INL/EXT-11-22523
Rev. 1

A SCOPING STUDY:

Development of Probabilistic Risk Assessment Models for Reactivity Insertion Accidents During Shutdown In U.S. Commercial Light Water Reactors

S. Khericha

June 2011

The INL is a U.S. Department of Energy National Laboratory
operated by Battelle Energy Alliance

INL/EXT-11-22523
Rev. 1

A SCOPING STUDY:

Development of Probabilistic Risk Assessment Models for Reactivity Insertion Accidents During Shutdown In U.S. Commercial Light Water Reactors

S. Khericha

June 2011

Idaho National Laboratory
Idaho Falls, Idaho 83415

http://www.inl.gov

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

ABSTRACT

This report documents the scoping study of developing generic simplified fuel damage risk models for quantitative analysis from inadvertent reactivity insertion events during shutdown (SD) in light water pressurized and boiling water reactors. In the past, nuclear fuel reactivity accidents have been analyzed both mainly deterministically and probabilistically for at-power and SD operations of nuclear power plants (NPPs). Since then, many NPPs had power up-rates and longer refueling intervals, which resulted in fuel configurations that may potentially respond differently (in an undesirable way) to reactivity accidents. Also, as shown in a recent event, several inadvertent operator actions caused potential nuclear fuel reactivity insertion accident during SD operations. The set inadvertent operator actions are likely to be plant- and operation-state specific and could lead to accident sequences. This study is an outcome of the concern which arose after the inadvertent withdrawal of control rods at Dresden Unit 3 in 2008 due to operator actions in the plant inadvertently three control rods were withdrawn from the reactor without knowledge of the main control room operator. The purpose of this Standardized Plant Analysis Risk (SPAR) Model development project is to develop simplified SPAR Models that can be used by staff analysts to perform risk analyses of operating events and/or conditions occurring during SD operation. These types of accident scenarios are dominated by the operator actions, (e.g., misalignment of valves, failure to follow procedures and errors of commissions). Human error probabilities specific to this model were assessed using the methodology developed for SPAR model human error evaluations. The event trees, fault trees, basic event data and data sources for the model are provided in the report.

The end state is defined as the reactor becomes critical.

The scoping study includes a brief literature search/review of historical events, developments of a small set (8) of comprehensive event trees and fault trees and recommendation for future work.

Record of Revisions

Revision 0	May 2011	Original report issued by INL as a task in JCN N6660	S. Khericha
Revision 1	June 2011	Editorial revisions; also added Appendices A, B, and C	S. Sancaktar

CONTENTS

FIGURES

TABLES

A SCOPING STUDY:
Development of Probabilistic Risk Assessment Models for Reactivity Insertion Accidents During Shutdown In U.S. Commercial Light Water Reactors

1. INTRODUCTION

Generally, the analysis of accidents and release of radioactive material from nuclear reactors has been concentrated only during power operation. During full power operation, the "defense-in-depth" concept is used to ensure prevention of fission product release by the multiple safety systems and barriers. These are designed into the plant to the control rod scram and emergency water injection systems, as well as to containment physical barriers (fuel cladding, primary piping, and primary and secondary containments). In recent years, the Idaho National Laboratory has been developing the risk models for the Nuclear Regulatory Commission (NRC) during the SD to better understand and manage the risk. The consequences of reactivity insertion accidents such as control rod ejection events at power are generally evaluated deterministically and appropriate safeguards are developed to mitigate or reduce the consequences.

In the past, nuclear fuel reactivity accidents have been analyzed both deterministically and probabilistically for at-power and SD operations of NPPs. Since then, many NPPs had power up-rates and longer refueling intervals, which resulted in fuel configurations that may potentially respond differently (in an undesirable way) to reactivity accidents. Also, as shown in a recent event at Dresden 3, operator actions caused inadvertent withdrawal of three control rods from the reactor. The potential for inadvertent criticalities of BWRs during SD received increased attention from both the industry and the NRC in the aftermath of the Chernobyl accident. For SD and refueling conditions, both the events of interest and the means of protection are very different from those when at power. Due to the low amount of energy stored in the fuel and the relatively low rate of reactivity insertion, a large early-release type of event that threatens the containment integrity is not a concern. Note that a rod-ejection type of event which results in rapid reactivity insertion are not considered as these types of events are normally analyzed as part of the required safety analysis. However, reactivity controls are still important. During refueling, some of the normal barriers such as open containment to fission product release are removed to gain access to the core. The drywell and reactor pressure vessel are open to the containment and personnel are on the refueling floor above the reactor. Additionally, some of the automatic protection systems may not be available during refueling. In this situation, good engineering practices are relied upon to supplement the available automatic systems. The desired safety standards are achieved by using combinations of procedures and automatic systems to ensure that the reactor stays subcritical during all phases of refueling operations. Various approaches are used to meet this requirement, with administrative controls (written procedures) playing an important role in helping to maintain acceptable levels of risk.

The accident at Chernobyl was a reactivity insertion event. The Chernobyl reactor had a positive void reactivity coefficient. U.S. LWRs have a different design which has a negative void coefficient and a Chernobyl-type of event cannot occur. However, due to inadvertent withdrawal of control rods or incorrect manipulation of fuel, there is a potential that localized prompt-criticality or criticality can occur. It is very likely that the barriers are removed and the presence of personnel on the refueling floor could result in personnel receiving a significant radiation dose.

This report documents the scoping study of developing generic simplified fuel damage risk models for quantitative analysis from inadvertent reactivity insertion events during SD in LWPs and BWRs. The study is an outcome of the concern which arose after the inadvertent partial withdrawal of three control rods at Dresden Unit 3 in 2008 and an event in the Japanese BWR reactor Shika 1 in 1997 which was revealed in 2007. In the Dresden event, no emergency response was required and operators terminated the event and secured the reactor. However, in the Shika-1 event, the reactor became critical due to the unexpected withdrawal of three control rods. The purpose of this Standardized Plant Analysis Risk (SPAR) model development project is to develop simplified SPAR models that can be used by staff analysts to perform risk analyses of operating events and/or conditions occurring during SD operation. These types of accident scenarios are dominated by the operator actions as initiators by misalignment of valves, failure to follow procedures, and errors of commissions.

The certain inadvertent or unplanned reactivity insertion events are analyzed for the required safety analysis for LWR. The following table lists the events which are analyzed as part of the safety analysis and were not considered as part of this scoping study.

Events Causing an Increase in Reactivity that are Part of a Required Safety Analysis*

PWR

- Increase in heat removal by the secondary system
- Uncontrolled control assembly bank withdrawal
- Startup of an inactive reactor coolant pump in an idle loop
- Inadvertent boron dilution of the core at power
- Single control rod withdrawal at power
- Steamline break
- Control rod ejection

BWR

- Overpressurization due to closure of steamline valves
- Core coolant temperature reductions at power
- Control rod withdrawal error at power
- Increase in core coolant flow rate at power
- Free-fall of control rod (rod drop event)

* NUREG-5368.

The objective of this scoping study is to develop SPAR probabilistic risk assessment (PRA) models for the analysis of reactivity accidents during SD and prepare a draft report. The past studies on the reactivity insertion accidents were reviewed and a brief License Events Report (LER) search was conducted to identify the events which resulted in positive reactivity insertion during SD operation. Not all control rod(s) withdrawal, boron dilution or other positive reactivity insertion (e.g.; misfuel loading) events would necessarily result in criticality or a technical specification violation; therefore, such events would not be reported. Therefore, it is not easy to determine what fraction of positive reactivity insertion events would result in the delayed or prompt criticality and it strictly depends on several parameters such as reactor configuration, fresh fuel versus irradiated fuel, RCS temperature. This would require thermal-hydraulic and/or neutronic analysis which is beyond the scope of this study.

2. DATA ANALYSIS

This section discusses the LER search conducted for reported events occurring between 1980 and the present. In the U.S., nuclear reactors shut down to refuel once every 18 to 24 months. In the 1980s and early 1990s the average refueling outage lasted approximately three months. Over the last ten years, the refueling outage period has been reduced significantly to about a month. Total numbers of PWR and BWR plants refueling outages from 2000 to present are listed in Tables 2.1 and 2.2. The PWR plants experienced a total of 446 refueling outages, for an average of 0.62 refueling outages per year. The PWR plants were in the refueling mode on the average 41 days (984 hours). The BWR plants experienced a total of 238 refueling outages, for an average of 0.54 refueling outages per year. The BWR plants were in the refueling mode on the average 32 days (768 hours). Therefore, for the purpose of this analysis, the mean refueling outage duration for PWRs is assumed to be 984 hours and for BWRs is assumed to be 768 hours.

For PWR reactivity insertion events, a search was conducted for four different time periods using the phrase “Boron AND Dilution AND Shutdown”. The data were collected from 1980 to present, 1990 to present, 1995 to present and 2000 to present and provided the trending of the data. The results are provided in Table 2.3. The number of reportable events has decreased significantly in the last ten years.

Similarly, the LER search on “Control Rod Drift” resulted in only four hits. No LER was found on fuel misloading in the core during refueling. However, two fuel misloading events in the spent fuel pool are recorded. The event descriptions are provided in Appendix A.

The review of the data indicated that only those boron concentration reduction events which met the 10.CFR 50.73 criteria or resulted in a condition which is prohibited by the technical specification limit for that operating mode are reported. Similarly, in the case of BWRs only those control rods withdrawal events which met the 10.CFR 50.73 criteria or resulted in a condition which is prohibited by the technical specification limit for that operating mode are reported. This made it difficult to evaluate how many total inadvertent actual boron dilution events or total inadvertent control rods withdrawal events took place. Further detailed review of the events from 2000 to 2010 showed that only four events are related to actual boron dilution events and one inadvertent control rod withdrawal event.

Appendix A lists the some of the reactivity insertion events of interest for U.S. plants. Table C-4 lists Japanese nuclear industry reactivity insertion experiences.

Table 2.1. Refueling Outages in BWR Plants Since 2000

PL_NAME	AVG Refueling Outage Days	Number Refueling Outage Since 2000
BROWNS FERRY 1	34	2
BROWNS FERRY 2	36	5
BROWNS FERRY 3	30	6
BRUNSWICK 1	37	6
BRUNSWICK 2	37	5
CLINTON 1	25	6
COLUMBIA	41	5
COOK 1	38	6
COOK 2	44	7
COOPER STATION	40	7
DRESDEN 2	22	5
DRESDEN 3	22	6
DUANE ARNOLD	35	6
FERMI 2	37	8
FITZPATRICK	28	6
GINNA	25	7
GRAND GULF	26	7
HATCH 1	34	6
HATCH 2	46	6
HOPE CREEK	34	8
LASALLE 1	25	5
LASALLE 2	24	6
LIMERICK 1	19	6
LIMERICK 2	18	5
MONTICELLO	25	6
NINE MILE PT. 1	24	5
NINE MILE PT. 2	24	6
OYSTER CREEK	54	6
PEACH BOTTOM 2	27	6
PEACH BOTTOM 3	22	5
PERRY	25	5
PILGRIM	29	5
QUAD CITIES 1	43	5
QUAD CITIES 2	47	6
RIVER BEND	46	8
ST. LUCIE 1	37	7
ST. LUCIE 2	21	7
SUSQUEHANNA 1	42	6
SUSQUEHANNA 2	33	5
VERMONT YANKEE	32	8
	BWR AVG = 32	Total = 238

Table 2.2. Refueling Outages in PWR Plants Since 2000

PL_NAME	AVG Refueling Outage Days	Number Refueling Outage Since 2000
ARKANSAS 1	36	7
ARKANSAS 2	34	8
BEAVER VALLEY 1	40	8
BEAVER VALLEY 2	32	7
BRAIDWOOD 1	21	8
BRAIDWOOD 2	19	7
BYRON 1	23	7
BYRON 2	20	7
CALLAWAY	46	7
CALVERT CLIFFS 1	49	6
CALVERT CLIFFS 2	38	6
CATAWBA 1	37	7
CATAWBA 2	37	8
COMANCHE PEAK 1	32	7
COMANCHE PEAK 2	26	7
CRYSTAL RIVER 3	32	4
DAVIS-BESSE	200	5
DIABLO CANYON 1	40	7
DIABLO CANYON 2	44	6
FARLEY 1	45	8
FARLEY 2	43	7
FORT CALHOUN	57	7
HARRIS	36	8
INDIAN POINT 2	30	5
INDIAN POINT 3	27	5
KEWAUNEE	43	8
MCGUIRE 1	38	9
MCGUIRE 2	39	7
MILLSTONE 2	41	7
MILLSTONE 3	37	7
NORTH ANNA 1	32	8
NORTH ANNA 2	44	8
OCONEE 1	54	7
OCONEE 2	44	7
OCONEE 3	42	8
PALISADES	39	7
PALO VERDE 1	48	7
PALO VERDE 2	50	7
PALO VERDE 3	50	8
POINT BEACH 1	43	7
POINT BEACH 2	50	7
PRAIRIE ISLAND 1	46	6
PRAIRIE ISLAND 2	37	7
ROBINSON 2	46	7
SALEM 1	32	7
SALEM 2	38	7

PL_NAME	AVG Refueling Outage Days	Number Refueling Outage Since 2000
SAN ONOFRE 2	78	6
SAN ONOFRE 3	68	6
SEABROOK	41	7
SEQUOYAH 1	40	8
SEQUOYAH 2	30	7
SOUTH TEXAS 1	38	7
SOUTH TEXAS 2	35	7
SUMMER	58	7
SURRY 1	36	8
SURRY 2	33	7
THREE MILE ISL 1	49	5
TURKEY POINT 3	39	8
TURKEY POINT 4	35	7
VOGTLE 1	33	7
VOGTLE 2	33	7
WATERFORD 3	36	7
WATTS BAR 1	40	7
WOLF CREEK	41	7
	PWR AVG = 41	Total = 446

Table 2.3. Results of LER Search on “Boron AND Dilution AND Shutdown”

Period	Operating Experience (Yrs)	Number of Events	Avg. Number of Events/Year
1980 to April 2011	2500	176	0.070
1990 to April 2011	2100	57	0.027
1995 to April 2011	1500	28	0.019
2000 to April 2011	1000	10	0.010

3. EVENT TREE MODELS

This section presents the SPAR model event trees. Event trees are provided for diverse positive reactivity insertion scenarios for PWRs and BWRS during SD. The positive reactivity insertion events during low-power physics tests are not included.

There have been numerous events in U.S. and foreign reactors related to positive reactivity insertion during SD. There are multiple ways positive reactivity can be inserted in the reactor during SD, some of which are discussed in the following paragraphs.

A review of PWR and BWR operating history (LERs) and the reactivity insertion accident analysis reports were reviewed to identify the positive reactivity insertion events[1,2,3]. No attempt was made to identify additional paths or processes which would lead to positive reactivity insertion on the basis of a plant's detailed system diagrams or procedures. This is beyond the scope of this scoping study.

The general approach in developing the event tree models was to first identify the failure modes or the reactor configuration which leads to positive reactivity insertion and consequently identify the systems or operator actions which can be used to mitigate further undesired consequences. Each initiating event starts with frequency of being in refueling mode per year. For PWRs, an initiating event is defined as IE-PWR-REF-FREQ, and for BWRs, an initiating event is IE-BWR-REF-FREQ.

The end state is defined as reactor achieves re-criticality (RXC).

3.1 Malfunction in Chemical and Volume Control System

Event Tree – Boron Dilution Due to Malfunction in CVCS (LOVCT)

This event tree represents a boron dilution event resulting from a malfunction in the CVCS due to random equipment failure, operator action of misalignment of valves or failure to maintain proper boron concentration in the VCT. The CVCS system is designed to maintain the boron concentration at a desired level. The classic safety analysis boron dilution event postulates opening of the primary water makeup control valve and either a controller or mechanical failure of the blend system. The reduction in core boron concentration results in an increase in neutron count rate as indicated on the source range monitors. Since the charging flow (makeup) rate is only in few gpm, it is not expected that a reactor will become prompt-critical in a short period of time. This event also includes inadvertent dilution below the desired limit while reducing the born concentration in the RCS for the Mode change.

IE-PWR-CVCS: This event tree represents reactivity insertion as a result of insertion of diluted boron water via CVCS system. Initiating event frequency represents PWR shutdown frequency for refueling yearly.

VCT-DIL: This event represents the conditions during refueling outages that boron concentration in the VCT tank is reduced below the technical specification limit. The potential boron dilution can occur during the makeup of the VCT inventory if the operator misaligns valves; the operator inadvertently injects unborated water in the reactor; or the RCP seal cooling heat exchanger is leaking. This event also includes inadvertent over-dilution below the technical specification limit while reducing the born concentration in the RCS for the Mode change.

Three events related to inadvertent boron dilution exceeding the desired limit has been reported since 2000. Assuming one boron dilution activity per refueling, the probability of boron dilution is estimated to be 6.7E-3 (3/446).

VCT-REC: This top event represents that an operator fails to recognize the diluted boron concentration in the VCT or the diluted boron water is injected via the CVCS system into the core. The VCT level and boron concentration are monitored in the main control room.

RCS-FLOW-PWR: The second top event represent the RCS flow is available, i.e., the RHR pump is running. The diluted boron water compared to PCS boron concentration enters into the primary system via the charging system. Since the RHR pump is running, flow in the RCS is sufficient to create uniform mixing of the coolant throughout the system. As a result boron concentration will be reduced. The reduction in boron concentration results in reduction of the subcriticality margin. The reduction of the subcriticality margin depends on how much boron concentration is reduced. As the power increases, the reactor coolant temperature will also increase. Since most U.S. LWRs are designed to have a negative moderator temperature coefficient, the positive reactivity insertion from boron dilution will be somewhat negated. The failure probability of loss of RHR is used from SD PRA, 1.1E-06/hour. Therefore, unavailability of RHR during refueling mode is calculated to be 5.4E-04 using 492 hours (outage duration 984/2).

SRM-PWR: This is a third top event on the event trees. The source range monitors (SRM) monitor the neutron counts. The neutron counts will increase as the boron dilution is progressing. As the source range count rate increases, numerous indications and alarms will alert the operator that reactor power is increasing or the subcriticality margin is decreasing. Also the status of the CVCS system is readily available in the MCR.

VCT-ISO: This top event represents termination of boron dilution. Once the operator diagnoses the problem, he can restore the boron concentration and terminate the transient. It is assumed that no formal procedures (incomplete) or training exist to recover from this event.

RX-CRIT-PWR: This top event represents if this specific configuration is vulnerable to achieving re-criticality. Steady state evaluation of neutronic behavior of this specific configuration needs to be performed to determine that this scenario could lead to a critical condition. The reactor achieving criticality is dependent on the flow rate, rate of boron dilution, duration of dilution, reactor fuel configuration, fresh fuel, and RCS temperature, etc. To determine if the reactor achieves the criticality would require thermal-hydraulic and neutronic analysis which is beyond the scope of this study. No criticality event as a result of boron dilution has been reported.

