INL/EXT-20-60104
Revision 0

Light Water Reactor Sustainability Program

Flexible Plant Operation and Generation Probabilistic Risk Assessment of a Light Water Reactor Coupled with a High-Temperature Electrolysis Hydrogen Production Plant

October 2020

U.S. Department of Energy

Office of Nuclear Energy

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Probabilistic Risk Assessment of a Light Water Reactor Coupled with a High-Temperature Electrolysis Hydrogen Production Plant

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October 2020

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy

EXECUTIVE SUMMARY

Two generic probabilistic risk assessments (PRA) are performed for the addition of a heat extraction system (HES) to a light water reactor (LWR)—one for a pressurized-water reactor (PWR) and one for a boiling water reactor (BWR). The results investigate the applicability of the potential licensing approaches which might not require a full U.S. Nuclear Regulatory Commission (NRC) licensing amendment review (LAR). The PRAs are generic, and therefore some assumptions are made to preserve generality. Many conservative assumptions from the preliminary PWR PRA report were eliminated using design data for both the HES and the high-temperature electrolysis facility (HTEF). The results of the PRA indicate that application using the licensing approach in 10 CFR 50.59 is justified because of the minimal increase in initiating event frequencies for all design basis accidents (DBAs), none exceeding 5.6%. The PRA results for core damage frequency (CDF) and large early release frequency (LERF) support the use of Regulatory Guide 1.174 as further risk information that supports a change without a full LAR. Further insights provided through hazard analysis and sensitivity studies confirm with high confidence that the safety case for licensing an HES addition and an HTEF sited at 1.0 km from the nuclear power plant is strong and that the placement of an HTEF at 0.5 km is a viable case. Site-specific information can alter these conclusions.

ACKNOWLEDGEMENTS

The authors of this report would like to thank Austin Glover and his team at Sandia National Laboratory for outstanding support in the hydrogen plant safety analysis leak rates, frequencies, and overpressure estimates. We also would like to thank the heat extraction modeling team, especially Stephen Hancock and Tyler Westover for tremendous support. Many thanks also to the participants in the Failure Modes and Effects Analysis team, and especially to our PRA guru, Bob Buell.

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ACRONYMS

AFW	Auxiliary feedwater
ATWS	Anticipated Transient Without Scram
BWR	Boiling water reactor
CCF	Common Cause Failures
CCG	Common Cause Group
CDF	Core damage frequency
CFR	Codes of federal regulations
CST	Condensate storage tanks
DBA	Design basis accidents
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
HES	Heat extraction system
HPI	High Pressure Injection
HTEF	High-temperature electrolysis facility
HTF	Heat-transfer fluids
IE	Initiating event
IPEEE	Individual Plant Examination of External Events
LAR	Licensing amendment review
LERF	Large early release frequency
LOOP	Loss of switchyard components means a loss-of-offsite-power
LPI	Low-pressure injection
LWR	Light water reactor
LWRS	Light Water Reactor Sustainability
MCA	Maximum Credible Accident
MLOCA	Medium-Size-Loss-of-Coolant-Accident
MSIV	Main steam isolation valves
NPP	Nuclear power plants
NRC	Nuclear Regulatory Commission
P&ID	Piping and instrumentation diagram
PCS	Power Conversion System
PRA	Probabilistic risk assessment
PSF	Performance Shaping Factors
PWR	Pressurized-water reactor

RCP	Reactor Coolant Pump
RPN	Risk priority number
RPS	Reactor Protection System
RWST	Refueling water storage tanks
SBO	Station blackout
SME	Subject matter experts
SNL	Sandia National Laboratory
SSC	Structures, systems, and components
TPD	Thermal power delivery
UFSAR	Updated final safety analysis report

1. OVERVIEW

1.1 Purpose

Penetration of variable renewable power plants and low natural gas prices are threatening the profitability of already existing, paid off, nuclear power plants (NPPs). The Nuclear Energy Institute (NEI) [1] reported that the total generating cost for nuclear energy of existing LWR plants in 2017 was $33.50/MWh, This relatively low operating cost is quite competitive to other energy sources. However, there are other economic factors that need to be considered due to the intrinsic nature of the LWR power generation process. The LWR NPPs are typically run at full power during unfavorable over-supply electric market situations caused by fair weather and low electricity demands. This is caused by the need to avoid reactor shutdowns which lead to time delays in restarting. On the other hand, NPPs generally have superior reliability which allows operators to continue running them without frequent shutdowns. As a result, while the current LWR fleet consists of 10% of the operating capacity of electricity generation, it is consistently run at a much higher capacity than other technologies and provides 20% of the electricity sold in the U.S. This is one of the benefits NPPs provide to the electric grid, which is not adequately compensated, thereby disrupting their finances and sustainability in operating in such a baseload manner. During these times, NEI reports [1] that NPP operators only recoup the U.S. government subsidy of $23/MWh, essentially causing operators to pay for the electricity they create. No substantial governmental policy has been put into place to support the sustainable operation of NPPs as reliable baseload providers.

To increase the utility and profitability of the current fleet of LWR NPPs, the Light Water Reactor Sustainability (LWRS) Program is evaluating the feasibility of using part of the heat from an NPP for use in other industrial applications. Steel manufacturing, chemical processing, desalination, and hydrogen production are examples of industrial applications that could utilize heat from an LWR NPP. The co-located industrial facility will benefit from lower cost process heat and the NPP will benefit from a steadier income from its consistent production of energy. The feasibility of installing a modification of an LWR NPP to export process heat to an industrial facility is broken into two parts: economic viability and the safety case. The economic benefit will determine if the modification is desired. The safety case will determine if the modification is allowed through licensing by the U.S. Nuclear Regulatory Commission (NRC). This report concentrates on the probabilistic safety case of the use of LWR-extracted heat in hydrogen production by electrolysis of water. Hydrogen production is chosen because of the large demand for hydrogen across various markets and the added benefit of less carbon in the hydrogen production cycle. Currently, most of the commercial hydrogen produced uses steam methane reforming, which utilizes natural gas as a source of hydrogen and produces CO_2 as waste. Electrolysis utilizes water as the source of hydrogen.

For the suggested change to the LWR design and operation to be approved, the NRC requires a demonstration that the safety of the NPP will not be affected adversely. Probabilistic risk assessment (PRA) is used to risk-inform the decision for change acceptance by the NRC. PRA is a process by which risk is numerically estimated by computing probabilities of what can go wrong and the consequences of those undesired events. The quantitative results of the PRA are compared to guidelines set by the NRC which determine if the design and operation are safe enough for approval or if changes need to be made to increase its safety.

1.2 Background

A PRA for an NPP is broken into three levels. A Level 1 PRA estimates the frequency per year of accidents damaging the reactor core, referred to as core damage frequency (CDF). This is done using two types of logical structures—event trees (ETs) and fault trees (FTs). An ET represents the possible pathways that can occur due to an undesired outcome. The initial undesired event is called an initiating event (IE). After the IE, the ET uses the results of FT models representing responding systems that prevent core damage. These FTs are known as the top events of the ET. The event tree sequences of events result in end states which are indicative of the state of the reactor. The end state of interest is core damage. All basic events of component or human action failures have associated probabilities of failure that are used in relation to one another as defined by the logic trees. The sum of the probabilities associated to all the sequences leading to the core damage end state, represent the CDF.

Top-down methods are typically used to define IE frequencies. This uses data of recorded events to calculate the event frequency.

The probability of failure for top events of FTs are calculated using a bottom-up method. Bottom-up methods rely on knowing the exact componentry and controls of a system, that are then translated into a FT. Typically, this is accomplished by referencing a piping and instrumentation diagram (P&ID) of the system and a list of operator actions, then identifying how each of those components and/or actions could fail in a way that leads to a failure event in the ET. The FTs are created and integrated into ETs by identifying within what IE the system failure would be used either as an initiator itself or as a modification to one of the responding systems.

2. OBJECTIVE

The objective of this PRA is to further refine and expand upon the preliminary generic LWR pressurized-water reactor (PWR) PRA presented in INL/EXT-19-55884, "Preliminary Probabilistic Risk Assessment of a Light Water Reactor Supplying Process Heat to a Hydrogen Production Plant" [2] and remove as many conservatisms and assumptions as possible. This PRA includes both boiling water reactor (BWR) and PWR generic models to provide an example for starting a site-specific PRA for the purpose of pursuing a licensing pathway with the NRC using 10 CFR 50.59,"Changes, Tests, and Experiments" [3] supported by RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" [4].

3. PROJECT SCOPE

The scope of this report is a Level 1 PRA that models the risk of core damage by quantifying the CDF associated with removing heat from the process steam of an LWR. This result is then carried forward for use in adding a hydrogen production plant that uses high-temperature electrolysis. Within the PRA, the high-temperature electrolysis facility (HTEF) is treated as both a potential internal and external event hazard upon the LWR. The IE frequencies associated with the addition of the LWR's heat extraction system (HES) and the HTEF will be compared against the guidelines set in 10 CFR 50.59 and the CDFs and large early release frequencies (LERF) calculated from the PRA will be compared against the guidelines set in RG 1.174. Recommendations for the applicability of the results to this licensing path will be given.

The primary internal event concern for increased risk when heat removal is added to a standard LWR is the loss of steam inventory by a steam line break. Ultimately, the loss of steam results in the average temperature of the secondary system cooling down, thus causing a positive temperature coefficient that leads to reactivity insertion, which leads to a reactor power spike. The following increased temperature of the reactor core is what can lead to a reactor trip or core damage. Thus, large steam line break failures are considered the major risk added by the addition of the HES. Increases in the IE frequency of the large steam line break are quantified in this report. In addition to these events, the increase in transients caused by smaller steam line leaks, control system faults, etc., are also considered.

Hydrogen production poses a threat to the reactor core in large detonation accidents where the overpressure impulse (i.e., shock wave), fire, or shrapnel comes into contact with the reactor building or other critical structures on the site. While deflagration events have consequences local to the HTEF, those are not consequential outside of the facility.

The physical specifications of the proposed HES and HTEF are also detailed. These specifications are used to add on to the generic PRA models.

4. NPP WITH HES AND COLLOCATED HTEF SYSTEM DESCRIPTION

There are two designs considered for the HES. One is a two-phase to two-phase transfer design where the heat-transfer medium in the thermal power delivery (TPD) loop enters a vapor phase when heated to operating temperatures. The other design is a two-phase to one-phase transfer where the heat-transfer medium stays in the liquid phase. Steam-to-steam heat transfer will always use the two-phase to two-phase design. Heat-transfer fluids (HTF), many times incorrectly referred to as "heating oil," can be used in two-phase or single-phase operating states, depending on their physical characteristics and the desired operating temperature (Section 5.1.4.4). Note that there is no actual HES system at the time this research is done and therefore these are conceptual designs that are based on those used in the LWRS report "Incorporation of Thermal Hydraulic Models for Thermal Power Dispatch into a PWR Power Plant Simulator" [5].

4.1 Two-Phase to Two-Phase HES Design

A P&ID diagram of the proposed HES line for steam in the TPD loop is shown in Figure 4-1 as adapted from [5]. The nuclear plant's steam line (main steam header) taps steam from the main steam line downstream from the main steam isolation valves (MSIVs). The steam condition available for extraction at the main steam header is saturated steam with a total mass flow rate of 5.8×10^{6} kg/hr (1.3×10^{7} lb/hr) at 69.5 bar (1,008.5 psia). HES-1 as the main control valve for the HES line, and therefore has the largest effect on reactivity control. During steady-state operations, the steam in the HES line is condensed to avoid sending high-pressure steam to the condenser, which would decrease plant operating efficiency. The extraction heat exchangers required for heat transfer to the hydrogen production plant are located at the NPP site. The HES is also near the turbine system, but not necessarily within the turbine building, to reduce losses and minimize the amount of additional steam inventory that is cycled through the NPP. Two HES isolation valves are modeled in series (IV-1 and IV-2), mimicking the configuration of a typical MSIV arrangement. For the option in which superheated steam or a vapor-phase HTF is used in the TPD loop, the extraction heat exchangers comprise a two-stage system because there will be a phase change in both the hot and cold fluids.

The first heat exchanger HES-EHX-1 is a once-through steam generator (OSTG). The saturated steam is on the tube side of the heat exchanger, and the delivery steam is evaporated completely and superheated on the shell side. The reason for this design choice is the fact that the OTSG provides slightly superheated steam from a subcooled liquid inlet in a single heat exchanger. This combined with the vertical nature of the heat exchanger makes it reasonable for providing the desired heat transfer and fluid conditions. The TPD loop is superheated by about 45°F if steam is used as the heat-transfer medium (vapor-phase HTF superheated temperatures would vary) to assist thermal delivery to the hydrogen plant approximately a kilometer away with minimal condensation.

TPD-EHX-2 has a design like a feedwater heater. The wet steam from the NPP enters the heat exchanger on the shell side to be condensed and subcooled by the condensate from the TPD loop. The condensate in the TPD loop is preheated in the tube side of the heat exchanger before being fully evaporated and superheated in HES-EHX-1. The subcooled liquid is designed to exit HES-EHX-2 at 193.3°C (380°F) at a high pressure of 68.3 bar (980 psi). This liquid is throttled to condenser pressures through an orifice. There is a check valve prior to the orifice which requires a high differential pressure to open. This helps to ensure that the HES line remains pressurized in the event of a system malfunction to protect the chemistry of the nuclear steam in the case of a substantial tube leak in either of the extraction heat exchangers.

As the steam in the hydrogen production plant is pumped through the tubes of HES-EHX-2, it is preheated to saturation, then boils and superheats as it passes through the shell side of HES-EHX-1. The maximum flow rate of steam exiting the extraction heat exchangers and moving toward the hydrogen plant is 2.715×10^5 kg/hr (5.986×10^5 lb/hr) and the temperature is 252°C (485°F). This steam travels approximately 1 km to the hydrogen plant via a pipe equipped with steam traps to ensure dry steam is sent to the hydrogen plant's steam generator. The condensate is then pumped back to the HES heat exchangers, where it is boiled into steam again. Several valves in Figure 4-1 are highlighted in blue. This highlight indicates they are design options. A sensitivity analysis is conducted in Section 6.4 to analyze the safety benefits of these options, and to select the optimal option in terms of safety and costs.

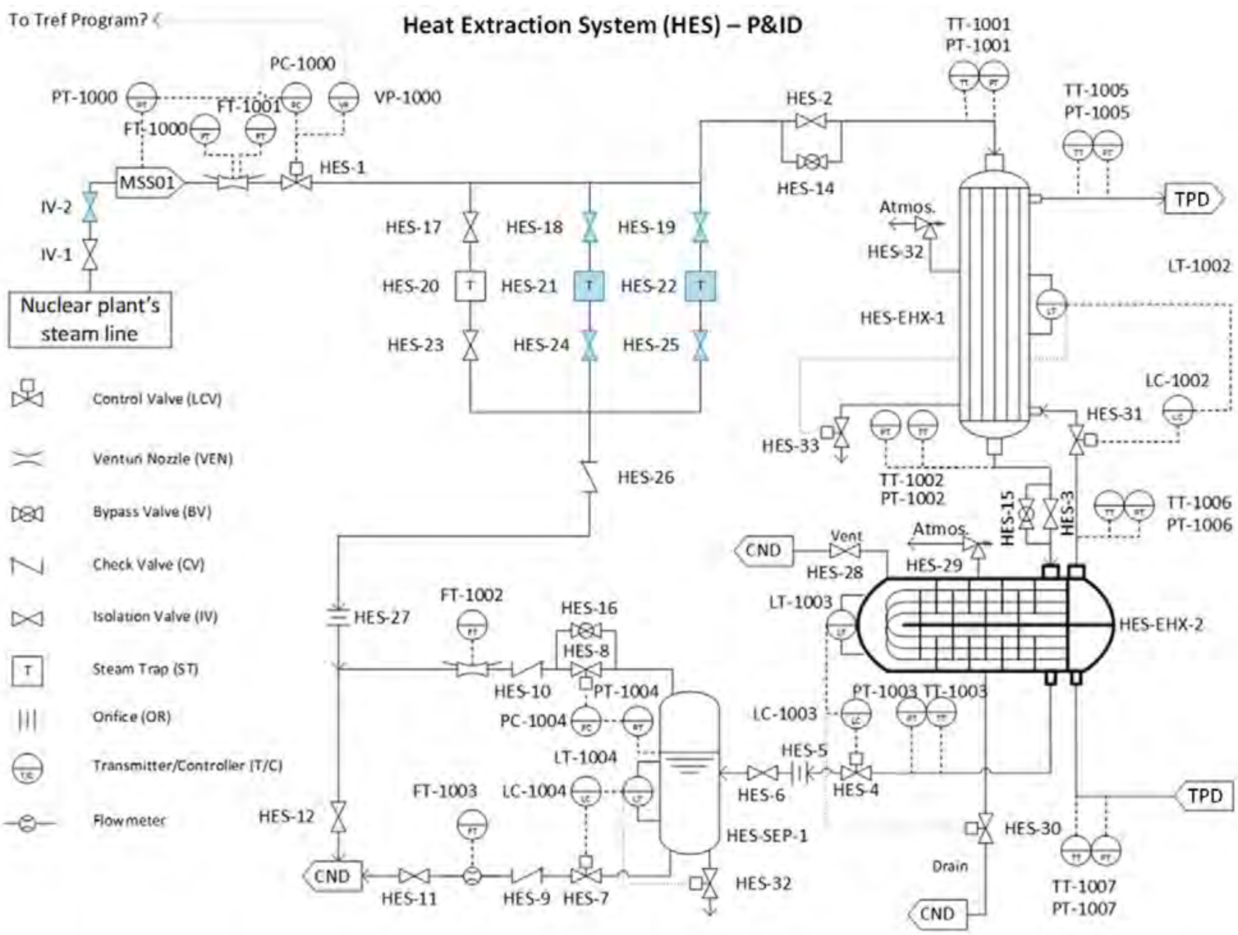

Figure 4-1. Piping and instrumentation diagram of two-phase to two-phase HES.

4.2 Two-Phase to One-Phase HES Design

The P&ID for the HES for constant liquid phase in the TPD loop is shown Figure 4-2 [5]. The design shown is the same as described in Section 4.1 with the following exceptions:

Steam traps are not used as a bypass configuration. Instead, HES-7 in the main extraction line downstream from HES-1 removes condensate that forms while saturated steam travels to the extraction heat exchangers. HES-EHX-1 condenses the steam in the HES steam line and is equipped with a hotwell (HES-HW-1). HES-HW-1 is a reservoir equipped with valves to control the condensate level in HES-EHX-1. At a specified condensate level, a valve opens to allow condensate to flow to the HES-EHX-2. This design ensures that only liquid water can flow to HES-EHX-2 when using fluid-to-fluid heat transfer. HES-EHX-1 has a vent to the condenser for use while the water level is building to the desired level. HES-HW-1 also has a drain to the condenser to allow for extra draining, if necessary. The steam is in the shell side of HES-EHX-1. HES-EHX-2 is a normal shell-and-tube heat exchanger with the water in the tubes and the HTF in the shell. This heat exchanger serves to sub-cool the water to allow for maximum heat dispatch. After the condensate exits HES-EHX-2, it flows to the condenser.

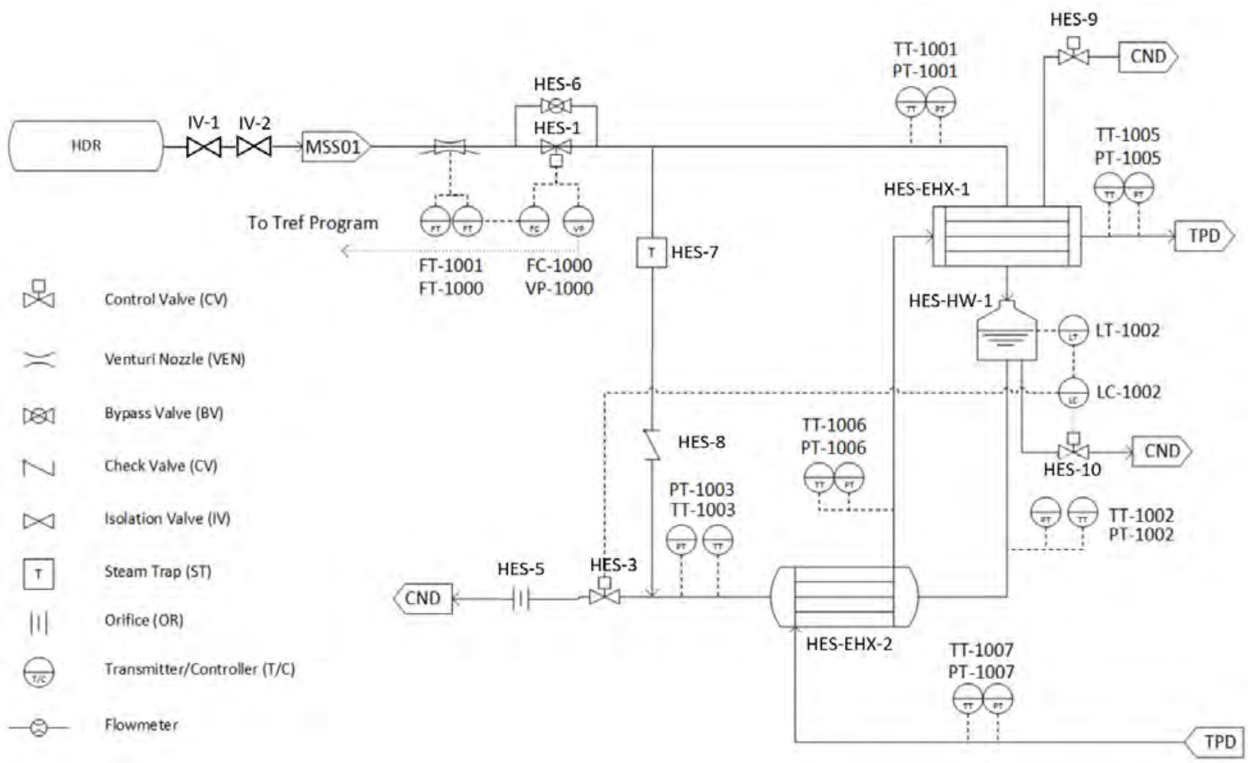

Figure 4-2. Piping and instrumentation diagram of two-phase to one-phase HES.

5. HAZARD ANALYSIS

The hazards considered potentially affect the frequency of internal and external events of the NPP. To define internal events in an NPP connected through a thermal loop to an HTEF, the jurisdictional boundary must be defined where the NRC's regulation of the nuclear facility ends. A report issued to address colocation of facilities at advanced nuclear reactor sites, INL/EXT-20-57762, "Establishing Jurisdictional Boundaries at Collocated Advanced-Reactor Facilities" [6], summarizes the following points applicable to jurisdiction:

- NRC would retain full oversight authority over SSCs needing protection under physical-security regulations. These security elements would be part of the nuclear facility.
- All SSCs that perform nuclear-safety-related or risk-significant functions would be included within the nuclear facility boundary and under NRC jurisdiction.
- Energy-conversion system(s) located within the nuclear protected-area boundary, are integral to the nuclear facility, and/or are operated by the nuclear facility control room, should be considered

part of the nuclear facility. Energy-conversion system(s) located outside the protected-area boundary and separated from the nuclear facility by a transfer system with appropriate interface criteria could be excluded from nuclear facility scope. Interface criteria must ensure the nuclear facility is not dependent upon or adversely affected by industrial facility events.

- Nuclear safety analysis would be required of all nuclear and industrial systems with respect to potential missiles, security issues, flooding issues, or any other impacts that may influence SSCs that perform a nuclear safety function.

- The regulatory boundary between the nuclear and industrial facilities can be defined by describing the boundary in the nuclear facility system design, transfer-system(s) design, and interface descriptions with appropriate interface requirements, and pertinent downstream conceptual-design information. Interface requirements must address industrial facility systems transients and failures. Requirements must ensure that no portion of the industrial energy-transfer system performs or adversely affects a nuclear safety function. Appropriate monitoring and detection systems are to be employed. Radioactive material releases from energy-transfer system(s) must meet applicable limits.

    - Interface requirements would demonstrate a robust ability to maintain safe nuclear operation. Site-related requirements and assumptions associated with the standard design would be shown as met along with all criteria-pertinent standard design safety. These requirements are also focused on preserving SSC nuclear safety functions.

These principles hold true for existing LWR facilities as well. A generalized NRC regulatory jurisdiction boundary is summarized in [6].

Most events that can interfere with the operation and safety of the NPP affected by the location of the HTEF outside of the regulatory jurisdiction (shown in Figure 5-1) are treated as external events. The exception is the reactivity feedback that would occur if there were a sudden large leak in the TPD that services the HTEF. External events are added to the NPP site by the potential for industrial interrupts and accidents at the HTEF. Other external events specific to the site are assumed to already be covered adequately by the existing NPP Level 1 PRA.

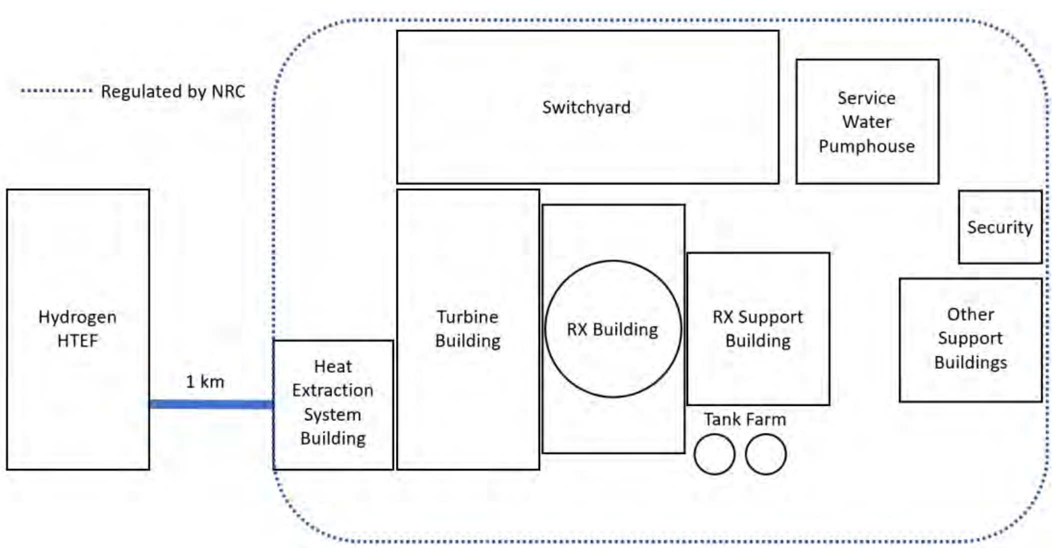

Figure 5-1. NRC jurisdictional boundary for LWR servicing an HTEF.

Hazard analyses were performed for both the NPP and the HTEF. The NPP hazard analysis included the envelope beyond that postulated by [6] by considering the heating loop provided by the NPP to the HTEF and the temperature drop negative reactivity feedback that would occur if the loop were to experience a sudden break in the piping. The HTEF hazard analysis started on the secondary side of the heat exchanger after the delivery of thermal energy to the HTEF.

5.1 Nuclear Power Plant with HES Hazard Analysis

The hazards associated with the addition of the HES to the existing NPP were considered through interviews with subject matter experts (SMEs) and available design drawings and options of the proposed HES.

5.1.1 Design Options and Assumptions

The HES design options and assumptions considered for the representative NPP, HES, and HTEF are listed in Table 5-1. HES design options reference the P&ID. Other assumptions are made based on physical properties and a generic geographic region.

Hydrogen detonation overpressure is a fraction-of-a-second impulse. Multiple detonations provide follow-on impulses. While it is reasonable to assume that a first impulse may weaken a structure and a following impulse might damage it, the fragility curves we use in this report are evaluated at the point of zero fragility to the impulse-equivalent psi. For multiple high-pressure jet detonations, it is possible that the first detonation would break another line, providing the opportunity for another high-pressure jet detonation of the same overpressure. An accumulated hydrogen cloud detonation would not cause another hydrogen cloud detonation because the facility is assumed to not have hydrogen storage.

Table 5-1. HES design options and assumptions.

Component/Parameter	**Identification (Figure 4-1)**	**Options**	**Assumptions**
Isolation Valve	IV-1, IV-2	One or two valves in series	Isolation valves will follow design of NPP MSIVs
Bypass Valve Trains	HES-17 through HES-25	One, two, or three trains	None
Heating Medium	TPD loop out and in	Steam or Heating Fluid	Steam is the standard
HES placement	Not Applicable (NA)	House the HES in the turbine building or in a dedicated building	HES is placed in a dedicated building (FMEA recommended).
Hydrogen Storage and Transfer Facility	NA		HTEF will pipe the production hydrogen to a storage and transfer facility 5 km distant from the NPP critical structures.
Electrical Power Linkage from NPP to HTEF	NA	Direct linkage, load following or connection to the grid then to the HTEF	The NPP is connected to the grid to buffer upsets from HTEF.

Component/Parameter	Identification (Figure 4-1)	Options	Assumptions
HTEF Ventilation	NA	Is there an HTEF industrial building ceiling ventilation of the hydrogen leak	A dedicated industrial building ceiling ventilation is not considered in base PRA case.
LOOP Frequency	NA		LOOP frequency is the same for the generic BWR and PWR model, assuming the same geographical region.
Multiple Detonations at HTEF	NA		Bounding accident is assumed for the first detonation overpressure. Ensuing detonations will not exceed bounding accident. Structures will not be weakened in the first detonation overpressure.
Temperature of the thermal delivery loop	NA		≤600°F

5.1.2 Nuclear Power Plant Safety-Critical Structures

The reactor building is the primary critical structure at an NPP. It is also the most well-protected from any external forces such as blast impulse shock waves. Nuclear-grade concrete walls encase the containment and provide significant protection to the reactor internal structures in addition to providing significant protection from accidental release of ionizing radiation. Critical structures external to the reactor building are typically designed to withstand postulated local wind and seismic loads. These include refueling water storage tanks (RWST) and condensate storage tanks (CST).

5.1.2.1 Reactor Containment Structure Fragility to Overpressure Events

Reactor building concrete walls were characterized in EGG-SSRE-9747, "Improved Estimates of Separation Distances to Prevent Unacceptable Damage to Nuclear Power Plant Structures from Hydrogen Detonation for Gaseous Hydrogen Storage" [7]. The lowest static pressure capacity of nuclear concrete identified is 1.5 psi. This conservative estimate was used for the blast analyses performed in the separation study INL/EXT-05-00137, "Separation Requirements for a Hydrogen Production Plant and High-Temperature Nuclear Reactor" [8] and [2] and is adopted as the static pressure capability of nuclear concrete walls in this study as well.

5.1.2.2 Safety Critical External Structures Fragility to Overpressure Events

Critical structures outside of the reactor building have been identified when assessing high winds fragility for PRA. For most BWRs, these include at least one CST. Many times, there is an auxiliary (sometimes called emergency) feedwater tank, service water pump house(s) and intakes, and the electrical switchyard. For PWRs, there is typically a refueling water storage tank (RWST), an auxiliary or emergency feedwater tank, and/or a CST, service water pump house(s) and their associated intakes, and a switchyard. Many wind-pressure and wind-missile fragility studies have been performed for NPPs. The individual plant examination of external events (IPEEE) studies in the 1990s produced a wealth of information on wind fragilities. The Duane Arnold IPEEE [9] was selected to act as a baseline for these fragilities. An updated high-wind fragility analysis performed by Applied Research Associates (ARA) [10] determined the mean fragilities components commonly found in the switchyard. These wind pressure fragilities of 6-second gusts were transformed into blast overpressure impulse fragilities in SAND2020-7946, "Final Report on Hydrogen Plant Hazards and Risk Analysis Supporting Hydrogen Plant Siting near Nuclear Power Plants" [11].

External water tanks are located close to the reactor building for use in providing condensate storage and coolant for routine and emergency operations. In some cases, there are concrete walls placed around the external tanks for protection, but some NPPs choose not to include external protection other than the tank's own construction. These tanks are built to extreme standards. According to [9] and other IPEEEs, they are equivalent in structural integrity against wind pressure to a Category I Structure. This means that the tanks are nearly as durable as the reactor building itself and nearly as durable as reactor containment when it comes to handling pressure. The CST and other storage tanks are assumed to be Category II structures when considering susceptibility to wind missiles. The probability of failure per instance of overpressure for storage tanks and Category I Structures are listed in Table 5-2. An overpressure event is a fraction-of-a-second impulse, so correlation between wind speed pressure fragility to overpressure requires proper scaling.

Service water intakes are solid structures and their failure modes typically involve the buildup of debris on the screens instead of physical damage; however, the pump house is not typically built to withstand tornadic or hurricane winds. In some NPP PRAs, a loss of service water is itself an initiator that challenges the NPP to shut down safely. The probability of failure per instance of wind speed for a typical pump house is listed in Table 5-2.

Loss of switchyard components means a loss-of-offsite-power (LOOP) event which challenges the NPP to shut down safely. Switchyard components are fragile to wind pressure, and therefore also fragile to an overpressure event. The resulting overpressure fragilities for the switchyard are shown in Table 5-2.

Table 5-2. Blast overpressure fragilities of switchyard components.

SSC	Effective Pressure (psi)	Equivalent Windspeed (mph)	Total Fragility (Wind and Missiles)
All Category I Structures	0.59	182	0
	0.97	234	4.00E-04
	1.49	290	4.60E-03
	2.16	349	4.00E-02
Storage Tanks (CST, RWST, etc…)	0.59	182	2.10E-03
	0.97	234	2.80E-03
	1.49	290	1.60E-02
	2.16	349	5.40E-02

SSC	Effective Pressure (psi)	Equivalent Windspeed (mph)	Total Fragility (Wind and Missiles)
Circulating Water/Service Water Pump Area in Pump House	0.10	75	8.00E-04
	0.20	105	5.80E-02
	0.28	125	1.50E-01
	0.59	182	5.20E-01
	0.97	234	9.40E-01
	1.49	290	1.0
	2.16	349	1.0
Switchyard, General	0.32	135	3.78E-01
	0.48	165	9.74E-01
	0.71	200	1.0
Transmission Tower	0.10*	75*	0.0*
	0.16*	95*	0.0*
	0.20*	105*	0.8*
	0.32	135	9.18E-01
	0.48	165	1.0
	0.71	200	1.0
Standby Auxiliary Transformer	0.32	135	1.99E-01
	0.48	165	2.68E-01
	0.71	200	3.11E-01
Note: * Updated and lower wind speed and pressure values taken from "Fragility Analysis and Estimation of Collapse Status for Transmission Tower Subjected to Wind and Rain Loads" [12].			

5.1.2.3 Non-Safety Critical External Structures

In addition to critical structures, some other structures that affect operations, but not typically the ability to safely shut down the reactor, are located in the plant yard as well: circulating water and standby service water pump houses, demineralized water storage tank(s), cooling towers, well water pump houses, liquid nitrogen tank, and hydrogen and nitrogen gas cylinders, which present stored energy in the form of chilled and pressurized gas.

Further, the day-to-day operations of the NPP would be affected by damage to the turbine building, administrative building, and maintenance support buildings located throughout the site.

5.1.2.4 Example Site Plans with External Structures for PWR and BWR

Several NPPs were reviewed for external safety-critical and non-safety-critical structures. Calvert Cliffs NPP was chosen as a representative PWR site and Columbia NPP was chosen as a representative BWR site.

