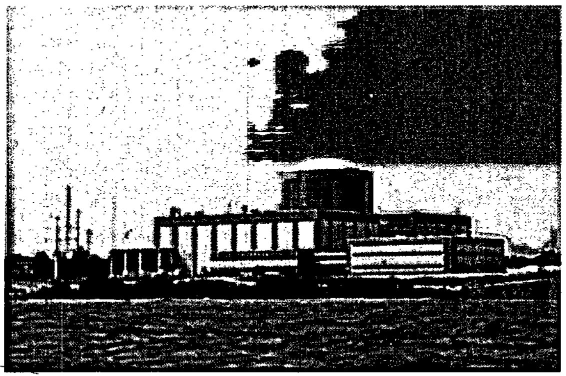

Safety Significance Evaluation of Kewaunee Power Station Turbine Building Internal Floods

Volume 1

Revision 1

December 1, 2005

Dominion

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Executive Summary

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A performance deficiency was identified in NRC Inspection Report 05000305/2005011 regarding internal flooding design features. The inspectors found that there was inadequate design control to ensure Class I equipment was protected against damage from the rupture of a pipe or tank resulting in serious flooding or excessive steam release to the extent that the Class I equipment's function was impaired. Specifically, the design did not ensure that the auxiliary feedwater (AFW) pumps, 480-volt (V) safeguards buses, safe shutdown panel, emergency diesel generators (EDGs) 1A and 1B, and 4160-V safeguards buses 1-5 and 1-6 would be protected from random or seismically-induced failures of non-Class I systems in the turbine building. Flood paths were present which would allow flood water from the turbine building to flow into the safeguards alley compartments containing the identified Class I equipment. These flood paths included floor drains without check valves, doors with sufficient bottom clearances to allow water to pass through, and open floor trenches which communicate between safeguards alley compartments.

The past safety significance of this performance deficiency was evaluated by performing a probabilistic risk assessment (PRA) of the subject internal flooding scenarios leading to core damage. The flood initiating events considered included: random pipe breaks, condenser expansion joint failures, steam line breaks with fire sprinkler actuation, feedwater line breaks with fire sprinkler actuation, seismic-induced breaks, turbine-missile induced breaks, and tornado-induced breaks. The scenarios were analyzed based on: surveyor floor measurements, dynamic flood level analysis using GOTHIC, equipment survivability evaluations, room heatup calculations using GOTHIC, simulator exercises, review of operator training materials, testing of 480-V breakers in simulated flooding conditions, and seismic fragility assessments. The turbine building flood sources capable of causing failure of Class I equipment in safeguards alley were determined to be: circulating water, service water, firewater, feedwater, condensate, and the condensate and reactor makeup water storage tanks.

The total contribution to core damage frequency (CDF) from this deficiency based on the plant design in 2004 was evaluated to be 7.2E-05 per year, which would be classified as Yellow in the NRC Reactor Oversight Process (ROP) Significance Determination Process (SDP) risk determination. The total large early release frequency (LERF) contribution from this deficiency was estimated to be at least a factor of ten below the CDF, and thus not limiting in the NRC ROP SDP risk determination. Sensitivity evaluations were performed to determine the impact of changes in key assumptions such as initiating event frequencies and human error probabilities.

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1 Introduction

A performance deficiency was identified in NRC Inspection Report 05000305/2005011 regarding internal flooding design features (Ref. 1). The inspectors found that there was inadequate design control to ensure Class I equipment was protected against damage from the rupture of a pipe or tank resulting in serious flooding or excessive steam release to the extent that the Class I equipment's function was impaired. Specifically, the design did not ensure that the auxiliary feedwater (AFW) pumps, 480-volt (V) safeguards buses, safe shutdown panel, emergency diesel generators (EDGs) 1A and 1B, and 4160-V safeguards buses 1-5 and 1-6 would be protected from random or seismically induced failures of non-Class I systems in the turbine building. Flood paths were present which would allow flood water from the turbine building to flow into the safeguards alley compartments containing the identified Class I equipment. These flood paths included floor drains without check valves, doors with sufficient bottom clearances to allow water to pass through, and open floor trenches which communicate between safeguards alley compartments.

2 Conclusions

The total contribution to core damage frequency (CDF) from this deficiency based on the plant design in 2004 was evaluated to be 7.2E-05 per year, which would be classified as Yellow in the NRC Reactor Oversight Process (ROP) Significance Determination Process (SDP) risk determination. The total large early release frequency (LERF) contribution from this deficiency was estimated to be at least a factor of ten below the CDF, and thus not limiting in the NRC ROP SDP risk determination. Sensitivity evaluations were performed to determine the impact of changes in key assumptions such as initiating event frequencies and human error probabilities.

3 Evaluation

3.1 Flood Sources

In this analysis, failures of non-Class I water system piping and equipment at Kewaunee Power Station (KPS) that can flood the turbine building and subsequently impact Class I components have been evaluated. Systems with sufficient inventory and flow rates to fail Class I equipment in safeguards alley were determined to be: circulating water, service water, firewater, feedwater, condensate, and condensate and reactor makeup water storage tanks. Twelve different random (9), tornado-induced (1), turbine-missile induced (1), and seismic-induced (1) flooding initiating events listed in Table 3-1 were evaluated. The frequencies of these flooding events were determined based on plant-specific analyses and industry references.

The critical flood levels for Class I equipment in safeguards alley potentially-impacted by turbine building floods are listed in Table 3-2. These levels were determined by measurements, engineering evaluations, and tests of equipment in flooded conditions.

3.2 Accident Scenarios

Based on identification and analysis of internal flood areas in the KPS turbine building and safeguards alley (including consideration of unoccupied floor space, risk-significant components and associated submergence depths, drainage paths and capacity, detection methods, operator actions, and propagation paths to/from other flood areas), accident scenarios were developed for each of the flooding initiating events described above. The accident scenarios for each initiating event are very similar with differences only in detection method and time to fail Class I equipment. For each initiating event the propagation paths into safeguards alley and the subsequent component damage are the same.

A flooding event due to a non-Class I break would be indicated by a turbine building miscellaneous sump level high alarm in the control room due to high level in either the turbine building or screenhouse sump. The drains and sumps alarm procedure instructs the operator to dispatch personnel to locally investigate the sump when this alarm sounds. Indication may also be provided by alarms related to the system with the break (e.g., low condenser vacuum, service water low discharge pressure, fire pump running or fire protection header pressure low, or steam generator low level depending on the break). The break would deposit water from the circulating water, service water, fire water system, or condensate and reactor makeup water storage tanks onto the turbine building floor. In addition, a break in the feedwater or main steam system that actuates the fire sprinklers would increase the temperature in the turbine building, which would impact the timing for investigation and isolation of the leak.

The water levels in the 480 V switchgear bus 61 and 62 room, the motor-driven auxiliary feedwater pump 1B (MDAFP 1B) room, the turbine-driven auxiliary feedwater pump (TDAFP) room, the MDAFP 1A room, and the CO2 storage tank 1B room would closely match the water level in the turbine building because the drain lines that connect these rooms to the turbine building sump do not contain check valves and would allow water to flow from the sump to these rooms. The water level in the 480 V switchgear bus 51 and 52 (bus 51/52) room would be lower than the turbine building because water would be entering this room via leakage under the doors from adjacent compartments. Water would rise in the bus 51/52 and diesel generator 1A (DG 1A) rooms simultaneously due to the trench connecting the two rooms. The only drainage from the DG 1A room would be leakage to the screenhouse pipe tunnel via the gap under the door and a four-inch opening into the trench. The DG 1A room drain line would not remove any of the flood water because its drain line (which contains a check valve) empties into the turbine building sump, which would already be above this level. If the water level in DG 1A exceeds a depth of 4 inches, 4 kV bus 5 and 480 V Buses 51 and 52 (which are powered from 4 kV bus 5) are conservatively evaluated to fail.

The water level in DG 1B room would also be fed by leakage under a door. The only drainage from the DG 1B room would be leakage under the door leading to the screenhouse pipe tunnel, because the room drain line (which contains a check valve) leads to the turbine building sump. Prior to late 2004, there was a six-inch curb in the DG 1B room that protected the diesel generator and 4 kV bus 6 from floods below six-inches. This curb was removed in late 2004. The curb has minimal impact on the analysis based on the dynamic water level evaluation and was not credited in the analysis.

Although propagation of water from the turbine building to the 4 kV buses would require some period of time, without a procedure or equipment for removing water from the room, it would have been inevitable for the water to eventually reach the buses if the flood source was not isolated.

3.3 Accident Sequence Progression

From the flooding initiating events and damage scenarios described above, the accident sequence progression has been analyzed. The accident sequence progression for each flooding event considers the response of the plant and operators to the initiating events and subsequent equipment failures, and is represented with an event tree. The flooding event trees are based on the KPS internal events PRA model event tree for loss of feedwater. In each case, if the operator successfully terminates the flood prior to failure of any buses, the accident progression would be identical to that of the existing loss of feedwater sequences except for equipment failed by spray from the initiating line break.

As with the accident scenarios, the accident sequence progression for each initiating event is very similar with differences only in the operator actions needed (i.e., isolation of the appropriate flood source) and the time required and available for those actions. The accident sequence progression following failure to isolate the flood before failure of any buses is described below.

A circulating water break would be isolated by manually tripping the circulating water pumps. For a service water break, the operator would isolate the turbine building header by closing valves SW-4A and -4B. For a high energy line break leading to fire sprinkler actuation, the operator would implement a procedure to isolate the discharge from the fire water system into the turbine building by isolating the fire sprinklers on the turbine building mezzanine level, and isolating deluge and fire sprinkler valves in the turbine building basement. Also, the operators could trip the fire pumps locally at the 480V breakers or locally close the pump discharge valves to stop flow, but the operators were conservatively not credited to pass through flooded switchgear areas in safeguards alley to perform these actions.

If the operator fails to isolate a flood before all RCP seal cooling systems are lost, then an RCP seal LOCA could occur. The response to the RCP seal LOCA would depend on the leakage rate. The WOG 2000 RCP seal LOCA model as modified by the NRC was used for this evaluation.

If the operator fails to isolate the break initially, the water level would continue to rise in safeguards alley. Although 4 kV bus 5 motor-loads would fail, buses 51/52, 4 kV bus 6 and associated 480 V buses 61/62 would still be available, as well as the TDAFP. There is a second isolation opportunity in order to prevent eventual failure of the TDAFP's ability to start due to submergence of the associated auxiliary lube oil pump (at 9 inches). A third isolation opportunity exists to prevent eventual failure of 4 kV bus 6 (at 4 inches) and associated 480 V buses 61/62 (at 11 inches). The total volume of water required in the turbine building to flood 4 kV bus 6 is almost equal to that required to flood 480 V buses 61/62. A fourth isolation is also modeled to prevent submergence failure of the MDAFPs at 13 inches. This isolation also ensures that power to 480 VAC buses will remain available.

If the second or third isolation opportunity were successful, 4 kV power would be available to the already operating MDAFP 1B. If continued operation of this MDAFW pump succeeds, the operator performs RCS cooldown and depressurization by opening a SG PORV (which if necessary can be performed locally) to reduce RCP seal leakage. If cooldown fails, the operator could still remove decay heat by restoring RCS inventory using the available SI pump and throttling SI flow to conserve the water in the RWST per procedure.

If the available MDAFP fails, the TDAFP would be available to provide secondary heat removal. Successful cooldown using the TDAFP also requires opening a SG PORV. Additionally, long-term instrument power must be available to allow the operator to monitor SG level and prevent overfilling the SG and failing the TDAFP. Because the normal battery chargers would be unavailable due to the loss of the 480 V buses, providing long-term DC power for steam generator level indication and auxiliary feedwater control is credited by a number of means, including automatic or manual transfer of the inverters source from the batteries to their alternate source (offsite power), which would be available in many scenarios. In addition, a normal or spare battery charger could be powered from offsite power or the Technical Support Center (TSC) diesel to restore long term battery capacity and provide SG level indication. Due to the long time to steam generator dryout due to reduced decay heat levels at the earliest point the batteries might be depleted (eight hours), much more than eight hours would be available in the most limiting cases to implement these recovery actions (e.g., a minimum of 24 hours of battery life is available if the inverters are transferred to their alternate source at four hours).

A final isolation opportunity can prevent the water level in the turbine building from reaching 18 inches. If the water level reaches this height, core damage is assumed since the electrical connections of the reserve auxiliary transformer (RAT) to 4 kV buses 1 and 2 will be submerged leading to a loss of offsite power and the eventual failure of all safety-related buses. Additionally, this water level will result in the failure of the diesel generators since their air supply fans are powered from 480 V buses 51/52 and 61/62.

Seismic-induced floods were analyzed based on the EPRI 1989 hazard curve and associated spectra, detailed fragility assessments of the systems capable of causing critical floods in the turbine building impacting Class I components in safeguards alley, and random failures taken from the PRA models from the random pipe break analyses. Combinations of breaks which could occur in seismic events were explicitly considered in the analysis.

3.4 Operator Actions

As described above, the accident sequence progression for each initiating event is very similar with differences only in the operator actions and the time required and available for those actions. Most of these operator actions fall into one of three groups: isolation of the flood source before 4 kV bus 5 fails, isolation of the flood source before the TDAFP auxiliary lube oil pump fails, or isolation of the flood source before 4 kV bus 6 and associated 480 V buses 61/62 fail, or isolation before submergence of the motor-driven AFW pumps.

The human error probabilities (HEPs) for these actions vary for each flooding initiating event, based on the specific actions to be taken to isolate the particular flood source, the time required to complete those actions, the time available to complete those actions (based on the flow rate of the source), and the environment in which the actions must be performed. As noted above, the hot water and/or steam released from a feedwater or main steam line break would impact the operators' ability to investigate and isolate the flood. The impact of these conditions and dependencies among these three actions are also considered.

3.5 Results

The turbine building flooding analysis summarized above represents a conservative assessment for occurrence, plant response, and operator response to a flooding event in the turbine building. Quantification of this conservative analysis provides the core damage frequency (CDF) for the plant configuration in the year 2004. Table 3-3 presents the individual and total CDFs for each of the flooding scenarios.

The total contribution to CDF from the deficiency for the analyzed turbine building flood scenarios was calculated to be 7.2E-05. More than 79% of the CDF is due to four flood scenarios: large breaks in an inlet circulating water expansion joint (50%), feedwater line breaks that results in full flow discharge from the fire pumps (14%), main steam line breaks that results in full flow discharge from the fire pumps (15%), and seismic induced breaks of firewater, service water and condensate and reactor makeup water storage tanks (9%).

3.6 Conservatisms

Development of the initiating events, accident scenarios, accident sequence progression, and human error probabilities for turbine building floods in some cases required the use of conservative modeling methods or conservative assumptions. The noteworthy conservatisms inherent in the KPS turbine building flooding analysis are summarized below.

1. The impact of tripping the feedwater and condensate pumps prior to emptying the hotwell was not evaluated. Instead it was conservatively determined that the entire feedwater and condensate inventory of 80,000 gallons would be pumped onto the turbine building floor. The feedwater pumps would likely be tripped early (within approximately ten minutes per the emergency operating procedures), and an extremely large break size (8,000 gpm) would be required to discharge 80,000 gallons within that period. A smaller break size would result in less water discharged and allow more time to isolate the break to prevent failure of risk significant components.
2. Credit for operators isolating the firewater pumps following high energy line break events by either tripping the firewater pumps at the 480V switchgear, closing the firewater pump discharge valves, or initiating a manual safety injection signal (which automatically trips the firewater pumps) were not included in the analysis.

3.7 Sensitivity Analyses

Development of the initiating events, accident scenarios, accident sequence progression, and human error probabilities for turbine building floods requires many assumptions. To help characterize the modeling and data uncertainty due to assumptions made for this evaluation, a series of sensitivity analyses were performed and are summarized in Table 3-4.

Table 3-1. **Flood Initiating Events and Frequencies**

Initiating Event	Frequency (per year)
Large random circulating water inlet expansion joint break (58,000 gpm)	3.7E-05
Large random circulating water outlet expansion-joint break (14,000 gpm)	2.9E-05
Small random circulating water expansion joint failure (6,000 gpm)	7.3E-05
Random service water system break with equivalent diameter greater than four inches	3.2E-05
Random fire water line with equivalent diameter greater than four inches	7.1E-05
Random feedwater or condensate high-energy line break that actuates sufficient turbine building fire sprinklers for full fire water flow	1.4E-04
Random feedwater or condensate high-energy line break that actuates 100 turbine building fire sprinklers	4.7E-05
Random main steam high-energy line break that actuates sufficient turbine building fire sprinklers for full fire water flow	2.5E-04
Random main steam high-energy line break that actuates 100 turbine building fire sprinklers	1.9E-05
Tornado-induced break of circulating water lines, firewater lines, service water lines, feedwater, condensate, and condensate and reactor makeup water storage tanks	Negligible
Turbine-missile induced break of circulating water lines, firewater lines, service water lines, feedwater, condensate, and condensate and reactor makeup water storage tanks	Negligible
Seismic-induced break of circulating water lines, firewater lines, service water lines, feedwater, condensate, and condensate and reactor makeup water storage tanks	EPRI,1989 Hazard Curve (see Appendix F, Table 3-1) (1)

(1)Changes in the human error probabilities and safety-related bus failure heights from the earlier revision of this analysis (Ref. 2) were evaluated for the seismic-induced floods and found to result in a small decrease (1E-07 per year) in core damage frequency. Therefore, the seismic-induced flood analysis documentation was not revised from the earlier revision.

Table 3-2. **Flood Levels Impacting Class I Equipment**

Train A/B 480V switchgear (buses 51, 52, 61, 62)
• 2.75" flood level trips bottom row of breakers
• 4" flood level control power lost
• 11" flood level bus stabs covered and bus fails
Train A/B 4kV switchgear (buses 5 and 6 located in respective EDG rooms)
• 4" flood level control power connections covered, 4kV motor loads will receive lockout signal, and breaker control fails (however, supply to 480V buses will remain energized)
• 18" flood level bus stabs covered and bus fails
Turbine-driven AFW pump
• 9" flood level auxiliary lube oil pump fails
• 18" flood level pump fails
Motor-driven AFW pumps
• 9" flood level auxiliary lube oil pump fails
• 13" flood level pump fails
Instrument air compressors (A, B, C)
• 11" flood level compressor fails
Emergency diesel generators and dedicated shutdown panel
• Equipment is above 6" flood level, however associated 4kV buses fail @ 6" flood level

Note: Flood levels impacting equipment failure were conservatively assessed from measured levels to allow for measurement uncertainty (typically ¼" to ½" less than measurement). Flood levels provided in this table are relative to floor elevation at equipment. Flood levels used in analysis were relative to sea level.

Table 3-3. **Flood Scenario Contributors to Turbine Building Flooding Results**

Flood Scenario	Total CDF (per r)
Large random circulating water inlet expansion joint break (58,000 gpm)	3.7E-05
Large random circulating water outlet expansion joint break (14,000 gpm)	4.4E-06
Small random circulating water expansion joint failure (6,000 gpm)	1.9E-06
Random service water system break with equivalent diameter greater than four inches	1.3E-06
Random fire water line with equivalent diameter greater than four inches	1.9E-06
Random feedwater or condensate high-energy line break that actuates sufficient turbine building fire sprinklers for full fire water flow	9.1E-06
Random feedwater or condensate high-energy line break that actuates 100 turbine building fire sprinklers	1.2E-07
Random main steam high-energy line break that actuates sufficient turbine building fire sprinklers for full fire water flow	9.7E-06
Random main steam high-energy line break that actuates 100 turbine building fire sprinklers	5.0E-08
Tornado induced break of circulating water lines, firewater lines, service water lines, feedwater, condensate, and condensate and reactor makeup water storage tanks	Negligible
Turbine missile induced break of circulating water lines, firewater lines, service water lines, feedwater, condensate, and condensate and reactor makeup water storage tanks	Negligible
Seismic induced break of circulating water lines, firewater lines, service water lines, feedwater, condensate, and condensate and reactor makeup water storage tanks	6.6E-06
Total	7.2E-05

Table 3-4: Sensitivity Cases

Analysis Case[(1)]	Total CDF
Baseline	7.2E-05
HEPs for operator actions with less than a 30-minute time window available from initiating event increased by factor of 5	8.4E-05
HEPs for operator actions with less than a 30-minute time window available from initiating event increased by factor of 10	8.4E-05
HEPs for operator actions with less than a one-hour time window available from initiating event increased by factor of 5	1.0E-04
HEPs for operator actions with less than a one-hour time window available from initiating event increased by factor of 10	1.2E-04
HEPs for unproceduralized operator actions increased by factor of 5	1.3E-04
HEPs for unproceduralized operator actions increased by factor of 10	1.3E-04
High energy (main steam and feedwater) line break frequencies increased by factor of 5	1.5E-04
High energy (main steam and feedwater) line break frequencies increased by factor of 10	2.4E-04
Circulating water expansion joint break frequencies increased by factor of 5	1.9E-04
Circulating water expansion joint break frequencies increased by factor of 10	3.4E-04
Random pipe break frequencies increased by factor of 5	1.4E-04
Random pipe break frequencies increased by factor of 10	2.3E-04
First HEP for firewater isolation following high energy line break changed to 0.1	6.9E-05
First HEP for firewater isolation following high energy line break changed to 0.3	7.0E-05
First HEP for firewater isolation following high energy line break changed to 0.6	7.1E-05
HEPs for firewater isolation following high energy line breaks assume 38 versus 32 minute average isolation time(2), first HEP for firewater isolation following high energy line break changed to 0.1	7.9E-05
HEPs for firewater isolation following high energy line breaks assume 38 versus 32 minute average isolation time(2), first HEP for firewater isolation following high energy line break changed to 0.3	8.0E-05
HEPs for firewater isolation following high energy line breaks assume 38 versus 32 minute average isolation time(2), first HEP for firewater isolation following high energy line break changed to 0.6	8.1E-05
Large condenser outlet and small circulating water expansion joint break frequencies increased by factor of 5	9.5E-05
Large condenser outlet and small circulating water expansion joint break frequencies increased by factor of 5; all HEPs for large condenser outlet circulating water expansion joint break isolations set to 0.05	8.3E-05
Large condenser outlet and small circulating water expansion joint break frequencies increased by factor of 5; all HEPs for large condenser outlet circulating water expansion joint break isolations set to 0.12	9.4E-05
Large outlet and small circulating water expansion joint break frequencies increased by factor of 10	1.2E-04
Large condenser outlet and small circulating water expansion joint break	1.0E-04

Analysis Case[1]	Total CDF
frequencies increased by factor of 10; all HEPs for large condenser outlet circulating water expansion joint break isolations set to 0.05	
Large condenser outlet and small circulating water expansion joint break frequencies increased by factor of 10; all HEPs for large condenser outlet circulating water expansion joint break isolations set to 0.12	1.2E-04
Kewaunee IPE frequency of circulating water expansion joint break (2E-04 per year) used for large condenser inlet, large condenser outlet, and small circulating water expansion joint break frequencies	2.8E-04
Kewaunee IPE frequency of circulating water expansion joint break (2E-04 per year) used for large condenser inlet circulating water expansion joint break frequency; large condenser outlet, and small circulating water expansion joint break frequencies set to zero	2.4E-04

(1) HEPs in sensitivities were increased to a maximum of 0.5.

(2) Average time is for isolation of mezzanine fire sprinkler valves. GOTHIC analysis (Ref 3) indicates an additional 10 minutes is available to isolate the basement firewater valves before additional Class I equipment failures occur.

4 References

1. Letter, USNRC (Mark Satorius) to Dominion (David Christian), NRC Inspection Report 05000305/2005011(DRP) Preliminary Greater than Green Finding Kewaunee Power Station, October 6, 2005.
2. Letter, Dominion Energy Kewaunee, Inc (Michael G. Gaffney) to USNRC, "Kewaunee Power Station Flooding Significance Determination Process Risk Assessment Report," October 31, 2005.
3. MPR Associates Inc. Calculation No. 0064-0515-LYS-01, "Evaluation of Flooding Levels for Various PRA Cases", Rev. 1, November 18, 2005.

Appendix A

Initiating Events Analysis for Turbine Building Floods

INTERNAL FLOODING – Initiating Events Analysis for Turbine Building Floods

Owner's Acceptance: Thomas G Hook

Signature

THOMAS G HOOK

Print Name

12/01/05

Date

Kewaunee Power Station

Initiating Events Analysis for Turbine Building Floods

Revision No. 1

Effective Date: November 2005

S. E. GUOKAS VIA EMAIL 12/01/05
Prepared By: S. E. Guokas Date

R. J. DREMEL VIA EMAIL 12/01/05
Reviewed By: R. J. Dremel Date

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1.0 PURPOSE

The purpose of the internal flooding initiating events analysis is to define, quantify, and document the frequency results for potential internal flooding initiating events caused by breaks of non-safety-related piping/components in the Turbine Building before February 2005. That is, the analysis considers the plant prior to installation of the flood mitigation modifications installed in and around safeguards alley. Flooding events caused by earthquakes are considered separately.

The following information is identified, correlated, and developed as part of this analysis:

- Identification of pipe breaks of concern
- Quantification of the frequency expected for pipe breaks in those systems.

2.0 MODEL DEVELOPMENT

Internal flooding analysis encompasses the effects from the accumulation of fluids arising from the rupture, cracking or incorrect operation of piping/components within the station. In practice, major internal floods have occurred in nuclear power plants, from the rupture of pipes, valves and expansion joints as well as from operator errors during plant maintenance activities. All potential internal flood sources in the turbine building are considered in this analysis.

The steps for conducting the internal flooding initiating events analysis are described in the following section.

2.1 Steps for Turbine Building Internal Flooding Initiating Events Analysis

The analysis of the Turbine Building internal flooding initiating events analysis consists of the following steps:

1. Determine the volume of water that can be released before failure of equipment in safeguards alley would be expected.

2. Screen from consideration, those systems that cannot be significant contributors to the overall turbine flooding risk. Screen from consideration systems that are not capable of causing failure of equipment even if the entire system volume is released or if a break in the system was allowed to flow for a long period of time.

3. Review information collected from the internal flooding walkdown and screening analysis [NB01] to identify potential flood sources. Review drawings to identify other potential flood sources not included in [NB01].

4. Identify the specific piping and components that can cause an internal flood. For these pipes and components, calculate the frequency for flooding events of concern.

The results from each of these steps are presented in Section 3.0.

Development of the flood scenarios and accident sequence progression for each of the identified initiating events is documented in a separate report.

2.2 Turbine Building Internal Flooding Initiating Events Major Assumptions

The key assumptions that were made during the internal flooding initiating events analysis are discussed in Section 3.0 for each of the specific flooding scenarios. In addition, the following general assumptions apply:

1. Actuation of fire sprinkler heads can also occur due to localized heating from operating equipment, aging failure, or impact damage from maintenance activities. Inadvertent actuation will result in discharge from a single sprinkler head, with a maximum rate of 30 gpm [CALC01]. The low flow rate from actuation of a single sprinkler head is assumed to be too low to cause equipment damage outside of the immediate area and, therefore it would be no more severe than a loss of main feedwater event. Therefore, it is concluded that flooding events that result only in failure of equipment located in the Turbine Building can be considered subsumed by the frequency of loss-of-main-feedwater transient events.

2. All piping systems in the Turbine Building are assumed to be non-safety related. Therefore, all pipes are initially considered as potentially causing an initiating event.

3. All flooding events in the Turbine Building are assumed to cause a loss of main feedwater and, therefore, result in a reactor trip. If a flooding event does not cause a reactor trip, the flood could be excluded as an initiator. The effect of this assumption is that all pipe breaks are initially considered as potentially causing an initiating event.

4. The service water return lines are assumed to operate at the same pressure as the supply headers. The impact of this assumption is that some breaks in service water return lines that may be screened as initiating events are included in the overall initiating event frequency. The impact of this assumption is expected to result in only a slight increase in the overall initiating event frequency.

3.0 TURBINE BUILDING FLOODING INITIATING EVENTS ANALYSIS

Identification and quantification of Turbine Building internal flooding initiating events is discussed below.

3.1 Determination of Water Volume to Fail Equipment in Safeguards Alley

For this analysis, failure of non-safety related systems in the Turbine Building are considered. A flooding event which does not result in failure of equipment outside the Turbine Building would be no more severe than a loss of main feedwater event. Although some equipment used to mitigate a loss of main feedwater event could be failed by the flooding event, the expected impact of these additional failures would be bounded by the loss of main feedwater event modeled in the internal events PRA for the following reasons.

First, other than main feedwater, the only potentially risk significant plant equipment located in the Turbine Building basement are the service air compressors and plant equipment water pumps. The plant equipment water pumps are located on the far southwest corner of the basement area such that a flooding event that would spray those pumps would be unable to spray any other equipment included in the PRA models. In addition, plant equipment water cooling is provided with a backup from service water so failure of these pumps would not directly cause failure of other equipment. The service air compressors are located in the north end of the turbine basement area such that a flooding event that would spray the service air compressors would be unable to spray any other equipment included in the PRA models. Also, the service compressors in the Turbine Building are provided with backup from instrument air compressors located in safeguards alley. Therefore, failure of the service air compressors located in the Turbine Building basement would not directly cause failure of other equipment. On the mezzanine level, non-safety related switchgear, Bus 3, Bus 4, and associated 480 VAC switchgear, and steam dump valves 11A and 11B are located. In the PRA models, the non-safety related switchgear is used only for equipment that otherwise would be failed by the Turbine Building flood. Failure of the steam dump valves can be mitigated by using the steam generator power operated relief valves (PORVs).

The frequency of Turbine Building flooding events is much less than the frequency of loss-of-main-feedwater transient events. Therefore, it is concluded that flooding events that result only in failure of equipment located in the Turbine Building can be considered subsumed by the frequency of loss-of-main-feedwater transient events.

Water released to the Turbine Building will flow to the basement. Drain lines and gaps in doors allow the water to flow to the rooms in the safeguards alley. If the total volume of water released from a pipe break is less than the volume of water needed to fail enough equipment located in the safeguards alley that accident mitigation response is significantly impaired, then the pipe break can be excluded from consideration in the internal flooding events analysis.

Water flowing from the Turbine Building basement to the safeguards alley could potentially fail instrument air compressors, auxiliary feedwater (AFW) pumps, 480 VAC switchgear buses 51, 52, 61, and 62, 4kVAC buses 5 and 6, and diesel-generators 1A and 1B. The first impact that a flooding event will have on equipment in the safeguards alley is when level reaches 2.75 inches on Bus 62 [CALC02] when the bottom row of breakers on the bus would open [CALC03] and the loads listed in Figure 3-2 of [CALC03] would be lost. The next impact of the flood would be when water level reaches 2.75 inches of water on Bus 52 [CALC02] when the bottom row of breakers on the bus would open [CALC03] and the loads listed in Figure 3-1 of [CALC03] would be lost. After loss of the bottom row of breakers on the 480 VAC safety buses, the next impact of a turbine building-flooding event would be loss of motor loads [CALC02] when level reaches 4 inches on Bus 5 [CALC03]. Note that the lockout relays submerged at the 4-inch depth on Bus 5 will only trip the breakers to the motor loads on Bus 5; the transformers to 480 VAC switchgear buses 51 and 52 will not be affected and buses 51 and 52 will still have power.

Reviewing the loads supplied from the bottom row of breakers in the 480 VAC safety buses shows that their loss would not present an immediate challenge to the ability of the operators to mitigate a reactor trip provided that the flood is isolated prior to the flooding event causing failure of other equipment in safeguards alley. The battery chargers are lost when the bottom row of breakers open. Therefore, actions to ensure longer-term availability of DC power must be taken. If the flood is isolated before the A-train electrical safety buses would be failed, then the instrument inverters, BRA-111. BRA-112, BRB-111, and BRB-112, could be powered from their alternate power supply. An evaluation in Attachment 1 to Appendix D shows that adequate time is available to switch inverter power supplies and maintain battery capacity in excess of twenty-four hours. Therefore, this analysis will screen from consideration any flooding event that does not result in water level reaching 4 inches on 4kVAC safety Bus 5.

Analyses show that if 131,000 gallons of water is released to the turbine building in 10 seconds, water level would reach only 2.9 inches on Bus 5 and 3.1 inches on Buses 61/62 [CALC02]. The same analyses show that a release of 200,000 gallons of water into the turbine building in 10 seconds would cause level to reach 5.7 inches on Buses 61/62 and 4.3 inches on Bus 5. Interpolating between the two flood volumes above gives a flood volume of 185,000 gallons as the volume that would just fail the motor loads on Bus 5 and present the first significant challenge to the ability of the operators to mitigate a reactor trip. Therefore, any event that releases less than 185,000 gallons of water is screened from further consideration and the event can be considered subsumed by the loss of main feedwater event analyzed in the internal events PRA.

3.2 Screening of Systems as Potential Turbine Building Flooding Initiating Events

Not all flooding events that release greater than 185,000 gallons of water need to be considered as initiating events. Any pipe break where the flowrate from the break would require more than one hour to release 185,000 gallons is eliminated from consideration. It is reasonable to expect these pipe breaks can be detected and isolated within one hour for the following reasons. First, a Miscellaneous Sump Level High alarm would be received. The alarm response procedure for that alarm [PROC01] directs the operators immediately to the Miscellaneous Drains and Sumps Abnormal Operation procedure [PROC02], which specifies that an operator be sent to investigate this alarm. The Miscellaneous Sump Level High alarm would be actuated before water exceeded the capacity of the turbine room sump and spilled onto the floor. The alarm is received infrequently (See Attachment 1) and typically only during evolutions where excessive water is being directed to the sump. From [PROC02] the operators would enter the appropriate abnormal operating procedure for the affected system.

For a system with a nominal pressure of 100 psig, a break with a three-inch equivalent diameter in a 4-inch line would result in a flow rate of 2100 gpm and a 3-inch equivalent diameter break in a 6-inch line would result in a flow rate of 1800 gpm (See Addendum 1 for details of the associated flow calculations). These flow rates are what would be expected from a sharp orifice-like break in a pipe and do not include any flow reduction that may occur due to head losses in the pipe from the pump to the break. With these flow rates, 88 and 102 minutes respectively would be available for the operators to isolate the break before equipment in safeguards alley would be threatened to the point that the ability of the operators to mitigate a reactor trip would be seriously challenged by the failure of Bus 5 motor loads. For lines smaller than 4-inches, the release rate would be much less, allowing significantly longer than one hour to isolate the break.The service water system supply headers are maintained at a nominal pressure of between 90 and 100 psig [REPORT06]. The service water return lines operate at a lower pressure, but will be assumed to operate at the same pressure as the supply headers. The fire protection system, when in standby is maintained at a pressure between 128 and 143 psig [REPORT01].

Although the volume of the potable water and service water pre-treatment systems is essentially unlimited, the systems contain only small-diameter lines and operate at pressures generally lower than 100 psig. A break in these systems would be expected to result in a release rate that would allow significantly longer than one hour to isolate the break. Therefore, these systems are eliminated from further consideration as causing a negligible increase in flooding risk.

The turbine oil systems contain less than 185,000 gallons and, therefore, are eliminated from further consideration.

The reactor makeup storage tanks have a maximum capacity of 80,000 gallons and, therefore, are eliminated from further consideration.

The condensate storage tanks (CSTs) have a maximum capacity of 150,000 gallons and, therefore, are eliminated from further consideration.

Therefore, all systems except the circulating water, fire protection water, service water, and high-energy line breaks (HELBs) that result in fire protection water system actuation are screened from consideration as flooding sources.

Turbine Missile-Induced Flooding Events

A flooding event could be caused if failure of the turbine generates a missile which then impacts and fails a system capable of causing a significant flooding event. An evaluation of turbine missile effects is presented in Appendix B.9 of the Kewaunee Power Station (KPS) Updated Safety Analysis Report (USAR) and is used as the basis for this analysis.

The probability of turbine missile generation due to fatigue has been determined to be much less than 1.0E-08. For stress corrosion, the probability of failure and missile generation by the original low-pressure turbine rotors is determined to be 1.64E-03 at rated speed and 1.49E-05 for overspeed [CALC05]. Note that the latter value is lower than the former because the latter includes the probability of the overspeed condition. The total probability of turbine missile generation is the sum of these two values or:

$$P_{TotMiss} = P_{MissRate} + P_{MissOver}$$

$$P_{TotMiss} = 1.64E\text{-}03 + 1.49E\text{-}05$$

$$P_{TotMiss} = 1.65E\text{-}03$$

These failure probability values are based on a five-year inspection interval so the frequency of turbine missile generation is determined as follows:

$$F_{TotMiss} = P_{ToteMiss} / 5 \text{ years}$$

$$F_{TotMiss} = 1.65E\text{-}03 / 5 \text{ years}$$

$$F_{TotMiss} = 3.30E\text{-}04 \text{ per year.}$$

Since the performance of the analysis that generated the above values, the low-pressure rotors have been replaced. As stated in USAR section 9.1, the probability of failure of the new rotors is less than the original rotors so the frequency calculated above is bounding for the current plant configuration.