However, three boron dilution events have been reported and none of them resulted in criticality. No criticality event as a result of RWST boron dilution has been reported. Using Bayesian update for non-informative prior, the probability of criticality would be 0.25.

PROMPT-CRITICAL: This top event defines that if the reactor achieves re-criticality, what fraction of events would result in prompt-critical. No criticality event as a result of boron dilution has been reported. Therefore, for the purpose of this analysis, it is assumed that for every ten criticality events one event results in prompt-criticality.

3.2 Misalignment/Failure of the Valves in RWST

Event Tree – Boron Dilution Due to Malfunction in RWST (LORWST)

This event tree represents boron dilution event due to a malfunction in HPI/LPI injection lines from RWST either due to random equipment failure, operator action of misalignment of valves or the gross leakage through the valves. During refueling the reactor is depressurized. Under this circumstance the safety injection flow rate can be appreciable if one of the valves in the safety injection line fails to open. The reduction in core boron concentration results in an increase in neutron count rate as indicated on the source range monitors.

IE-PWR-RWST: This event tree represents reactivity insertion as a result of insertion of diluted boron water via the RWST system. Initiating event frequency represents PWR SD frequency for refueling yearly.

BORON-DIL-RWST: This event represents the condition during refueling outage where boron concentration in the RWST is reduced below the technical specification limit. A simplified fault tree is developed, identifying failure modes which can lead to this condition during refueling outage, due to the unavailability of detailed system drawings and procedures.

RWST-REC: This top event represents that an operator fails to recognize the diluted boron concentration in the RWST.

RWST-MOV: This event represents that one of the valves in the RCS injection line from the RWST fails to open while the reactor is depressurized or an operator inadvertently opens the valve during maintenance. The RWST boron concentration is monitored in the main control room. Any substantial amount of in-leakage from the accumulators to the reactor coolant system should be detected if the sensors and/or alarms are operable.

RCS-FLOW: A second top event represents the RCS flow is available, i.e., the RHR pump is running. The diluted boron water compared to PCS boron concentration enters into the primary system via a charging system. Since the RHR pump is running, flow in the RCS is sufficient to achieve uniform mixing of the coolant throughout the system. As a result, boron concentration will be reduced. The reduction in boron concentration results in a reduction of the subcriticality margin. The reduction in the subcriticality margin depends on how much boron concentration is reduced. As the power increases, the reactor coolant temperature will also increase. Since most U.S. LWRs are designed to have a negative moderator temperature coefficient, the positive reactivity insertion from boron dilution will be somewhat negated.

SRM-PWR: This is a third top event on the event trees. The source range monitors (SRM) monitor the neutron counts. The neutron counts will increase as the boron dilution is progressing. As the source range count rate increases, numerous indications and alarms will alert the operator that reactor power is increasing or the subcriticality margin is decreasing. Also the status of the CVCS system is readily available in MCR.

RWST-ISO: This top event represents termination of boron dilution. Once the operator diagnoses the problem, he can restore the boron concentration and terminate the transient.

RX-CRIT-PWR: This top event represents if this specific configuration is vulnerable to achieving re-criticality. Steady state evaluation of neutronic behavior of this specific configuration needs to be performed to determine that this scenario could lead to a critical condition. The reactor achieving criticality is dependent on the flow rate, rate of boron dilution, duration of dilution, reactor fuel configuration, fresh fuel versus irradiated fuel, RCS temperature etc. To determine if the reactor achieves the criticality would require thermal-hydraulic and neutronic analysis which is beyond the scope of this study. No criticality event as a result of RWST boron dilution has been reported. However, three boron dilution events have been reported and none of them resulted in criticality. Using a Bayesian update for non-informative prior, the probability of criticality would be 0.25.

PROMPT-CRITICAL: This top event defines that if the reactor achieves re-criticality, what fraction of events would result in prompt-critical. No criticality event as a result of boron dilution has been reported. Therefore, for the purpose of this analysis, it is assumed that for every ten criticality events one event results in prompt-criticality.

3.3 Failure of Accumulator Injection Valve (Accumulator Diluted)

Event Tree: Loss of Accumulator MOVs Integrity

This event tree represents a boron dilution event due to a malfunction in accumulator injection lines from accumulator tanks either due to random equipment failure, operator action of misalignment of valves or the gross leakage through the valves. In a typical four loops PWR, there are four accumulators each attached to cold leg. Typically, accumulators are pressurized above 600 psig. During the loss of coolant accident events (medium or large LOCA), the borated water from the accumulators would flow to the RCS when RCS pressure drops below the accumulator pressure. During normal power operation each accumulator is isolated from the reactor coolant system by two check valves in a series. During SD before the reactor is depressurized, the accumulator is isolated from the reactor coolant system by the MOV located between the two check valves. Typically during SD the accumulators are isolated and assumed to have normal operating pressure and boron concentration within the technical specification limit.

IE-PWR-ACC: This event tree represents reactivity insertion as a result of insertion of diluted boron water via the accumulator system. Initiating event frequency represents PWR SD frequency for refueling yearly.

ACC-DIL: This event represents the condition during refueling outage where boron concentration in the accumulator is reduced below the technical specification limit after the test and maintenance. A simplified fault tree is developed, identifying failure modes which can lead to this condition during refueling outage, due to the unavailability of detailed system drawings and procedures.

ACC-REC: This top event represents that an operator fails to recognize the diluted boron concentration in the accumulator.

ACC-MOV: This top event represents that one of the valves in the RCS injection line from the accumulator fails to open while the reactor is depressurized or an operator inadvertently opens the valve during maintenance. The accumulator level and pressure are monitored in the main control room. Any substantial amount of in-leakage from the accumulators to the RCS should be detected if the sensors and/or alarms are operable.

RCS-FLOW-PWR: This top event represent the RCS flow is available, i.e., the RHR pump is running. The diluted boron water compared to PCS boron concentration enters into the primary system via charging system. Since the RHR pump is running, flow in the RCS is sufficient to achieve uniform mixing of the coolant throughout the system. As a result, boron concentration will be reduced. The reduction in boron concentration results in a reduction of the subcriticality margin. The reduction in subcriticality margin depends on how much boron concentration is reduced. As the power increases, the reactor coolant temperature will also increase. Since most U.S. LWRs are designed to have a negative moderator temperature coefficient, the positive reactivity insertion from boron dilution will be somewhat negated.

SRM-PWR: This is a third top event on the event trees. The SRMs monitor the neutron counts. The neutron counts will increase as the boron dilution is progressing. As the source range count rate increases, numerous indications and alarms will alert the operator that reactor power is increasing or the subcriticality margin is decreasing. Also the status of the CVCS system is readily available in MCR. A simplified fault tree was developed for this top event.

RWST-ISO: This top event represents termination of boron dilution. Once the operator diagnoses the problem, he can restore the boron concentration and terminate the transient.

RX-CRIT-PWR: This top event represents if this specific configuration is vulnerable to achieving re-criticality. Steady state evaluation of neutronic behavior of this specific configuration needs to be performed to determine if that scenario could lead to a critical condition. The reactor achieving criticality is dependent on the flow rate, rate of boron dilution, duration of dilution, reactor fuel configuration, fresh fuel versus irradiated fuel, RCS temperature etc. To determine if the reactor achieves the criticality would require thermal-hydraulic and neutronic analysis which is beyond the scope of this study. No criticality event as a result of accumulator boron dilution has been reported. However, three boron dilution events have been reported and none of them resulted in criticality. Using a Bayesian update for non-informative prior, the probability of criticality would be 0.25. This top event also defines that if the reactor achieves re-criticality, what fraction of events would result in prompt-critical.

3.4 Unborated Water Injection in the RCS During Steam Generator Maintenance

Event Tree: Loss of Coolant from SG to Primary (LOSG)

Several RCS dilution events have resulted from equipment failures and/or human errors associated with steam generator (SG) maintenance. In this scenario, operator action results in the leakage of secondary unborated cooling in the primary piping. It is assumed that unborated water would accumulate in the primary piping until the RCS pump is started and enters into the core region diluting the boron concentration, consequently inserting positive reactivity into the core. This is a postulated event. No event has been reported. NUREG/CR-2798 cited two events which involved secondary water leaking to the primary due to operator actions. Both events resulted in the reduction of boron concentration in the RCS.

IE-PWR-SG: This event tree represents reactivity insertion as a result of insertion of unborated water from the SG (secondary) system. Initiating event frequency represents PWR SD frequency for refueling yearly.

SG-LEAK: This top event defines inadvertent leakage of secondary water to the primary side during testing and maintenance of the SG. It is assumed that SG maintenance is performed every-other outage. The LER search from 2000 until present resulted in no SG leak events to RCS. However, prior to 2000, _____ such events have been reported. NUREG/CR-2798 cited two events which involved secondary water leaking to the primary due to operator actions. Therefore, the probability of this event using a Bayesian update is estimated to be 1.4E-02.

SG-ISO: This event represents an operator recognizes and isolates the SG leak.

RCS-FLOW-SG: This top event represent that the RCS flow is available, i.e., the RHR pump is running. The diluted boron water compared to PCS boron concentration enters into the primary system via a charging system. Since the RHR pump is running, flow in the RCS is sufficient to achieve uniform mixing of the coolant throughout the system. As a result, boron concentration will be reduced. The reduction in boron concentration results in reduction of subcriticality margin. The reduction in the subcriticality margin depends on how much boron concentration is reduced. As the power increases, the reactor coolant temperature will also increase. Since most U.S. LWRs are designed to have a negative moderator temperature coefficient, the positive reactivity insertion from boron dilution will be somewhat negated.

It is not clear whether the unborated water can accumulate in the primary piping while the RHR system is operating. Due to a lack of detailed isometric system drawings and procedures, for the purpose of this analysis it is assumed that there is fifty-fifty chance that unborated water will accumulate in the primary piping while the RHR is running. A simplified fault tree was developed for this top event at the system level.

SRM-PWR: This is a second top event on the event trees. The SRMs monitor the neutron counts. The neutron counts will increase as the boron dilution is progressing. As the source range count rate increases, numerous indications and alarms will alert the operator that reactor power is increasing or the subcriticality margin is decreasing. Also the status of the CVCS system is readily available in MCR. A simplified fault tree was developed for this top event at the system level.

OEP-1: This top event represents termination of boron dilution. Once the operator diagnoses the problem, he can restore the boron concentration and terminate the transient.

RX-CRIT-SG: This top event represents if this specific configuration is vulnerable to achieving re-criticality. Steady state evaluation of neutronic behavior of this specific configuration needs to be performed to determine if that scenario could lead to a critical condition. The reactor achieving criticality is dependent on the flow rate, rate of boron dilution, duration of dilution, reactor fuel configuration, fresh fuel versus irradiated fuel, RCS temperature etc. To determine if the reactor achieves the criticality would require thermal-hydraulic and neutronic analysis which is beyond the scope of this study. No criticality event as a result of boron dilution from steam generator leakage has been reported. This top event also defines that if the reactor achieves recriticality, what fraction of events would result in prompt-critical.

3.5 Unplanned Control Rod Withdrawal from Loaded Fuel Cell or Fuel Loading In Uncontrolled Cell

Event Tree: One CRD Removed for TM from Loaded Fuel Cell (CRD1OUT)

In the last decade, the refueling outage period has been reduced from an average of three months to one month due to efficient outage management and availability of advanced management risk evaluation tools. With an increased emphasis on minimizing outage times, there is a strong incentive to perform refueling activities in parallel. Sometimes these parallel activities require bypassing or altering the safety or protective system, e.g., the use of temporary circuit alterations ("jumpers" or bypass) to allow operation of equipment under conditions not considered during the original design or development of safety procedures. Typically, some form of administrative controls are applied during the time that interlocks are jumpered to bypass interlock function and to perform parallel activities that may impact the ability of administrative controls to provide compensating safety function. This scenario describes when the control rod is withdrawn from the fuel cell a second control rod is withdrawn from this "uncontrolled loaded fuel cell". This will often require bypassing the position indication for the CRD to allow movement of other control rods. The withdrawal of the second control rod from a full fuel loaded cell could lead to a critical or prompt-critical condition. Steady state evaluation of neutronic behavior of a specific configuration needs to be performed to determine that if scenario could lead to critical or prompt-critical condition.

IE-BWR-CRD1OUT: This event tree represents reactivity insertion during testing and maintenance on one CRD. Initiating event frequency represents BWR SD frequency for refueling yearly.

CRD-TM: This is an initiating event. This event describes the vulnerable configuration created during an outage. It is common practice during refueling to perform maintenance on a fraction of the CRD while the fuel cell is loaded. First, a control rod is withdrawn from the fuel cell that is loaded without bypassing the interlock system. An additional control rod cannot be withdrawn from this fuel cell as the interlock will prevent it. This is an accepted practice. It is assumed that CRD maintenance is performed every-other outage.

CRD-TWO: This top event represents control rod withdrawal from the uncontrolled loaded fuel cell. More than one control rod cannot be withdrawn from the loaded fuel cell without bypassing the interlock or mechanical failure of the interlock. However, as discussed earlier, to minimize the outage period, it is common practice to perform parallel activities by installing jumpers to bypass the interlock. This requires communication between field operation and the main control room as no movement of a CRD is allowed without knowledge of the main control room.

SRM-BWR: This is a third top event on the event trees. The neutron counts will increase as the control rod is moved out of the core. As the source range count rate increases, numerous indications and alarms will alert the operator that reactor power is increasing or the subcriticality margin is decreasing. Also the status of the CRD system is readily available in MCR. A simplified fault tree was developed for this top event at the system level.

RX-CRIT-BWR: This top event represents if this specific configuration is vulnerable to critical or prompt-critical. Steady state evaluation of neutronic behavior of this specific configuration needs to be performed to determine if that scenario could lead to a critical condition. The reactor achieving recriticality is dependent on the flow rate, reactor core configuration, fresh fuel versus irradiated fuel, and RCS temperature, etc. To determine if the reactor achieves the criticality would require thermal-hydraulic and neutronic analysis which is beyond the scope of this study. Control rod insertion events during testing and maintenance have been reported; however, no criticality event has been reported during CRD testing and maintenance in U.S. BWRs. However, Japanese BWRs have reported several criticality events during testing and maintenance prior to 2000 (see Appendix A). This would require thermal-hydraulic and/or neutronic analysis which is beyond the scope of this study. This top event also defines that if the reactor achieves recriticality, what fraction of events would result in prompt-critical.

3.6 Unplanned Fuel Loading In Uncontrolled Fuel Cell (NOLFC)

Event Tree: Two or More CRD Removed from Unloaded Fuel Cell

This event tree is parallel to the previous event tree except it is assumed that fuel and two adjacent control rods from the cells are removed. A control rod can be withdrawn during refueling from a cell that may contain fuel. To perform the maintenance on additional control rod(s), the interlock function must be defeated. This scenario describes inadvertent loading of fuel while the two adjacent control rods are withdrawn from the fuel cell. The loading of fuel in the cell could lead to a critical or prompt-critical condition. Steady state evaluation of neutronic behavior of this specific configuration needs to be performed determine if that scenario could lead to a critical or prompt-critical condition.

IE-BWR-NOLFC: This event tree represents reactivity insertion during loading uncontrolled fuel cell. Initiating event frequency represents BWR shutdown frequency for refueling yearly.

CRD-TWO: This is an initiating event. This event describes the vulnerable configuration created during an outage. It is common practice during refueling to remove more than one CRD while fuel is removed from the fuel cell. First, a control rod is withdrawn from the fuel cell that is loaded without bypassing the interlock system. An additional control rod can be withdrawn from this fuel cell if the interlock is bypassed and fuel is removed from the cell.

FUEL-IN: This top event defines the inadvertent loading of fuel in an uncontrolled fuel cell.

SRM-BWR: This is a third top event on the event trees. The SRMs monitor the neutron counts. The neutron counts will increase as the control rod is moved out of the core. As the source range count rate increases, numerous indications and alarms will alert the operator that reactor power is increasing or the subcriticality margin is decreasing. Also the status of the CRD system is readily available in MCR.

FUEL-OUT: This top event defines an operator action. Once the SRM alarms indicate the unexpected increase in neutron account, an operator recognizes the vulnerable situation and stops inserting the fuel and pulls it back out.

RX-CRIT-BWR: This top event represents if this specific configuration is vulnerable to critical or prompt-critical. Steady state evaluation of neutronic behavior of this specific configuration needs to be performed determine if that scenario could lead to a critical condition. The reactor achieving re-criticality is dependent on the flow rate, reactor core configuration, fresh fuel versus irradiated fuel, and RCS temperature, etc. Control rod insertion events during testing and maintenance have been reported; however, no criticality event has been reported during CRD testing and maintenance in U.S. BWRs. However, Japanese BWRs have reported several criticality events during test and maintenance prior to 2000 (Appendix A). This would require thermal-hydraulic and/or neutronic analysis which is beyond the scope of this study. This top event also defines that if the reactor achieves re-criticality, what fraction of events would result in prompt-critical.

3.7 Isolation of CRD HCU (CRDHU)

Event Tree: Isolation of CRD HCU (Testing and Maintenance)

It is common practice to perform multiple tasks during a refueling outage to reduce the outage duration while the control rods are fully inserted in the core. One of the scheduled task utility performed is the alignment of the control rod drive system in preparation for hydro-lazing the scram discharge volume. During the process, the operator would isolate scram discharge valves 101 and 102. The procedure “Discharging of CRD Accumulators with Mode switch in Shutdown or Refuel” requires the operator to monitor the cooling water and exhaust header pressures every 10 HCU after 50 HCUs have been isolated. The Operating Experiences (OE) exists specific to this event. The Institute of Nuclear Power Operation (INPO) Significant Event Notice (SEN)-264, “Unplanned BWR Control Rod Withdrawals While Shutdown,” dated April 2007, detailed historical events at several BWRs between 1978 and 2000 where single or multiple control rods unexpectedly moved out of the core without an intentional withdrawal signal. As the HCUs are isolated, the delta P across the piston in non-isolated HCU will increase upon closing of valve 101. This differential pressure increase across the piston is very small. This increase in delta pressure depends on the CRD pump flow rate. However, by the time the operator isolates all except the last four or five HCUs on isolation of valve 101, the differential pressure will be large enough to drive the control rod out until valve 102 to the related HCU is closed.

IE-BWR-CRDHU: This event tree represents reactivity insertion during testing and maintenance on all hydraulic units. Initiating event frequency represents BWR shutdown frequency for refueling yearly.

CRD-HU: This top event represents the testing and maintenance event of alignment of the control rod drive system in preparation for hydro-lazing the scram discharge volume.

MCR-OP-1: This top event represents the control room operator actions. The control rod position and differential pressure indications are available in the control room. The MCR operator should verify the differential pressure after so many HCUs are isolated and adjust CRD pump flow rate as required. If the MCR operator is aware of the HCUs maintenance, he can monitor the control rods movement in the CR.