Calvert Cliffs NPP was chosen because it is a good example of a shoreline NPP where the placement of an industrial complex is limited to 180 degrees around the NPP due to the water source. It also has many natural obstructions due to the woods in the area. The overhead view of Calvert Cliffs NPP (Figure

5-2) shows the possible location at 1-km distance denoted by the red circle where a co-located industrial plant may be placed. Once a choice of siting is made, the origination and direction of an overpressure event can be determined along with attenuating obstructions. The analysis performed for this report did not consider attenuating obstructions to remain a generic model, but this feature is pointed out as something to consider for an actual site if conservatism is not desired or warranted. Figure 5-3 shows an aerial view of Calvert Cliffs NPP with the critical structures labeled. This gives a good perception of the sizes of the tanks and the geography of the surrounding area. Figure 5-4 shows the Calvert Cliffs site plan with the critical structures labeled. Other structures of interest are the water storage tanks alongside the CSTs and the liquid nitrogen storage in the northeast corner of the tank farm where the CSTs are located.

Figure 5-2. Calvert Cliffs NPP 1 km from reactor building overhead view, © listed in image.

Figure 5-3. Calvert Cliffs NPP critical structures labeled on aerial view, image from the NRC.

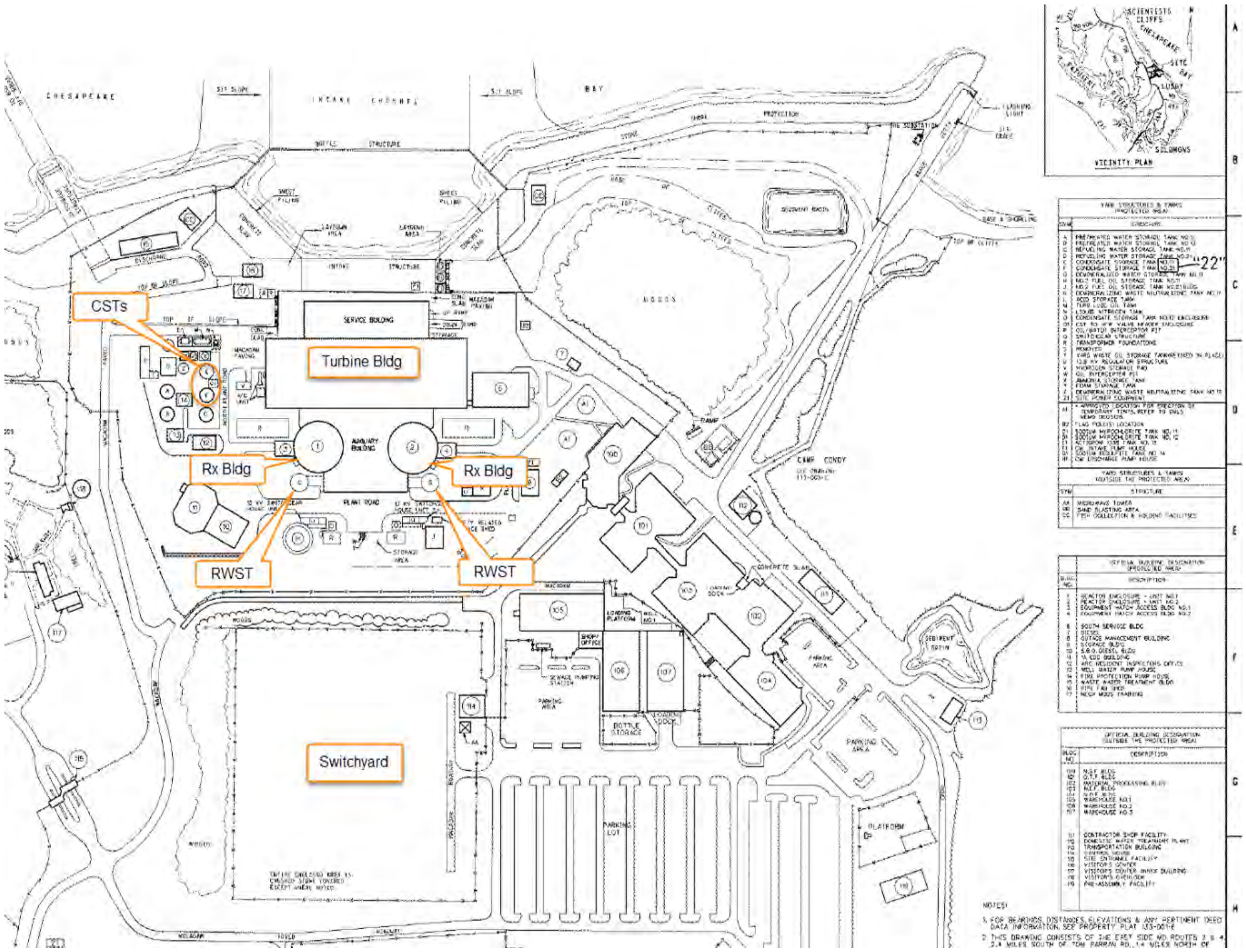

Figure 5-4. Calvert Cliffs NPP Site layout from IPE.

Columbia Generating Station NPP was chosen as an example BWR for several reasons. Even though the Columbia River is in close proximity, the Columbia NPP is a good example of an inland NPP site using man-made ponds. The site has no natural obstructions within the 1-km area specified for a co-located industrial site. There are two abandoned NPP projects immediately to the east which could potentially be an industrial site location. The overhead view of Columbia NPP (Figure 5-5) shows the possible orientation within 1 km where a co-located industrial plant may be placed. Once a choice of siting is made, the origination and direction of an overpressure event can be determined along with attenuating obstructions. As stated previously, attenuation of an overpressure event was not considered in the analysis, but attenuation should be considered for an actual site if conservatism is not desired or warranted. Figure 5-6 shows an aerial view with the critical structures labeled. Figure 5-7 shows the Columbia NPP site plan with the structures labeled. The CSTs, the transformer yard, and the switchyard are critical structures. Other structures of interest are the standby service water pumphouses.

Figure 5-5. Columbia NPP 1-km boundary from reactor building overhead view, © listed in image.

Figure 5-6. Columbia NPP site overhead view © Google Maps, Maxar Technologies, USDA.

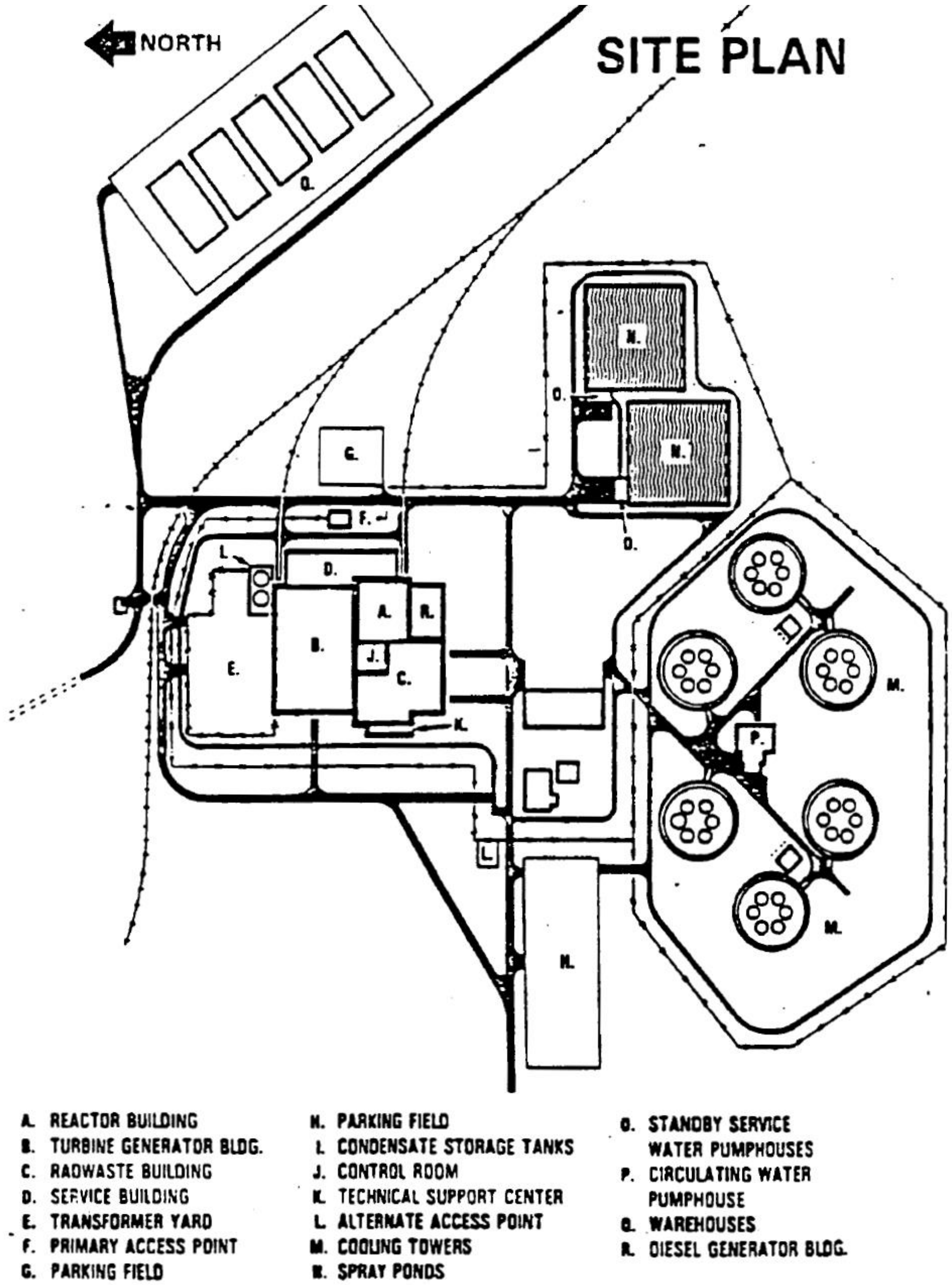

Figure 5-7. Columbia NPP site plan from IPE.

5.1.3 NPP Hazard Analysis

A group of SMEs were gathered for a Failure Modes and Effects Analysis (FMEA). The team included SMEs with experience in PRA and reliability engineering, PWR operations, BWR operations, detailed design knowledge of the hydrogen HTEF proposed for this study, chemical, and controls experts. Information gathered from the SNL report [11] was used to determine the external events which could possibly affect the NPP. These included external overpressure events, heating medium (steam or HTF) leakage at the HTEF, and electrical power load loss from the HTEF.

An outline of the topics considered for the FMEA included:

- External overpressure event effects on NPP
- Thermal and electrical load effects on NPP
    - Thermal and Electrical load power profiles supplied by the NPP to the HTEF
- Hot standby mode
- Steam vs. HTF
- A list of the HTFs under consideration
- Placement of the HES
- Unique risks of BWR
- Unique risks of PWR
- Production hydrogen routing options and effects on risk.

Possible external overpressure events effects on the NPP were summarized to include the damage to the containment, damage to external coolant storage tanks, LOOP, damage to above water spray mechanisms in spray ponds, debris in spray pond or cooling tower pond, and service water pump house damage. The results of the SNL report on Maximum Credible Accident (MCA) at 1-km distance were known prior to the performance of this FMEA. The team was therefore able to quantify a risk priority number (RPN) for each of the components considered based on the overpressure created from the MCA.

Possible thermal and electrical load effects on the NPP were summarized as a load-drop feeding back negative reactivity into the NPP, possibly causing a reactor trip.

Hot standby mode discussion was centered around the thermal and electrical load effects.

Differences were considered between steam and HTF as the heat transfer fluid in the secondary HES loop providing thermal energy to the HTEF. Steam is identified as the preferred heat-transfer medium from anecdotal evidence and a discussion with Electrical Power Research Institute (EPRI) BWR and PWR experts in January 2020. This preference is far and away due to familiarity of working with steam. There are benefits to using HTF in comparison to steam. The HTF maintains heat for a longer period of time, it can operate in a steady state or from a liquid to a vapor, therefore there is much less chance of cavitation of pumps, if used. Finally, the heat exchanger for a steam system would be larger and therefore more expensive than the heat exchangers for HTF.

The HES was considered for placement within the turbine building or in a building separate from the turbine building. The benefit of placement in the turbine building (if room in the existing NPP is available) is lower cost. The benefit of placement within its own structure is increased safety, as the FMEA results (Appendix C) identify.

Unique risks were considered for BWR and PWRs for each of the hazards identified.

Hydrogen production and storage was discussed as a potential hazard. The current model consists of piping the hydrogen to a transfer facility at least 5 km away from the NPP. This facility would consist of truck transfer and other pipeline transfer, including the possibility of mixing with natural gas.

5.1.4 List of NPP Hazards Identified

The NPP FMEA results are listed in Appendix C, Table C-1 and Table C-2. The RPN for each identified hazard was calculated and ranked. RPNs for this exercise are used as risk information. There is no RPN cutoff at which the hazard will not be modeled in the PRA. All risks identified are evaluated in the sections that follow. Those not screened by engineering evaluation are mapped into the respective event trees and the IE frequency for these event trees are re-quantified for the respective BWR and PWR models based on the increased frequency of occurrence caused by the addition of the HES and the location of the HTEF at 1-km distance.

The hazards either affected or added to the PRA by the addition of the HES and the HTEF are listed in Table 5-3. Also listed in the table is the event tree that the hazard would map to and the status (include or screen from the PRA) from the FMEA panel. There are five potential hazards considered in adding the HES and locating the HTEF at 1-km distance: hydrogen detonation at the HTEF causing an overpressure event at the NPP site, an unisolable steam pipe leak in the HES outside of the NPP MSIVs, a heat exchanger leak in the HES, ignition of the heating medium, and the prompt loss of thermal load to the HES.

Table 5-3. FMEA potential failures from hazards and PRA event tree assignment.

Hazards	Potential NPP Process Functions Affected	Potential PRA Event Tree Assignment	FMEA Hazard Status
H2 detonation at HTEF (high-pressure jet detonation, cloud accumulation detonation)	Loss of Offsite Power	Switchyard Centered LOOP (LOOPSW)	Included
	Loss of Service Water (Spray Pond damage or debris, Cooling Tower Pond debris, Service Water Pump House, Forced Air Cooling)	Loss of Service Water System (LOSWS) (BWR) No generic PWR tree affected	Included
	Critical Structure Damage (Reactor Containment, CST, or other coolant supply tanks)	HTEF-H2-DETONATION[1]	Included
HES steam pipe rupture outside of NPP MSIVs	Missile damage in turbine building (if HES located in turbine building)	Main (Large) Steam Line Break in HES (MSLB-HES), TRANSIENT (MSLB-HES bounding)	Included (screened if HES is not in the turbine building)
	Main (large) steam line rupture, unisolable steam leak	MSLB-HES	Included
HES heat exchanger leak	Large Leak/Rupture: Main steam line unisolable steam leak	MSLB-HES	Included
	Small Leak: Contamination of the HTEF heating loop (steam or HTF)	Not a design basis event. Economic risk. BWR is a higher risk to contaminate the HTEF heating loop.	Screened for Level-1 PRA. There is an economic and environmental concern.

[1] Potential new event tree if evaluated overpressure damages critical structures.

Hazards	Potential NPP Process Functions Affected	Potential PRA Event Tree Assignment	FMEA Hazard Status
Ignition of heating medium	Steam, non-flammable HTFs: flammable	None	Screened for steam Not considered for HTFs
Prompt steam diversion loss, feedback	5% thermal diversion	None. NPP can handle 30% prompt load loss. Screened out.	Screened
HES steam rupture in the turbine building	Turbine building SSC damage, possible safety bus damage, depending on plant configuration.	TRANSIENT, emergency power capability	Screened out by recommendation to not place HES in turbine building

5.1.4.1 Hydrogen Detonation at the HTEF

The hydrogen detonation at the HTEF is the focus of the study performed by SNL [11]. The leak frequency was determined by analyzing the P&IDs for the proposed pilot HTEF used for this project using industrial leak rate data for the individual components. The overall leak rate for leak sizes scaled from 1 = full line break is reproduced from [11] in Table 5-4. The overpressure felt at the NPP from a high-pressure jet leak detonation or a hydrogen cloud accumulation detonation were determined based on 15 leakage scenarios. No credit was given for attenuation of the shock wave made by buildings, wooded areas, or other topography. The bounding case presented in [11] used the largest leak size and therefore this frequency (5.19E-02 /y) was used in the PRA IE development. Calculations were made for the next largest leak size, denoted 0.1, and the most fragile components of the NPP were not affected by the overpressures created from either the high-pressure jet or hydrogen cloud detonation. According to "Methodology for Assessing the Safety of Hydrogen Systems: HyRAM 1.1 Technical Reference Manual" [13] the highest probability of detonation of a hydrogen leak, given an ignition source, is 0.35. This conservative value was used for the determination of detonation frequency, given a leak, in the PRA model.

Table 5-4. HTEF System Leak Frequency (/y) from [11].

Leak Size	HTEF System Frequency			
	Mean	5th	Median	95th
0.0001	2.28E+01	7.95E+00	1.70E+01	5.48E+01
0.001	4.19E+00	1.13E+00	3.32E+00	9.89E+00
0.01	1.37E+00	1.45E-01	7.47E-01	4.16E+00
0.1	1.33E-01	3.34E-02	1.01E-01	3.20E-01
1	5.19E-02	2.51E-03	2.18E-02	1.83E-01

High Pressure Jet Detonation: The high-pressure jet detonation frequency is not determinant on the human action to isolate the leak. The hydrogen is immediately available for detonation at the strength calculated. The maximum overpressure from a credible accident felt at 1 km distance from a high-pressure jet detonation is 0.056 psi [11]. The total fragility of switchyard components resulting from wind pressure and tornado-generated missiles is listed in Table 5-2 from [9] and [12]. This fragility data is used to determine the failure probability of these components when a hydrogen detonation event occurs. The fragility data points are shown in Figure 5-8. Fragility estimates between the known data points are interpolated linearly. The most fragile component in the switchyard is the transmission tower. The probability for damaging a transmission tower goes to zero at approximately 0.16 psi [12]. For reference, windows will break at an incident overpressure between 0.15 and 0.22 psi (Federal Emergency Management Agency, citing Kinney and Graham, "Explosive Shocks in Air" [14]). We use this data to screen out the high-pressure jet detonation as a safety concern in the PRA.

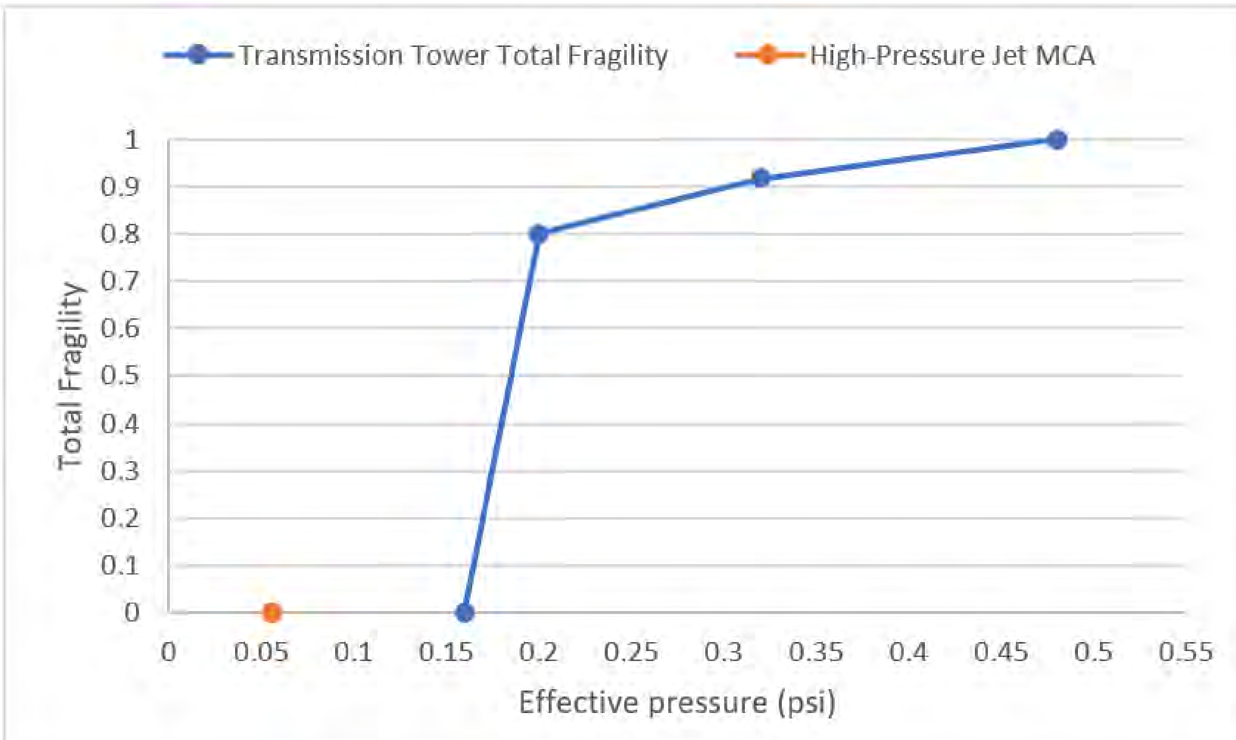

Figure 5-8. Switchyard components fragility as a function of wind pressure.

Hydrogen Cloud Detonation: The hydrogen cloud detonation frequency is determinant on the ability of to allow the hydrogen to accumulate within the building. This is determined by the failure of the building ventilation system to vent the leak to atmosphere and the failure of human action to isolate the leak within the specified time noted in [11]. For the MCA, this time is 120 minutes. The human action probability of failure was determined using the SPAR-H methodology within SAPHIRE to be conservatively 1.0E-2, given nominal time to perform the action and all other performance shaping factors listed as nominal. A less conservative approach, giving expansive time to perform the action was calculated as a probability of failure of 1.0E-04. The failure of all modes of an industrial building ventilation system was noted to be 2.4E-05/h in INEEL-EXT-99-001318, "Ventilation Systems Operating Review for Fusion Systems" [15]. The probability of detonation, given a leak is 0.35, as noted above. These probabilistic events, along with the yearly frequency of 5.19E-02/y for the full leak creating the MCA, were modeled in a fault tree to determine the frequency per year of the cloud detonation MCA event. This fault tree is the branch beginning with the AND logic gate IE_LOOPSC-HES-MCA in Figure 6-23. The resulting frequency is 4.15E-09/y. This is 7 orders of magnitude below the loss-of-offsite-power switchyard-centered (LOOPSC) IE frequency of 1.34E-02/y (basic event IE-LOOP-SC) for both the BWR and PWR models described below and five orders of magnitude below the IEFT-LOSW IE result of 1.80E-04/y in the BWR model for service water failure. We use the results of this IE fault tree to screen out the hydrogen cloud detonation as a safety concern in the PRA.

5.1.4.2 HES Inisolable Steam Pipe Rupture

A large steam line break is the most common hazard introduced by adding the HES to the NPP. The HES P&ID (Figure 4-2) shows there are two isolation valves for the HES, set in a series configuration. The success of these valves is the first line of defense of a steam line rupture within the HES after the NPP's main steam MSIVs. Rupture of the isolation valves are also a possibility that needed modeling. After the isolation valves, all of the other active components in the P&ID are evaluated in the fault tree of the HES (Section 6.1). The result of the fault tree was added to the IE for a large steam line break, as described in Section 6.2.1 for PWR and Section 6.3.1 for BWR.

5.1.4.3 HES Heat Exchanger Leak

Two types of heat exchanger leaks are considered for the PRA. One is a slow leak that is not a prompt safety concern to the operation of the NPP. The other is a heat exchanger rupture.

Slow Leak of an HES Heat Exchanger: The heat-transfer loop to the HTEF will always be operating at lower pressure than the NPP steam loop through the HES. This prevents the contamination of the NPP steam loop. Small leaks in the heat exchanger may contaminate the heat-transfer loop to the HTEF. This can cause a cleanup problem if there is enough activity transferred to the heat-transfer loop. For most NPPs this will not be a problem. PWR steam loops are less likely to have radioisotopes of any measure. BWR steam loops are more likely than PWR, but their steam loops are typically very clean as well. This a unique potential hazard to the LWR NPPs considering this modification. There are prevention, detection, and mitigation measures that obviously would need to be in place to monitor for and react to any small leaks. This hazard can cause economic issues for the cleanup, including shutdown of the reactor, and cause environmental concerns in the public. This study is concerned with reactor safety and did not consider the architecture of a representative system.

Rupture of an HES Heat Exchanger: There are two HES heat exchangers. Reference Figure 4-2. HES-EHX-1 heats the heating medium (steam or HTF) to its operating temperature. HES-EHX-2 pre-heats the returning heating medium and helps to chill NPP steam as after it exits HES-EHX-1. An HES heat exchanger rupture failure maps to the HES large steam line break event and is treated as an event within the IE fault tree for PWR (Section 6.2.1) and BWR (Section 6.3.1).

5.1.4.4 Ignition of Leaked Heat-Transfer Medium

The use of steam as the heat-transfer medium screens this hazard out from consideration. If HTF is used, it is dependent on the type of HTF. Four HTFs were provided by the designers of the proposed HES and are considered for this hazard: Therminol 66, Dowtherm A, Dowtherm G, and Therminol VP-1. As stated in Section 5.1.3, Dowtherm A and Therminol VP-1 operate in vapor states at their higher operating temperatures. Ignition of the HTF would result from a leak with an ignition source at a temperature above the flash point or over-heating the HTF to the auto-ignition temperature in the presence of oxygen. Leakage of the HTF was not determined for this study. Ignition probability was also not determined in this study; however, the flammability parameters and notes are listed in Table 5-5. The operating temperature of the HTEF thermal transfer loop is assumed to be ≤600°F.

A leak and fire within the HES building could damage the equipment and cause the NPP to isolate the HES. If the fire is severe enough, there is a possibility of damaging the ability to isolate the HES without closing the NPP's MSIVs.

Table 5-5. Heat-transfer fluid properties.

Heat-Transfer Fluid	Max Operating Temperature (°F)	Flash Point (°F)	Auto-Ignition (°F)
Dowtherm A	494 (liquid) 495–750 (vapor)	236	1110
Dowtherm G	675 (liquid)	280	810
Therminol 66	650 (liquid)	338	705
Therminol VP-1	256 (liquid) 257–750 (vapor)	230	1114

5.1.4.5 Prompt Steam Diversion Loss Causes Feedback

The addition of the HES to the NPP provides a new steam loop that must be evaluated for safety. The design considered for this study assumes that the amount of steam diversion is limited to 5% of the total steam production. This screens out one of the hazards which was postulated (Table 5-3, above), that the prompt load drop was felt by the NPP and pushed to the turbines, even with the successful closing of the HES isolation valves. The FMEA team determined that LWR NPPs can withstand up to 30% load drop without having to trip.

6. PROBABILISTIC RISK ASSESSMENT MODEL

Two generic PRAs were prepared for this report, one is a PWR and the other is a BWR. The difficulty in preparing a generic PRA for existing LWRs is that there are many differences in the existing LWR fleet and the geographical effects on LOOP events and external events. To remain generic, external events other than those created by the addition of a HTEF in close proximity to the NPP were not calculated. A Mark I containment BWR and a two-loop PWR were modeled. All mitigating fault trees were left intact except where affected by the addition of the HES or the effects on internal events of the HTEF. The external event of the HTEF detonation was considered for licensing under 10 CFR 50.59 as causing an increase in the LOOPSC IE frequency. The hydrogen detonation was also analyzed for inclusion in the PRA on its own as potentially damaging to critical structures not related to causing a LOOP, as noted in Section 5.1.4.1.

The sections that follow detail the HES model and the PRA modifications made to the generic models to assess the effects of the HES and HTEF on the NPP.

6.1 HES PRA Model

Since the HES is a separate, isolable steam system on the secondary side of the nuclear power plant, a steam line break in this system may not require a reactor trip if it can be isolated from the main steam line. For that reason, the steam leakage events and the failure to isolate HES system events are modeled altogether and the resulting probabilistic failure events are linked to the existing PRA model.

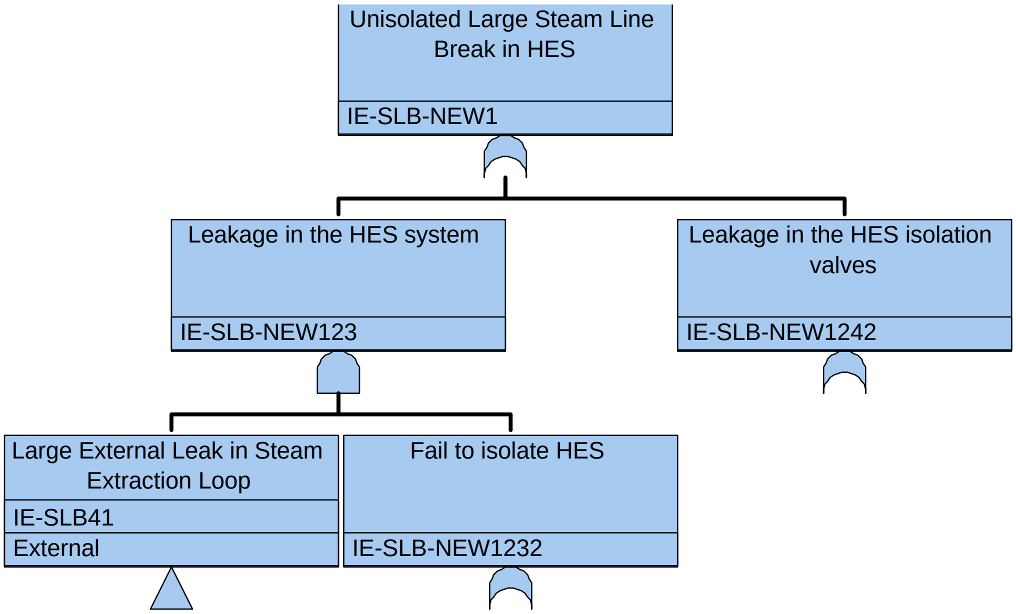

Figure 6-1. Unisolated large steam line break in HES Fault Tree (IE-SLB-NEW1).

The intermediate events that contribute to the unisolated large steam line break in HES system are shown in Figure 6-1. Possible leakage events were categorized based on the leakage locations, either in the isolation valves themselves or in the HES system. The latter consists of a large external leak coupled with the failure of isolation valves. The basic events in the former are shown in Figure 6-2. This tree consists of events in the different design options, whether the system uses an isolation valve or two in series. A House Flag event HES-ISOV-FLAG was paired in an AND logic gate with the basic events. The basic events when two valves are used in series include the event when IV-2 ruptures and IV-1 fails to stop the steam flow to the ruptured valve, or when IV-1 ruptures. Meanwhile with only one isolation valve, the possible rupture event exists for that valve only. The House Flag event is set as a complement in the one-valve subtree, such that only one configuration is active at a time, either the double- or the single-valve configuration.

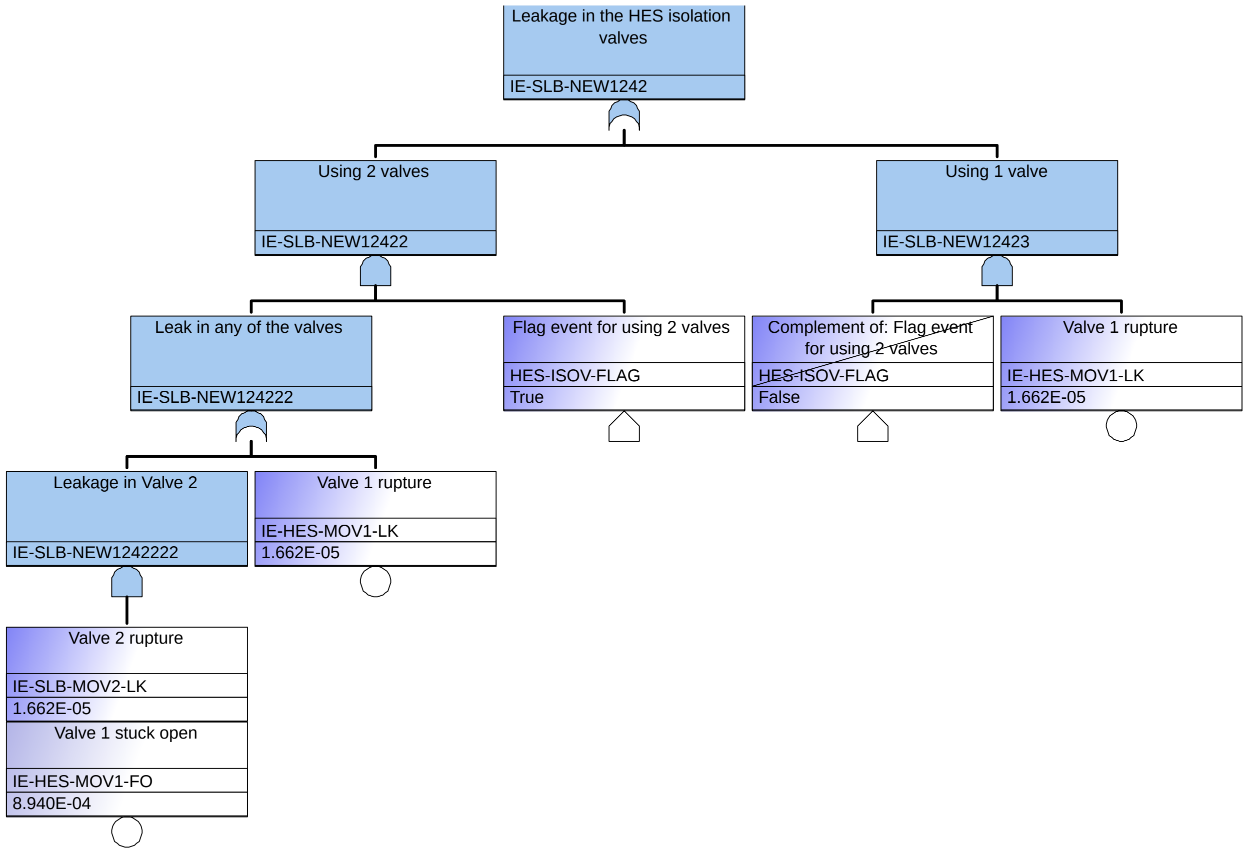

Figure 6-2. Leakage in HES isolation valves Fault Tree (IE-SLB-NEW1242).

The IE-SLB-NEW1232 intermediate event in Figure 6-1 is expanded in Figure 6-3. The tree structure is similar to Figure 6-2, which considers the two design options of using double- and single-isolation valves and uses a House Flag event to select the design option for analysis. The failure to close in the double valves design consists of independent failures from both valves and Common Cause Failures (CCFs). The CCF event was constructed using the Alpha-factor method with a Staggered testing scheme. In this tree, it was assumed that the Common Cause Group (CCG) consists of only the two isolation valves.

