 \times 10^{-6} + 4.9 \times 10^{-5}$$

$$= 5.1 \times 10^{-5}$$

$$P(T_PKDF_BG_D) = F(T_PK)\ Q(DF_BG_D)$$

$$= (1.7 \times 10^{-3})(5.1 \times 10^{-5})$$

$$= 8.7 \times 10^{-8} \text{ per reactor-year.}$$

4.2.10 Transients Without PCS with Failure to Scram (T_UB)

In this sequence, a transient occurs that makes the main condenser unavailable as a heat sink. Failure of the reactor to scram allows reactor power to remain high. As a result, the pressure increases until the relief valves open. The HPCI and RCIC systems are not capable of providing makeup to the reactor as fast as steam is being lost to the torus via the relief valves. Therefore, the core uncovers and melts. The initial value for this sequence is 5.1×10^{-5}, based on an initiating frequency of 1.70 per reactor-year and an unavailability of 3×10^{-5} for failure to scram. The sequence is shown on the systemic event tree, Figure C-30.

Since core uncovery in this scenario will occur within the first 10 min (depending on power level), the recovery guidelines do not allow for considering operator action to correct the condition. Therefore, the final sequence value is the same as the initial value of 5.1×10^{-5}.

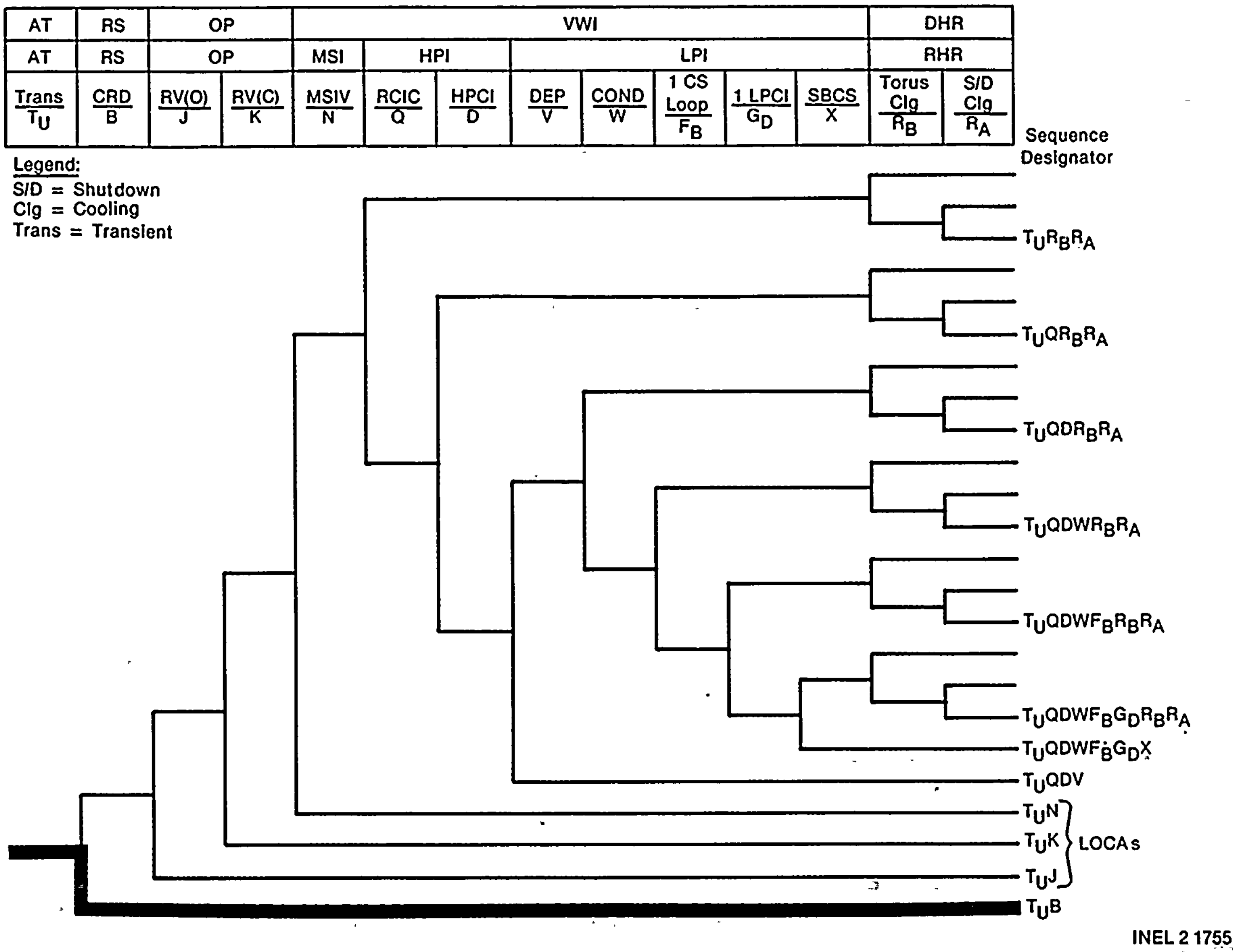


Figure C-30. Systemic event tree showing the T_UB sequence.

The unavailability for the reactor scram function is 3.0×10^{-5}. This number was taken from the ATWS document NUREG-0460. This analysis did not evaluate the probability of failure to scram using the fault tree methodology. As noted in WASH-1400 and NUREG-0460, the exact number of rods that must fail to insert and the relative position of those rods is not easily calculated and was considered beyond the scope of this analysis. Thus, the NUREG-0460 probability value for failure to reach subcriticality was used in lieu of a specific evaluation of the reactor subcriticality function.

4.2.11 Transients with PCS with Failure to Scram and Recirculation Pump Trip Failure (T_ABM)