Given that a turbine missile is generated, the probability that it impacts and fails a system capable of causing a significant flood must be considered. Missiles that occur on the operating deck may result in a steam release and could potentially impact the feedwater piping located on the southwest side of the building. Analyses [CALC06] have concluded that steam breaks on the turbine operating deck do not actuate sufficient fire protection sprinklers to present a flooding concern. Therefore, a turbine missile that impacts steam pipe on the operating deck does not present a flooding concern.

The feedwater piping on the operating deck is located on the southwest end of the building across from the southernmost low-pressure turbine. Between the turbine and feedwater piping is a moisture separator reheater (MSR), steam piping, and building structural supports. Only a very small portion of the piping could be impacted by a turbine missile that does not first impact the intervening equipment and structures. Assuming that a missile that impacts the intervening equipment will not cause failure of the feedwater piping on the operating deck, it is estimated based on visual inspections that only 5% of the missiles would be capable of impacting the feedwater piping. Assuming that all turbine missiles that impact the feedwater piping cause failure of the piping and actuate fire protection sprinklers, the frequency of such events is:

(3.30E-04 per year) * 0.05 = 1.65E-05 per year.

As described above, this frequency is bounding because the probability of failure for the new rotors is less than that of the old rotors on which these values are based. Also, this value assumes that all missiles that impact the feedwater piping penetrate the piping. Therefore, the frequency of turbine-missile-induced failures of feedwater piping on the operating deck would be negligible.

Turbine missiles that exit below the turbine shaft would be stopped by the concrete turbine support structure or imbedded in the condenser structure itself. Given the physical configuration of the turbine support structure and the condenser, a turbine missile would need to exit downward at a near vertical trajectory to imbed in the condenser. In doing so, the missile would contact the in-condenser feedwater heaters prior to contacting the circulating water tubes. If the missile did contact the circulating water tubes, such a failure would allow flow of circulating water back to the lake. Therefore, it is concluded that the flooding risk posed by turbine missiles that exit below the turbine rotor is considered negligible.

As described above, a conservative analysis of turbine-generated missiles concludes that the frequency of flooding events initiated by turbine missiles is sufficiently small as to be excluded from further analysis.

Tornado-Induced Flooding Events

Flooding events in the Turbine Building potentially could be initiated by the occurrence of a tornado which could fail systems either directly by wind loading or indirectly by causing a tornado-induced missile to impact and perforate a fluid system. Unlike random pipe failures where only a single system failure is considered at a time, a tornado could affect multiple systems simultaneously, thereby increasing the resulting flood height.

As described above, all systems except the circulating water, fire protection water, service water, and high-energy line breaks (HELBs) were screened from consideration as flooding sources. The systems were screened from consideration either because they contained insufficient inventory to damage equipment outside of the turbine building or because the flow rate that would result from any break would be low enough so that a very long time would be available for operator action to isolate any flooding event prior to equipment damage outside the Turbine Building.

The screening of systems above is still valid with two exceptions; the condensate storage tanks (CSTs) and the reactor makeup storage tanks (RMSTs). When considered individually, the volume for each of these two sources is low enough that a flood which released their contents could not damage enough equipment outside the Turbine Building to seriously impair the ability of the operators to mitigate a reactor trip. Because the two sources are located near each other, a tornado could cause near simultaneous failure of all the tanks.

The primary flood risk in a tornado is due to a failure of the RMSTs and the CSTs in the tank room to the south of the auxiliary building. [CALC07] shows that the RMSTs would fail at lower wind loads than the CSTs. The capacity of each RMST is 40,000 gallons. Although some water could spill to other locations, such as outside, the maximum amount of water released from both RMSTs is 80,000 gallons. As discussed above, at least 185,000 gallons must be deposited in the turbine building basement to result in equipment failures in safeguards alley. Therefore, winds severe enough to fail the RMSTs, but not the CSTs, would not result in a significant risk increase. Since the combined capacity of the CSTs is 150,000 gallons, there is a potential of damage to equipment in safeguards alley due to flooding from the combination of the four tanks.

[CALC08] shows that the frequency of CST damage due to direct tornado impact is 6.7E-7 per year. This reference also includes a discussion of tornado missiles. Specifically, the document states that tornado missiles are not a concern with wind speeds below 212 mph, which corresponds to an exceedance frequency of 7.1E-6 per year. It also points out that most missiles would hit the upper portion of the tank, resulting in less that the full 150,000 gallons being released into the basement. Furthermore, for a missile to puncture the tank, the pipe must strike the tank nearly end-on along a radial line of the tank diameter. Any object that strikes slightly off normal or off the radial line would not be expected to penetrate the tank, but rather would be expect to glance off the tank without perforating it. Of the potential missiles that come within striking distance of the CST, only a fraction of them would be expected to strike the tank in such a manner as to be able to penetrate the tank. Therefore, the frequency of a tornado missile causing a flood of greater than 185,000 gallons of water to enter the turbine building basement is negligible.

Tornado-induced failure of the circulating water system is considered unlikely for several reasons. First, the majority of the piping is located in the basement under the main turbine. The turbine building is designed such that it will not collapse (although the panels may fail) following a tornado so it is unlikely that the piping would be failed directly by the tornado. Secondly, the circulating water pumps are powered from the non-safety buses which require offsite power. It is likely that a tornado severe enough to threaten the circulating water piping would also cause a loss of offsite power, thereby removing the motive force for system flow and stopping the flood. Third, tornado missile-induced failure is unlikely. A tornado missile risk analysis of the Kewaunee Power Station (KPS) was performed using the TORMIS methodology [CALC09]. In that study, the yearly probability of a tornado missile hitting either the diesel oil day tank vents, diesel exhaust stacks, or the turbine-driven auxiliary feedwater pump exhaust pipe is 9.5E-06 per year and the probability of damaging one of the targets 1.7E-06 per year. These values are dominated by the concrete paver blocks located on the Turbine Building roof. Since all the circulating water piping is located below the turbine operating deck and, therefore, protected from such missiles, it is concluded that the tornado missile-induced failure probability is negligible.

The fire protection water header is located entirely in the Turbine Building basement, below grade. Several branch lines do extend to the mezzanine level to deluge valves and other equipment supporting system operation. Once on the mezzanine level, piping size reduces quickly. Only very short lengths of small-diameter piping to hose stations are located on the operating deck. As with the circulating water system, the fire protection water piping would be protected from direct failure in a tornado because of the ability of the Turbine Building to remain standing following such an event. The failure of fire protection water piping by missile impact is considered to be much lower than that calculated in [CALC07] and discussed above. Therefore, it is concluded that the risk from fire protection water flooding events initiated by tornados is negligible.

As with the fire protection water system, the majority of service water piping is located in the Turbine Building basement, below grade. No service piping is located on the operating deck. Service water piping located on the mezzanine level is generally smaller in size, e.g., less than six inches nominal pipe size. Because the Turbine Building is designed to not collapse under tornado winds, direct failure of the service water piping is not expected. Failure of service water piping due to missile impact is considered to be a negligible contribution to risk as discussed above. Also, the turbine header isolation valves would be available to isolate the Turbine Building header following a tornado. Therefore, it is concluded that the risk from service water flooding events initiated by tornados is negligible.

For a tornado to cause a HELB, the event must first expose the Turbine Building to the outside winds. Because the Turbine Building contains blowout panels that are designed to fail, it is likely that the building would be open to the outside winds. The analysis of sprinkler actuation due to HELB [CALC06] shows that Turbine Building temperatures are reduced rapidly once the blowout panels fail. For a tornado-induced HELB, the blowout panels would fail prior to the HELB and the tornado winds would help mitigate any temperature rise caused by steam release. Therefore, the number of sprinklers actuated for any HELB caused by a tornado would be much less than a similar size break initiated internally to the Turbine Building. Also, feedwater, condensate, and steam piping of concern to flooding events is designed for very high pressures and, therefore, much less likely than the diesel exhaust stacks to be damaged by tornado missiles. Therefore, it is concluded that the risk from tornado-induced HELBs that actuate the fire protection system is negligible.

3.3 Identification of Systems as Potential Flooding Sources

For piping in the turbine building, only the service water, circulating water, and fire protection water contain sufficient volume or lines large enough to release fluid to the point that equipment in safeguards alley would be threatened in less than one hour. As described above, all other systems were screened as negligible contributors to flooding risk. Further analysis of these systems as potential flooding initiators is given in the sections that follow.

3.3.1 Service Water Flooding Events

This initiating event will assume that all service water piping in the turbine building is supplied from the 20-inch turbine building header and is downstream of motor-operated valves SW-4A and SW-4B. There is service water piping that is in the turbine building but is not supplied from the turbine building header. Examples include auxiliary feedwater pump room cooler return lines to the standpipe, diesel cooling return lines, and air compressor cooling lines. With the exception of the diesel cooling return lines, piping in the turbine building that is not supplied from the turbine building header is small, e.g., 1.5-inches or less. Any leak from such lines would result in a low flow rate thereby providing the operators with a long time period to isolate the break using manually-operated valves local to the component. The diesel cooling return lines are normally isolated so any break in those lines would not result in a flooding event.

As discussed in Section 3.2, service water lines with a nominal diameter of less than four inches would not release of sufficient water in one hour to threaten enough equipment in safeguards alley that accident mitigation would be significantly impaired. Therefore, only breaks in service water lines four inches or greater are considered as potential initiating events.

3.3.2 Circulating Water Flooding Events

A break from the circulating water system could result in the release of a very large amount of water in a short period of time. Calculations [CALC10] show that rupture of an expansion joint on the circulating water supply lines could be expected to release up to 58,000 gpm of flow. Because the pressure on the return lines is less and because gravitational effects would tend to direct flow to the return header, a break in the circulating water return lines would release less flow to the turbine building. A rupture of an expansion joint on the circulating water return lines could be expected to release up to 14,000 gpm to the Turbine Building basement [CALC10]. Because there is significant difference in the rate of release for the two locations, a large break in each location is considered as a unique initiating event. A break of the piping will be assumed to result in the same flow rate as the largest flow from a rupture of the expansion joint. In addition to the largest break sizes, an expansion joint rupture that results in less than the maximum flow is considered. For circulating water expansion joint ruptures less than the maximum flow, break sizes which lead to ruptures with leak flows between 2,000 and 10,000 gpm are considered.

3.3.3 Fire Protection Water Flooding Events

The flooding event could be caused by an uncontrolled release of water from the fire protection system either because of a random break in the system or as a consequential release caused by a high energy line break (HELB). As discussed in Section 3.2, fire protection water lines with a nominal diameter of less than four inches would not release sufficient water in one hour to threaten equipment in safeguards alley. Therefore, only random breaks in fire protection water lines four inches or greater are considered as potential initiating events.

A HELB could raise temperatures in the Turbine Building to the point that fire protection sprinklers or deluge systems actuate. If a large number of sprinklers actuate, the potential exists to threaten equipment in safeguards alley. Breaks in the feedwater or condensate lines release a large quantity of water to the Turbine Building in addition to actuating fire protection systems. Breaks in the steam systems do not result in an appreciable quantity of water being released to the Turbine Building. Therefore, steam line breaks are considered separately from feedwater and condensate line breaks.

Steam Line Breaks

Analyses show that steam line breaks greater than nine inches equivalent diameter and upstream of the turbine building throttle valves will result in a safety injection (SI) signal [CALC06]. Because a SI signal inhibits operation of the fire pumps [REPORT01], large breaks in the main steam system can be excluded as initiating events. In addition, the same analyses show that steam line breaks on the operating deck of the turbine building and less than nine-inches in diameter will not actuate any fire sprinklers. Therefore, all steam lines on the operating deck can be excluded as initiating events.

For steam line breaks below the operating deck, calculations show that breaks smaller than two inches equivalent diameter actuate no fire protection sprinklers [CALC06], however, for the highest pressure main steam lines, i.e., upstream of the turbine throttle valves, a three-inch equivalent diameter break will actuate enough sprinklers that the fire pumps can be assumed to be providing full flow to the system.

For the extraction steam supply to the 15 feedwater heaters, a four-inch equivalent diameter break would actuate about 100 sprinklers while a six-inch or larger break would actuate enough sprinklers that the fire pumps can be assumed to be providing full flow to the system.

After steam exits the high-pressure turbine, a four-inch equivalent diameter break would actuate no fire protection systems while a six-inch break would actuate about 100 sprinklers.

Based on these results, two initiating events are analyzed for flooding events. The first is a steam line break that actuates enough fire sprinklers to result in full flow from both fire pumps to the Turbine Building. This event includes any break upstream of the turbine throttle valves with an equivalent diameter less than nine inches but greater than two inches, any break in the extraction steam line greater than six inches, and any break in a line after exiting the high-pressure turbine with an equivalent diameter of six inches or greater.

The second event is a steam line break that actuates approximately 100 sprinklers. The Turbine Building HELB models show that 100 sprinklers is representative of moderate releases. This event includes breaks in the extraction steam lines with an equivalent break size between two and six inches, and breaks in a line after exiting the high-pressure turbine and having an equivalent diameter of two to six inches.

Feedwater and Condensate Line Breaks

This event initially considers breaks in any pipe containing main turbine working fluid above saturation conditions and includes all piping from the outlet of second feedwater heaters (12A and 12B). Analyses show that breaks upstream of the fourth feedwater heaters (14A and 14B) do not actuate any fire protection systems [CALC06]. In addition, the volume of water released from such breaks is less than the 185,000 gallons needed to threaten any equipment in safeguards alley. Therefore, all breaks upstream of the fourth feedwater heaters can be excluded from further consideration.

For piping between the 14 and 15 feedwater heaters, breaks smaller than four inches equivalent diameter actuate no sprinklers. A six-inch equivalent diameter break in these lines would actuate about 100 sprinklers and a nine-inch equivalent break would actuate enough sprinklers that the fire pumps can be assumed to be providing full flow to the system.

For piping after the 15 feedwater heaters, a two-inch or smaller equivalent diameter break would actuate no fire protection systems. A four-inch break would actuate enough sprinklers that the fire pumps can be assumed to be providing full flow to the system.

Based on these results, two initiating events are analyzed for flooding events. The first is a feedwater or condensate line break that actuates enough fire sprinklers to result in full flow from both fire pumps to the Turbine Building. This event includes any break between the 14 and 15 feedwater heaters with an equivalent diameter of greater than six inches or any break downstream of the 15 feedwater heaters with an equivalent diameter greater than two inches.

The second event is a feedwater or condensate line break that actuates approximately 100 sprinklers. The Turbine Building HELB models show that 100 sprinklers is representative of moderate releases. This event includes breaks in the lines between the 14 and 15 feedwater heaters with an equivalent diameter between four and six inches.

3.3.4 Summary of Turbine Building Internal Flooding Events

For internal flooding events in the turbine building, nine different initiating events have been defined for further analysis. The first is a break in the service water system in the Turbine Building and having an equivalent diameter of greater than four inches. The second event is a break in the circulating water supply lines. The third is a break in the circulating water return lines. The fourth is a circulating water break between 2,000 and 10,000 gpm. The fifth is a random break in fire protection water piping with the break having an equivalent diameter of greater than four inches. The sixth is a steam line break that actuates enough fire sprinklers to result in full flow from both fire pumps to the Turbine Building. The seventh is a steam line break that actuates approximately 100 fire sprinklers. The eighth is a feedwater or condensate line break that actuates enough fire sprinklers to result in full flow from both fire pumps to the Turbine Building. The ninth is a feedwater or condensate line break that actuates approximately 100 fire sprinklers.

3.4 Quantification of Internal Flooding Initiating Event Frequencies

Quantification of the initiating event frequency for each of the nine events discussed above is performed in the following sections. Described within each section is the source of data used for system break frequency determination and how that data was used to calculate the initiating event frequency.

3.4.1 Service Water-Initiated Flooding Events

To determine the frequency of service water-initiated flooding events, the frequency of pipe breaks is calculated using the methodology presented in EPRI TR 102266, "Pipe Failure Study Update", April 1993 [REPORT02]. Newer data sources that can be used to determine internal flooding initiating event frequency values have recently been published, i.e., EPRI TR 1012302, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments (PRAs)," [REPORT04]. However, service water initiating event frequency values calculated using the data and methodology of [REPORT04] are not expected to be significantly different from those calculated using [REPORT02]. Generally, it is expected that lower initiating event frequency values will result if calculated using [REPORT04] instead of [REPORT02]. In addition, the pipe segment data needed to calculate initiating event frequency values using the methodology of [REPORT02] is already available. A significant effort would be needed to determine the pipe length data needed to employ the methodology of [REPORT04]. In addition, service water-initiated flooding events have been shown in prior, scoping studies to be a small contribution to overall risk from turbine building floods.

Therefore, the frequency of service water-initiated flooding events will be calculated using the methodology presented in [REPORT02].

Using that methodology, pipe breaks are categorized as large, medium, and small. A break in a large pipe will not always be categorized as large. There is a probability that a large pipe will have a break in the medium or small category. Similarly, a medium pipe may have a break in the small category. When determining the frequency of breaks that result in the different categories, the recommended values from [REPORT02] will be used to determine the probability of equivalent break sizes.

The frequency for failure of components such as valves and heat exchangers is calculated using data from Eide, S.A. et al., "Component External Leakage and Rupture Frequency Estimates", EGG-SSRE-9639 [REPORT03]. The following table gives the component rupture frequencies from [REPORT03] that are used in this analysis:

Component Rupture Frequencies

Component Type	**Rupture/Leakage**	**Rate (/hr)**
Valve	Leakage	1.0E-08
	non-PCS Rupture[1]	4.0E-10
	PCS Rupture	1.0E-10
Pump	Leakage	3.0E-08
	non-PCS Rupture	1.2E-09
	PCS Rupture	3.0E-10
Flange	Leakage	1.0E-08
	Rupture	1.0E-10
Heat Exchanger Tube Side	Leakage	1.0E-07
	non-PCS Rupture	4.0E-09
	PCS Rupture	1.0E-09
Heat Exchanger Shell Side	Leakage	1.0E-08
	non-PCS Rupture	4.0E-10
	PCS Rupture	1.0E-10
Tank	Leakage	1.0E-08
	non-PCS Rupture	4.0E-10
	PCS Rupture	1.0E-10

[1] PCS = Primary Cooling System

It was assumed that the rupture of valves, pump casings, and other components have the same conditional probability of small, medium, large ruptures as for piping.

The initiating event frequency for service water-initiated flooding events in the turbine building will consider breaks in all pipes with a nominal size greater than four inches. Service water pipes and components are tabulated by size in [NB01]. As shown in Appendix F of [NB01], service water piping in the turbine is either four inches or smaller or six inches or greater. Twenty-seven pipe segments and nine valves were identified in the six-inch-or-larger category.

It will be assumed that large-bore piping breaks with an equivalent break diameter in the medium (two-to-six-inch) category are not large enough to be of concern because breaks that size in large-bore piping have a sufficiently low flow rate to allow more time for recovery and, therefore, are not included in the total frequency of service water flooding events. Therefore, the frequency of large service water initiated flooding events in the Turbine Building was calculated to be:

$F_{SW} = F_{SWPipe} + F_{SWValve}$

F_{SW} = ((27 pipe segments) * (1.39E-10 / pipe segment-hour) + (9 valves) * (4.0E-10 / valve-hour)) * 0.5 conditional probability of a large break [REPORT02]

F_{SW} = (3.75E-09 / hour + 3.6E-09 / hour) * 0.5

F_{SW} = 3.78E-09 / hour

F_{SW} = 3.22E-05 per year.

The contribution of maintenance-induced flooding events is considered negligible for several reasons. First, the maintenance event must be such that the event breaches the service water system pressure boundary but still permits operation of the plant and the turbine building header. Actions such as cleaning heat exchanger water boxes could be performed. However, most valves in the systems could not be breached without securing the entire header. Therefore, the frequency of maintenance events is expected to be small. Second, the isolation valves for the service water-cooled heat exchangers are all manual valves located near the component being serviced. Should a breach of an unisolated component occur, the maintenance personnel would be able to quickly isolate the leak.

3.4.2 Large Circulating Water Inlet Line-Initiated Flooding Events

Large flooding events from the circulating water system inlet lines could occur due to three causes, failure of the expansion joints, rupture of the piping and components in the system, or maintenance errors. The frequency of large failures, i.e., greater than 10,000 gpm, of expansion joints is documented in Attachment 2, which provides a failure frequency of 6.08E-06 per year per expansion joint. With four inlet expansion joints, the total frequency of expansion joint failures is calculated to be:

$F_{CWINExp}$ = 2.43E-05 per year.

As with service water-initiated flooding events, the frequency of system breaks (excluding expansion joint breaks) is calculated using the methodology of [REPORT02]. Use of this methodology over the newer methodology recently published in [REPORT04] is judged to be acceptable for the same reasons explained in Section 3.4.1.

Circulating water pipes and components are tabulated in [NB01]. As shown in Appendix F of [NB01], circulating water inlet piping contains ten pipe segments and four valves. Therefore, the frequency of large circulating water inlet-initiated pipe rupture events was calculated to be:

$F_{CWINPipe} = F_{CWPINipe} + F_{CWINValve}$

$F_{CWINPipe}$ = ((10 pipe segments) * (1.39E-10 / pipe segment-hour) + (4 valves) * (4.0E-10 / valve-hour)) * 0.5 conditional probability of a large break [REPORT02]

$F_{CWINPipe}$ = (1.39E-09 / hour + 1.60E-09 / hour) * 0.5

$F_{CWINPipe}$ = 1.49E-09 / hour

$F_{CWINPipe}$ = 1.31E-05 per year.

A flooding event could be initiated during maintenance operations if the following conditions exist or events occur. First, operation of at least one circulating water pump must continue through the maintenance event. This would be expected for power operations. Second, the circulating water system pressure boundary must be breached. A breach would be expected for events such as cleaning water boxes. Third, a failure must occur so as to breach the isolation boundary from the circulating water inlet header to the maintenance opening. Isolation failures are described in more detail below.

Only breaks greater than six inches equivalent diameter are considered because the circulating water system operates at a very low pressure and the flow rate from breaks less than six inches would be expected to allow a significant time period for operators to isolate the break. The only isolation failures that would be of concern are the condenser inlet isolation valves. These motor-operated valves are controlled from local push button stations. During the maintenance event, the valve would be closed, the breaker opened, and then the open breaker and valve hand wheel would be danger tagged. In addition, the push button station would be caution tagged. Therefore, inadvertent opening of the valve would require that the danger tags be disregarded. Then the valve must be manually opened sufficiently to allow flow to endanger turbine building equipment. Since the valves are located just below the water box inlets, it is unlikely that an operator would open a valve without noticing that water was being released. Similarly, if maintenance is attempted on an unisolated water box, then the operators would be expected to notice flow from the system as the pressure boundary is being unbolted. When leakage occurs, the operators can be expected to secure the area and investigate. Random failures of the valve disk are considered negligible. Therefore, flooding events initiated by maintenance on the circulating water system are considered negligible contributors to the overall initiating event frequency and are neglected.

The total frequency of large breaks in the circulating system inlet piping is the sum of the frequency of expansion joint ruptures and the frequency of large pipe ruptures, or,

$F_{CWIN} = F_{CWINExp} + F_{CWINPipe}$.

F_{CWIN} = 2.43E-05 per year + 1.31E-05 per year

F_{CWIN} = 3.74E-05 per year.

3.4.3 Large Circulating Water Outlet Line-Initiated Flooding Events

Flooding from the circulating water system outlet lines could occur due to three causes, failure of the expansion joints, rupture of the piping in the system, or maintenance errors. Failure of expansion joints used the information from Attachment 2 that provided a failure frequency of 6.08E-06 per year per expansion joint for failures with flow greater than 10,000 gpm. With four outlet expansion joints, the total frequency of expansion joint failures is calculated to be:

$F_{CWOUTExp}$ = 2.43E-05 per year.

As with service water-initiated flooding events, the frequency of system breaks (excluding expansion joint breaks) is calculated using the methodology of [REPORT02]. Use of this methodology over the newer methodology recently published in [REPORT04] is judged to be acceptable for the same reasons explained in Section 3.4.1.

Circulating water pipes and components are tabulated in [NB01]. As shown in Appendix F of [NB01], circulating water outlet piping contains eight pipe segments but no components other than the expansion joints discussed above. Therefore, the frequency of large circulating water outlet-initiated pipe rupture events was calculated to be:

$F_{CWOUTPipe}$ = ((8 pipe segments) * (1.39E-10 / pipe segment-hour)) * 0.5 conditional probability of a large break [REPORT02]

$F_{CWOUTPipe}$ = (1.11E-09 / hour) * 0.5

$F_{CWINPipe}$ = 4.87E-06 per year.

A flooding event could be initiated during maintenance operations if the following conditions exist or events occur. First, operation of at least one circulating water pump must continue through the maintenance event. This would be expected for power operations. Second, the circulating water system pressure boundary must be breached. A breach would be expected for events such as cleaning water boxes. Third, a failure must occur so as to breach the isolation boundary from the circulating water inlet header to the maintenance opening. Isolation failures are described in more detail below.

Only breaks greater than six inches equivalent diameter are considered because the circulating water system operates at a very low pressure and the flow rate from breaks less than six inches would be expected to allow a significant time period for operators to isolate the break. The only isolation failures that would be of concern are the condenser inlet isolation valves. These motor-operated valves are controlled from local push button stations. During the maintenance event, the valve would be closed, the breaker opened, and then the open breaker and valve hand wheel would be danger tagged. In addition, the push button station would be caution tagged. Therefore, inadvertent opening of the valve would require that the danger tags be disregarded. Then the valve must be manually opened sufficiently to allow flow to endanger turbine building equipment. Since the valves are located just below the water box inlets, it is unlikely that an operator would open a valve without noticing that water was being released. Similarly, if maintenance is attempted on an unisolated water box, then the operators would be expected to notice flow from the system as the pressure boundary is being unbolted. When leakage occurs, the operators can be expected to secure the area and investigate. Random failures of the valve disk are considered negligible. Therefore, flooding events initiated by maintenance on the circulating water system are considered negligible contributors to the overall initiating event frequency and are neglected.

The total frequency of large breaks in the circulating system outlet piping is the sum of the frequency of expansion joint ruptures and the frequency of large pipe ruptures, or,

$F_{CWOUT} = F_{CWOUTExp} + F_{CWOUTPipe}$.

F_{CWOUT} = 2.43E-05 per year + 4.87E-06 per year

F_{CWOUT} = 2.92E-05 per year.

3.4.4 Small Circulating Water Expansion Joint Flooding Events

Flooding from the circulating water system could result in break flow rates less than the maximum used flow described above. Such events could occur in either the inlet or outlet lines. Because all pipe breaks are assumed to result in the maximum flow and the pipe break frequency is included in the first two circulating water events, pipe breaks are not considered in this event. Therefore, this event considers only failures of the circulating water expansion joints that result in less than the maximum flow, which for this analysis is between 2000 and 10,000 gpm. (A 6000-gpm break was deemed to be most representative of a small rupture.) The frequency for such events is documented in Attachment 2 which provides a failure frequency of 9.17E-06 per year per expansion joint. With four inlet expansion joints and four outlet expansion joints, the total frequency of expansion joint failures is calculated to be:

F_{CWExp} = 7.34E-05 per year.

3.4.4 Random Breaks in Fire Protection Water Piping

As with service water-initiated flooding events, the frequency of system breaks is calculated using the methodology of [REPORT02]. Use of this methodology over the newer methodology recently published in [REPORT04] is judged to be acceptable for the same reasons explained in Section 3.4.1.

The initiating event frequency for random breaks in the fire protection water system considers breaks in all pipes with a nominal size greater than four inches. Piping drawings for the fire protection water system were reviewed and piping and components that are located in the turbine building and that cause a flooding event of concern were tabulated by pipe size. The piping tabulation in Addendum 2 identified 40 piping segments, 20 valves, and 26 flanges with a nominal size greater than four inches. Assuming that fire protection water piping is classified in the "other safety related" category used in [REPORT02], the frequency of fire protection water-initiated flooding events is calculated to be:

$F_{FPR} = F_{\mathbf{FPPipe}} + F_{\mathbf{FPValve}} + F_{\mathbf{FPFlange}}$

F_{FPR} = ((40 pipe segments) * (1.39E-10 / pipe segment-hour) + (20 valves) * (4.0E-10 / valve-hour) + (26 Flanges) * (1.0E-10 / flange-hour)) * 0.5 conditional probability of a large break [REPORT02]

F_{FPR} = (5.56E-09 / hour + 8.00E-09 / hour + 2.6E-09 / hour) * 0.5

F_{FPR} = 8.08E-09 / hour

F_{FPR} = 7.08E-05 per year.

It will be assumed that large-bore piping breaks with an equivalent break diameter in the two-to-six-inch category are not large enough to be of concern because breaks that size in large-bore piping have a sufficiently low flow rate to allow more time for recovery and are not included in the total frequency of fire protection water flooding events.

The contribution of maintenance-induced flooding events is considered negligible for several reasons. First, the maintenance event must be such that the event breaches the fire protection water system pressure boundary but still allows the system to be pressurized. There are very few large-bore components that would permit such maintenance. Potentially, certain deluge valves could be breached. Next, the maintenance must be such that the breach would allow flooding to continue undetected for a significant time period following any operator error that resulted in an inadvertent breach. For deluge valves in the fire protection system, their associated isolation valve is immediately adjacent to the valve. Therefore, should a breach of an unisolated component occur, the maintenance personnel would be able to quickly isolate the leak. For these reasons, the frequency of maintenance events to the fire protection flooding initiating event frequency is considered negligible.

3.4.5 Steam Line Breaks Causing Large Fire Protection System Actuations

The first step in determining the frequency of steam line breaks that cause large fire protection system actuations is to determine the length and location of the steam pipes of concern. Piping layout drawings were reviewed and the dimensions indicated on them were used to determine the length of steam pipes that are of concern to turbine building flooding events. Details of the pipe length data are listed in Addendum 3. Summing the lengths of high-pressure main steam piping located on the mezzanine and basement levels gives a total of 884.6 linear feet of piping. Summing the lengths of extraction steam piping located on the mezzanine and basement levels gives a total of 176.5 linear feet of piping. Summing the lengths of lower-pressure steam piping located on the mezzanine and basement levels gives a total of 621.7 linear feet of piping. All other steam piping was located either on the operating deck or in the Auxiliary Building. (Note all of the steam piping tabulated is at least 6-inch diameter, and therefore of sufficient size to have the break flow required for a large fire protection actuation.)

Because not all high-energy line breaks would result in a turbine building flooding event, a separate analysis was performed to determine the frequency for steam line breaks of interest. This separate analysis uses the data of [REPORT04] and is documented in [REPORT05]. Refer to [REPORT05], which is included as Attachment 3, for details of the calculations. From that analysis, the frequency of steam line breaks (including failures of valves, flanges, etc.) that result in large fire protection system actuations, F_{SLBL}, is:

F_{SLBL} = **2.53E-04 per year**

3.4.6 Steam Line Breaks Causing Intermediate Fire Protection System Actuations

Calculation of the frequency of this event is performed in [REPORT05] using the pipe length data described in Section 3.4.5 for large steam line breaks. From [REPORT05], the frequency of steam line breaks (including failures of valves, flanges, etc.) that result in intermediate fire protection system actuations, F_{SLBM}, is:

F_{SLBM} = **1.87E-05 per year**

3.4.7 Feedwater and Condensate Line Breaks Causing Large Fire Protection System Actuations

As discussed in Section 3.3.3, this event includes any break with an equivalent diameter greater than two inches in piping downstream of the 15 feedwater heaters and any break with an equivalent diameter greater than six inches between the 14 and 15 feedwater heaters.

The first step in determining the frequency of feedwater and condensate line breaks that cause large fire protection system actuations is to determine the length and location of the pipes of concern. Piping layout drawings were reviewed and the dimensions indicated on them were used to determine the length of feedwater and condensate pipes that are of concern to turbine building flooding events. Details of the pipe length data are listed in Addendum 3. Summing the lengths of feedwater piping downstream of the 15 feedwater heaters gives a total of 331.56 linear feet of piping. Summing the lengths of feedwater and condensate piping located between the 14 and 15 feedwater heaters gives a total of 696.65 linear feet of piping. (Note all of the feedwater and condensate piping tabulated is at least 12-inch diameter, and therefore of sufficient size to have the break flow required for a large fire protection actuation.)

Because not all high-energy line breaks would result in a turbine building flooding event, a separate analysis was performed to determine the frequency for feedwater and condensate line breaks of interest. This separate analysis uses the data of [REPORT04] and is documented in [REPORT05]. Refer to [REPORT05], which is included as Attachment 3, for details of the calculations. From that analysis, the frequency of feedwater and condensate line breaks (including failures of valves, pumps, heat exchangers, etc.) that result in large fire protection system actuations, F_{FLBL}, is:

F_{FLBL} = 1.35E-04 per year

3.4.8 Feedwater and Condensate Line Breaks Causing Intermediate Fire Protection System Actuations

As discussed in Section 3.3.3, this event includes any break with an equivalent diameter between four and six inches between the 14 and 15 feedwater heaters. Identification and tabulation of the pipe lengths is described above.

Because not all high-energy line breaks would result in a turbine building flooding event, a separate analysis was performed to determine the frequency for feedwater and condensate line breaks of interest. This separate analysis uses the data of [REPORT04] and is documented in [REPORT05]. Refer to [REPORT05], which is included as Attachment 3, for details of the calculations. From that analysis, the frequency of feedwater and condensate line breaks (including failures of valves, pumps, heat exchangers, etc.) that result in large fire protection system actuations, F_{FLBM}, is:

F_{FLBM} = 4.69E-05 per year

4.0 SUMMARY

For the analysis of internal flooding caused by pipe and component failures in the turbine building that potential threaten equipment in safeguards alley, nine initiating events have been identified and their associated frequency values quantified. These events are summarized in the table below.

Event	Consequence	Frequency (per year)
Random Service Water Break	Releases a large flow of Service Water to the Turbine Building	3.22E-05
Large Circulating Water Inlet Piping Break	Releases 58,000 gallons per minute to the Turbine Building	3.74E-05
Large Circulating Water Outlet Piping Break	Releases 14,000 gallons per minute to the Turbine Building	2.92E-05
Small Circulating Water Expansion Joint Failure	Releases 6,000 gallons per minute to the Turbine Building	7.34E-05
Random Fire Protection Water Break	Releases full flow from both fire water pumps to the Turbine Building	7.08E-05
Large Steam Line Break	Actuates enough fire sprinklers that full fire protection water flow is released to the Turbine Building	2.53E-04
Intermediate Steam Line Break	Actuates 100 fire sprinklers that release fire protection water flow is released to the Turbine Building	1.87E-05
Large Feedwater or Condensate Line Break	Actuates enough fire sprinklers that full fire protection water flow is released to the Turbine Building	1.35E-04
Intermediate Feedwater or Condensate Line Break	Actuates 100 fire sprinklers that release fire protection water flow is released to the Turbine Building	4.69E-05

5.0 REFERENCES

[CALC01] X-K-204-1696, Revised 4/5/1972.

[CALC02] MPR Calculation 064-0515-LYS-01, "Evaluation of Flooding Levels for Various

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[CALC03] MPR Report 2823, "Past Operability Evaluation of Electrical Equipment Due To Flooding in Kewaunee Nuclear Power Plant," Revision 2.

[CALC05] "Results of Probability Analyses of Disc Rupture and Missile Generation," CT-24108, Revision 0, August 1980, Westinghouse Electric Corporation.

[CALC06] MPR Calculation 0064-0509-DEC-03, Revision 0, "Turbine Building HELB Fire Protection Actuation Analysis."

[CALC07] Stevenson & Associates Calculation CAL-002: Evaluation of CST and RMST Wind Load.

[CALC08] PRA Application 05-13: Core Damage Frequency due to Tornadoes Damaging Condensate Storage Tanks

[CALC09] "Tornado Missile Risk Analysis for Kewaunee Nuclear Plant," July 2005, Applied Research Associates, Inc.

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[PROC02] A-MDS-30, Miscellaneous Drains and Sumps Abnormal Operation," Revision N, 3/5/2002.

[REPORT01] KPS System Description, "Fire Protection System," Rev. 2.

[REPORT02] Electric Power Research Institute, EPRI TR 102266, "Pipe Failure Study Update," April 1993.

[REPORT03] Eide, S.A. et al., 1991, "Component External Leakage and Rupture Frequency Estimates", EGG-SSRE-9639, INEL, November 1991.

[REPORT04] Electric Power Research Institute, EPRI TR 1012302, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments (PRAs)," September, 2005.

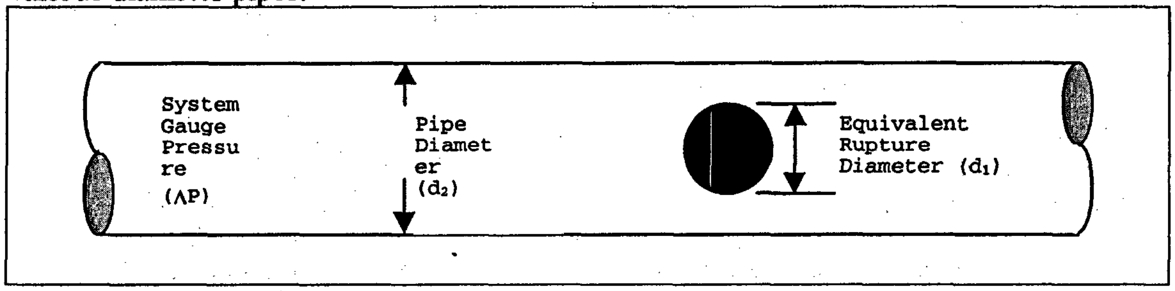
[REPORT05] Fleming, K.N. et al, "High Energy Line Break Initiating Event Frequencies for the Kewaunee PRA", October 2005.