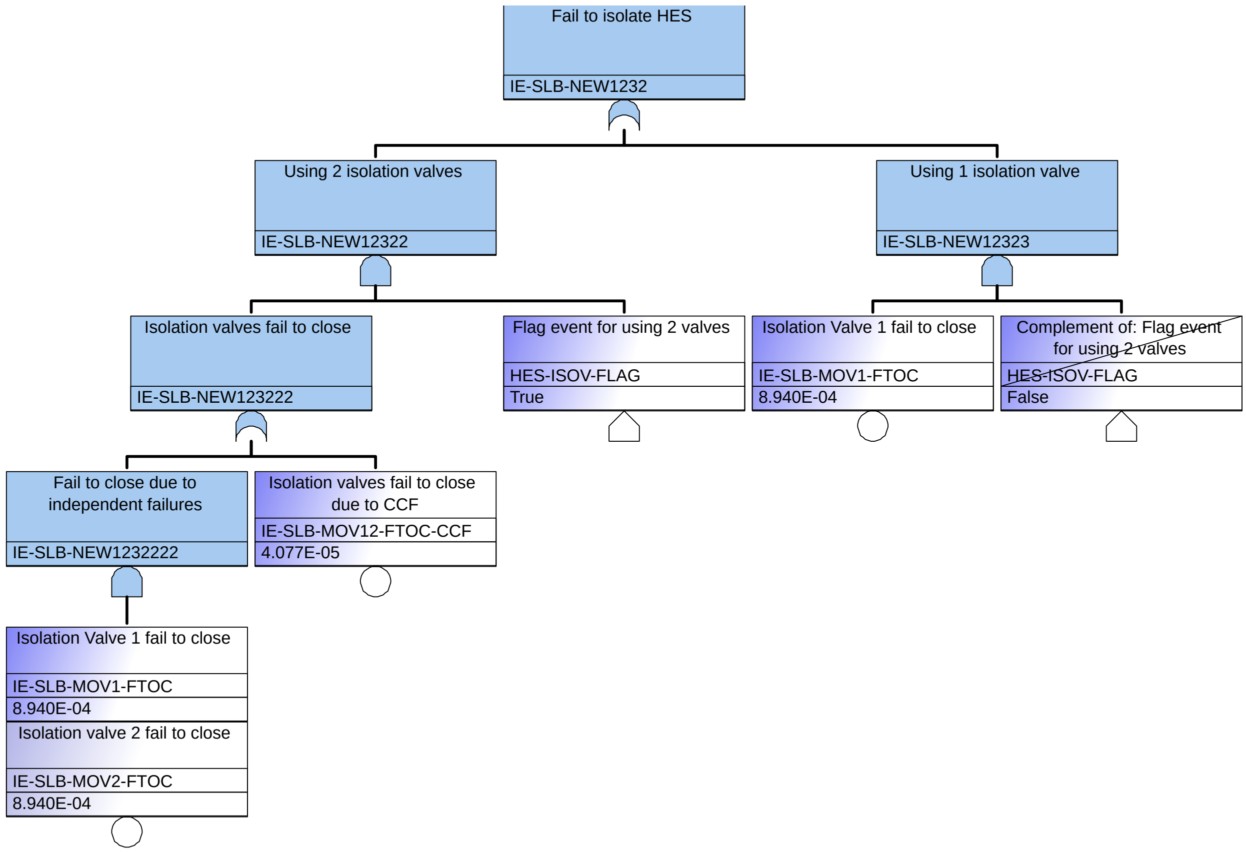

Figure 6-3. Failure to isolate HES fault tree (IE-SLB-NEW1232).

The IE-SLB41 intermediate event in Figure 6-1 is shown in Figure 6-4. In this tree, the HES-1 control valve was assumed to function as an isolation valve for the system. This valve is normally set to allow 5% of the main steam to be diverted to the heat extraction loop. It may close upon demand in case the isolation valve IV-1 and IV-2 fail to function. Therefore, the leak events in this tree may occur when HES-1 valve ruptures or when there is a leakage downstream of HES-1 and HES-1 fails to close.

The leakage in HES intermediate event (i.e., IE-SLB4132), is shown in Figure 6-5. The leakage in HES system downstream of the HES-1 control valve is categorized into sections based on the leak location. These sections include the three bypass trains, the components downstream of the bypass trains, the EHX-1 heat exchanger subsystem, the EHX-2 heat exchanger subsystem, and the SEP-1 tank subsystem. It was assumed there were no CCGs across these subsystems.

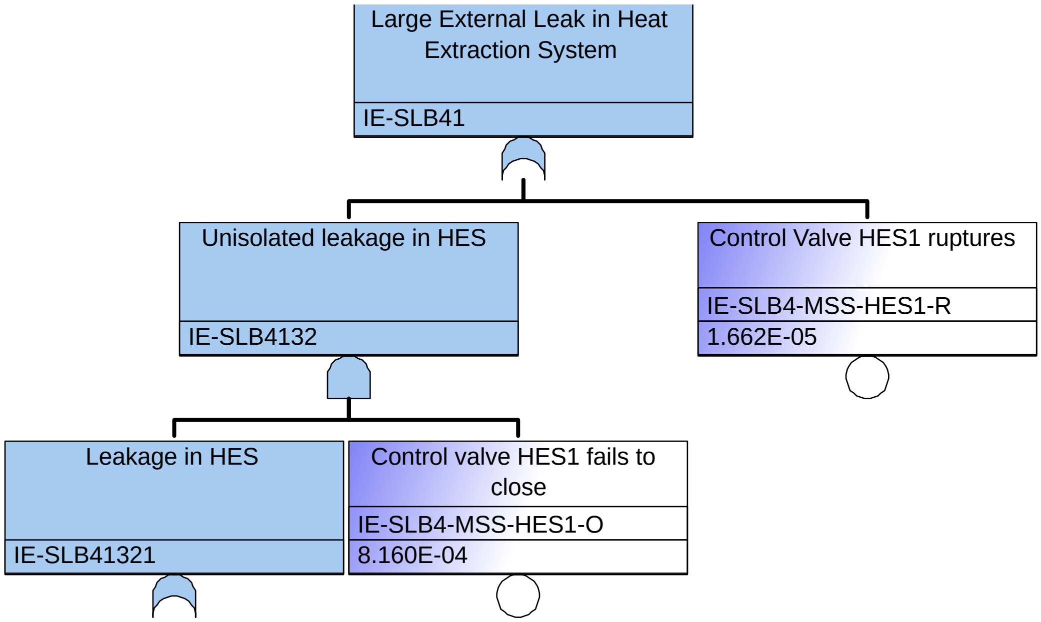

Figure 6-4. Large external leak in HES fault tree (IE-SLB41).

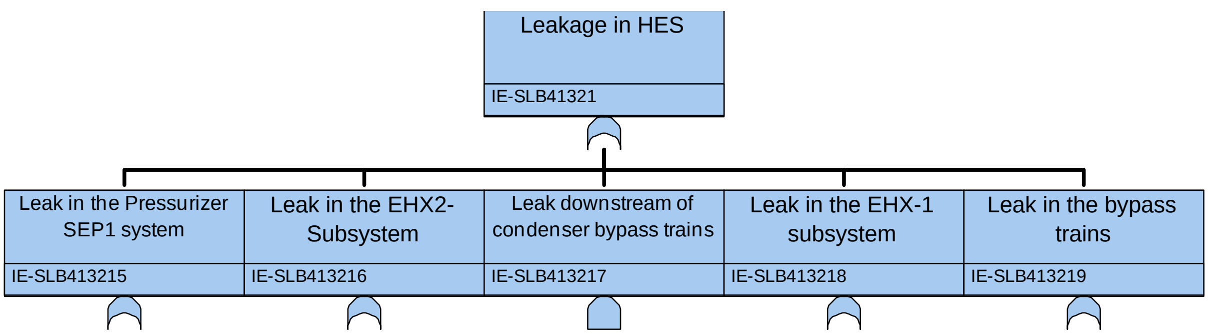

Figure 6-5. Leakage in HES fault tree (IE-SLB41321).

Steam leakage in the bypass trains may happen at any of the three trains, as shown in Figure 6-6. The subtree for the first train is shown in Figure 6-7. The leakage events in this train may occur when the upstream HES-17 valve ruptures, or when the downstream components rupture and the upstream valves fail to close. This logic applies likewise to the other trains as shown in Figure 6-8 and Figure 6-9. However, in these two trees, the logic structure was coupled to the House Flag event for that train in an AND gate. This modeling approach allows the analyst to evaluate risks by using single-to-triple redundant bypass trains in the HES system.

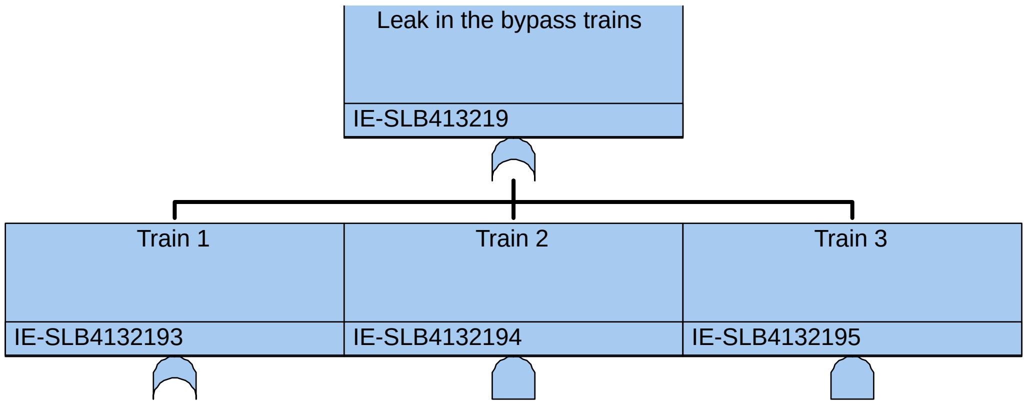

Figure 6-6. Leakage in the bypass trains fault tree (IE-SLB413219).

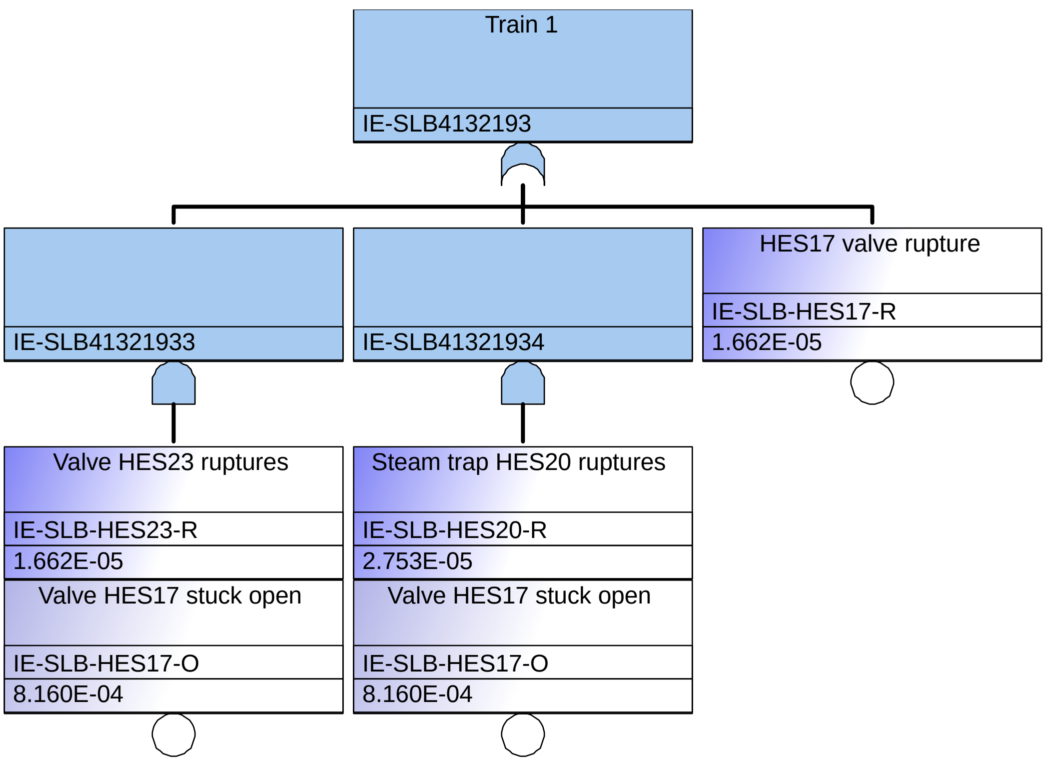

Figure 6-7. Leakage in the bypass train number 1 fault tree (IE-SLB4132193).

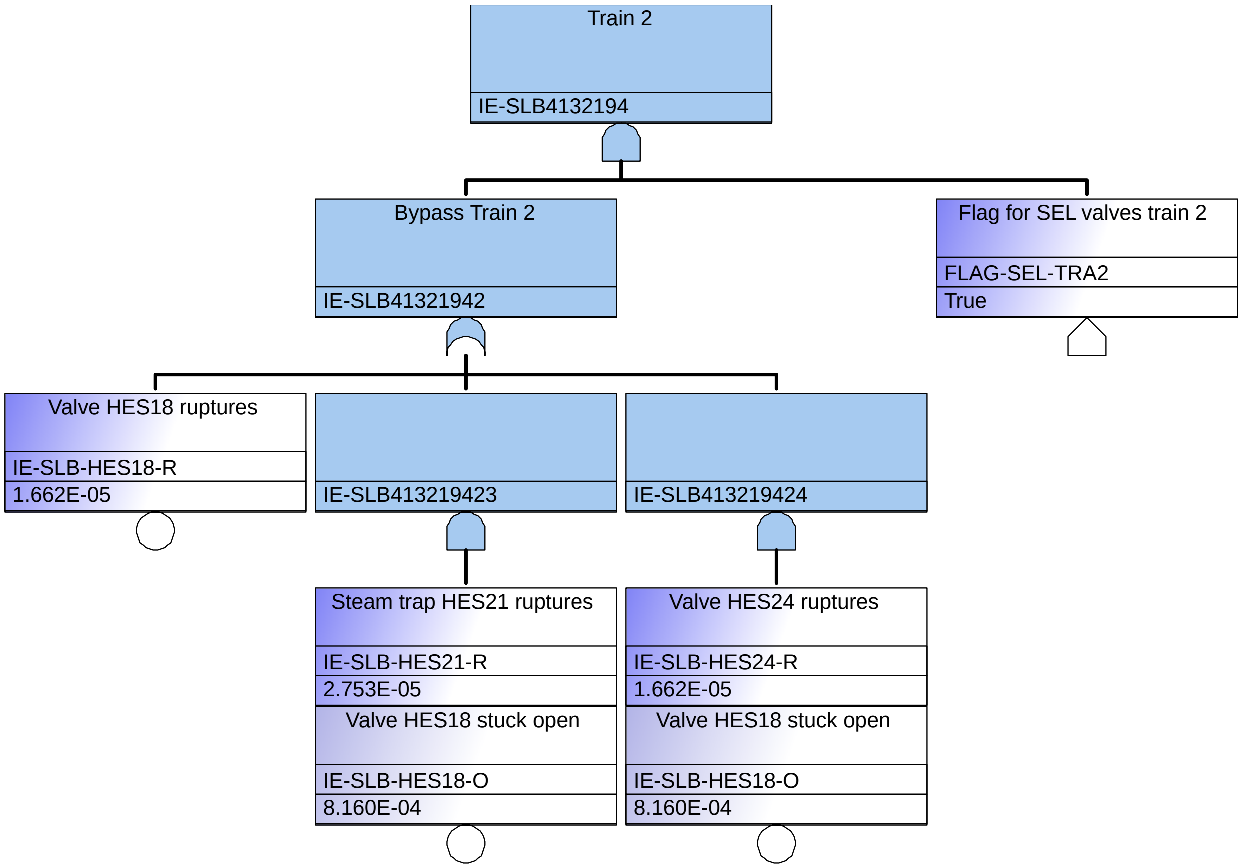

Figure 6-8. Leakage in the bypass train number 2 fault tree (IE-SLB4132194).

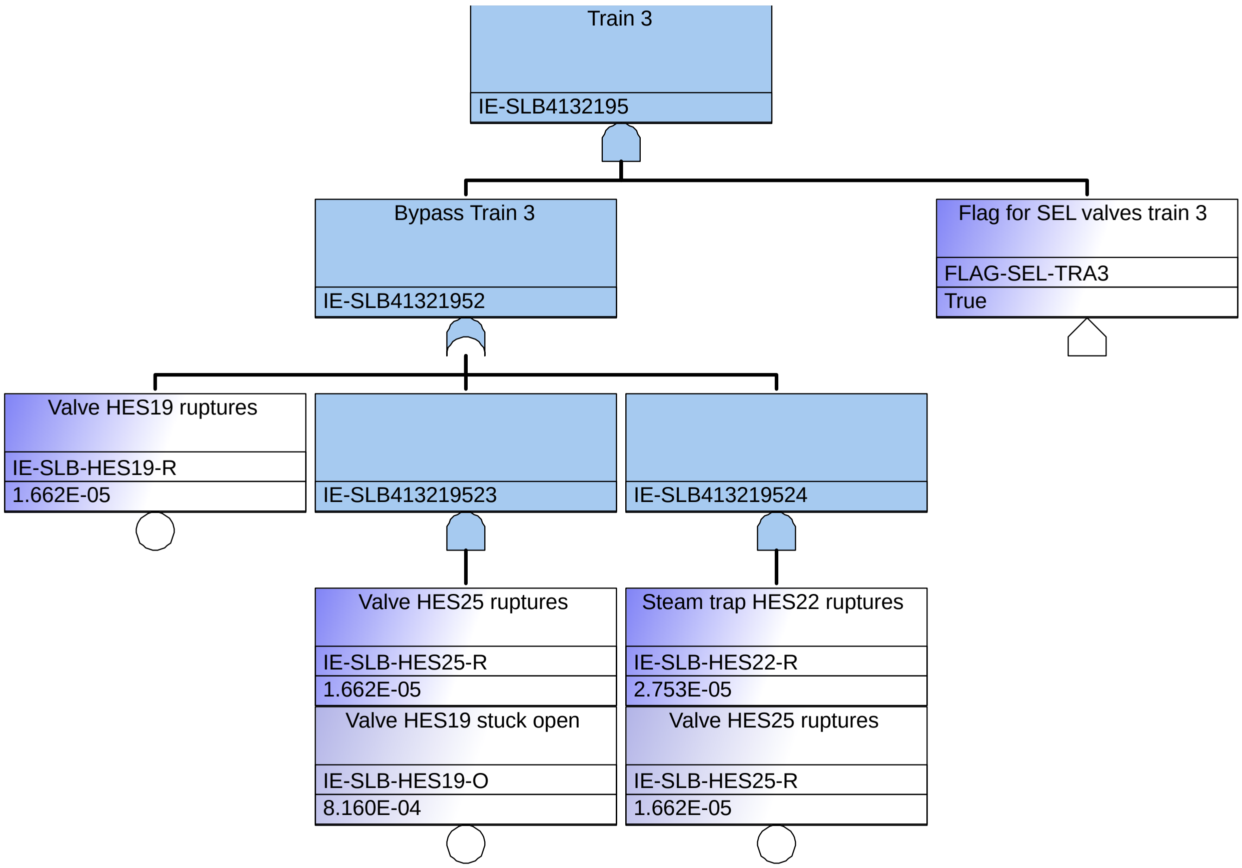

Figure 6-9. Leakage in the bypass train number 3 fault tree (IE-SLB4132195).

The fault tree showing leakage events in the EHX-1 heat exchanger subsystem is shown in Figure 6-10. Leakage may occur at the upstream valves (i.e., HES-2 and HES-14), or at the heat exchanger and is not isolated. Because HES-2 and HES-14 are installed in parallel, their failures are set in an OR gate. As can be inferred from the figure, although the probabilities for leakage events in the heat exchanger are relatively higher, they are coupled in an AND gate to the isolation failure events so the resulting probability for an unisolated leak is less significant than the valves' leakage probabilities.

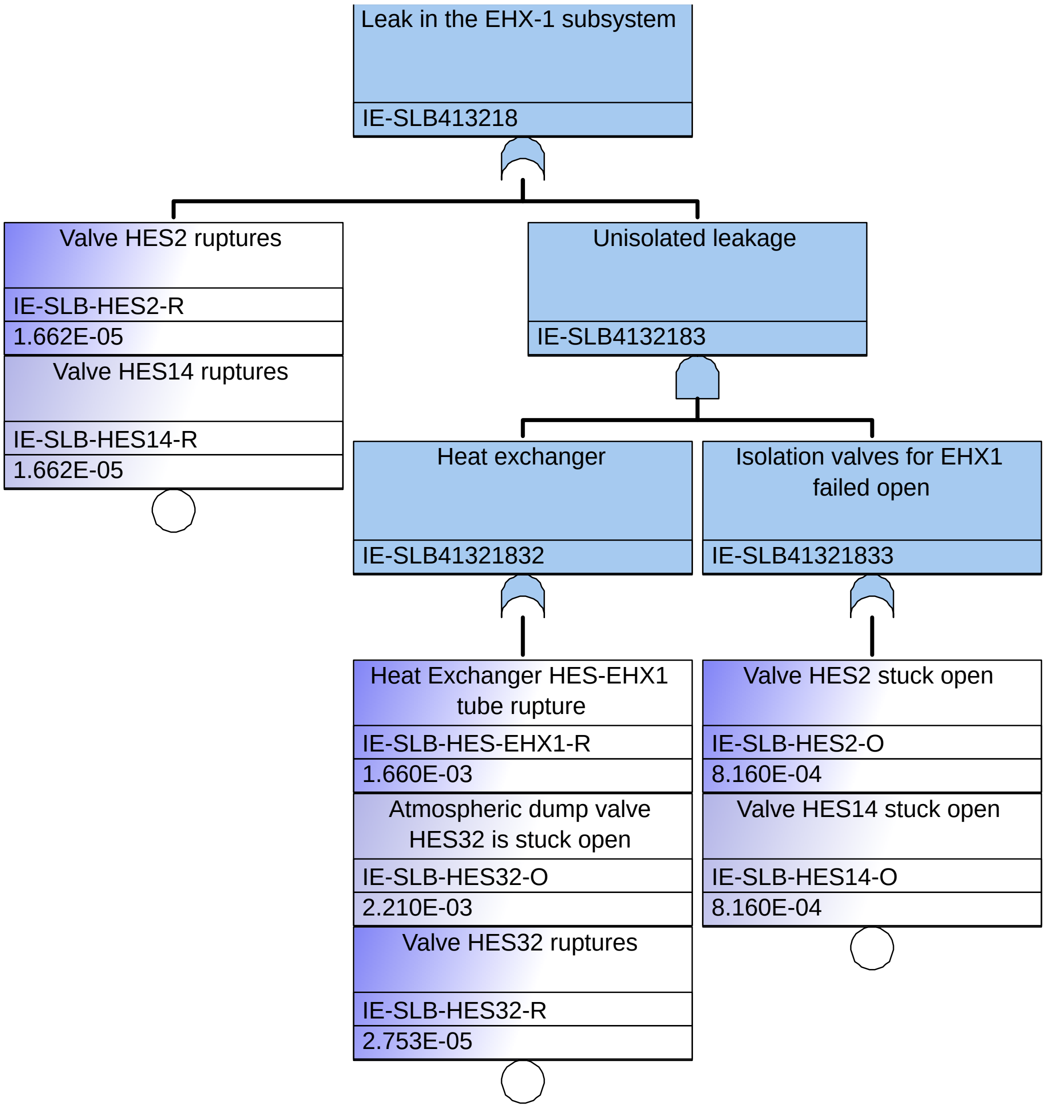

Figure 6-10. Leak in the first heat exchanger fault tree (IE-SLB413218).

The fault tree describing the leakage events at the downstream of condenser bypass trains is shown in Figure 6-11. The leak events may be caused by the rupture of components alongside the failure of all bypass trains to close. Because the bypass trains are designed in a parallel manner, the failure of a single train allows steam from the main steam line to flow to the break location. In such case, up to 5% of main steam flowrate may leak out from the secondary coolant inventory.

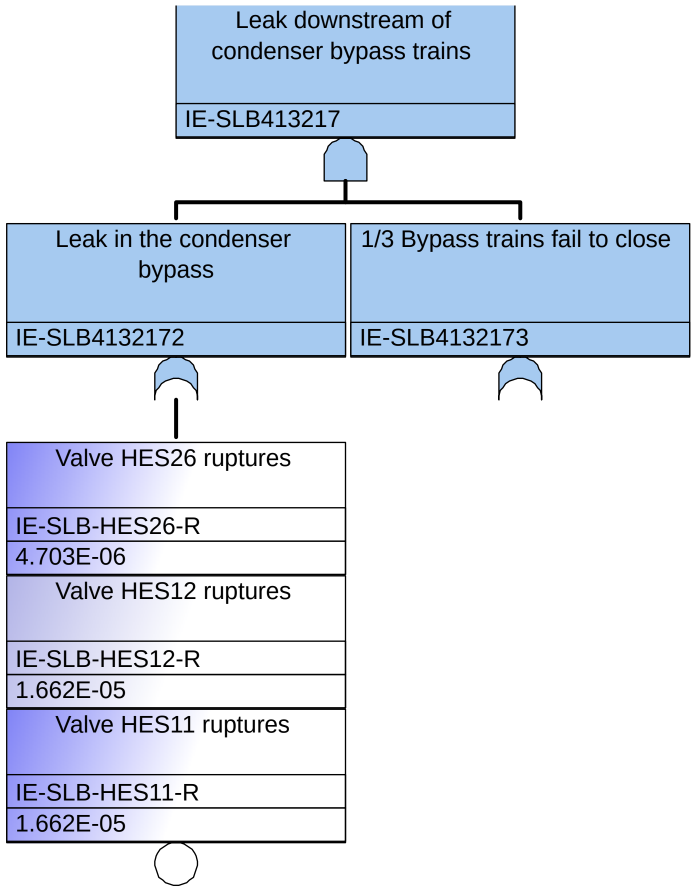

Figure 6-11. Leak downstream of condenser bypass trains fault tree (IE-SLB413217).

The fault tree describing the failure of bypass trains to close and isolate the downstream leakage is shown in Figure 6-12. The structure of this tree is straightforward where each train is represented by an AND gate of individual valve rupture events. The second and third trains are additionally coupled with their respective House Flag events to activate or deactivate the trains during the sensitivity analysis phase. An intermediate CCF event is included in the tree, which is shown in detail in Figure 6-13. The common cause events are constructed using the staggered Alpha-factor method. A combination of Flag events and their complements are added to select the active design configuration for the purpose of sensitivity analysis. A CCF for 1, 2, and 3 trains have a CCG of size 2, 4, and 6 respectively.

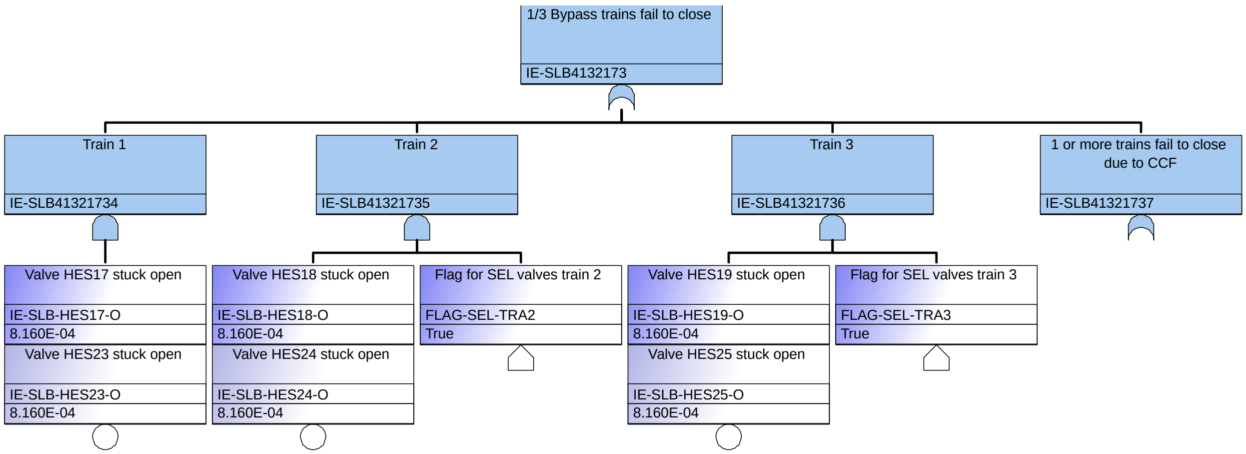

Figure 6-12. Fault tree of 1-out-of-3 bypass train fail to close (IE-SLB4132173).

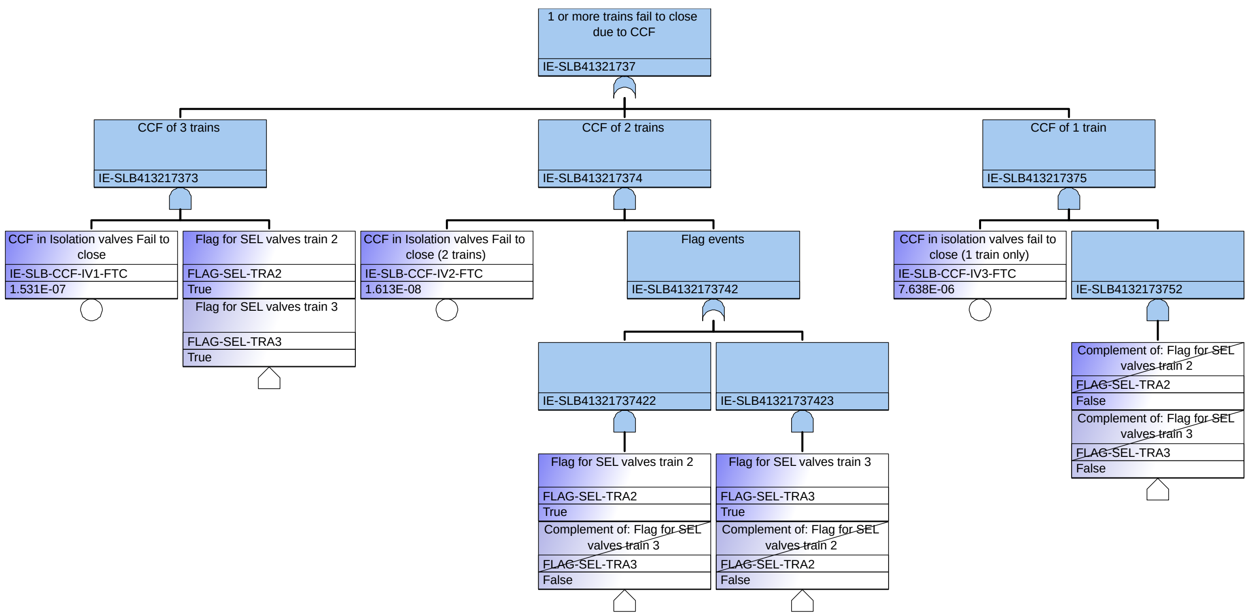

Figure 6-13. Fault tree of fail-to-close events due to CCF in the HES bypass trains (IE-SLB41321737).

Leakage events in the second heat exchanger (EHX-2) subsystem is shown in Figure 6-14. The steam loss may originate from the rupture of the isolation valves (i.e., HES-3 and HES-15), or an unisolated leakage as a combination of leak events downstream the isolation valves and the failure of those valves to close and terminate the steam flow. The most probable leakage event in this tree is the HES-29 atmospheric relief valve's failure to close, releasing steam from the EHX-2. However, this event is coupled with the isolation failures from the HES-3 and HES-15 valves in an AND gate. Therefore, the resulting risk contribution from this event is reduced.

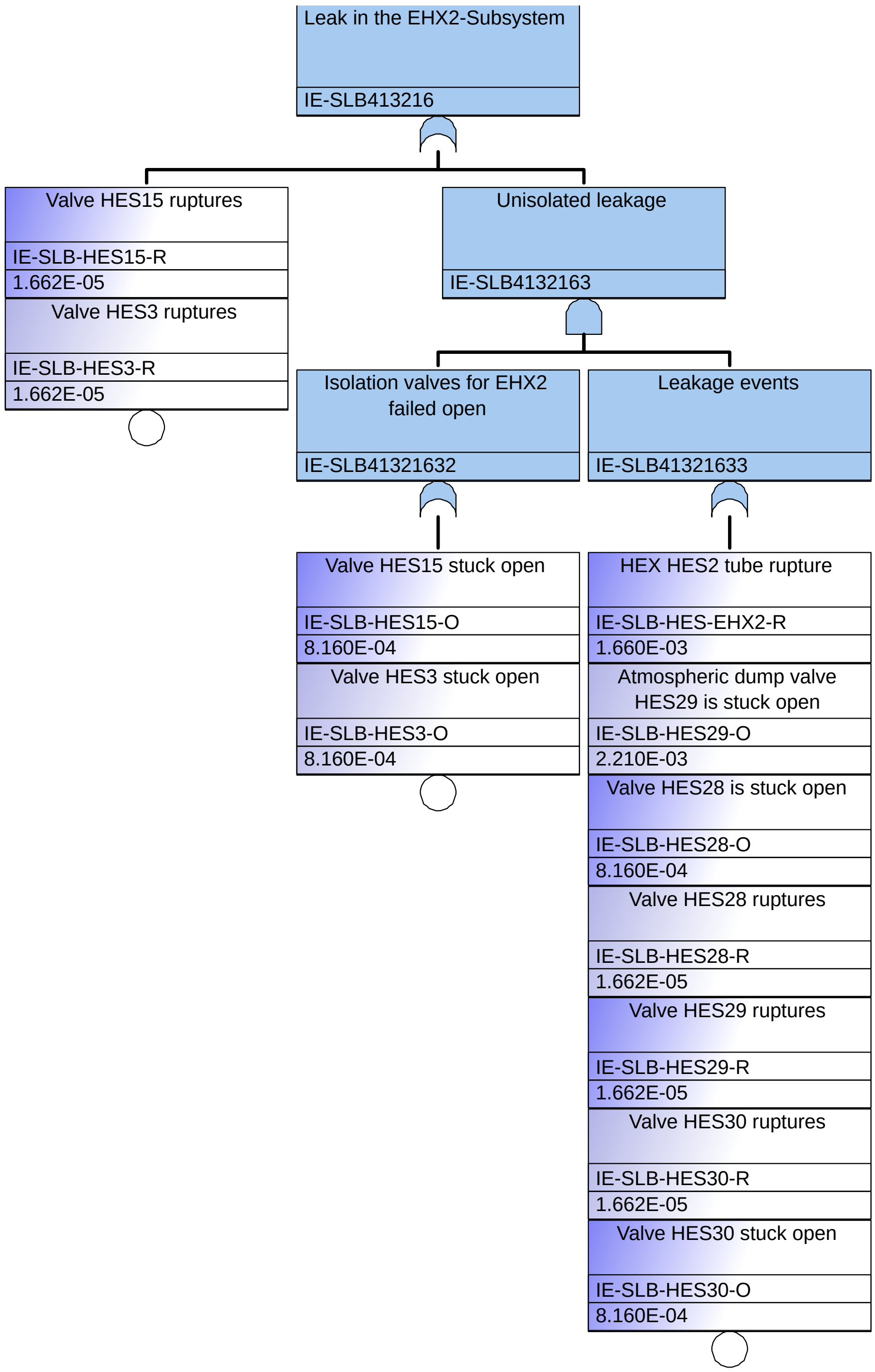

Figure 6-14. Leakage in the secondary heat exchanger fault tree (IE-SLB413216).