In this sequence, a transient occurs that does not cause the MSIVs to close. The PCS is available both as a heat sink and a source of makeup water to the reactor. If the RPT is successful, the resulting reactor power level is within the capacity of the bypass valves to remove heat from the reactor. Failure of the RPT allows reactor power level to remain above the capacity of the bypass valves. Therefore, reactor pressure increases until the relief valves open. The feed pumps are able to maintain level but the steam going through the relief valves to the torus does not return to the condenser to be reinjected to the core. Thus, condensate storage tank level decreases until the condensate and feed systems trip. At this point, reactor water level decreases until the MSIVs close making the PCS unavailable. Level continues to drop until core uncovery occurs and a core melt ensues. The initial value for this sequence is 3.7×10^{-6}, based on an initiating frequency of 1.68 per reactor-year and 2.2×10^{-6} for the unavailability of the combination of B and M. The sequence is shown on the systemic event tree, Figure C-31. Figure C-32 is a sequence evaluation diagram showing the dominant contributors to the mitigating systems unavailability.

The dominant contributor to failure of both the reactor scram system and the RPT is failure of the reactor protection system to initiate either one. This value was taken to be 1.9×10^{-6} from the WASH-1400 report, since no analysis of the reactor protection system was done for the present report.

The potential recovery actions for this sequence involve manually scramming the reactor or operator trip of the recirculation pumps. The time available to do either of these is a function of the reactor power level. Since the reactor power/bypass valve mismatch could be as high a 70% of full power, the time available for operator action would be minimal. Therefore, no credit is taken for operator action to prevent a core melt for this sequence. The final sequence value is then 3.7×10^{-6}.

4.3 Dominant Sequences

Those sequences from Table C-9 that have final sequence frequencies greater than 1.0×10^{-6} per reactor-year are the dominant sequences. There are eight dominant accident sequences. Six of these are transient sequences, while the other two are transient-induced LOCAs. Table C-14 lists these sequences in decreasing order of frequency.