CRD-DP (CRD-DP-1, CRD-DP-2): This top event represents availability of the differential pressure (DP) indictor. If the MCR operator is aware of the testing and maintenance on HCU, he can respond to unusual fluctuation in DP as well as rod movement.

SRM-BWR: This is a third top event on the event trees. The SRMs monitor the neutron counts. The neutron counts will increase as the boron dilution is progressing. As the source range count rate increases, numerous indications and alarms will alert the operator that reactor power is increasing or the subcriticality margin is decreasing.

SCRAM: This top event represents termination of the control withdrawal transient. Depending on the availability of the scram function, an operator can scram the reactor or manually terminate the transient.

RX-CRIT-BWR: This top event represents if this specific configuration is vulnerable to critical or prompt-critical. Steady state evaluation of neutronic behavior of this specific configuration needs to be performed to determine if that scenario could lead to a critical condition. The reactor achieving re-criticality is dependent on the flow rate, reactor core configuration, fresh fuel versus irradiated fuel, and RCS temperature, etc. Control rod insertion events during testing and maintenance have been reported; however, no criticality event has been reported during CRD testing and maintenance in U.S. BWRs. However, Japanese BWRs have reported several criticality events during testing and maintenance prior to 2000 (Appendix A). This would require thermal-hydraulic and/or neutronic analysis which is beyond the scope of this study. This top event also defines that if the reactor achieves re-criticality, what fraction of events would result in prompt-critical.

3.8 Reactivity Insertion During Manual Reactor Shutdown (MANSD)

Event Tree: Manual Reactor Shut Down

It is a common practice while entering in a scheduled (non-emergency) SD, the operator will reduce the reactor power by slowly inserting the control rods into the reactor. This process continues in steps until all the rods are inserted in the reactor achieving the subcriticality. As the control rods are inserted into the reactor, power level is reduced and consequently the coolant temperature will decrease. However, this would lead to positive reactivity feedback consequently increasing power and negating some of the negative reactivity insertion from the control rod. Depending on the insertion and cooldown rate, reactor configuration (beginning or end of cycle), and how long the operator waits between the rods insertion, it is possible to re-achieve criticality. Note that when all the control rods are fully inserted in the reactor, it has sufficient negative reactivity to overcome this positive reactivity feedback and maintain sufficient sub-criticality margin as required. This event took place in the early 1980s at Big Rock Point.

IE-MAN-SD-FREQ: This top event is an initiating event. This event represents the frequency of manual shutdowns per year. Normally manual shutdown occurs when a plant is scheduled to be in shutdown. It is assumed that all refueling outages are scheduled shutdowns.

MCR-OP-2: This top event represents operator error of not paying attention to procedures.

SRM: This is a third top event on the event trees. The SRMs monitor the neutron counts. The neutron counts will increase as the boron dilution is progressing. As the source range count rate increases, numerous indications and alarms will alert the operator that reactor power is increasing or the subcriticality margin is decreasing. Also the RCS temperature and power will rise.

SCRAM: This top event represents termination of the transient. Depending on the availability of the scram function, the operator would either scram the reactor or manually terminate the transient.

RX-CRIT-MNSD: This top event represents if this specific configuration is vulnerable to critical or prompt-critical. Steady state evaluation of neutronic behavior of this specific configuration needs to be performed determine if that scenario could lead to a critical condition. The reactor achieving re-criticality is dependent on the flow rate, reactor core configuration, fuel, RCS temperature, and CRD withdrawal positions, etc. In the 1980s during manual shutdown at Big Rock Point the reactor achieved criticality while operator attention got diverted; however, no criticality event has been reported since then. This would require thermal-hydraulic and/or neutronic analysis which is beyond the scope of this study. This top event also defines that if the reactor achieves re-criticality, what fraction of events would result in prompt-critical.

References:

1. Burnett, T.; D. Kummeth, J. Andrachek, 'Risk of PWR Inadvertent Criticality During Shutdown and Refueling; EPRI Outage Risk Assessment and Management (ORAM) Program', NSAC/183, Electric Power Research Institute, December 1992.
2. Anderson, M. J., Glazier, J. R., Stirn, R. C., G. A. Walford, 'Guidelines for BWR Reactivity Control During Refueling', NSAC/164L, Electric Power Research Institute, April 1992.
3. Diamond, D. J., C. J. Hsu, R. Fitzpatrick, 'Reactivity Accidents, A reassessment of the Design Basis Events' NUREG/CR-5368 (BNL-NUREG-52198), Brookhaven National Laboratory, January 1990.