The fault tree describing leakage events in the SEP-1 tank subsystem is shown in Figure 6-15. The logic in this tree is quite similar to the leakage tree in the EHX-2 subsystem described earlier. The events consist of rupture of isolation valves (i.e., rupture of HES-4 and HES-6), and unisolated leakage in the tank and subsequent components following those valves. Looking at the tree structure, the latter leakage events have negligible statistical probability less than 1E-8 due to the AND logical gate connecting the basic events. Therefore, the significant contributor of a steam-loss event in this tree comes from the isolation valves themselves.

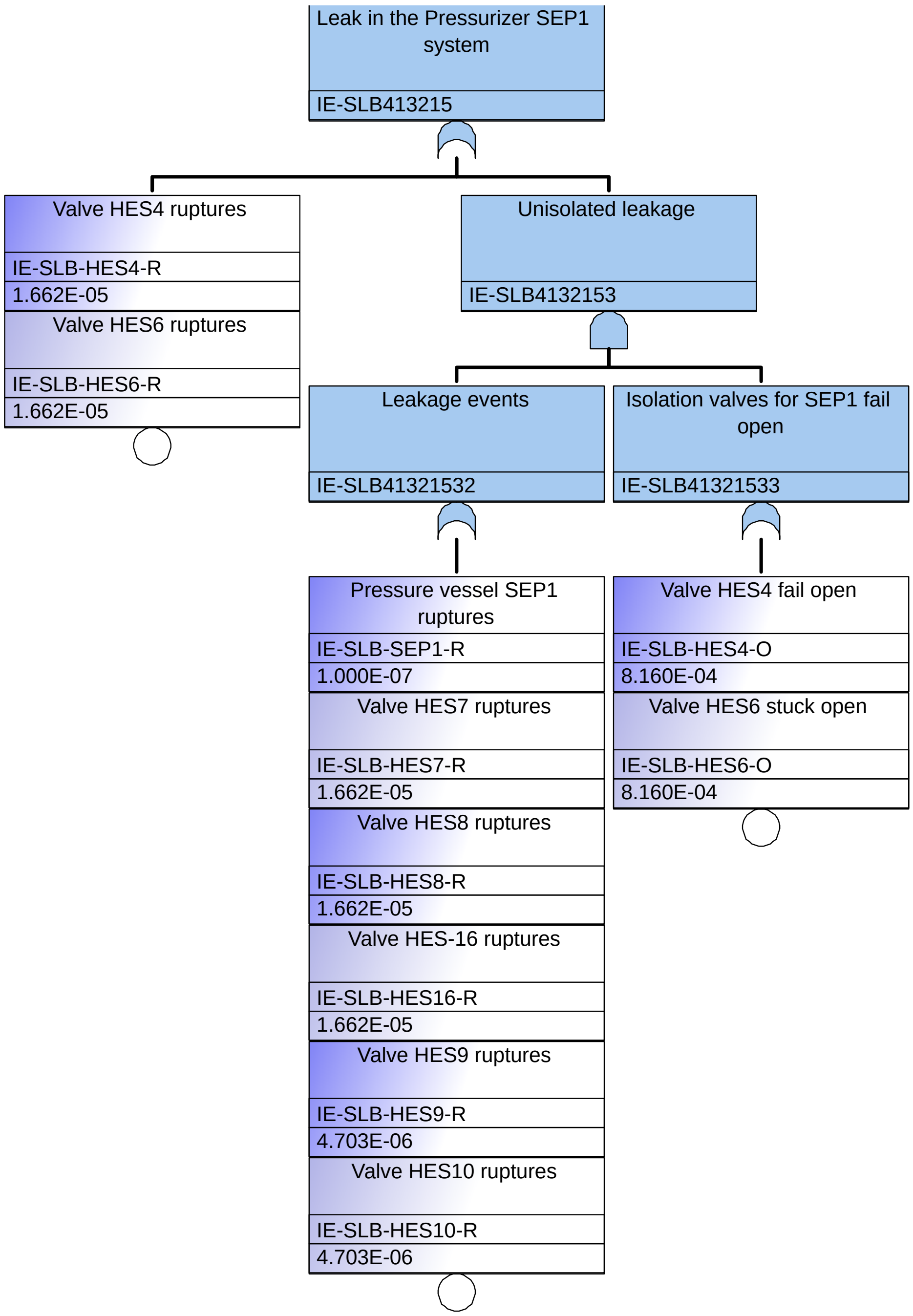

Figure 6-15. Leakage in the SEP1 subsystem fault tree (IE-SLB413215).

6.2 Generic PWR Model

The addition of an HES system into the steam line creates more venues for the steam to leak out either through pipe breaks or component ruptures. Therefore, one of the possible hazards considered in this study is an increased probability for steam leakage through the new system. In this study, a two-loop generic PWR model is used as a reference. The Event Tree for the Main Steam Line Break initiator is shown in Figure 6-16. A break in the main steam line causes the loss of ultimate heat sink and therefore the reactor must be tripped. The removal of reactor decay heat depends on whether steam generators are ruptured because of the steam line break. If steam generators are functioning, the Auxiliary Feedwater

(AFW) system supplies feedwater to the steam generators while the main steam/feedwater line is isolated. If the main steam line cannot be isolated, the AFW system cannot inject water due to the high pressure in the line and the High Pressure Injection (HPI) is used in its place. In case AFW system fails, the reactor heat is removed using the feed and bleed mechanism on the primary cooling line. The failure event of steam generators requires mitigation actions as prescribed in the Steam Generator Tube Rupture Event Tree. Meanwhile, the failure of the reactor trip requires mitigation procedures laid out in the Anticipated Transient Without Scram (ATWS) Event Tree. These Event Trees are provided in Appendix A.

Additionally, the existence of a hydrogen production plant near the NPP may create another hazard (i.e., hydrogen explosion). This explosion may cause significant blast pressure and missiles that may damage surrounding structures including the plant's switchyard components. The loss of switchyard components may trigger a LOOP event that may cause a transient to the reactor. This event has been taken into consideration in the PRA model as shown in Figure 6-18. The LOOP IE trips the reactor and brings the emergency power online. The auxiliary feedwater system is then activated to maintain cooling on the secondary coolant loop. If the pressure-operated safety relief valves are closed and Reactor Coolant Pump (RCP) seal cooling is maintained, this mitigation action is sufficient to safely shut down the reactor. If RCP seal cooling fails, the mitigation procedure switches to the LOOP-1 Event Tree, shown in Figure 6-20. This procedure involves activating a controlled bleed-off in the primary cooling system while maintaining the reactor coolant subcooling. This action should prevent the RCP seal from failing due to overpressure and shuts down the reactor safely. If the RCP seal fails, the operator has 1 hour to recover power before the situation can be declared as a Medium-Size-Loss-of-Coolant-Accident (MLOCA). If power is recovered within that timeline, the operator can proceed with the HPI to make up the inventory of the primary cooling system until the reactor is brought to a safe shutdown state.

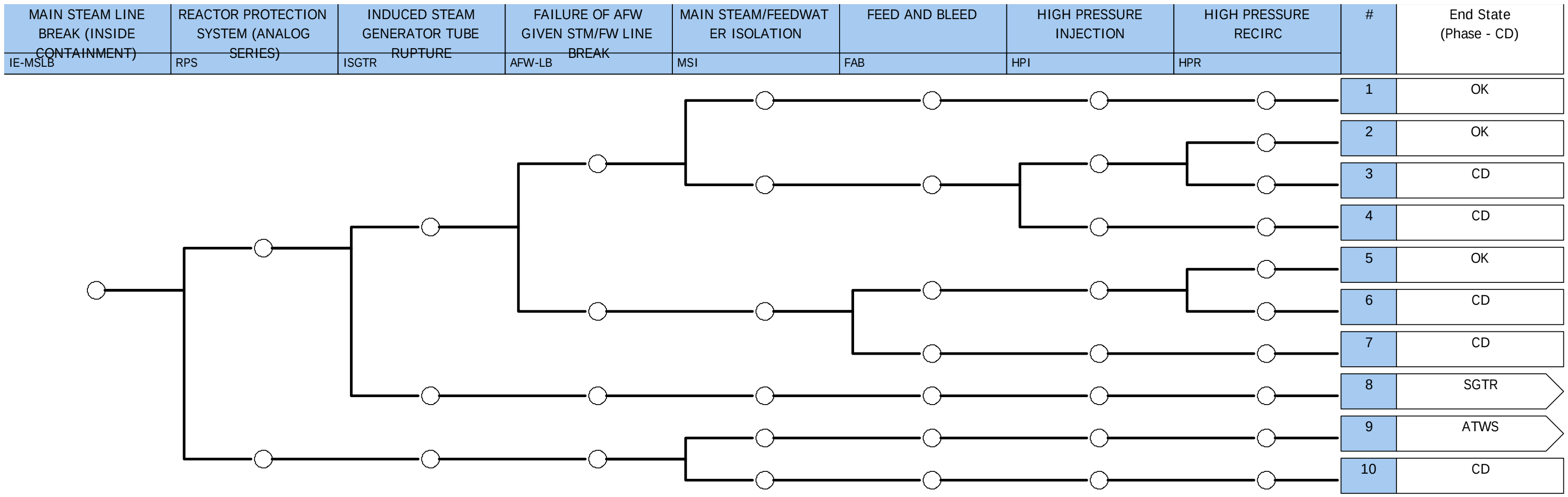

Figure 6-16. Main Steam Line Break Event Tree (IE-MSLB).

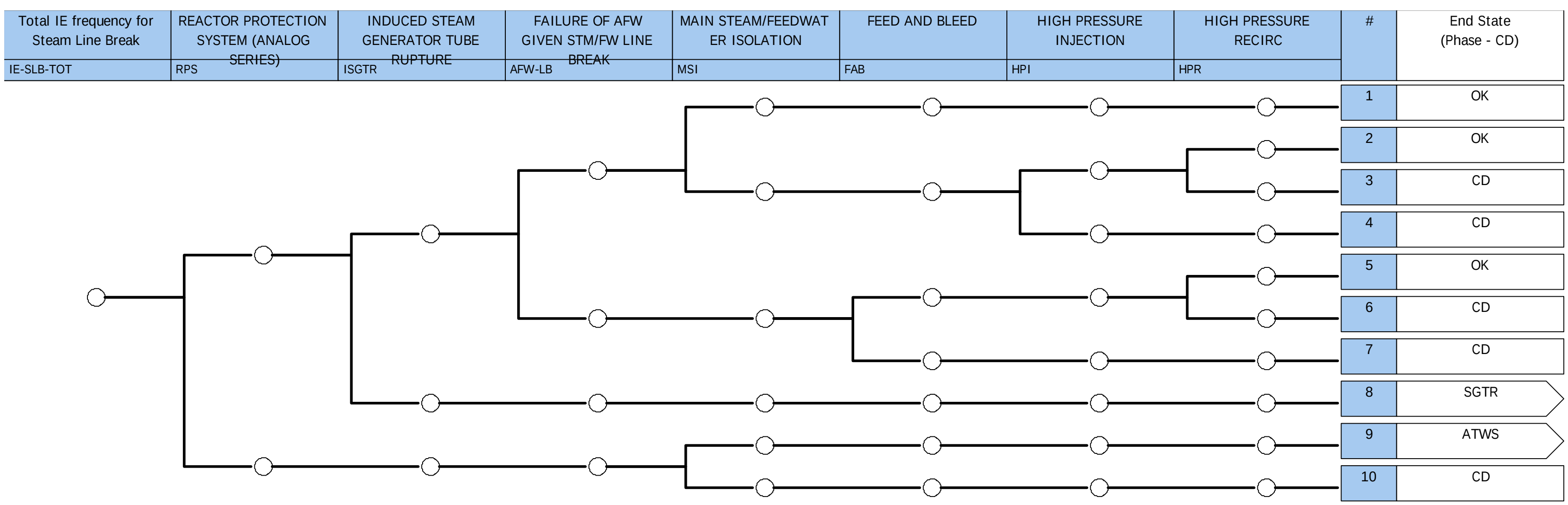

Figure 6-17. Main Steam Line Break Event Tree with HES system (IE-SLB-TOT).

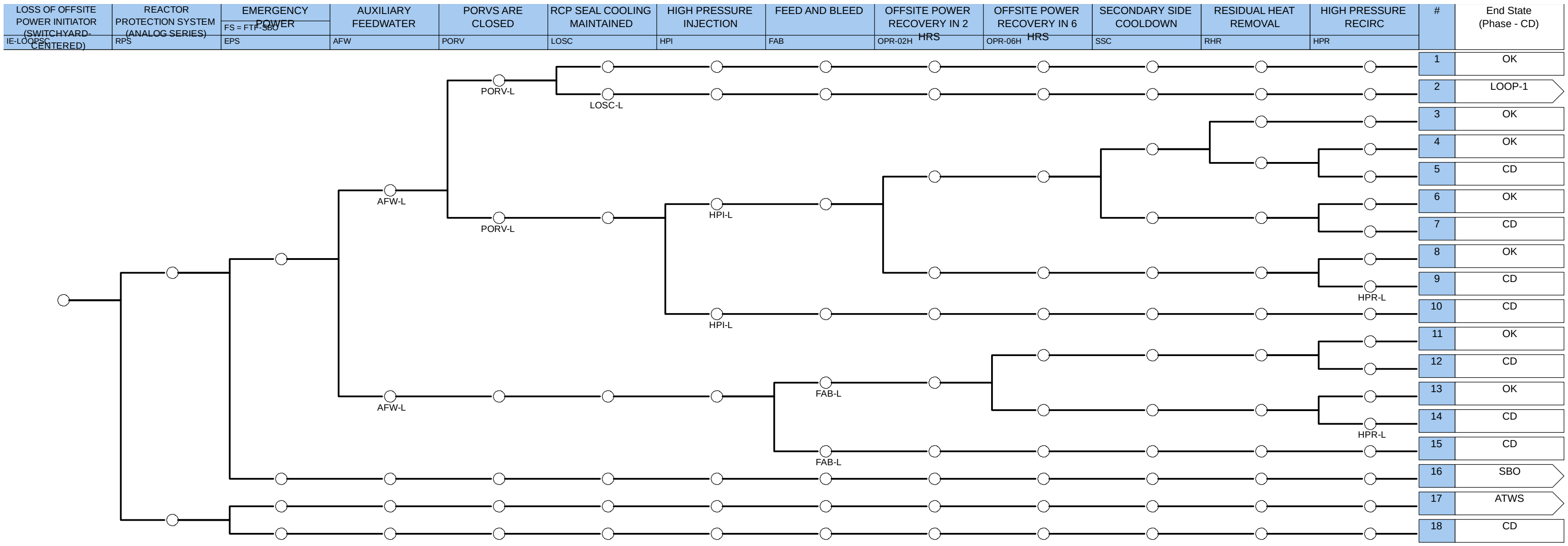

Figure 6-18. LOOPSC Event Tree (IE-LOOPSC).

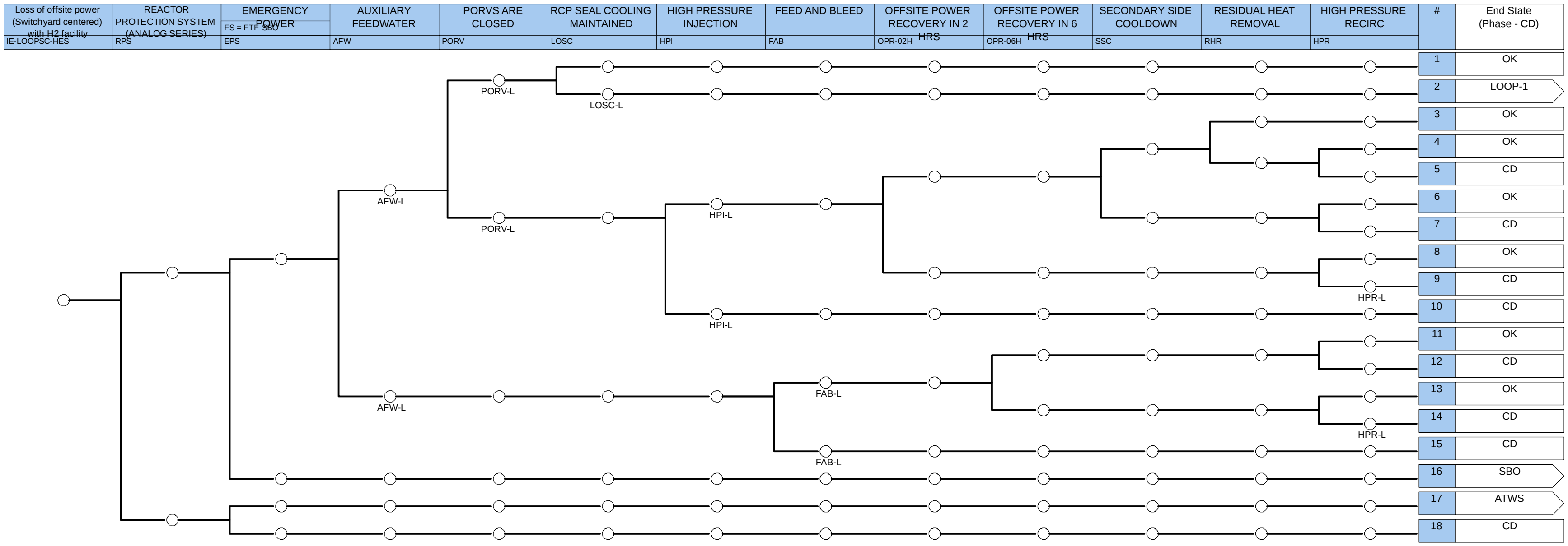

Figure 6-19. LOOPSC with HES Event Tree (IE-LOOPSC-HES).

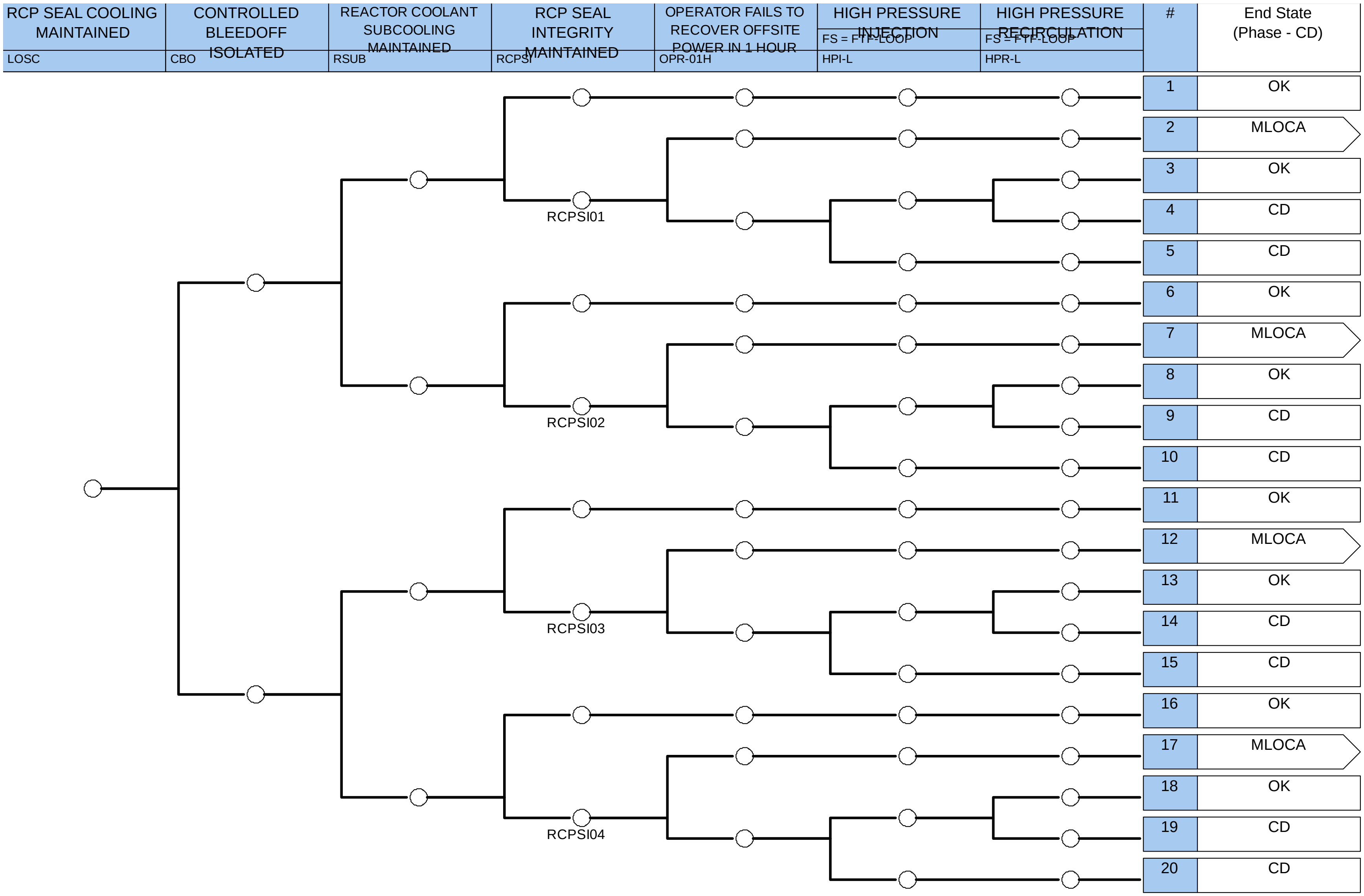

Figure 6-20. LOOP-1 Event Tree (LOSC).

6.2.1 HES linkage into the PWR Model

The addition of the HES that taps into the main steam line of a nuclear power plant creates additional points where steam may leak out of the secondary cooling loop. The frequency of steam leak in the HES system is estimated using the fault tree described in the previous section. The additional frequency from HES is added to the existing base IE frequency of the steam line break event tree using a fault tree, as shown in Figure 6-22. The top event of this tree becomes the total steam line break IE frequency, which is used as the initiator for the new steam line break Event Tree as shown in Figure 6-17.

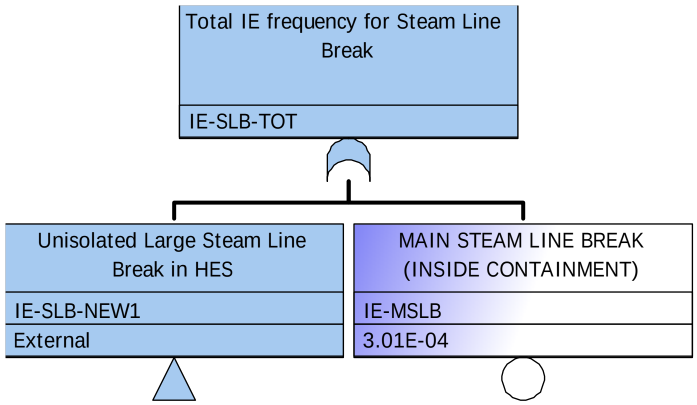

Figure 6-21. Fault Tree for Total Initiating Event frequency for main steam line break (IE-SLB-TOT).

Another possible hazard identified in the previous section is the switchyard failure event due to hydrogen leakage and explosion. This switchyard failure may cause a LOOP event. The severity of hydrogen explosion and its annual frequency was calculated in a reference report [11]. The conservative leak frequency estimate from that reference is adopted in this work. A fault tree is constructed, as shown in Figure 6-23, to model this additional risk. The switchyard component may fail when a hydrogen leak occurs, plant operator fails to isolate the leakage within 2 hours, the building ventilation fails to disperse the hydrogen to the atmosphere, and a spark occurs igniting the accumulated hydrogen cloud. This is the MCA scenario highlighted in Figure 6-21, which is assumed to be the bounding accident to damage the switchyard components. The hydrogen ignition probability is a function of hydrogen leakage rate [13]; however, in this fault tree, a conservative probability value of 0.35 is selected for the event. This scenario ignites a total of 13.2 kilograms of hydrogen and creating an overpressure of 0.39 psi to the nuclear plant structures located 1 km from the hydrogen plant. This overpressure may fail the switchyard components with a statistical probability of 0.95 and create a LOOP event. As with the steam line break hazard, the top event of this fault tree is set as the total initiator frequency for the new LOOP event tree as shown in Figure 6-19.

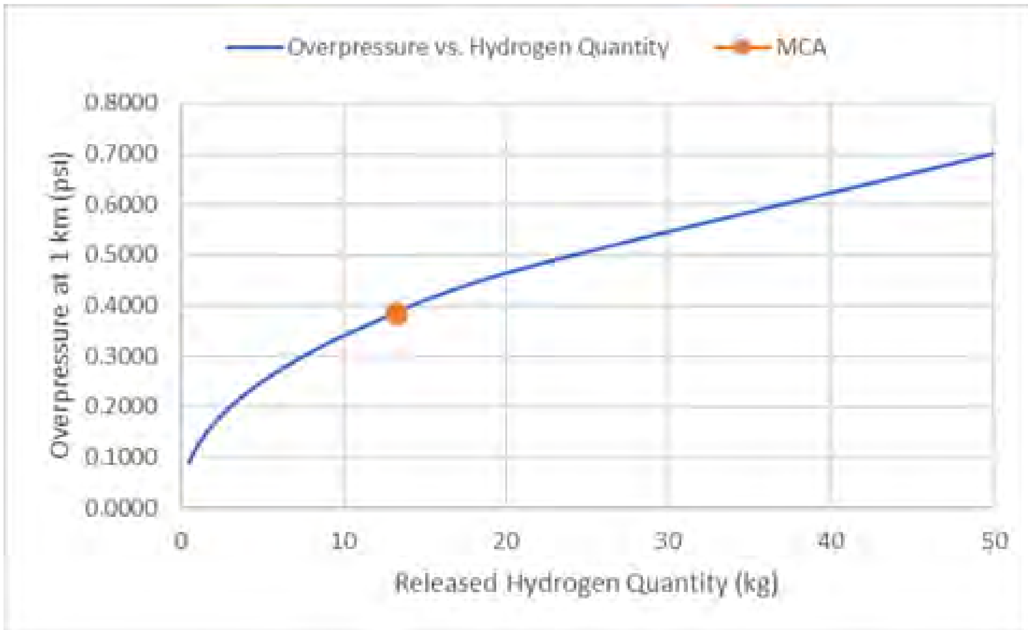

Figure 6-22. Overpressure at a distance of 1 km due to hydrogen detonation.

It is conservatively assumed that the hydrogen cloud detonation scenario always leads to the MCA scenario. With this assumption, the probability for an MCA scenario is 1 whenever there is an unmitigated hydrogen leakage. This conservative assumption is taken because of the absence of data available on the time distribution of uncertainty sources affecting the hydrogen leakage time (i.e., operator's timing to isolate the leakage, timing of spark occurrences, and actuation timing of building ventilation). These uncertainties may lower the probability for an MCA event. For example, if the leakage time is assumed to occur uniformly between 5 to 120 minutes, the total fragility may be calculated by uniformly sampling the quantity of released hydrogen in Figure 6-22 up to the MCA scenario and performing a look-up conversion of the detonation's overpressure to the switchyard fragility using Figure 5-8 (above). The total switchyard fragility estimated using a Monte Carlo simulation of 10,000 samples is found as 0.76, which is less than the fragility for the MCA event (IE-LOOPSC-HES2144A). For that reason, it is reasonable to accept that the MCA detonation assumption is conservative.

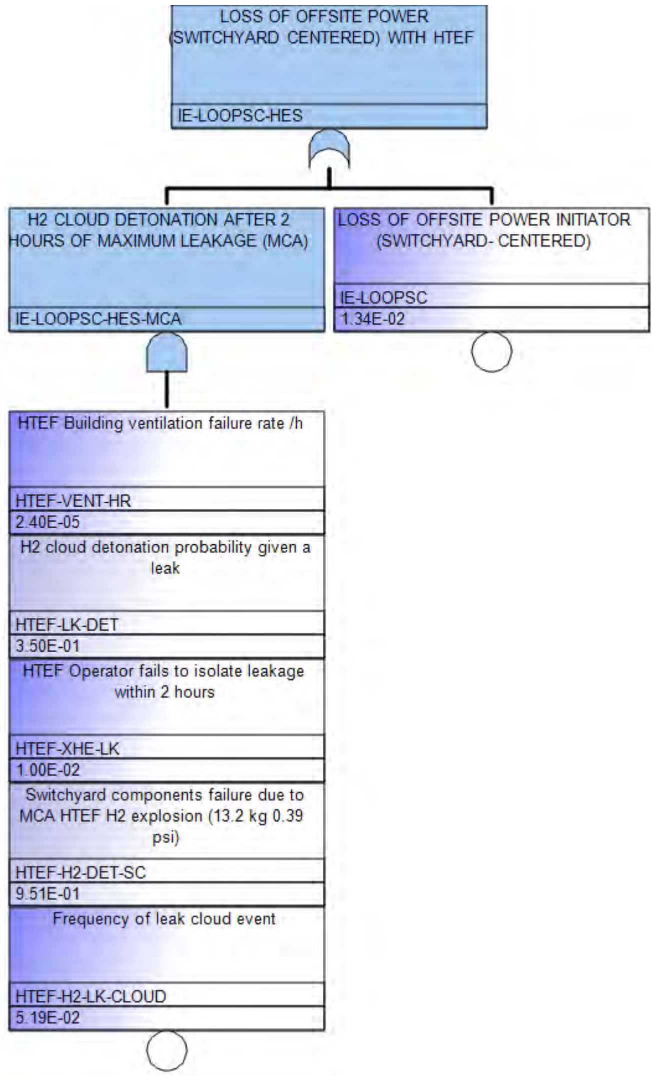

Figure 6-23. Total frequency of LOOP with Hydrogen Production Facility (IE-LOOPSC-HES).

6.3 Generic BWR Model

Similar to the PWR, the HES system in the BWR taps steam from the main steam line after the MSIVs. A loss of up to 5% of steam flow rate due to a leakage event in the HES may lead to a general transient event. The mitigation procedure for this event is shown in Figure 6-24. The transient can be mitigated safely if reactor power generation is shut down, the offsite power is available, the safety relief valves remain closed to preserve coolant inventory, and the power conversion system is running. If this power conversion system fails, the HPI system is activated followed by suppression pool cooling. Without the automatic suppression pool cooling, operators need to depressurize the reactor manually and perform the control rod drive injection. Further mitigation sequences can be deducted from the figure, in which various redundant measures are available including a low-pressure injection (LPI) system, shutdown cooling, containment spray, and containment venting.

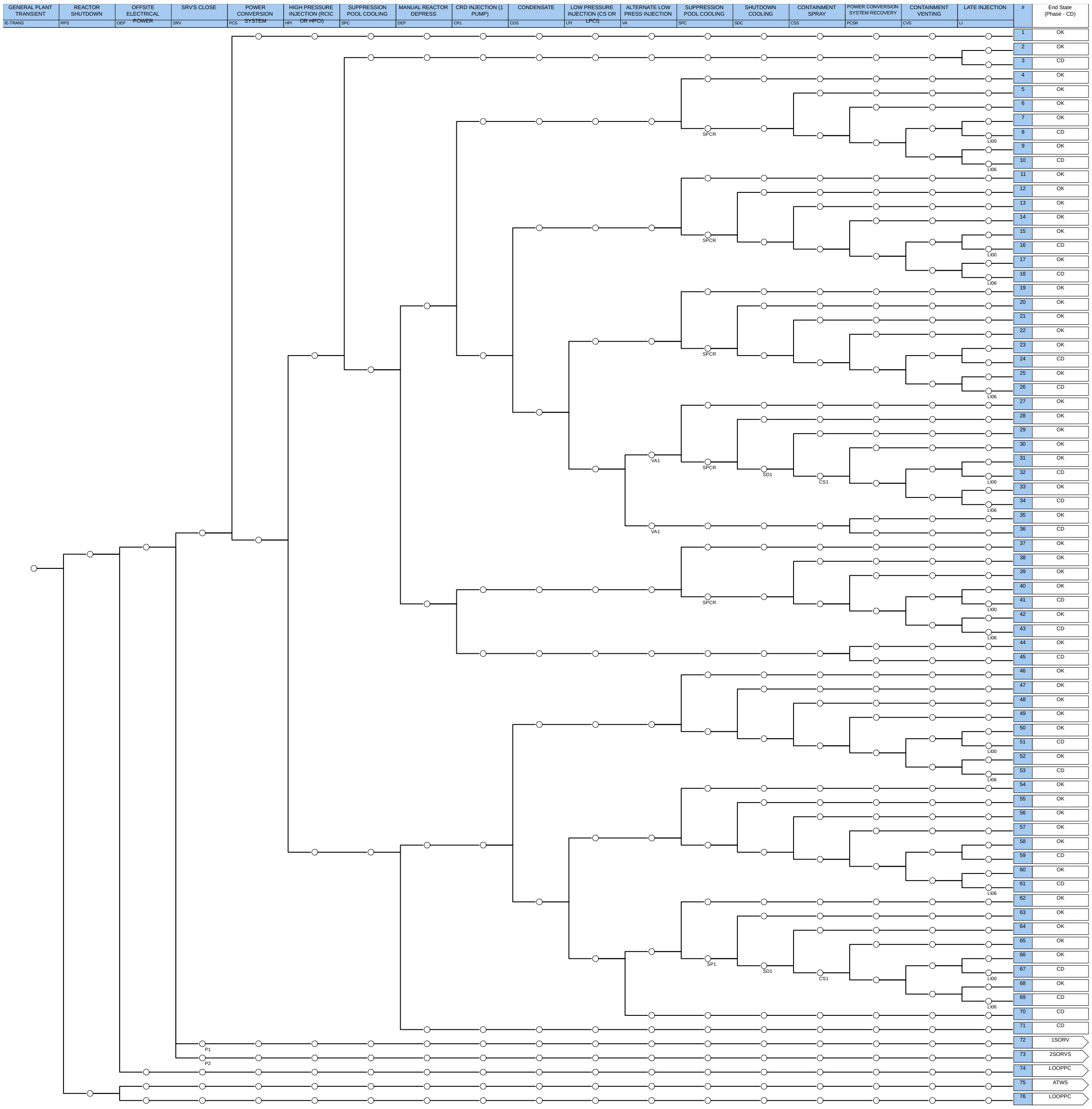

Figure 6-24. General Transient Event Tree (IE-TRANS).

As with the PWR plant, the presence of the hydrogen facility near the BWR plant may cause a hydrogen leakage that leads to an explosion. This event may create a blast pressure that damages the switchyard components. When it happens, a LOOP event may occur. The mitigation procedure due to a switchyard-related LOOP IE is shown in Figure 6-25. Upon a LOOP event, the reactor is shut down and emergency power is activated. If safety relief valves remain closed while the HPI system and the suppression pool cooling actuate, the reactor will be in a safe shutdown state. The tree logic is quite similar to the general transient tree. Redundant safety measures are incorporated in the tree including manual depressurization followed by an LPI, an alternate LPI, shutdown cooling, containment spray, and containment venting to prevent an overpressure event.