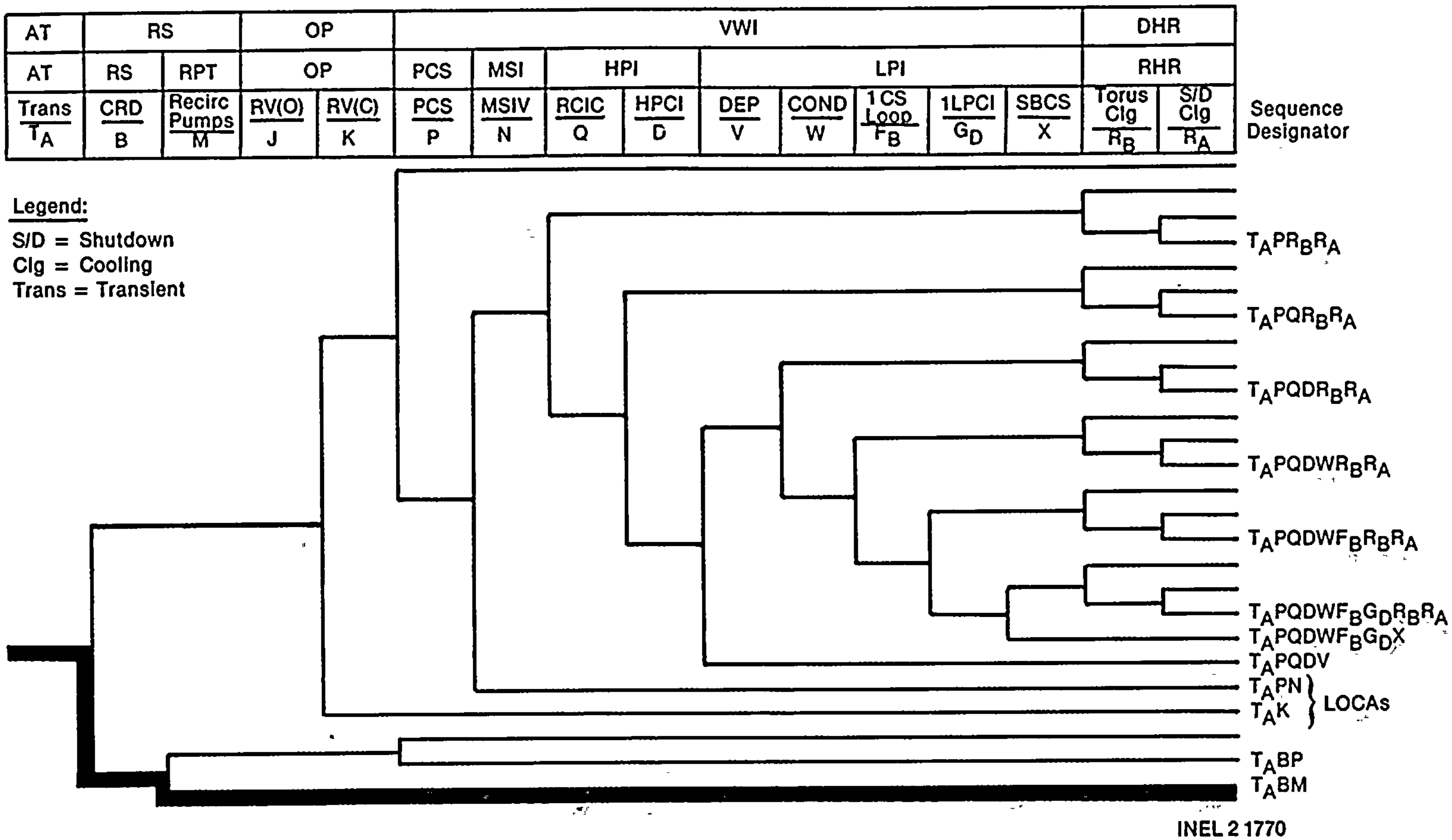


Figure C-31. Systemic event tree showing the T_ABM sequence.