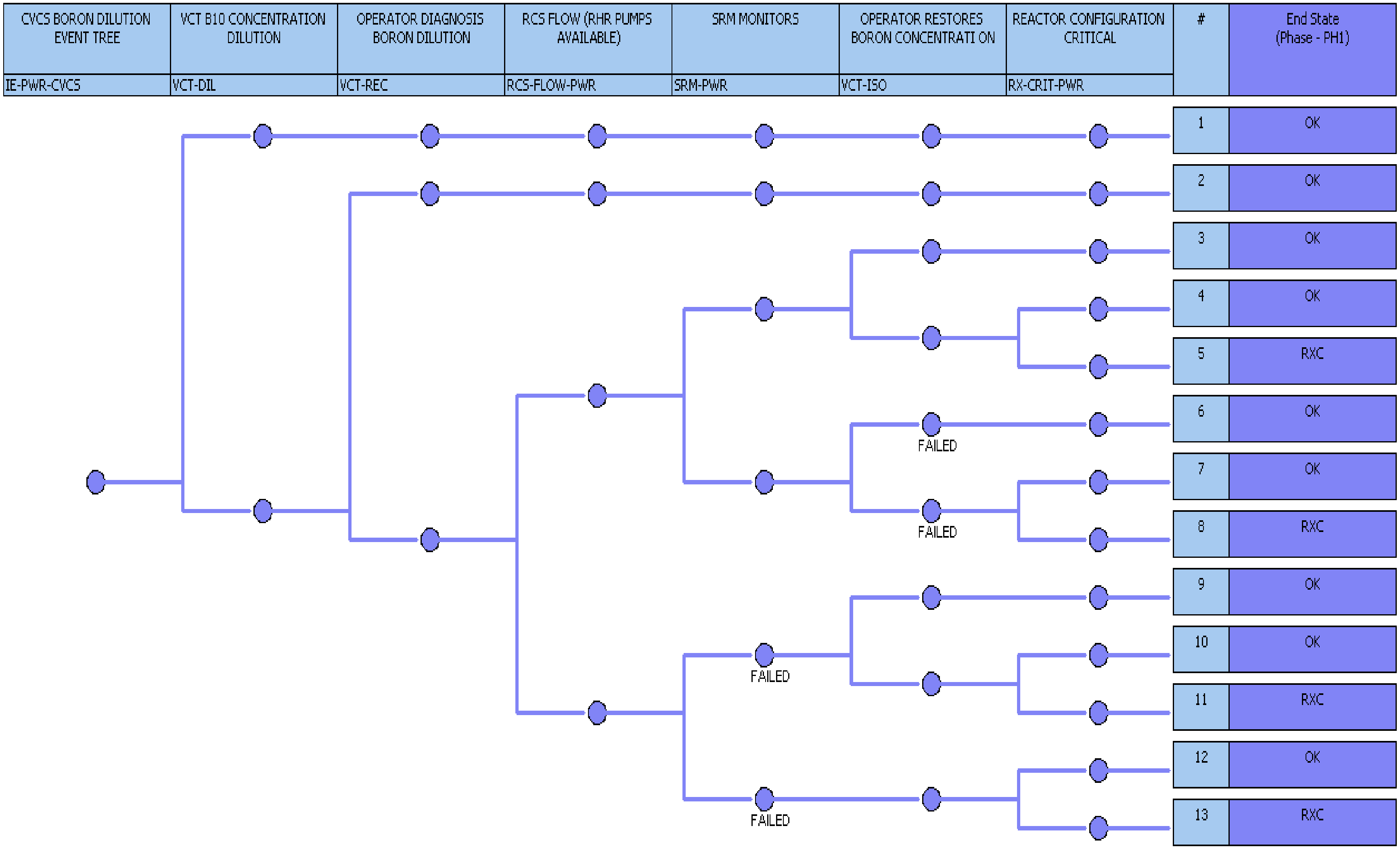

Figure 3-1. Boron Dilution Due to Malfunction in CVCS

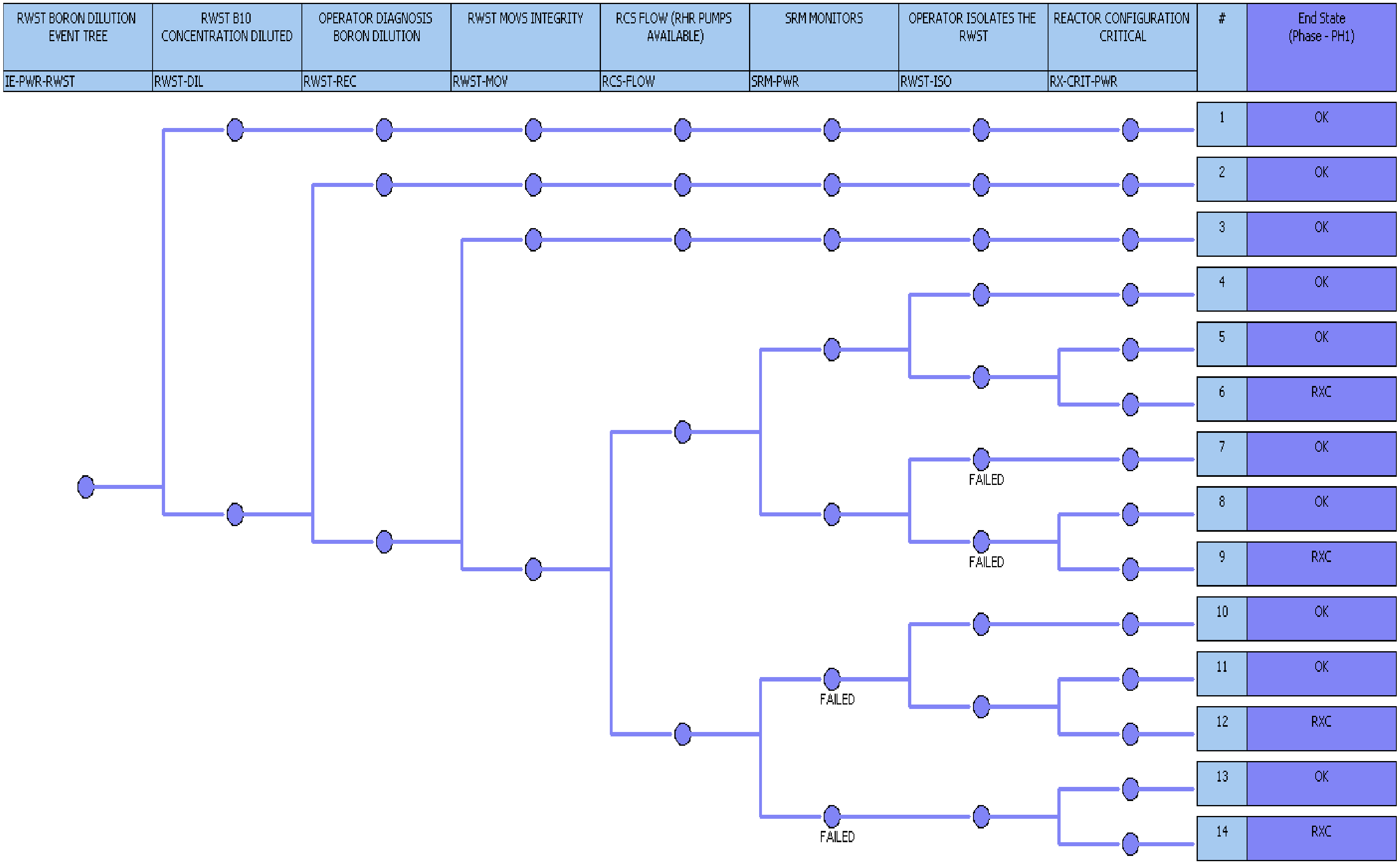

Figure 3-2. Loss of RWST MOVS Integrity

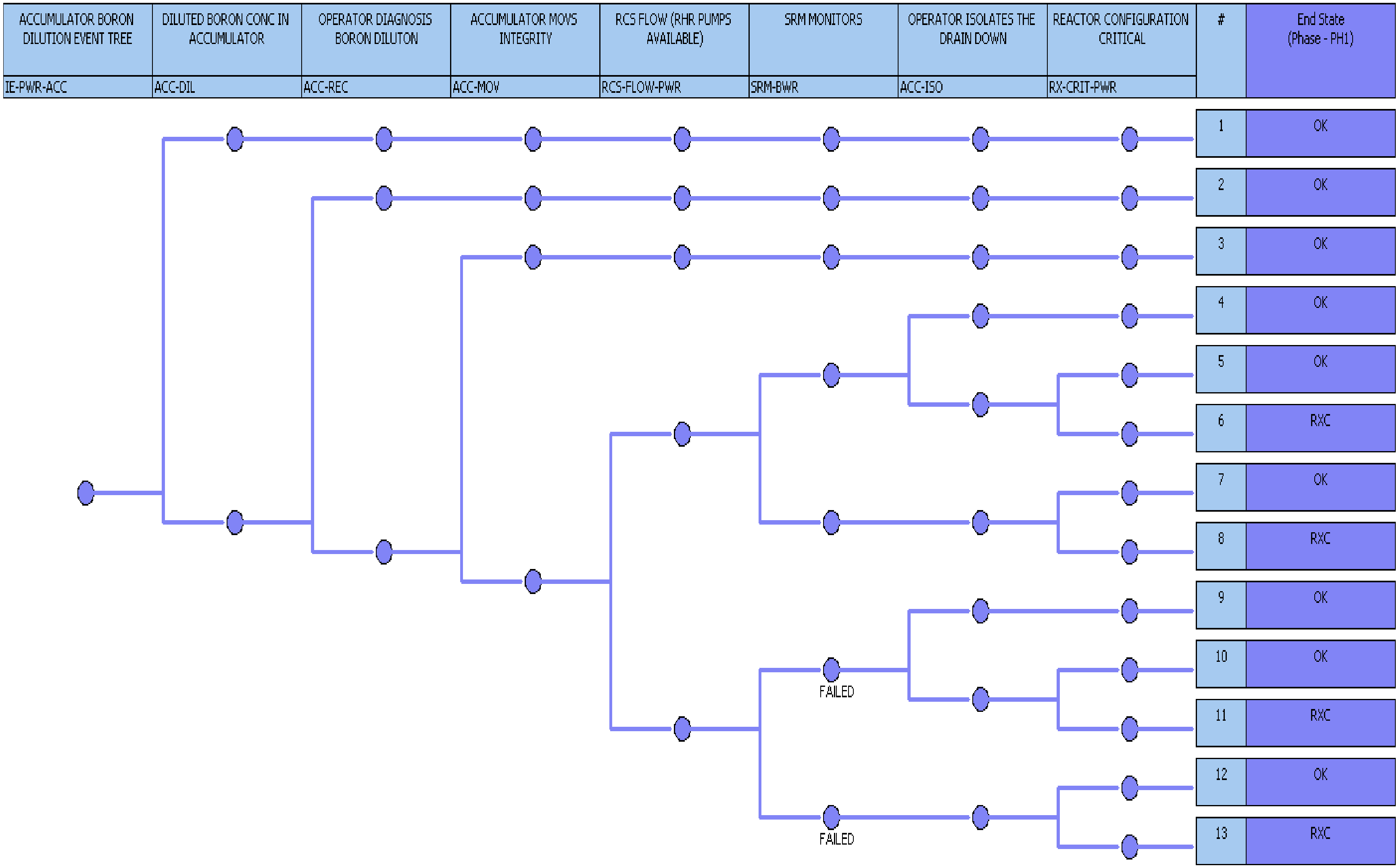

Figure 3-3. Loss of Accumulator MOVs Integrity

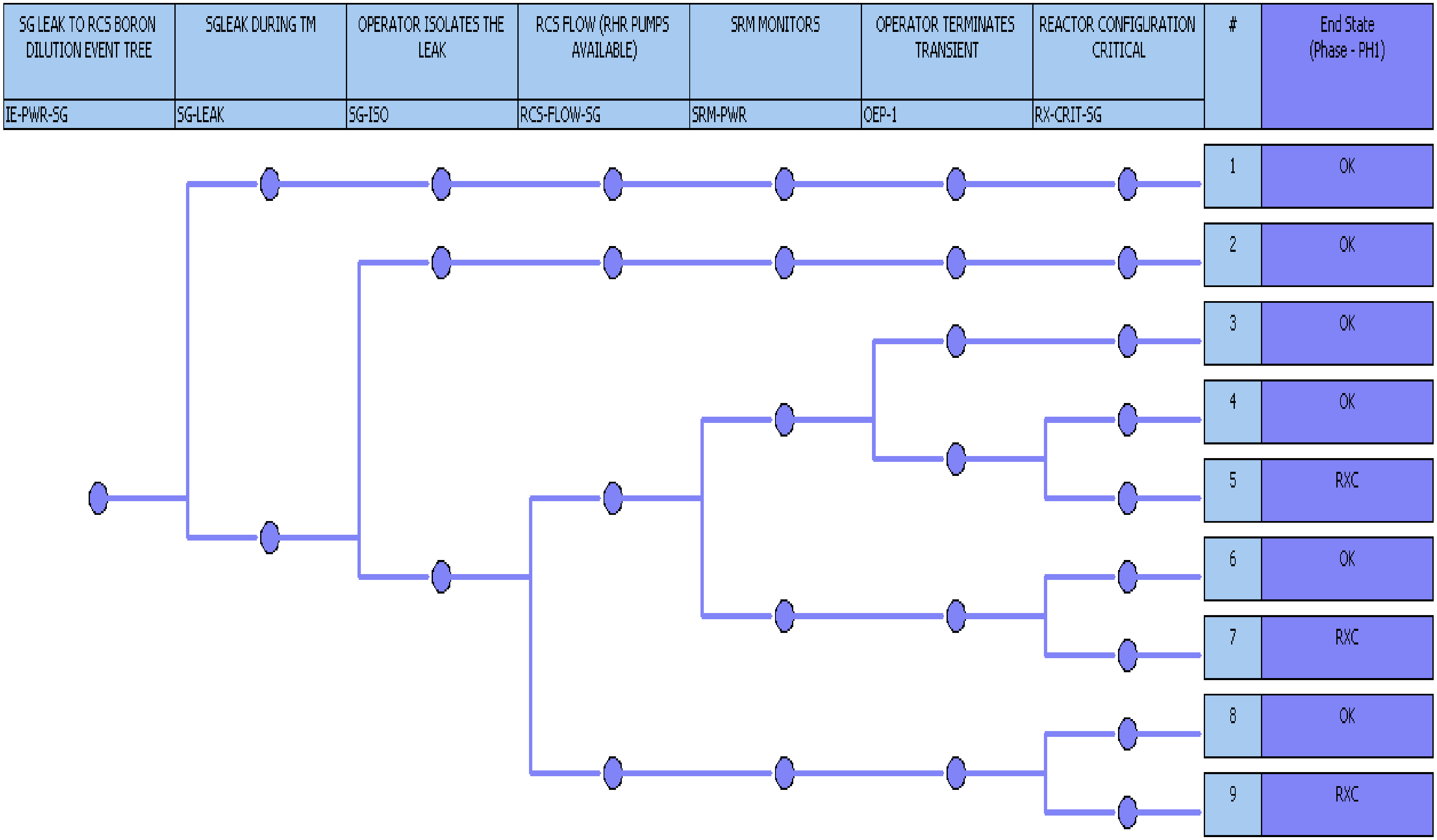

Figure 3-4. Loss of Coolant from SG to Primary

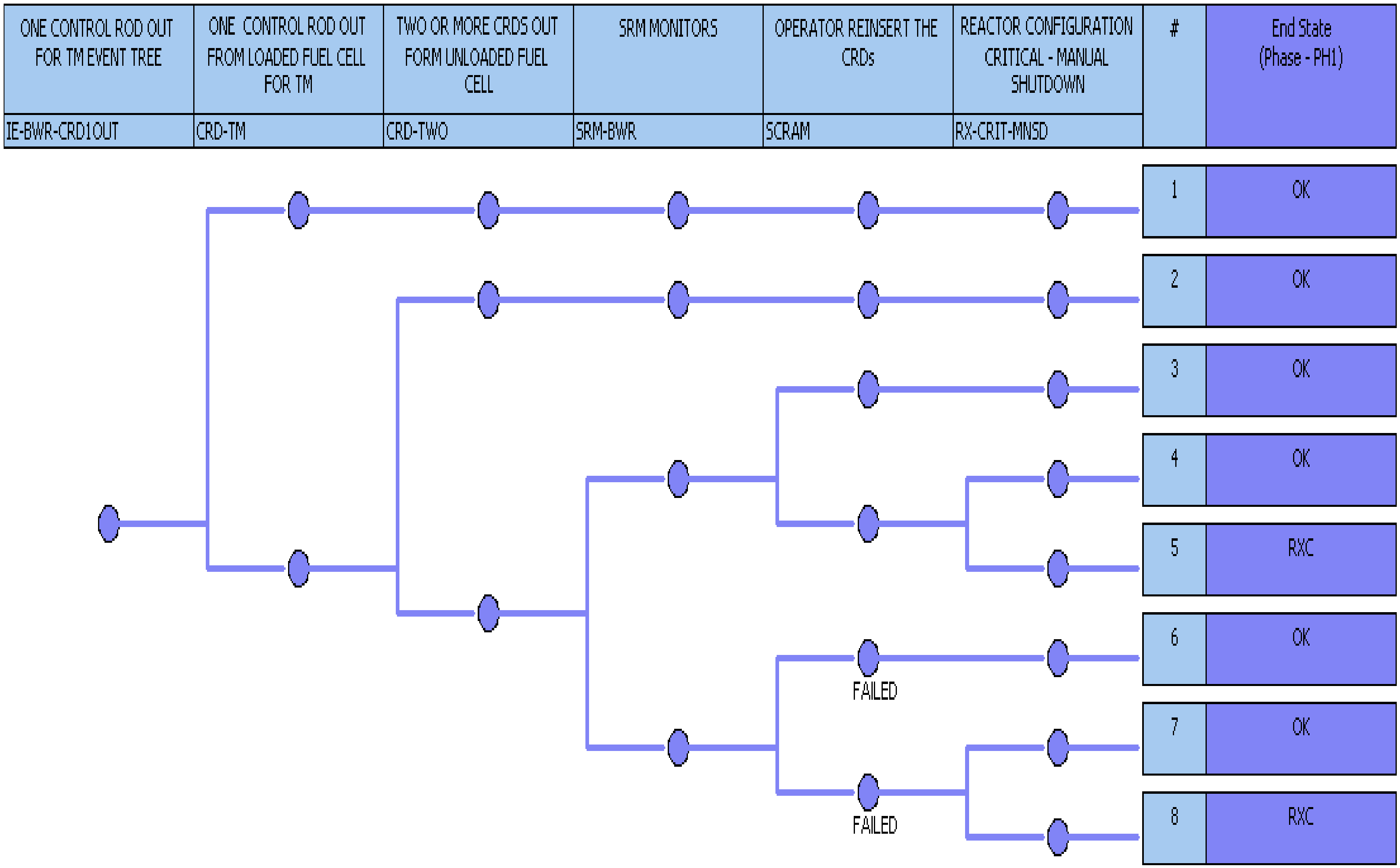

Figure 3-5. One CRD Removed for TM – BWR

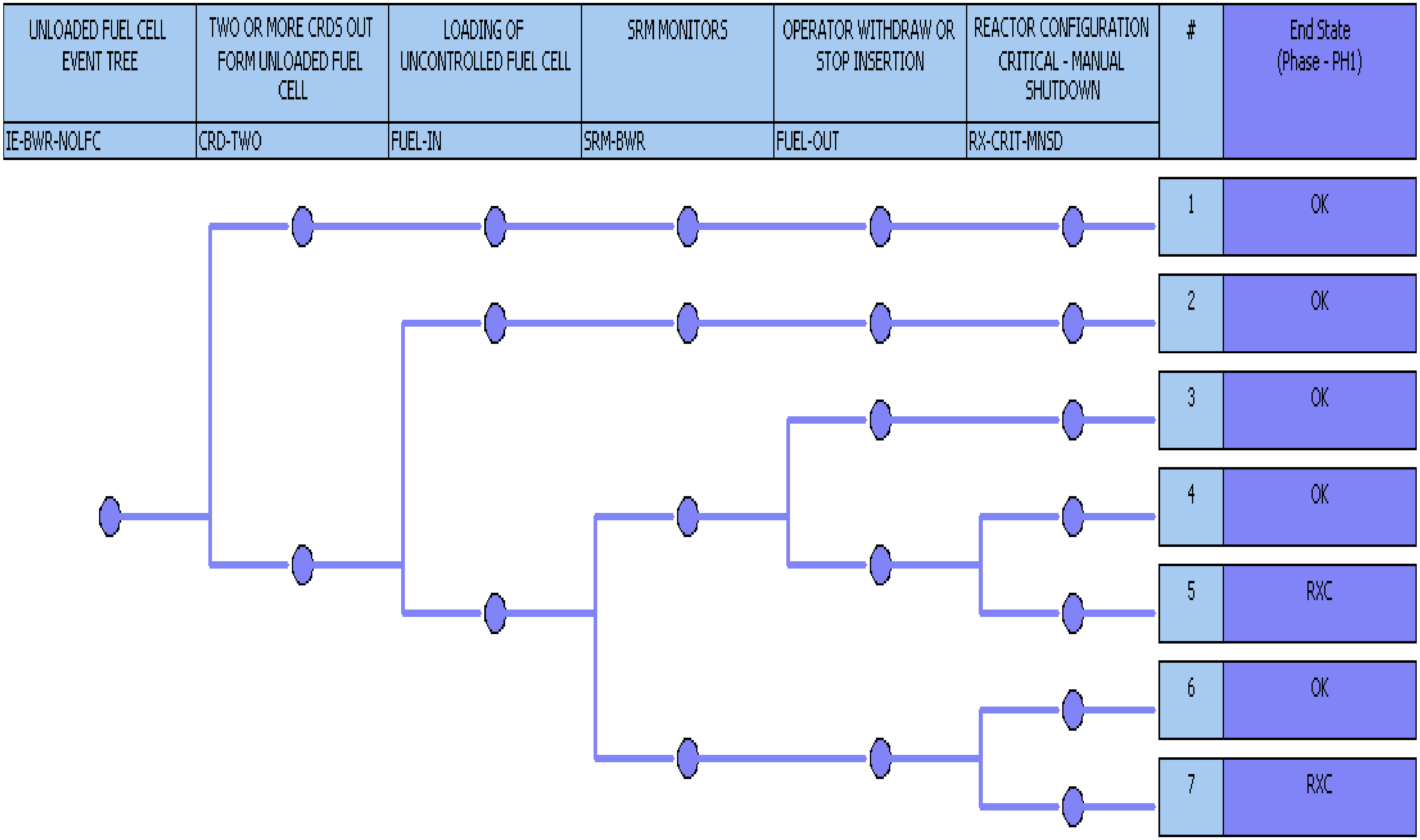

Figure 3-6. Two or More CRD Removed from Unloaded Fuel Cell

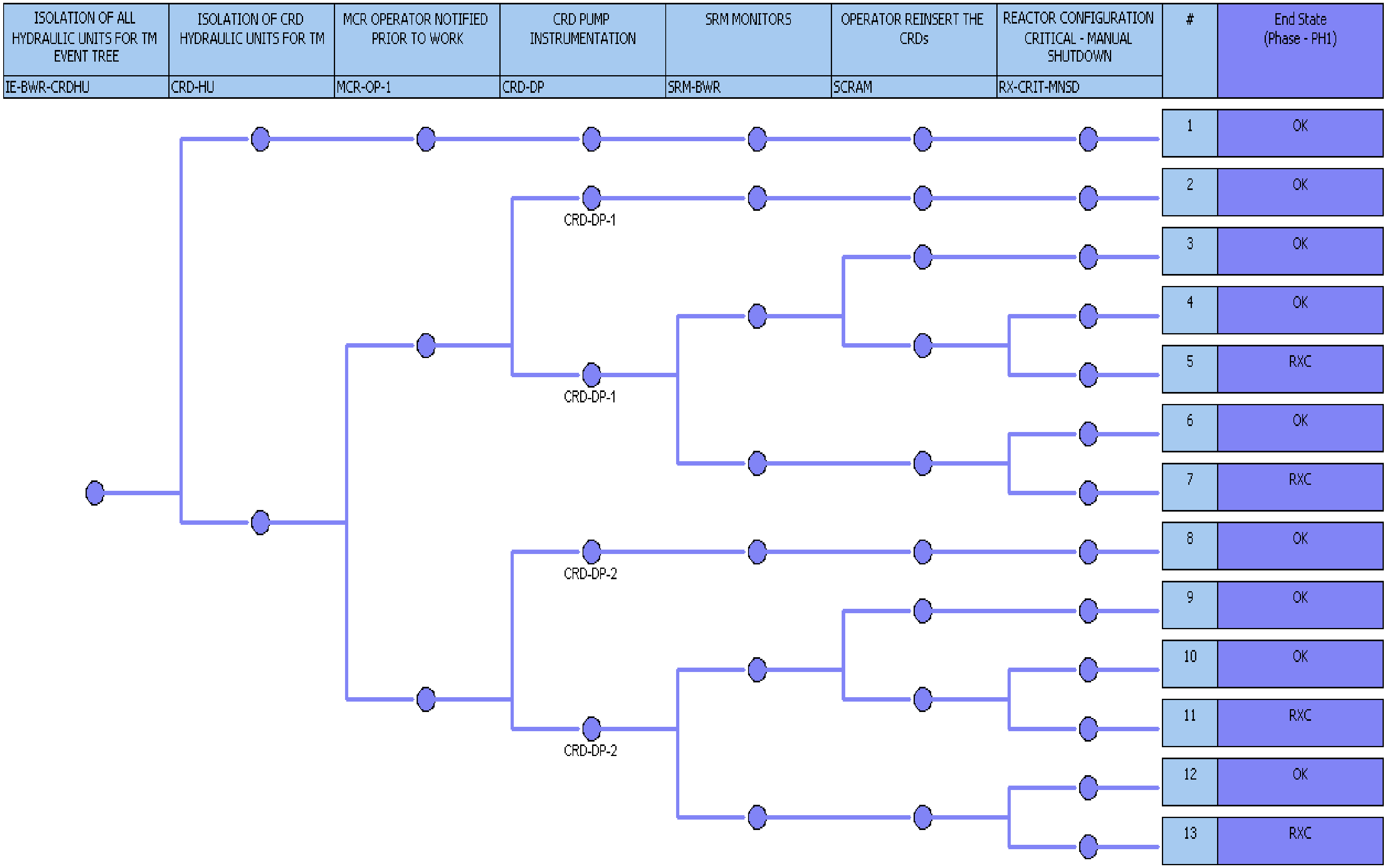

Figure 3-7. Isolation of CRDs Hydraulic Units – BWR

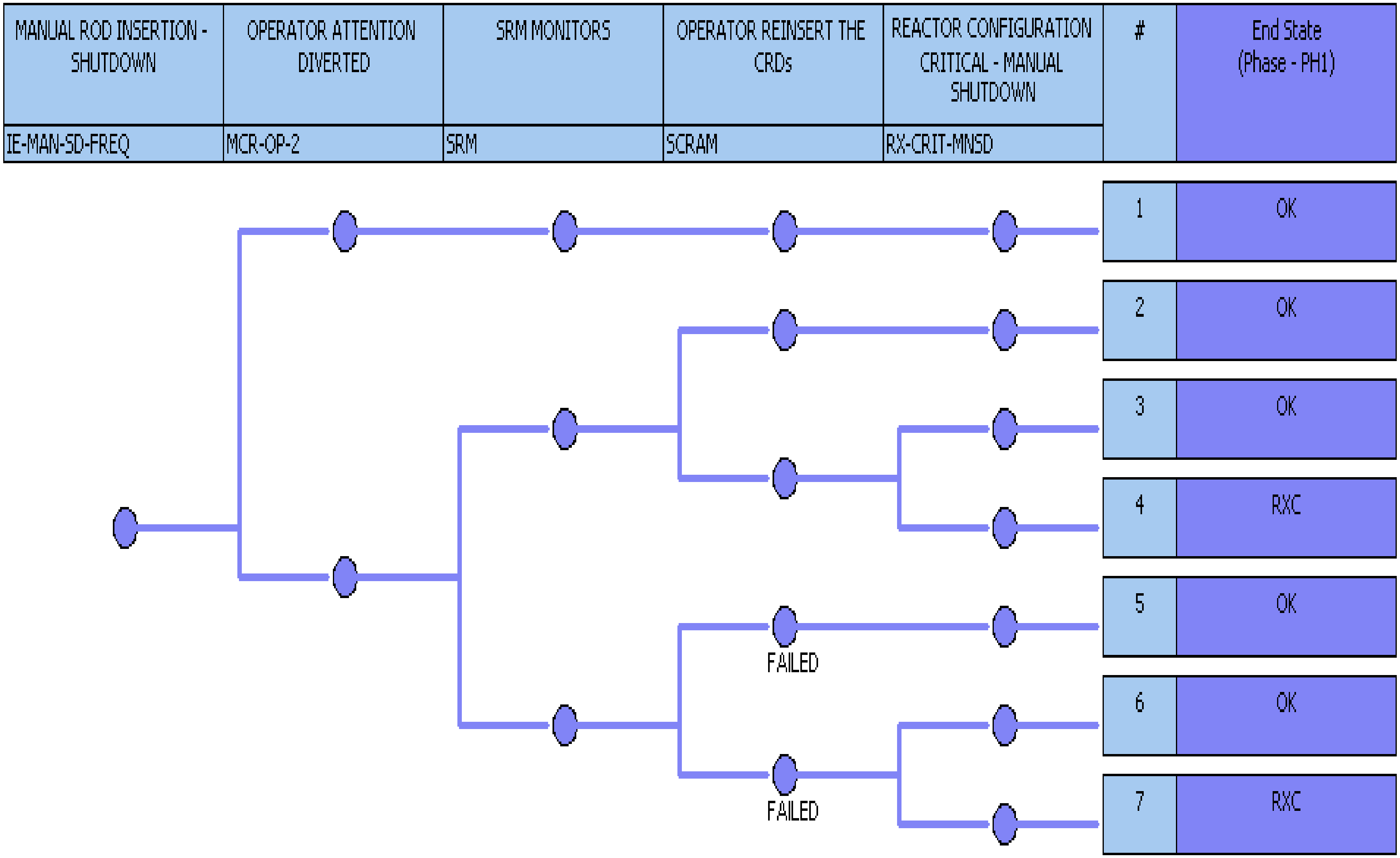

Figure 3-8. Manual Rod Insertion - Shutdown

4. FAULT TREE MODELS

This section presents the fault tree models developed for the event trees presented in the previous section. It was not possible to build reasonably generic train level fault tree models for the event tree top events as system P&IDs, power supplies; isometric drawings and procedures were not available. Some fault trees were built based on the similar studies performed in the past. No supporting system dependencies have been explicitly included in the fault trees.

5. BASIC EVENT DATA

The basic event data used to quantify the event trees and fault trees from the preceding sections are described in the following pages. The basic event values are generally independent hardware failure probabilities, common cause failure probabilities, test and maintenance unavailabilities, or human error probabilities for pre-accident and post-accident operator actions. A complete listing of all the non-human basic events is provided in Table 5-1. This section deals primarily with the independent hardware failure probabilities. No common cause or power-related events are modeled. Section 6 provides additional explanation of the human error probability calculations.

The basic event probability data in Table 5-1 is either stored or calculated. The stored probabilities are taken from a number of different sources and provides the basic event name, description, probability, uncertainty distribution type and parameter, and source of the data value if not ASEP generic. The calculated probabilities are calculated from input data using built-in SAPHIRE reliability equations (e.g., a MOV closed valve fails to open from a failure rate and a mission time), or a complex calculation using a plug in software module (e.g., a common cause failure probability calculated using the Alpha Factor Method). Templates and compound event calculation are described in the following sections.

5.1 Template Events

Template events are basic events that most often represent a particular failure mode for a particular component type (e.g., check valve fails to open, motor operated valve fails to close, etc.). Note that templates are not restricted to basic event probability values. They are also used for failure rate parameters, alpha factors, and anywhere else that a given data value may need to be reused many times. The template events have all of the information needed to calculate the failure probability for a given component and failure mode. By creating template events and using them in the database, the probability and uncertainty parameter for a given component type and failure mode need only be entered once instead of separately for each specific component in the model. Once the probability and uncertainty parameter have been entered, those basic events representing components of the same type, with the same failure mode, will reference the template. The advantage of using templates is if a parameter changes, the parameter only has to be changed once, at the template event, and all components that are similar will be updated automatically. Table 5-1 lists the template events that are used in the model and provides the template event name, description, probability, uncertainty distribution type and parameter, and source of the data value if not SPAR generic[5-1].

5.2 Compound Events

Compound events can be viewed as super-component basic events which combine other basic events according to some rule or equation to obtain a failure probability. The most common use of compound events is to develop a train failure probability from the pump and valve failure events comprising the train super-component. In this case the compound event would use a SAPHIRE utility module to add a list of basic events composing the train. The compound event calculation provides both the point estimate for the train and is able to provide sampled values during uncertainty analysis. The basic events used in a compound event will not appear in the cut sets, and therefore always represent event groups that could also be modeled with independent sub-trees. The compound event feature is used primarily to minimize the number of basic events in the cut sets, while also allowing automated uncertainty analysis.