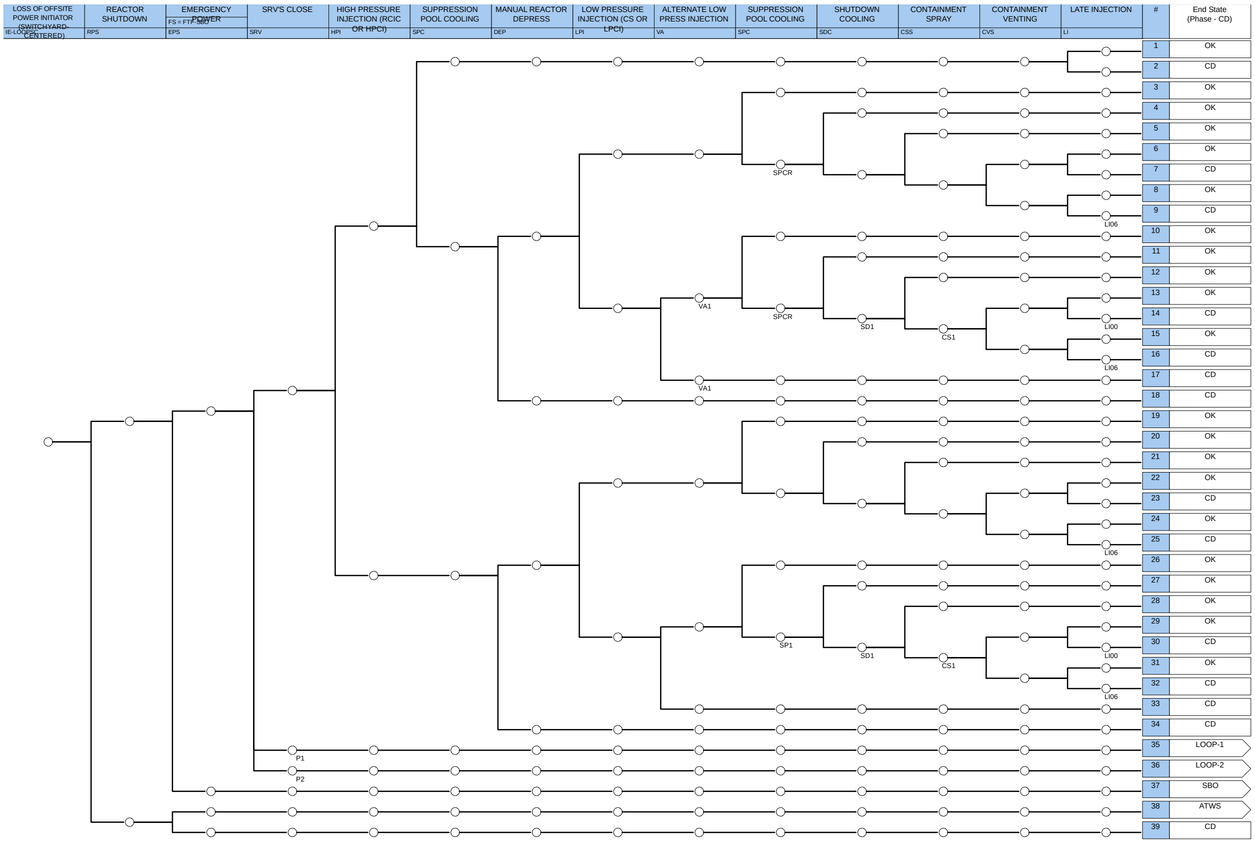

Figure 6-25. Loss-of-offsite-power (LOOP) (Switchyard-centered) Event Tree (LOOPSC).

6.3.1 HES Linkage into the BWR Model

The mitigation procedure for a steam line break in the HES system is shown in Figure 6-26. When the event occurs, the core will be damaged if the Reactor Protection System (RPS) fails, or if the MSIVs fail to close. If both systems function properly, the mitigation tree transfers to the General Transient event tree as shown in Figure 6-24 (above). However, since the General Transient tree is used as is, there needs to be a set of linkage rules to customize the tree based on the initiator (i.e., a steam line break in the HES). This linkage rules are set as pictured in Figure 6-27. It instructs SAPHIRE to activate the LSSB-HES Flag Set when the initiator is IE-LSSB-HES. This instruction is also carried over to the transfer event tree, i.e. General Transient. The LSSB-HES Flag set is set up as shown in Figure 6-28. It activates the HE-SLB-TOT House event and changes its state from False to True.

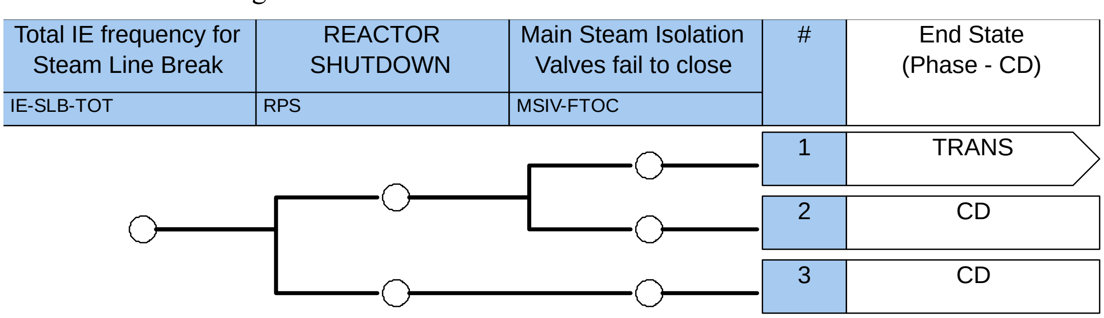

Figure 6-26. Initiating Event for Steam Line Break in the HES system (IE-SLB-TOT).

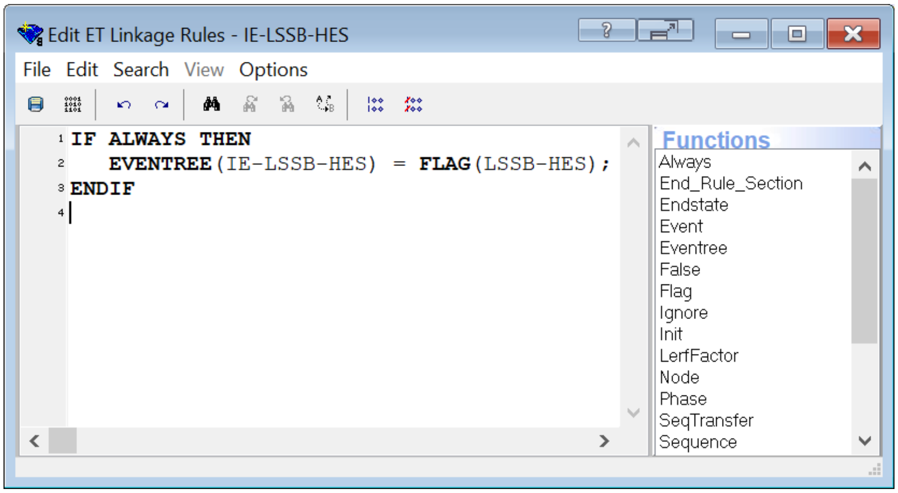

Figure 6-27. Linkage rules for the IE-LSSB-HES event tree

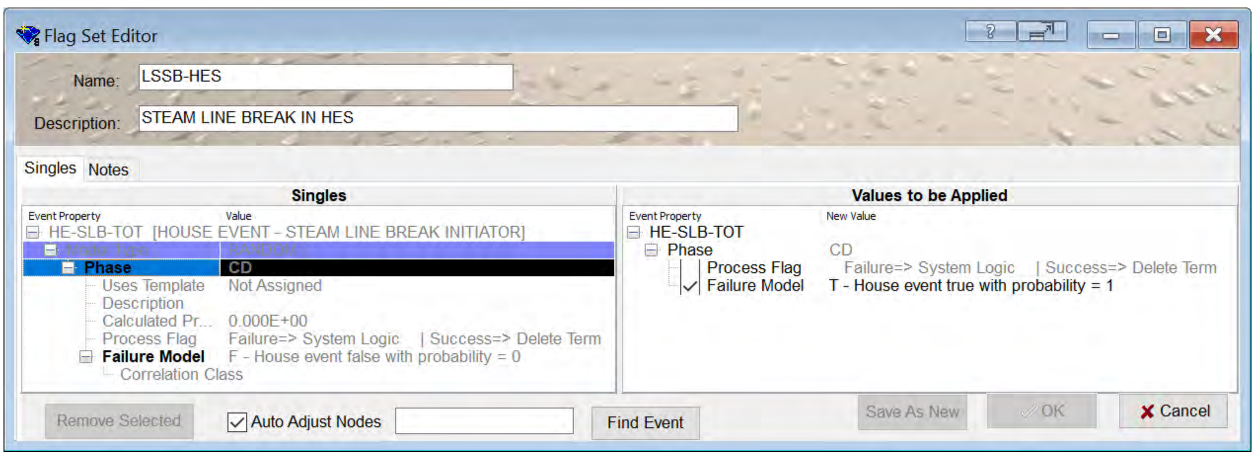

Figure 6-28. LSSB-HES flag editor.

As indicated in Figure 6-26, the IE-SLB-TOT event tree transitions to the TRANS tree only when RPS functions successfully. For that reason, the RPS top event in the TRANS tree should not be evaluated again when the sequence originates from IE-SLB-TOT. This logic is made possible by adding a complement of HE-SLB-TOT as shown in Figure 6-29. This event is coupled in an AND gate with the other events that may cause RPS to fail. With this configuration, when the IE-SLB-TOT event tree transitions to the TRANS tree, the LSSB-HES Flag is activated and the HE-SLB-TOT House Event is set to true. Therefore, its complement becomes false and the RPS failure top event does not occur. Meanwhile, when the TRANS tree is activated after the MSIV is closed, the Power Conversion System (PCS) is always off. This logic is implemented by adding the HE-SLB-TOT house event in an OR gate to the PCS and PCS recovery fault trees as shown in Figure 6-30 and Figure 6-31 respectively.

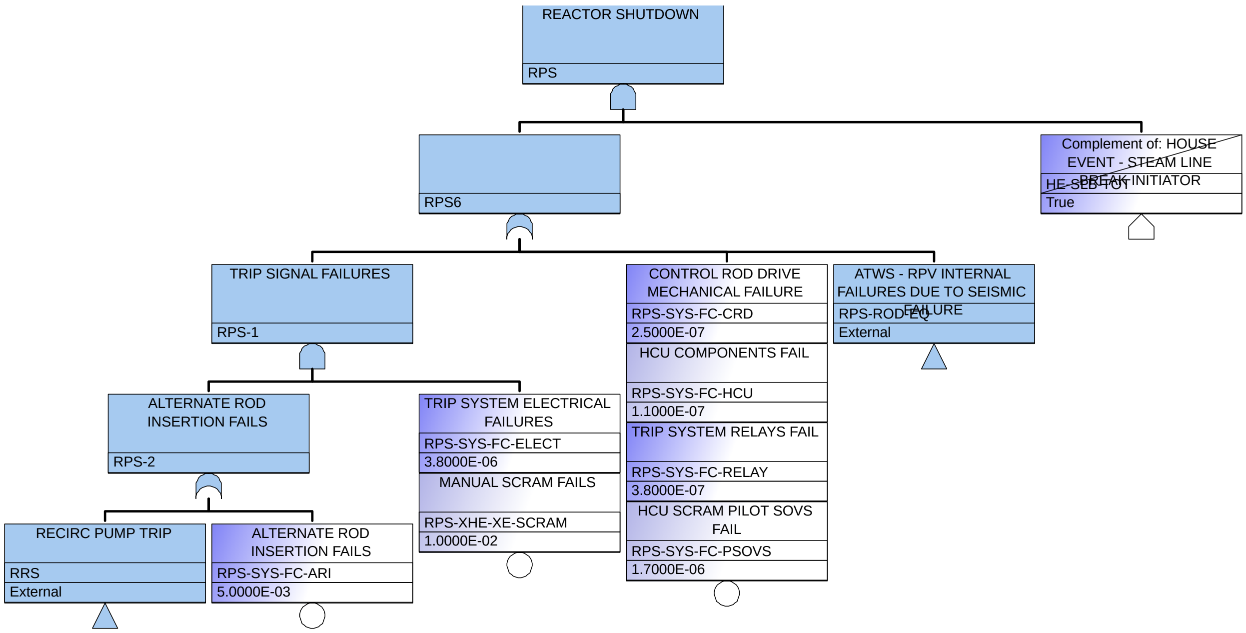

Figure 6-29. Reactor Protection System fault tree (RPS)

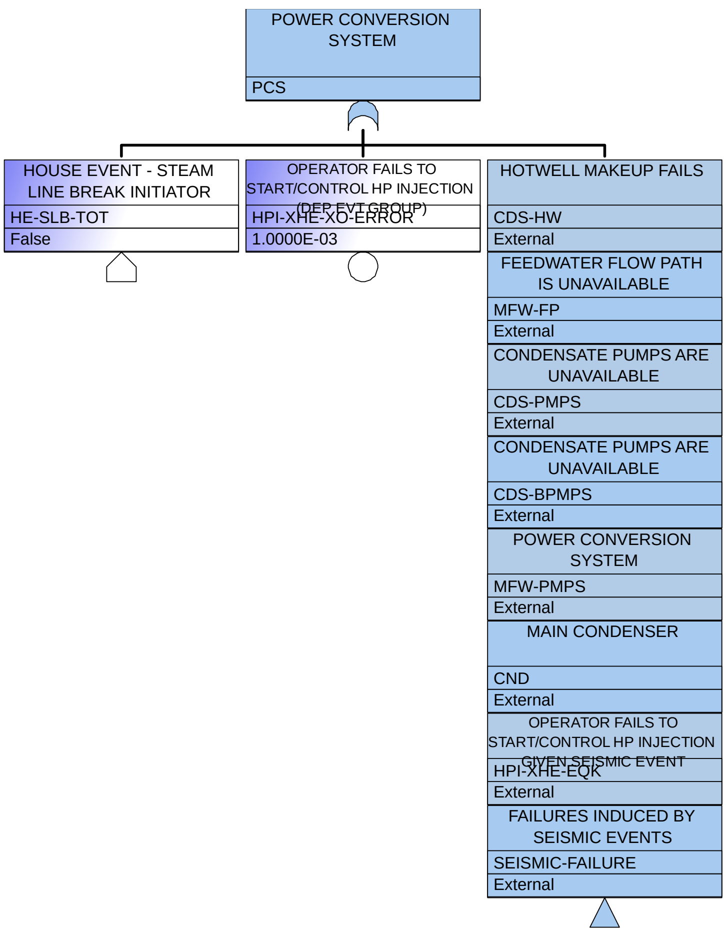

Figure 6-30. Power Conversion System fault tree (PCS)

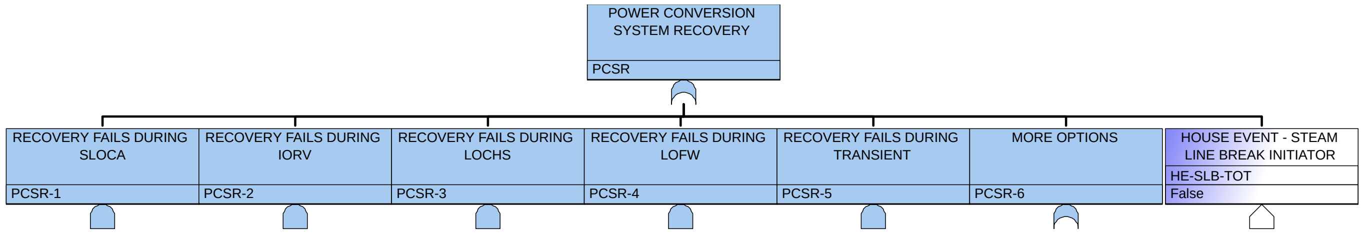

Figure 6-31. Power Conversion System Recovery fault tree (PCSR).

6.4 Sensitivity Studies

This section describes several sensitivity studies conducted in the risk analysis. The factors considered in the sensitivity analysis are:

1. Whether to use one or two isolation valves for HES
2. Whether to use one, two or three bypass trains in the HES system
3. Whether to equip dedicated ceiling ventilation system at the hydrogen plant to vent leaked hydrogen from inside the building to the atmosphere.

A Change Set is used to alter the HES design configuration from two isolation valves to only one. This Change Set is shown in Figure 6-32. When this change set is activated, the HES-ISOV-FLAG switches state from True to False, which affects the fault trees associated with the HES isolation valves. Meanwhile, the change sets for Train number 2 and 3 of the HES steam bypass trains are shown in Figure 6-33 and Figure 6-34. These change sets alter the state of FLAG-SEL-TRA2 and FLAG-SEL-TRA3 respectively from True to False. The initial configuration of the HES system consists of two isolation valves and three trains.

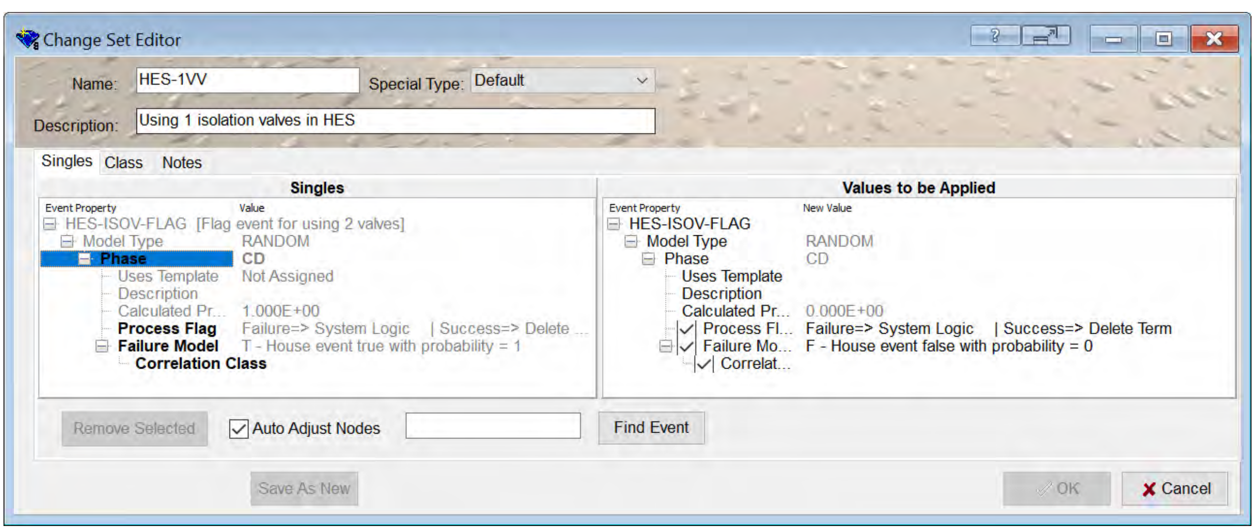

Figure 6-32. Change Set for HES isolation valves.

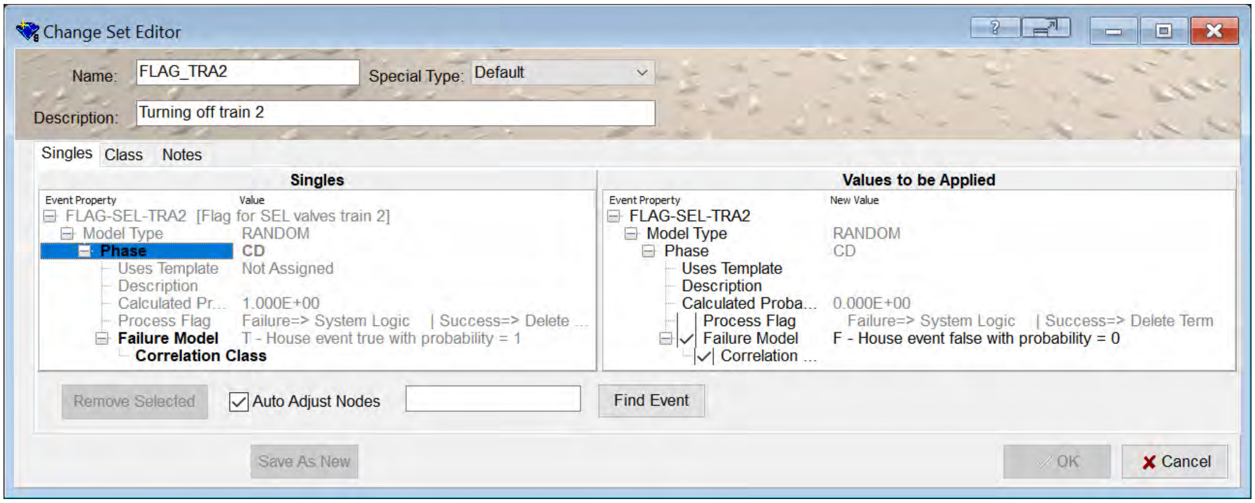

Figure 6-33. Change Set for Train 2 of HES bypass trains.

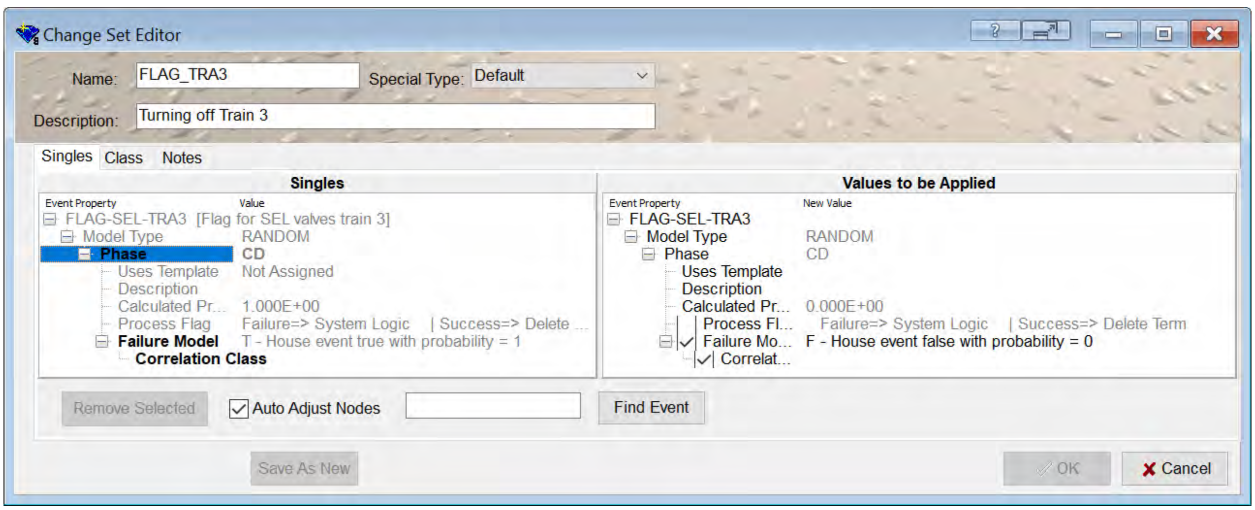

Figure 6-34. Change Set for Train 3 of HES bypass trains.

7. PRA RESULTS

7.1 PWR PRA Results

The initial IE frequency for Main Steam Line Break in the PWR model 3.01E-4/year and the CDF from this event is 2.542E-7/year. With the installation of the HES system, the resulting frequency for this event is 3.18E-4/year, or an increase of 5.6% from the initial value. The new CDF is 2.667E-7/year, or an increase of 4.9% from its initial frequency.

For the switchyard-related LOOP event, the initiator frequency is determined by the operator's performance to seal the leak within 2 hours as the bounding time for the MCA event. In this model, the SPAR-H human reliability model was utilized to estimate the operator's failure probability. If all the Performance Shaping Factors (PSFs) are set at their nominal values, the operator's failure to isolate the leakage in 2 hours is 1E-2. With this value, and without the presence of a dedicated ceiling ventilation system to vent out the hydrogen leakage, the IE frequency increases slightly by 1.3% from 1.34E-2 to

1.357E-2. Even so, this estimate may be rather conservative, because 2 hours is a reasonably ample time to actuate a valve isolating the leakage. Furthermore, it is informed in reference [11] that 2 hours is the longest time for the operator action in this scenario, which indicates that it is more than the average time required to perform such action. With that consideration, a more realistic operator failure probability is estimated as 1E-4 by setting the available time PSF as expansive. With this estimate, an increase in the IE frequency is not significantly observed. The variations on the dedicated ceiling ventilation system is performed to investigate the possible design options on the hydrogen plant. The highest risk increase of 1.4% rise from the initial CDF is observed when the SPAR-H timing is set at the nominal value and there is no dedicated ceiling ventilation system to vent the leaked hydrogen. These results are summarized in Table 7-1.

Table 7-1. Summary of PRA results for PWR.

Risk metric	Case	Initiating Event Frequency (/y) (Δ%)	Core Damage Frequency (/y)	Cutsets
Steam line break IE frequency	Nominal	3.01E-4		1
Steam line break IE frequency with HES system	Base assumptions	3.18E-4 **(+5.6 %)**		95
Switchyard-related LOOP frequency	Nominal	1.34E-2		1
Switchyard-related LOOP frequency with HES system, conservative SPAR-H timing, without dedicated ceiling ventilation system	Base assumptions	1.357E-2 **(+1.3%)**		2
Switchyard-related LOOP frequency with HES system, conservative SPAR-H timing, and dedicated ceiling ventilation system	Sensitivity	1.34E-2		2
Switchyard-related LOOP frequency with HES system, realistic SPAR-H timing, and dedicated ceiling ventilation system	Sensitivity	1.34E-2		2
Switchyard-related LOOP frequency with HES system, realistic SPAR-H timing, without dedicated ceiling ventilation system	Sensitivity	1.34E-2		2
CDF due to steam line break	Nominal		2.542E-7	1912
CDF due to steam line break with HES system	Base assumptions		2.667E-7 **(+4.9 %)**	1931
CDF due to switchyard-related LOOP	Nominal		2.749E-7	6183
CDF due to switchyard-related LOOP with HES system, conservative SPAR-H timing, without dedicated ceiling ventilation system	Base assumptions		2.787E-7 **(+1.4%)**	6243
CDF due to switchyard-related LOOP with HES system, conservative SPAR-H timing, and dedicated ceiling ventilation system	Sensitivity		2.749E-7	6183

Risk metric	Case	Initiating Event Frequency (/y) (Δ%)	Core Damage Frequency (/y)	Cutsets
CDF due to switchyard-related LOOP with HES system, realistic SPAR-H timing, and dedicated ceiling ventilation system	Sensitivity		2.749E-7	6183
CDF due to switchyard-related LOOP with HES system, realistic SPAR-H timing, without dedicated ceiling ventilation system	Sensitivity		2.749E-7	6183

Based on the results in Table 7-1, the plant total CDF and Large Early Release Frequency (LERF) are calculated using the conservative assumption. Those 2 hours are the nominal time to locate and seal hydrogen leakage (a conservative SPAR-H timing) and in which the hydrogen plant does not have a dedicated ceiling ventilation system. Furthermore, the base design of HES system is selected (i.e., an HES system with two isolation valves and three bypass trains) and the base assumptions listed in Table 5-1 are followed. These results are shown in Table 7-2. The flexible NPP operation with an HES system increases CDF by 5.47E-7 (6.56%) and LERF by 6E-10 (0.07%).

Table 7-2. Risk metric for PWR.

	Total CDF (/y)	Total LERF (/y)
NPP without HES	8.334E-6	8.039E-7
NPP with HES	8.881E-6	8.045E-7

The sensitivity analyses for possible configurations of a dedicated ceiling ventilation system and expansive time to isolate the hydrogen leak at the HTEF show that a minimal increase in safety is achieved of 1.4%.

7.2 BWR PRA Results

PRA results for the reference BWR reactor are summarized in Table 7-3. The addition of steam line break IE frequency to the existing general transient initiator is trivial. Likewise, the additional CDF due to steam line break in HES system is less than 1%. Meanwhile the IEs related to a switchyard-induced LOOP are the same with the PWR model because such events are indifferent to the reactor types, but are a function of the geographical region in which the reactor resides in. The increase in CDF due to switchyard-related LOOP resulting from the hydrogen MCA event is negligible. The highest risk increase of 1.17% CDF is observed when the SPAR-H timing of 2 hours is assumed nominal, and when there is no dedicated ceiling ventilation system to vent the leaked hydrogen to the atmosphere.

Table 7-3. Summary of PRA results for BWR.

Risk metric	Case	Initiating Event Frequency (/y) (Δ%)	Core Damage Frequency (/y)	Cutsets
General transient frequency (steam line break is modeled within general transient for the BWR)	Nominal	7.4E-01		1

Risk metric	Case	Initiating Event Frequency (/y) (Δ%)	Core Damage Frequency (/y)	Cutsets
Steam line break IE frequency with HES system	Base assumptions	1.663E-5 **(+0.002%)**		3
Switchyard-related LOOP IE frequency	Nominal	1.34E-02		1
Switchyard-related LOOP frequency with HES system, conservative SPAR-H timing, without dedicated ceiling ventilation system	Base assumptions	1.357E-2 **(+1.3%)**		2
Switchyard-related LOOP frequency with HES system, conservative SPAR-H timing, and dedicated ceiling ventilation system	Sensitivity	1.34E-02		2
Switchyard-related LOOP frequency with HES system, realistic SPAR-H timing, and dedicated ceiling ventilation system	Sensitivity	1.34E-02		2
Switchyard-related LOOP frequency with HES system, realistic SPAR-H timing, without dedicated ceiling ventilation system	Sensitivity	1.34E-2		2
CDF due to general transient initiator	Nominal		3.886E-06	5200
CDF due to steam line break with HES system	Base assumptions		8.003E-10 **(+0.02%)**	1931
CDF due to switchyard-related LOOP	Nominal		5.787E-7	5083
CDF due to switchyard-related LOOP with HES system, conservative SPAR-H timing, without dedicated ceiling ventilation system	Base assumptions		5.855E-7 **(+1.17%)**	5133
CDF due to switchyard-related LOOP with HES system, conservative SPAR-H timing, and dedicated ceiling ventilation system	Sensitivity		5.787E-7	5083
CDF due to switchyard-related LOOP with HES system, realistic SPAR-H timing, and dedicated ceiling ventilation system	Sensitivity		5.787E-7	5083
CDF due to switchyard-related LOOP with HES system, realistic SPAR-H timing, without dedicated ceiling ventilation system	Sensitivity		5.787E-7	5083

Using the results in Table 7-3, the plant risk measures are calculated on the conservative assumption that 2 hours are the nominal time to locate and seal hydrogen leakage (a conservative SPAR-H timing) and in which the hydrogen plant does not have a dedicated ceiling ventilation system. The base design of HES system as discussed in Section 7.1 and the base assumptions listed in Table 5-1 are also selected for this analysis. The results are shown in Table 7-4.

It is found that both the total CDF and LERF increase by 1E-8 (0.03%) when a BWR NPP is coupled with a hydrogen production facility.

Table 7-4. Risk metric for BWR.

	Total CDF (per year)	Total LERF (per year)
NPP without HES	2.839E-5	2.807E-5
NPP with HES	2.840E-5	2.808E-5

The sensitivity analyses for possible configurations of a dedicated ceiling ventilation system and expansive time to isolate the hydrogen leak at the HTEF show that a minimal increase in safety is achieved of 1.3%.

7.3 Extended Sensitivity Analysis Results

Results of extended sensitivity analyses on the risk of multiple HES design options are summarized in Table 7-5. These results are obtained with a 1E-15 cutoff value set in the PRA solver settings. The steam line break IE frequency does not change much between the various HES design options. This is because the IE-SLB41 intermediate event in Figure 6-1 (above) is in the order of 1E-5 regardless of the selected design option. When this event is combined with the isolation failure event (IE-SLB-NEW1232), the resulting probability becomes an order of 1E-9. This value is insignificant compared to the event where isolation valves ruptures (IE-SLB-NEW1242) are of an order of 1E-5. With such trivial variations, the CDF due to steam line break is analyzed only for the two extreme design options (i.e., one with two isolation valves and three bypass trains and another with one isolation valve and one bypass train). Results show that there is no significant difference in CDF between these design options. The design options do not affect the IE frequency and CDF due to switchyard-centered LOOP event. For that reason, the design option with one isolation valve and one bypass train is deemed better due to having fewer components and, consequently, less cost.

Table 7-5. Sensitivity analyses for PWR.

Risk metric	Initiating Event Frequency (/y)	Core Damage Frequency (/y)	Cutsets
Steam line break IE frequency with HES system (2 isolation valves and 3 bypass trains)	3.18E-4		39
Steam line break IE frequency with HES system (2 isolation valves and 2 bypass trains)	3.18E-4		37
Steam line break IE frequency with HES system (2 isolation valves and 1 bypass trains)	3.18E-4		35
Steam line break IE frequency with HES system (1 isolation valves and 3 bypass trains)	3.18E-4		47
Steam line break IE frequency with HES system (1 isolation valves and 2 bypass trains)	3.18E-4		44
Steam line break IE frequency with HES system (1 isolation valves and 1 bypass trains)	3.18E-4		42
CDF due to steam line break, with 2 isolation valves and 3 bypass trains		2.69E-7	11228
CDF due to steam line break, with 1 isolation valves and 1 bypass train		2.69E-7	11228

Sensitivity analysis results for BWR reactor are summarized in Table 7-6. Similar to the PWR, the variations on IE frequency between the design options are trivial. The CDF due to steam line break is analyzed for the two extreme design options, just as with PWR. The change in CDF is found to be negligible. With these considerations, the HES system with one isolation valve and one bypass train may be preferred in terms of risk analysis, system complexity and costs.

Table 7-6. Sensitivity analysis for BWR

Risk metric	Initiating Event Frequency (/y)	Core Damage Frequency (/y)	Cutsets
Steam line break IE frequency with HES system (2 isolation valves and 3 bypass trains)	1.66E-5		37
Steam line break IE frequency with HES system (2 isolation valves and 2 bypass trains)	1.66E-5		35
Steam line break IE frequency with HES system (2 isolation valves and 1 bypass trains)	1.66E-5		33
Steam line break IE frequency with HES system (1 isolation valves and 3 bypass trains)	1.66E-5		46
Steam line break IE frequency with HES system (1 isolation valves and 2 bypass trains)	1.66E-5		44
Steam line break IE frequency with HES system (1 isolation valves and 1 bypass trains)	1.66E-5		41
CDF due to steam line break, with 2 isolation valves and 3 bypass trains		8.227E-10	624
CDF due to steam line break, with 1 isolation valves and 1 bypass train		8.228E-10	624

The distance of hydrogen plant to the nuclear power plant is taken as 1 km in this study, following the overpressure analysis conducted by Sandia National Laboratories [11]. The study suggested that 1 km is a safe separation distance based on a set of conservative assumptions. An additional sensitivity study is conducted to analyze the effect of separation distance to the fragility of transmission towers, which may affect the switchyard-induced LOOP frequency. Figure 7-1 shows the overpressure and total fragility curves of switchyard components as a function of separation distance between the hydrogen and the switchyard. The distance around 845 meters marks the critical fragility for switchyard components, below which their fragility is 1.