[REPORT06] KPS System Description, "Service Water System," Rev. 3.

[REPORT07] Fleming, K. N. et al., "Expansion Joint Failure Rates for the Kewaunee PRA", Final Report, November 2005.

ADDENDUM 1, SERVICE WATER AND FIRE PROTECTION SYSTEM PIPING LEAK RATE CALCULATION

Infinite flood sources, such as Service Water and Fire Protection system, have been analyzed to determine the equivalent size of a pipe rupture that will potentially overwhelm the drainage capacity of a designated flood area. The analysis was performed using an Excel spreadsheet that calculates the flow equations listed below to determine flow rates from various rupture sizes in various diameter pipes.



Calculating the volumetric flow rate can be done by applying the following equation:

$$q_{ft^3/sec} = C * A * \sqrt{\frac{2g * 144 * \Delta P}{\rho}} \qquad \text{[MAN01], Eqn. 2.23}$$

or expressed in Gallons per Minute (GPM)

$$Q_{GPM} = \left(448.8 \frac{GPM}{ft^3/Sec}\right) * (C * A) * \sqrt{\frac{2g * 144 * \Delta P}{\rho}}$$

Where:

ΔP = System Gauge Pressure (psig)
A = Equivalent Rupture Area (ft^2)
C = Flow Coefficient (dimensionless)
ρ = Density of Water (lb/ft^3)
g = Gravity (32.17 ft/sec^2)

The Flow Coefficient (C) for an orifice is calculated using the equation

$$C = \frac{C_d}{\sqrt{(1 - \beta^4)}} \qquad \text{[MAN01], Page A-20}$$

As stated in [MAN01], Table 3.10, the Discharge Coefficient (C_d) for a sharp-edged orifice is

0.62.

The ratio of small to large diameter in an orifice (β) is defined as:

$$\beta = \frac{(d_1)}{(d_2)}$$ [MAN01], Page A-20

For calculation of the flow rates from a ruptured Service Water System, the following constants are used:

Piping Inside Diameters
1" Standard Schedule 40 Pipe: 1.049" [MAN01], Page B-16
2" Standard Schedule 40 Pipe: 2.067" [MAN01], Page B-16
4" Standard Schedule 40 Pipe: 4.026" [MAN01], Page B-16
6" Standard Schedule 40 Pipe: 6.065" [MAN01], Page B-16

Pressure
Normal Service Water System Pressure (ΔP): 90-100 psig [REPORT06]

Density
Water Density at 54°F: 62.39 lb/ft3 [MAN01], Page A-6
Water Density at 74°F*: 62.27 lb/ft3 [MAN01], Page A-6

For calculation of the flow rates from a ruptured Fire Protection System, the following constants are used:

Pressure
Fire Protection System Pressure (ΔP) (standby): 128-143 psig [REPORT01]

Density
Water Density at 85°F: 62.17 lb/ft3 [MAN01], Page A-6

The table below shows the resultant flow rates for various rupture sizes in pipes with diameters 1 inch, 2 inch, 4 inch, and 6 inch for the service water and fire protection system pipes. The calculations used to develop the table used a pressure of 108 psig which is representative of the average pressure expected in both the service water and fire protection water systems. Since the flowrate is a function of the square route of the pressure, any differences on pressure have a minor impact on the overall results.

Table A1-1: Fire Protection System Piping Rupture Flow Rates (Pressure = 108 psig @ 85°F)

Pipe Inside Diameter (in)	Pipe Cross-Sectional Area (ft^2)	Equivalent Rupture Diameter (in)	Equivalent Rupture Area (ft^2)	Diameter Ratio Beta Factor (β)	Flow Coefficient (C)	Rupture Flow Rate (q) (ft^3/sec)	Rupture Flow Rate Q (GPM)
1 (ID = 1.049)	0.0060	0.50	0.0014	0.4766	0.6366	0.1101	49.4270
		0.75	0.0031	0.7150	0.7214	0.2808	126.0101
2 (ID = 2.067)	0.0233	0.50	0.0014	0.2419	0.6211	0.1074	48.2171
		0.75	0.0031	0.3628	0.6254	0.2434	109.2536
		0.90	0.0044	0.4354	0.6315	0.3539	158.8363
		1.00	0.0055	0.4838	0.6377	0.4413	198.0387
		1.50	0.0123	0.7257	0.7293	1.1355	509.6005
4 (ID = 4.026)	0.0884	0.50	0.0014	0.1242	0.6201	0.1073	48.1402
		0.75	0.0031	0.1863	0.6204	0.2415	108.3678
		0.90	0.0044	0.2235	0.6208	0.3479	156.1508
		1.00	0.0055	0.2484	0.6212	0.4298	192.9053
		1.50	0.0123	0.3726	0.6261	0.9747	437.4454
		2.00	0.0218	0.4968	0.6398	1.7708	794.7316
		3.00	0.0491	0.7452	0.7455	4.6425	2083.5477
6 (ID = 6.065)	0.2006	0.50	0.0014	0.0824	0.6200	0.1073	48.1356
		0.75	0.0031	0.1237	0.6201	0.2413	108.3152
		0.90	0.0044	0.1484	0.6202	0.3476	155.9935
		1.00	0.0055	0.1649	0.6202	0.4292	192.6091
		1.50	0.0123	0.2473	0.6212	0.9671	434.0229
		2.00	0.0218	0.3298	0.6237	1.7263	774.7457
		3.00	0.0491	0.4946	0.6394	3.9821	1787.1590
		4.00	0.0873	0.6595	0.6885	7.6230	3421.2024
		5.00	0.1364	0.8244	0.8452	14.6210	6561.8827

ADDENDUM 2, FIRE PROTECTION PIPE SEGMENT TABULATION

Table A2-1: Turbine Building Fire Protection Water Piping

		ID => 6"				2" <= ID < 6"				0.5" <= ID < 2"			
		Large Pipe				Medium Pipe				Small Pipe			
Piping ID	Diameter	From	To	Valves	Flanges	From	To	Valves	Flanges	From	To	Valves	Flanges
TU-1	10	Wall	T to FP 5-3										
TU-2	10	T to FP 5-3	T to FP 5-2										
TU-3	10	T to FP 5-2	T to FP 5-4										
TU-4	10	T to FP 5-4	T to FP 28-2										
TU-5	10	T to FP 28-2	T to FP 5-5										
TU-6	10	T to FP 5-5	T to FP 5-6										
TU-7	10	T to FP 5-6	FP 1-1										
TU-8	10	T at FP 1-1 North to Turbine Building Wall	T to FP 15-1										
TU-9	10	T at FP 1-1 South to T at FP 22-1	T at FP 22-1										
TU-10	10	T to FP 22-1	T to FP 5-11										
TU-11	10	T to FP 5-11	T to FP 28-1										
TU-12	10	T to FP 28-1	T to FP 3-6										
TU-13	10	T to FP 3-6	T to FP 3-5										
TU-14	10	T to FP 3-5	T to FP 3-4										
TU-15	10	T to FP 3-4	T to FP 3-3										
TU-16	10	T to FP 3-3	T to FP 3-2										
TU-17	10	T to FP 3-2	10 to 6-inch reducer to FP 3-1										
TU-18	2.5					T to FP 5-3	FP 5-3	1	2				
TU-19	2.5					FP 5-3	T to hose connection lines						
TU-20	1.5									T to hose connection lines	Up to FP 90-14	1	
TU-21	1.5									T to hose connection lines	down to FP 90-7	1	
TU-22	2.5					T to FP 5-2	FP 5-2	1	2				
TU-23	2.5					FP 5-2	T to FP 90-4						
TU-24	1.5									T to FP 90-4	Down to FP 90-4	1	
TU-25	2.5					T to FP 90-4	Up to T to FP 90-13						
TU-26	1.5									T to FP 90-13	FP 90-13	1	
TU-27	1.5									T to FP 90-13	Up to FP 90-3		
TU-28	2.5					T to FP 5-4	FP 5-4	1	2				
TU-29	2.5					FP 5-4	Strainer and Deluge Valve	2	4				
TU-30	8	T to FP 28-2	FP 28-2	1	2								
TU-31	4					FP 28-2	FP 56-1						

Table A2-1: Turbine Building Fire Protection Water Piping

Piping ID	Diameter	ID => 6"				2" <= ID < 6"				0.5" <= ID < 2"			
		Large Pipe				Medium Pipe				Small Pipe			
		From	To	Valves	Flanges	From	To	Valves	Flanges	From	To	Valves	Flanges
TU-32	4					FP 56-1	Wall to TSC						
TU-33	8	FP 28-2	T to Mezzanine Sprinkler Isolation Valve										
TU-34	6	T to Mezzanine Sprinkler Isolation Valve	Basement Sprinkler Isolation Valve										
TU-35	2.5					T to FP 5-5	FP 5-5	1	2				
TU-36	2.5					FP 5-5	Reducing T						
TU-37	1.5									Reducing T	FP 90-9	1	
TU-38	2.5					Reducing T	FP 90-15	1					
TU-39	1.5									FP 90-15	FP-91-5	1	
TU-40	2.5					T to FP 5-6	FP 5-6	1	2				
TU-41	2.5					FP 5-6	T to FP 90-10						
TU-42	2.5					T to FP 90-10	T to FP 90-16	1					
TU-43	1.5									T to FP 90-16	Hose Station 21		
TU-44	1.5									T to FP 90-10	FP 90-10		
TU-45	2									T to FP 22-1	FP 22-1	1	
TU-46	2.5					T to FP 5-11	FP 5-11	1	2				
TU-47	2.5					FP 5-11	T to Hose Station 10						
TU-48	1.5									T to Hose Station 10	Hose Station 10	1	
TU-49	1.5									T to Hose Station 10	Hose Station 16	1	
TU-50	1.5									FP 5-11	Hose Station 1	1	
TU-51	8	T to FP 28-1	FP 28-1	1	2								
TU-52	8	FP 28-1	T to Sprinkler Branch Lines										
TU-53	6	T to Sprinkler Branch Lines	Basement Sprinkler Isolation Valve										
TU-54	6	T to FP 3-6	FP 19-6	2	2								
TU-55	6	T to FP 3-5	FP 19-5	2	2								
TU-56	6	T to FP 3-4	FP 19-4	2	2								
TU-57	6	T to FP 3-3	FP 19-3	2	2								
TU-58	6	T to FP 3-2	FP 19-2	2	2								
TU-59	6	Elbow to FP 3-1	FP 19-1	2	2								
TU-60	2.5					T to FP 5-12	FP 5-12	1	2				
TU-61	2.5					FP 5-12	T to FP 90-2						

Table A2-1: Turbine Building Fire Protection Water Piping

		ID => 6"				2" <= ID < 6"				0.5" <= ID < 2"			
		Large Pipe				Medium Pipe				Small Pipe			
Piping ID	Diameter	From	To	Valves	Flanges	From	To	Valves	Flanges	From	To	Valves	Flanges
TU-62	2.5					T to FP 90-2	T to FP 90-12 and FP 90-17						
TU-63	1.5									T to FP 90-12 and FP 90-17	FP 90-12	1	
TU-64	1.5									T to FP 90-2	FP 90-2	1	
TU-65	6									T to FP 90-17	FP 90-17	1	
TU-66	6	T to Mezzanine Sprinkler Isolation Valve Line TU-33)	T at column 8 that splits to 3-inch header and 5-inch header	2	3								
TU-67	5	T at column 8 that splits to 3-inch header and 5-inch header	Riser										
TU-68	5	Riser	T and Riser labeled Q										
TU-69	5	T and Riser labeled Q	T and riser to branch 317										
TU-70	5	T and riser to branch 317	Riser labeled K										
TU-71	5	Riser labeled K	Riser labeled G										
TU-72	6	Basement Sprinkler Isolation Valve	T to lines 21 and 15	2	3								
TU-73	6	T to lines 21 and 15	T to Riser labeled J										
TU-74	6	Basement Sprinkler Isolation Valve (line TU-53)	T to riser labeled S	2	3								
TU-75	10	T to FP 15-1	T to FP 5-10										
TU-76	10	T to FP 5-10	Wall		1								
TU-77	3					T to FP 5-10	Wall	1	4				
TU-78	2.5					T to FP 5-10	FP 23-1	3	4				
78	Totals	40		20	26	22		15	26	16		13	0

ADDENDUM 3, HIGH-ENERGY LINE PIPE LENGTH TABULATIONS

Table A3-1: High-Pressure Main Steam Piping

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
MAIN STEAM PIPING LOCATED IN THE AUXILIARY BUILDING (ALL LEVELS)								
"A" Train								
Aux	M-238	622'-0"	30	A3 - "1A" Train MS Piping from MSIV MS1A to 90° Bend		Horiz.	7.63	5N-6S, N-M
Aux	M-238	622'-0"	30	A3 - "1A" Train MS Piping from 90° Bend to 90° Bend		Horiz.	16.52	6S, M
Aux	M-238, -240	622'-0" to 639'-6"	30	A3 - "1A" Train MS Piping from 90° Bend to Floor Penetration at 639'-6"	"15-15"	Vert.	17.50	6S, M
Aux	M-238	639'-6" to 664'-11 7/16"	30	A3 - "1A" Train MS Piping from Floor Penetration at 639'-6" to 90° Bend		Vert.	27.45	6S, M
Aux	M-238	664'-11 7/16"	30	A3 - "1A" Train MS Piping from 90° Bend to 40° Bend		Horiz.	29.43	6S-6, M-L
Aux	M-238	664'-4 3/4"	30	B4 - "1A" Train MS Piping from 40° Bend to 90° Bend		Horiz.	98.43	6, L-H
Aux	M-238	664'-4 3/4"	30	E4 - "1A" Train MS Piping from 90° Bend to 90° Bend		Horiz.	34.52	6-7, H
Aux	M-238, -240	664'-3" to 652'-11 3/4"	30	E4 - "1A" Train MS Piping from 90° Bend to 90° Bend	"16-16"	Vert.	11.27	7, H
Aux	M-238, -240	652'-11 3/4"	30	E5 - "1A" Train MS Piping from 90° Bend to Turbine Building Wall Penetration (Oper. Deck Level)	"16-16"	Horiz.	58.99	7, H-G
"B" Train								
Aux	M-238	620'-0"	30	F2 - "1B" Train MS Piping from MSIV MS1B to 90° Bend		Horiz.	35.92	4, HE-G
Aux	M-238	620'-0"	30	G2 - "1B" Train MS Piping from 90° Bend to 90° Bend		Horiz.	31.46	4-5, G
Aux	M-238	620'-0"	30	G2 - "1B" Train MS Piping from 90° Bend to Turbine Building Wall Penetration (Mez. Level)		Horiz.	5.49	5, G
				Linear FT on 622'+ Level			**277.59**	
				Linear FT on 622' - Level			**97.02**	
				Linear FT on Basement Level			**0.00**	
				Total Length (Linear FT)			**374.61**	

Table A3-1: High-Pressure Main Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
MAIN STEAM PIPING LOCATED IN THE TURBINE BUILDING OPERATING DECK AND MEZZANINE LEVELS (30 INCH PIPE)								
"A" Train								
TB	M-984-1	Oper Deck	30	E8 - "1A" Train MS Piping Through TB Wall (Approx 660' elev.) 90° Elbow Through Pipe Chase to TB Mez.		Vert.	28.19	7-8, G-F
TB	M-984-1	Mez.	30	F8 - "1A" Train MS Piping From Oper. Deck Opening Thru Mez. 90° Elbow to 90° Towards HP Turbine		Horiz.	73.04	7-8, G-C
TB	M-984-1	Mez.	30	H6 - "1A" Train MS Piping From Mez. 90° Elbow to Oper. Deck. Penetration Towards HP Turbine		Vert.	9.42	7-8, D-C
TB	M-984-1	Mez.	30	H6 - "1A" Train MS Piping From Oper. Deck. Penetration Towards HP Turbine to 90° Elbow		Vert.	4.75	7-8, D-C
TB	M-984-1	Oper Deck	30	H6 - "1A" Train MS Piping From 90° Elbow at Oper. Deck. Penetration Towards HP Turbine Stop Valve Inlet Connection		Horiz.	16.06	7-8, D-C
TB	X-K-101-30	Oper Deck	30	"1A" MS Piping From Valve MS-3A to Oper. Deck Floor Penetration (U-PIPE)		Vert.	4.06	6-7,D-C
TB	X-K-101-30	Mez.	30	"1A" MS Piping From Oper. Deck Floor Penetration to 90° Elbow		Vert.	19.50	6-7,D-C
TB	X-K-101-30	Mez.	30	"1A" MS Piping From 90° Elbow to 90° Elbow		Horiz.	5.75	6-7,D-C
TB	X-K-101-30	Mez.	30	"1A" MS Piping From 90° Elbow to HP Turbine		Vert.	13.30	6-7,D-C
"B" Train								
TB	M-985-1, -2	Mez.	30	D8/D9 - "1B" Train MS Piping Thru TB Wall into Mez. Level to 90° Elbow to Oper. Deck		Horiz.	117.08	5-8, G-D
TB	M-985-2	Mez.	30	D1 - "1B" Train MS Piping From 90° Elbow in the Mez. to Oper. Deck Penetration		Vert.	4.00	7-8, E-D

Table A3-1: High-Pressure Main Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-985-2	Oper Deck	30	C7 - "1B" Train MS Piping From Oper. Deck Penetration to 90° Elbow		Vert.	10.33	7-8, E-D
TB	M-985-2	Oper Deck	30	C7 - "1B" Train MS Piping From 90° Elbow at Oper. Deck. Penetration Towards HP Turbine Stop Valve Inlet Connection		Horiz.	13.69	7-8, E-D
TB	X-K-101-30	Oper Deck	30	"1B" MS Piping From Valve MS-3A to Oper. Deck Floor Penetration (U-PIPE)		Vert.	4.06	6-7,E-D
TB	X-K-101-30	Mez.	30	"1B" MS Piping From Oper. Deck Floor Penetration to 90° Elbow		Vert.	19.50	6-7,E-D
TB	X-K-101-30	Mez.	30	"1B" MS Piping From 90° Elbow to 90° Elbow		Horiz.	5.75	6-7,E-D
TB	X-K-101-30	Mez.	30	"1B" MS Piping From 90° Elbow to HP Turbine		Vert.	13.30	6-7,E-D
				Linear FT on Oper. Deck			**76.39**	
				Linear FT on Mez. Level			**285.39**	
				Linear FT on Basement Level			**0.00**	
				Total Length (Linear FT)			**361.78**	
MAIN STEAM PIPING (STEAM DUMP) LOCATED IN THE TURBINE BUILDING OPERATING DECK AND MEZZANINE LEVELS (18 INCH PIPE)								
"A" Train [illegible]								
TB	M-239	Oper Deck	18	A6 - "1A" Train Steam Dump Piping from Tee with 30" MS Line To 90° Elbow Bend	"8-8"	Horiz.	3.68	7-8, F-G
TB	M-239	Oper Deck	18	A6 - "1A" Train Steam Dump Piping 90° Elbow Bend Thru Oper. Deck Floor to Mez.	"8-8"	Vert.	2.97	7-8, F-G
TB	M-239	Mez.	18	A6 - "1A" Train Steam Dump Piping From Oper. Deck Floor to Mez. Level 90° Elbow Bend	"8-8"	Vert.	4.10	7-8, F-G
TB	M-239	Mez.	18	A6 - "1A" Train MS Steam Dump Piping From 90° Elbow Bend to 90° Elbow Bend		Horiz.	13.55	7-8, F-G

Table A3-1: High-Pressure Main Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-239	Mez.	18	A6 - "1A" Train Steam Dump Piping From 90° Bend Thru 30° Down Angle Towards "7" Line	"8-8"	Angle	5.17	7-8, F-G
TB	M-239	Mez.	18	A6 - "1A" Train Steam Dump Piping From End of 30° Down Angle To 90° Elbow Bend		Horiz.	35.38	6-7, F-G
TB	M-239, -240	Mez.	18	B4 - "1A" Train Steam Dump Piping From 90° Bend to Capped End	"19-19"	Horiz.	15.17	5-6, F-G
"B" Train								
TB	M-239	Mez.	18	A3 - "1B" Train Steam Dump Piping from Tee in 30" MS Line 45° Elbow Bend at "5" Line		Horiz.	4.96	5-4, F-G
TB	M-239, -241	Mez.	18	D8 - "1B" Train Steam Dump Piping from 45° Elbow Bend to 45° Elbow Bend	"25-25"	Angle	3.18	5-4, F-G
TB	M-239	Mez.	18	A3 - "1B" Train Steam Dump Piping from 45° Elbow Bend to 90° Elbow Bend		Horiz.	14.40	4-5, F-G
TB	M-239, -241	Mez.	18	A3 - "1B" Train Steam Dump Piping from 90° Thru a 45° Declined Angle to 45° Elbow Bend	"20-20"	Angle	3.03	4-5, F-G
TB	M-239	Mez.	18	A3 - "1B" Train Steam Dump Piping from 45° Elbow Bend to Capped End		Horiz.	18.17	4-5, E-F
				Linear FT on Oper. Deck			6.65	
				Linear FT on Mez. Level			117.11	
				Linear FT on Basement Level			0.00	
				Total Length (Linear FT)			123.76	
MAIN STEAM PIPING (STEAM DUMP) LOCATED IN THE TURBINE BUILDING OPERATING DECK AND MEZZANINE LEVELS (8 INCH PIPE)								
"A" Train - 8" MS LINE FROM 18" STEAM DUMP 1A (Valves SD1-1, SD2-1, FCV-484A)								
TB	M-239, -240	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping (1 of 3) From 18" Main Header to Capped Tee	"19-19"	Vert.	12.00	5-6, E-F
TB	M-239, -240	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping (1 of 3) From 8" Tee to 16"x8" Reducer	"19-19"	Horiz.	7.50	5-6, E-F

Table A3-1: High-Pressure Main Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
"A" Train - 8" MS LINE FROM 18" STEAM DUMP 1A (Valves SD1-3, SD2-3, FCV-484B)								
TB	M-239, -241	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping From 18" Main Header to 90° Elbow (2 of 3)	"22-22"	Horiz.	2.08	5-6, E-F
TB	M-239, -241	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping (2 of 3) From 90° Elbow to 90° Elbow	"22-22"	Vert.	3.63	5-6, E-F
TB	M-239, -241	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping (2 of 3) From 90° Elbow to 90° Elbow	"22-22"	Horiz.	1.67	5-6, E-F
TB	M-239, -241	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping (2 of 3) From 90° Elbow to 90° Elbow	"22-22"	Vert.	7.87	5-6, E-F
TB	M-239, -241	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping (2 of 3) From 90° Elbow to Tee in 16" Line	"22-22"	Horiz.	16.67	5-6, E-F
"A" Train - 8" MS LINE FROM 18" STEAM DUMP 1A (Valves SD1-2, SD2-2, FCV-484C)								
TB	M-239, -241	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping From 18" Main Header to 90° Elbow (3 of 3)	"22-22"	Horiz.	3.75	5-6, E-F
TB	M-239, -241	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow to 90° Elbow	"22-22"	Vert.	2.21	5-6, E-F
TB	M-239, -241	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow to 90° Elbow	"22-22"	Horiz.	2.25	5-6, E-F
TB	M-239, -241	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow to 90° Elbow	"22-22"	Vert.	9.29	5-6, E-F
TB	M-239, -241	Mez.	8	B4 - "1A" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow to Tee in 16" Line	"22-22"	Horiz.	18.83	5-6, E-F
"B" Train - 8" MS LINE FROM 18" STEAM DUMP 1B (Valves SD1-4, SD2-4, FCV-484D)								
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (1 of 3) From 18" Main Header to 90° Elbow	"20-20" "18-18"	Vert.	2.00	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (1 of 3) From 90° Elbow Thru 180° Return to 90° Elbow	"18-18"	Horiz.	22.00	3-5, E-F

Table A3-1: High-Pressure Main Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (1 of 3) From 90° Elbow at 45° Declined Angle to 45° Elbow	"20-20"	Angle	0.00	(FT included in 9.15 below) 4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (1 of 3) From 45° Elbow to Capped Tee	"20-20"	Vert.	9.15	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (1 of 3) From 8" Tee to 16"x8" Reducer	"20-20"	Horiz.	9.00	4-5, E-F
"B" Train - 8" MS LINE FROM 18" STEAM DUMP 1B (Valves SD1-5, SD2-5, FCV-484E)								
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (2 of 3) From 18" Main Header to 90° Elbow	"21-21"	Vert.	1.83	4-5, E-F
TB	M-239	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (2 of 3) From 90° Elbow to 90° Elbow		Horiz.	3.58	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (2 of 3) From 90° Elbow to 90° Elbow	"21-21"	Vert.	4.13	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (2 of 3) From 90° Elbow Thru 180° Bend to 90° Elbow	"21-21"	Horiz.	30.13	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (2 of 3) From 90° Elbow to 90° Elbow	"21-21"	Horiz.	2.50	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (2 of 3) From 90° Elbow Thru SD1-5 to 90° Elbow	"21-21"	Vert.	7.69	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (2 of 3) From 90° Elbow Thru FCV-484E and SD2-5 to 90° Elbow	"21-21"	Horiz.	6.17	4-5, E-F
TB	M-239	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (2 of 3) From 90° Elbow to Tee in 16" Line		Horiz.	3.50	4-5, E-F
"B" Train - 8" MS LINE FROM 18" STEAM DUMP 1B (Valves SD1-6, SD2-6, FCV-484F)								
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (3 of 3) From 18" Main Header to 90° Elbow	"21-21"	Vert.	3.00	4-5, E-F

Table A3-1: High-Pressure Main Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (In)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow to 90° Elbow	"21-21"	Horiz.	10.45	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow to 90° Elbow	"21-21"	Vert.	7.58	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow to 90° Elbow	"21-21"	Horiz.	7.13	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow to 90° Elbow	"21-21"	Horiz.	7.25	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow to 90° Elbow	"21-21"	Horiz.	11.83	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow to 90° Elbow	"21-21"	Horiz.	4.00	4-5, E-F
TB	M-239, -241	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow Thru Valve SD1-6 to 90° Elbow	"21-21"	Vert.	6.08	4-5, E-F
TB	M-239	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow Thru Valves FCV-484F and SD2-6 to 90° Elbow		Horiz.	6.17	4-5, E-F
TB	M-239	Mez.	8	B3 - "1B" Train 8" Steam Dump Piping (3 of 3) From 90° Elbow to Tee in 16" Line		Horiz.	5.75	4-5, E-F
8" MS LINE FROM 30" MS HEADER 1A TO MOISTURE SEPARATORS/REHEATERS A1 AND A2								
TB	M-239	Mez.	8	C6 - "1A" Train 8" Steam Line From 30" Main Header to 90° Elbow	"5-5" "5A-5A"	Horiz.	7.42	D-C, 7-8
TB	M-239	Mez.	8	C6 - "1A" Train 8" Steam Line From 90° Elbow to 90° Elbow	"5A-5A"	Vert.	2.29	D-C, 7-8
TB	M-239	Mez.	8	C6 - "1A" Train 8" Steam Line From 90° Elbow to 90° Elbow		Horiz.	14.50	B-C, 7-8
TB	M-239	Mez.	8	C6/D6 - "1A" Train 8" Steam Line From 90° Elbow to 8"x4" Reducer	"6-6"	Horiz.	99.00	B-C, 8-4
						Linear FT on Oper. Deck	**0.00**	
						Linear FT on Mez. Level	**381.87**	
						Linear FT on Basement Level	**0.00**	

Table A3-1: High-Pressure Main Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (In)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
					Total Length (Linear FT)		381.87	
MAIN STEAM PIPING (TO MSRs) LOCATED IN THE TURBINE BUILDING OPERATING DECK AND MEZZANINE LEVELS (6 INCH PIPE)								
6" MS LINE FROM 18" STEAM DUMP HEADER 1A TO MOISTURE SEPARATORS/REHEATERS B1								
TB	M-239, -240	Mez.	6	B4 - "1A" Train 6" Steam Line from 18" Steam Dump to 90° Elbow	"17-17"	Horiz.	12.25	5-7, F-E
TB	M-239, -240	Mez.	6	B4/B5 - "1A" Train 6" Steam Line from 90° Elbow to Oper. Deck Level Penetration	"17-17"	Vert.	6.75	5-7, F-E
TB	M-239, -240	Oper Deck	6	B4/B5 - "1A" Train 6" Steam Line from Oper. Deck Level Penetration to 90° Elbow	"17-17"	Vert.	2.00	5-7, F-E
TB	M-203, -239, -240	Oper Deck	6	B4/B5 - "1A" Train 6" Steam Line from 90° Elbow Thru Valves (MS200B1, MS201B1) to 90° Elbow	"17-17"	Horiz.	7.65	5-7, F-E
TB	M-239, -240	Oper Deck	6	B4/B5 - "1A" Train 6" Steam Line from 90° Elbow to 90° Elbow	"17-17"	Vert.	8.34	5-7, F-E
TB	M-239, -240	Oper Deck	6	B5 - "1A" Train 6" Steam Line from 90° Elbow Thru Orifice to 90° Elbow	"17-17"	Horiz.	11.50	5-7, F-E
TB	M-239, -240	Oper Deck	6	B5 - "1A" Train 6" Steam Line from 90° Elbow to 90° Elbow	"17-17"	Horiz.	6.75	5-7, F-E
TB	M-239, -240	Oper Deck	6	B5 - "1A" Train 6" Steam Line from 90° Elbow to Moisture Sep/Reheater B1	"17-17"	Angle	2.00	5-7, F-E
6" MS LINE FROM 18" STEAM DUMP HEADER 1B TO MOISTURE SEPARATORS/REHEATERS B2								
TB	M-239, -240	Mez.	6	B3 - "1B" Train 6" Steam Line from 18" Steam Dump to 90° Elbow	"18-18"	Vert.	3.08	5-3, F-E
TB	M-239, -240	Mez.	6	B2/B3 - "1B" Train 6" Steam Line from 90° Elbow to 90° Elbow	"18-18"	Horiz.	12.63	5-3, F-E
TB	M-239, -240	Mez.	6	B2/B3 - "1B" Train 6" Steam Line from 90° Elbow to Oper. Deck Level Penetration	"18-18"	Vert.	3.25	5-3, F-E
TB	M-239, -240	Oper Deck	6	B2/B3 - "1B" Train 6" Steam Line from Oper. Deck Level Penetration to 90° Elbow	"18-18"	Vert.	2.00	5-3, F-E

Table A3-1: High-Pressure Main Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-239, -240	Oper Deck	6	B2/B3 - "1B" Train 6" Steam Line from 90° Elbow Thru Valves (MS200B2, MS201B2) to 90° Elbow	"18-18"	Horiz.	7.65	5-3, F-E
TB	M-239, -240	Oper Deck	6	B2/B3 - "1B" Train 6" Steam Line from 90° Elbow to 90° Elbow	"18-18"	Vert.	8.34	5-3, F-E
TB	M-239, -240	Oper Deck	6	B2 - "1B" Train 6" Steam Line from 90° Elbow Thru Orifice to 90° Elbow	"18-18"	Horiz.	11.50	5-3, F-E
TB	M-239	Oper Deck	6	B2 - "1B" Train 6" Steam Line from 90° Elbow to 90° Elbow		Horiz.	6.75	5-3, F-E
TB	M-239, -240	Oper Deck	6	B2 - "1B" Train 6" Steam Line from 90° Elbow to Moisture Sep/Reheater B2	"18-18"	Angle	2.50	5-3, F-E
6" MS LINE FROM 8" STEAM HEADER 1A TO MOISTURE SEPARATORS/REHEATERS A1								
TB	M-239	Mez.	6	D4/E4 - "1A" Train 6" Steam Line from 8" Steam Supply Line to Oper. Deck Penetration	"9-9"	Vert.	5.19	6-7, B-C
TB	M-239	Oper Deck	6	D4/E4 - "1A" Train 6" Steam Line from Oper. Deck Penetration to 90° Elbow	"9-9"	Horiz.	2.00	6-7, B-C
TB	M-239	Oper Deck	6	D5/E5 - "1A" Train 6" Steam Line from 90° Elbow Thru Valves (MS200A1, MS201A1) to 90° Elbow	"9-9"	Horiz.	7.65	6-7, B-C
TB	M-239	Oper Deck	6	D5/E5 - "1A" Train 6" Steam Line from 90° Elbow to 90° Elbow	"9-9"	Vert.	8.34	6-7, B-C
TB	M-239	Oper Deck	6	D5/E5 - "1A" Train 6" Steam Line from 90° Elbow Thru Orifice to 90° Elbow	"9-9"	Horiz.	11.50	6-7, B-C
TB	M-239	Oper Deck	6	D5/E5 - "1A" Train 6" Steam Line from 90° Elbow to 90° Elbow		Horiz.	6.75	6-7, B-C
TB	M-239	Oper Deck	6	D5/E5 - "1A" Train 6" Steam Line from 90° Elbow to Moisture Sep/Reheater A1	"9-9"	Angle	2.50	6-7, B-C
6" MS LINE FROM 8" STEAM HEADER 1A TO MOISTURE SEPARATORS/REHEATERS A2								
TB	M-239	Mez.	6	D2/E2 - "1A" Train 6" Steam Line from 8" Steam Supply Line to Oper. Deck Penetration	"6-6"	Vert.	5.52	4-3, B-C

Table A3-1: High-Pressure Main Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-239	Oper Deck	6	D2/E2 - "1A" Train 6" Steam Line from Oper. Deck Penetration to 90° Elbow	"6-6"	Horiz.	2.00	4-3, B-C
TB	M-239	Oper Deck	6	D2/E2 - "1A" Train 6" Steam Line from 90° Elbow Thru Valves (MS200A2, MS201A2) to 90° Elbow	"6-6"	Horiz.	7.65	4-3, B-C
TB	M-239	Oper Deck	6	D2/E2 - "1A" Train 6" Steam Line from 90° Elbow to 90° Elbow	"6-6"	Vert.	8.34	4-3, B-C
TB	M-239	Oper Deck	6	D2/E2 - "1A" Train 6" Steam Line from 90° Elbow Thru Orifice to 90° Elbow	"6-6"	Horiz.	11.50	4-3, B-C
TB	M-239	Oper Deck	6	D2/E2 - "1A" Train 6" Steam Line from 90° Elbow to 90° Elbow		Horiz.	6.75	4-3, B-C
TB	M-239	Oper Deck	6	D2/E2 - "1A" Train 6" Steam Line from 90° Elbow to Moisture Sep/Reheater A2	"6-6"	Angle	2.50	4-3, B-C
				Linear FT on Oper. Deck			**154.46**	
				Linear FT on Mez. Level			**48.67**	
				Linear FT on Basement Level			**0.00**	
				Total Length (Linear FT)			**203.13**	
MAIN STEAM HEADER "1A"/"1B" EQUALIZING LINE (20" PIPE)								
TB	M-985-2	Mez.	20	B6 - Equalizing Line Between Main "A" and "B" Steam Headers		Horiz.	42.25	7-8, F-E
				Linear FT on Oper. Deck			**0.00**	
				Linear FT on Mez. Level			**42.25**	
				Linear FT on Basement Level			**0.00**	
				Total Length (Linear FT)			**42.25**	
STEAM LINE SECTION FROM 16"X8" REDUCER FROM 8" STEAM DUMP LINES TO 16" NOZZLE (16" PIPE)								
TRAIN "1A" MS LINE FROM 16"X8" REDUCER FROM 8" STEAM DUMP LINES TO 16" NOZZLE (16" PIPE)								
TB	M-239, -240	Mez.	16	B4/C4 - "1A" Train 16" Steam Line From 16"x8" Reducer to Low Pressure Turbine	"19-19"	Horiz.	4.67	5-6, E
TRAIN "1B" MS LINE FROM 16"X8" REDUCER FROM 8" STEAM DUMP LINES TO 16" NOZZLE (16" PIPE)								
TB	M-239, -240	Mez.	16	B3/C3 - "1B" Train 16" Steam Line From 16"x8" Reducer to Low Pressure Turbine	"19-19"	Horiz.	4.67	4-5, E

Table A3-1: High-Pressure Main Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
				Linear FT on Oper. Deck			0.00	
				Linear FT on Mez. Level			9.34	
				Linear FT on Basement Level			0.00	
				Total Length (Linear FT)			9.34	