5.3 Uncertainty

One motivation for performing an uncertainty analysis using the SPAR model results is to estimate the variability of the analysis results. This variability arises from uncertainties in model inputs including basic event probabilities, initiating event frequencies, model structure, analysis assumptions and others. In general, a SPAR model can be divided into two types, aleatory and deterministic. Aleatory models (i.e., random) include basic event probabilities and initiating event frequencies. For example, the "fails to run" model for a diesel generator is based upon a Poisson process which is used to yield a probability of failure. Deterministic models are based upon deterministic equations and include such things as the thermal-hydraulic calculations that support the event tree structure, fault tree success criteria and the event/fault tree logic model itself.

Each of these two input types of models has uncertainty associated with it. For example, in a thermal-hydraulic calculation, the temperature and pressure for a particular accident sequence spans a range. For the diesel generator, data may be collected on operation time to determine a failure rate, but the rate is not exactly known (even if we collect a large amount of data). Further, the representation selected for either model type may not be exact, thus indicating a degree of model uncertainty. This "state of knowledge" uncertainty (i.e., model uncertainty) is referred to as epistemic uncertainty.

Another type of epistemic uncertainty is related to the variability in model parameters. This type of uncertainty is frequently called parameter uncertainty. When an analyst indicates that they have done an uncertainty analysis in PRA, they generally mean that they evaluated only the epistemic parameter uncertainty for just the aleatory portion of the PRA. The SPAR logic models incorporate basic events which have uncertainties (parameter uncertainty) developed during the data derivation process. All SPAR models are evaluated for uncertainty and a summary of these results are provided in the results section of this document. SPAR model results are only meaningful when the results of an uncertainty evaluation are considered along with the point estimate.

Deterministic and model uncertainty is much more difficult to evaluate and is typically not included in PRA uncertainty calculations. In other words, the formal treatment of model uncertainty is a “state-of-the-art” practice. However, the SPAR modeling project has made an attempt to identify key issues/parameters that have significant variability within PRAs associated with commercial nuclear power plants. These issues were identified during plant visits made as part of the Significance Determination Process (SDP) benchmarking effort. Using a rudimentary quantitative process, these issues were ranked with respect to variability within the industry and the potential CDF impact.

5.4 References

5-1. NUREG/CR-6928, Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants, U.S. Nuclear Regulatory Commission, February 2007.

Table 5-1. Reactivity Insertion Accident Analysis Basic Event Data

Event Name	Event Description	Model Type	Failure Rate (Per year)	Mission time (hr)	Distribution Type	Dist. Parameter	Probability	Ref #
ACC-INS-FO-MCR	ACCUMULATOR INSTRUMENTATION FAILURE IN MCR	RANDOM			CNI	0.00E+0	1.00E-3	
ACC-MOV-CO-VALVE	ACCUMULATOR MOV VALVE FAILURE	RANDOM	4.00E-8	3.60E+2			1.44E-5	
ACC-MOV-OO-VALVE	OPEN ACC VALVE FAILS TO CLOSE	RANDOM			CNI	0.00E+0	1.00E-3	
ACC-XHE-XD-MCR	MCR OPERATOR FAILS TO DIAGNOSE THE BORON DILUTION EVENT	RANDOM			Log Normal	1.00E+1	1.00E-2	1
ACC-XHE-XE-6HR	FAILURE TO CHECK CONC. SIX HRS AFTER ACCUMULATOR FILL	RANDOM			Log Normal	1.00E+1	5.00E-3	1
ACC-XHE-XE-MONTHLY	FAILURE TO PERFORM MONTHLY TEST	RANDOM			Log Normal	1.00E+1	1.00E-3	1
ACC-XHE-XE-TEST	OPERATOR MADE ERROR IN TESTING	RANDOM			Log Normal	1.00E+1	5.00E-3	1
ACC-XHE-XE-TEST1	OPERATOR MADE AN ERROR IN SCHEDULED TESTING	RANDOM			Log Normal	1.00E+1	1.47E-1	1
ACC-XHE-XM-ISO	OPERATOR FAILS TO TERMINATE BORON DILUTION EVENT	RANDOM			Log Normal	1.00E+1	1.00E-3	1
ACC-XHE-XM-PWR	OPERATOR FAILS TO ISOALTE POWER	RANDOM			Log Normal	1.00E+1	1.00E-3	1
ACC-XHE-XM-VALVE	MCR OPERATOR INADVERTENTLY OPENS ACC VALVE TO RCS	RANDOM			Log Normal	1.00E+1	1.00E-3	1
CRD-HCU-TM-REFUEL	TM IS PERFORMED ALL HCU	RANDOM			Log Normal	1.00E+1	2.00E-1	
CRD-INS-FO-LIGHT	INDICATOR LIGHTFOR CRD OUT FAILS	RANDOM			CNI	0.00E+0	1.00E-3	
CRD-PUMP-FO-DP	DIFFERENTAIL PRESSURE INSTRUMENT FAILURE	RANDOM			CNI	0.00E+0	1.00E-3	
CRD-XHE-XD-ERROR	FIELD OPERATOR FAILS TO DIAGNOSE UNUSUAL DP	RANDOM			Log Normal	1.00E+1	1.00E-2	1
CRD-XHE-XD-MCR	MCR OPERATOR FAILS TO DIAGNOSE EXCESS DELTA-P	RANDOM					1.00E-2	1
CRD-XHE-XM-MCR	MCR OPERATOR FAILURE TO FOLLOW PROCEDURE	RANDOM			Log Normal	1.00E+1	1.00E-3	1
CRD-XHE-XM-OEP	FIELD OPERATOR FAILS TO FOLLOW PROCEDURE	RANDOM					1.00E-3	1
CVS-HTX-LK-RCP	RCP SEAL HEAT EXCHANGER LEAKING	RANDOM			CNI	0.00E+0	1.00E-4	
CVS-INS-FO-LEVEL	VCT LEVEL INST FAILURES	RANDOM			CNI	0.00E+0	1.00E-3	
CVS-MOV-CO-CHGA	MOV IN CHARGING LINE A FAILS OPEN	RANDOM			CNI	0.00E+0	2.00E-5	2
CVS-MOV-CO-CHGB	MOV IN CHARGING LINE A FAILS OPEN	RANDOM			CNI	0.00E+0	2.00E-5	2
CVS-XHE-XD-VCTLVL	OPERATOR FAILURE TO NOTICE INCREASED VCT LEVEL	RANDOM			Log Normal	1.00E+1	1.00E-2	1
CVS-XHE-XM-DILUTION	OPERATOR INADVERTENTLY DILUTE RCS WATER BELOW THE DESIRED LIMIT	RANDOM			Log Normal	1.00E+1	1.00E-3	1

Table 5-1 (Continued).

Event Name	Event Description	Model Type	Failure Rate (Per year)	Mission time (hr)	Distribution Type	Dist. Parameter	Probability	Ref #
CVS-XHE-XM-ISO	OPERATOR FAILS TO TERMINATE BORON DILUTION EVENT FROM CVCS INJECTION	RANDOM			Log Normal	1.00E+1	6.00E-2	1
CVS-XHE-XM-MISALIGN	OPERATOR INADVERTENTLY MISALIGN THE VALVES	RANDOM			Log Normal	1.00E+1	1.00E-3	1
CVS-XHE-XM-TEST	OPERTOR FAILS TO TEST RCS WATER CHEMISTRY	RANDOM			Log Normal	1.00E+1	1.00E-3	1
FAILED-EVENT	BLANK BASIC EVENT FOR NEW ENTRY	RANDOM					1.00E+0	
HPI-MOV-CO-HPI	MOV IN HPI LINE FAILS OPEN	RANDOM			CNI	0.00E+0	2.00E-5	3
IE-BWR-REF-FRQ	REACTOR SHUTDOWN FOR REFUELING FREQUENCY PER YEAR	RANDOM					5.40E-1	
IE-MAN-SD-FREQ	MANUAL ROD INSERTION - SHUTDOWN	RANDOM					5.90E-1	
IE-PWR-REF-FREQ	REACTOR SHUTDOWN FOR REFUELING FREQUENCY PER YEAR	RANDOM					6.20E-1	
LPI-MOV-CO-LPIA	MOV IN LPI LINE FAILS OPEN	RANDOM			CNI	0.00E+0	2.00E-5	2
LPI-MOV-CO-LPIB	MOV IN LPI LINE FAILS OPEN	RANDOM			CNI	0.00E+0	2.00E-5	2
MCR-XHE-XD-HIDP2	MCR OPERATOR FAILURE TO DIAGNOSE HI-DP INDICATION (NOT INFORMED)	RANDOM			Log Normal	1.00E+1	1.00E-2	1
MCR-XHE-XD-SRM	OPERATOR FAILS TO DIAGNOSE HI-COUNTS	RANDOM			Log Normal	1.00E+1	1.00E-2	1
MCR-XHE-XE-SRM	OPERATOR IGNORES THE SRM INDICATIONS/MISCOMUNICATES	RANDOM			Log Normal	1.00E+1	1.00E-3	1
MCR-XHE-XM-DIVERT	MCR OPERATOR ATTENTION IS DIVERTED	RANDOM			Log Normal	1.00E+1	1.00E-2	1
MCR-XHE-XM-FUELOUT	OPERATOR FAILS TO STOP INSERTION OR PULL FUEL BACKUP	RANDOM			Log Normal	1.00E+1	1.00E-3	1
MCR-XHE-XM-TRANS	OPERATOR TERMINATES TRANSIENT	RANDOM			Log Normal	1.00E+1	2.00E-2	1
OPR-XHE-XE-CRD1	OPERATOR FAILS TO VERIFY ONE CRD OUT	RANDOM			Log Normal	1.00E+1	1.00E-3	1
OPR-XHE-XE-CRD1-DEP	SECOND OPERATOR FAILS TO VERIFY	RANDOM			Log Normal	1.00E+1	1.44E-1	1
OPR-XHE-XE-FLOAD	OPERATOR-1 FAILS TO CHECK FUEL LOADING PATTERN	RANDOM			Log Normal	1.00E+1	1.00E-3	1
OPR-XHE-XE-FLOAD-DEP	OPERATOR-2 FAILS TO CHECK FUEL LOADING PATTERN	RANDOM			Log Normal	1.00E+1	1.44E-1	1
OPR-XHE-XE-IGNORELK	OPERATR IGNORES SG LEAK	RANDOM			Log Normal	1.00E+1	1.00E-3	1
OPR-XHE-XE-PATTERN	LOADING PATTERN ERROR	RANDOM			Log Normal	1.00E+1	1.00E-3	1
OPR-XHE-XM-COMM	FIELD SUPERVISSOR FAILED TO NOTIFY MCR OPERATOR	RANDOM			Log Normal	1.00E+1	1.00E-3	1
OPR-XHE-XM-WJUMP	OPERATOR REMOVES WRONG JUMPER	RANDOM			Log Normal	1.00E+1	1.00E-3	1

Table 5-1 (Continued).

Event Name	Event Description	Model Type	Failure Rate (Per year)	Mission time (hr)	Distribution Type	Dist. Parameter	Probability	Ref #
OPR-XHE-XM-WJUMP-DEP	SEOND TECH FAILS TO VERIFY	RANDOM			Log Normal	1.00E+1	1.44E-1	1
RCS-CNF-CR-BWR	REACTOR CONFIGURATION CRITICAL - BWR	RANDOM			CNI	0.00E+0	1.00E-1	
RCS-CNF-CR-MNSD	REACTOR CONFIGURATION CRITICAL - MANUAL SHUTDOWN	RANDOM			CNI	0.00E+0	1.00E-1	
RCS-CNF-CR-PWR	REACTOR CONFIGURATION CRITICAL-PWR	RANDOM			CNI	0.00E+0	2.50E-1	
RCS-CNF-CR-SG	REACTOR CONFIGURATION CRITICAL DURING SG SCENARIO	RANDOM			CNI	0.00E+0	1.00E-2	
RCS-DN-CRITICAL	DELAYED NEUTRON CRITICAL	RANDOM			CNI	0.00E+0	9.00E-1	
RCS-PN-CRITICAL	PROMPT CRITICAL	RANDOM			CNI	0.00E+0	1.00E-1	
RCS-WATER-ACCUMULATES	UNBORATED WATER ACCUMULATES IN RCS PIPING	RANDOM			Log Normal	1.00E+1	5.00E-2	
RHR-MDP-FO-BWR	RHR SYSTEM IS NOT RUNNING (FROM SD PRA 1.3E-5/HR)	RANDOM			CNI	0.00E+0	5.00E-3	
RHR-MDP-FO-PWR	RHR SYSTEM IS NOT RUNNING (FROM SD PRA 1.1E-6/HR)	RANDOM			CNI	0.00E+0	5.40E-4	
RHR-XHE-XD-RHR	OPERATOR FAILS TO DAIGNOSE LOSS OF RHR	RANDOM			Log Normal	1.00E+1	1.00E-2	1
RPS-CRD-TM-ONE	ONE CONTROL ROD OUT FOR TM	RANDOM					5.00E-1	
RPS-ELC-FO-SCRAM	ELECTRICAL SYSTEM FAILS	RANDOM			CNI	0.00E+0	1.00E-6	
RPS-MEC-FO-SCRAM	MECHANICAL SYSTEM FAILS	RANDOM			CNI	0.00E+0	1.00E-6	
RPS-XHE-XM-SCRAM	OPERATOR FAILS TO SCRAM THE REACTOR	RANDOM			Log Normal	1.00E+1	1.00E-4	1
RWS-MOV-OO-VALVE	OPEN VALVE FAILS TO CLOSE	RANDOM			CNI	0.00E+0	1.00E-3	
RWS-SYS-DL-OTHER	RWST B10 DILUTION FROM NON TM-EVENTS	RANDOM			CNI	0.00E+0	1.00E-2	
RWS-SYS-FO-BLEND	MECAHNICAL FAILURES IN BLENDING SYSTEM	RANDOM			CNI	0.00E+0	1.00E-3	
RWS-SYS-FO-TEST	FAILURE OF TESTING EQUIPMENT	RANDOM			CNI	0.00E+0	7.00E-5	
RWS-XHE-XD-MCR	MCR OPERATOR FAILS TO DIAGNOSE THE BORON DILUTION EVENT FROM RWST	RANDOM			Log Normal	1.00E+1	1.00E-2	1
RWS-XHE-XE-MONTH	FAILURE TO PERFORM MONTHLY TEST	RANDOM			Log Normal	1.00E+1	1.00E-3	1
RWS-XHE-XE-TEST	OPERATOR MADE AN ERROR IN TESTING	RANDOM			Log Normal	1.00E+1	5.00E-3	1
RWS-XHE-XE-TEST1	OPERATOR MADE AN ERROR IN SCHEDULED TESTING	RANDOM			Log Normal	1.00E+1	5.00E-3	1
RWS-XHE-XE-TEST2	FAILURE TO CHECK RWST CONC AFTER	RANDOM			Log Normal	1.00E+1	1.00E-2	

Table 5-1 (Continued).

Event Name	Event Description	Model Type	Failure Rate (Per year)	Mission time (hr)	Distribution Type	Dist. Parameter	Probability	Ref #
	BLENDING (TM)							
RWS-XHE-XM-BLEND	OPERATOR ERROR IN MIXING BLENDER	RANDOM			Log Normal	1.00E+1	5.00E-3	1
RWS-XHE-XM-ISO	OPERATOR FAILS TO TERMINATE BORON DILUTION EVENT FROM RWST INJECTION	RANDOM			Log Normal	1.00E+1	1.00E-3	1
SGS-SYS-LK-LEAK	STEAM GENERATOR LEAKS TO PRIMARY DURING TM	RANDOM			CNI	0.00E+0	1.40E-2	
SGS-SYS-TM-SG	SG IN TM	RANDOM					5.00E-1	
SGS-XHE-XD-SGLEAK	FAILURE TO DIAGNOS LEAK FROM SG TO RCS	RANDOM			Log Normal	1.00E+1	5.00E-2	1
SRM-INS-FO-SRM1	NEARBY SRM failed	RANDOM			CNI	0.00E+0	1.00E-3	
SRM-INS-FO-SRM2	SRM FAILS TO SEE INCREASE IN COUNTS	RANDOM					5.00E-1	
SRM-INS-FO-SRMS	SRM FAILS TO SEE INCREASE IN COUNTS	RANDOM			CNI	0.00E+0	1.00E-4	
SRM-INS-FO-TEMP	PRIMARY RCS TEMPERATURE INST FAILURE	RANDOM			CNI	0.00E+0	1.00E-4	

6. HUMAN RELIABILITY MODEL

The human actions included in the SPAR model consist of both pre-accident failures to restore systems following testing or maintenance, and of post-accident failures to align systems, to control or operate systems, and to recover system hardware failures. Pre-accident failures to restore systems following testing or maintenance are quantified using generic ASEP data, data from NUREG-1150 studies, and engineering judgment. Post-accident failures to align, control, or operate systems are addressed in Section 6.1. Post-accident failures to recover system hardware failures are addressed in Section 6.2.

Table 6-1 provides a listing of the human actions in the model. The following general naming scheme for the basic event component code and failure mode code for operator action events was adopted:

XHE-XE	Failure to perform a manual operation.
XHE-XL	Failure to recover a hardware failure locally (outside of the control room) by manipulation of the failed component to achieve the desired alignment or operation of the component.
XHE-XM	Failure to manually align and actuate (a manually controlled system)
XHE-XO	Failure to operate or control a system adequately to achieve required performance
XHE-XR	Failure to restore from testing or maintenance. Failure to restore events are considered pre-accident events and not evaluated using the formal HRA procedures described in this section.

SAPHIRE recovery rules are used to make the basic event substitutions required to implement the formal dependency calculations. The SAPHIRE rule file is provided in Listing 1 at the end of this section. The recovery rules perform two functions:

1. The recovery rules remove combinations of testing and maintenance events that are disallowed by the plant Technical Specifications. The cut sets that must be removed are identified in the output of the ME-TECHSPEC fault tree described in Section 4.

2. The recovery rules apply the result of the formal dependency calculations by removing the independent basic event and replacing it with its dependent version.

6.1 Alignment, Control, and Operate Events

This model includes operator actions to manually align systems and to control or operate systems. Examples of manual alignments are the operator actions required to initiate service water or firewater injection. Examples of control or operate events are the operator actions required to maintain the reactor level between the high and low level interlocks to prevent the automatic cycling of HPCI/RCIC injection valves or steam isolation valves (which systems studies have shown to result in higher system unreliability than the case with proper control of reactor level). HEPs for these types of actions have generally been calculated using the Human Error Worksheets as indicated in Table 6-1. It is the dependency between operator error events that introduces complications and requires some simplifying assumptions to make the model quantifiable. Previous large-scale PRAs have addressed the dependency problem by using screening values for the operator actions, solving the core damage equations, and then evaluating the human error dependency in any dominant cut sets that might appear as a result. In the SPAR models the screening value approach leads to excessively long calculation times and a huge number of cut sets that would need to be evaluated for dependence. So instead the operator error events are first evaluated without considering dependence then, after solution of the core damage equations, event substitutions are used to account for dependency between events in a given cut set. The advantage of this approach is that it results in fast run times. The disadvantage is that it is possible that some cut sets will be truncated that should not be when dependence is accounted for. The potential loss of cut sets is minimized by the use of a very low cutoff frequency when the core damage equations are solved.