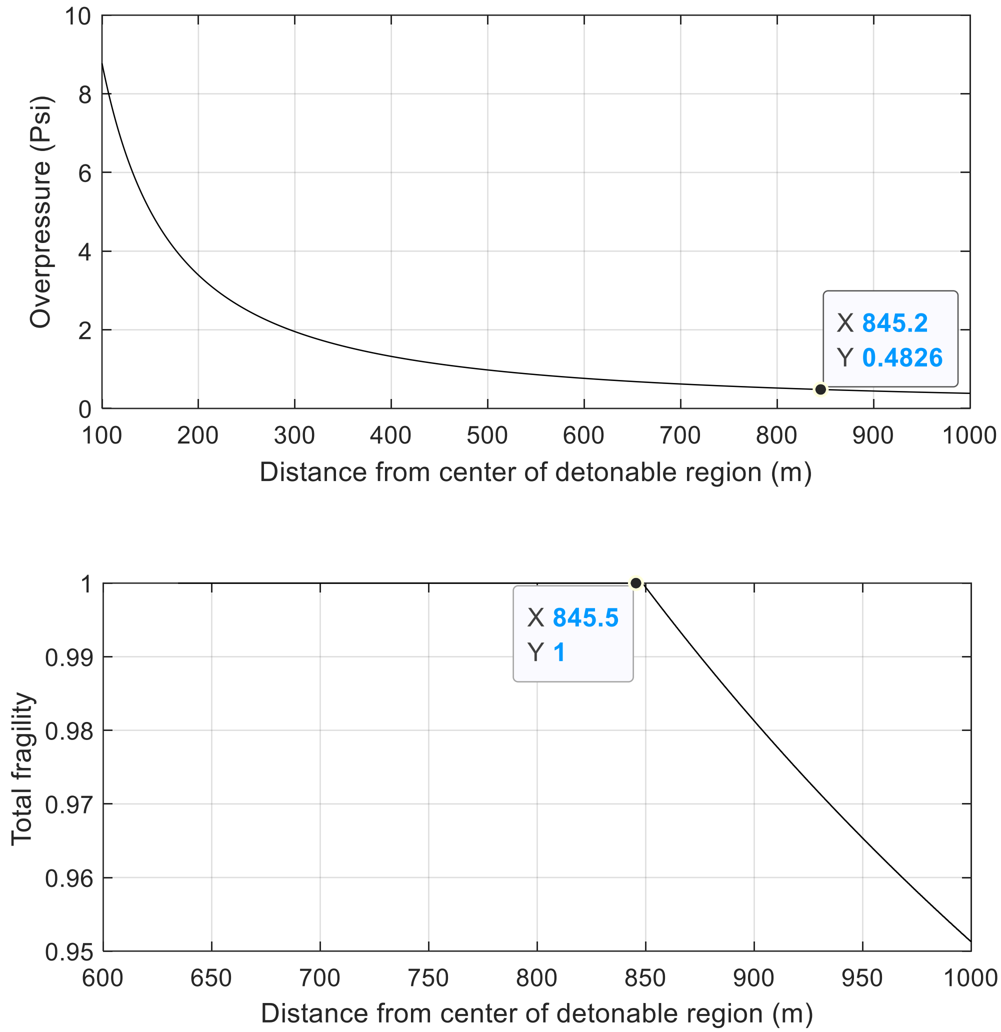

Figure 7-1. MCA overpressure (top) and total switchyard fragility (bottom) as a function of separation distance between the hydrogen and the nuclear plant.

The hydrogen detonation event considered in Section 6 is the cloud detonation event, in which leaked hydrogen accumulates indoors for 2 hours before it finally ignites and detonates. There is another possibility of ignition immediately following leakage, which creates a high-pressure hydrogen jet detonation event. This event was excluded from the PRA model on the basis that it cannot create a significant overpressure to damage a transmission tower 1 km away as shown in Figure 5-8 (above). However, if the separation distance is reduced, the overpressure from the high-pressure hydrogen jet may damage the transmission tower and create a LOOP event. For that reason, a sensitivity analysis is conducted to find the minimum safe distance. The LOOP initiator fault tree in Figure 6-23 (above) is modified to include the high-pressure jet event as shown in Figure 7-2. The switchyard failure probability due to jet detonation (IE-LOOPSC-SC-JET-F) is initially set to 0 at a separation distance of 1 km. If a 15% increase in IE frequency is set as the safety limit with considerations discussed in Section 8.1, the IE-LOOPSC-SC-JET-F event should have a probability of 0.11. Meanwhile, if a 5% increase in IE frequency is used such that the change in IE-LOOPSC-HES frequency is comparable to the increase in Steam Line break frequency, the probability for IE-LOOPSC-SC-JET-F event is 0.037.

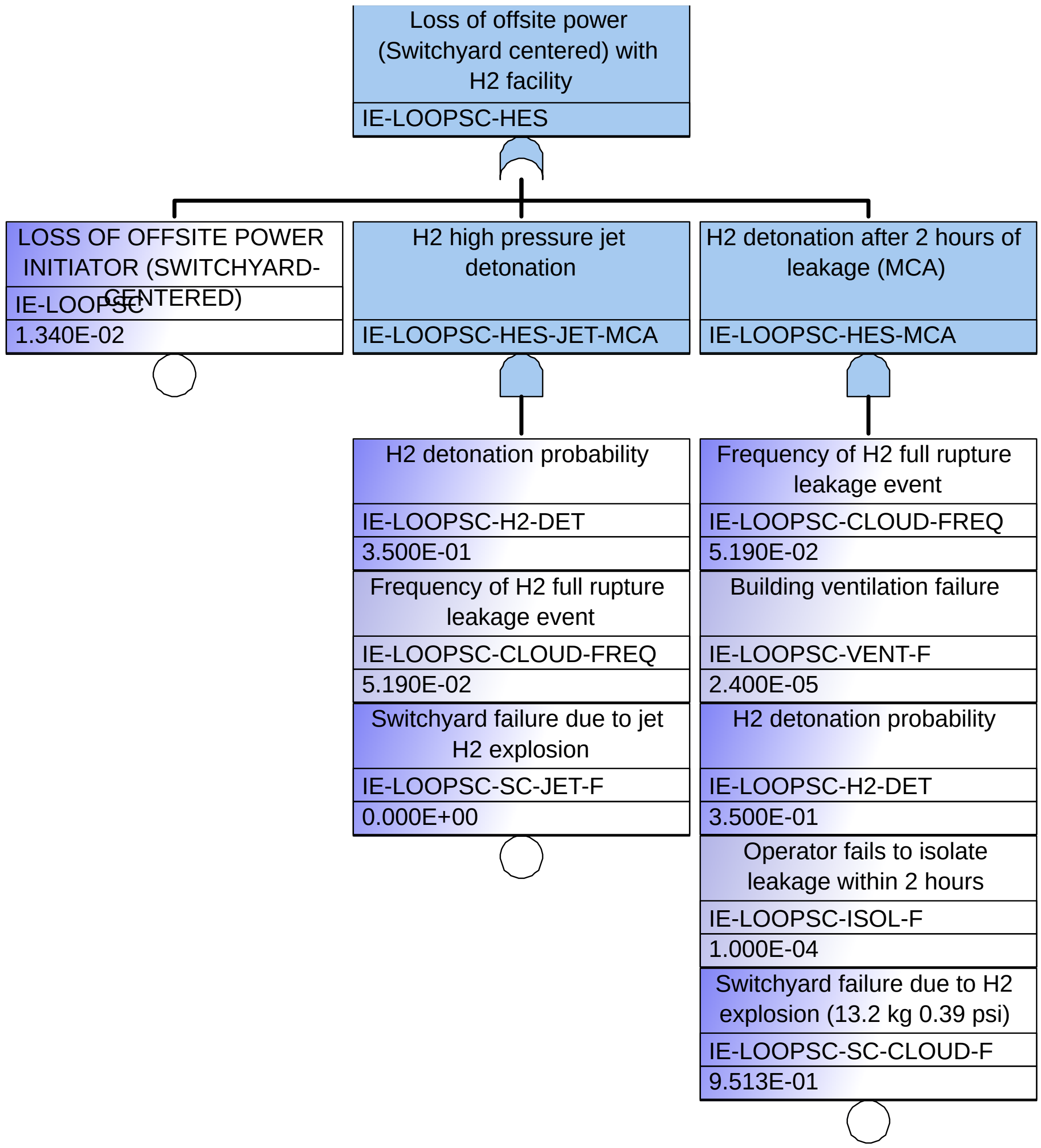

Figure 7-2. LOOP fault tree with high-pressure hydrogen jet event.

A reference study [11] has assessed various hydrogen jet detonation scenarios and identified the most conservative scenario of a 200 mm break with a temperature of 50° C and pressure of 7 MPa. By combining data from this reference and Figure 5-8 (above), a graph of transmission tower fragility versus the separation distance between the hydrogen plant and transmission towers is plotted in Figure 7-3. The data points for IE-LOOPSC-SC-JET-F to fulfill the 5% and 15% IE increase are highlighted on the plot. The figure suggests that a minimum separation distance lies at around the 450-meter mark to meet the safety criteria explained in the previous paragraph. When the transmission tower is spaced at least 500 meters away from the hydrogen plant, the LOOP risk due to high-pressure hydrogen jet detonation is nullified.

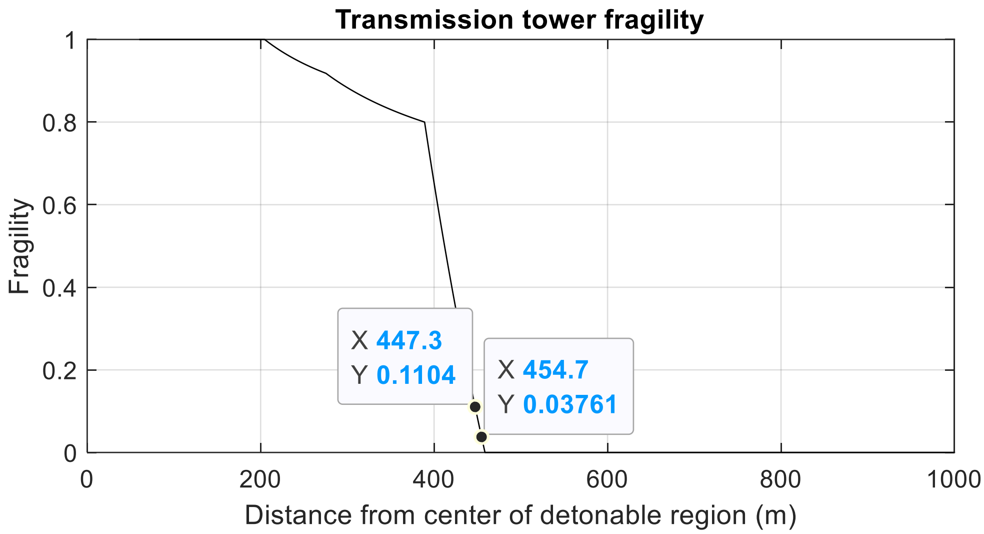

Figure 7-3. Fragility curve of transmission tower.

8. LICENSING PATHWAY SUPPORT FROM PRA

The NRC develops various regulatory guides to assist license applicants' implementation of NRC regulations by providing evaluation techniques and data used by the NRC staff. Two distinct pathways through guides and codes of federal regulations (CFR) are identified for use in the proposed LWR plant configuration change approval.

One pathway utilizes 10 CFR 50.59 [3], to review the effects on frequencies of design basis accidents (DBAs), amendment of the updated final safety analysis report (UFSAR), and determination of whether a licensing amendment review (LAR) is required. This pathway is dependent on the IE frequency, which is on the front end of the PRA.

A supporting pathway utilizes RG 1.174 [4] through use of risk-informed metrics to approve a plant configuration change based on the effect on the overall CDF of an approved PRA. This pathway is dependent on the tail end, the CDF-resulting metric of the PRA.

The final pathway is the LAR process, which would utilize PRA results as well; however, the process utilizes 10 CFR 50.90, "Application for amendment of license or construction permit at request of holder" [16] and should be avoided if possible due to lengthy review and monetary burden.

8.1 Licensing Process Through 10 CFR 50.59

The pathway that utilizes an evaluation of the change in DBA frequencies first uses 10 CFR 50.59 [3] to determine if an LAR would be required via 10 CFR 50.90 [16]. Changes that meet the requirements of 10 CFR 50.59 do not require additional NRC review and approval. In a study commissioned by the LWRS: Swindlehurst, "Safety Evaluation of Modification for Process Steam Supply Capability in Pressurized Water Reactors, Rev. 1" [17] the effects on DBAs of a PWR with the addition of an HES were evaluated for adherence to the following eight criteria:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated)

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the final safety analysis report (as updated)

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated)

4. Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report (as updated)

5. Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report (as updated)

6. Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated)

7. Result in a design basis limit for a fission product barrier as described in the Final Safety Analysis Report (FSAR) (as updated) being exceeded or altered

8. Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.

If the above criteria are not met, the 10 CFR 50.59 process cannot be used to implement the plant modification and an LAR must be submitted to the NRC for review and approval.

As noted in [17], nearly all criteria are readily met for a modification such as the HES, but there was not enough data available at the time to determine if item 1 (minimal increase in DBA frequency) is met when considering a minimal increase is traditionally understood to be ≤15%. This PRA found the largest increase in a DBA yearly IE frequency to be 5.6% (Large Steam Line Break for the PWR), thus meeting the criteria for 10 CFR 50.59.

8.2 Licensing Support Through RG 1.174

RG 1.174 [4] provides general guidance concerning analysis of the risk associated with proposed changes in plant design and operation. Specifically, thresholds and guidelines are provided for comparison with Level 1 PRA results for CDF and LERF.

As seen in Figure 8-1, CDF should be below 1E-5 overall and the change in overall CDF should be below a magnitude of 1E-5. Any plant which starts at a 1E-4 or more CDF requires less than 1E-6 increase in CDF to be considered. If these metrics are met, the NRC most likely considers this a small change which is consistent with the intent of the Commission's Safety Goal Policy Statement and a detailed quantitative assessment of the base values of CDF is not necessary for the license review.

If the above criteria for CDF are not met, then an LAR must be submitted to the NRC for review and approval.

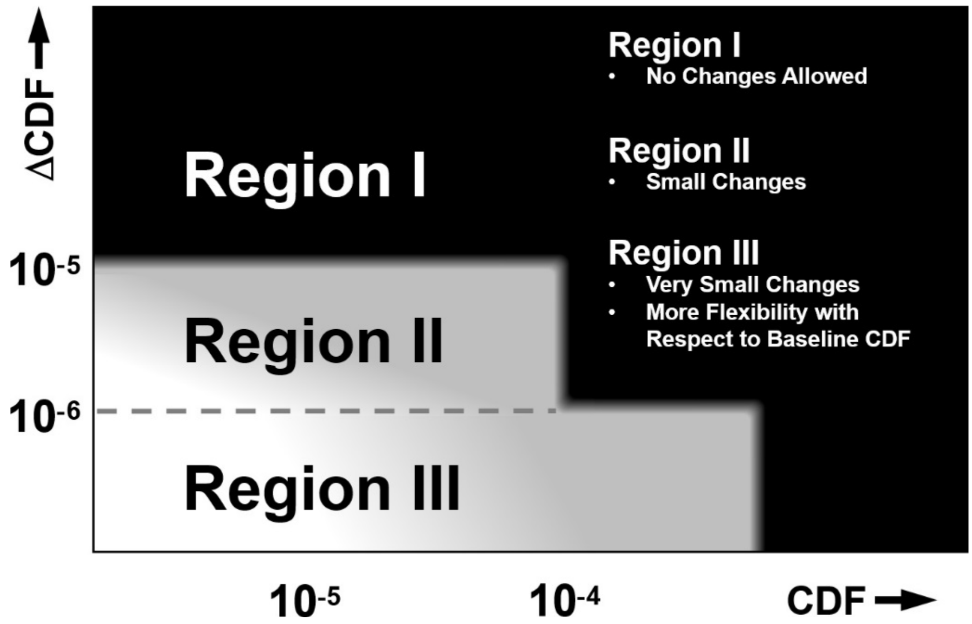

Figure 8-1: Acceptance guidelines for CDF.

As seen in Figure 8-2, LERF should be below 1E-6 overall and the change in overall LERF should be below a magnitude of 1E-6. If these metrics are met, the NRC most likely considers this a small change which is consistent with the intent of the Commission's Safety Goal Policy Statement and a detailed quantitative assessment of the base values of CDF is not necessary for the license review.

If the above criteria for LERF are not met, then an LAR must be submitted to the NRC for review and approval.

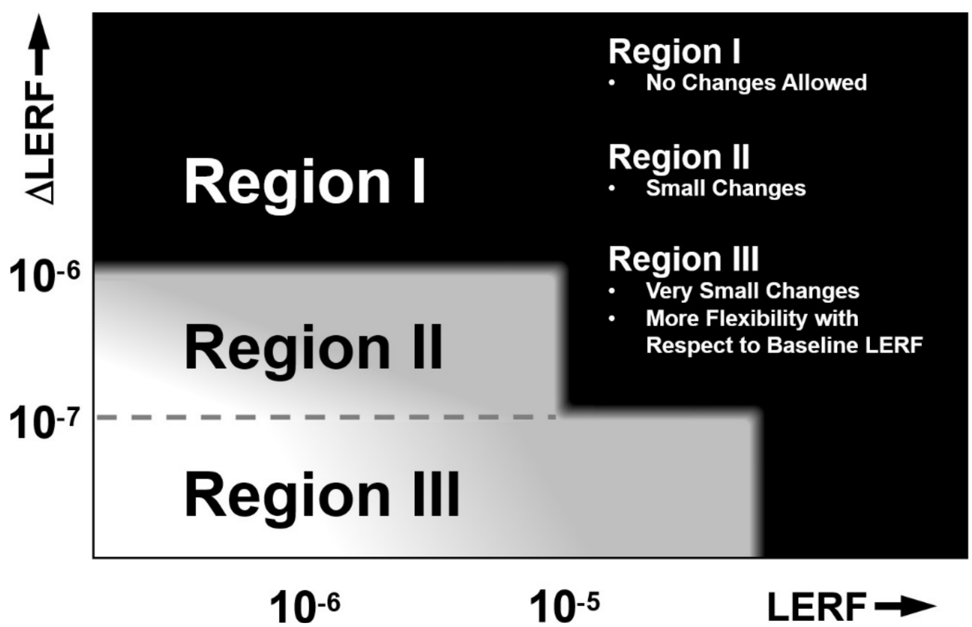

Figure 8-2: Acceptance guidelines for large early release frequency.

As noted in Table 7-2 (above), the generic PWR being considered for this study has a nominal CDF of 8.34E-06 /y and the increase after addition of the HES and HTEF is to 8.88E-06 /y for ΔCDF of 5.47E-07 /y, which is well within Region III of the acceptance guidelines shown in Figure 8-1. The nominal LERF frequency for the PWR is 8.04E-07 /y and the increase after addition of the HES and HTEF is to 8.88E-06 /y for ΔLERF of 6.00E-010 /y, which is well within Region III of the acceptance guidelines shown in Figure 8-2.

As noted in Table 7-4 (above), the generic BWR being considered for this study has a nominal CDF of 2.839E-05 /y and the increase after addition of the HES and HTEF is to 2.840E-05 /y for ΔCDF of 1.000E-07 /y, which is well within Region III of the acceptance guidelines shown in Figure 8-1. The nominal LERF frequency for the BWR is 2.807E-05 /y and the increase after addition of the HES and HTEF is to 2.808 E-05 /y for ΔLERF of 1.00E-08 /y, which is well within Region III of the acceptance guidelines shown in Figure 8-2.

8.3 Licensing Amendment Review Process

Should the prior two processes fail to approve a change in the LWR, the last resort would be a detailed request for an LAR. As stated in [17], "10 CFR 50.90 is the governing regulation for the process undertaken by the licensee to develop and submit an LAR. This regulation states that the application fully describes the changes desired and is to follow the form prescribed for the original UFSAR submittal. An LAR is required when a change to the technical specifications is desired for whatever purpose. The LAR is developed by the licensee staff and is reviewed by internal committees and management to ensure that the technical content is correct and meets management approval".

The NRC review of the LAR is extensive and typically involves meetings with the licensee and the opportunity for public meetings per 10 CFR 50.91, "Notice for Public Comment; State Consultation" [18]. The NRC issues requests for additional information to obtain responses from the licensee as a result of the NRC review. 19 CFR 50.92, "Issuance of Amendment" [19] includes a no significant hazards consideration to determine if any of the following conditions exist based on the NRC review of the LAR:

- Involve a significant increase in the probability or consequences of an accident previously evaluated
- Create the possibility of a new of different kind of accident from any accident previously evaluated
- Involve a significant reduction in margin of safety.

Provided these regulatory requirements are met the NRC issues, a safety evaluation that approves the LAR including the technical specification revisions."

9. CONCLUSIONS

Two generic PRAs for the addition of an HES addition to an LWR are performed, one for a PWR and one for a BWR. The results investigate the applicability of the potential licensing approaches which do not require a full NRC licensing review. The PRAs are generic, and some assumptions are made (Table 5-1). Many conservative assumptions from the preliminary PWR PRA report [2] were eliminated through the use of design data for both the HES and the HTEF. The results of the PRA indicate that the 10 CFR 50.59 licensing approach is justified due to the minimal increase in IE frequencies for all DBAs, none exceeding 5.6%. The PRA results for CDF and LERF support the use of RG 1.174 as further risk information that supports a change without a full LAR.

This PRA investigation outlines a successful pathway to follow when moving to the site-specific case.

The hazard analysis performed to support the PRAs in this report provides insights that built the nominal case of safety and some economic and non-safety hazards:

- The HES should be placed in its own building for protection of the turbine building SSCs and possible safety buses should there be a large steam line rupture.
- The high-pressure jet detonation hazard at the HTEF can be screened out as a hazard based on the low overpressures experienced at 1 km.
    - The high-pressure jet detonation hazard was the bounding hazard for safely decreasing the distance between the HTEF and the NPP (switchyard transmission towers). This distance was found to be 455 meters.

Sensitivity studies performed on the nominal case provided the following insights:

- The addition of a dedicated ceiling ventilation at the HTEF and using a less conservative time allotment to isolate the hydrogen leak added approximately 1.3% to the safety margin for the LWR licensing case, however the licensing case is strong without these additions.
- The mitigation of the larger hydrogen cloud detonations with a dedicated ceiling ventilation in the HTEF makes the placement of the HTEF viable at much closer ranges than 1 km.
- The safety case for using one isolation valve in the HES, rather than mimicking the NPP's MSIV paired configuration is a valid one, with negligible increase in the CDF.
- The safety case for using one bypass train, rather than three in the HES is a valid one, with negligible increase in the CDF.

This report confirms with high confidence the safety case for licensing an HES addition and an HTEF sited at 1.0 km from the NPP is strong and that the placement of a HTEF at 0.5 km is a viable case. Site-specific information can alter these conclusions (e.g., using blast barriers and other modifications).

10. REFERENCES

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[18] 10 CFR 50.91, "Notice for Public Comment; State Consultation", U.S. Nuclear Regulatory Commission, https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0091.html, published August 29, 2017, accessed October 8, 2020.

[19] 10 CFR 50.92, "Issuance of Amendment", U.S. Nuclear Regulatory Commission, https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0092.html, published August 29, 2017, accessed October 8, 2020.

Appendix A: Generic PWR PRA Model

This Appendix shows PWR Event Trees which are transfers of the accident mitigation Event Trees described in the body of this report.

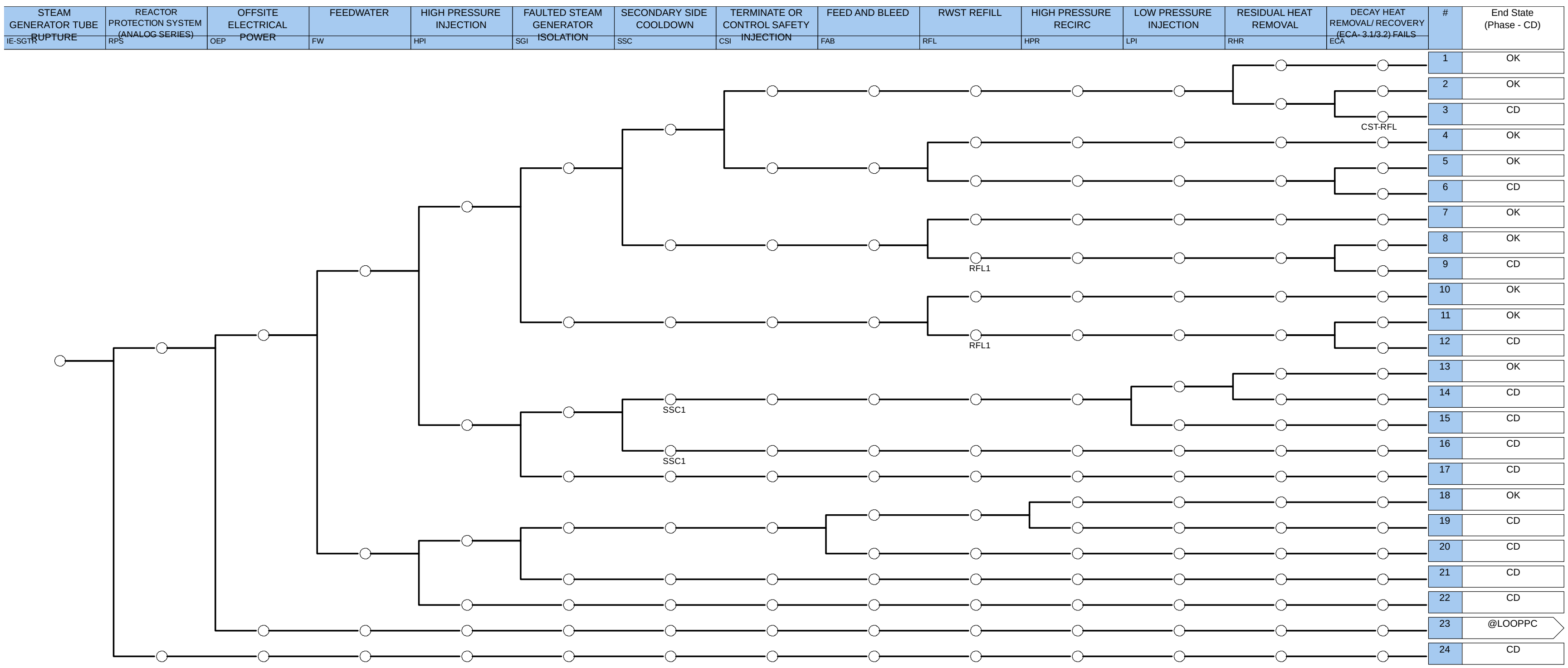

Figure A-1. Steam Generator Tube Rupture Event Tree (SGTR)

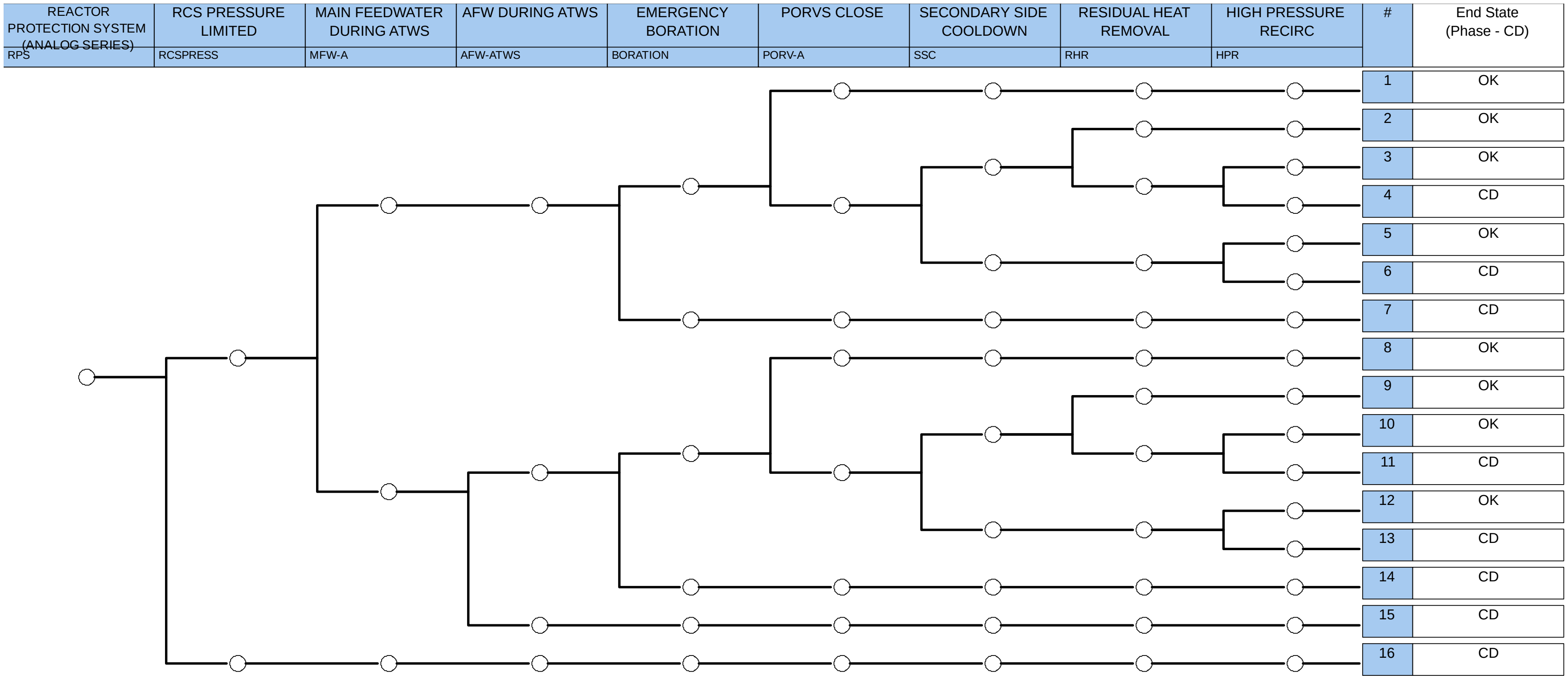

Figure A-2. Anticipated Transient Without Scram Event Tree (ATWS)

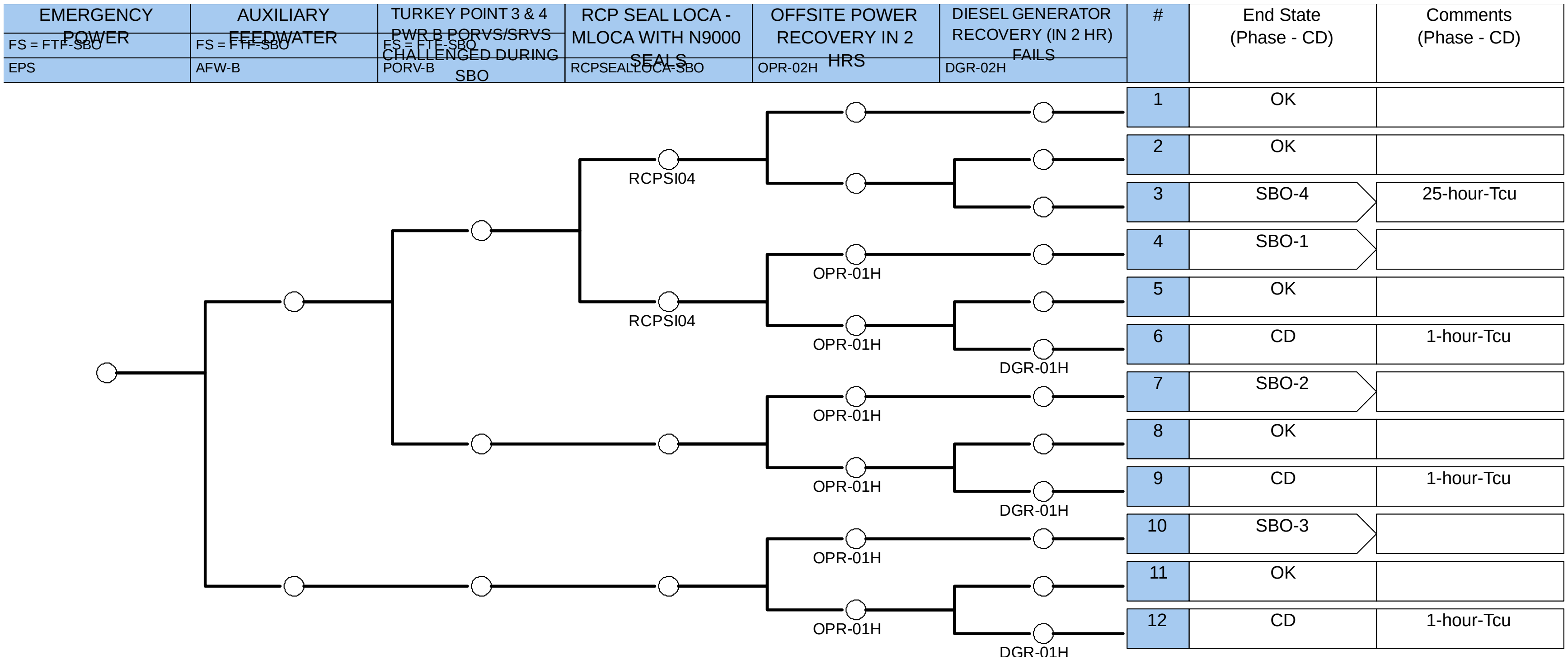

Figure A-3. Station Blackout Event Tree (SBO).

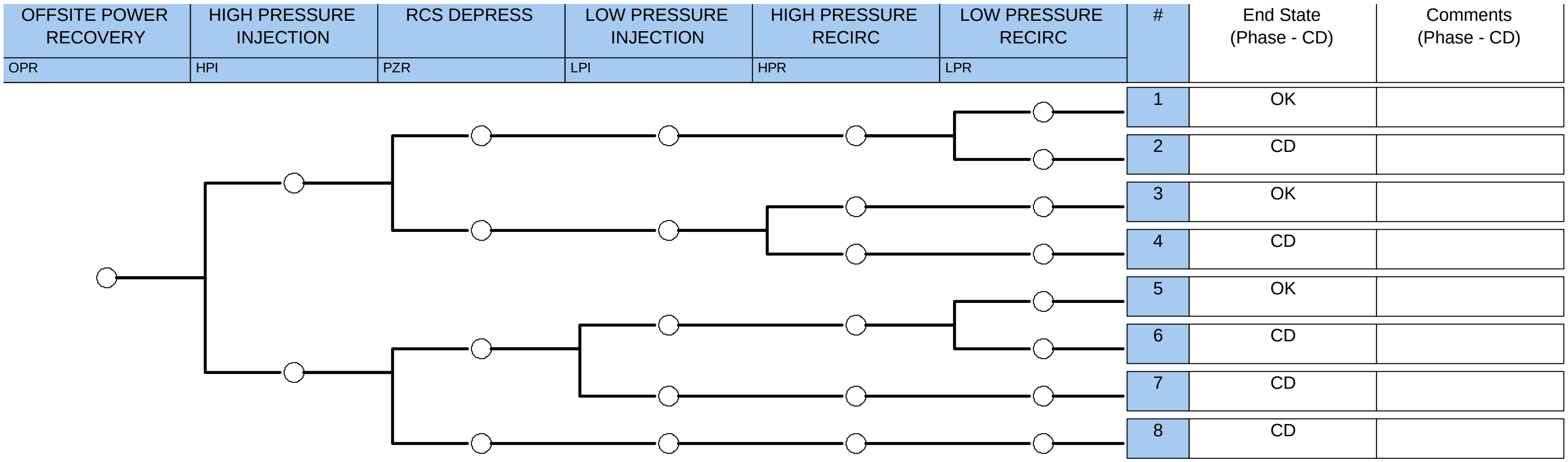

Figure A-4. Station Blackout-1 Event Tree (SBO-1).