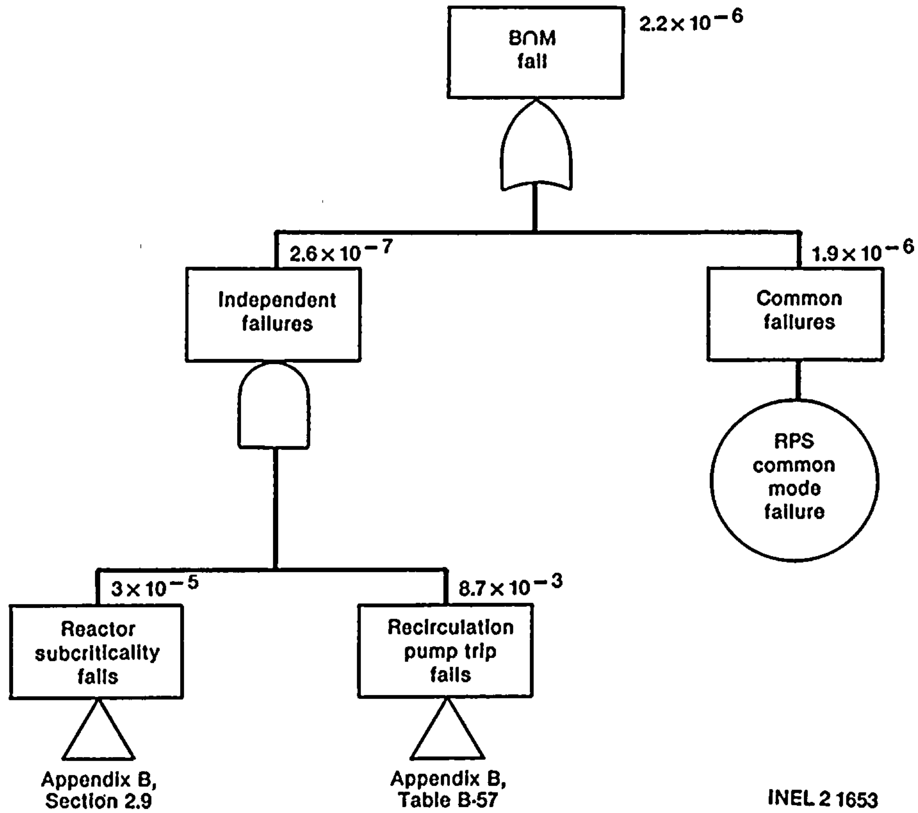


Figure C-32. Dominant contributors to the unavailability of the CRD and RPT systems following a transient where the PCS is available.

TABLE C-14. DOMINANT SEQUENCES

Sequence Initiator	Sequence Designator	Frequency
Transient without PCS	$T_UR_BR_A$	9.7×10^{-5}
Transient with PCS	T_UB	5.1×10^{-5}
Loss of offsite power	$T_PR_BR_A$	2.8×10^{-5}
Transient-induced LOCAs	TKR_BR_A	9.3×10^{-6}
Transient without PCS	$T_UQR_BR_A$	4.1×10^{-6}
Transient with PCS	T_ABM	3.7×10^{-6}
LOSP-induced LOCAs	$T_PKR_BR_A$	1.6×10^{-6}
Loss of offsite power	$T_PQR_BR_A$	1.2×10^{-6}

5. UNCERTAINTY ANALYSIS

5.1 Introduction

The point estimate of the frequency of each dominant sequence appears in Table C-14. In addition to knowing the point estimate of the frequency for these sequences, it is also useful to understand the uncertainty associated with each point estimate. The uncertainty analysis of this report only propagates errors associated with the given basic event failure rates and initiating event frequencies. Other uncertainties associated with quality assurance, success criteria, etc., are not included. The results of this analysis, therefore, only evaluate the uncertainty associated with the failure rate data base. The main purpose of the uncertainty analysis is to provide those using this report with additional perspective on the results. When evaluating potential design or operational changes, it may well be useful to examine changes to the uncertainty bounds as well as to the point-value estimates.

5.2 Methodology

The uncertainty bounds for each sequence were determined by assigning an uncertainty bound and a distribution to each basic event and sequence initiator. The MOCARS computer code[9] uses this information along with the cut sets for the systems to perform a Monte Carlo simulation that describes the resulting distribution for the systems. In much the same way as point estimates were obtained, the COMCAN code was used to identify cut sets common to two or more systems. These cut sets were also evaluated using the MOCARS code and appropriately combined to generate the distribution for the sequence.

The upper bound value was chosen to be the value of the sequence distribution at the 95% quantile. In other words, 95% of the distribution values generated by MOCARS were less than or equal to the upper bound value. Rather than expressing this upper bound as a fixed value, it is associated with the point estimate by an error factor that equals the upper bound divided by the point estimate.

5.3 Data Base

Table C-4 gives the error factors used for the basic event point estimates for most of the basic events. For human error probabilities and the generic control circuit models, an error factor of 10 was used. Most hardware failure data had error factors of three. Using an error factor of 10 for these two cases is, therefore, more conservative and puts the uncertainty for these events on the same level as short circuits, valve ruptures, and similar passive failures where the data base is sparse.

The lognormal distribution was chosen as the distribution for each basic event and for the initiating events. The lognormal distribution is commonly used in analyses where the uncertainty associated with the data is expressed in orders of magnitude differences from the point estimate.

The dominant sequence initiators are all transients or transient-induced LOCAs. The point estimates for these initiators came from EPRI

NP-801 as noted in Appendix A, Section 2.2. While EPRI NP-801 did assign 95% upper bound values on transient initiator frequencies, these bounds were assigned on a generic basis (i.e., BWRs, PWRs). Since the point estimates used for sequence frequency calculations were Browns Ferry-specific and EPRI NP-801 did not assign uncertainty bounds on a plant-specific basis, an error factor of three with a lognormal distribution was assigned for each initiator. Compared with the EPRI NP-801 generic data, an error factor of three is more conservative as is the assumption of a lognormal distribution.