To simplify the dependency calculations four dependent groups of operator error events have been identified. They are: 1) high-pressure injection alignment and control, 2) low-pressure injection alignment and control, 3) RHR and venting alignment and control, and 4) post-venting injection alignment and control. Table 6-2 identifies the human error events in each dependent grouping. The model is quantified by assuming that the first event in the group will always be correctly represented by its independent value. Then the following dependent events in the group are removed from the core damage cut sets by the recovery rules in Listing 1 and replaced with appropriate dependent events. Dependencies between events in different event groups, such as between the high pressure injection and the low pressure injection group, are considered negligible.

6.2 System Hardware Recovery Events

This model is similar to most full-scope PRAs and IPEs in that recovery of hardware failures and components unavailable because of testing or maintenance is not generally credited. However, the SPAR HRA methods assume that a given action is doable in the time available if the operators successfully diagnose and act. This will not be true of most system hardware repairs.

The exceptions to the above involve cases where detailed studies of system reliability have provided the probability that a given failure mode can be recovered. In this model the recoverable failures are in the HPCI and RCIC systems. The HPCI and RCIC recovery events are summarized in Table 6-1. Dependency between the HPCI and RCIC system recovery events has also been evaluated. The RCIC recovery failures are assumed to occur first and to therefore be independent. The HPCI recovery events are therefore considered to be dependent on the RCIC events in the combinations shown in Table 6-2. Dependency between recovery events other than those in the HPCI/RCIC systems is considered negligible, as is dependency between recovery events and operator actions to align and control systems. Tables 6-1 and 6-2 are placed in Appendix B due to their length.

6.3 References

6-1. H.S. Blackman and J.C. Byers, ASP Human Reliability Methodology Development, INEL-95/0139, April 1995.

6-2. A.D. Swain, and H. Gutman, Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278, August 1983, (THERP).

6-3. R.J. Belles, et al., Precursors to Potential Severe Core Damage Accidents: 1994 A Status Report, NUREG/CR-4674, Vol. 21, December 1995, pg. A.1-5.

6-4. K.D. Russell, et al., Systems Analysis Programs for Hands-on Integrated Reliability Evalulations (SAPHIRE) Version 5.0, NUREG/CR-6116, Vol 2. December 1993.

6-5. J. W. Minarick, et al., Precursors to Potential Severe Core Damage Accidents: 1986 A Status Report, NUREG/CR-4674, Vol. 5, May 1988.

6-6. G.M. Grant, et al., High-Pressure Coolant Injection (HPCI) System Performance, 1987-1993, INEL-94/0158, February 1995.

6-7. J.P. Poloski, et al., Reactor Core Isolation Cooling System Reliability, 1987-1993, INEL-95/0196, June 1997.

6-8. A.M. Kolaczkowski, et al., Analysis of Core Damage Frequency: Peach Bottom, Unit 2 Internal Events, NUREG/CR-4550, Vol. 4, Rev. 1, Part 1., August 1989.

6-9. D. Gertman, H. Blackman, J. Marble, J. Byers, C. Smith, *The SPAR-H Human Reliability Analysis Method*, NUREG/CR-6883, August 2005.

7. BASELINE RESULTS

Baseline results are shown in Tables 7-1, 7-2, and 7-3. Table 7-1 presents each event tree potential re-criticality (RXC) frequency along with its percent contribution to the overall re-criticality frequency. Table 7-2 presents baseline sequence re-criticality frequency by descending value. Table 7-3 presents basic event importance measures by descending risk increase value. The sequence cut sets were calculated using a truncation value of 1.0E-13.

Table 7-1. Reactivity Insertion Accident Analysis Initiating Event Contribution to Overall Re-criticality Frequency.

Initiating Event	IE Frequency (Per year)	Re-Criticality Frequency (Per year)	Percent contribution to RXC	Cumulative contribution to RXC
MANUAL ROD INSERTION - SHUTDOWN	5.9E-1	6.55E-6	96.94%	96.94%
SG LEAK TO RCS BORON DILUTION EVENT TREE	6.2E-1	1.80E-7	2.66%	99.60%
CVCS BORON DILUTION EVENT TREE	6.2E-1	2.50E-8	0.37%	99.97%
ISOLATION OF ALL HYDRAULIC UNITS FOR TM EVENT TREE	5.4E-1	1.90E-9	0.03%	100.00%
ONE CONTROL ROD OUT FOR TM EVENT TREE	5.4E-1	8.99E-11	0.00%	100.00%
ACCUMULATOR BORON DILUTION EVENT TREE	6.2E-1	5.35E-11	0.00%	100.00%
RWST BORON DILUTION EVENT TREE	6.2E-1	2.04E-11	0.00%	100.00%
UNLOADED FUEL CELL EVENT TREE	5.4E-1	0.00E+0	0.00%	100.00%
TOTAL		6.76E-6	100.00%	100.00%

Table 7-2. Reactivity Insertion Accident Analysis Baseline Results.

Event Tree Name	Sequence Name	RXC Frequency (Per year)
MANSD	7	6.5E-6
LOCA-SG	9	1.1E-7
MANSD	4	6.0E-8
LOCA-SG	5	4.4E-8
LOCA-SG	7	2.5E-8
LOVCT	05	2.1E-8
LOVCT	08	3.9E-9
CRDHU	07	1.4E-9
CRDHU	13	5.2E-10
CRD1OUT	8	8.9E-11
LOACC	06	2.7E-11
LOACC	08	2.6E-11
CRDHU	05	1.2E-11
LORWST	06	1.1E-11
LORWST	09	9.8E-12
CRDHU	11	4.3E-12
LOVCT	13	1.5E-12
CRD1OUT	5	7.0E-13
TOTAL		6.76E-6

Table 7-3. Reactivity Insertion Accident Analysis y 1 Baseline Importance Measure Results

Basic Event Name	Num. of Occ.	Probability	Fussel-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
ACC-MOV-OO-VALVE	4	1.00E-3	3.37E-7	1.00E+0	1.00E+0	2.28E-9
ACC-XHE-XD-MCR	4	1.00E-2	3.37E-6	1.00E+0	1.00E+0	2.28E-9
ACC-XHE-XE-6HR	11	5.00E-3	3.96E-6	1.00E+0	1.00E+0	5.35E-9
ACC-XHE-XE-TEST	11	5.00E-3	3.96E-6	1.00E+0	1.00E+0	5.35E-9
ACC-XHE-XE-TEST1	22	1.47E-1	7.91E-6	1.00E+0	1.00E+0	3.63E-10
ACC-XHE-XM-ISO	4	1.00E-3	3.37E-7	1.00E+0	1.00E+0	2.28E-9
ACC-XHE-XM-VALVE	22	1.00E-3	7.91E-6	1.00E+0	1.01E+0	5.35E-8
CRD-HCU-TM-REFUEL	58	2.00E-1	2.82E-4	1.00E+0	1.00E+0	9.52E-9
CRD-INS-FO-LIGHT	38	1.00E-3	6.89E-5	1.00E+0	1.07E+0	4.66E-7
CRD-PUMP-FO-DP	24	1.00E-3	5.56E-5	1.00E+0	1.06E+0	3.76E-7
CRD-XHE-XD-ERROR	10	1.00E-2	3.66E-6	1.00E+0	1.00E+0	2.47E-9
CRD-XHE-XD-MCR	8	1.00E-2	1.85E-4	1.00E+0	1.02E+0	1.25E-7
CRD-XHE-XM-MCR	21	1.00E-3	3.89E-5	1.00E+0	1.04E+0	2.63E-7
CRD-XHE-XM-OEP	8	1.00E-3	1.85E-4	1.00E+0	1.19E+0	1.25E-6
CVS-HTX-LK-RCP	19	1.00E-4	1.76E-4	1.00E+0	2.76E+0	1.19E-5
CVS-INS-FO-LEVEL	19	1.00E-3	3.42E-5	1.00E+0	1.03E+0	2.31E-7
CVS-MOV-CO-CHGA	9	2.00E-5	6.03E-7	1.00E+0	1.03E+0	2.04E-7
CVS-MOV-CO-CHGB	9	2.00E-5	6.03E-7	1.00E+0	1.03E+0	2.04E-7
CVS-XHE-XD-VCTLVL	19	1.00E-2	3.42E-5	1.00E+0	1.00E+0	2.31E-8
CVS-XHE-XM-DILUTION	24	1.00E-3	1.76E-3	1.00E+0	2.76E+0	1.19E-5
CVS-XHE-XM-ISO	18	6.00E-2	3.12E-3	1.00E+0	1.05E+0	3.52E-7
CVS-XHE-XM-MISALIGN	24	1.00E-3	1.76E-3	1.00E+0	2.76E+0	1.19E-5
CVS-XHE-XM-TEST	26	1.00E-3	3.43E-3	1.00E+0	4.42E+0	2.31E-5
HPI-MOV-CO-HPI	9	2.00E-5	6.03E-7	1.00E+0	1.03E+0	2.04E-7
IE-BWR-CRD1OUT	14	5.40E-1	1.33E-5	1.00E+0	1.00E+0	1.67E-10
IE-BWR-CRDHU	58	5.40E-1	2.82E-4	1.00E+0	1.00E+0	3.53E-9
IE-MAN-SD-FREQ	12	5.90E-1	9.69E-1	3.27E+1	1.67E+0	1.11E-5
IE-PWR-ACC	22	6.20E-1	7.91E-6	1.00E+0	1.00E+0	8.62E-11
IE-PWR-CVCS	67	6.20E-1	3.70E-3	1.00E+0	1.00E+0	4.03E-8
IE-PWR-RWST	45	6.20E-1	3.02E-6	1.00E+0	1.00E+0	3.29E-11
IE-PWR-SG	23	6.20E-1	2.66E-2	1.03E+0	1.02E+0	2.90E-7
LPI-MOV-CO-LPIA	9	2.00E-5	6.03E-7	1.00E+0	1.03E+0	2.04E-7
LPI-MOV-CO-LPIB	9	2.00E-5	6.03E-7	1.00E+0	1.03E+0	2.04E-7
MCR-XHE-XD-HIDP2	10	1.00E-2	3.66E-6	1.00E+0	1.00E+0	2.47E-9
MCR-XHE-XD-SRM	63	1.00E-2	8.77E-1	8.15E+0	8.78E+1	5.93E-4
MCR-XHE-XE-SRM	52	1.00E-3	8.77E-2	1.10E+0	8.86E+1	5.93E-4
MCR-XHE-XM-DIVERT	12	1.00E-2	9.69E-1	3.27E+1	9.70E+1	6.55E-4
MCR-XHE-XM-TRANS	4	2.00E-2	6.55E-3	1.01E+0	1.32E+0	2.21E-6
OPR-XHE-XE-CRD1	7	1.00E-3	6.66E-6	1.00E+0	1.01E+0	4.50E-8
OPR-XHE-XE-CRD1-DEP	7	1.44E-1	6.66E-6	1.00E+0	1.00E+0	3.13E-10
OPR-XHE-XE-IGNORELK	11	1.00E-3	5.21E-4	1.00E+0	1.52E+0	3.52E-6
OPR-XHE-XM-COMM	21	1.00E-3	3.89E-5	1.00E+0	1.04E+0	2.63E-7
OPR-XHE-XM-WJUMP	7	1.00E-3	6.66E-6	1.00E+0	1.01E+0	4.50E-8
OPR-XHE-XM-WJUMP-DEP	7	1.44E-1	6.66E-6	1.00E+0	1.00E+0	3.13E-10
RCS-CNF-CR-MNSD	84	1.00E-1	9.70E-1	3.30E+1	9.73E+0	6.55E-5

Basic Event Name	Num. of Occ.	Probability	Fussel-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
RCS-CNF-CR-PWR	134	2.50E-1	3.71E-3	1.00E+0	1.01E+0	1.00E-7
RCS-CNF-CR-SG	23	1.00E-2	2.66E-2	1.03E+0	3.63E+0	1.80E-5
RCS-DN-CRITICAL	139	9.00E-1	9.00E-1	1.00E+1	1.10E+0	6.76E-6
RCS-PN-CRITICAL	102	1.00E-1	1.00E-1	1.11E+0	1.90E+0	6.76E-6
RCS-WATER-ACCUMULATES	4	5.00E-2	1.64E-2	1.02E+0	1.31E+0	2.21E-6
RHR-MDP-FO-PWR	5	5.40E-4	1.99E-6	1.00E+0	1.00E+0	2.49E-8
RHR-XHE-XD-RHR	5	1.00E-2	1.99E-6	1.00E+0	1.00E+0	1.34E-9
RPS-CRD-TM-ONE	14	5.00E-1	1.33E-5	1.00E+0	1.00E+0	1.80E-10
RPS-ELC-FO-SCRAM	2	1.00E-6	8.73E-5	1.00E+0	8.83E+1	5.90E-4
RPS-MEC-FO-SCRAM	2	1.00E-6	8.73E-5	1.00E+0	8.83E+1	5.90E-4
RPS-XHE-XM-SCRAM	16	1.00E-4	8.73E-3	1.01E+0	8.83E+1	5.90E-4
RWS-MOV-OO-VALVE	5	1.00E-3	1.03E-7	1.00E+0	1.00E+0	6.98E-10
RWS-SYS-DL-OTHER	67	1.00E-2	1.09E-5	1.00E+0	1.00E+0	7.38E-9
RWS-SYS-FO-TEST	22	7.00E-5	2.40E-4	1.00E+0	4.42E+0	2.31E-5
RWS-XHE-XD-MCR	15	1.00E-2	1.35E-6	1.00E+0	1.00E+0	9.15E-10
RWS-XHE-XE-MONTH	10	1.00E-3	4.13E-7	1.00E+0	1.00E+0	2.79E-9
RWS-XHE-XE-TEST1	35	5.00E-3	2.60E-6	1.00E+0	1.00E+0	3.52E-9
RWS-XHE-XM-ISO	5	1.00E-3	1.03E-7	1.00E+0	1.00E+0	6.98E-10
SGS-SYS-LK-LEAK	23	1.40E-2	2.66E-2	1.03E+0	2.87E+0	1.28E-5
SGS-SYS-TM-SG	23	5.00E-1	2.66E-2	1.03E+0	1.03E+0	3.59E-7
SGS-XHE-XD-SGLEAK	12	5.00E-2	2.60E-2	1.03E+0	1.49E+0	3.52E-6
SRM-INS-FO-SRM1	20	1.00E-3	1.29E-5	1.00E+0	1.01E+0	8.69E-8
SRM-INS-FO-SRM2	20	5.00E-1	1.29E-5	1.00E+0	1.00E+0	1.74E-10
SRM-INS-FO-SRMS	18	1.00E-4	3.88E-5	1.00E+0	1.39E+0	2.62E-6
SRM-INS-FO-TEMP	2	1.00E-4	8.73E-7	1.00E+0	1.01E+0	5.90E-8

8. CONCLUSION AND RECOMMENDATIONS

Potential for inadvertent reactivity insertion during shutdown test and maintenance activities has been modeled and evaluated, using recent precursor events that occurred in various plants. For SD and refueling conditions, both the events of interest and the means of protection are very different from those when at power. Due to the low amount of energy stored in the fuel and the relatively low rate of reactivity insertion, a large early-release type of event that threatens the containment integrity may not be a major concern. However, reactivity controls are still important. During refueling, some of the normal barriers such as open containment to fission product release are removed to gain access to the core. The drywell and reactor pressure vessel are open to the containment and personnel are on the refueling floor above the reactor. Additionally, some of the automatic protection systems may not be available during refueling. In this situation, good engineering practices are relied upon to supplement the available automatic systems. The desired safety standards are achieved by using combinations of procedures and automatic systems to ensure that the reactor stays subcritical during all phases of refueling operations. Various approaches are used to meet this requirement, with administrative controls (written procedures) playing an important role in helping to maintain acceptable levels of risk.

Due to limited scope of this study, brief review of LER search resulted in few in advertent boron dilution events and one control rod withdrawal event. None of these incident resulted in recriticality events or any human injury or fuel damage. Detailed international data search needs to be conducted as in 2007 it was revealed that in Japan, there were ten inadvertent control rods withdrawal events during shutdown activities events took place in 1990s and some of them resulted in criticality. None of these incident resulted in recriticality events or any human injury or fuel damage.

During this study, two issues of possible further interest became apparent:

1) Lack of ET and sequence success criteria for what constitutes recriticality; for example how many rods need to withdrawn to what degree to declare a failed end state for a sequence. Note that such an end state may be a local exposure of workers at the rim above an open reactor cavity to radiation from a rapid recriticality.

2) Importance of multiple operator action failures, or commission errors that can lead to "failed end states". Quantification of the overall human error probability in a single sequence of such multiple errors may require additional analysis capabilities or assumptions to avoid non-conservative estimates.

It was not possible to build robust and generic train level fault tree models for the event tree top events as system P&IDs, power supplies; isometric drawings and procedures were not available. Some fault trees were built based on the similar studies performed in the past. Most operator action HEP are assumed to nominal as SPAR human reliability calculator is not designed to calculate the operator actions of commission or lack of communications.

Earlier analyses for studying re-criticality events using one dimensional codes seemed to suggest that an inadvertent withdrawal of as few as two control rods in a BWR may cause recriticality. However, later analyses with multidimensional codes seem to indicate that those previous analyses may be conservative. Since the fuel designs have changed in the last decades to accommodate longer fuel cycles and power uprates, the question of their effect on recriticality margins comes to mind. It may be useful to study the recriticality "failure criteria", for example, how many rods inadvertently drawn out by what degree would cause recriticality during shutdown operations. Such studies are outside the scope of a risk analyst, but are needed to determine the success criteria to model event sequences.

This report outlines and carries forward a risk analysis framework for potential recriticality accidents during shutdown operations, using precursor events already occurred in the plants. The next logical step could be to make reactivity analyses to determine realistic and current failure criteria for such sequences, to allow quantification of risk for specific sequences.

APPENDIX A. LER DATA FOR REACTIVITY INSERTION EVENTS

LER 2692002002

On July 11, 2002 an engineer referenced the Operations Emergency Operating Procedure (EOP) steps for alignment of an alternate post-LOCA Boron Dilution path if the primary path failed. He recognized that the guidance allowed alignment at Reactor Coolant System temperatures and pressures which might result in damage to the flow path. At 2202 hours on July 11, 2002 the NRC was notified pursuant to 10CFR50 72(b)(93)(v)(D). On July 12, 2002 the EOP was revised to require engineering evaluation of plant parameters prior to establishing the alternate flow path during any event. The root cause of this deficient procedure was inadequate design documentation which resulted in a deficient procedure change package for a revision approved December 20, 2001. This event is considered to have no significance with respect to the health and safety of the public.