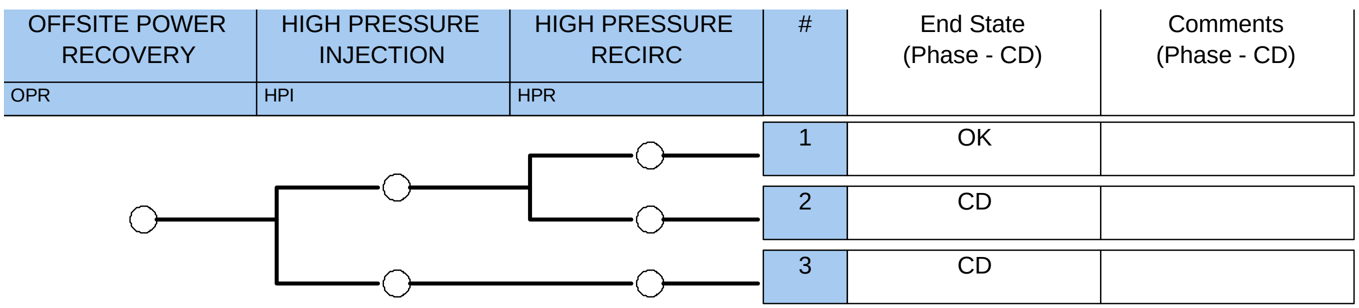

Figure A-5. Station Blackout-2 Event Tree (SBO-2).

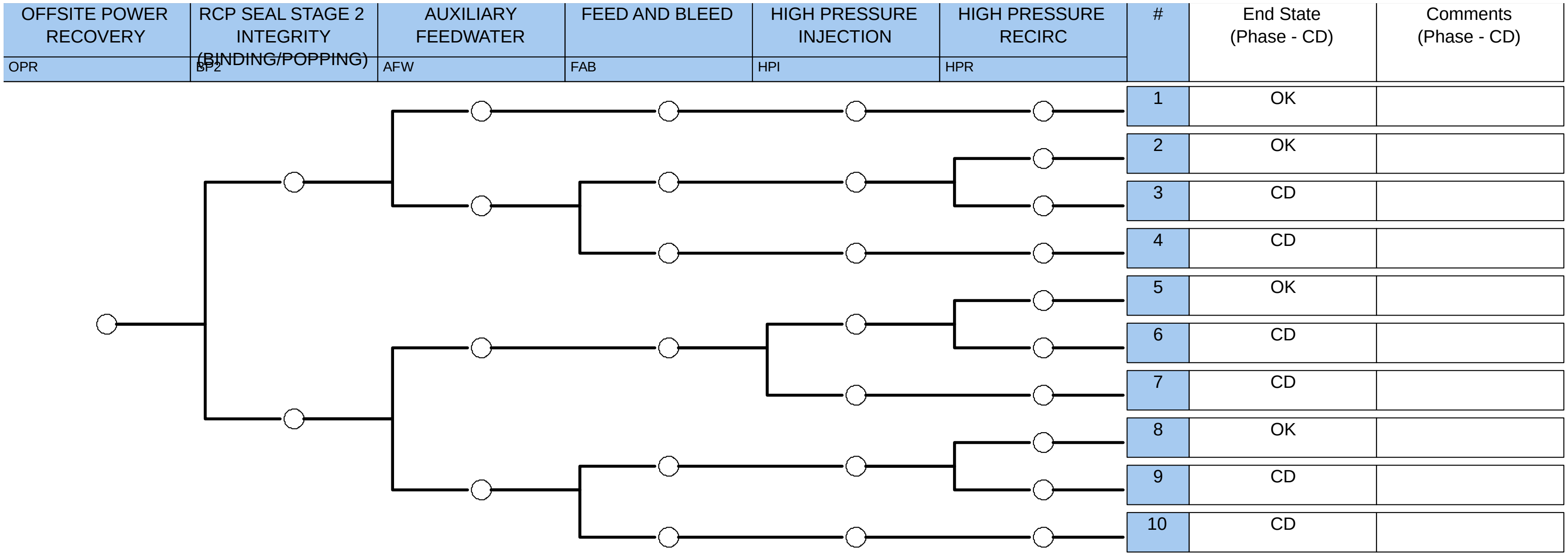

Figure A-6. Station Blackout-3 Event Tree (SBO-3).

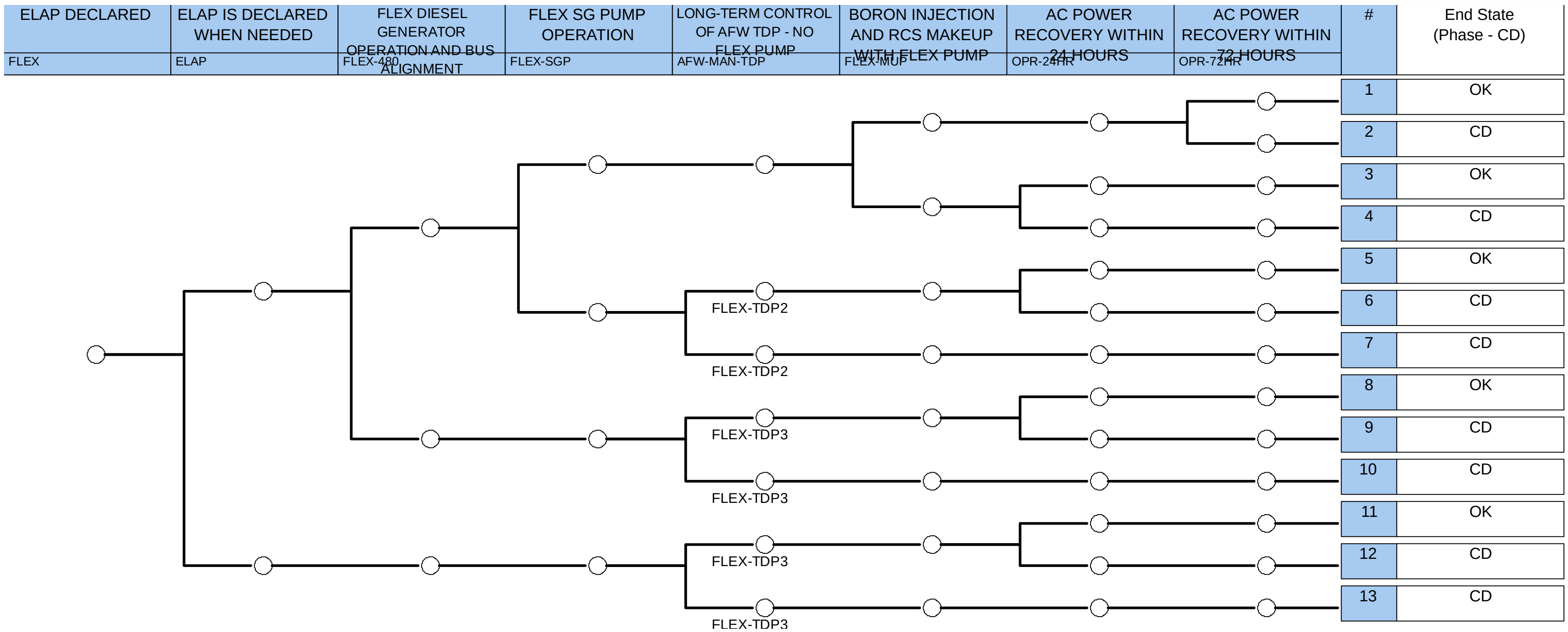

Figure A-7. Station Blackout-4 Event Tree (SBO-4).

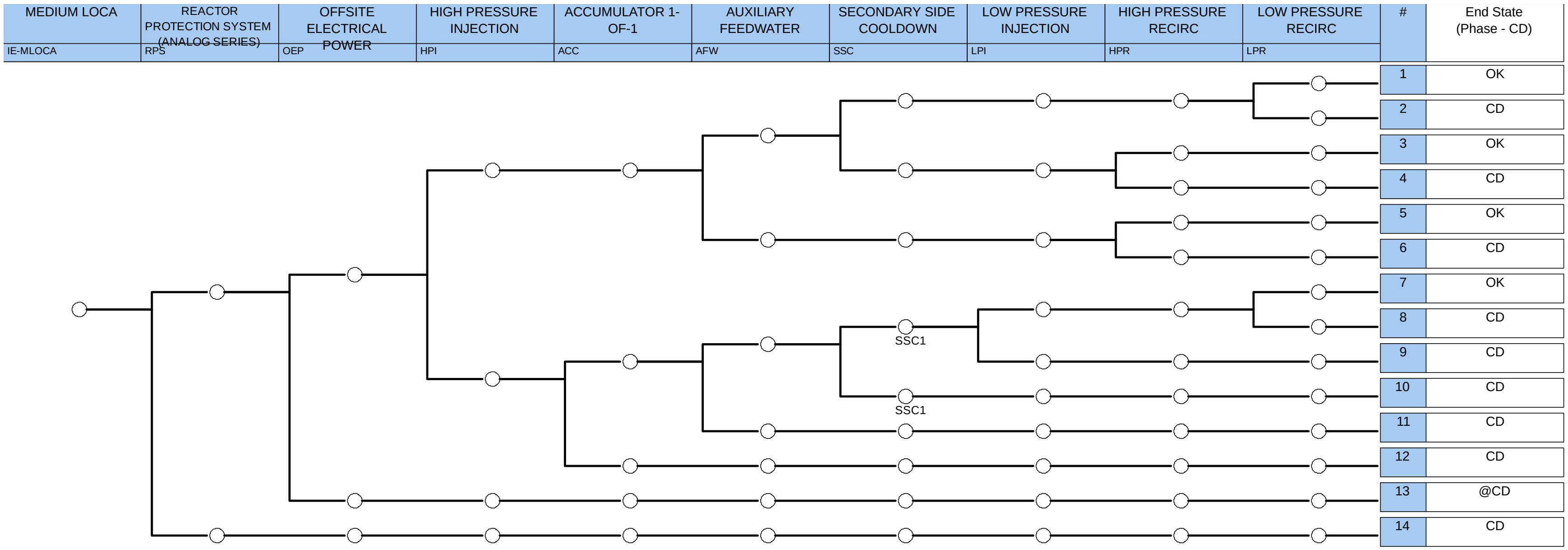

Figure A-8. Medium Loss of Coolant Accident Event Tree (MLOCA).

Appendix B: Generic BWR PRA Model

This Appendix shows BWR Event Trees which are transfers of the accident mitigation Event Trees described in the body of this report. The General plant transient event tree previously shown in Section 6.3 is truncated and displayed in several parts here for a better readability. The one stuck-open relief valve event tree is likewise shown in multiple parts for the same reason.

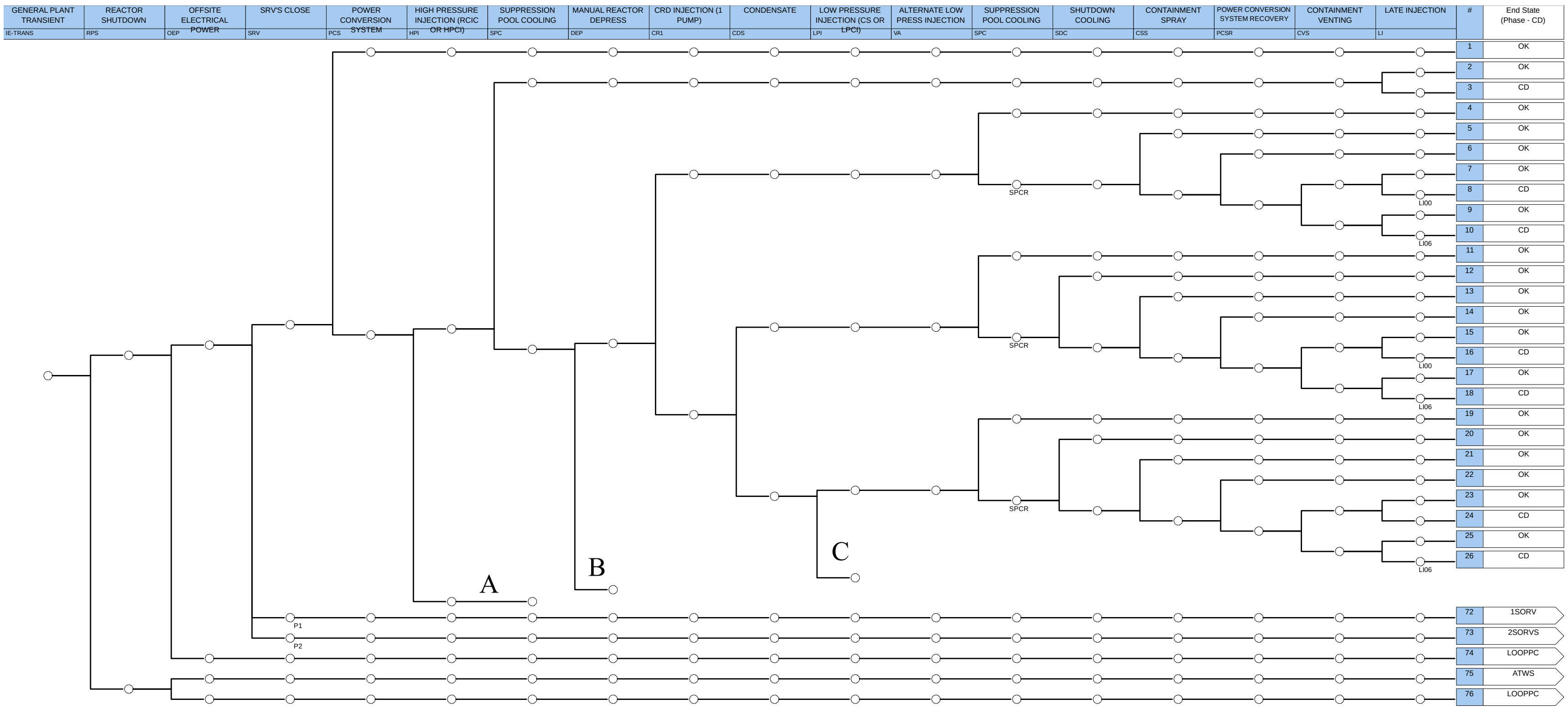

Figure B-1. General plant transient event tree (IE-TRANS) part 1 showing three truncated branches (i.e., branch A, B, and C).

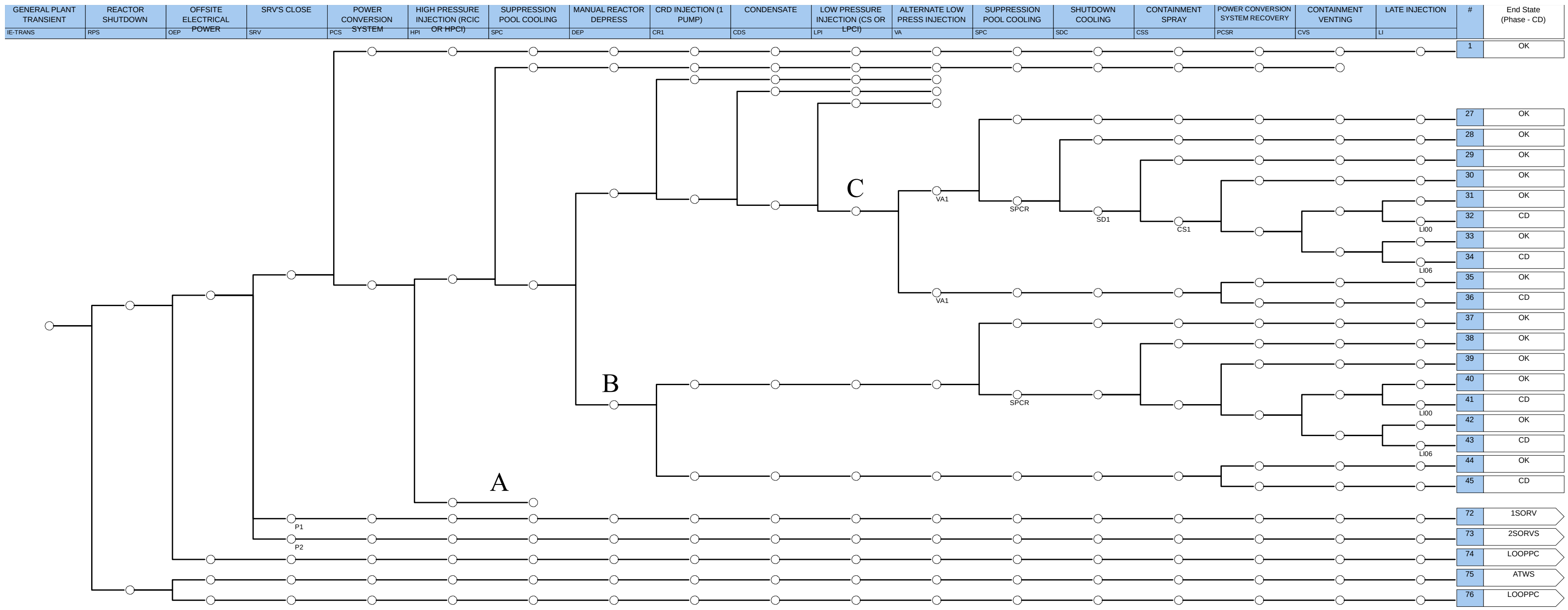

Figure B-2. General plant transient event tree (IE-TRANS) part 2 revealing branch B and C.

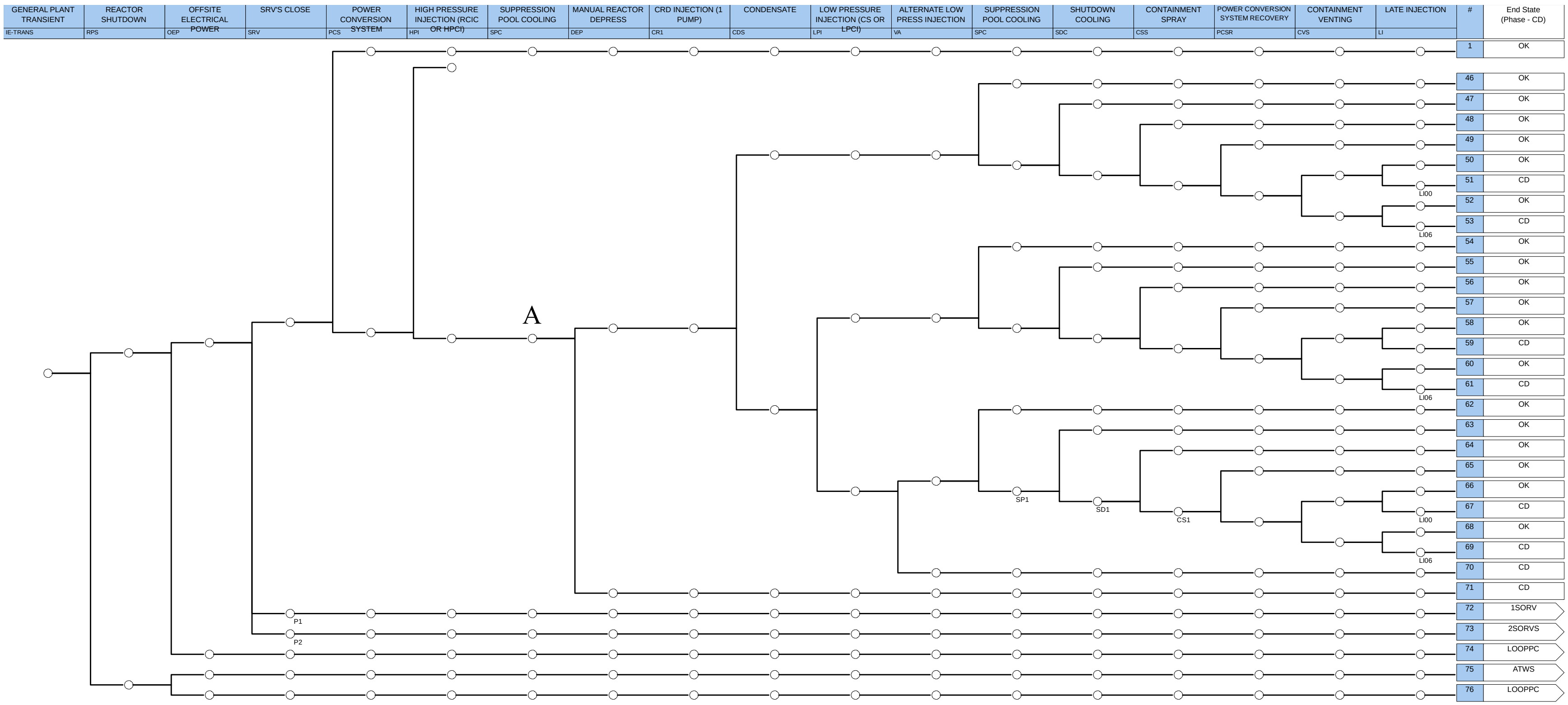

Figure B-3. General plant transient event tree (IE-TRANS) part 3 revealing branch A.

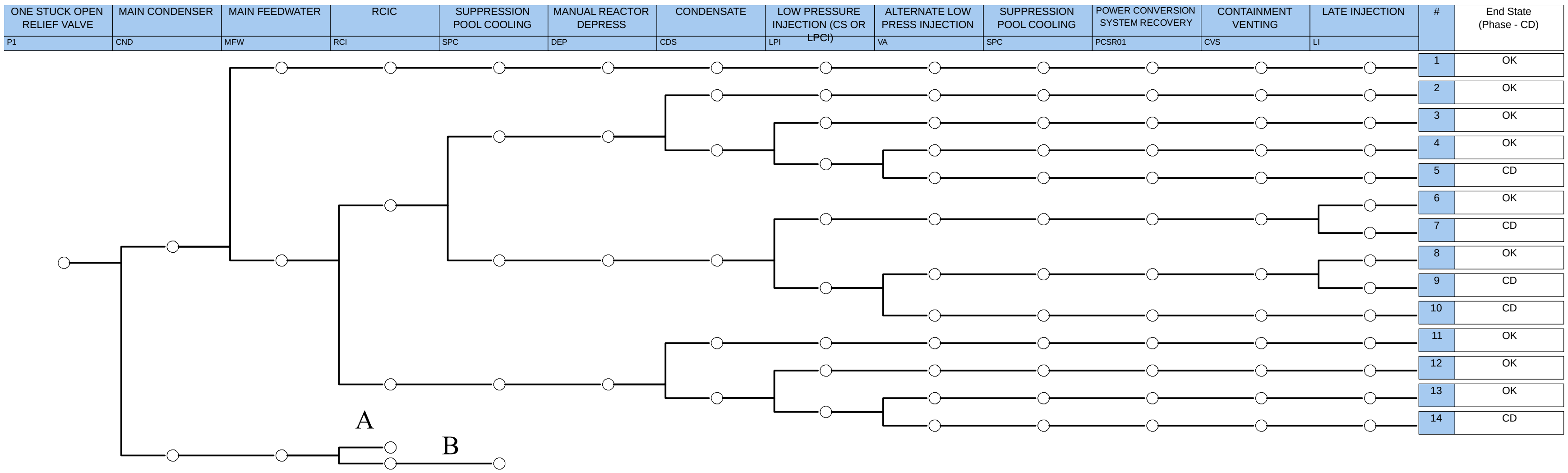

Figure B-4. One stuck-open relief valve event tree (P1) part 1 showing a truncated branch.

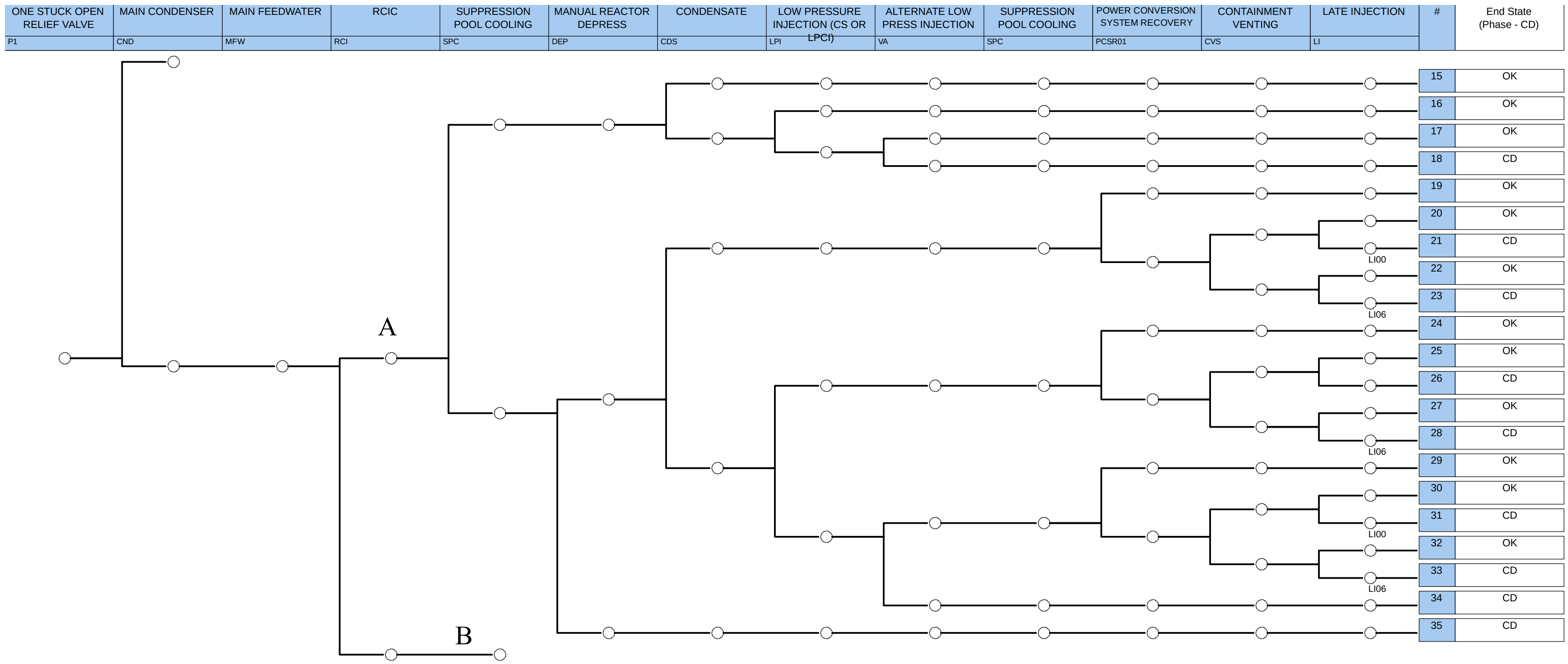

Figure B-5. One stuck-open relief valve event tree (P1) part 2 revealing branch A.

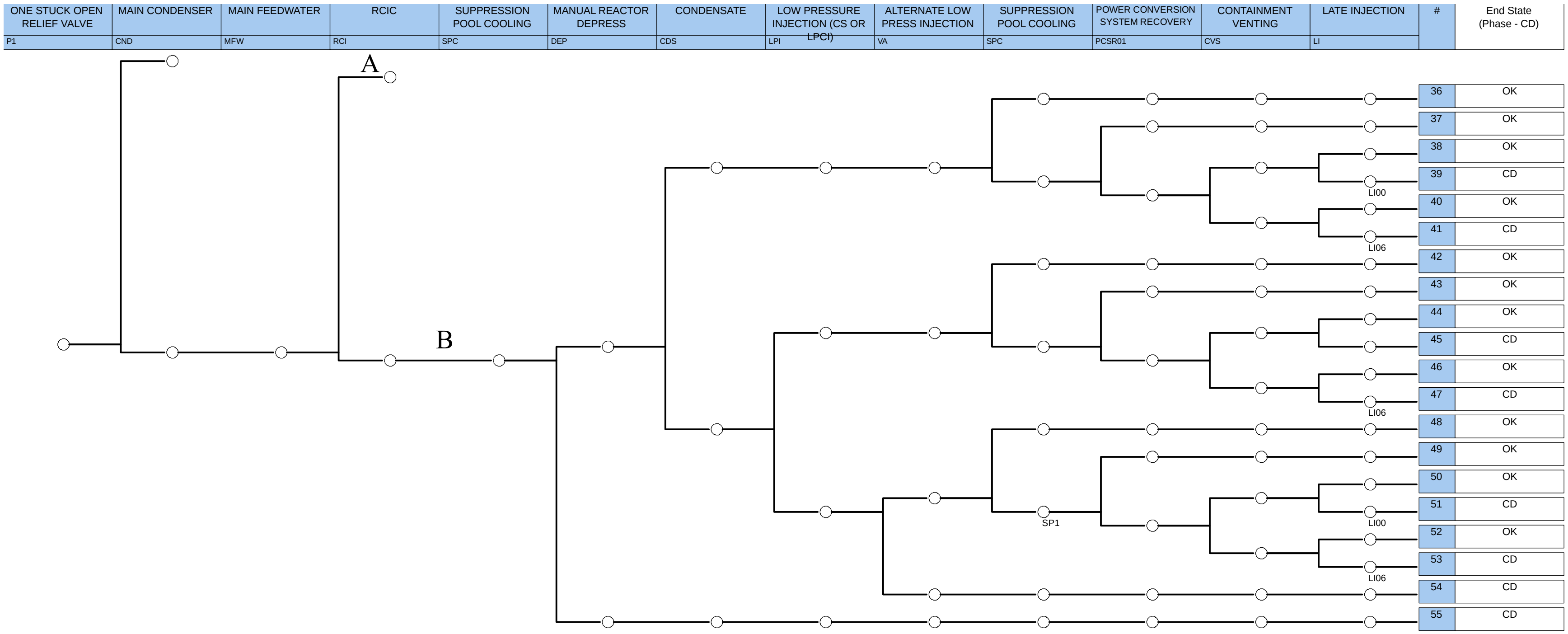

Figure B-6. One stuck-open relief valve event tree (P1) part 3 revealing branch B.

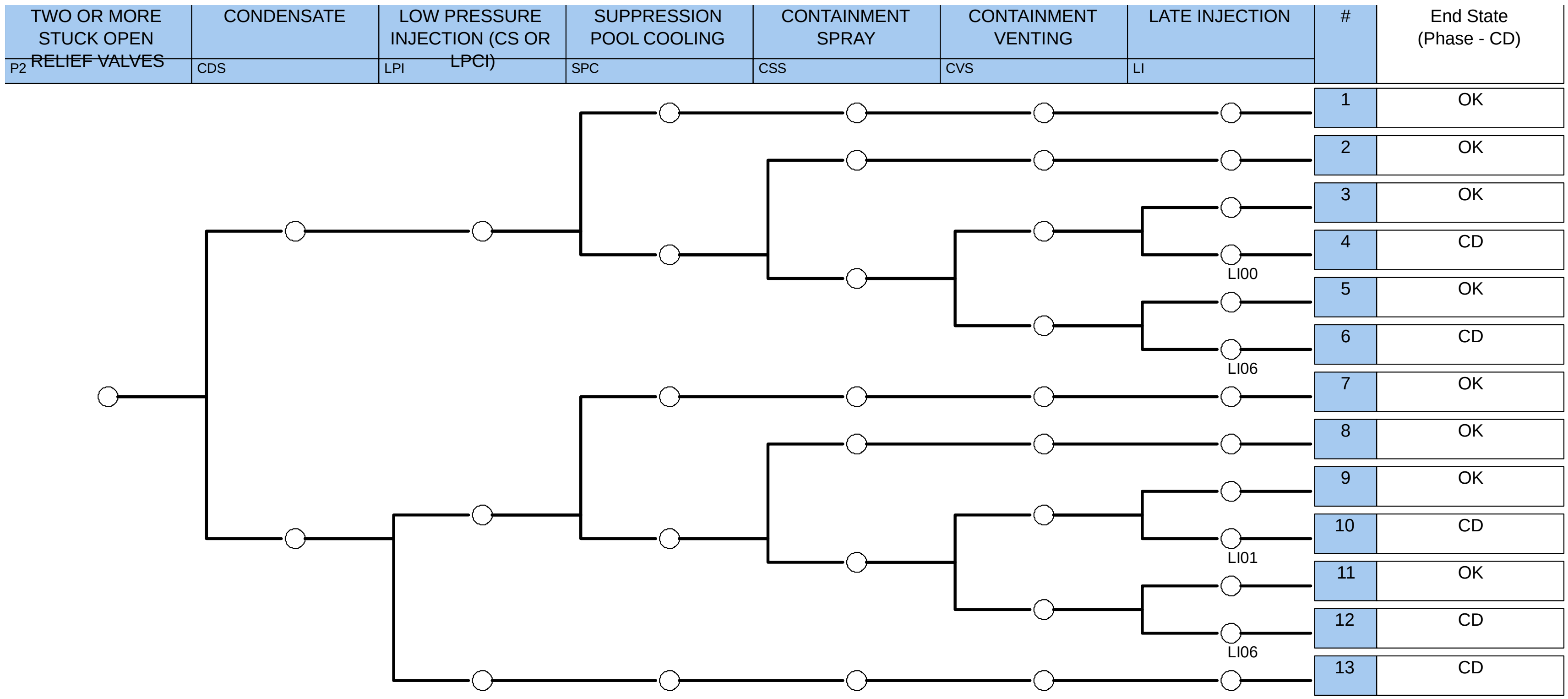

Figure B-7. Two or more stuck-open relief valves (P2).