5.4 Results

Table C-15 summarizes the results of the uncertainty analysis. Each dominant sequence is listed in descending order of the final frequency. Both the initial sequence frequency and error factor and the final sequence frequency (considering recovery) and error factor appear in the table. In addition, the sum of the dominant sequence frequencies and its associated error factor is shown. The error factor for the sum represents the result of a MOCARS evaluation of the sum of the dominant sequence distributions.

5.5 Insights on Uncertainty Analysis

In Section 4, the effect of control circuit faults on sequence frequencies involving failure of torus cooling and shutdown cooling is discussed. Table C-15 shows another aspect of the control circuit fault contribution. This contribution is to the uncertainty. Because of the assumption of an ĕrror factor of 10 for control circuit faults and their dominance in the point estimate of some sequences, control circuit faults

TABLE C-15. DOMINANT SEQUENCE UNCERTAINTIES

Sequence Designator	Initial Frequency	Error Factor	Final Frequency	Error Factor
$T_UR_BR_A$	1.3×10^{-4}	20.5	9.7×10^{-5}	8.7
T_UB	5.1×10^{-5}	5.0	5.1×10^{-5}	5.0
$T_PR_BR_A$	1.5×10^{-3}	5.6	2.8×10^{-5}	2.8
TKR_BR_A	1.2×10^{-5}	21.5	9.3×10^{-6}	9.0
$T_UQR_BR_A$	5.5×10^{-6}	36.3	4.1×10^{-6}	15.3
T_ABM	3.7×10^{-6}	4.6	3.7×10^{-6}	4.6
$T_PKR_BR_A$	8.3×10^{-5}	6.7	1.6×10^{-6}	2.8
$T_PQR_BR_A$	6.2×10^{-5}	10.7	1.2×10^{-6}	4.7
Total	1.9×10^{-3}	5.8	2.0×10^{-4}	5.6

are large contributors to the error factor associated with the initial frequency of these sequences. After adjusting for recovery, but keeping an error factor of 10, the error factor for the final frequency is reduced considerably. These control circuit faults were considered to be recoverable whenever there is enough time (a) to repair or bypass the control circuits, (b) to manually operate a valve, or (c) to valve in another pump, as is the case with torus cooling and shutdown cooling (long-term decay heat removal). Therefore, their contribution to the final sequence frequency was less than their contribution to the initial sequence frequency. Similarly, the final frequency error factor is less sensitive to control circuit faults than its corresponding initial frequency error factor.

Another example of this particular sensitivity is that the uncertainty for sequence $T_U R_B R_A$ initially is quite a bit higher than for sequence $T_P R_B R_A$. In the case of sequence $T_P R_B R_A$, the dominant faults were combinations of diesel generator faults (error factor of three) instead of control circuit faults (error factor of 10) as in sequence $T_U R_B R_A$.

Thus, it is apparent that the high error factors in some dominant sequence initial values are associated with the conservatism in the choice of the error factor of 10 for control circuit faults. Furthermore, when recovery is considered, the uncertainty diminishes by approximately a factor of two even when the conservative error factor of 10 is carried through.

To further demonstrate the conservative nature of the error factor used for control circuit faults, a MOCARS evaluation of the generic model using the data of Table C-4 was performed. This analysis provided an error factor of 2.1, which is considerably less than the assumed value of 10.

Another interesting insight comes from the sequence totals before and after recovery is considered. Even when control circuits are considered in their conservative case, the total core melt frequency error factor is only 5.8. After considering recovery, the error factor drops to 5.6. Thus, despite the fact that some sequences have relatively high error factors, their effect on the cumulative core melt frequency error factor is relatively modest. Furthermore, consideration of recovery actions reduces the cumulative frequency by approximately one order of magnitude while maintaining approximately the same error factor. This tends to indicate that the error factor for the cumulative core melt frequency is not significantly affected by recovery factors or by the wide error spread of a few sequences.