Table A3-2: Extraction Steam Piping

BLDG	Dwg. No.	Building Level	Nom. Dia. (In)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
HP TURBINE STEAM SUPPLY TO HIGH PRESSURE FEEDWATER HEATERS 15A/15B								
BLEED STEAM HEADER TO FEEDWATER HEATERS 15A/15B								
TB	M-1258	Mez.	12	B11 - Bleed Steam Piping From Turbine Shell Insulation to 90° Elbow		Vert.	6.27	6-7, D
TB	M-1258	Mez.	12	B11/A10 - Bleed Steam Piping From 90° Elbow to 45° Declined Angle Bend		Horiz.	18.98	6-7, D-E
TB	M-1258	Mez.	12	A10 - Bleed Steam Piping From 45° Declined Angle Bend to 16"x12" 90° Reducing Elbow		Angle	3.67	6-7, E
TB	M-1258	Mez.	16	A10/E3 - Bleed Steam Piping From 16"x12" 90° Reducing Elbow to 90° Elbow		Horiz.	71.09	7-4, E
TB	M-1258	Mez.	16	E3/D2 - Bleed Steam Piping From 90° Elbow to 16"X10" 90° Reducing Elbow		Horiz.	23.80	4, E-F
TB	M-1258	Mez.	12	C11 - Bleed Steam Piping From Turbine Shell Insulation to 90° Elbow		Vert.	2.83	6-7, D
TB	M-1258	Mez.	12	C11/C10 - Bleed Steam Piping From 90° Elbow to 90° Elbow		Horiz.	8.04	6-7, E-D
TB	M-1258	Mez.	12	C10 - Bleed Steam Piping From 90° Elbow to 90° Elbow		Vert.	4.58	6-7, E-D
TB	M-1258	Mez.	12	C10 - Bleed Steam Piping From 90° Elbow to 45° Declined Angle Bend		Horiz.	7.50	6-7, E-D
TB	M-1258	Mez.	12	C9 - Bleed Steam Piping From 45° Declined Angle Bend to 45° Angle Bend		Angle	3.67	6-7, E-D
TB	M-1258	Mez.	12	C9 - Bleed Steam Piping From 45° Angle Bend to 45° Angle Bend		Horiz.	4.85	6-7, E
TB	M-1258	Mez.	12	C9 - Bleed Steam Piping From 45° Angle Bend to 45° 16"x12" Lateral Reducer		Horiz.	3.54	6-7, E
STEAM TO FOR FEEDWATER HEATER 15A								
TB	M-1258	Mez.	10	D1/E1 - Bleed Steam Piping From 16"X16"X10" Tee to 12"x10" 90° Reducing Elbow		Horiz.	6.96	4, E-F

Table A3-2: Extraction Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-1258	Mez.	12	E1 - Bleed Steam Piping From 12"x10" 90° Reducing Elbow Top of FD WTR HTR 15A		Vert.	2.00	4, E-F
STEAM TO FOR FEEDWATER HEATER 15B								
TB	M-1258	Mez.	10	D2 - Bleed Steam Piping From 16"X16"X10" Tee to 12"x10" 90° Reducing Elbow		Horiz.	6.69	4-F
TB	M-1258	Mez.	12	E2 - Bleed Steam Piping From 12"x10" 90° Reducing Elbow Top of FD WTR HTR 15B		Vert.	2.00	4-F
				Linear FT on Oper. Deck			0.00	
				Linear FT on Mez. Level			176.46	
				Linear FT on Basement Level			0.00	
				Total Length (Linear FT)			176.46	

Table A3-3: Lower-Pressure Steam Piping

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
HP TURBINE STEAM SUPPLY TO HIGH PRESSURE FEEDWATER HEATERS 14A/14B								
REHEAT STEAM FROM REHEAT STEAM CROSSUNDER PIPE TO FOR FEEDWATER HEATER 14A/14B								
TB	M-242	Mez.	16	E7 - Reheat Steam From 30" Crossunder Piping to 90° Elbow		Horiz.	1.75	6, D
TB	M-242, -423	Mez.	16	E7 - Reheat Steam From 90° Elbow to 90° Elbow	"C-C"	Vert.	20.73	6-7, D-E
TB	M-242, -423	Mez.	16	E7 - Reheat Steam From 90° Elbow to 45° Angle Bend Into 24" Reheat Steam Header		Horiz.	5.66	6-7, D-E
TB	M-242	Mez.	16	F8 - Reheat Steam From 30" Crossunder Piping to 90° Elbow		Horiz.	1.75	6-7, D
TB	M-242, -423	Mez.	16	F8 - Reheat Steam From 90° Elbow to 24"x16" 90° Reducing Elbow	"C-C"	Vert.	20.73	6-7, D
TB	M-242	Mez.	24	F8 - Reheat Steam From 24"x16" 90° Reducing Elbow to 90° Elbow		Horiz.	28.50	6-7, D-E
TB	M-242	Mez.	24	D8 - Reheat Steam From 90° Elbow to 90° Elbow		Horiz.	14.33	6-7, E
TB	M-242	Mez.	24	D9 - Reheat Steam From 90° Elbow to 90° Elbow		Horiz.	6.00	7, E
TB	M-242, -423	Mez.	24	D9/C9 - Reheat Steam From 90° Elbow to 90° Elbow	"C-C"	Vert.	14.21	7, E-F
TB	M-242, -423	Mez.	24	D9/C9 - Reheat Steam From 90° Elbow to 24"x16" 90° Reducing Elbow For FD WTR HTR 14B	"C-C"	Horiz.	14.92	7, E-F
STEAM TO FOR FEEDWATER HEATER 14A (16" PIPE)								
TB	M-242	Mez.	16	C9 - Reheat Steam from 24"x24"x16" Tee to 90° Elbow		Horiz.	9.17	6-7, E-F
TB	M-242, -423	Mez.	16	C8 - Reheat Steam from 90° Elbow To FD WTR HTR 14A	"C-C"	Vert.	2.00	6-7, E-F
STEAM TO FOR FEEDWATER HEATER 14B (16" PIPE)								
TB	M-242	Mez.	16	B9 - Reheat Steam from 24"x16" 90° Reducing Elbow to 90° Elbow For FD WTR HTR 14B		Horiz.	9.17	6-7, F

Table A3-3: Lower-Pressure Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-242, -423	Mez.	16	B8 - Reheat Steam from 90° Elbow To FD WTR HTR 14B	"C-C"	Angle	2.15	6-7, F
						Linear FT on Oper. Deck	0.00	
						Linear FT on Mez. Level	151.07	
						Linear FT on Basement Level	0.00	
						Total Length (Linear FT)	151.07	
HEATING STEAM FROM FEEDWATER HEATERS 14A/14B 24" STEAM SUPPLY LINE (10" PIPE)								
HEATING STEAM FROM 24" REHEAT STEAM (10" PIPE)								
TB	M-242, -423	Mez.	10	C9 - Heating Steam from 24" Reheat Steam Line Tee to 90° Elbow	"C-C"	Vert.	7.10	7, E-F
TB	M-242	Mez.	10	C9 - Heating Steam From 90° Elbow to 90° Elbow		Horiz.	4.00	7, E-F
TB	M-242	Mez.	10	C9 - Heating Steam From 90° Elbow to 45° Bend		Horiz.	6.04	7, E-F
TB	M-242	Mez.	10	C8 - Heating Steam From 45° Bend to 45° Bend		Horiz.	9.90	7-6, F
TB	M-242	Mez.	10	B8 - Heating Steam From 45° Bend to 90° Declined Elbow		Horiz.	47.00	7-5, F
TB	M-242, -423	Mez.	10	B4 - Heating Steam From 90° Declined Elbow to 90° Elbow	"C-C"	Angle	8.44	5, F-G
TB	M-242	Mez.	10	B4 - Heating Steam From 90° Elbow to 60° Bend		Horiz.	56.50	5-3, F-G
TB	M-242	Mez.	10	B1 - Heating Steam From 60° Bend to 90° Elbow		Horiz.	9.50	3, F-G
TB	M-242, -423	Mez.	10	A1 - Heating Steam From 90° Elbow to Mez. Floor Penetration	"J-J"	Vert.	11.02	3, G
TB	M-242, -423	Basement	10	A1 - Heating Steam From Mez. Floor Penetration to 90° Elbow	"J-J"	Vert.	4.81	3, G
TB	M-242, -423	Basement	10	A1 - Heating Steam From 90° Elbow to 90° Elbow	"J-J"	Horiz.	20.25	3-2, G
TB	M-423	Basement	10	H4 - Heating Steam From 90° Elbow to Aux. Bld Wall Pen.		Horiz.	10.00	3-2, G-GG
						Linear FT on Oper. Deck	0.00	

Table A3-3: Lower-Pressure Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
						Linear FT on Mez. Level	159.50	
						Linear FT on Basement Level	35.06	
						Total Length (Linear FT)	194.56	
MAIN STEAM CROSSUNDER PIPING (30" PIPE)								
STEAM CROSSUNDER PIPING FROM HP TURBINE TO "A" MSRs - FRONT PIPE								
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Crossunder Piping (to "A" MSRs - Front Pipe) from HP Turbine to 90° Elbow		Vert.	10.56	6-7, C-D
STEAM CROSSUNDER PIPING FROM HP TURBINE TO "A" MSRs - REAR PIPE								
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Crossunder Piping (to "A" MSRs - Rear Pipe) from HP Turbine to 90° Elbow		Vert.	10.56	6-7, C-D
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Crossunder Piping (to "A" MSRs - Rear Pipe) from 90° Elbow to 90° Elbow		Horiz.	8.86	6-7, C-D
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Crossunder Piping (to "A" MSRs - Rear Pipe) from 90° Elbow to 42" Crossunder Piping to "A" MSRs		Horiz.	8.54	6-7, C-D
STEAM CROSSUNDER PIPING FROM HP TURBINE TO "A" MSRs - HEADER PIPE								
TB	XK-101-30, XK-101-33	Mez.	30, 42	Reheat Steam Crossunder Piping (to "A" MSRs - Header) from 30" 90° Elbow Thru 42"x30" Reducer to 42" 90° Elbow		Horiz.	25.00	7-5, C-B
TB	XK-101-30, XK-101-33	Mez.	30, 42	Reheat Steam Crossunder Piping (to "A" MSRs - Header) from 42" 90° Elbow Thru 42"x30" Reducer to 30" 90° Elbow		Horiz.	53.38	7-5, C-B
STEAM CROSSUNDER PIPING TO MSR 1A								
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Piping to MSR "1A" from 42" Crossunder Header to Oper. Deck Floor Penetration		Vert.	10.56	5, B-C
TB	XK-101-30, XK-101-33	Oper Deck	30	Reheat Steam Piping to MSR "1A" from Oper. Deck Floor Penetration to 30" 90° Elbow		Vert.	4.75	5, B-C

Table A3-3: Lower-Pressure Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	XK-101-30, XK-101-33	Oper Deck	30	Reheat Steam Piping from 30" 90° Elbow to MSR "1A"		Horiz.	11.75	5, B-C
STEAM CROSSUNDER PIPING TO MSR 2A								
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Piping to MSR "2A" from 42" Crossunder Header to Oper. Deck Floor Penetration		Vert.	10.56	5, B-C
TB	XK-101-30, XK-101-33	Oper Deck	30	Reheat Steam Piping to MSR "2A" from Oper. Deck Floor Penetration to 30" 90° Elbow		Vert.	4.75	5, B-C
TB	XK-101-30, XK-101-33	Oper Deck	30	Reheat Steam Piping from 30" 90° Elbow to MSR "2A"		Horiz.	11.75	5, B-C
STEAM CROSSUNDER PIPING FROM HP TURBINE TO "B" MSRs - FRONT PIPE								
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Crossunder Piping (to "B" MSRs - Front Pipe) from HP Turbine to 90° Elbow		Vert.	10.56	6-7, D-E
STEAM CROSSUNDER PIPING FROM HP TURBINE TO "B" MSRs - REAR PIPE								
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Crossunder Piping (to "B" MSRs - Rear Pipe) from HP Turbine to 90° Elbow		Vert.	10.56	6-7, D-E
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Crossunder Piping (to "B" MSRs - Rear Pipe) from 90° Elbow to 90° Elbow		Horiz.	8.86	6-7, D-E
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Crossunder Piping (to "B" MSRs - Rear Pipe) from 90° Elbow to 42" Crossunder Piping to "B" MSRs		Horiz.	8.54	6-7, D-E
STEAM CROSSUNDER PIPING FROM HP TURBINE TO "B" MSRs - HEADER PIPE								
TB	XK-101-30, XK-101-33	Mez.	30, 42	Reheat Steam Crossunder Piping (to "A" MSRs - Header) from 30" 90° Elbow Thru 42"x30" Reducer to 42" 90° Elbow		Horiz.	25.00	7-5, E-F

Table A3-3: Lower-Pressure Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	XK-101-30, XK-101-33	Mez.	30, 42	Reheat Steam Crossunder Piping (to "B" MSRs - Header) from 42" 90° Elbow Thru 42"x30" Reducer to 30" 90° Elbow		Horiz.	53.38	7-5, E-F
STEAM CROSSUNDER PIPING TO MSR 1B								
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Piping to MSR "1B" from 42" Crossunder Header to Oper. Deck Floor Penetration		Vert.	10.56	5, E-F
TB	XK-101-30, XK-101-33	Oper. Deck	30	Reheat Steam Piping to MSR "1B" from Oper. Deck Floor Penetration to 30" 90° Elbow		Vert.	4.75	5, E-F
TB	XK-101-30, XK-101-33	Oper. Deck	30	Reheat Steam Piping from 30" 90° Elbow to MSR "1B"		Horiz.	11.75	5, E-F
STEAM CROSSUNDER PIPING TO MSR 2B								
TB	XK-101-30, XK-101-33	Mez.	30	Reheat Steam Piping to MSR "2B" from 42" Crossunder Header to Oper. Deck Floor Penetration		Vert.	10.56	5, E-F
TB	XK-101-30, XK-101-33	Oper. Deck	30	Reheat Steam Piping to MSR "2B" from Oper. Deck Floor Penetration to 30" 90° Elbow		Vert.	4.75	5, E-F
TB	XK-101-30, XK-101-33	Oper. Deck	30	Reheat Steam Piping from 30" 90° Elbow to MSR "2B"		Horiz.	11.75	5, E-F
				Linear FT on Oper. Deck			66.00	
				Linear FT on Mez. Level			276.04	
				Linear FT on Basement Level			0.00	
				Total Length (Linear FT)			342.04	
STEAM CROSSOVER PIPING FROM MOISTURE SEPARATORS 1A/1B/2A/2B TO LOW PRESSURE TURBINE (30" PIPE)								
STEAM CROSSOVER PIPING FROM MSR 1A								
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from MSR 1A to 90° Elbow		Vert.	13.42	6, B-C
TB	X-K-101-30, X-K-	Oper. Deck	30	Steam Crossover Piping from 90° Elbow to 90° Elbow		Horiz.	12.00	6-5, B-C

Table A3-3: Lower-Pressure Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
	101-33							
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from 90° Elbow to Bend into LP Turbine #1		Horiz.	27.50	6-5, B-D
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from Bend into LP Turbine #1 to Common Inlet		Vert.	3.50	6-5, D
STEAM CROSSOVER PIPING FROM MSR 1B								
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from MSR 1B to 90° Elbow		Vert.	13.42	6, E-F
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from 90° Elbow to 90° Elbow		Horiz.	12.00	6-5, E-F
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from 90° Elbow to Bend into LP Turbine #1		Horiz.	27.50	6-5, E-D
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from Bend into LP Turbine #1 to Common Inlet		Vert.	3.50	6-5, D
STEAM CROSSOVER PIPING FROM MSR 2A								
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from MSR 2A to 90° Elbow		Vert.	13.42	4, B-C
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from 90° Elbow to 90° Elbow		Horiz.	12.00	4-5, B-C
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from 90° Elbow to Bend into LP Turbine #2		Horiz.	27.50	4-5, B-D
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from Bend into LP Turbine #2 to Common Inlet		Vert.	3.50	4-5, D
STEAM CROSSOVER PIPING FROM MSR 2B								

Table A3-3: Lower-Pressure Steam Piping (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (In)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from MSR 2B to 90° Elbow		Vert.	13.42	4, E-F
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from 90° Elbow to 90° Elbow		Horiz.	12.00	4-5, E-F
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from 90° Elbow to Bend into LP Turbine #2		Horiz.	27.50	4-5, E-D
TB	X-K-101-30, X-K-101-33	Oper. Deck	30	Steam Crossover Piping from Bend into LP Turbine #2 to Common Inlet		Vert.	3.50	4-5, D
				Linear FT on Oper. Deck			225.68	
				Linear FT on Mez. Level			0.00	
				Linear FT on Basement Level			0.00	
				Total Length (Linear FT)			225.68	

Table A3-4: Piping Upstream of 15 Feedwater Heaters

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
CONDENSATE PIPING LOCATED BETWEEN HEATERS 14 AND MAIN FEEDWATER PUMPS								
TB	M-245	Basement	12	B10 - 12-inch bypass header centerline of valve C20-1 to elbow that angles down to main 20-inch header	D-D, E-E	Horiz.	16.75	7-7, F-F
TB	M-245	Basement	12	B10 - 12-inch bypass header centerline of valve C20-1 angling down to main 20-inch header	D-D, E-E	Horiz.	5.23	7-7, F-F
TB	M-245	Mezzanine	16	B10 - Horizontal distance from outlet of heater 14A to centerline of vertical pipe down	D-D, E-E	Horiz.	2.00	7-7, F-F
TB	M-246	Basement	16	C4 - 16-inch header piping down from outlet of heater 14A to mezzanine floor level	D-D	Vert.	3.67	7-7, F-F
TB	M-246	Mezzanine	16	C4 - 16-inch header piping down from mezzanine floor level to centerline of header	D-D	Vert.	5.50	7-7, F-F
TB	M-245	Basement	14	B10 - Centerline of vertical pipe down from heater 14A outlet to inlet of main 20-inch header	D-D, E-E	Horiz.	9.50	7-7, F-F
TB	M-245	Mezzanine	16	B10 - Horizontal distance from outlet of heater 14B to centerline of vertical pipe down	D-D, E-E	Horiz.	2.00	7-7, F-F
TB	M-246	Mezzanine	16	C4 - 16-inch header piping down from outlet of heater 14B to mezzanine floor level	D-D	Vert.	3.67	7-7, F-F
TB	M-246	Basement	16	C4 - 16-inch header piping down from mezzanine floor level to centerline of header	D-D	Vert.	5.50	7-7, F-F
TB	M-245	Basement	14	B10 - Centerline of vertical pipe down from heater 14B outlet to 90 degree elbow to the east	D-D, E-E	Horiz.	9.50	7-7, F-F
TB	M-245	Basement	14	B10 - Header pipe east from centerline of C15-2 to 20-inch header	D-D, E-E	Horiz.	12.25	7-7, F-F
TB	M-245	Basement	20	B10 - Header pipe east from reducer east to centerline of main 20-inch header south	D-D, E-E	Horiz.	8.25	7-7, F-F

Table A3-4: Piping Upstream of 15 Feedwater Heaters (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-245	Basement	12	B10 - 12-inch bypass header from centerline of valve C9-1 south to beginning of pipe bend	D-D, E-E	Horiz.	8.00	7-7, F-F
TB	M-245	Basement	12	B10 - 12-inch bypass header the beginning of pipe bend angling to the 14-inch header	D-D, E-E	Horiz.	7.78	7-7, F-F
TB	M-245	Basement	20	B10 - 20-inch header south toward feedwater pumps to reducer	E-E	Horiz.	36.50	7-6, F-F
TB	M-246	Basement	16	D6 - 16-inch header piping down from main header to main feedwater pump 1A inlet	F-F	Vert.	6.25	6-6, F-F
TB	M-246	Basement	16	B8 - 16-inch pipe west to main feedwater pump 1A suction	F-F	Horiz.	16.92	6-6, F-F
TB	M-246	Basement	16	B8 - 16-inch pipe south to main feedwater pump 1A suction	F-F	Horiz.	4.00	6-6, F-F
TB	M-246	Basement	16	D6 - 16-inch header piping down into main feedwater pump 1A inlet	F-F	Vert.	2.00	6-6, F-F
TB	M-245	Basement	16	B8 - 16-inch header south from reducer toward main feedwater pump 1B to 90 degree elbow down	G-G	Horiz.	39.00	6-4, F-F
TB	M-246	Basement	16	E12 - 16-inch header piping down from main header	G-G	Vert.	12.50	4-4, F-F
TB	M-245	Basement	16	B6 - 16-inch header south from vertical pipe down to 90 degree elbow up	G-G	Horiz.	10.00	4-4, F-F
TB	M-246	Basement	16	E11 - 16-inch header piping up from main header	G-G	Vert.	6.25	4-4, F-F
TB	M-245	Basement	16	B5 - 16-inch pipe west to main feedwater pump 1B suction	H-H	Horiz.	18.92	4-4, F-F
TB	M-246	Basement	16	D6 - 16-inch header piping down into main feedwater pump 1B inlet	H-H	Vert.	2.00	4-4, F-F
TB	M-252	Basement	12	B10 - Horizontal distance from the discharge of heater drain pump 1A east to 90 degree elbow up	Q-Q	Horiz.	1.93	7-7, F-F

Table A3-4: Piping Upstream of 15 Feedwater Heaters (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-252	Basement	12	B10 - Horizontal distance from the discharge of heater drain pump 1B east to 90 degree elbow up	Q-Q	Horiz.	1.93	7-7, F-F
TB	M-253	Basement	12	B10 - Vertical distance from centerline of heater drain pump 1A discharge to 14-inch line	Q-Q	Vert.	7.15	7-7, F-F
TB	M-253	Basement	12	B10 - Vertical distance from centerline of heater drain pump 1B discharge to 14-inch line	Q-Q	Vert.	7.15	7-7, F-F
TB	M-252	Basement	14	B10 - Horizontal distance from the discharge of heater drain pump 1A south to 90 degree elbow turning to the east	Q-Q	Horiz.	19.75	7-6, F-F
TB	M-252	Basement	14	B10 - Horizontal distance from centerline of 14-inch pump discharge header east to 90 degree elbow up	Q-Q	Horiz.	5.52	6-6, F-F
TB	M-253	Basement	14	B10 - Vertical distance from centerline of 14-inch pump header to centerline of 20-inch feedwater header	Q-Q	Vert.	6.19	6-6, F-F
TB	M-252	Basement	20	B10 - Horizontal distance from the centerline of heater drain tank to the discharge of heater drain pump 1A	Q-Q	Horiz.	13.46	7-7, F-F
TB	M-252	Basement	20	B10 - Horizontal distance from the centerline of heater drain tank to the discharge of heater drain pump 1B	Q-Q	Horiz.	13.46	6-6, F-F
TB	M-253	Basement	20	B10 - Vertical distance from heater drain tank outlet to centerline of heater drain pump 1A inlet	Q-Q	Vert.	4.00	7-7, F-F
TB	M-253	Basement	20	B10 - Vertical distance from heater drain tank outlet to centerline of heater drain pump 1B inlet	Q-Q	Vert.	4.00	7-7, F-F
				Linear FT on Basement			**325.31**	
				Linear FT on Mez. Level			**13.17**	
				Total Length (Linear FT)			**338.48**	
PIPING LOCATED BETWEEN MAIN FEEDWATER PUMPS AND 15 HEATERS								

Table A3-4: Piping Upstream of 15 Feedwater Heaters (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-249	Basement	16	B3 - Main feedwater pump 1A outlet up to 90 degree elbow to the east	E-E	Vert.	9.25	6-6, F-F
TB	M-247	Basement	16	C5 - Horizontal distance from centerline pump 1A discharge east to 90 degree elbow to the north	E-E	Horiz.	4.00	6-6, F-F
TB	M-247	Basement	16	C5 - Horizontal distance north on pump 1A discharge piping	E-E	Horiz.	9.50	6-6, F-F
TB	M-247	Basement	16	C6 - Horizontal distance west on pump 1A discharge piping	E-E	Horiz.	12.25	6-6, F-F
TB	M-247	Basement	16	C6 - Horizontal distance south on pump 1A discharge piping	E-E	Horiz.	19.50	6-5, F-F
TB	M-247	Basement	16	C6 - Horizontal distance east on pump 1A discharge piping to 90 degree elbow angling down to the south	E-E	Horiz.	16.00	5-5, F-F
TB	M-247	Basement	16	D5 - Pump 1A discharge piping south toward 15 feedwater heaters	E-E	Horiz.	42.88	5-4, F-F
TB	M-247	Basement	16	D2 - Pump 1A discharge piping angling 45 degrees to 22-inch header	G-G	Horiz.	6.36	4-4, F-F
TB	M-249	Basement	16	B6 - Main feedwater pump 1B outlet up to 90 degree elbow to the west	F-F	Vert.	6.50	4-4, F-F
TB	M-247	Basement	16	C3 - Horizontal distance from centerline pump 1B discharge west to 90 degree elbow angling up and to the north	F-F	Horiz.	8.75	4-4, F-G
TB	M-249	Basement	16	B6 - Horizontal distance for the pipe angling up from the 90 degree elbow to the centerline of valve F2-2	F-F	Horiz.	5.50	4-4, G-G
TB	M-247	Basement	16	C3 - Horizontal run of Pump 1B discharge piping north through valve F2-2	F-F	Horiz.	18.53	4-5, G-G
TB	M-247	Basement	16	C4 - Horizontal run of Pump 1B discharge piping downstream of valve F2-2 to the east	F-F	Horiz.	6.00	5-5, G-G
TB	M-247	Basement	16	C4 - Horizontal run of Pump 1B discharge piping north	F-F	Horiz.	7.71	5-5, G-G

Table A3-4: Piping Upstream of 15 Feedwater Heaters (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-247	Basement	16	C4 - Horizontal run of Pump 1B discharge piping east to the elbow angling down	E-E	Horiz.	13.50	5-5, G-F
TB	M-249	Basement	16	B6 - Main feedwater pump 1B discharge piping down to header towards 15 feedwater heaters	E-E	Vert.	2.67	5-5, F-F
TB	M-247	Basement	16	B6 - Main feedwater pump 1B discharge piping south towards 15 feedwater heaters up to the reducer	G-G	Horiz.	41.60	5-4, F-F
TB	M-247	Basement	22	B2 - 22-inch header for main feedwater discharge piping from reducer to T ant 15 heaters	G-G	Horiz.	7.79	4-4, F-F
TB	M-247	Basement	22	D2 - Centerline of T in 22-inch header east to reducing elbow	G-G	Horiz.	8.50	3-3, F-F
TB	M-247	Basement	16	D2 - From centerline of reducing elbow south through valve F3-1 to 90 degree reducing elbow to the west	G-G	Horiz.	17.04	3-3, F-F
TB	M-247	Basement	20	D2 - Straightline distance north through the two 90 degree elbows toward the 15A heater	G-G	Horiz.	5.00	3-3, F-F
TB	M-247	Basement	20	D2 - 20-inch piping north toward heater 15A	G-G	Horiz.	13.29	3-3, F-F
TB	M-249	Basement	20	C12 - Vertical piping from 20-inch horizontal pipe up to mezzanine floor toward heater 15A	H-H	Vert.	5.92	3-3, F-F
TB	M-249	Basement	20	C12 - Vertical piping from mezzanine floor to heater 15A inlet	H-H	Vert.	1.52	3-3, F-F
TB	M-249	Basement	14	F1 - Vertical 14-inch bypass piping from 20-inch horizontal pipe up to mezzanine floor toward heater 15A	H-H	Vert.	5.92	3-3, F-F
TB	M-249	Mez.	14	CF1 - Vertical piping from mezzanine floor to valve F11-1 inlet	H-H	Vert.	1.50	3-3, F-F
TB	M-247	Basement	22	D2 - Centerline of T in 22-inch header west to reducing elbow	G-G	Horiz.	11.00	3-3, F-F

Table A3-4: Piping Upstream of 15 Feedwater Heaters (cont.)

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-247	Basement	16	C2 - From centerline of reducing elbow south through valve F3-2 to 90 degree reducing elbow to the east	G-G	Horiz.	17.04	3-3, F-F
TB	M-247	Basement	20	C2 - Straightline distance south through the two 90 degree elbows toward the 15B heater	G-G	Horiz.	5.00	3-3, F-F
TB	M-247	Basement	20	C2 - 20-inch piping north toward heater 15B	G-G	Horiz.	13.29	3-3, F-F
TB	M-249	Basement	20	C12 - Vertical piping from 20-inch horizontal pipe up to mezzanine floor toward heater 15A	H-H	Vert.	5.92	3-3, F-F
TB	M-249	Basement	20	C12 - Vertical piping from mezzanine floor to heater 15A inlet	H-H	Vert.	1.52	3-3, F-F
TB	M-249	Basement	14	F1 - Vertical 14-inch bypass piping from 20-inch horizontal pipe up to mezzanine floor toward heater 15A	H-H	Vert.	5.92	3-3, F-F
TB	M-249	Mez.	14	CF1 - Vertical piping from mezzanine floor to valve F11-1 inlet	H-H	Vert.	1.50	3-3, F-F
						Linear FT on Basement	**355.17**	
						Linear FT on Mez. Level	**3.00**	
						Total Length (Linear FT)	**358.17**	

Table A3-5: Piping Downstream of 15 Feedwater Heaters

BLDG	Dwg. No.	Building Level	Nom. Dia. (In)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
PIPING LOCATED BETWEEN 15 HEATERS AND FW-7A AND FW-7B IN TURBINE BUILDING								
TB	M-249	Mez.	18	B9 - Outlet of heater 15A up to 90 degree elbow to the south	G-G, H-H	Vert.	7.04	F-F, 3-3
TB	M-247	Mez.	18	D2 - Horizontal piping running to the south from the outlet of heater 15A	G-G, H-H	Horiz.	9.63	F-F, 3-3
TB	M-247	Mez.	18	D2 - Horizontal distance east through the 90 degree elbows on heater 15A outlet piping	G-G, H-H	Horiz.	5.00	F-F, 3-3
TB	M-247	Mez.	18	D2 - Horizontal distance north to the reducing elbow to the west	G-G, H-H	Horiz.	13.38	F-F, 3-3
TB	M-247	Mez.	22	D2 - Horizontal distance from the reducing elbow west toward turbine building wall to elbow up	G-G, H-H	Horiz.	28.16	F-G, 3-3
TB	M-249	Mez.	14	F1 - Heater 15A bypass line from F11-1 to 18-inch pipe on heater outlet	G-G	Vert.	12.54	F-F, 3-3
TB	M-249	Mez.	18	B9 - Outlet of heater 15B up to 90 degree elbow to the south	G-G, H-H	Vert.	7.04	F-F, 3-3
TB	M-247	Mez.	18	C2 - Horizontal piping running to the south from the outlet of heater 15B	G-G, H-H	Horiz.	9.63	F-F, 3-3
TB	M-247	Mez.	18	C2 - Horizontal distance east through the 90 degree elbows on heater 15B outlet piping	G-G, H-H	Horiz.	7.00	F-F, 3-3
TB	M-247	Mez.	18	D2 - Horizontal distance north from elbow to intersection of 45 degree pipe to 22-inch header	G-G, H-H	Horiz.	6.38	F-F, 3-3
TB	M-247	Mez.	18	D2 - *Horizontal distance northwest of* heater 51B outlet piping angling at 45 degrees into 22-inch header	G-G, H-H	Horiz.	9.90	F-F, 3-3
TB	M-249	Mez.	14	A1 - Heater 15B bypass line from F11-2 to 90 degree elbow angling toward 22-inch header	G-G	Vert.	12.54	F-F, 3-3
TB	M-247	Mez.	14	C2 - Horizontal distance northwest of heater 15B bypass piping angling at 45 degrees into 22-inch header	G-G, H-H	Horiz.	3.36	F-F, 3-3

Table A3-5: Piping Downstream of 15 Feedwater Heaters

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
TB	M-249	Mez.	24	B8 - Vertical distance from centerline of 22-inch header to operating deck. Assumes that all pipe is 24 inches from elbow on	G-G	Vert.	7.95	G-G, 3-3
TB	M-249	Oper.	24	B8 - Vertical distance from operating deck to 90 degree elbow into auxiliary building. Assumes that all pipe is 24 inches from elbow on	G-G	Vert.	11.00	G-G, 3-3
TB	M-247	Oper.	24	C2 - Horizontal distance north from T in vertical 24-inch header	G-G	Horiz.	4.42	G-G, 3-3
TB	M-247	Oper.	24	A2 - Vertical distance from centerline of line 224 to 90 degree elbow angling out from wall	A-A	Vert.	10.25	G-G, 3-3
TB	M-247	Oper.	24	C2 - Horizontal distance of pipe angling out 45 degrees from wall to header	G-G	Horiz.	7.42	G-G, 3-4
TB	M-247	Oper.	24	C2 - Horizontal distance north to elbow up	G-G	Horiz.	41.92	G-G, 4-5
TB	M-247	Oper.	24	B5 - Vertical distance from centerline of header through valve V38-8 to the 90 degree elbow to the south	N/A	Vert.	11.08	G-G, 5-5
TB	M-247	Oper.	22	C2 - Horizontal distance from centerline of valve F38-8 south to elbow down	G-G	Horiz.	64.77	G-G, 5-3
TB	M-249	Oper.	22	B8 - Vertical distance from centerline of header to centerline of valve V38-9	N/A	Vert.	19.42	G-G, 3-3
TB	M-247	Oper.	22	C2 - Horizontal distance from centerline of header pipe down to the reducing elbow turning west	G-G	Horiz.	12.50	G-G, 3-3
TB	M-247	Oper.	22	C2 - Horizontal distance from centerline of F38-9 into header T downstream of F38-7	G-G	Horiz.	5.25	G-G, 3-3
TB	M-247	Oper.	22	C2 - Horizontal distance from centerline of F38-7 to auxiliary building wall	G-G	Horiz.	4.00	G-G, 3-3
						Linear FT on Oper. Deck	192.03	
						Linear FT on Mez. Level	139.53	

Table A3-5: Piping Downstream of 15 Feedwater Heaters

BLDG	Dwg. No.	Building Level	Nom. Dia. (in)	Drawing Coordinates/Description	Detail Section	Horiz./Vert. /Angle	Pipe Length (Linear FT)	Quad #,Letter (floor plan quads)
					Total Length (Linear FT)		331.56	

Appendix A

Initiating Events

Attachment 1 – Turbine Sump Alarm History

Turbine Building Sump Alarm History

Prepared by: ______________________ George E Baldwin
Signature Print Name

10/18/05
Date

Reviewed by: ______________________ JEFFREY T. STAFFORD
Signature Print Name

10-18-05
Date

Turbine Building Sump Alarm History

Annunciator 47033-P, *Miscellaneous Sump Level High*, represents three sumps, the Screenhouse sump (SER point 1593), Turbine Building sump (SER point 1594), and the Waste Area sump (SER point 1595). In review of the Sequence Event Recorder (SER) output from January 2003 through April 2005 (28 months), Annunciator 47033-P actuated on 70 days. This represents an alarm on the average of once every 12 days. There are periods of close to 2 months without an alarm and short periods with daily alarms. Alarms are frequently less than 1 minute and clear when operator acknowledges the annunciator. Of the 156 annunciator activations, 103 were at power. The annunciator was active for 1100 minutes with three times greater than 1 hour. The activations at-power average length of time was 11 minute but the three longer times account for 572 minutes. The average time, excluding the three long periods, is approximately 5 minutes per alarm. With the infrequency and length of time of the annunciator, the operators would respond in a timely fashion with concern if the alarm does not immediately clear.