LER 2822000002

On August 21, 2000, with the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 operating at 100% power, a representative of the current PINGP nuclear fuel assembly vendor informed PINGP staff by telephone that recent batches of fuel assemblies delivered to PINGP contained fuel pellets with nominal densities greater than the 95% of theoretical density assumed in the criticality analysis submitted in support of Amendments 129/121 to the PINGP Unit 1 and 2 Operating Licenses DPR-42 and DPR-60 . This increased nominal fuel pellet density results in the calculated 95/95 Kell exceeding the Technical Specification 5.6.A.1 .b limit (Kett <1 .0 if the spent fuel pool were fully flooded with unborated water) for fuel assemblies stored in both unrestricted and restricted (3x3 checkerboard) configurations. Actions have been taken to provide increased assurance that any significant dilution event will be detected before the boron concentration in the spent fuel pit could be reduced below 750 ppm. The investigation into the root causes of this event is still in progress and a final determination of the root causes and corrective actions has not been made.

LER 3052009008

On October 10, 2009, while the reactor was in refueling shutdown mode, the boron concentration of the reactor coolant system was inadvertently reduced below the 2500 ppm limit required by Kewaunee Power Station (KPS) Technical Specifications (TS), resulting in a violation of TS requirements. At the start of the event, the reactor vessel head was removed, the reactor was defueled, and the refueling cavity was flooded. Operators had begun diluting reactor coolant system (RCS) water in the residual heat removal (RHR) flow path to reduce the boron concentration in the refueling cavity. This activity inadvertently reduced the reactor coolant boron concentration in the RHR system below the minimum value required by TS. With RCS boron below the minimum TS limit, operators began transferring fuel assemblies from the spent fuel pool into the reactor. A sample of RCS water obtained from the RHR system shortly after commencement of refueling activities showed a boron concentration of about 2300 ppm. Upon confirmation of the boron concentration, fuel handling was suspended and boron concentration of the RCS was restored to required limits. This event is being reported pursuant to 10 CFR 50.73 (a)(2)(i)(B), any operation or condition which was prohibited by TS.

LER 3362008002

On April 13, 2008 at 0805 hours with the plant shutdown in MODE 6, while performing control room operator rounds, a Licensed Operator identified three charging pumps were aligned so they were capable of injection with the RCS temperature less than 300°F. The plant was in this configuration for approximately six hours. Upon discovery of this condition, TS Action Statement 3.1.1.3.b was immediately entered and then exited when the swing charging pump was made incapable of injection into the RCS. The cause of this event was determined to be inadequate configuration control because the tag for the charging pump removed from service to comply with the Boron Dilution TS 3.1.1.3.b. did not provide adequate guidance. Subsequent reconfiguration of the charging system to support electrical work activities aligned three charging pumps so they were capable of injection with the RCS temperature less than 300°F. Since the plant was in a configuration prohibited by the TS, this event is reportable in accordance with 10CFR50.73(a)(2)(i)(B) as any operation or condition prohibited by the plant's TS.

LER 3612008005

On June 9, 2008 at about 1314 hours PDT, San Onofre Nuclear Generating Station Unit 2 entered Mode 2 from Mode 3 in accordance with startup procedures. At about 1443 PDT, the Control Room Supervisor (CRS) recognized that the Control Element Assembly (CEA) [AA] Alignment Surveillance Requirements (SR) 3.1.5.1 and SR 3.1.5.2 had not been completed prior to Mode change. TS 3.1.5 is applicable in Modes 2 and 1, but not in Mode 3 or lower. Because SR 3.0.4 prevents entry into Modes without completion of all applicable SRs, SCE is reporting this occurrence in accordance with 10CFR50.73(a)(2)(i)(B). Operators completed the surveillances with satisfactory results. Consequently, there was no safety significance to this occurrence. This event was caused by (1) a lack of detail in the applicable procedure and (2) lack of oversight by the CR Supervisor (Utility, Licensed). SCE has coached the Operator's (Utility, Licensed) involved and will revise the affected procedure. LER 3-2006-005, dated January 26, 2007, reported an instance of exceeding 20% power without all required SRs completed. That event resulted from a boron dilution error during startup. LER 2-2007-002, dated August 17, 2007, reported a missed TS SR required with the plant entered Mode 3 following a failure of the instrument air system and consequential plant trip.

LER 3622006005

On December 11, 2006, plant operators were in the process of starting up Unit 3 following a refueling outage. To support main turbine testing, about 1954 hours PST, operators began raising power from about 15% to 18% using boron dilution. The two operators involved erred when determining the amount of demineralized water necessary. Consequently, at about 2017 hours PST power briefly exceeded 20%. Operations had not yet completed all the required surveillance testing for exceeding 20% power which violated Technical Specification Requirement (SR) 3.0.4. Operators returned power below 20% about 2024 hours PST. As a result, there is little or no safety consequence of this occurrence. The event occurred because operators estimated boron dilution based upon an extension of the dilutions for xenon control already in progress. Operators did not follow procedures which require that the amount of reactivity insertion be obtained from a controlled source during planned load changes. Additionally, the Shift Manager and the CR Supervisor did not ensure that a reactivity pre-job brief was conducted prior to adding the dilution water for the power increase, which likely would have prevented the error. SCE has taken corrective actions including a stand-down with each crew, enhancements to reactivity management requirements and appropriate disciplinary action.

LER 4132002002

On April 30, 2002, at approximately 0100 hours with Unit 1 in Mode 6, vital AC instrument panel board 1ERPD was removed from service for scheduled maintenance. Removing 1ERPD from service rendered train B of the Boron Dilution Mitigation System (BDMS) inoperable and it also disabled the Source Range Nuclear Instrument N-32 High Flux at Shutdown alarm function. Operations did not recognize that these functions were inoperable until approximately 0530 hours. TS 3.9.2 requires two trains of BDMS operable in Mode 6. If one train becomes inoperable, there are several required actions that have to be completed. Therefore, since the required actions of TS 3.9.2 were not completed within the time allowed, this is a condition prohibited by TS and reportable to the NRC as an LER. The root cause was less than adequate use of available information to properly identify the effects of isolating 1ERPD. Corrective actions include discussing this event with the personnel involved, development of Operations Management guidance, and reviewing this event in Licensed Operator Training. This event is considered to be of no significance with respect to the health and safety of the public.

LER 4562010002

At 0241 hours, Unit 1 received a volume control tank high level alarm, causing a loss of the boron dilution protection system (BDPS) function. Entry into TS Limiting Condition for Operation (LCO) 3.3.9 Conditions A and C for BDPS was required, which includes a one-hour Required Action to close non-borated water source isolation valves. This action was not completed within the one-hour requirement. The root causes were determined to be 1) the procedure for reactor trip response does not provide adequate guidance to alert operators to potential LCO 3.3.9 entry during plant transients, and 2) the BDPS annunciators are not adequately human factored. The corrective action to prevent recurrence is to revise the reactor trip response procedure to include guidance for potential LCO 3.3.9 entry. An additional corrective action is to evaluate human factors improvements for the plant annunciator system. There were no actual safety consequences impacting plant or public safety as a result of the event. This event is reportable under 10 CFR 50.73(a)(2)(i)(B), any operation or condition which is prohibited by the plant's TS.

LER 4832003004

On April 11, 2003 while at 100% power, it was discovered that a note contained in TS 3.3.9 for the BDMS had been inappropriately applied during past reactor startups. This had been interpreted to allow blocking BDMS while withdrawing SD Bank rods in Mode 3. This action is not allowed in Mode 3 per Final Safety Analysis Report (FSAR) accident analysis Section 15 .4 .6.2 where BDMS is credited for automatically terminating a dilution event while in Mode 3. Wording of TS 3.3.9 and TS 3.3.9 Bases did not provide clear guidance as to what constitutes reactor startup. The Bases indicate BDMS could be blocked prior to withdrawing rods for startup. These words do not delineate between control banks and SD banks. Based on this unclear guidance, procedure OTG-ZZ-0001A was incorrectly revised allowing the blocking of BDMS prior to withdrawing SD banks. The discovery of the unclear TS wording was the result of requested procedure enhancements to clarify when it was allowable to block BDMS. A review of reactor startups within the last three years indicated that BDMS was inappropriately blocked on three separate startups. The first occurred on November 24, 2002, the second on December 17, 2002, and the third on April 2, 2003. Plant procedures governing reactor startup were revised to remove statements allowing blocking BDMS while withdrawing SD Bank rods in Mode 3.

LER 5292008001

On May 21, 2008, plant staff noted that the Boron Dilution Alarm System (BDAS) had been in alarm (alarm window flashing) for approximately one hour when no actual dilution was occurring. When a BDAS channel is in an alarm condition, the channel loses its ability to alert the control room operators of an inadvertent boron dilution event until the channel is reset manually. Review of alarm logs indicated that during the period from May 8 - 21, 2008 when Unit 2 was in Modes 5 and 6, either a single BDAS channel or both channels were in alarm on multiple occasions for extended periods without being reset. During these periods, required TS 3.3.12 Actions A.1, B.1 and C.1 were not performed and TS LCO 3.0.4 was not met when the Operating Mode was changed from Mode 6 to Mode 5. The cause of the event was that licensed operators treated the alarm as a nuisance alarm and failed to reset the BDAS alarms in a timely manner. A night order was issued on May 22, 2008 as an immediate corrective action to direct timely reset of the BDAS alarms. The associated alarm response procedure was subsequently revised adding a note on the significance of the BDAS alarm function. No similar conditions have been by reported by PVNGS in the past three years.

LER 2062980034

While performing steam generator tube removal, unexpected water in the secondary side intruded into the reactor coolant system. A positive' 'reactivity insertion of 44c occurred as a result of a 35ppm boron I dilution. This event occurred without containment integrity as required. Investigation revealed an ongoing gradual RCS dilution. There was no degradation to public safety.

LER 2061982016

With the plant in cold SD (Mode 5), a freshly charged mixed-bed demineralizer was placed in service without boron saturating the new resin per the applicable procedure. This resulted in an RCS boron dilution of 211 ppm, causing a positive reactivity insertion prohibited by TS 3.6.1.8(3). A SD margin >10% was maintained throughout the event. There were no adverse effects on public health or safety.

LER 3100982049

At 0130 hours with the RCS partially drained in Mode 6, hydrolasing water filled the SG past the nozzle lip and may have diluted the RCS boron concentration. Conservative boron samples indicate a possible dilution of 107 ppm (+1.1% reactivity, TS 6.9.1.8.d). Shutdown margin remained greater than 22%. CEA's were inserted and fuel shuffle had not commenced.

LER 461982021

During the final stages of RCS cooldown for the Refueling Outage, the water injected to make up for the RCS inventory shrinkage was of a lower-than-expected boron concentration. This was due to a failure to completely close the demineralized water makeup valve. Although the reactor maintained at least 14% SD, TS 6.9.1.8.d requires a report be prepared whenever “an unplanned reactivity insertion of more than 0.5%4 K/K” occurs. The minimum boron concentration 'after the dilution was 1698 ppm, which is well above the 600 ppm minimum to maintain. The cause of the occurrence was a combination of personnel and procedure error. The operators involved were counseled and the event will be reviewed with all operators.

LER 2492008003

On November 3, 2008 at approximately 1036 hours CST with Unit 3 in a refuel outage, Dresden Nuclear Power Station operations personnel observed an unplanned withdrawal of control rod D-7. The control rod withdrawal stopped at position 06 with no actions taken by main control room personnel. An unplanned withdrawal of control rod E-6 to position 18 and control rod E-7 to position 16 also occurred and stopped with no actions by main control room personnel. All control rods were re-inserted to the full-in position per procedure on November 3, 2008, at approximately 1156 hours CST. The root cause of the unplanned control rod withdrawals is attributed to latent procedure deficiencies in DOP 0500-05, “Discharging CRD Accumulators with Mode Switch in Shutdown or Refuel,” Revision 4 that were not identified during an Operating Experience Review of the Significant Event Notification (SEN) 264, “Unplanned BWR Control Rod Withdrawals While Shutdown,” per procedure LS-AA-115, “Operating Experience Procedure.” Corrective actions to address this event include procedure revisions to DOP 0500-05 and LS-AA-115.

LER 2851993016

On November 13, 1993 a surveillance test was being performed on the Secondary Control Element Assembly Position Indication System (SCEAPIS), during a refueling outage, with the plant in Mode 5 (Refueling Shutdown). At approximately 2200 hours, a CEA began to withdraw from the reactor core. A “Continuous Rod Motion” alarm was received, and a licensed operator placed the CEA mode selector switch in Off, which prevents CEA movement. During troubleshooting, Rod 31 became fully withdrawn. Rod 31 was then driven back into the core and the mode selector switch returned to Off. The reactor was then manually tripped to ensure rods were at their lower stop and to return the system to its pretest configuration. Troubleshooting identified multiple electrical grounds associated with control rod drive and position indication circuitry, and a wiring deficiency. These conditions resulted in an unintended electrical circuit that energized the raise contactor for Rod 31. The root cause of the event was determined to be the lack of a ground detection system for the associated power supplies.

LER 3111993008

During Unit 2 Reactor startup activities, following the unit's seventh refuel outage, it was determined on June 4, 1993 that a postulated single failure concern existed where failure of one Rod Control System slave cycler decoder card, in conjunction with a rod motion command signal, may cause an unplanned Rod Control Cluster Assembly (RCCA) withdrawal. At 1734 hours, all control rods were inserted, the reactor trip breakers were opened, and the Unit was stabilized in MODE 3. On May 27, 1993, at 1844 hours, rod 1SA3 had withdrawn approximately 15 steps from fully-inserted following a manual insertion command. Rod control power was then deenergized to fully insert the rod. The RCS single failure concern is attributed to RCS design. 1SA3 withdrew as the result of inappropriate current orders to the RCCA. Integrated circuit chips on two slave cycler decoder cards had failed due to the relay driver circuit card connector Pin No. 4 not making electrical contact with the surge suppression diode. Pin No. 4 was repaired and the slave cycler cards were replaced to restore operability of rod 1SA3. An additional corrective action was installation of suppression diodes on the rod step counters of the RCS circuitry of each unit to mitigate consequences of an open or bad connection on the relay driver circuit card connector pin No. 4. All Unit 2 RCS logic cards were replaced and satisfactorily tested and all RCS Power Cabinet cards were pulled, visually inspected, and retested satisfactorily.

LER 2551980031

During performance of control rod interlock testing, group 1 rods withdrew to approximately 121 inches. This resulted in unplanned reactivity insertion of 1.6 delta K/K, and is reportable per TS 6.9.2.A. During the period that rods were withdrawn, SD margin requirements were met, and the reactor remained sub-critical. Cause of excessive withdrawal was loss of both Primary and secondary data loggers (see LER 80-32).

APPENDIX B. HRA TABLES 6-1 AND 6-2

Table 6-1. Reactivity Insertion Accident Analysis - Human Action Summary

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
ACC-XHE-XD-MCR	RANDOM	MCR OPERATOR FAILS TO DIAGNOSE THE BORON DILUTION EVENT	Log Normal	1.0E-2		
		Diagnosis is modeled.		1.0E-2		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
ACC-XHE-XE-6HR	RANDOM	FAILURE TO CHECK CONC. SIX HRS AFTER ACCUMULATOR FILL	Log Normal	5.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Highly complex	100%	5.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
ACC-XHE-XE-MONTHLY	RANDOM	FAILURE TO PERFORM MONTHLY TEST	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
		Dependency is not modeled.				
ACC-XHE-XE-TEST	RANDOM	OPERATOR MADE ERROR IN TESTING	Log Normal	5.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Highly complex	100%	5.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
ACC-XHE-XE-TEST1	RANDOM	OPERATOR MADE AN ERROR IN SCHEDULED TESTING	Log Normal	1.5E-1		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Highly complex	100%	5.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is modeled.		Dep. = (1+6*P)/7	Moderate Dependenc e	
					Different Crew, Close in Time, Same Location, No Additional Cues	
ACC-XHE-XM-ISO	RANDOM	OPERATOR FAILS TO TERMINATE BORON DILUTION EVENT	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
ACC-XHE-XM-PWR	RANDOM	OPERATOR FAILS TO ISOALTE POWER	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
ACC-XHE-XM-VALVE	RANDOM	MCR OPERATOR INADVERTENTLY OPENS ACC VALVE TO RCS	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
CRD-XHE-XD-ERROR	RANDOM	FIELD OPERATOR FAILS TO DIAGNOSE UNUSUAL DP	Log Normal	1.0E-2		
		Diagnosis is modeled.		1.0E-2		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
CRD-XHE-XD-MCR	RANDOM	MCR OPERATOR FAILS TO DIAGNOSE EXCESS DELTA-P	Point Value	1.0E-2		
		Diagnosis is modeled.		1.0E-2		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
CRD-XHE-XM-MCR	RANDOM	MCR OPERATOR FAILURE TO FOLLOW PROCEDURE	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
CRD-XHE-XM-OEP	RANDOM	FIELD OPERATOR FAILS TO FOLLOW PROCEDURE	Point Value	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
CVS-XHE-XD-VCTLVL	RANDOM	OPERATOR FAILURE TO NOTICE INCREASED VCT LEVEL	Log Normal	1.0E-2		
		Diagnosis is modeled.		1.0E-2		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
CVS-XHE-XM-DILUTION	RANDOM	OPERATOR INADVERTENTLY DILUTE RCS WATER BELOW THE DESIRED LIMIT	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
CVS-XHE-XM-ISO	RANDOM	OPERATOR FAILS TO TERMINATE BORON DILUTION EVENT FROM CVCS INJECTION	Log Normal	6.0E-2		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Low	100%	3.0	
		Procedures	Incomplete	100%	20.0	
		Ergonomics/HMI	Nominal	100%	1.0	

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
CVS-XHE-XM-MISALIGN	RANDOM	OPERATOR INADVERTENTLY MISALIGN THE VALVES	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
CVS-XHE-XM-TEST	RANDOM	OPERTOR FAILS TO TEST RCS WATER CHEMISTRY	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
MCR-XHE-XD-HIDP2	RANDOM	MCR OPERATOR FAILURE TO DIAGNOSE HI-DP INDICATION (NOT INFORMED)	Log Normal	1.0E-2		
		Diagnosis is modeled.		1.0E-2		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
MCR-XHE-XD-SRM	RANDOM	OPERATOR FAILS TO DIAGNOSE HI-COUNTS	Log Normal	1.0E-2		
		Diagnosis is modeled.		1.0E-2		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
MCR-XHE-XE-SRM	RANDOM	OPERATOR IGNORES THE SRM INDICATIONS/MISCOMUNICATES	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
MCR-XHE-XM-DIVERT	RANDOM	MCR OPERATOR ATTENTION IS DIVERTED	Log Normal	1.0E-2		
		Action is modeled.		1.0E-3		
		Available Time	Just enough time	100%	10.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
MCR-XHE-XM-FUELOUT	RANDOM	OPERATOR FAILS TO STOP INSERTION OR PULL FUEL BACKUP	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
MCR-XHE-XM-TRANS	RANDOM	OPERATOR TERMINATES TRANSIENT	Log Normal	2.0E-2		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Incomplete	100%	20.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
OPR-XHE-XE-CRD1	RANDOM	OPERATOR FAILS TO VERIFY ONE CRD OUT	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
OPR-XHE-XE-CRD1-DEP	RANDOM	SECOND OPERATOR FAILS TO VERIFY	Log Normal	1.4E-1		
		Action is modeled.		1.0E-3		