6. SENSITIVITY ANALYSIS

6.1 Introduction

After selection of the dominant sequences and evaluation of the uncertainties associated with each, it is important to examine the assumptions and uncertainties that went into the original values. A sensitivity analysis can aid in understanding the contributors to dominant sequence frequencies. The method of performing such an analysis is to identify potential uncertainties and recalculate the sequence frequencies to show how much variations in selected input parameters change the final value.

6.2 Scope of Analysis

Review of the dominant sequences revealed several areas where a sensitivity analysis would be desirable. These areas are summarized below.

1. The RHR trees assumed that failure of the minimum-flow bypass valves to close would disable the RHR loops. Since about 90% of the flow per loop would not be diverted by such a failure, what would be the effect on sequence frequency if such failures did not disable the RHR loops?

2. For the LOSP initiated sequences, failure of EECW was an important contributor to the sequence frequencies. The analysis assumed that three of four pumps were needed to supply adequate cooling. Since two of four pumps provides up to 91% of the necessary cooling, what change to the sequence frequency would occur if the EECW model were changed to require only two of four pumps for successful cooling?

3. The transient-induced LOCA initiator frequencies were derived from the transient systemic event trees using the WASH-1400 failure data for relief valves. What would be the change in these sequences if the generic stuck open relief valve frequency from EPRI NP-801 was used instead?

4. Unavailabilities for valve and pump control circuits were based on analysis of typical systems. A more detailed analysis of the corresponding systems would be possible. In particular, what would be the effect of modeling differences between AC- and DC-powered valve control circuits, and of modeling the effect of 4160 V rather than 480 V AC motor control circuits?

Other areas considered for sensitivity analysis include the usage of cross-connects between the three units at Browns Ferry in recovery actions for the dominant sequences. Cross-connects are described in Appendix B Section 1.2, but no credit was taken in the analysis for their use. While they do represent a potential resource for cooling the core, their components are tested less frequently than ECCS and operators must follow complicated, seldom-used procedures to bring them online. Their impact on recovery possibilities is thus judged to be minimal, and sensitivity studies to consider their effect were not performed.

The remaining sections describe the sensitivity analysis results for the four topics listed above.

6.3 Evaluation

In order to answer the questions previously noted, the fault trees or the initiator values were changed. The resulting sequence frequencies are presented for comparison.

6.3.1 Exclusion of Minimum-Flow Bypass Valves

Removal of minimum-flow bypass valve faults from the RHR fault trees reduces torus cooling unavailability from 3.1×10^{-3} to 1.7×10^{-3}. Shutdown cooling unavailability decreases from 2.0×10^{-2} to 1.0×10^{-2}. The commonalities between torus cooling and shutdown cooling are reduced to 2.4×10^{-6} when the bypass valves are removed, since the original values contained both support system and minimum-flow bypass valve faults. Therefore, the unavailability of torus cooling and shutdown cooling becomes 2.0×10^{-5}. This value is approximately 3.8 times less than the value obtained with the minimum-flow bypass valves considered in the RHR model.

Considering potential recovery further reduces the unavailability of torus cooling and shutdown cooling without the bypass valves. Of the 1.0×10^{-2} unavailability for shutdown cooling, approximately 2.3×10^{-3} represents nonrecoverable faults. The remaining 7.7×10^{-3} is potentially recoverable. Applying the recovery guidelines discussed previously in Section 3.3 produces a final unavailability for shutdown cooling of 2.4×10^{-3}. Of the torus cooling unavailability of 1.7×10^{-3}, approximately 1.1×10^{-3} is nonrecoverable. The remaining 6.0×10^{-4} is potentially recoverable. The resulting torus cooling unavailability is then 1.1×10^{-3}. The commonalities of torus cooling and shutdown cooling are also recoverable. Therefore, the resulting unavailability is 2.6×10^{-6}. This value is approximately 22 times lower than the unavailability after recovery with the bypass valves included.

Because the minimum-flow bypass valves are common to both the torus cooling and shutdown cooling fault trees, exclusion of these two valves reduces the prerecovery unavailability of the systems. Since many of the minimum-flow bypass valve faults were not recoverable, postrecovery unavailabilities are not affected as much when the valves remain in the tree (7.6×10^{-5} to 5.7×10^{-5}) as when they are removed (2.0×10^{-5} to 2.6×10^{-6}). This indicates that the torus cooling and shutdown cooling unavailabilities are sensitive to minimum-flow bypass valve faults, especially when recovery is considered.