Date	SER Point	IN	Out	Note
27-Feb-03	1594	0758	0758	
	1594	0758	0800	
	1594	0836	0836	
	1594	0836	0837	
	1594	0837	0840	
	1594	0840	0840	
	1594	0840	0845	
20-Mar-03	1593	1443	1443	
	1593	1443	1443	
	1593	1443	1443	
	1593	1443	1443	
	1593	1443	1443	
25-Mar-03	1594	0931	0932	
26-Mar-03	1594	0619	0619	
	1594	0619	0619	
	1594	0619	0619	
15-Apr-03	1593	1340	1343	First during Outage
20-Apr-03	1593	0903	1046	
06-May-03	1594	2328	0102	Day change
07-May-03	1594	0102	0154	
	1594	0102	0102	
	1594	0154	0154	
	1594	0154	0154	
	1594	0204	0230	
	1594	1633	2119	
08-May-03	1594	1239	1631	Day change
10-May-03	1594	1304	1311	Last during Outage
17-May-03	1593	0819	0824	
	1593	1859	1902	
18-May-03	1593	0039	0042	
15-Jun-03	1593	1129	1132	
	1593	2243	2246	
07-Jul-03	1593	0102	0104	
	1594	0953	0953	
22-Jul-03	1593	2244	2249	
23-Jul-03	1593	1309	1312	
24-Jul-03	1593	0549	0550	
27-Jul-03	1593	1819	1821	
29-Jul-03	1593	2100	2103	
30-Jul-03	1593	0521	0524	
20-Sep-03	1593	0512	0518	
	1593	1419	1423	
	1593	1729	1933	
25-Sep-03	1594	2113	2145	
	1594	2145	2149	
02-Oct-03	1594	1340	1340	
03-Dec-03	1593	1141	1143	
04-Dec-03	1594	0220	0220	
08-Dec-03	1593	0136	0141	
	1594	0326	0326	
09-Dec-03	1594	0144	0144	
	1593	0437	0441	
12-Dec-03	1593	0913	0916	
16-Dec-03	1593	2208	2212	
18-Dec-03	1593	0018	0021	

Date	SER Point	IN	Out	Note
22-Dec-03	1594	1241	1241	
25-Dec-03	1593	1136	1138	
28-Dec-03	1593	1400	1402	
	1593	1412	1415	
	1593	1724	1726	
29-Dec-03	1593	1243	1246	
	1593	2246	2249	
31-Dec-03	1593	1509	1513	
01-Jan-04	1593	0632	0635	
	1593	0924	0926	
10-Jan-04	1593	1723	1727	
13-Jan-04	1593	0341	0344	
13-Jan-04	1593	0645	0649	
	1593	2118	2121	
15-Jan-04	1593	1828	1832	First during Outage
	1593	1939	1943	
16-Jan-04	1593	0149	0153	
	1593	0533	0537	
	1593	1257	1300	
	1593	1412	1415	
28-Jan-04	1593	1245	2349	
	1593	2145	2148	
29-Jan-04	1593	0312	0318	
	1594	2256	2300	
	1594	2300	0038	Day change
30-Jan-04	1594	0038	0039	Last during Outage
	1594	0039	0039	
	1594	0039	0132	
31-Jan-04	1594	0303	0314	
	1594	0359	0406	
	1594	0508	0516	
	1593	0559	0603	
	1594	0802	0810	
01-Feb-04	1593	1647	1650	
02-Feb-04	1593	0053	0057	
	1593	0236	0240	
	1593	0921	0924	
	1593	1209	1213	
	1593	1430	1435	
	1593	1550	1556	
01-Mar-04	1594	1246	1251	
02-Mar-04	1594	0753	0753	
	1594	0759	0800	
	1594	0759	0759	
	1594	0946	0946	
	1594	0947	0947	
	1594	0947	0954	
02-Mar-04 Cont.	1594	0955	1007	
	1594	1043	1043	
	1594	1043	1044	
	1594	1107	1107	
	1594	1129	1132	
	1594	1152	1152	
	1594	1154	1154	
	1594	1156	1159	
	1594	1304	1304	
	1594	1311	1311	
	1594	1311	1311	
14-Jun-04	1594	0833	0833	
18-Jun-04	1593	0832	0832	
16-Aug-04	1594	1230	1230	
	1594	1230	1230	
	1594	1230	1230	
	1594	1230	1230	
	1594	1230	1230	
20-Sep-04	1594	1346	1414	
01-Oct-04	1594	1333	1633	
	1594	1655	1655	
	1594	1718	1758	
20-Oct-04	1593	2237	2332	First during Outage
02-Nov-04	1594	1209	1210	
	1594	1210	1213	
	1594	1313	1314	
	1594	1314	1314	
	1594	1314	1314	
	1594	1314	1332	
	1594	1314	1314	
	1594	1332	1421	
04-Nov-04	1594	1405	1405	
06-Dec-04	1594	0537	0610	Last during Outage
09-Feb-05	1595	1043	1044	
	1595	1044	1044	
	1595	1044	1044	
	1595	1049	1049	
	1595	1117	1117	
	1595	1117	1117	

Date	SER Point	IN	Out	Note
23-Feb-05	1593	2157	2158	First during Outage
	1593	2157	2157	
10-Mar-05	1593	0827	0834	SERIES
13-Mar-05	1594	1101	1104	
14-Mar-05	1594	0808	0808	
17-Mar-05	1593	0925	0925	
	1593	1018	1018	
22-Mar-05	1593	1739	1743	
02-Apr-05	1594	1451	1457	
03-Apr-05	1594	0626	0650	
04-Apr-05	1594	1225	1236	
05-Apr-05	1594	0629	0644	
06-Apr-05	1594	0616	0634	
09-Apr-05	1594	1546	1547	
	1594	1550	1550	

Appendix A

Initiating Events

Attachment 2 – Circulation Water Expansion Joint Rupture Frequency

Circulation Water Expansion Joint Rupture Frequency

Owner's Acceptance: Thomas G Hook

Signature

THOMAS G HOOK

Print Name

12/01/05

Date

EXPANSION JOINT FAILURE RATES FOR THE KEWAUNEE PRA

Final Report

Prepared for

Dominion Energy
Kewaunee Power Station

By

Karl N. Fleming
Bengt O.Y. Lydell

In Cooperation with
Maracor Software & Engineering, Inc.

Karl N. Fleming Consulting Services LLC
A California Limited Liability Company
616 Sereno View Road
Encinitas, CA 92024
United States of America

November, 2005

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1. INTRODUCTION

1.1 *Purpose*

The purpose of this report is to document the derivation of failure rates for different failure modes for rubber expansion joints of the type used in LWR circulating water systems. This work was performed via Subcontract to Maracor Software Engineering, Inc. on behalf of Dominion Energy's Kewaunee Power Station. This report is prepared to be an integral part of the overall turbine building internal flooding initiating events analysis.

1.2 *Scope*

The scope of work covered in this report includes:

- Development of failure rates and rupture frequencies for rubber expansion joints of the type used in the Kewaunee Circulating Water System
- Development of point estimates and probability uncertainty distributions for all parameters subject to data uncertainties

1.3 *Objectives*

The objective is to perform a state of the art data analysis that is consistent with the applicable requirements of ASME PRA Standard Capability Category II for data analysis and initiating event frequency development. Consistent with this objective, the report is intended to provide a traceable basis for the calculations so that the results could be independently reproduced from the information provided.

1.4 *Report Guide*

A major part of this report is devoted to the development of a set of failure rates and rupture frequencies for use in the internal flooding initiating event development. The technical approach to developing these failure rates and rupture frequencies is summarized in Section 2. In Section 3 the failure rates for rubber expansion joints are developed for different expansion joint failure modes including leakage and ruptures with flow rates less than 2,000 gpm, ruptures with flow rates greater than 2,000 and ruptures with flow rates greater than 10,000 gpm. Section 4 lists the references used as inputs to the data development and methodology. Supporting details are provided in the Appendices.

2. TECHNICAL APPROACH

2.1 *Overview*

The model used to estimate piping component failure frequencies for the initiating event models in this calculation is the same as that used in a recent EPRI report on internal flooding initiating event frequencies [1], and similar to that used in recent NRC studies regarding loss of coolant accident (LOCA) initiating event frequencies [2] [3]. The source of pipe failure and exposure data used to quantify the failure rates used in these models is known as "PIPExp-2004" [4]. A summary of this database is provided in Appendix A.

2.2 *Uncertainty Treatment*

Uncertainties in these failure rates were quantified using a Bayes' methodology that was developed in the EPRI RI-ISI program [5] and approved by the NRC for use in applied RI-ISI evaluations [6]. An independent review of this pipe failure rate uncertainty treatment was performed to support the NRC Safety Evaluation and results of this favorable review are provided in Reference [7]. An earlier EPRI report [8] developed a set of pipe failure rates for use in the EPRI RI-ISI applications which was also approved and independently reviewed in References [6] and [7]. These earlier failure rate estimates were derived from a pipe failure database that had been developed in Reference [10]. During subsequent work in applying these estimates in applied RI-ISI evaluation, a significant number of data classification errors in the original data source [10] were identified and improved estimates of the exposure population became available. These factors, as discussed more fully in Reference [9], were the prime motivation to switch to the more comprehensive and validated "PIPExp-2004" database when Reference [1], was developed. The most recent NRC sponsored work on LOCA frequencies [3] is also based in part on the "PIPExp-2004" database.

2.3 *Component Rupture Model*

The model used for relating failure rates and rupture frequencies for piping components uses the following simple model that is widely used in piping reliability assessment and was used in recent updates of recommended Loss of Coolant Accident frequencies [6]. The failure modes included in the estimation of failure rates include leaks and ruptures and, in some cases, cracks may also be included depending on the application. The model is expressed in the following equation:

$$\rho_{ijx} = \sum_{k=1}^{M} \rho_{ijkx} = \sum_{k=1}^{M} \lambda_{ijk} P_{ijk}\{R_x|F\} \qquad (2.1)$$

Where:

ρ_{ijx} = total rupture frequency of rupture size x for pipe size i in system j

ρ_{ijkx} = rupture frequency of rupture size x for pipe of size i in system j due to damage mechanism k

λ_{ijk} = failure rate of pipe of size i in system j due to damage mechanism k

$P_{ik}\{R_x|F\}$ = conditional probability of rupture size *x* given failure for pipe size i in system j and damage mechanism k

M = Number of different damage mechanisms

In general, a point estimate of the frequency of pipe failures, λ_{ijk}, is given by the following expression:

$$\lambda_{ijk} = \frac{n_{ijk}}{f_{ijk} N_{ij} T_{ij}} \tag{2.2}$$

Where

n_{ijk} = the number of failures (cracks, wall thinning, leaks and ruptures) events for pipe size i in system j due to damage mechanism k

T_{ij} = the total time over which failure events were collected for pipe size i in system j

N_{ij} = the number of components that provided the observed pipe failures for size i in system j

f_{ijk} = the fraction of number of components of size i in system j that are susceptible to failure from damage mechanism k for conditional failure rates given susceptibility to damage mechanism k, 1 for unconditional failure rates

Note that all failure modes that result in pipe repair are included in the failure rate and that all failures thus defined are regarded as precursors to rupture. Some events that have no evidence of leakage are screened out prior to the calculation of failure rates. The events counted as ruptures are based on a specific definition of rupture which is application specific. For internal flooding applications, we seek unconditional failure rates and hence we can combine these equations under the condition: $f_{ijk} = 1$ to obtain the following expression for the point estimate of the rupture frequency.

$$\rho_{ijx} = \sum_{k=1}^{M} \rho_{ijkx} = \sum_{k=1}^{M} \lambda_{ijk} P_{ik}\{R_x|F\} = \sum_{k=1}^{M} \frac{n_{ijk}}{N_{ij} T_{ij}} P_{ik}\{R_x|F\} \tag{2.3}$$

In the development of Bayes' uncertainty distributions for these parameters, prior distributions are developed for the parameters λ_{ijk} and $P_{ik}\{R/F\}$ and these prior distributions are updated using the evidence from the failure and exposure data as in standard Bayes' updating. The exposure terms (denominator of the fractions on the right hand side of Equation (2.3) also have uncertainty as these terms must be estimated for the entire nuclear industry that provides the number of failures for the failure rate estimation. This uncertainty is treated in this process by adopting three hypotheses about the values of the exposure terms which requires three Bayes updates for each failure rate. The resulting posterior distributions for each parameter on the right hand side of Equation (2.3) are then combined using Monte Carlo sampling to obtain uncertainty distributions for the pipe rupture frequencies. A picture of this process is shown in Figure 2-1. This flow chart shows the full treatment of uncertainty needed for the RISI formulation in Equation (2.2). For the internal flooding formulation of Equation (2.3) the damage mechanism susceptibility fractions (f_{ijk}) do not come into play. The specific way in which this flow chart is applied is discussed in Section 4 for each system and failure mode.

In Reference [1] rupture frequencies were developed for three rupture sizes that were selected to support internal flooding analysis. These sizes include water sprays with flood rates of up to 100 gpm, flooding with flood rates of 100 to 2000 gpm, and major flooding with flood rates greater than 2000 gpm. For the Kewaunee internal flooding models, a somewhat different rupture size model had to be developed as the criteria for producing the consequences of interest are based on specific rupture sizes that were determined to produce the assumed flooding consequences.

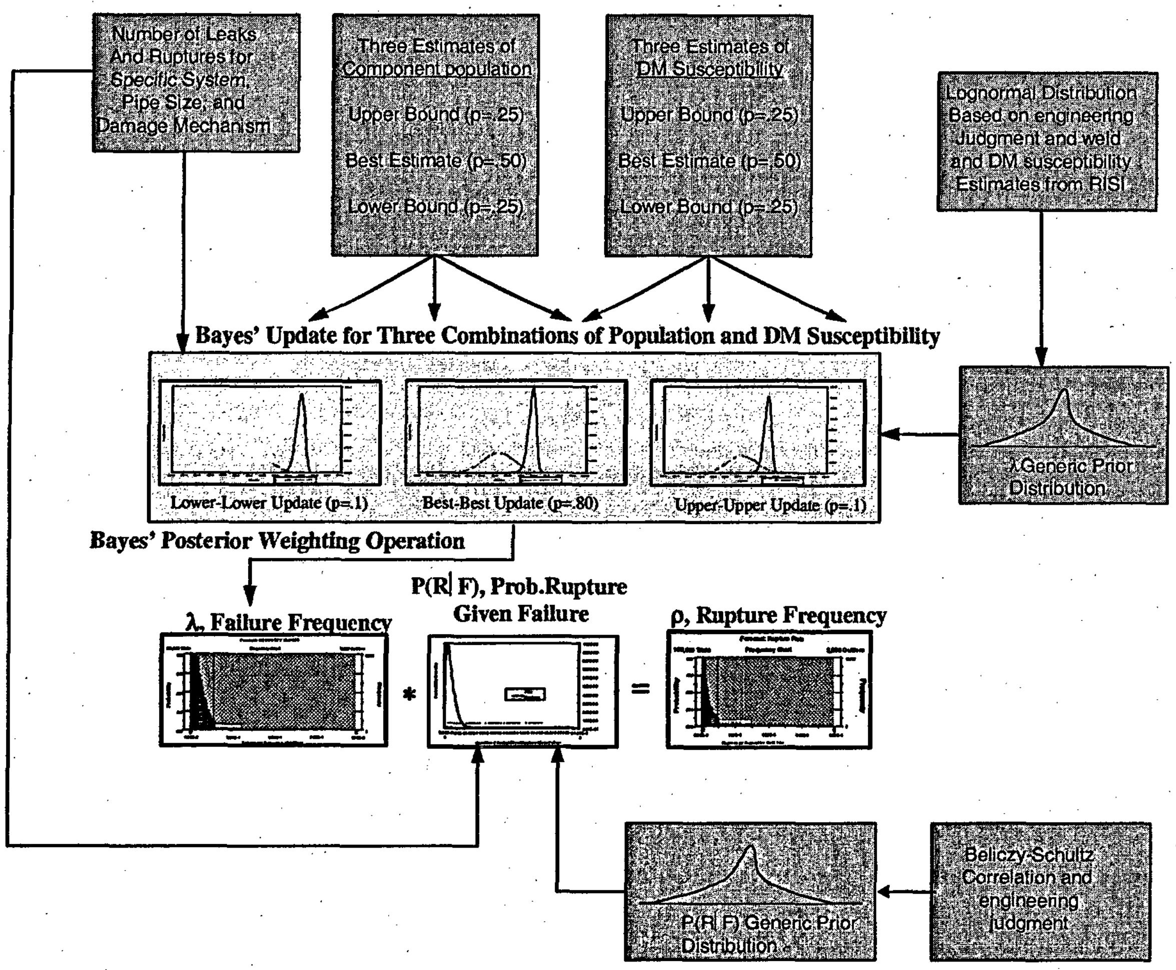

Figure 2-1 Flow Chart for Bayes' Estimates of System, Size, and Damage Mechanism Specific Pipe Failure Rates (λ) and Rupture Frequencies (ρ)

2.4 *Definition of Expansion Joint Failure Mode Cases*

Failure rates are developed in this report for the following cases.

1. Total failure rate for all failure modes involving leaks and ruptures
2. Rupture frequency for leaks and ruptures with flow rates less than or equal to 2,000 gpm
3. Rupture frequency for ruptures with flow rates greater than 2,000 gpm
4. Rupture frequency for ruptures with flow rates greater than 10,000

Note that Case 3 is inclusive of Case 4, i.e. Case 3 includes leak flows less than, equal to, and greater than 10,000 gpm.

3. CIRCULATING WATER SYSTEM EXPANSION JOINT FAILURE RATES

3.1 *Background*

Failure rates and rupture frequencies for circulating water system expansion joints were developed in Reference [1] by the authors of this report. The results obtained in that analysis are summarized in Table 3-1 reproduced from Table A-35 in Reference [1] below.

Table 3-1 CW-Expansion Joint Failure Rate & Rupture Frequency (Reproduced from Table A-35 from Reference [1])

Component & Failure Mode		Uncertainty Distribution [1/EXJ.YR]			
Type	**Failure Mode**	**Mean**	**5th Percentile**	**Median**	**95th Percentile**
CW Rubber EXJ	Spray	1.11E-04	4.13E-05	9.51E-05	2.38E-04
CW-Rubber EXJ	Major Flooding	1.49E-05	3.92E-06	1.20E-05	3.61E-05

The evidence used to develop these results consisted of the following:

The failure rate for sprays was based on 4 events involving LWR circulating water system rubber expansion joint failures that occurred at Comanche Peak (1 event), LaSalle (1 event), and Catawba (2 events). The exposure term was estimated based on 2899 LWR reactor years of service data in the PIPExp database through 2004 and an estimate of 12 rubber expansion joints per LWR circulating water system.

The prior distribution used for the analysis was based on the failure rate developed in the Oconee PRA [19] whose mean value is 2.5×10^{-4} per component year and a range factor of 100 was assumed.

To estimate the conditional probability of rupture for major flooding given failure, which was defined in Reference [1] to be a rupture with flooding in excess of 2,000 gpm, a larger population of expansion joint failure events covering different systems and including an HTGR event at Ft. St. Vrain was developed. This population had a total of 35 events including the 4 events used in the above described failure rate calculation. One of these ruptures was the LaSalle event considered in the failure rate calculation and the other 3 ruptures in this population occurred at Ft. St. Vrain, Beaver Valley, and Comanche Peak. Note that the information presented in Reference [1] did not identify the 3 rupture events other than the one at LaSalle. Also note that the events at Beaver Valley and Comanche Peak were not in the circulating water service system nor were 31 of the 35 events considered in this larger population of expansion joint failures. The approach of specializing the failure rate data to the system of interest and then using a larger sample size for the conditional rupture probability is consistent with the approach used in Reference [1] for all the piping system failure rates that were developed.

3.2 *Revised Data Analysis*

In the current study, more information was collected on the events that were analyzed in Reference [1] and additional expansion joint events were identified. This was accomplished by augmenting the data queries from the PIPExp database that was used in Reference [1] by consulting additional sources including those of References [20] through [34]. In addition to the 35 events used in Reference [1], the revised analysis included an additional 7 events that are summarized in Table 3-2. As a result of this additional information, the 43 events in Table 3-2 are analyzed as follows by tabulating the assessments in the last column of Table 3-2.

Total No. Events	42
Events screened out due to non-leakage	6
Events involving leaks or ruptures in LWR Circ. water systems	5
Events involving Leaks	27
Events involving Ruptures with leak flows < 2,000gpm	6
Events involving Ruptures with leak flows 2,000-10,000gpm	2
Events involving Ruptures > 10,000 gpm	1
Total Number of Faillures (leaks + ruptures)	36

3.3 *Revised Expansion Joint Failure Rate*

Consistent with the methodology adopted in Reference [1], the failure rate for the circulating water system expansion joints is based on data from LWR circulating water systems only and does not include data from expansion joints in other systems. This approach is used to capture system specific factors that may impact the degradation mechanism responsible for failure and is expected to influence the likelihood of failure. In the Reference [1] analysis of expansion joint failure rates 4 events were classified as LWR Circulating water system failures and in the revised analysis this is increased to 5. The assumptions regarding population exposure and the assumed prior distribution are unchanged from Reference [1] in this revised analysis: namely that there are on the average of 12 circulating water expansion joints per reactor unit and that the reactor years of exposure data that was estimated in Reference [1] of 2899 reactor operating years is still valid. According to the methodology described in Section 2, uncertainty in this exposure term estimate was accounted for by admitting hypotheses that these estimates are 50% higher and 50% lower than this best estimate. The net result of the revised circulating water system failure counts are an increase in the assessed failure rate by a factor of 1.25. The resulting failure rate distribution is presented in Table 3-3 and this distribution was used for each of the different failure mode cases described below.

3.4 *Revised Conditional Probability of Rupture*

Consistent with the methodology adopted in Reference [1], the data set for the estimation of conditional rupture probabilities is expanded to include expansion joints in other systems because this parameter is viewed to be more a function of the properties of the component than is the case with the failure rate. System specific factors may influence the failure rate, but the conditional probability of rupture given failure is not expected to vary from system to system. The use of different data sets for the failure rate and the conditional probability of rupture given failure is consistent with the pipe failure data handling methodology that was developed for the EPRI RI-ISI evaluations in Reference [8] and approved by the NRC in Reference [7]. In that reference, data from different systems was pooled to support the conditional rupture probabilities, but such pooling was not performed for the failure rates. The motivation is to have a statistically significant sample size for each parameter. More discussion on this point can be found in Reference [9].

In the Reference [1] analysis there was one rupture case developed for ruptures with leak flows greater than 2,000 gpm. The evidence for that analysis was four rupture events in 35 failure events. In the updated analysis there is a total of 36 events involving leaks and ruptures. For for leak flows greater than 2,000 gpm, there are 3 events including 2 events with leak flows between 2,000 gpm and 10,000 gpm and 1 event with leak flow greater than 10,000. One of the events classified as rupture in the Reference [1] analysis at Comanche Peak was determined by contacting Plant personnel (Reference [33]) to be a leak with a flow rate substantially less than 2,000 gpm which flooded a small room over a protracted period of time and hence in this revised analysis it was classified as a leak. The remaining 3 ruptures in Reference [1] remain so classified here. The most severe was the expansion joint rupture at Ft. St. Vrain which had a reported leak rate of 15,000 gpm.

So the evidence for the conditional probability of rupture is 3 events out of 36 failure events involving leak flows greater than 2,000 gpm, 2 events out of 36 failure events for leak flows between 2,000 gpm and 10,000 and 1 event out of 36 failure events involving leak flows greater than 10,000.

In the Reference [1] analysis a Beta distribution was used to characterize the uncertainty in the conditional probability of rupture. The prior distribution was assumed to be a flat prior indicating a non-informative state of knowledge. Given the current knowledge based on the Reference [1] results we now know that the vast majority of expansion joint failures are leaks and not major ruptures. In the revised analysis, the A and B parameters for the prior Beta distribution are set at 1 and 9, respectively consistent with an assumed mean conditional rupture probability of 0.1. This yields the following mean values of the Bayes' updated Beta Distribution.

$$Mean\{Rupture > 2{,}000gpm\} = \frac{A_{Posterion}}{A_{Posterior} + B_{Posterior}} = \frac{(A_{Prior}+3)}{(A_{Prior}+3)+(B_{Prior}+33)} = \frac{(1+3)}{(1+3)+(9+33)} = .087$$

$$Mean\{Rupture 2{,}000 to 10{,}000 gpm\} = \frac{A_{Posterion}}{A_{Posterior} + B_{Posterior}} = \frac{(A_{Prior} + 2)}{(A_{Prior} + 2) + (B_{Prior} + 35)} = \frac{(1+2)}{(1+2)+(9+35)} = .065$$

$$Mean\{Rupture > 10{,}000 gpm\} = \frac{A_{Posterion}}{A_{Posterior} + B_{Posterior}} = \frac{(A_{Prior} + 1)}{(A_{Prior} + 1) + (B_{Prior} + 35)} = \frac{(1+1)}{(1+1)+(9+35)} = .044$$

3.5 *Results*

The results using the methodology of Reference [1] and summarized in Section 2 were obtained using Crystal Ball and yielded the failure rates and rupture frequencies for the Circulating Water system rubber expansion joints given in Table 3-3. The frequency distribution from the Monte Carlo analysis for the expansion joint rupture frequency with leak flows greater than 10,000 gpm is shown in Figure 3-1.

3.6 *Plan to Update EPRI Report of Reference [1]*

This revised analysis of Circulating Water System Expansion Joint failure rates and rupture frequencies will be included in an update of the EPRI internal flooding frequency report of Reference [1] which will be published in early 2006.

Table 3-2 Analysis of Events Involving Expansion Joint Failures

Plant	Date	Data Source	Event Description	System	Classification
ANO 2	3/10/1992	Reference [23]	30-in. expansion joint condensate pump. Expansion joint inspected and found to have pinholes because of aging. Evidence of leakage	Condensate (CND)	Leak
Beaver Valley	10/15/1990	Reference [23]	A SW expansion joint collapsed. Information was obtained directly from Beaver Valley (January 2000) that documented expansion joint seepage amounting to little more than a wet spot along an intermittent four foot long crack at the base of an arch. SW expansion joint was found deformed during routine operator rounds. An analysis of the expansion joint could not demonstrate its operability. The expansion joint failed due to the effect of water hammer / column separation. This condition was attributed to the failure of the pump's discharge vacuum breaker to open following the pump's shutdown. Vacuum relief is required due to the design of the system. Evidence of leakage	Service Water (SW)	Leak
Beaver Valley 1	12/18/1995	LER 1995-010	A 24-inch diameter, 2-foot long SW expansion joint ruptured because of erosion of tube. Steel belt was corroded. The expansion joint was 10 years old. The rupture was approximately 4.5 inches by 3 inches. Approximately 40,000 gallons of water spilled in a very short time. The failure was attributed to erosion of the inner rubber wall which caused corrosion of the expansion joint belts. (Reference [28]).	SW	Rupture > 2,000 gpm
Beaver Valley 2	8/11/1986	Reference [23]	30-in. condensate pump suction expansion joint deformed and partially collapsed, no leakage.	CND	Screened out; no leak
Beaver Valley 2	8/9/1991	Reference [23]	30-in. suction expansion joint for condensate pump found to have pinhole leak.	CND	Leak
Brunswick 1	1/4/1990	Reference [23]	Rupture of screen wash pump expansion joint.	Screen Wash pump (SWP)	Rupture < 2,000gpm
Byron 1	11/19/1991	Reference [23]	Condensate pump expansion joint cover found torn 180 degrees. Evidence of leakage	CND	Leak
Calvert Cliffs 1	2/1/1996	Reference [23]	24-in. suction side of condensate pump expansion joint - tube imploded and core of joint visible. Evidence of leakage	CND	Leak
Catawba-1	1/1/2001	IR 50-413/2001-02	CW Expansion joint leakage	Circ. Water Systrem (CWS)	Leak
Catawba-2	1/1/2001	IR 50-	CW Expansion joint leakage; contact with Catawba plant personnel	CWS	Screened out as a

Plant	Date	Data Source	Event Description	System	Classification
		414/2001-02	revealed that this event is the same event as noted above for Catawba-1 and is not counted separately. (Reference [30])		separate event
Clinton 1	8/15/1989	Reference [23]	Inlet expansion joint to condenser leaking (1 cup/hr.), replaced expansion joint later outage.	CWS	Leak
Clinton 1	3/18/1990	Reference [23]	Condenser water box over-pressurized, damage to water box expansion ,joint, no leakage.	CND	Screened out; no leak
Comanche Peak 1	6/6/1993	Reference [23]	Comanche Peak Unit 1 was at 85% power when rubber expansion joint on the circulation lube water pump discharge leaked resulting in six feet of water in the circulating discharge room over a period of time. Expansion joint failed under vacuum when the pump was stopped. Failure attributed to normal aging. Circulation pump 02 rubber expansion joint was replaced and the unit brought to 100% power. This event involved leakage but not a catastrophic rupture (Reference [33]).	Auxiliary System (AUX)	Leak
Crystal River 3	8/14/1985	Reference [23]	10-in., SW pump suction expansion joint failed - hole in joint, aging (12 years). A crack in the joint was discovered to be weeping. The expansion joints were replaced (Reference [26]). Evidence of leakage	SW	Leak
D.C. Cook 1	7/29/1990	Reference [23]	8-inch diameter expansion joint header connected to the Condensate Storage Tank leaked because of 4-in. gash in the joint, cause unknown. Expansion joint is composed of fabric and is two-feet long. Fabric 2' long.	CND	Leak
D.C. Cook 2	8/28/1987	Reference [23]	Expansion Joint XJ-54N in the discharge header west essential service water pump ruptured. Most probable cause of failure attributed to time / function degradation aggravated by the impure quality of the raw water in this system. Replaced expansion joint. A phone conversation with a cognizant engineer indicated the expansion joint is 8 inches in diameter. Nominal operating pressure is 80 psi. During the summer, at the time of this event, operating pressure was likely higher, at 85 to 90 psi. The expansion joint is in a line that serves as a minimum flow path during normal operations. The flow through the joint during normal operations is typically 2000 gpm (Reference [34]).	SW	Rupture < 2,000gpm

Plant	Date	Data Source	Event Description	System	Classification
Diablo Canyon 1,2	12/1/1998	Reference [23]	Two cooling water system synthetic expansion joints experienced catastrophic failures. One failure caused a 500 to 1000 pgm leak from a 16" elastomeric expansion joint in a connection between screenwash water system and a pipe embedded in the intake building wall that is connected to the turbine building cooling system (a configuration allowed by procedure). Eighteen hours later, a 6" elastomeric expansion joint in the closed loop intake cooling water system (cools the circulating water pump motors) occurred. These expansion joints are installed in locations not obviously visible and involve a wet (saltwater) location. Root cause was degradataion due to corrosion of metal in the joint; saltwater introduction from the exterior of the joint. Expansion joints were 23 years old and were not part of the inspection program. Expansion joints are designed similar to a tire, inside out tire waterproof internal, cotton core that wraps it for pressure control; steel stiffening rings; protective cover had breached	SWP	2 ruptures < 2000 gpm
Ft. St. Vrain 1[b]	4/7/1988	LER 88-006	CW expansion joint failed because of degradation, 54", 15 years old; 15" tear that resulted in a 15,000 gpm leak flow (Reference [22]).	CWS	Rupture > 10,000 gpm
Hatch 1	2/27/1977	Reference [35]	A forced shutdown resulted from a recirculation pump trip followed by a condensor bellows rupture; not clear if this is a rubber expansion joint	CND	Rupture < 2,000gpm
Indian Point 3	6/15/1988	Reference [23]	30-in. condensate pump expansion joint leaking.	CND	Leak
Indian Point 3	12/8/1992	Reference [23]	30-in. expansion joint on suction side of condensate pump had minor leakage (excessive forces).	CND	Leak
Indian Point 3	1/14/1993	Reference [23]	Expansion joint on suction side of condensate pump deformed and leaked because of degradation.	CND	Leak
Indian Point 3	2/3/1993	Reference [23]	Expansion joint on suction side of condensate pump deformed and leaked because of degradation.	CND	Leak
LaSalle 1 &2	5/31/1985	LER 1985-045	108-in. cir. water pump expansion joint failed resulting in flooding '(2000 gpm), due to water hammer (LER 50-302/1989-011). Failure occurred in the Lake Screen House.	CWS	Rupture > 2,000 gpm
Limerick 1	1/3/1998	Reference [23]	Leak in expansion joint on suction side of condensate pump.	CND	Leak

Plant	Date	Data Source	Event Description	System	Classification
Limerick 2	12/15/1998	Reference [23]	Limerick Unit 2 while at full load found a leak on the ESW EXJ after starting the ESW pump. A small V shaped tear was observed at approximately 6:00 outside of the direct flow path. A failure analysis was performed on the EXJ after removal and it was concluded that the leak was due to an age related (end of life) failure. The failure worked its way through the expansion joint's two plies and an arc of approximately 120 degrees before exiting the outer protective layer.	SW	Leak
Millstone 2	4/28/1977	Reference [23]	24"-in SW pump expansion joint had ballooned out because of leakage thru tube.	SW	Leak
Oyster Creek		Reference [35]	After the ESW pump "A" was started during a surveillance test, the rubber expansion joint ruptured. The reactor was shutdown for refueling	SW	Rupture < 2,000gpm
Sequoyah 1	8/25/1994	Reference [23]	Main feedwater pump turbine condenser pump expansion joint developed leak because of high temp.	CND	Leak
St. Lucie 1	7/28/1986	Reference [23]	30-inch intake cooling water system, rubber steel-reinforced expansion joint on the intake cooling water system (essentially equivalent to a SW system) pump outlet developed a leak (<100gpm). Failure was caused by aging and cyclic fatigue. The intake cooling water system is a sea water system that cools component cooling water and turbine cooling water, and is essentially equivalent to a SW system [CORRESP07]. Evidence of leakage.	SW	Leak
St. Lucie 1	2/18/1990	Reference [23]	Intake 30-in. cooling water expansion joint had through wall leak, aging.	SW	Leak
Turkey Point 3	6/14/1988	Reference [23]	30-in. suction expansion joint for condensate pump leaking.	CND	Leak
VC Summer	2/4/1991	Reference [23]	36-in. expansion joint on suction side of condensate pump had a hole in liner with small leakage.	CND	Leak
VC Summer	2/14/1991	Reference [23]	36-in. expansion joint on suction side of condensate pump developed leak, cyclic fatigue.	CND	Leak
VC Summer	3/30/1993	Reference [23]	36-in. expansion joint on suction side of condensate pump developed leak, cyclic fatigue.	CND	Leak

Plant	Date	Data Source	Event Description	System	Classification
Surry 2	6/17/1986	LER 1986-06	While at 100% power, operations personnel discovered a service water leak in Unit 2 containment. The leak of approximately one gpm was in an expansion joint on the service water return line from a recirculation spray heat exchanger. The inlet and outlet service water valves were closed to isolate the leak. It was determined that repairs could not be made with the 72 hour LCO, therefore a rampdown was commenced and an unusual event was declared. The leak in the expansion joint was caused by galvanic corrosion.	SW	Leak
Beaver Valley 1	2/2/1996	Reference [20]	Load reduction to 90% for expansion joint replacement at the outlet of the D waterbox of the main condenser.	CWS	Screened out; no leak
Susquehanna 1	3/29/1989	Reference [29]	Manual shutdown due to circulation water system expansion joint leak - early refueling. Expansion joints leaked starting with minor dripping to ultimately about 1 gpm. Expansion joints are steel reinforced. During prior refueling outage, wear was noted on the interior of the joints (Reference [29]).	CWS	Leak
Catawba 1	6/12/1993	Reference [30]	Shutdown to repair expansion joint leakage. This event involved leakage of a rubber expansion joint reinforced by stainless steel (Reference [30]). Contributing causes: wear and arrangement.	CWS	Leak
River Bend 1	2/14/1989	Reference [20]	Load reduction to repair waterbox B expansion joint. No evidence of leakage	CWS	Screened out; no leak
Catawba 2	10/21/2001	Reference [20]	Refuel outage delay for condenser circulating water expansion joint repair. No evidence of leakage	CWS	Screened out; no leak
Clinton 1	5/9/1990	LER 1990-010	The unit was shutdown when diesel generators (Division 1 & 2) were declared inoperable because expansion joints in the shutdown service water system piping for the DG heat exchangers did not have required tie rods installed to prevent expansion beyond design limitations. The cause was a construction/installation error. The expansion joints are installed between the SWS piping and the DG heat exchangers to isolate vibratory motion between SWS piping and the DGs. The tie rods should have been located between the flanges of the expansion joints to prevent their expansion beyond design limitations.	SW	Screened out; no leak

Table 3-3 Uncertainty Distribution Results for Expansion Joint Failure Rates

Component & Failure Mode		Uncertainty Distribution [1/EXJ.YR]			
Type	**Failure Mode**	**Mean**	**5th Percentile**	**Median**	**95th Percentile**
CWS Rubber Expansion Joint	Failures (leaks + ruptures)	1.40E-04	5.69E-05	1.23E-04	2.84E-04
	Ruptures with leak flows > 2,000 gpm	1.22E-05	2.92E-06	9.75E-06	2.97E-05
	Ruptures with Leak flows from 2,000 to 10,000 gpm	9.17E-06	1.82E-06	7.10E-06	2.33E-05
	Ruptures with leak flows > 10,000 gpm	6.08E-06	8.81E-07	4.44E-06	1.67E-05

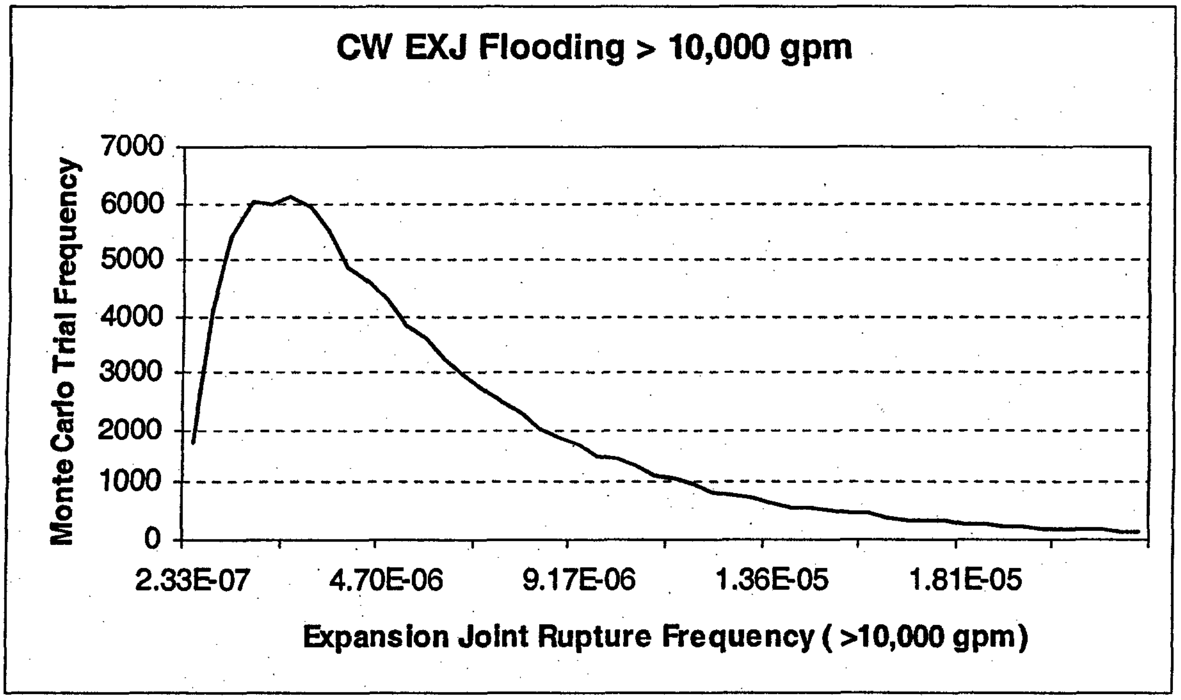

Figure 3-1 Uncertainty Distribution Expansion Joint Rupture Frequency (>10,000 gpm)

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Appendix A

Initiating Events

Attachment 3 – High Energy Line Break Report

High Energy Line Break Report

Owner Acceptance: THOMAS G HOOK THOMAS G. HOOK

Signature Print Name

10/27/2005

Date

HIGH ENERGY LINE BREAK INITIATING EVENT FREQUENCIES FOR THE KEWAUNEE PRA

Final Report

Prepared for

Dominion Energy
Kewaunee Power Station

By

Karl N. Fleming
Bengt O.Y. Lydell

In Cooperation with
Maracor Software & Engineering, Inc.