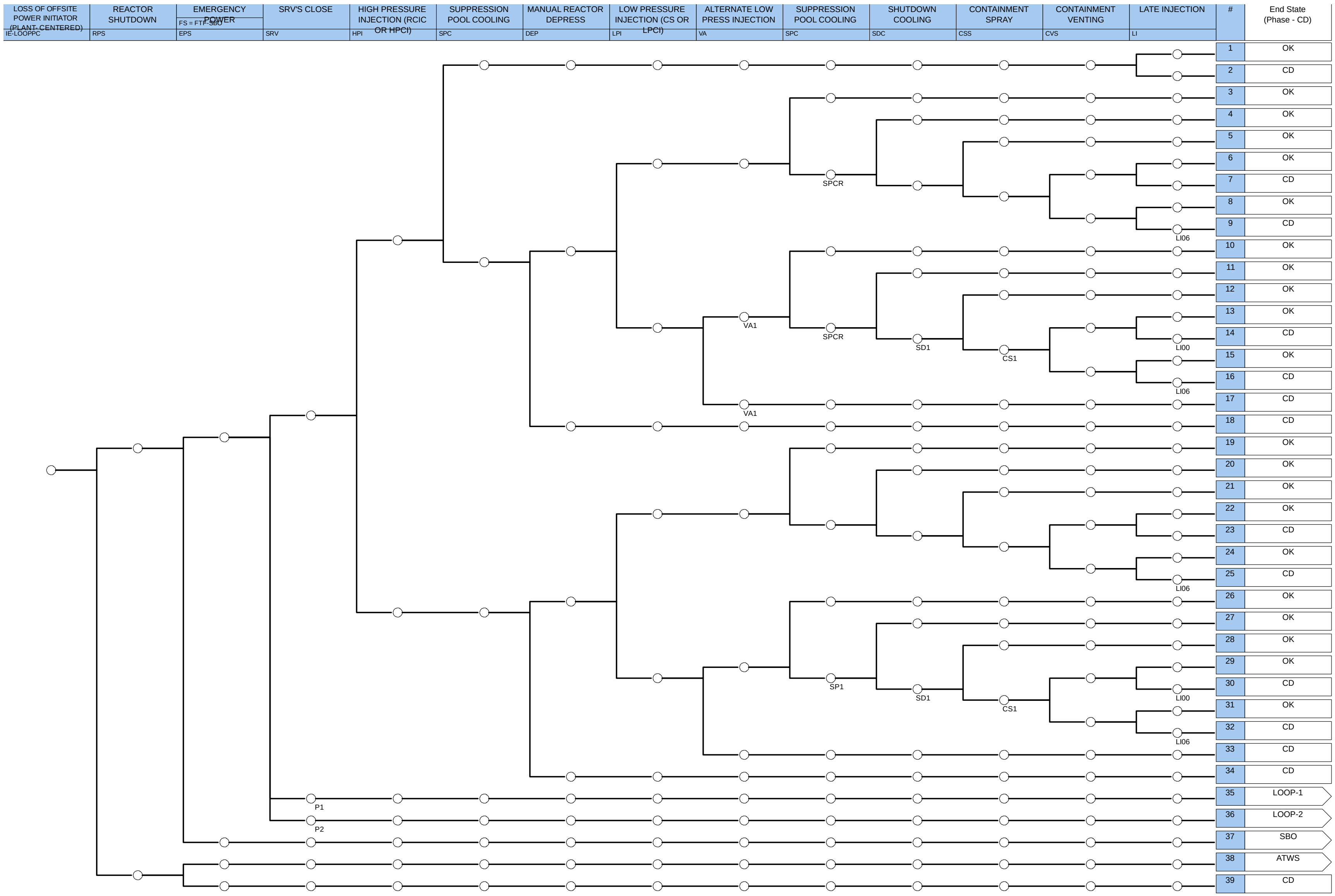

Figure B-8. Loss of offsite power (plant-centered) event tree (IE-LOOPPC).

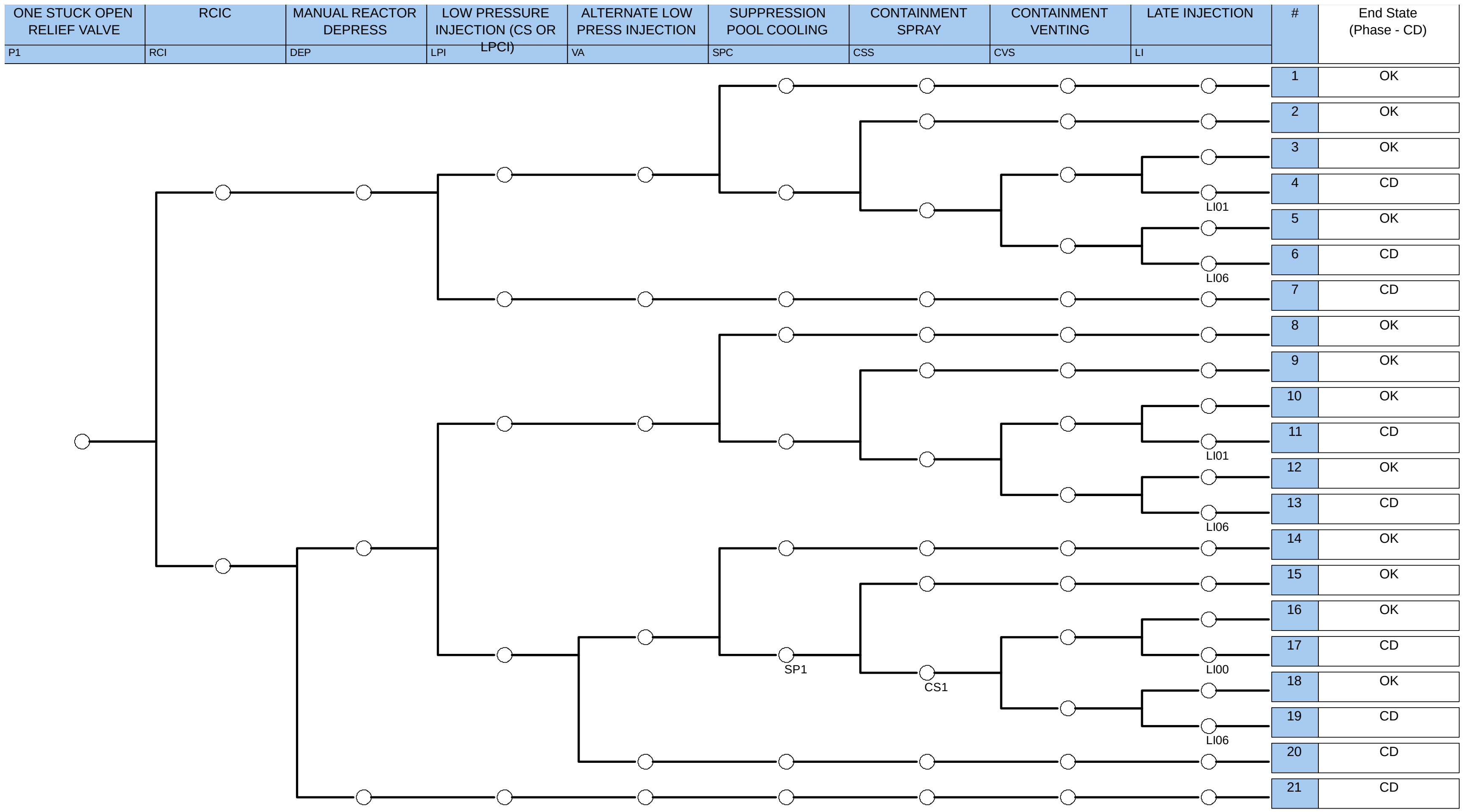

Figure B-9. LOOP-1 event tree (P1).

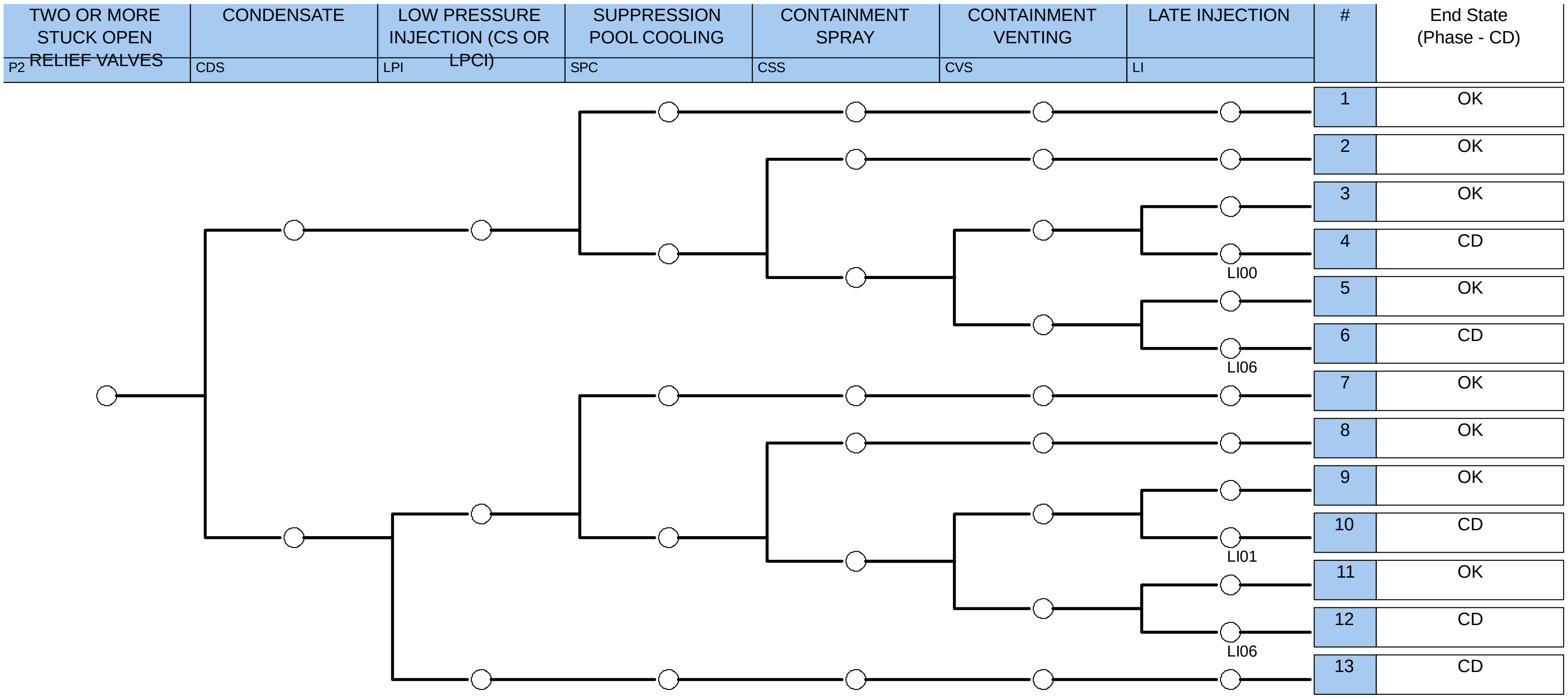

Figure B-10. LOOP-2 event tree (P2).

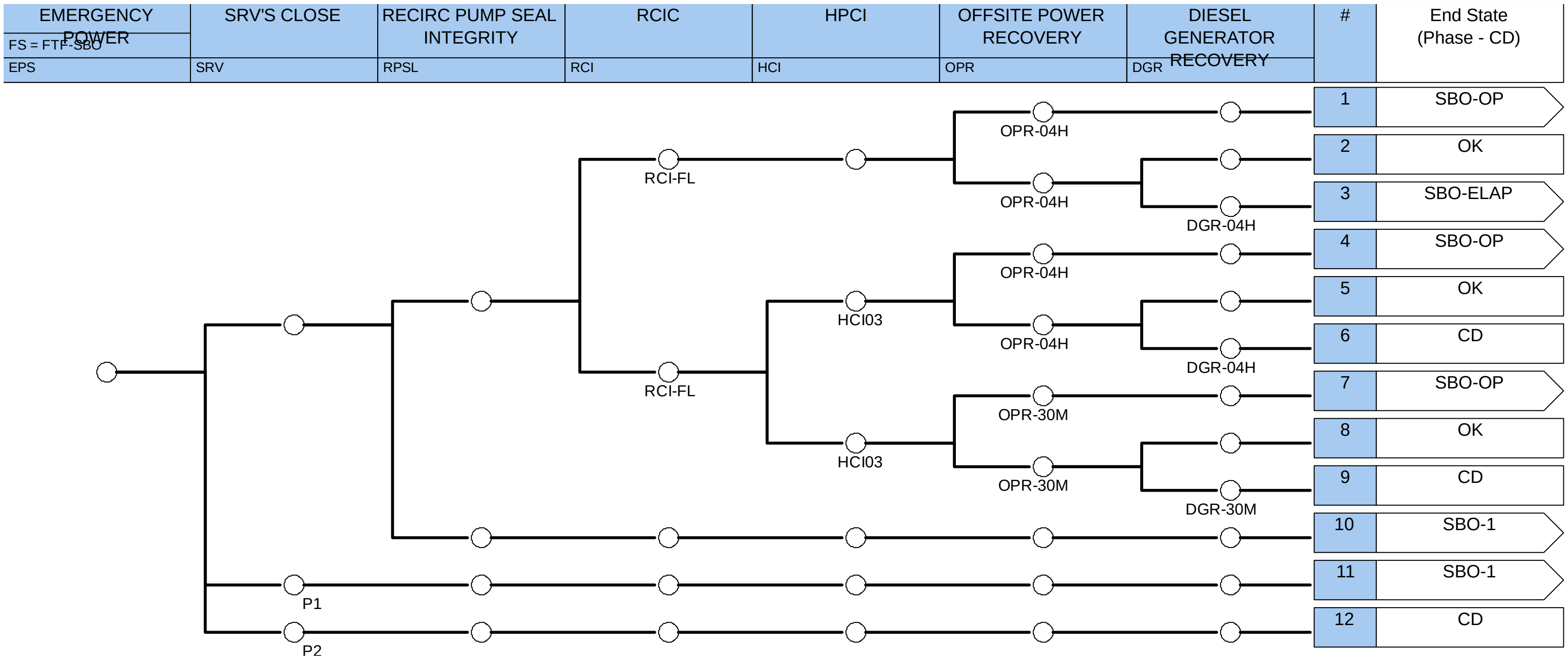

Figure B-11. Station blackout event tree (SBO).

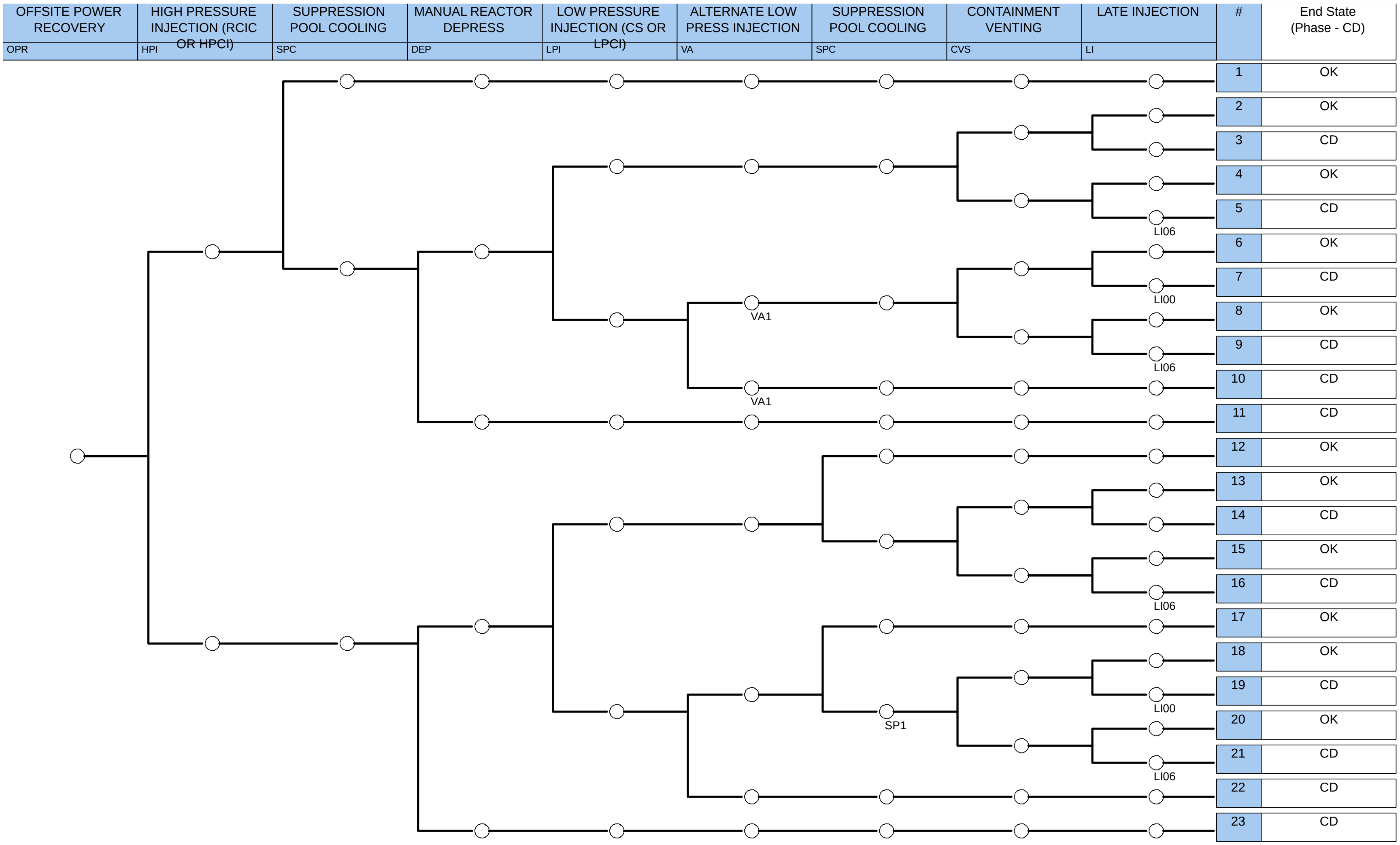

Figure B-12. SBO-OP event tree (SBO-OP).

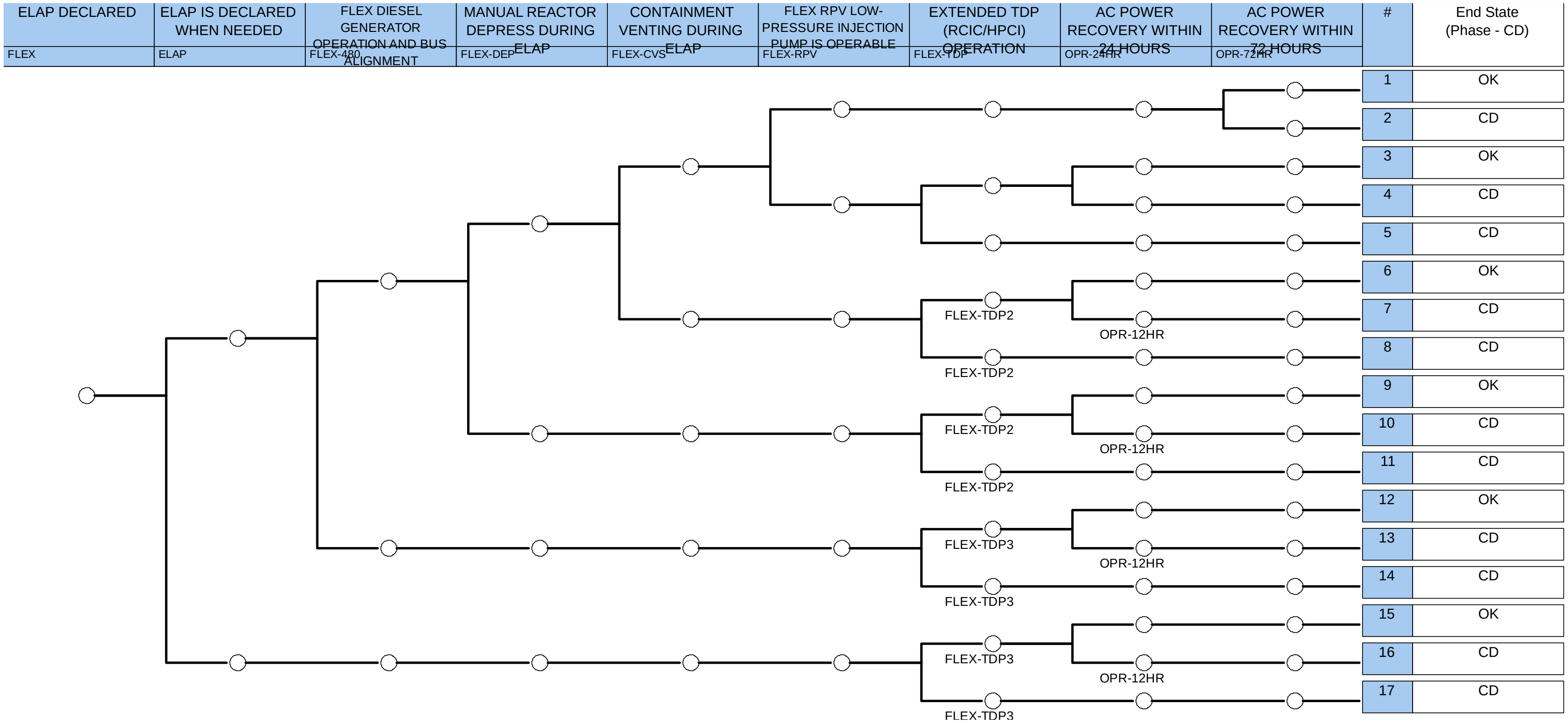

Figure B-13. SBO-ELAP event tree (SBO-ELAP).

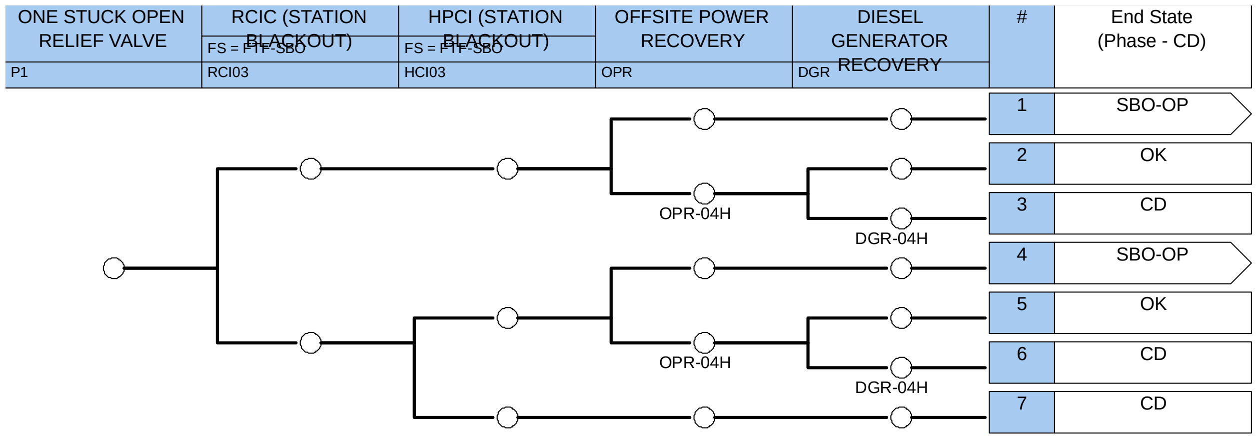

Figure B-14. SBO-1 event tree (SBO-1).

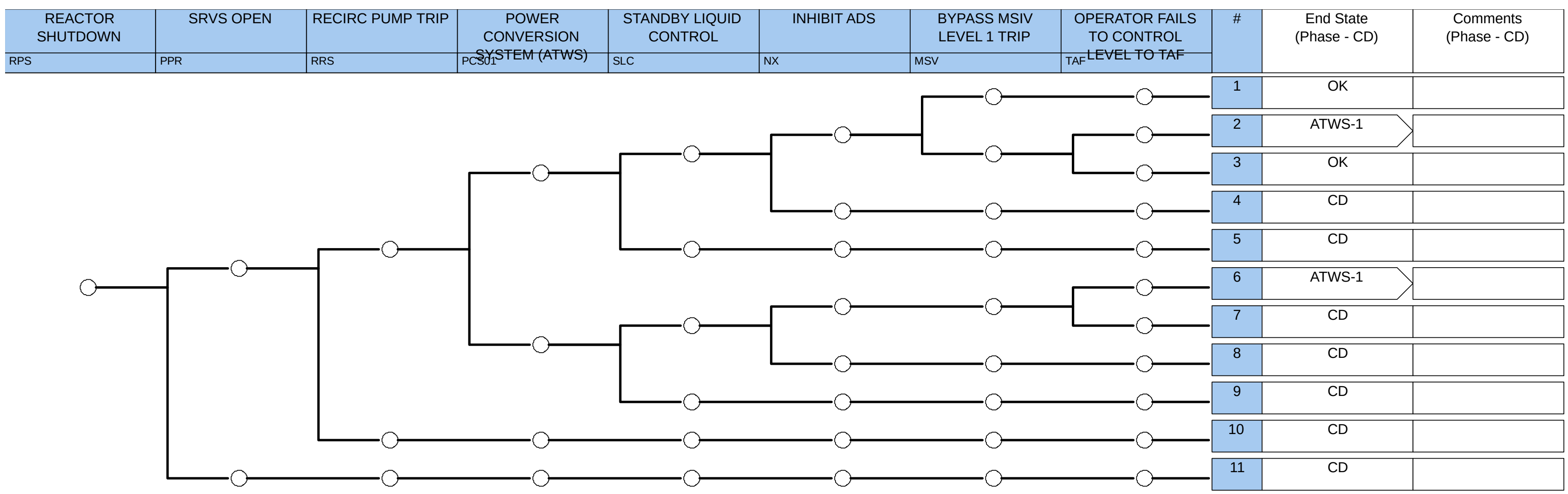

Figure B-15. Anticipated Transient Without Scram event tree. (ATWS)

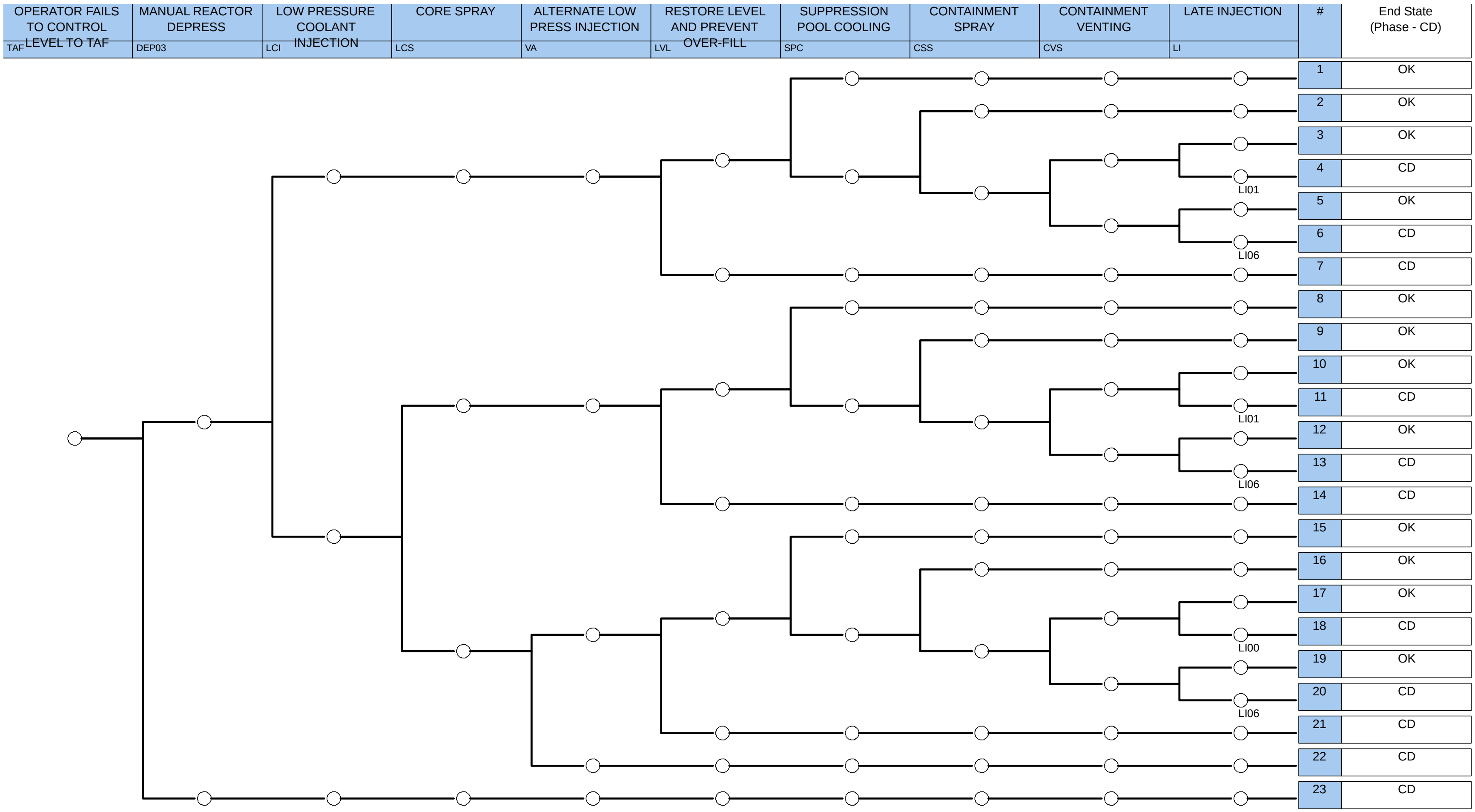

Figure B-16. ATWS-1 event tree (ATWS-1).

Appendix C: FMEA Results

The FMEA results for BWR and PWR are presented on the following pages.

Table C-1 BWR FMEA Results.

Process Function	Potential Failure Mode	Potential Causes/ Mechanisms of Failure	Existing Event Tree?	Ranking Scale (1-10) Severity to CD	Frequency	Detection	RPN	Safety / Economic	General Notes	BWR Unique
External Power	Loss of offsite power	H2 detonation at HTEF	LOOP	3 to 9	3	1	9 to 27	S, E	Severity highly dependent on NPP. Number of plants where a LOOP is a really bad day. It depends on the configuration of Emergency Power	
Spray pond		H2 detonation at HTEF		3	3	3	27	S, E	Debris and above water spray mechanisms, ultimate heat sink	
Cooling Tower pond		H2 detonation at HTEF		3	3	3	27	S, E	Debris in ultimate heat sink	
Primary loop transport of process steam	Pipe Rupture after MSIV	Placement of HES in the turbine building: Damage to turbine building equipment, possibly safety power buses, depending on the plant	STM-LINE-BREAK	7	3	1	21	S, E	Recommend placement of HES in a dedicated building. This study will model PRA with that assumption.	Severity is higher for BWR, 7, Need to isolate in HES and dump steam to condenser
Primary loop transport of process steam	Pipe Rupture after MSIV	Operational vibration seismic, erosion,	STM-LINE-BREAK	4	3	1	12	S, E		
Service water pump house		H2 detonation at HTEF		3	3	1	9	E	As sited 1 km distance NPP to HTEF	

Process Function	Potential Failure Mode	Potential Causes/ Mechanisms of Failure	Existing Event Tree?	Ranking Scale (1-10) Severity to CD	Frequency	Detection	RPN	Safety / Economic	General Notes	BWR Unique
	Heat Exchanger Leak	Contamination of the tertiary HTF loop with process steam	STM-LINE-BREAK	7	1	1	7	E	Steam in Therminol 66. Decrease lifetime through emulsion, cracking hydrocarbons. Talk with chemist. Ec severity: 5	Contamination of oil. Therminol will bind tritium. Mostly N-16 out of BWR steam. Possibly C-60, but little chance. Ec severity: 7
	Heat Exchanger Leak	Over pressurization of tertiary loop		2	3	1	6	E	Relief valve in tertiary loop	
External Supply Tanks integrity	Damage to CST, other supply tanks	H2 detonation at HTEF		2	3	1	6	S, E	As sited 1 km distance NPP to HTEF	
Forced air cooling		H2 detonation at HTEF		2	3	1	6	S, E		
Turbine load of up to 90%	Loss of 90% load immediately if used in following			2	2	1	4	S, E	Depends on the way power is placed on grid or to facility. Possibility of turbine trip. Need to quantify for PRA.	
H2 in NPP process	H2 piped back to NPP		TRANSIENT	1	1	2	2	S,E		BWR uses more H2. Already in risk model for local generation
	Flammability of HTF?			2	1	1	2	E	Data sheet shows minimal flammability	

Process Function	Potential Failure Mode	Potential Causes/ Mechanisms of Failure	Existing Event Tree?	Ranking Scale (1-10) Severity to CD	Frequency	Detection	RPN	Safety / Economic	General Notes	BWR Unique
Steam diversion load roughly 5% thermal	Loss of 5% load immediately			0	2	2	0	S,E	NPP can handle up to 30% prompt load loss, so not a hazard	
Critical structure integrity	Damage to reactor building walls	H2 detonation at HTEF	No fragility	10	0	1	0	S, E	As sited 1 km distance NPP to HTEF	
H2 to transfer facility	pipeline failure leaks H2 close to electrical lines			UNK	UNK	5	UNK	Potential S, E	Where would the tank farm be? How many tanks? Or only a pipeline to other facilities?	

Table C-2 PWR FMEA Results.

Process Function	Potential Failure Mode	Potential Causes/ Mechanisms of Failure	Existing Event Tree?	Ranking Scale (1-10) Severity to CD	Frequency	Detection	RPN	Safety / Economic	General Notes	PWR Unique
Spray pond		H2 detonation at HTEF		3	3	3	27	S, E	Debris and above water spray mechanisms, ultimate heat sink	
Cooling Tower pond		H2 detonation at HTEF		3	3	3	27	S, E	Debris in ultimate heat sink	
External Power	Loss of offsite power	H2 detonation at HTEF	LOOP	3 to 9	3	1	9 to 27	S, E	Severity highly dependent on NPP. Number of plants where a LOOP is a really bad day. It depends on the configuration of Emergency Power.	
Primary loop transport of process steam	Pipe Rupture after MSIV	Placement of HES in the turbine building: Damage to turbine building equipment, possibly safety power buses, depending on the plant	STM-LINE-BREAK	4	3	1	12	S, E	Recommend placement of HES in a dedicated building. This study will model PRA with that assumption.	Severity less in PWR
Primary loop transport of process steam	Pipe Rupture after MSIV	Operational vibration seismic, erosion,	STM-LINE-BREAK	4	3	1	12	S, E		
Service water pump house		H2 detonation at HTEF		3	3	1	9	E	As sited 1 km distance NPP to HTEF	
Forced air cooling		H2 detonation at HTEF		2	3	1	6	S, E		

Process Function	Potential Failure Mode	Potential Causes/ Mechanisms of Failure	Existing Event Tree?	Ranking Scale (1-10) Severity to CD	Frequency	Detection	RPN	Safety / Economic	General Notes	PWR Unique
	Heat Exchanger Leak	Overpressurization of tertiary loop		2	3	1	6	E	Relief valve in tertiary loop	
External Supply Tanks integrity	Damage to CST, other supply tanks	H2 detonation at HTEF		2	3	1	6	S, E	As sited 1 km distance NPP to HTEF	
	Heat Exchanger Leak	Contamination of the tertiary oil loop with process steam	STM-LINE-BREAK	5	1	1	5	E	Steam in Therminol 66. Decrease lifetime through emulsion, cracking hydrocarbons. Talk with chemist. Ec severity: 5	
Turbine load of up to 90%	Loss of 90% load immediately if used in following			2	2	1	4	S, E	Depends on the way power is placed on grid or to facility. Possibility of turbine trip. Need to quantify for PRA.	
H2 in NPP process	H2 piped back to NPP		TRANSIENT	1	1	2	2	S,E		PWR less of a hazard. H2 levels are low and are in risk models of applicable NPPs
	Flammability of heating oil?			2	1	1	2	E	Data sheet shows minimal flammability	
Steam diversion load roughly 5% thermal	Loss of 5% load immediately			0	2	2	0	S,E	NPP can handle up to 30% prompt steam load loss, so not a hazard	

Process Function	Potential Failure Mode	Potential Causes/ Mechanisms of Failure	Existing Event Tree?	Ranking Scale (1-10) Severity to CD	Frequency	Detection	RPN	Safety / Economic	General Notes	PWR Unique
Critical structure integrity	Damage to reactor building walls	H2 detonation at HTEF		10	0	1	0	S, E	As sited 1 km distance NPP to HTEF	
Primary loop transport of process steam	Heat Exchanger Leak	Erosion, vibration	STM-LINE-BREAK				0			
H2 to transfer facility	pipeline failure leaks H2 close to electrical lines			UNK	UNK	5	UNK	Potential S, E	Where would the tank farm be? How many tanks? Or only a pipeline to other facilities?	