Therefore, for those dominant accident sequences involving transients other than LOSP where shutdown and torus cooling fail (R_BR_A), the final sequence frequencies would be reduced approximately by a factor of 22 if faults associated with the minimum-flow bypass valves were not considered. For LOSP-initiated sequences, failure of R_BR_A is dominated by faults other than those associated with the bypass valves, and no change in sequence frequency would be realized.

6.3.2 Modification of EECW Success Criteria

As noted in the discussions of candidate dominant sequences, for LOSP initiators, the EECW system represents a common mode failure for all the AC systems. The success criteria for EECW in these sequences was three of four pumps operating. Since two of four pumps can provide up to 91% of the design flow requirements, it would be desirable to understand how those sequence frequencies would be affected if two of four pumps were sufficient.

Evaluation of the EECW system with a success criteria of two of four pumps under a LOSP condition reduces the unavailability from 2.0×10^{-2} to 2.3×10^{-3}. For the three LOSP initiated dominant sequences, this change would reduce the unavailability of torus cooling and shutdown cooling from 4.9×10^{-2} to 3.1×10^{-2}, thereby reducing the initial sequence frequency for these sequences by a factor of 1.6. Since the EECW contribution after recovery is considered negligible for these sequences (even with the original 2.0×10^{-2} value for three of four pumps), the final sequence frequency for these sequences would not be affected by the change in EECW success criteria to two of four pumps.

6.3.3 Transient-Induced SORV Initiator

The frequency of transient-induced stuck open relief valves in this analysis is based on the EPRI NP-801 frequencies for transients and the failure data for failure of the relief valves to reclose after a demand (see the treatment of System K in Appendix B, Section 2.6). It is desirable to investigate how using the EPRI NP-801 value for SORV frequency would change these sequence frequencies.

From the EPRI NP-801 data, the frequency of a SORV for BF1 is 0.95 per reactor-year compared to an average of 0.2 per reactor-year for General Electric (GE) plants. The transient event tree analysis for BF1 yielded a frequency of SORV initiators of 0.16 per reactor-year. Using the BF1-specific number would increase the sequence frequency of transient induced SORVs by a factor of 5.9. Using the GE average only increases the frequency of a factor by 1.25.

This information tends to indicate that the event tree frequency determination for SORVs matches well with the industry average data but not with the BF1-specific data. It should be noted that the three-stage relief valves originally installed at BF1 are being replaced by two-stage versions. Therefore, the previous plant-specific data for SORVs may now be unrepresentative of the current design. Also, the EPRI NP-801 data for BF1 was based on the first 37 months of operation. Accounting for subsequent operation may change the plant-specific frequency. In fact, EPRI document NP-2230[10] contains updated information and revisions to the original EPRI NP-801 data. This document reflects a much larger data base than EPRI NP-801, but the GE average value changes only from 0.20 to 0.21. The BF1-specific value is reduced to 0.05, and the average of BF-1, -2, and -3 is 0.31. In light of the GE average and updated BF1 specific data, the BF1 event tree determined frequency (0.16) seems to be reasonable.

The impact of this frequency on the overall BF1 core melt frequency estimate is insignificant, since SORV-initiated sequences contribute only 5% to the dominant sequence total and the final frequency estimate has a large error factor.

6.3.4 Use of Generic Control Circuit Unavailabilities

The generic control circuit analysis for valves in Appendix B, Section 5, is based on AC power supplies. The resulting unavailability estimates were also used for DC valve control circuits in the PRA. The effect of this assumption on dominant sequence frequencies was investigated by identifying the main differences between AC and DC valve control circuits and computing generic unavailability estimates for DC circuits. The details of this analysis are reported in Appendix B, Section 5.2. The result is that DC valve control circuit unavailability is 15% less than the corresponding AC unavailability with monthly testing and 19% less with quarterly testing.

Similarly, a generic 4160 V AC motor control circuit was analyzed to assess the difference in unavailability associated with the higher voltage system as opposed to the 480 V AC generic motor control circuit orginally used for all motor control circuits in the PRA. This analysis, documented in Appendix B, Section 5.3, shows no change in unavailability for the circuits with monthly testing. The unavailability with quarterly testing was 8.4×10^{-3} for the generic motor control circuit originally analyzed and 7.1×10^{-3} for a 4160 V AC circuit, which represents a drop in unavailability of 15%.

These results show that, for both the generic control circuits analyzed, the differences in power assumptions do not have a significant impact on system unavailabilities.

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