Karl N. Fleming Consulting Services LLC
A California Limited Liability Company
616 Sereno View Road
Encinitas, CA 92024
United States of America

October, 2005

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1. INTRODUCTION

1.1 *Purpose*

The purpose of this report is to document the derivation of initiating event frequencies that will be used as input to the turbine building internal flooding risk assessment at Dominion Energy's Kewaunee Power Station. Specifically, initiating event frequency values associated with ruptures of high-energy lines that in turn cause actuation of fire protection systems will be determined. This work was performed via Subcontract to Maracor Software Engineering, Inc. on behalf of Dominion Energy's Kewaunee Power Station. This report is intended to be an integral part of the overall turbine building internal flooding initiating events analysis.

1.2 *Scope*

The scope of work covered in this report includes:

- Development of pipe failure rates and rupture frequencies for high energy piping (i.e. piping with water or steam above saturation temperature) in PWR plants including the following systems:
    - steam, including high pressure, low pressure, and extraction steam systems
    - feedwater system, including feedwater heaters and drain systems
    - condensate system
- Development of point estimates and probability uncertainty distributions for all parameters subject to data uncertainties
- Calculation of Kewaunee HELB initiating event frequencies including point estimates and probability uncertainty distributions based on information provided by Kewaunee and Maracor on initiating event success criteria and piping lengths

1.3 *Objectives*

The objective is to perform a state of the art data analysis that is consistent with the applicable requirements of ASME PRA Standard Capability Category II for initiating event frequency development. Consistent with this objective, the report is intended to provide a traceable basis for the calculations so that the results could be independently reproduced from the information provided.

1.4 *Report Guide*

A major part of this report is devoted to the development of a set of failure rates and rupture frequencies for use in the turbine building HELB-initiated internal flooding initiating event development. The technical approach to developing these failure rates and rupture frequencies is summarized in Section 2. In Section 3 the HELB-initiated internal flooding initiating event models for the Kewaunee PRA are described including the success criteria for screening pipe locations and break sizes that apply to each event. The details of the development of the break sizes and locations for these events and break sizes are provided in Section 3.2 in the turbine building internal flooding initiating events analysis, into which this report is to be integrated. The information in Section 3 of this report is based on information in Section 3.2 of the main report and was provided to the authors by Kewaunee and Maracor. The development of failure rates and rupture frequencies for this model using the methodology of Section 2 is documented in Section 4. The results for the initiating event frequencies including point estimates and uncertainty distributions are summarized in Section 5. Section 6 lists the references used as inputs to the data development and methodology. Supporting details are provided in the Appendices.

2. TECHNICAL APPROACH

2.1 *Overview*

The model used to estimate pipe break frequencies for the initiating event models in this calculation is the same as that used in a recent EPRI report on internal flooding initiating event frequencies [1], and similar to that used in recent NRC studies regarding loss of coolant accident (LOCA) initiating event frequencies [2] [3]. The source of pipe failure and exposure data used to quantify the failure rates used in these models is known as "PIPExp-2004" [4]. A summary of this database is provided in Appendix A.

2.2 *Uncertainty Treatment*

Uncertainties in these failure rates were quantified using a Bayes' methodology that was developed in the EPRI RI-ISI program [5] and approved by the NRC for use in applied RI-ISI evaluations [6]. An independent review of this pipe failure rate uncertainty treatment was performed to support the NRC Safety Evaluation and results of this favorable review are provided in Reference [7]. An earlier EPRI report [8] developed a set of pipe failure rates for use in the EPRI RI-ISI applications which was also approved and independently reviewed in References [6] and [7]. These earlier failure rate estimates were derived from a pipe failure database that had been developed in Reference [10]. During subsequent work in applying these estimates in applied RI-ISI evaluation, a significant number of data classification errors in the original data source [10] were identified and improved estimates of the exposure population became available. These factors, as discussed more fully in Reference [9], were the prime motivation to switch to the more comprehensive and validated "PIPExp-2004" database when Reference [1], was developed. The most recent NRC sponsored work on LOCA frequencies [3] is also based in part on the "PIPExp-2004" database.

2.3 *Pipe Rupture Model*

The model used for relating failure rates and rupture frequencies uses the following simple model that is widely used in piping reliability assessment and was used in recent updates of recommended Loss of Coolant Accident frequencies [6]. The failure modes included in the estimation of failure rates include leaks and ruptures and, in some cases, cracks may also be included depending on the application. The model is expressed in the following equation:

$$\rho_{ijx} = \sum_{k=1}^{M} \rho_{ijkx} = \sum_{k=1}^{M} \lambda_{ijk} P_{ijk}\{R_x|F\} \tag{2.1}$$

Where:

ρ_{ijx} = total rupture frequency of rupture size x for pipe size i in system j

ρ_{ijkx} = rupture frequency of rupture size x for pipe of size i in system j due to damage mechanism k

λ_{ijk} = failure rate of pipe of size i in system j due to damage mechanism k

$P_{ik}\{R_x|F\}$ = conditional probability of rupture size x given failure for pipe size i in system j and damage mechanism k

M = Number of different damage mechanisms

In general, a point estimate of the frequency of pipe failures, λ_{ijk}, is given by the following expression:

$$\lambda_{ijk} = \frac{n_{ijk}}{f_{ijk} N_{ij} T_{ij}} \tag{2.2}$$

Where

n_{ijk} = the number of failures (cracks, wall thinning, leaks and ruptures) events for pipe size i in system j due to damage mechanism k

T_{ij} = the total time over which failure events were collected for pipe size i in system j

N_{ij} = the number of components that provided the observed pipe failures for size i in system j

f_{ijk} = the fraction of number of components of size i in system j that are susceptible to failure from damage mechanism k for conditional failure rates given susceptibility to damage mechanism k, 1 for unconditional failure rates

Note that all failure modes that result in pipe repair are included in the failure rate and that all failures thus defined are regarded as precursors to rupture. The events counted as ruptures are based on a specific definition of rupture which is application specific. For internal flooding and HELB applications, we seek unconditional failure rates and hence we can combine these equations under the condition: f_{ijk} =1 to obtain the following expression for the point estimate of the rupture frequency.

$$\rho_{ijx} = \sum_{k=1}^{M} \rho_{ijkx} = \sum_{k=1}^{M} \lambda_{ijk} P_{ik}\{R_x|F\} = \sum_{k=1}^{M} \frac{n_{ijk}}{N_{ij} T_{ij}} P_{ik}\{R_x|F\} \tag{2.3}$$

In the development of Bayes' uncertainty distributions for these parameters, prior distributions are developed for the parameters λ_{ijk} and $P_{ik}\{R/F\}$ and these prior distributions are updated using the evidence from the failure and exposure data as in standard Bayes' updating. The exposure terms (denominator of the fractions on the right hand side of Equation (2.3) also have uncertainty as these terms must be estimated for the entire nuclear industry that provides the number of failures for the failure rate estimation. This uncertainty is treated in this process by adopting three hypotheses about the values of the exposure terms which requires three Bayes updates for each failure rate. The resulting posterior distributions for each parameter on the right hand side of Equation (2.3) are then combined using Monte Carlo sampling to obtain uncertainty distributions for the pipe rupture frequencies. A picture of this process is shown in Figure 2-1. This flow chart shows the full treatment of uncertainty needed for the RISI formulation in Equation (2.2). For the internal flooding and HELB formulation of Equation (2.3) the damage mechanism susceptibility fractions (f_{ijk}) do not come into play. The specific way in which this flow chart is applied is discussed in Section 4 for each system and failure mode.

In Reference [1] rupture frequencies were developed for three rupture sizes that were selected to support internal flooding analysis. These sizes include water sprays with flood rates of up to 100 gpm, flooding with flood rates of 100 to 2000 gpm, and major flooding with flood rates greater than 2000 gpm. For the Kewaunee HELB-initiated internal flooding models, a somewhat different rupture size model had to be developed as the criteria for producing the consequences of interest are based on specific rupture sizes that were determined in a deterministic calculation, based on the energy required to activate fire protection system sprinklers.

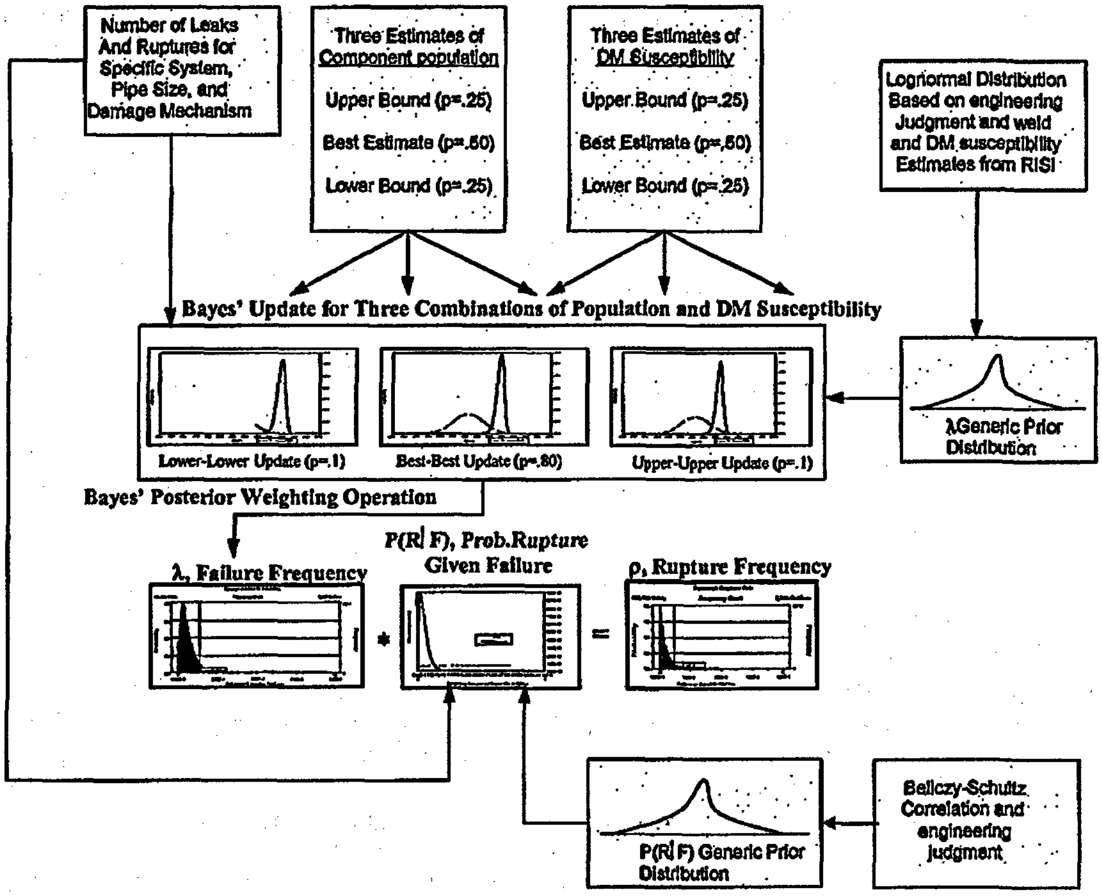

Figure 2-1 Flow Chart for Bayes' Estimates of System, Size, and Damage Mechanism Specific Pipe Failure Rates (λ) and Rupture Frequencies (ρ)

2.4 Definition of Pipe Failure Rate Cases

To support the baseline calculations and some sensitivity calculations that were selected to develop risk management insights, a set of 24 analysis cases were devised as shown in Table 2-1. The variables used to define these cases include the piping system, rupture size, and data screening assumptions.

A failure rate and a rupture frequency had to be developed for each case and, hence, a total of 48 parameter distributions were developed. As discussed more fully in Section 4, the dominant failure mechanism in HELB piping is flow accelerated corrosion (FAC). The piping systems were put into 4 major categories based on their general susceptibility to FAC. The systems in the HELB category that are susceptible to FAC include the feedwater and condensate systems and the steam systems with relatively wet steam conditions with carbon steel pipe. Based on insights from service experience and the piping design parameters, the high-pressure steam piping between the steam generators and the inlet of the high pressure turbine is generally not susceptible to FAC. The reasons for this include the use of thick walled pipe, dry steam conditions, and relatively straight bend free runs of pipe. In the PIPExp database there have been no instances of FAC in this part of the main steam system. Hence the high-pressure steam piping is set aside as one category so that the remaining categories represent the FAC sensitive pipe. The FAC sensitive pipe was further broken down into 3 categories based on the relative susceptibility to FAC; two categories for steam and one for feedwater and condensate. The two steam categories in include the low-pressure steam pipe downstream of the HP turbine outlet and the extraction steam.

For each of the four system categories described in the preceding paragraphs, rupture frequencies were developed for two rupture size cases: Ruptures with equivalent break sizes between 2-inches and 6-inches diameter, and ruptures with equivalent break sizes greater than 6-inches in diameter. The estimation of the rupture frequencies for each of these break size cases required the estimation of two parameters: a failure rate and a conditional probability that the break would be in the specified size range. The failure rate for each break size range is different because only pipes with a pipe diameter of at least 6-inches can produce a break size greater than 6-inches, whereas pipes as small as 2-inches in diameter can produce break sizes of 2-inches and greater. To support the estimation of these parameters, separate queries of the pipe failure database had to be made for pipe failures (cracks, leaks, wall-thinning, and ruptures) and ruptures in the prescribed break size ranges. Then, these queries had to be matched up against the appropriate estimate of the pipe component population exposure terms. The parameter estimation for these failure rates and conditional rupture probabilities is documented in Section 4.

Consideration was given to the development of system-specific failure rates and rupture frequencies separately for the feedwater system and for the condensate system as was performed in Reference [1] for the internal flooding application. It was decided to develop a composite set of failure rates and rupture frequencies for both systems combined for several reasons: One is that there are inconsistencies in the way in which system boundaries are established between feedwater and condensate that would give rise to inconsistencies between how the data was classified and how it is applied to Kewaunee.

Second, there are a variety of different operating conditions within the condensate system and the feedwater system that give rise to different susceptibilities to the predominant damage mechanism, flow accelerated corrosion. For example there are normally several stages of feedwater heating in the condensate and additional stages in the feedwater system. Feedwater drains and heater and main feedwater and condensate lines have much different conditions.

Third, there is no noticeable trend in the failure and rupture service experience between the two systems. And finally, breaking up the data into separate systems reduces the statistical quality of each data cell, i.e. would subdivide the data cells too finely so that the frequency of events within each data cell are statistically insignificant.

Based on what was learned in this study, the authors plan to issue a revision to Reference [1] to replace the system specific rates in that reference with a composite set of rates for the feedwater and condensate systems.

Based on the success criteria discussed in Section 3, for each set of failure rates, two rupture modes had to be distinguished: those with equivalent break sizes between 2" and 6" and those with break sizes in excess of 6 inches. Depending on the location of the pipe break either or both of these rupture modes may contribute to a specific HELB-initiated internal flooding initiating event, as discussed more fully in Section 3. Separate conditional rupture probability models had to be developed to distinguish these cases.

Table 2-1 Pipe Failure Rate Analysis Cases

Case	System	Pipe Size	Data Screening Assumptions
KNPP01	FWC	≥ 2 inch	Post-1988 data only
KNPP02	FWC	> 6 inch	Post-1988 data only
KNPP03	FWC	≥ 2 inch	Data up to 1988 only
KNPP04	FWC	> 6 inch	Data up to 1988 only
KNPP05	FWC	≥ 2 inch	FAC events removed
KNPP06	FWC	> 6 inch	FAC events removed
KNPP07	Extraction Steam	≥ 2 inch	Post-1988 data only
KNPP08	Extraction Steam	> 6 inch	Post-1988 data only
KNPP09	Extraction Steam	≥ 2 inch	Data up to 1988 only
KNPP10	Extraction Steam	> 6 inch	Data up to 1988 only
KNPP11	Extraction Steam	≥ 2 inch	FAC events removed
KNPP12	Extraction Steam	> 6 inch	FAC events removed
KNPP13	Low Pressure Steam	≥ 2 inch	Post-1988 data only
KNPP14	Low Pressure Steam	> 6 inch	Post-1988 data only
KNPP15	Low Pressure Steam	≥ 2 inch	Data up to 1988 only
KNPP16	Low Pressure Steam	> 6 inch	Data up to 1988 only
KNPP17	Low Pressure Steam	≥ 2 inch	FAC events removed
KNPP18	Low Pressure Steam	> 6 inch	FAC events removed
KNPP19	High Pressure Steam	≥ 2 inch	Post-1988 data only
KNPP20	High Pressure Steam	> 6 inch	Post-1988 data only
KNPP21	High Pressure Steam	≥ 2 inch	Data up to 1988 only
KNPP22	High Pressure Steam	> 6 inch	Data up to 1988 only
KNPP23	High Pressure Steam	≥ 2 inch	FAC events removed
KNPP24	High Pressure Steam	> 6 inch	FAC events removed

A review of the piping service data as discussed more fully in Section 4 reveals a significant improvement in piping system performance around 1988. It is reasonable to assume that this trend in performance is due to industry and NRC efforts to improve plant performance in general, and in particular to address flow accelerated corrosion in augmented inspection, repair and replacement programs. For the base case analysis only the service data since 1988 was used to calculate the failure rates as this data is viewed to be representative of current industry practice in managing piping system performance. As a contrast, the second case considered only the service data up to and including 1988. A third case was defined by screening out all the FAC related pipe failures. The purpose of the three cases was to understand the importance of the prevailing failure mechanism for experienced high energy line breaks.

Failure rates were specialized for the wet and dry steam systems, and for the feedwater and condensate systems, by specializing the data analysis for the failure rates. The data from the FAC sensitive steam, feedwater, and condensate systems were combined for the purposes of estimating the conditional rupture size probabilities. The justification for this is that essentially all the pipe ruptures in these systems are due to FAC and occur in similar carbon steel pipes. The system-specific factors that influence the rupture frequencies are judged to be adequately reflected in the specialized failure rates. The conditional probability of rupture size is viewed to be primarily related to properties of the pipe material and the damage mechanism and less related to the property of the system. The piping system materials for all the FAC sensitive piping are very similar. This is consistent with the data treatment in References [1], [3], and [8].

In summary, the piping failure rates and rupture frequencies developed in this study were quantified to address 4 different pipe system categories, 2 break size categories, and 3 data screening assumptions, giving rise to 24 data analysis cases. For each case, a pipe failure rate covering all failure modes, and a rupture frequency covering a specific break size range was developed and hence 48 parameters were developed.

3. KEWAUNEE HELB-INITIATED INTERNAL FLOODING INITIATING EVENTS

3.1 *Definition of Breaks*

Quantification of the HELB-initiated internal flood initiating event frequency values is performed for each initiating event defined in the turbine building internal flooding initiating events analysis. A summary of the HELB-related initiating events is provided below.

3.1.1 Steam Line Breaks

For steam line breaks, two HELB-initiated internal flooding initiating events are analyzed. The first is a steam line break that actuates enough fire sprinklers to result in full flow from both fire pumps to the Turbine Building. This event includes any break upstream of the turbine throttle valves below the operating deck with an equivalent diameter less than nine inches but greater than two inches, any break in the extraction steam line greater than six inches, and any break in a line after exiting the high-pressure turbine with an equivalent diameter of six inches or greater.

The second event is a steam line break that actuates approximately 100 sprinklers. The Turbine Building HELB models show that 100 sprinklers are representative of moderate releases. This event includes breaks in the extraction steam lines with an equivalent break size between two and six inches, and breaks in a line after exiting the high-pressure turbine and having an equivalent diameter of two to six inches.

3.1.2 Feedwater and Condensate Line Breaks

For feedwater and condensate line breaks, two HELB-initiated internal flooding initiating events are analyzed. The first is a feedwater or condensate line break that actuates enough fire sprinklers to result in full flow from both fire pumps to the Turbine Building. This event includes any between the fourth and fifth feedwater heaters with an equivalent diameter of greater than six inches or any break downstream of the fifth feedwater heaters with an equivalent diameter greater than two inches.

The second event is a feedwater or condensate line break that actuates approximately 100 sprinklers. The Turbine Building HELB models show that 100 sprinklers are representative of moderate releases. This event includes breaks in the lines between the fourth and fifth feedwater heaters with an equivalent diameter between two and six inches.

3.2 *Break Frequency Calculations*

3.2.1 Steam Line Breaks Causing Large Fire Protection System Actuations

This analysis will use the pipe length values determined in the turbine building internal flooding initiating events analysis. For steam piping located upstream of the turbine throttle valve, a total of 884.6 linear feet of piping were identified on the mezzanine and basement levels. For extraction steam, a total of 176.5 linear feet of piping was identified on the mezzanine and basement levels. For steam lines after the exit of the high-pressure turbine, a total of 621.7 linear feet of piping was identified on the mezzanine and basement levels. All other steam piping was located either on the operating deck or in the Auxiliary Building.

For piping located upstream of the turbine throttle valve, the frequency of pipe ruptures includes all failures with an equivalent diameter of greater than two inches. The frequency of failures in steam piping upstream of the turbine throttle valve, F_{HPS}, can be calculated as follows:

$$F_{HPS} = L_{HPS}(\rho_{KNPP19} + \rho_{KNPP20}) = L_{HPS}(\lambda_{KNPP19}P\{2-6|F\} + \lambda_{KNPP20}P\{>6|F\}) \qquad (3.1)$$

Where:

L_X = Length of pipe in system X

ρ_j = Pipe Rupture Frequency for Case j (see Table 2-1)

λ_j = Pipe Failure Rate for Case j (see Table 2-1)

$P\{2-6|F\}$ = Conditional probability of pipe rupture of size 2" to 6" given pipe failure in pipe ≥ 2 inch in size

$P\{>6|F\})$ = Conditional probability of pipe rupture of size > 6" given pipe failure in a pipe > 6 inch in size

The systems and cases are defined in Table 2-1.

The above equation uses the pipe modeling methodology of Reference [1] in which all the failure modes of the metallic system pressure boundary components are averaged into a pipe system failure rate per linear foot of pipe. Since all the pressure boundary failure modes were included in the data analysis, there is no need to add separate terms to the equations to account for such components as valves, heat exchangers, pump bodies, and metallic expansion joints. This approach is also justified by the fact that most of the experienced pipe failures occur in pipes or where pipes are welded to other pipes or piping components.

For extraction steam piping, the frequency of pipe ruptures includes all failures with an equivalent diameter of greater than six inches. The frequency of failures in the extraction steam piping can be calculated as follows:

$$F_{ES} = L_{ES}\rho_{KNPP08} = L_{ES}(\lambda_{KNPP08}P\{>6|F\} \qquad (3.2)$$

For steam piping after the exit of the high-pressure turbine, the frequency of pipe ruptures includes all failures with an equivalent diameter of greater than six inches. The frequency of failures in this piping can be calculated as follows:.

$$F_{RSL} = L_{RS}\rho_{KNPP14} = L_{RS}(\lambda_{KNPP14}P\{>6|F\} \qquad (3.3)$$

The total frequency for steam line breaks that actuate enough fire protection sprinklers to result in full system flow to the turbine building is the sum of the three values calculated above or:

$$F_{SLBL} = F_{HPS} + F_{ES} + F_{RS} \qquad (3.4)$$

3.2.2 Steam Line Breaks Causing Intermediate Fire Protection System Actuations

Calculation of the frequency of this event is performed as shown in Section 3.2.1 for large steam line breaks. Pipe length data also are identified in that section.

For extraction steam piping, the frequency of pipe ruptures includes failures with an equivalent diameter of between two and six inches. The frequency of failures in the extraction steam piping can be calculated as follows:

$$F_{ESM} = L_{ES}\rho_{KNPP07} = L_{ES}\lambda_{KNPP07}P\{2-6|F\} \qquad (3.5)$$

For steam piping after the exit of the high-pressure turbine, the frequency of pipe ruptures includes all failures with an equivalent diameter of between two and six inches. The frequency of failures in this piping can be calculated as follows:

$$F_{RSM} = L_{RS}\rho_{KNPP13} = L_{RS}\lambda_{KNPP13}P\{2-6|F\} \qquad (3.6)$$

The total frequency for steam line breaks that actuate approximately 100 fire protection sprinklers is the sum of the two values calculated above or:

$$F_{SLBM} = F_{ESM} + F_{RSM} \tag{3.7}$$

3.2.3 Feedwater and Condensate Line Breaks Causing Large Fire Protection System Actuations

This analysis will use the pipe length values determined in the turbine building internal flooding initiating events analysis. As discussed in Section 3.1.1, this event includes any break with an equivalent diameter greater than two inches in piping downstream of the 15 feedwater heaters and any break with an equivalent diameter greater than six inches between the 14 and 15 feedwater heaters. For feedwater piping located downstream of the 15 feedwater heaters, a total of 331.56 feet of pipe was identified. For piping between the 14 and 15 feedwater heaters, a total of 696.55 feet of pipe was identified.

The failure frequency for these size breaks in this piping is calculated to be:

$$F_{FL15} = L_{FL15}(\rho_{KNPP01} + \rho_{KNPS02}) = L_{FL15}(\lambda_{KNPP01}P\{2-6|F\} + \lambda_{KNPP02}P\{>6|F\}) \tag{3.8}$$

For piping between the 14 and 15 feedwater heaters, only pipe breaks greater than six-inches equivalent diameter are included. The failure frequency for these size breaks in this piping is calculated to be:

$$F_{FL45L} = L_{FL45}\rho_{KNPP02} = L_{FL45}(\lambda_{KNPP02}P\{>6|F\} \tag{3.9}$$

The frequency of feeedwater and condensate line breaks for this initiating event is the sum of the two values above or:

$$F_{FLBL} = F_{FLB15} + F_{FLB45L} \tag{3.10}$$

3.2.4 Feedwater and Condensate Line Breaks Causing Intermediate Fire Protection System Actuations

Calculation of the frequency of this event is performed as shown in Section 3.2.3. Pipe length data also are identified in that section. As discussed in Section 3.1.2, this event includes any break with an equivalent diameter between two and six inches between the 14 and 15 feedwater heaters. Using that data and the methodology of this report, the failure frequency for these size breaks in this piping is calculated to be:

$$F_{FL45M} = L_{FL45}\rho_{KNPP01} = L_{FL45}\lambda_{KNPP01}P\{2-6|F\} \tag{3.11}$$

3.3 *Model Quantification*

The technical approach to model quantification is to develop uncertainty distributions for each of the parameters defined in this section and then to propagate these distributions through the equations using Monte Carlo simulation, a traditional approach to PRA uncertainty quantification. The development of the pipe failure rate and rupture frequency parameters in these models is documented in Section 4 and the results of the Monte Carlo analysis are provided in Section 5. The pipe length estimates described in the above section were provided to the authors by Maracor and are documented in the main body of the turbine building internal flooding initiating events report of which this analysis will be an attachment. Uncertainty in pipe length estimates is modeled using normal distributions with the estimated pipe lengths taken as the mean values and a standard deviation of 10% of these length estimates.

4. PIPE FAILURE RATES AND RUPTURE FREQUENCIES

4.1 *System Boundaries*

This evaluation is concerned with non-ASME Code piping systems inside the Turbine Building of Pressurized Water Reactor (PWR) plants. The following systems are considered:

- Feedwater & Condensate (FWC) piping: The Condensate piping system extends from the Condenser Hotwell up to and including the Low Pressure Heaters. It also includes the Drains and Vents System piping from the Low Pressure and High Pressure Heaters. The Feedwater piping system boundary considered in this evaluation consists of the piping from the Low Pressure Heaters, the Feedwater pump suction/discharge piping, High Pressure Heater inlet/outlet piping up to the outboard containment isolation valves. Due to comparable susceptibilities to flow accelerated corrosion (FAC) and plant to plant variabilities in how the boundaries between these systems is defined, a composite set of data parameters are developed for FWC piping.
- Steam Extraction piping: In a typical PWR the high pressure portion of the turbine has extraction connections for two stages of feedwater heating. The low pressure portion of the turbine has extraction connections for four stages of feedwater heating.
- Low Pressure Steam piping: In this evaluation, the low pressure steam piping includes piping between the high pressure (HP) and low pressure (LP) turbine stages, including steam cross-over and cross-under piping, and Moisture Separator Reheater (MSR) piping. The MSR piping is also located between the HP and LP turbines and it is used to extract moisture from the steam and reheat the steam to improve the turbine performance.
- HP Steam piping:. In this evaluation the HP steam piping is upstream of the HP turbine throttle valve and extends to the outboard containment isolation valves.

4.2 *Database Screening*

The pipe failure rates and rupture frequencies in this evaluation are derived from service data included in the PIPExp database (Appendix A). The full PIPExp includes on the order of 6,700 data records covering Code Class 1-3 and non-Code piping in commercial light water reactor plants. Input parameters to the pipe failure rate calculation in this evaluation are obtained through database queries that include filters for excluding any non-relevant service data:

- Initial screening on the basis of Code Class and PWR plant system. Retain failure data associated with non-Code piping in Turbine Building including the following systems:
    - Condensate System
    - Extraction steam piping
    - Feedwater heater drain and vent piping

  - Main Feedwater (from LP feedwater heaters to outboard containment isolation valves)
  - Main Steam (from outboard containment isolation valve to High Pressure turbine steam admission valve, and turbine cross-over/cross-under piping)
  - Moisture Separator Reheater piping
- Results of initial screening subjected to additional screening on the basis of nominal pipe size and through-wall flaw size:
- The evaluation considers piping of nominal pipe size (NPS) greater than 2-inch diameter as piping less than 2-inch is not within the scope of the HELB-initiated internal flooding initiating event models described in Section 3.

The service data involving through-wall flaws are reviewed in accordance with the Kewaunee HELB-initiated internal flooding initiating event analysis requirements (i.e., "moderate" versus "major" release). This means that the service data are screened further on the basis of flaw size ('equivalent diameter break size'). The results of this screening step are input to the derivation of posterior Beta distribution parameters for calculation of conditional pipe failure probabilities for 2" to 6" and greater than 6" break sizes.

4.3 *Database Query Results*

The results of the database queries are summarized in charts (Figures 4-1 and 4-2) and tables (Tables 4-1 and 4-2). Flow-accelerated corrosion (FAC) is a predominant degradation mechanism for the systems that are included in the study scope except for the high pressure steam system. Most if not all plant owners have implemented programs to mitigate FAC susceptibilities. These programs include implementing non-destructive examination (NDE) programs, pro-active monitoring of pipe wall wear rates, and replacing the original carbon steel piping with FAC-resistant piping material such as stainless steel, carbon steel clad on the inside diameter with stainless steel, or chrome-molybdenum alloy steel. The purpose of these initial data queries was to identify the appropriate data set to use that represents current industry practice for predicting the initiating event frequencies at Kewaunee. The use of time trend analysis is a requirement of the ASME PRA standard for Capability Category 3 analyses. In addition, evaluating the trending of events avoids important insights in the data that would be missed by simply averaging all the industry experience.

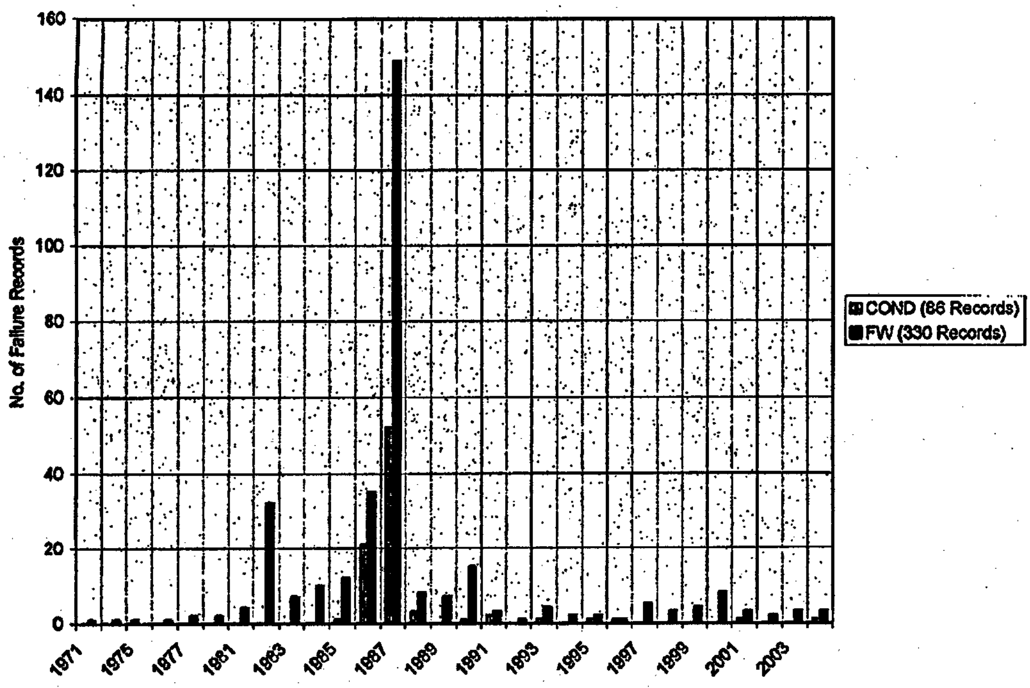

Figure 4-1 PWR Worldwide Experience with non-Code FWC Piping 1970-2004

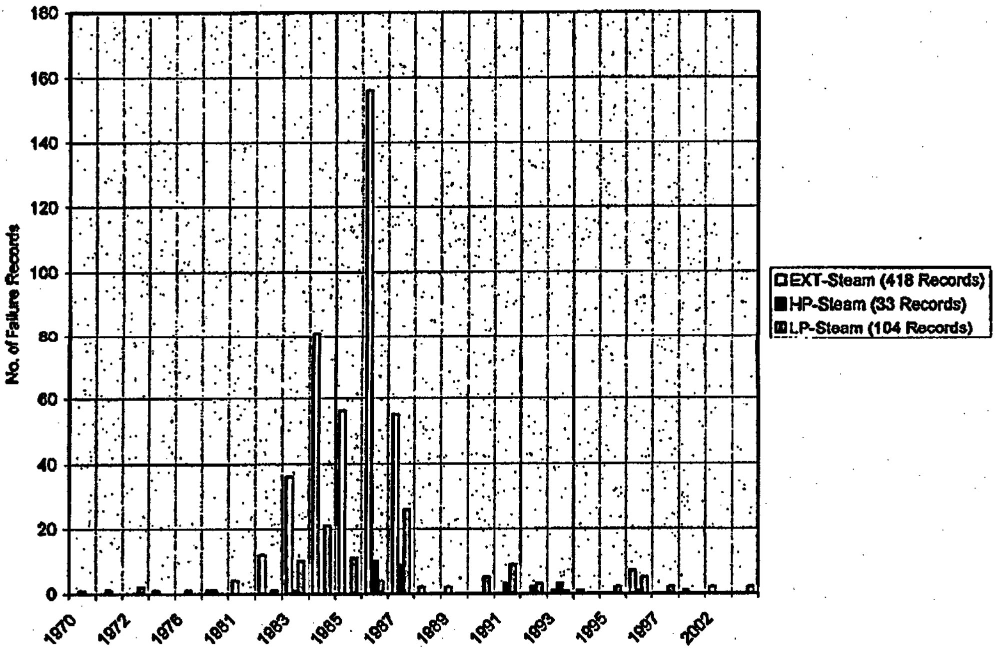

Figure 4-2 PWR Worldwide Experience with non-Code Steam Piping 1970-2004 [1]

The two charts above show a distinctly higher incident rate before 1988. The before/after-1988 trend in Figure 4-1 and 4-2 is accounted for in the quantitative evaluation of the service-data. The service data coverage in PIPExp corresponds to 858 PWR reactor years for the period 01/01/1970 – 12/31/1987 and 1666 PWR reactor years for the period 01/01/1988-12/31/2004. By the early- to mid-1980's the industry experienced several major failures of non-Code carbon steel piping (e.g., Trojan in March 1985 and Surry-2 in December 1986) (See References [11] through [14]). In response to these events as well as the industry-wide experience with pipe wall thinning and minor through-wall flaws attributed to FAC.