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is modeled.		Dep. = (1+6*P)/7	Moderate Dependence	
					Different Crew, Close in Time, Same Location, No Additional Cues	
OPR-XHE-XE-FLOAD	RANDOM	OPERATOR-1 FAILS TO CHECK FUEL LOADING PATTERN	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
OPR-XHE-XE-FLOAD-DEP	RANDOM	OPERATOR-2 FAILS TO CHECK FUEL LOADING PATTERN	Log Normal	1.4E-1		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is modeled.		Dep. = (1+6*P)/7	Moderate Dependenc e	
					Different Crew, Close in Time, Same Location, No Additional Cues	
OPR-XHE-XE-IGNORELK	RANDOM	OPERATR IGNORES SG LEAK	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
OPR-XHE-XE-PATTERN	RANDOM	LOADING PATTERN ERROR	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
OPR-XHE-XM-COMM	RANDOM	FIELD SUPERVISSOR FAILED TO NOTIFY MCR	Log Normal	1.0E-3		

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
		OPERATOR				
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
OPR-XHE-XM-WJUMP	RANDOM	OPERATOR REMOVES WRONG JUMPER	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
OPR-XHE-XM-WJUMP-DEP	RANDOM	SECOND TECH FAILS TO VERIFY	Log Normal	1.4E-1		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is modeled.		Dep. = (1+6*P)/7	Moderate Dependence	
					Different	

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
					Crew, Close in Time, Same Location, No Additional Cues	
RHR-XHE-XD-RHR	RANDOM	OPERATOR FAILS TO DAIGNOSE LOSS OF RHR	Log Normal	1.0E-2		
		Diagnosis is modeled.		1.0E-2		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
RPS-XHE-XM-SCRAM	RANDOM	OPERATOR FAILS TO SCRAM THE REACTOR	Log Normal	1.0E-4		
		Action is modeled.		1.0E-3		
		Available Time	Extra time	100%	0.1	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
RWS-XHE-XD-MCR	RANDOM	MCR OPERATOR FAILS TO DIAGNOSE THE BORON DILUTION EVENT FROM RWST	Log Normal	1.0E-2		
		Diagnosis is modeled.		1.0E-2		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
RWS-XHE-XE-MONTH	RANDOM	FAILURE TO PERFORM MONTHLY TEST	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
RWS-XHE-XE-TEST	RANDOM	OPERATOR MADE AN ERROR IN TESTING	Log Normal	5.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Highly complex	100%	5.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
RWS-XHE-XE-TEST1	RANDOM	OPERATOR MADE AN ERROR IN SCHEDULED TESTING	Log Normal	5.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Highly complex	100%	5.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	

HRA Basic Event Name	Model Type	Event Description/Shaping Factor	Distribution Type or PSF	Probability or Percentage	Initial or Multiplier	Basic Event Notes
		Dependency is not modeled.				
RWS-XHE-XM-BLEND	RANDOM	OPERATOR ERROR IN MIXING BLENDER	Log Normal	5.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Highly complex	100%	5.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
RWS-XHE-XM-ISO	RANDOM	OPERATOR FAILS TO TERMINATE BORON DILUTION EVENT FROM RWST INJECTION	Log Normal	1.0E-3		
		Action is modeled.		1.0E-3		
		Available Time	Nominal time	100%	1.0	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Nominal	100%	1.0	
		Procedures	Nominal	100%	1.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				
SGS-XHE-XD-SGLEAK	RANDOM	FAILURE TO DIAGNOS LEAK FROM SG TO RCS	Log Normal	5.0E-2		
		Diagnosis is modeled.		1.0E-2		
		Available Time	Extra time	100%	0.1	
		Stress/Stressors	Nominal	100%	1.0	
		Complexity	Nominal	100%	1.0	
		Experience/Training	Low	100%	10.0	
		Procedures	Available, but poor	100%	5.0	
		Ergonomics/HMI	Nominal	100%	1.0	
		Fitness for Duty	Nominal	100%	1.0	
		Work Processes	Nominal	100%	1.0	
		Dependency is not modeled.				

Table 6-2. Reactivity Insertion Accident Analysis - Dependent Human Actions

Original Events	Dependent Events	Dependent Value
RCI-XHE-XO-ERROR * HCI-XHE-XO-ERROR	HCI-XHE-XO-ERROR1	1.4E-1
CDS-XHE-XO-ERROR * OPR-XHE-XM-ALPI	OPR-XHE-XM-ALPI4	1.4E-1
CR1-XHE-XM-VLVS * OPR-XHE-XM-ALPI1	OPR-XHE-XM-ALPI5	1.4E-1
CR1-XHE-XM-VLVS * OPR-XHE-XM-ALPI2	OPR-XHE-XM-ALPI5	1.4E-1
RCI-XHE-XL-START * HCI-XHE-XL-START	HCI-XHE-XL-START1	5.4E-1
RCI-XHE-XL-START * HCI-XHE-XL-RUN	HCI-XHE-XL-RUN1	8.2E-1
RCI-XHE-XL-RUN * HCI-XHE-XL-START	HCI-XHE-XL-START1	5.4E-1
RCI-XHE-XL-RUN * HCI-XHE-XL-RUN	HCI-XHE-XL-RUN1	8.2E-1
RCI-XHE-XL-RSTRT * HCI-XHE-XL-START	HCI-XHE-XL-START1	5.4E-1
RCI-XHE-XL-RSTRT * HCI-XHE-XL-RUN	HCI-XHE-XL-RUN1	8.2E-1
RCI-XHE-XL-XFER * HCI-XHE-XL-START	HCI-XHE-XL-START1	5.4E-1
RCI-XHE-XL-XFER * HCI-XHE-XL-RUN	HCI-XHE-XL-RUN1	8.2E-1

APPENDIX C. TABLES OF INFORMATION FROM LITERATURE

Table C-1. Probabilistic Profile for Reactivity Accidents

Event	Consequences[a]	Probability[b]	Notes
Design-Basis Events and Extensions			
RDA (BWR)	No fuel damage	<1.0E-7	
REA (PWR)	No fuel damage	<1.0E-6	Fuel damage possible if criterion changes
Beyond-design-basis REA	Fuel damage	<1.0E-7	
Boron Dilution Events			
Reflux condensation during small LOCA	Fuel damage	No published results	Fuel damage dependent on many assumptions
Reactor restart scenario	Fuel damage	<1.0E-5	Many conservative assumptions
Boration after shutting off RCPs	Unknown	Unknown	Detailed analysis not done
SG tube rupture, backfill cooldown and start of RCP	Fuel damage	~1.0E-7	Two analyses done
Dilution during RCS filling	Fuel damage	<1.0E-6	Analysis plant dependent
Secondary water enters RCS and start of RCP	Fuel damage	<1.0E-7	Events that could be eliminated by procedures
Other events (9) in Table 2		<1.0E-7	Insignificant events
Events During Refueling			
BWR refueling accident	Fuel damage	<1.0E-7	Reduced frequency due to Tech Spec changes
PWR refueling accident	Fuel damage	<1.0E-6	Conservative assumptions used
Events Without Reactor Trip			
BWR ATWS events	No fuel damage	~1.0E-5	Fuel damage possible if criterion changes
LOCA with reflood water not borated	Fuel damage	<1.0E-6	Probability of control rod failure unknown

Notes
[a] The criterion for fuel damage in this column is a fuel enthalpy of 280 cal/g.
[b] Frequency of Occurrence $(RY)^{-1}$

Diamond 1998 (A Probabilistic Profile for Reactivity Accidents)

Table C-2. Examples of Boron Dilution Events

Rapid Boron Dilution Events	
1.	Reflux condensation during a small-break loss-of-coolant accident causes slug of diluted water to accumulate in the cold leg loop seals. Loop seal clearing or re-establishment of natural circulation causes diluted slug to move into the core.
2.	Loss of offsite power during boron dilution at startup causes trip of the reactor coolant pumps but charging pumps continue to pump diluted water from the volume control tank into the vessel. When power is restored RCPs may restart causing slug of diluted water to move into the core.
3.	At the beginning of a refueling shutdown if the RCPs are tripped before refueling boron concentration is reached there is a potential for a slug of diluted water to form.
4.	Steam generator tube rupture, backfill cooldown and restart of an RCP.
5.	The reactor coolant system (RCS) is being filled and diluted water inadvertently gets in through the RCP seals. When the RCPs are started the diluted slug moves into the core.
6.	Secondary water enters the RCS through leaking steam generator (SG) tubes and/or SG maintenance during shutdown. When the RCPs are started the diluted slug moves into the core.
7.	Loss-of-coolant accident (LOCA) with SG tube rupture and secondary diluting primary.
8.	LOCA with diluted emergency core cooling water.
9.	LOCA with sump water diluted.
10.	Inadvertent actuation of safety injection during shutdown when the refueling water storage tank (RWST) is diluted.
11.	Blowdown of an accumulator during shutdown with accumulator water diluted due to errors in maintenance.
12.	Startup of the residual heat removal system (RHR) using unborated water.
Slow Boron Dilution Events	
13.	Leakage of accumulator water (inadvertently diluted) during shutdown.
14.	Leakage of RWST water (inadvertently diluted) during shutdown.
15.	Failures in the chemical and volume control system which lead to uncontrolled boron dilution.

Diamond 1998 (A Probabilistic Profile for Reactivity Accidents)

Table C-3. Unexpected Criticality Events at Commercial BWRs

No	Date of occurrence	Country and unit	Operating condition	Event convergence measure	Title of case	Outline	Source
1	1973/11/7	Vermont Yankee (GE-BWR, USA)	During low-temperature shutdown Reactor vessel and reactor containment were open	Scram	The reactor went critical due to carelessness	When the adjacent control rods were fully withdrawn, the control rod was withdrawn and then the criticality event occurred due to carelessness. Through the intermediate region monitor (IRM) high high signal, the reactor scram took place and the output rise was stopped.	NRC IN88-21 1988/5/9
2	1976/11/12	Millstome -1 (GE-BWR, USA)	During low-temperature shutdown	Scram	The reactor went critical due to carelessness	During the shutdown margin test of the partially loaded reactor core, the operator withdrew the adjacent control rod by mistake, and then the criticality event occurred due to carelessness. The reactor went critical and the scram took place.	NRC IN88-21 1988/5/9
3	1987/7/24	OSKARSHAMN-3 (ASEA_Atom-BWR, Sweden)	During low-temperature shutdown	Insertion of control rods by the motor-driven insertion system	The reactor shutdown margin test while the hydraulic main scram system was not working	During the shutdown margin test, the unplanned criticality event occurred. The shutdown margin test was conducted in spite of the fact the fast activated hydraulic scram system was known to be not operational. When the first control rod was partially withdrawn, the reactor core reached criticality. The high flux signal stopped the control rod withdrawal. The control rod was re-inserted by the slowly-activated motor-driven insertion system.	NRC IN88-21 1988/5/9
4	1991/6/6	Monticello (GE-BWR, USA)	During operation for shutdown	Scram	Unexpected return to criticality during reactor shutdown	When the control rod was inserted during the reactor shutdown, the pressure and the temperature started decreasing unexpectedly. The operator stopped insertion of the control rod to inspect and evaluate the plant state, and then the reactor output increased. As a result, the intermediate region monitor (IRM) high high signal was issued and the reactor scram took place.	NRC IN92-39 1992/5/13
5	1991/11/30	Big Rock Point (GE-BWR, USA)	During operation for shutdown	Implemented the subcriticality measures	Unexpected return to criticality during reactor shutdown	During the planned shutdown, the control rod insertion was stopped for the period of takeover of operator shift with the main turbine disconnected and the reactor in the subcritical state. Then the reactor cooling system temperature continued decreasing and caused the reactor to go critical again. However, the operator took the subcriticality measure about 2 minutes later.	NRC IN92-39 1992/5/13

6	1991/12/30	Grand Gulf (GE-BWR, USA)	During operation for shutdown	Control rod insertion operation	Unexpected return to criticality during reactor shutdown	When the control rod temperature decrease occurred during the reactor shutdown, unexpected criticality occurred. Finally the control rod insertion was restarted and the reactor was shut down.	NRC IN92-39 1992/5/13

Table C-4. List of Control Rod Withdrawal Events in Japanese BWR

	Shika Unit 1	**Fukushima Daiichi Unit 3**	**Fukushima Daiichi Unit 5**	**Fukushima Daiichi Unit 2**	**Onagawa Unit 1**	**Hamaoka Unit 3**	**Fukushima Daini Unit 3**	**Kashiwazaki Kariwa Unit 6**	**Fukushima Daiichi Unit 4**	**Kashiwazaki Kariwa Unit 1**
Date of occurrence (Date of commissioning)	June 18, 1999 (July 30, 1993)	November 2, 1978 (March 27, 1976)	February 12, 1979 (April 18, 1978)	September 10, 1980 (July 18, 1974)	July 9, 1988 (June 1, 1984)	May 31, 1991 (August 28, 1987)	June 15, 1993 (June 21, 1985)	June 10, 1996 (November 7, 1996)	February 22, 1998 (October 12, 1978)	April 7, 2000 (September 18, 1985)
Outline of event	The criticality incident occurred due to control rod withdrawal.	The critical state occurred due to control rod withdrawal.	Control rod withdrawal occurred, but the neutron flux is considered to have remained unchanged.	Control rod withdrawal occurred, but the neutron flux is considered to have remained unchanged.	Control rod withdrawal occurred, but the neutron flux remained unchanged.	Control rod withdrawal occurred, but the neutron flux remained unchanged.	Control rod withdrawal occurred, but the neutron flux remained unchanged.	Control rod withdrawal occurred, but the neutron flux remained unchanged.	Control rod withdrawal occurred, but the neutron flux remained unchanged.	Control rod withdrawal occurred, but the neutron flux remained unchanged.
Reactor manufacturer	Hitachi	Toshiba	Toshiba	GE/Toshiba	Toshiba	Toshiba	Toshiba	GE/Toshiba	Hitachi	Toshiba
Plant state	During shutdown	During shutdown	During shutdown	During shutdown	During shutdown	During shutdown	During shutdown	During shutdown	During shutdown	During shutdown
State of reactor pressure vessel head	Open	Closed	Open	Closed	Closed	Open	Closed	Open	Closed	Closed
State of reactor containment head	Open	Open	Open	Closed	Closed	Open	Closed	Open	Open	Open
Details of work	During the work for preparation of the function verification test as an incident management measure	During the work in association with the reactor pressure vessel hydraulic test	During the work for preparation before the in-core shipping (under the isolation of HCU)	During the preparatory inspection of containment isolation system function	During the work for preparation of starting up the reactor	During the work for completion of the preparatory inspection for confirmation of the reactor protection system set values	During the preparation for the preparatory inspection of the reactor containment leakage rate	During the scheduled shutdown in the commissioning period before commercial operation	During the pressure test of the reactor pressure vessel	During the preparation for the preparatory inspection of the reactor containment leakage rate
Work being performed	While closing the isolation valve for other control rods than the	While closing the control rod isolation valve	While closing the control rod isolation valve	During isolation of the control rod isolation valve (not during the valve	While opening the fully-closed control rod isolation valve	While opening the fully-closed control rod isolation valve	While closing the fully-opened control rod isolation valve	During the performance verification test of the constant power controller (APR)	At mistakenly turning on the drive power supply for valve operation	While closing the fully-opened control rod isolation valve

	Shika Unit 1	Fukushima Daiichi Unit 3	Fukushima Daiichi Unit 5	Fukushima Daiichi Unit 2	Onagawa Unit 1	Hamaoka Unit 3	Fukushima Daini Unit 3	Kashiwazaki Kariwa Unit 6	Fukushima Daiichi Unit 4	Kashiwazaki Kariwa Unit 1
Date of occurrence (Date of commissioning)	June 18, 1999 (July 30, 1993)	November 2, 1978 (March 27, 1976)	February 12, 1979 (April 18, 1978)	September 10, 1980 (July 18, 1974)	July 9, 1988 (June 1, 1984)	May 31, 1991 (August 28, 1987)	June 15, 1993 (June 21, 1985)	June 10, 1996 (November 7, 1996)	February 22, 1998 (October 12, 1978)	April 7, 2000 (September 18, 1985)
	control rod under test			operation)						
Number of withdrawn control rods (Total number of control rods)	3 (89)	5 (137)	1 (137)	1 (137)	2 (89)	3 (185)	2 (185)	4 (205)	34 (137)	2 (185)
Position of withdrawn control rod (position) *1	26-39 (16) 30-39 (20) 34-35 (08)	14-43 (04) 18-43 (06) 22-43(10) 22-47 (12) 46-43 (08)	42-31 (28)	18-47 (20)	38-11 (02) 26-03 (18)	30-55 (06) 38-03 (48) 42-03 (16)	50-31 (22) 54-31 (12)	10-23 (128) 10-55 (128) 18-15 (128) 26-39 (128)	34 (02)	18-55 (24) 22-55 (10)
Accumulator state	Disabled	Enabled	Unknown (No scram signal occurred)	Unknown (No scram signal occurred)	Enabled	Enabled	Disabled	Enabled	Enabled	Disabled
Open/close state of reactor return valve	Closed	Closed	Closed	Closed	Closed	Closed	Closed	No target valve (due to the electric control rod drive mechanism)	Closed	Closed
Record of event	Falsified/ concealed	Falsified	Recorded properly	Recorded properly	Recorded properly	Recorded properly	Recorded properly	Recorded properly	Recorded properly	Recorded properly
Investigation of cause and prevention of recurrence	Not implemented	Seems to have been implemented at the same time with the measures for the subsequent case	Seems to have been implemented	Seems to have been implemented	Implemented	Implemented	Implemented	Implemented	Implemented	Implemented
Summary of event	Event during isolation of the hydraulic control unit (HCU)	Event during isolation of the hydraulic control unit (HCU)	Event during isolation of the hydraulic control unit (HCU)	Event during isolation of the hydraulic control unit (HCU)	Event during isolation of the hydraulic control unit (HCU)	Event during isolation of the hydraulic control unit (HCU)	Event during isolation of the hydraulic control unit (HCU)	Operational error of the control rod drive power supply	Operational error of the safety relief valve power supply	Event during isolation of the hydraulic control unit (HCU)

LAST PAGE