Tables 4-1 and 4-2 show the same data sets as those included in Figures 4-1 and 4-2 except that the data is organized by failure mode and pipe size to reflect the Kewaunee HELB-initiated internal flooding initiating event analysis requirements. The following failure mode definitions are used:

- Wall thinning; represents cases of severe wall thinning resulting in either weld overlay repair or preemptive replacement of affected piping section or fitting (e.g., elbow, tee).
- Leak; includes pinhole leak, leak or large leak resulting in isolation (where feasible) or manual reactor shutdown to effect repair or replacement.
- Rupture; significant through-wall flaw resulting in moderate or significant steam/water release and prompt manual shutdown or automatic turbine trip/reactor trip.

As will be discussed more fully below, in developing estimates of the conditional rupture size probabilities, a special query is made on the database to identify those ruptures that fit into two size categories: 2" to 6", and greater than 6" equivalent break sizes.

Table 4-1 Service Experience with non-ASME Code FWC Piping

Nominal Pipe Size (NPS) [Inch]	1970-1987				1988-2004			
	Total	Wall Thinning	Leak	Rupture	Total	Wall Thinning	Leak	Rupture
2" < NPS ≤ 6"	14	5	6	3	18	7	7	4
NPS > 6"	300	275	17	8	52	30	15	7
Total:	314	280	23	11	70	37	22	11

Notes:

- Service experience in Table 1 derived from 2524 reactor-years of PWR operation worldwide; 858 reactor-years pre-1988 and 1666 reactor-years post-1987
- Failure data includes contributions from FAC (dominant degradation mechanism), vibration-fatigue and water hammer
- The root cause of post-1987 events in many cases is attributed to programmatic errors or weaknesses in the Owner's FAC program
- Appendix A includes information on the coverage and completeness of the PIPExp database

Table 4-2 Service Experience with non-Code Steam Piping

System	Nominal Pipe Size (NPS) [Inch]	1970-1987				1988-2004			
		Total	Wall Thinning	Leak	Rupture	Total	Wall Thinning	Leak	Rupture
EXT-Steam	2" < NPS ≤ 6"	10	0	8	2	9	1	7	1
	NPS > 6"	392	385	4	3	7	2	2	3
LP-Steam	2" < NPS ≤ 6"	14	0	11	3	15	1	10	4
	NPS > 6"	61	60	1	0	14	2	9	3
HP-Steam	NPS > 2"	24	19	3	2	9	1	7	1
	Total:	501	464	27	10	54	7	35	12

Notes:

- 'EXT-Steam' includes HP & LP steam extraction piping. Most of this piping is > NPS6.
- 'LP-Steam' includes piping between the HP and LP turbine stages, including cross-over/under piping and Moisture Separator Reheater piping.
- 'HP-Steam' includes piping upstream of the HP turbine throttle valve.
- Service experience in Table 1 derived from 2524 reactor-years of PWR operation worldwide; 858 reactor-years pre-1988 and 1666 reactor-years post-1987
- Failure data includes contributions from FAC (dominant degradation mechanism), vibration-fatigue and water hammer
- The root cause of post-1987 events in many cases is attributed to programmatic errors or weaknesses in the Owner's FAC program
- Appendix A includes information on the coverage and completeness of the PIPExp database

4.4 *Exposure Term Data*

In pipe failure rate estimation, the exposure term is the product of either the number of components (e.g., fittings, welds) or total length of piping that provides the observed pipe failures and the total time over which failure events are collected. There is variability in the population counts. In part this variability stems from differences across NSSS types and balance of plant design differences, and in part it stems from different piping design and fabrication practices (e.g., use of cold bent piping versus use of welded fittings). Also, design modifications are implemented during the lifetime of a plant to enhance flow conditions, minimize system vibrations, and to improve the access for non-destructive examination (NDE), etc. Table 4-3 summarizes piping population data for the systems covered in the Kewaunee HELB-initiated internal flooding initiating events analysis.

Table 4-3 Piping Population Exposure Data

System / System Group	Linear ft of Piping	Information Source / Comment
FWC (> NPS2)	14,037 ft	EPRI TR-111880, Table A-5; in the failure rate calculation the given length is input as a median value
EXT-Steam	1,500 ft	Entergy Nuclear Northeast (Indian Point-3 FAC program information). In the failure rate calculation the given length is input as a median value.
LP-Steam	622 ft	Dominion Energy; the given length is for KNPP and in the failure rate calculation it is input as a lower bound value
HP-Steam	885 ft	Dominion Energy; the given length is for KNPP and in the failure rate calculation it is input as a lower bound value

4.5 *Conditional Pipe Failure Probability*

For FAC-susceptible piping the likelihood of rapid or unexpected flaw propagation given wall thinning is quite high and can be estimated directly from service data. In the case of pipe materials or systems that are not susceptible to FAC such as the high pressure main steam system at Kewaunee, there are much fewer events from which to derive the conditional rupture probability. In this case the estimation of the likelihood of sudden pipe failure relies on insights from service experience with different piping systems and materials under different loading conditions in combination with engineering judgment and fracture mechanics evaluations.

The likelihood of a through-wall flaw propagating to a significant structural failure is expressed by the conditional failure probability $P_{ik}\{R/F\}$. It is determined from service experience insights and engineering judgment, with the uncertainty treated using the Beta Distribution.

The beta distribution takes on values between 0 and 1 and is defined by two parameters, A and B (some texts refer to these as "Alpha" and "Beta"). It is often used to express the uncertainty in the estimation of dimensionless probabilities such as MGL common cause parameters and failure rates per demand. The mean of the Beta Distribution is given by:

$$Mean = \frac{A}{A+B} \tag{4.1}$$

If A = B = 1, the beta distribution takes on a flat distribution between 0 and 1. If A = B = ½, the distribution is referred to as a Jeffery's non-informative prior and is a U shaped distribution with peaks at 0 and 1. Expert opinion can be incorporated by selecting *A* and *B* to match up with an expert estimate of the mean probability. For example, to represent an expert estimate of 10^{-2}, A=1 and B=99 can be selected. These abstract parameters *A* and *B* can be associated with the number of failures and the number of successes in examining service data to estimate a failure probability on demand. *A* + *B* represents the number of trials.

The beta distribution has some convenient and useful properties for use in Bayes' updating. A prior distribution can be assigned by selecting the initial parameters for *A* and *B*, denoted as A_{Prior} and B_{Prior}. Then when looking at the service data, if there are *N* failures and *M* successes observed, the Bayes updated or posterior distribution is also a Beta distribution with the following parameters:

$$A = A_{Prior} + N \tag{4.2}$$

$$B = B_{Prior} + M \tag{4.3}$$

The above explains how the Beta distribution is used in this study to estimate conditional rupture probabilities. The priors are selected to represent engineering estimates of the probabilities "prior" to the collection of evidence. Equations (4.2) and (4.3) are used to compute the parameters of the Bayes' updated distribution after applying the results of the data queries to determine N and M. N corresponds to the number of ruptures in the specified size range and M corresponds to the number of pipe failures that do not result in a rupture in the specified size range.

A review of service data provides some insights about the conditional pipe failure probability for different types of piping systems. Figure 4-3 shows the conditional failure probability for different, observed through-wall flow rate threshold values. For comparison the Beliczey-Schulz correlation [15] is re-calibrated for through-wall flow threshold values rather than pipe size; this correlation only applies to Code Class 1 piping. According to Beliczey-Schulz, for 1-inch piping the conditional probability of a major structural failure (MSF) or rupture is on the order of 5.0×10^{-2} (corresponding to a liquid flow rate of about 800 gpm (completely severed pipe), which is well beyond the upper threshold value in Figure 4-3. This information is presented to help justify the prior distribution parameters A and B selected for this analysis.

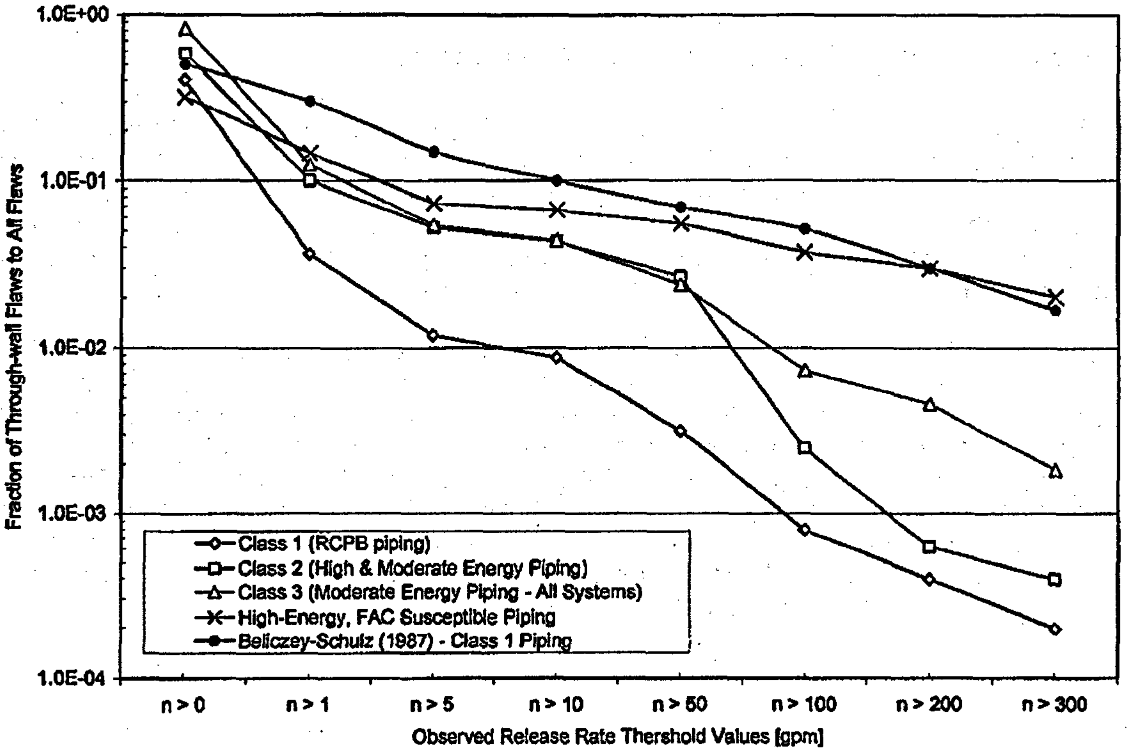

Figure 4-3
Empirical Conditional Probability of Pipe Failure as a Function of Type of Piping System & Through-Wall Flow Rate Threshold Value[1]

The "A" parameter of the Beta Distribution corresponds to a significant consequence (spray, internal flooding or major flooding event) and the "B" parameter corresponds to the remaining failure experience (significant wall thinning or through-wall flaw). The total number of failures in the database is equal to A+B. Table 4-4 is a summary of the prior and posterior Beta Distribution parameters for non-Code FWC and steam piping used in this report. The posterior distribution parameters are derived by performing a Bayes' update of the assumed prior distributions using service data from PIPExp and the conjugate properties of the Beta Distribution.

Part of the information presented in Table 4-4 is the screening of pipe ruptures in different break size ranges in the FAC sensitive piping. The 26 events with equivalent break sizes between 2" and 6" are listed in Table 4-5, and the 33 events with break sizes greater than 6-inches are in Table 4-6.

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[1] Plotted in the figure are the conditional probabilities of leak flow rates given pipe failure as estimated by the fraction of the pipe failures in the failure data population with the indicated leak flow rate.

4.6 *Results for Failure Rates and Rupture Frequencies*

Using the methodology described in Section 2, uncertainty distributions were developed for the failure rates and rupture frequencies for each of the analysis cases in Table 2-1. The mean values of these distributions are presented in Table 4-7. The full uncertainty distributions were propagated through the HELB-initiated internal flooding initiating event models that were described in Section 3 and the results are presented in Section 5. Parameters of these distributions are presented in Appendix B.

To support sensitivity calculations that are summarized in Section 5, comparisons were made among the data screening sensitivity cases for each system group that were identified. As seen in Figures 4-4 and 4-5 the results for the case using only data from prior to 1988 before FAC programs became effective would increase by more than an order of magnitude. Stated another way, the failure rates and rupture frequencies based on the service data before 1988 are more than an order of magnitude greater than those considering only data from events after 1988 when the FAC programs were in effect.. Conversely, if all the FAC-related events were precluded by some type of plant change, an order of magnitude reduction in the relevant pipe failure rates and rupture frequencies would be expected.

Table 4-4
Parameters of Posterior Beta Distribution for $P_{Ik}\{R|F\}$ for non-Code FAC-Susceptible High-Energy Piping & non-Code FAC-resistant High-Energy Piping

Analysis Case		Prior Beta Parameters			Posterior Beta Parameters		
Piping Material	**Equivalent Break Size (EBS)**	**Constraint**	A_{Prior}	B_{Prior}	A_{Post}	B_{Post}	**Mean**
Carbon Steel and FAC-susceptible	2" < EBS ≤ 6"	1.0E-2	1	99	27 (1)	1254	2.11E-02
	EBS > 6"	1.0E-2	1	99	34 (2)	1072	3.07E-02
Stainless Steel or FAC-resistant	2" < EBS ≤ 6"	1.0E-3	1	999	10	1062	9.33E-03
	EBS > 6"	1.0E-3	1	999	8	1036	7.66E-03

Notes:

(1) A through-wall flaw of size 2" < EBS ≤ 6" can occur in any FAC-susceptible piping of nominal pipe size (NPS) > 2". The database screening criteria include consideration of NPS and through-wall flaw size.

(2) A through-wall flaw of size EBS > 6" can occur in any FAC-susceptible piping of NPS > 6".

- EBS = Equivalent Break Size
- NPS = Nominal Pipe Size [inch]
- The posterior Beta distribution parameters are obtained from PIPExp database (accounts for service experience applicable to non-Code FWC and steam piping in Light Water Reactors):
  - $B_{Post} = B_{Prior} + (B_{Evidence} - A_{Evidence})$
  - $A_{Evidence}$ = Total number of ruptures in specified size range
  - $B_{Evidence}$ =Total number of failure records =1181 records (carbon steel FWC piping of nominal pipe size greater than 2". There are 1006 records for piping > 6" NPS.
  - $A_{Post\text{-}Large\ Leak} = A_{Prior} + A_{Evidence}$; the evidence is 26 records for which the through-wall defect is sufficient to create a significant outflow of steam/condensate corresponding to 2" < EBS ≤ 6" (Table 4-5).
  - $A_{Post\text{-}MSF} = A_{Prior} + A_{Evidence}$; the evidence is 33 records involving major structural failure of FAC-susceptible piping corresponding to EBS > 6-inch diameter (Table 4-6)
- The Beta distribution parameters for 'stainless steel or FAC resistant case' are obtained by screening out any data record involving degradation or failure caused by FAC. A total of 72 records involve non-FAC failures and of these, 44 records involve piping > NPS6.

Table 4-5 Summary of FAC-Susceptible Piping Rupture Events with Equivalent Break Size Between 2-inch Diameter and 6-inch Diameter (EBS1)

DATABASE RECORD NO.	EVENT DATE	PLANT NAME	COUNTRY	PLANT TYPE	SYSTEM	SYSTEM GROUP	NOMINAL PIPE SIZE [Inch]
2962	4/22/1995	Almaraz-1	ES	PWR	COND	FWC	6
15272	2/13/2001	Balakovo-2	RU	PWR	FW	FWC	3.2
2907	7/27/1993	Bohunice-3	SK	PWR	MS	STEAM	6
455	9/28/1983	Browns Ferry-1	US	BWR	MSR	STEAM	6
456	11/1/1977	Browns Ferry-3	US	BWR	EXT-Steam	STEAM	6
3722	8/10/1999	Callaway	US	PWR	FW	FWC	6
1166	9/25/1985	Dresden-2	US	BWR	COND	FWC	6
2787	11/17/1986	Fermi-2	US	BWR	FW	FWC	6
1425	4/28/1970	H.B. Robinson-2	US	PWR	MS	STEAM	6
1975	3/1/1977	Hatch-1	US	BWR	COND	FWC	4
1463	9/26/1989	Indian Point-2	US	PWR	MS	STEAM	4
2866	4/3/1987	Indian Point-2	US	PWR	FW	FWC	6
2498	11/24/1993	Kola-4	RU	PWR	MS	STEAM	4
999	1/1/1972	Millstone-1	US	BWR	MS	STEAM	4
494	12/30/1973	Millstone-1	US	BWR	COND	FWC	4
2161	12/31/1990	Millstone-3	US	PWR	MSR	STEAM	6
498	12/31/1990	Millstone-3	US	PWR	MSR	STEAM	6
501	3/19/1983	Oconee-2	US	PWR	MSR	STEAM	3
2949	12/15/1996	Paks-3	HU	PWR	EXT-STEAM	STEAM	6
478	7/29/1986	R.E. Ginna	US	PWR	MS	STEAM	6
850	11/18/1977	Ringhals-2	SE	PWR	FW	FWC	6
607	3/23/1990	Surry-1	US	PWR	MSR	STEAM	4
540	8/7/1972	Surry-1	US	PWR	MSR	STEAM	4
1536	1/9/1982	Trojan	US	PWR	EXT-STEAM	STEAM	6
697	8/1/1983	Zion-1	US	PWR	EXT-STEAM	STEAM	6
2458	7/28/1991	Zion-2	US	PWR	FW	FWC	3

Table 4-6 Summary of FAC-Susceptible Piping Rupture Events with Equivalent Break Size > 6-inch Diameter (EBS2)

DATABASE RECORD NO.	EVENT DATE	PLANT NAME	COUNTRY	PLANT TYPE	SYSTEM	SYSTEM GROUP	NOMINAL PIPE SIZE [Inch]
2865	12/18/1991	Almaraz-1	ES	PWR	MS	STEAM	8
445	4/18/1989	ANO-2 (Arkansas-2)	US	PWR	MS	STEAM	14
454	9/29/1982	Browns Ferry-1	US	BWR	MS	STEAM	8
453	6/24/1982	Browns Ferry-1	US	BWR	MSR	STEAM	8
15185	8/15/1983	Browns Ferry-1	US	BWR	MS	STEAM	8
462	11/20/1984	Calvert Cliffs-1	US	PWR	EXT-STEAM	STEAM	16
465	1/15/1988	Catawba-1	US	PWR	COND	FWC	8
2912	9/25/1987	Doel-1	BE	PWR	COND	FWC	8
2504	4/10/1993	Fermi-2	US	BWR	EXT-STEAM	STEAM	8
2785	4/21/1997	Fort Calhoun-1	US	PWR	FW	FWC	12
483	4/25/1986	Hatch-2	US	BWR	FW	FWC	20
37	6/27/1985	KMK Mülheim-Kärlich	DE	PWR	FW	FWC	18
2598	12/29/1984	Krsko	SLO	PWR	FW	FWC	14
2446	5/6/1991	Kuosheng-2	TW	BWR	COND	FWC	12
85	5/28/1990	Loviisa-1	FI	PWR	FW	FWC	12
76	2/25/1993	Loviisa-2	FI	PWR	FW	FWC	8
2928	6/14/1996	Maanshan-21	TW	PWR	MS	STEAM	16
20056	8/9/2004	Mihama-3	JP	PWR	FW	FWC	20
1307	11/6/1991	Millstone-2	US	PWR	MSR	STEAM	8
1320	8/8/1995	Millstone-2	US	PWR	Heater-Drain	FWC	8
500	6/23/1982	Oconee-2	US	PWR	EXT-STEAM	STEAM	24
865	1/1/1985	Oconee-2	US	PWR	FW	FWC	10
2701	9/24/1996	Oconee-2	US	PWR	MSR	STEAM	18
504	9/17/1986	Oconee-3	US	PWR	Heater-Drain	FWC	10
976	6/10/1974	Quad Cities-2	US	BWR	FW	FWC	18
2913	1/1/1989	Santa Maria de Garona	ES	BWR	FW	FWC	16
3092	2/9/1980	Santa Maria de Garona	ES	BWR	EXT-STEAM	STEAM	16

DATABASE RECORD NO.	EVENT DATE	PLANT NAME	COUNTRY	PLANT TYPE	SYSTEM	SYSTEM GROUP	NOMINAL PIPE SIZE [Inch]
2278	3/1/1993	Sequoyah-2	US	PWR	MS	STEAM	10
541	10/15/1983	Surry-1	US	PWR	FW	FWC	26
542	12/9/1989	Surry-1	US	PWR	Heater-Drain	FWC	10
595	12/9/1986	Surry-2	US	PWR	FW	FWC	18
545	3/9/1985	Trojan	US	PWR	FW	FWC	14
920	12/2/1971	Turkey Point-3	US	PWR	MS	STEAM	12

Table 4-7 Mean Values of Failure Rate and Rupture Frequency Parameters

Case	Description	Results – Mean Values	
		Failure Rate [1/ft.yr]	Rupture Frequency [1/ft.yr]
KNPP01	FWC, EBS1 with Post-1988 data	3.19E-06	6.72E-08
KNPP02	FWC, EBS2 with Post-1988 data	3.56E-06	1.09E-07
KNPP03	FWC, EBS1 with data through 1988	2.78E-05	5.85E-07
KNPP04	FWC; EBS2 with data through 1988	3.98E-05	1.22E-06
KNPP05	FWC; EBS1 with FAC events screened out	9.21E-07	8.60E-09
KNPP06	FWC; EBS2 with FAC events screened out	8.29E-07	6.35E-09
KNPP07	Steam Extraction piping; EBS1 with Post-1988 data	3.40E-06	7.17E-08
KNPP08	Steam Extraction piping; EBS2 with Post-1988 data	2.58E-06	7.93E-08
KNPP09	Steam Extraction piping; EBS1 with data through 1988	3.32E-04	6.99E-06
KNPP10	Steam Extraction piping; EBS2 with data through 1988	4.86E-04	1.49E-05
KNPP11	Steam Extraction piping; EBS1 with FAC events screened out	1.93E-07	1.80E-09
KNPP12	Steam Extraction piping; EBS2 with FAC events screened out	2.68E-07	2.07E-09
KNPP13	Steam piping downstream HP turbine, EBS1 Post-1988 data	1.33E-05	2.80E-07
KNPP14	Steam piping downstream HP turbine, EBS2 Post-1988 data	1.07E-05	3.29E-07
KNPP15	Steam piping downstream HP turbine, EBS1 with data through 1988	7.15E-05	1.51E-06
KNPP16	Steam piping downstream HP turbine, EBS2 with data through 1988	9.09E-05	2.79E-06
KNPP17	Steam piping downstream HP turbine, EBS1 with FAC events screened out	2.25E-07	2.10E-09
KNPP18	Steam piping downstream HP turbine; EBS2 with FAC events screened out	9.22E-07	7.05E-09
KNPP19	MS piping upstream HP turbine throttle valve, EBS1 Post-1988 data	3.25E-06	3.03E-08
KNPP20	MS piping upstream HP turbine throttle valve, EBS2 Post-1988 data	1.16E-06	8.90E-09
KNPP21	MS piping upstream HP turbine throttle valve, EBS1 with data through 1988	1.60E-05	1.49E-07
KNPP22	MS piping upstream HP turbine throttle valve, EBS2 with data through 1988	2.50E-05	1.91E-07
KNPP23	MS piping upstream HP turbine throttle valve, EBS1 with FAC events screened out	1.74E-07	1.64E-09
KNPP24	MS piping upstream HP turbine throttle valve, EBS2 with FAC events screened out	2.36E-07	1.80E-09

<u>Notes:</u>

- EBS = Equivalent (Diameter) Break Size
- EBS1: 2" < EBS ≤ 6" equivalent diameter break size – moderate energy release
- EBS2: EBS > 6" equivalent diameter break size – major energy release

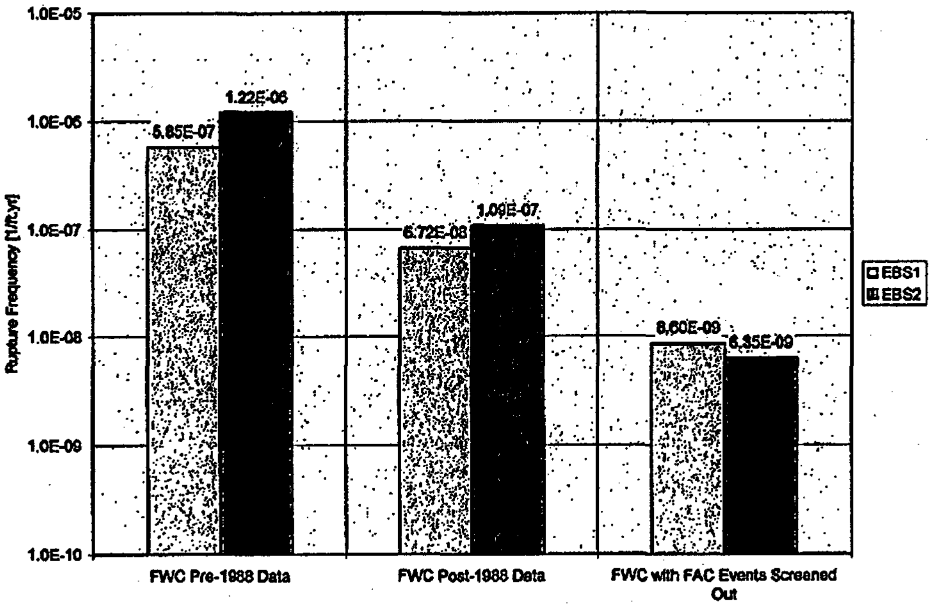


Figure 4-4
Impact of Different Data Screening Assumptions on FWC Piping Reliability

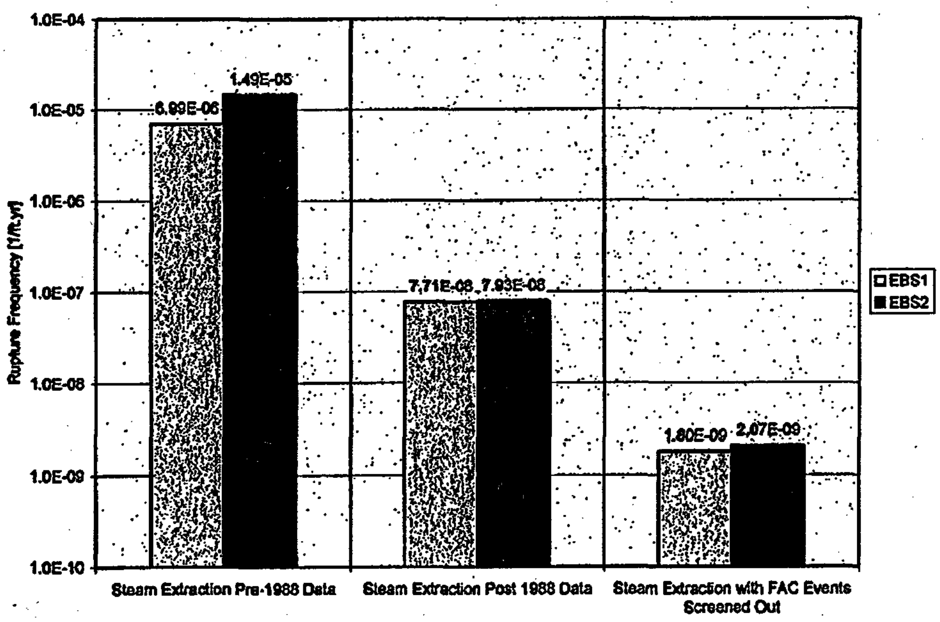


Figure 4-5
Impact of Different Data Screening Assumptions on Steam Extraction Piping Reliability

5.HELB INITIATING EVENT FREQUENCIES

5.1 *Calculation Steps*

The results for the initiating event frequencies were obtained using the equations in Section 3 and the data parameters developed in Section 4. The uncertainties were calculated using the technical approach described in Section 2 and is comprised of the following steps.

1. A prior distribution for each failure rate was obtained from Reference [1]. The prior is a lognormal distribution with a mean value of 1.50×10^{-4} failures per foot of pipe with a range factor of 100. The same prior was used for all 24 cases in Table 2-1.
2. For each case listed in Table 2-1, Bayes' updates were performed using the prior from Step 1, the number of failures obtained from the PIPExp database for each case, and estimates of the piping population exposures that are documented in Section 4. Bayes' updates were performed using the program BART™ developed by ERIN Engineering and Research, Inc.
3. To account for uncertainty in the population exposure estimates the Bayes' updates were performed for three estimates of the exposure: a best estimate with a probability weight of 80% and a high and low estimate with weights of 10% each.
4. A composite uncertainty distribution was developed for each of the 24 cases of failure rates using a posterior weighting procedure using Crystal Ball™ and Microsoft Excel.
5. The process listed in Steps 1-4 was repeated for two ranges of pipe size: one for pipes greater than or equal to 2", which could produce ruptures of size 2" and greater, and one for pipes sizes greater than 6" which could produce rupture sizes exceeding 6". Hence a total of 48 failure rate distributions were developed: one for 2" and greater, and one for 6" and greater pipe size ranges for each of the 24 cases in Table 2-1.
6. A Beta distribution was developed to represent the conditional probability of rupture for two rupture sizes: 2" to 6", and greater than 6" equivalent break size using the data described in Section 4. These beta distributions include prior distribution parameters that represent the authors expert judgment on the values of these probabilities, and service data experience that is documented in Section 4. Two sets of distributions were developed: one for FAC sensitive carbon steel pipe in systems subject to FAC, and the other for FAC resistant pipe or systems that are not susceptible to FAC, e.g., the high-pressure main steam piping upstream of the turbine throttle valves.
7. The rupture frequencies for rupture sizes between 2" and 6" were obtained by combining the failure rates for 2" and greater pipes and the conditional rupture probabilities developed in Step 6. The rupture frequencies for greater than 6" breaks were obtained by combining the failure rates for greater than 6" pipe sizes with the appropriate conditional rupture probability.
8. The HELB-initiated internal flooding initiating event frequencies were obtained by propagating the uncertainties in the appropriate rupture frequencies through the equations of Section 3 using the Monte Carlo process using Cystall Ball™ and Microsoft Excel. To properly treat the state of knowledge dependencies all the uncertainty calculations from the output of the Bayes' updates through Step 8 were performed in a single integrated Monte Carlo procedure. In each Monte Carlo trial a failure rate was sampled for each case and

pipe size by sampling from either a high, best estimate or low exposure term estimate. A conditional rupture probability for each rupture mode was sampled for each pipe size, and a sample initiating event frequency was calculated by propagating these samples through the equations for the pipe rupture frequencies and the equations for the HELB-initiated internal flooding initiating event frequencies. This process also made it unnecessary to perform a series of Monte Carlo calculations in which the results from each step would be fitted to a distribution for sampling in the next stage.

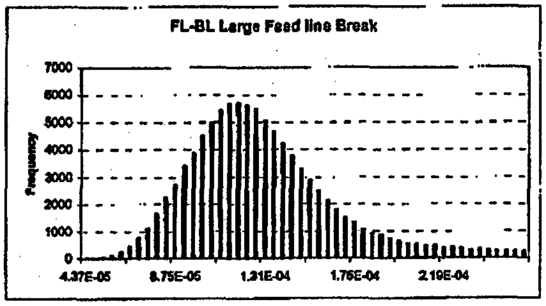
5.2 *Summary of Results*

The results for the initiating event frequencies are summarized in Table 5-1 for each of the equations listed in Section 3. The results listed in bold font are the initiating event frequencies; the remaining values are key intermediate results.

In Figures 5-1 through 5-4 the details of the uncertainty analysis are provided for Large Feedline Breaks, Moderate Feedline Breaks, Large Steamline Breaks, and Moderate Steamline breaks, respectively using as input reports that are generated by Crystal Ball™.

Table 5-1 Uncertainty Distribution Results for HELB-Initiated Internal Flooding Initiating Event Frequencies

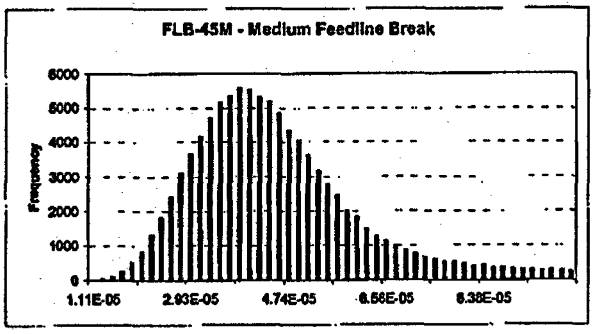
Event	Events per Reactor Operating Year			
	Mean	5%tile	50%tile	95%tile
F_{HP}, Large High Pressure SLB	3.47E-05	1.50E-05	3.11E-05	6.68E-05
F_{RSL}, Large Reheat SLB	2.04E-04	9.82E-05	1.84E-04	3.85E-04
F_{ESL}, Large Extraction SLB	1.40E-05	4.96E-06	1.19E-05	3.00E-05
F_{SLBL}, Large SLB	**2.53E-04**	**1.42E-04**	**2.33E-04**	**4.37E-04**
F_{RSML}, Moderate Reheat SLB	1.74E-04	8.63E-05	1.57E-04	3.25E-04
F_{ESM}, Moderate Extraction SLB	1.28E-05	5.32E-06	1.11E-05	2.58E-05
F_{SLBL}, Moderate Steam Line SLB	**1.87E-05**	**9.84E-05**	**1.71E-04**	**3.37E-04**
F_{FL15}, Large FLB downstream of FWH15	5.85E-05	3.67E-05	5.52E-05	9.40E-05
F_{FL45}, Large FLB between FWH14 and FWH15	7.67E-05	4.15E-05	7.01E-05	1.42E-04
F_{FLBL}, Large FLB	**1.35E-4**	**8.19E-05**	**1.26E-04**	**2.27E-04**
F_{FL45M}, Moderate FLB	**4.69E-05**	**2.47E-05**	**4.29E-05**	**8.63E-05**



Percentiles:	Forecast values
0%	4.15E-05
5%	8.19E-05
10%	9.04E-05
15%	9.65E-05
20%	1.02E-04
25%	1.06E-04
30%	1.10E-04
35%	1.14E-04
40%	1.18E-04
45%	1.22E-04
50%	1.26E-04
55%	1.30E-04
60%	1.34E-04
65%	1.39E-04
70%	1.44E-04
75%	1.50E-04
80%	1.58E-04
85%	1.69E-04
90%	1.87E-04
95%	2.27E-04
100%	4.90E-04

Statistics:	Forecast values
Trials	100,000
Mean	1.35E-04
Median	1.26E-04
Mode	—
Standard Deviation	4.50E-05
Variance	2.02E-09
Skewness	1.76
Kurtosis	10.76
Coeff. of Variability	0.33
Minimum	4.15E-05
Maximum	4.90E-04
Range Width	4.49E-04
Mean Std. Error	1.42E-07

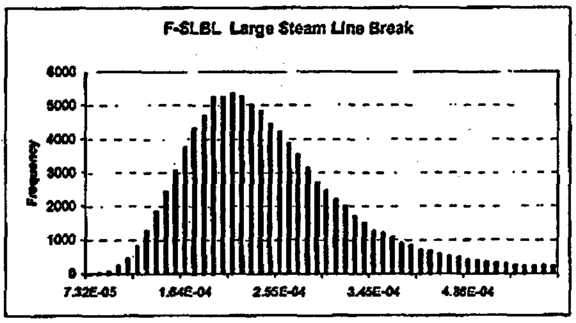
Figure 5-1 Crystal Ball Results for Large Feedline Break Frequency



Percentiles:	Forecast values
0%	1.02E-05
5%	2.47E-05
10%	2.81E-05
15%	3.05E-05
20%	3.27E-05
25%	3.45E-05
30%	3.62E-05
35%	3.79E-05
40%	3.95E-05
45%	4.11E-05
50%	4.29E-05
55%	4.46E-05
60%	4.65E-05
65%	4.86E-05
70%	5.09E-05
75%	5.36E-05
80%	5.70E-05
85%	6.17E-05
90%	6.96E-05
95%	8.63E-05
100%	2.16E-04

Statistics:	Forecast values
Trials	100,000
Mean	4.69E-05
Median	4.29E-05
Mode	—
Standard Deviation	1.94E-05
Variance	3.75E-10
Skewness	1.83
Kurtosis	11.10
Coeff. of Variability	0.41
Minimum	1.02E-05
Maximum	2.16E-04
Range Width	2.06E-04
Mean Std. Error	6.12E-08

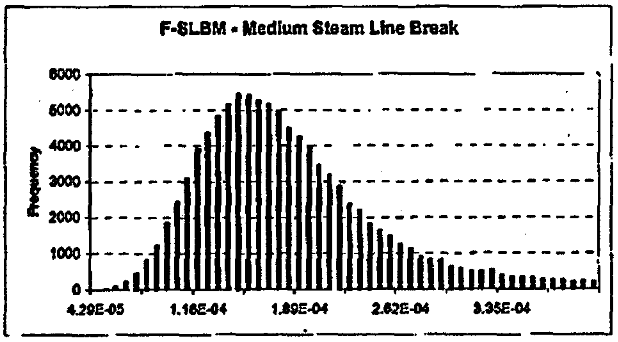
Figure 5-2 Crystal Ball Results for Intermediate Feedline Break Frequency



Percentiles:	Forecast values
0%	6.87E-05
5%	1.42E-04
10%	1.58E-04
15%	1.70E-04
20%	1.80E-04
25%	1.90E-04
30%	1.98E-04
35%	2.07E-04
40%	2.15E-04
45%	2.24E-04
50%	2.33E-04
55%	2.42E-04
60%	2.53E-04
65%	2.64E-04
70%	2.76E-04
75%	2.91E-04
80%	3.09E-04
85%	3.33E-04
90%	3.69E-04
95%	4.37E-04
100%	1.16E-03

Statistics:	Forecast values
Trials	100,000
Mean	2.63E-04
Median	2.33E-04
Mode	—
Standard Deviation	9.61E-05
Variance	9.24E-09
Skewness	1.82
Kurtosis	11.77
Coeff. of Variability	0.38
Minimum	6.87E-05
Maximum	1.16E-03
Range Width	1.10E-03
Mean Std. Error	3.04E-07

Figure 5-3 Crystal Ball Results for Large Steam Line Break



Percentiles:		Forecast values
	0%	3.93E-05
	5%	9.84E-05
	10%	1.11E-04
	15%	1.21E-04
	20%	1.29E-04
	25%	1.36E-04
	30%	1.43E-04
	35%	1.50E-04
	40%	1.57E-04
	45%	1.64E-04
	50%	1.71E-04
	55%	1.78E-04
	60%	1.86E-04
	65%	1.95E-04
	70%	2.04E-04
	75%	2.16E-04
	80%	2.30E-04
	85%	2.49E-04
	90%	2.78E-04
	95%	3.37E-04
	100%	1.02E-03

Statistics:	Forecast values
Trials	100,000
Mean	1.87E-04
Median	1.71E-04
Mode	—
Standard Deviation	7.75E-05
Variance	6.01E-09
Skewness	1.89
Kurtosis	12.03
Coeff. of Variability	0.41
Minimum	3.93E-05
Maximum	1.02E-03
Range Width	9.82E-04
Mean Std. Error	2.45E-07

Figure 5-4 Crystal Ball Results for Intermediate Steam Line Break

5.3 *Sensitivity Study*

As a sensitivity study, the initiating event frequencies were recalculated using different assumptions regarding how the data was screened as discussed in Section 4. This study was performed by propagating the results for the pipe failure rates and rupture frequencies for the different data screening strategies through the equations for the initiating event frequencies in Section 3. The results are summarized in Table 5-2 and Figure 5-5. As seen in these exhibits, the impact of using the service data from 1988 to represent the current industry practice and as a basis to predict the HELB-initiated internal flooding frequencies is approximately an order of magnitude compared with the case of using pre-1988 data. This shows the impact of industry improvement programs, particularly the FAC programs, which were responsible for reducing the frequency of pipe breaks since about 1988. Although these programs were effective in reducing the pipe break frequencies, as seen in the third case in which all the FAC related failures since 1988 were removed, FAC is still a dominant failure mechanism for these systems. The initiating event frequencies would be an order of magnitude lower if all the FAC related failures were removed from the data analysis.

Table 5-2 Impact of Alternative Assumptions Regarding Data Screening on HELB-Initiated Internal Flooding Initiating Event Frequencies

Initiating Event	Mean Initiating Event Frequency per Reactor Operating Year		
	Base Case Data after 1988 only	Data up to 1988 only	Data after 1988 with FAC events removed
F_{HP}, Large High Pressure SLB	3.47E-05	3.01E-04	3.04E-06
F_{RSL} , Large Reheat SLB	2.04E-04	1.73E-03	4.38E-06
F_{ESL}, Large Extraction SLB	1.40E-05	2.63E-03	4.77E-07
F_{SLBL}, Large SLB	**2.53E-04**	**4.67E-03**	**7.09E-06**
F_{RSM_L}, Moderate Reheat SLB	1.74E-04	9.39E-04	1.31E-06
F_{ESM}, Moderate Extraction SLB	1.28E-05	1.23E-03	3.18E-07
F_{SLBL}, Moderate Steam Line SLB	**1.87E-05**	**2.17E-03**	**1.62E-06**
F_{FL15}, Large FLB downstream of FWH15	5.85E-05	5.98E-04	4.96E-06
F_{FL45}, Large FLB between FWH14 and FWH15	7.67E-05	8.50E-04	4.42E-06
F_{FLBL}, Large FLB	**1.35E-4**	**1.45E-03**	**9.38E-06**
F_{FL45M}, Moderate FLB	**4.69E-05**	**4.07E-04**	**4.29E-05**

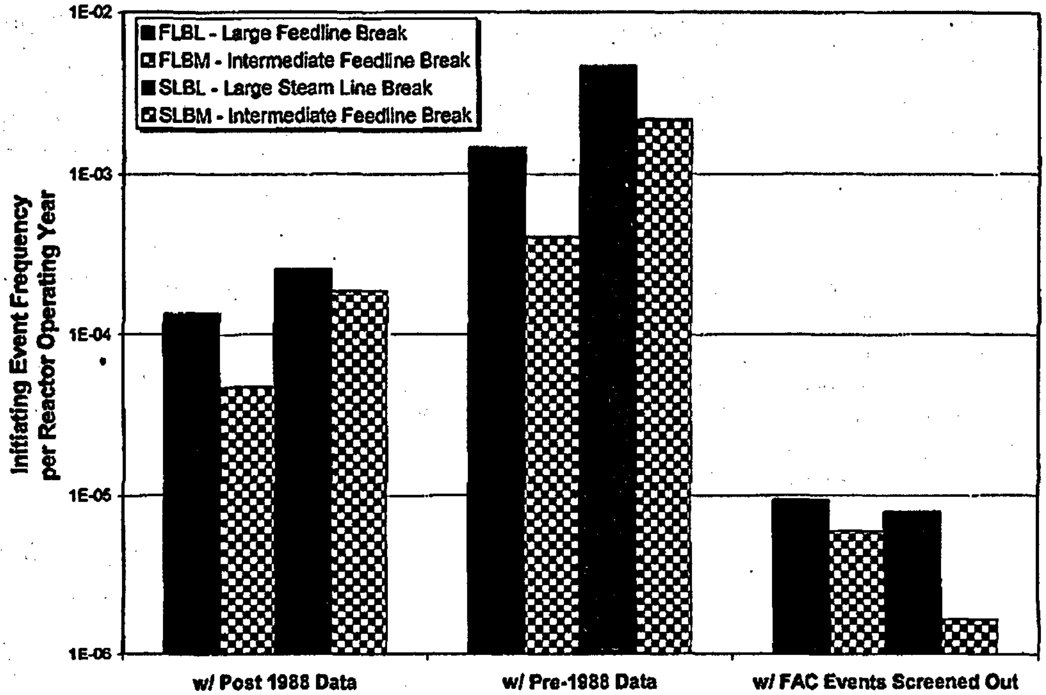


Figure 5-5 Impact of Alternative Data Screening Regarding FAC

6. REFERENCES

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APPENDIX A PIPExp DATABASE DESCRIPTION

A.0 PIPExp / OPDE OVERVIEW

This appendix describes the PIPExp database content and structure, and its relationship with the OECD Pipe Failure Data Exchange Project (OPDE). OPDE was established in 2002 as a cost-shared, multi-national co-operation in piping reliability. The initial objective of OPDE was to establish a comprehensive database on pipe failures in commercial nuclear power plants worldwide and to make the database available to project member organizations that provide data. The project is operated under the umbrella of the OECD Nuclear Energy Agency (NEA). A Clearinghouse is operating the database and provides the quality assurance function. The Clearinghouse is operated by one of the authors of this report.

A.1 Historical Background

The Swedish Nuclear Power Inspectorate (SKI) in 1994 launched a R&D project with the objective of advancing the state-of-art in piping reliability. The stated objective included the following tasks:

- Develop a high-quality, comprehensive database on the service history of piping systems in commercial nuclear power plants.
- In parallel with the database development, identify and develop a general framework for statistical analysis of the service data as recorded in the pipe failure database.
- Perform a pilot application to demonstrate how the pipe failure database and piping reliability analysis framework can be used to develop plant-specific loss of coolant accident (LOCA) frequencies.

A long term strategy for the pipe failure database was formulated during the discussions leading up to the project initiation in mid-1994. This strategy included considerations to establish an international cooperation to support the long term database maintenance and applications program. The R&D project was concluded at the end of 1998. Results of the project included:

- A pipe failure database in Microsoft ACCESS. At the time this database was referred to as "SKI-PIPE", a proprietary database. It included 2291 pipe failure records as of 31-Dec-1998. This version formed the basis of OPDE in 2002 (Figure A-1).
- A series of technical reports (e.g., SKI Reports 95:58, 97:26, 97:32 and 98:30, all available from www.ski.se.

Independent of SKI and in preparation for and support of an international cooperative effort, the maintenance and update of the pipe failure database has continued post-1998. Figure A-1 is a top-level summary of this post-1998 maintenance and update program including the relationship between PIPExp and OPDE. Insights from practical database applications have played a significant role in enhancing and restructuring the database to become tool for piping reliability assessments.

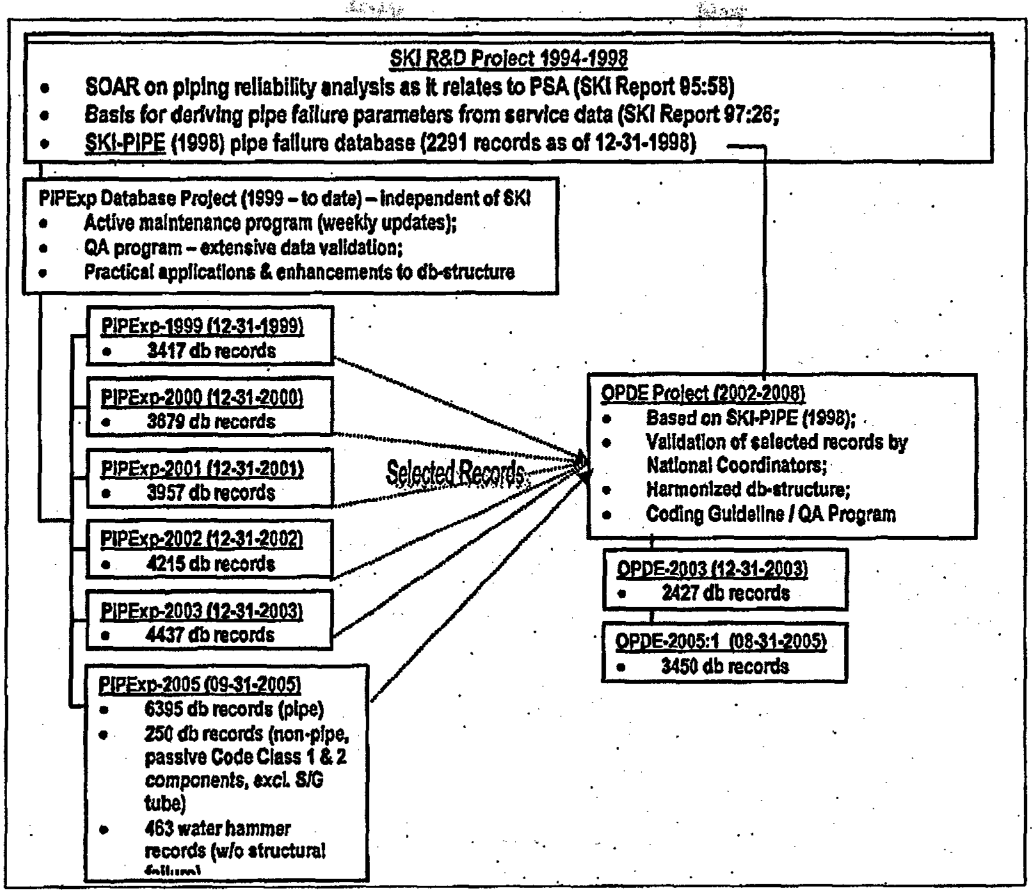


Figure A-1
Evolution of PIPExp Pipe Failure Database

A.2 PIPExp Quality Management

All work associated with database maintenance is controlled by a QA program. Source information including text files, drawings and photographs associated with each database record is stored in an electronic archive. Each data record in PIPExp is assigned a "Quality Index" (or completeness index) per the definitions in Table A-1. The Quality Index is used to assess the completeness and technical accuracy of the source information as well as the classified and coded information in the database. Table A-2 summarizes the evolution of the database since 1998.

Table A-1
Definition of Quality Index for Database Management

Quality-Index	Definition
1	Validated – all source data has been accessed & reviewed – no further action required
2	Validated – source data may be missing some, non-critical information – no further action anticipated
3	Validated – incomplete source data – assumptions made about material grade and/or exact flaw location – no further action anticipated
4	Validation based on incomplete information – depending on application requirements, further action may be necessary
5	Validation based on available, incomplete information – further action expected (e.g., retrieval of additional source data)
6	Not validated – validation is pending, or record is subject to deletion from database

Table A-2
Database Content by Quality Index

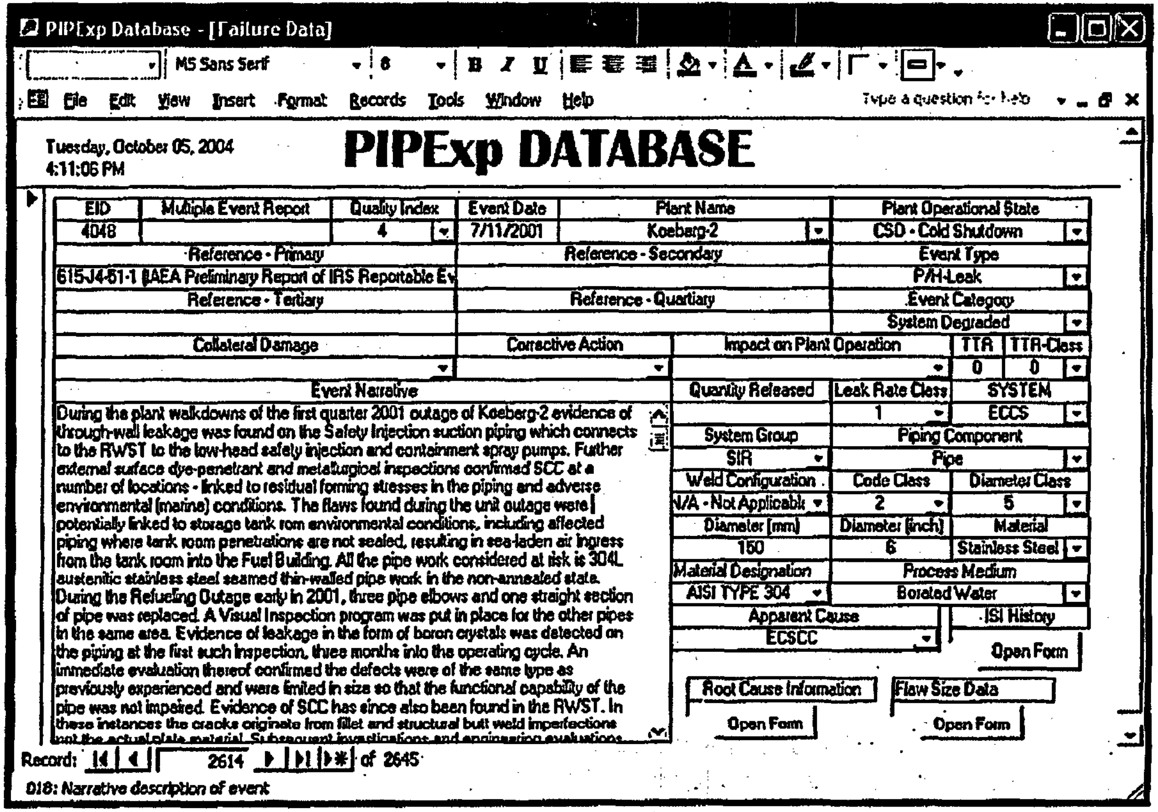
	Database as of 12-31-1998 No. Pipe Failure Records by Quality Index						
Plant Type	**Totals**	**1**	**2**	**3**	**4**	**5**	**6**
BWR	673	210	66	3	74	7	277
PHWR	100	30	3	–	56	1	10
PWR	1376	386	123	6	152	84	746
RBMK	57	3	6	–	19	28	1
	2291	629	198	9	301	120	1034
	Database as of 12-31-2002 No. Pipe Failure Records by Quality Index						
Plant Type	**Totals**	**1**	**2**	**3**	**4**	**5**	**6**
BWR	1872	1216	174	12	219	75	176
PHWR	106	51	2	–	42	11	–
PWR	2077	1011	198	6	351	233	278
RBMK	160	48	–	–	18	81	–
	4215	2290	379	22	721	349	454
	Database as of 09-30-2005 No. Pipe Failure Records by Quality Index						
Plant Type	**Totals**	**1**	**2**	**3**	**4**	**5**	**6**
BWR	2510	1489	300	172	282	204	63
GCR, HWLWR	12	8	–	2	1	1	–
PHWR	131	47	4	23	42	15	–
PWR	3563	1318	323	300	453	1070	99
RBMK	179	12	21	4	110	32	–
	6395	2874	648	501	888	1322	162

A.3 PIPExp Database Input Forms

This section gives and overview of the database input requirements. All data entry is done via the four forms (Form 1 through Form 4).

A.3.1 Form 1 – Event Descriptions

Form 1 is shown in Figure A-2. It consists of 35 fields; seven of which are free-format with the balance defined by roll-down menus with key words (or data filters). The data entry requirements are defined below:



Tuesday, October 05, 2004
4:11:06 PM

EID: 4048
Multiple Event Report:
Quality Index: 4
Event Date: 7/11/2001
Plant Name: Koeberg-2
Plant Operational State: CSD - Cold Shutdown
Reference - Primary: 615-J4-61-1 IAEA Preliminary Report of IRS Reportable Ev
Reference - Secondary:
Event Type: P/H-Leak
Reference - Tertiary:
Reference - Quartiary:
Event Category: System Degraded
Collateral Damage:
Corrective Action:
Impact on Plant Operation:
TTR: 0
TTR-Class: 0
Event Narrative: During the plant walkdowns of the first quarter 2001 outage of Koeberg-2 evidence of through-wall leakage was found on the Safety Injection suction piping which connects to the RWST to the low-head safety injection and containment spray pumps. Further external surface dye-penetrant and metallurgical inspections confirmed SCC at a number of locations - linked to residual forming stresses in the piping and adverse environmental (marine) conditions. The flaws found during the unit outage were potentially linked to storage tank rom environmental conditions, including affected piping where tank room penetrations are not sealed, resulting in sea-laden air ingress from the tank room into the Fuel Building. All the pipe work considered at risk is 304L austenitic stainless steel seamed thin-walled pipe work in the non-annealed state. During the Refueling Outage early in 2001, three pipe elbows and one straight section of pipe was replaced. A Visual Inspection program was put in place for the other pipes in the same area. Evidence of leakage in the form of boron crystals was detected on the piping at the first such inspection, three months into the operating cycle. An immediate evaluation thereof confirmed the defects were of the same type as previously experienced and were limited in size so that the functional capability of the pipe was not impaired. Evidence of SCC has since also been found in the RWST. In these instances the cracks originate from fillet and structural butt weld imperfections not the actual plate material. Subsequent investigations and engineering evaluations
Quantity Released:
Leak Rate Class: 1
SYSTEM: ECCS
System Group: SIR
Piping Component: Pipe
Weld Configuration: N/A - Not Applicable
Code Class: 2
Diameter Class: 5
Diameter [mm]: 150
Diameter [inch]: 6
Material: Stainless Steel
Material Designation: AISI TYPE 304
Process Medium: Borated Water
Apparent Cause: ECSCC
ISI History

Record: 2614 of 2645
018: Narrative description of event

Figure A-2
Event Descriptions – Form 1

Form 1 Data Entry Requirements

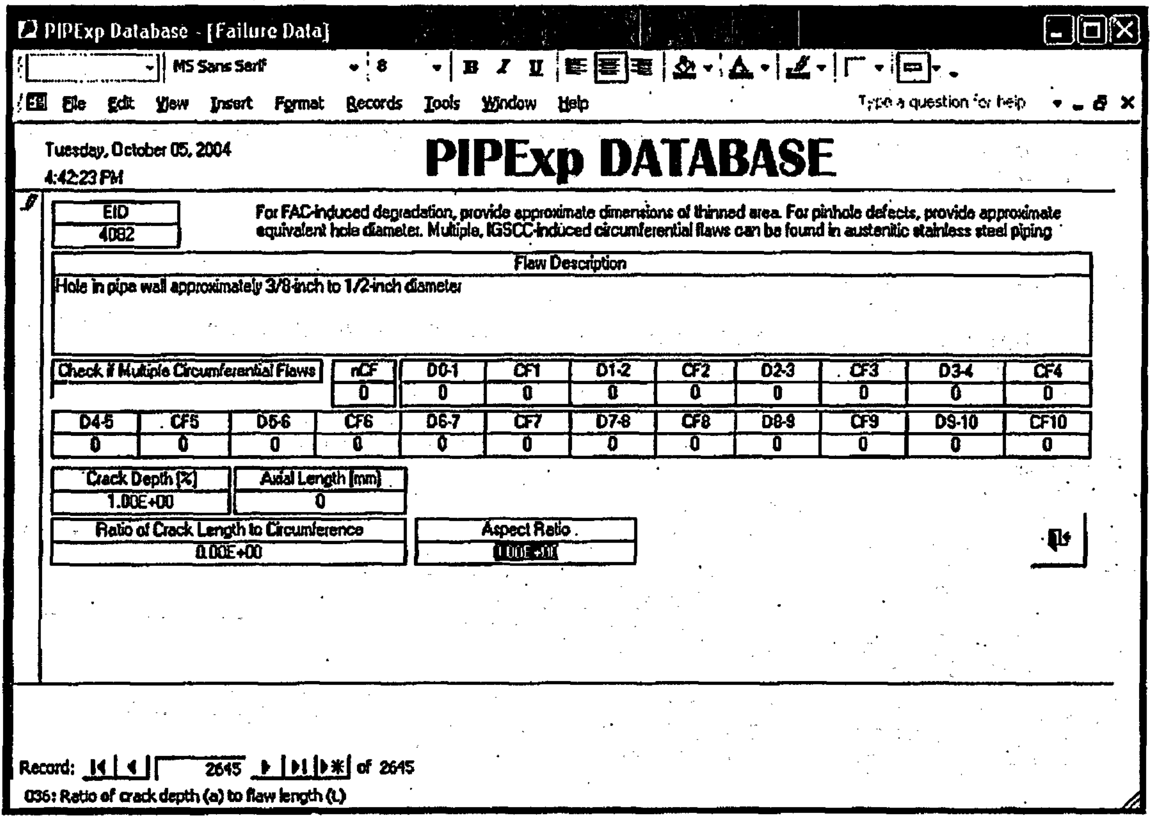
- EID (Event ID) is a uniquely defined database record number (or "primary key"); it is generated automatically by Access.
- Multiple Event Report is checked if one source document (reference) includes information about more than one pipe failure and at different piping system locations.

Mainly, this field supports database management activities (e.g., answer to question "have all pipe failures been adequately recorded in PIPExp?").

- Quality Index (a number 1 to 6); a roll-down menu defines the different options together with definitions.
- Event Date is always required.
- Plant Name; a roll-down menu with listing of all commercial nuclear power plants in NEA member and non-member countries.
- Plant Operational State; a roll-down menu defines the different options.
- Reference; there are four free-format fields for primary and supplemental references. Electronic copies of each reference are stored on CD.
- Event Type; a roll-down menu defines the different options.
- Event Category; a roll-down menu defines the different options.
- Collateral Damage; a roll-down menu defines the different options. "N/A – None" is used as the default.
- Corrective Action; a roll-down menu defines the different options. Note that the term "Temporary Repair" always implies that a "Code Repair" or "Replacement" be performed during the next scheduled outage lasting 30 days or more, but no later than the next refueling outage.
- TTR (Time to Repair) is for the repair time in hours.
- TT-Class is a data filter; a roll-down menu defines the different options with definitions.
- Event Narrative is a free-format memo field.
- Quantity Released is free format field; the dimension can be [lb], [kg], [ton], or [m^3].
- Leak Rate Class is a data filter; a roll-down menu defines the different options with definitions.
- System is a free format field for the system name; a roll-down menu includes a selection of BWR- and PWR-specific, English language names.
- System Group is a data filter; a roll-down menu defines the different options.
- Piping Component is a data filter; a roll-down menu defines the different options.
- Weld Configuration; a roll-down menu defines the different options.
- Code Class; a roll-down menu defines the different options. A cross-reference table compares the different national safety classifications with ASME Section III.
- Diameter Class is a data filter; a roll-down menu defines the different options and definitions.
- Diameter [mm] is used for the measured diameter.
- Diameter [inch] is used for the measured diameter.
- Material is a data filter; a roll-down menu defines the different options.
- Material Designation; a roll-down menu defines the different options. A cross-reference includes different carbon steel and stainless steal material designations.
- Process Medium, a roll-down menu defines the different options.
- ISI History (Form 3) is checked only if information is available.
- Root Cause Information (Form 4) is checked only if information is available.
- Flaw Size Information (Form 2) is checked only if flaw size (e.g., crack orientation, depth, length) information is available.

A.3.2 Form 2 – Flaw Size Information

Form 2 is shown in Figure A-3. It consists of 28 fields. The data entry requirements are defined below:



Tuesday, October 05, 2004
4:42:23 PM

PIPExp DATABASE

EID: 4082

For FAC-induced degradation, provide approximate dimensions of thinned area. For pinhole defects, provide approximate equivalent hole diameter. Multiple, IGSCC-induced circumferential flaws can be found in austenitic stainless steel piping

Flaw Description: Hole in pipe wall approximately 3/8-inch to 1/2-inch diameter

Check if Multiple Circumferential Flaws ☐

nCF	D0-1	CF1	D1-2	CF2	D2-3	CF3	D3-4	CF4
0	0	0	0	0	0	0	0	0

D4-5	CF5	D5-6	CF6	D6-7	CF7	D7-8	CF8	D8-9	CF9	D9-10	CF10
0	0	0	0	0	0	0	0	0	0	0	0

Crack Depth [%]	Axial Length [mm]
1.00E+00	0

Ratio of Crack Length to Circumference	Aspect Ratio
0.00E+00	0.00E+00

Record: 2645 of 2645

036: Ratio of crack depth (a) to flaw length (L)

Figure A-3
Flaw Size Information – Form 2

<u>Form 2 Data Entry Requirements</u>

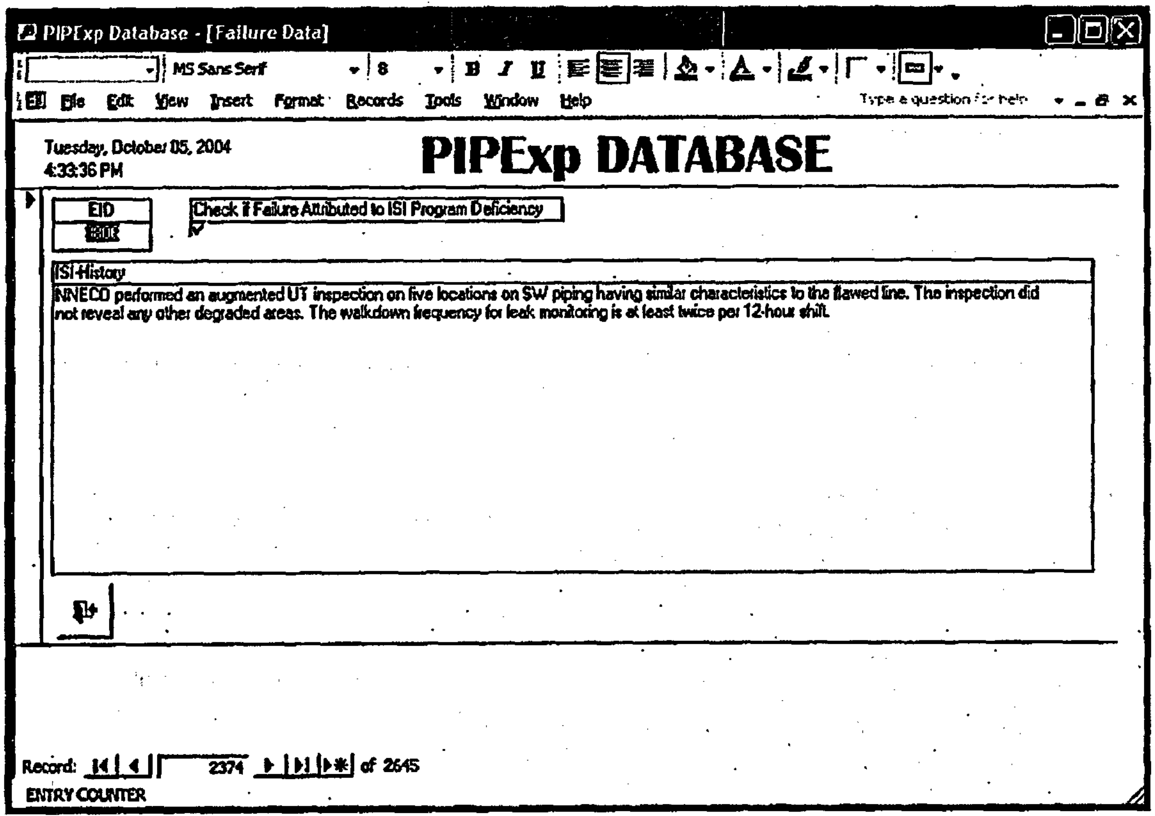
- Flaw Description is a free-format memo field. For through-wall flaws, information about dimensions (e.g., equivalent diameter) should be included in this field. For part through-wall flaws, this field should include information on flaw depth (a) and length (*l*), and orientation. For multiple flaws, the number of flaws and their lengths are recorded in the designated fields.
- Check if Multiple Circumferential Flaws. This check box typically applies to flaws attributed to IGSCC. In PIPExp, on the order of 15% of the records on IGSCC involve multiple, single plane circumferential cracks.
- nCF (number of Circumferential Flaws) includes the total number of flaws in an affected weld.
- D#-## is the distance, in [mm], between adjacent circumferential flaws; e.g., D0-1 is the distance from the TDC (12 o'clock) position to flaw #1, and D2-3 is the distance between

flaw #2 and flaw #3, etc. A blank field indicates that no information on the spacing is available in the database.

- CF-# is the length of circumferential flaw '#' [mm]. The flaw number is relative to the 0-degree position; CF-1 is the first circumferential flaw from the reference position, etc.
- Crack Depth [%] is the ratio of crack depth to pipe wall thickness.
- Axial Length [mm]; this field relates to the Flaw Description.
- Ratio of Crack Length to Circumference; this ratio should be relative to the inside pipe circumference.
- Aspect Ratio; this is the ratio of crack depth to crack length and relates to the information under Flaw Description.

A.3.3 Form 3 – ISI History

Form 3 is shown in Figure A-4. It consists of 3 fields. While primarily intended for ISI program weaknesses, the free-format field may be used to document any information pertaining to the ISI of the affected component, or ISI history such as time of most recent inspection.



Tuesday, October 05, 2004
4:33:36 PM

EID: [illegible]

Check if Failure Attributed to ISI Program Deficiency ☑

ISI History: NNECO performed an augmented UT inspection on five locations on SW piping having similar characteristics to the flawed line. The inspection did not reveal any other degraded areas. The walkdown frequency for leak monitoring is at least twice per 12-hour shift.

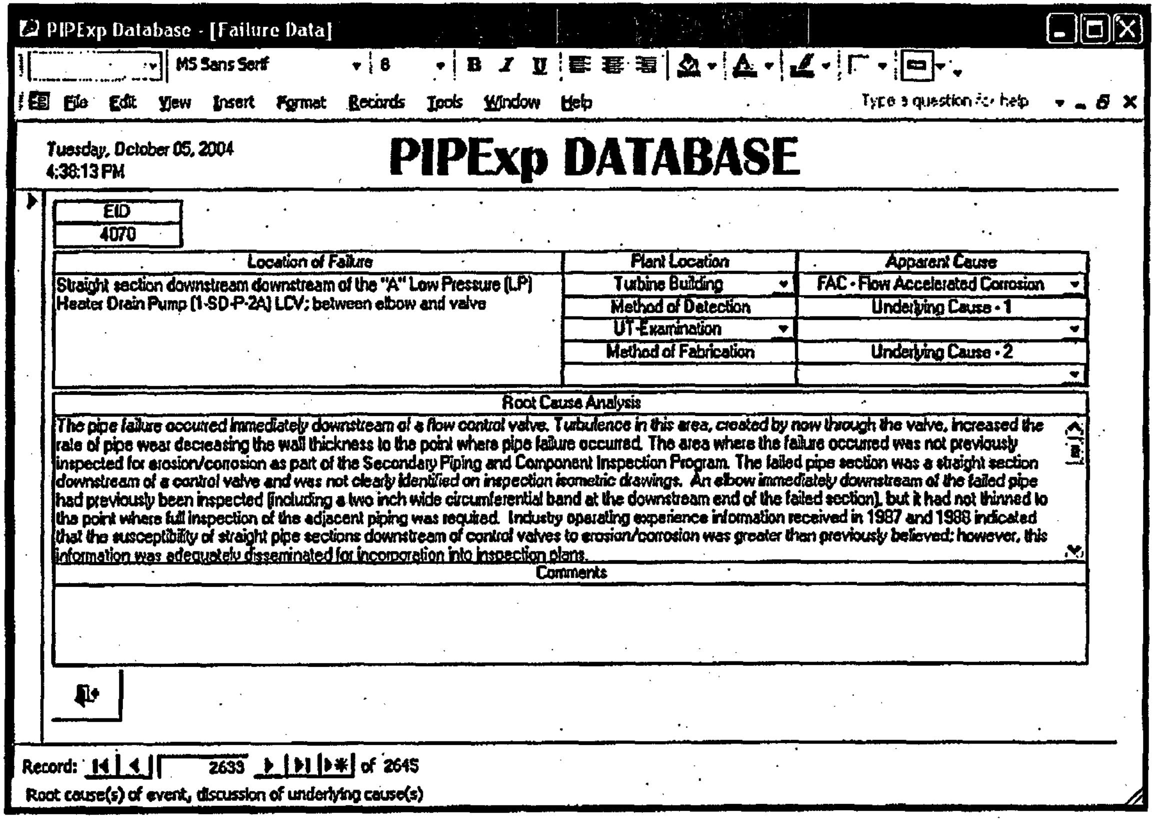
Record: 2374 of 2645

ENTRY COUNTER

Figure A-4
ISI History – Form 3

A.3.4 Form 4 – Root Cause Information

Form 4 is shown in Figure A-5. It consists of 9 fields. The data entry requirements are defined below:



Tuesday, October 05, 2004
4:38:13 PM

PIPExp DATABASE

EID: 4070

Location of Failure	**Plant Location**	**Apparent Cause**
Straight section downstream downstream of the "A" Low Pressure (LP) Heater Drain Pump (1-SD-P-2A) LCV; between elbow and valve	Turbine Building	FAC - Flow Accelerated Corrosion
	Method of Detection	**Underlying Cause - 1**
	UT-Examination	
	Method of Fabrication	**Underlying Cause - 2**

Root Cause Analysis

The pipe failure occurred immediately downstream of a flow control valve. Turbulence in this area, created by now through the valve, increased the rate of pipe wear decreasing the wall thickness to the point where pipe failure occurred. The area where the failure occured was not previously inspected for erosion/corrosion as part of the Secondary Piping and Component Inspection Program. The failed pipe section was a straight section *downstream of a control valve and was not clearly identified on inspection isometric drawings. An elbow immediately downstream of the failed pipe* had previously been inspected (including a two inch wide circumferential band at the downstream end of the failed section), but it had not thinned to the point where full inspection of the adjacent piping was required. Industry operating experience information received in 1987 and 1988 indicated *that the susceptibility of straight pipe sections downstream of control valves to erosion/corrosion was greater than previously believed; however, this* information was adequately disseminated for incorporation into inspection plans.

Comments

Record: 2633 of 2645

Root cause(s) of event, discussion of underlying cause(s)

Figure A-5
Root Cause Information – Form 4

<u>Form 4 Data Entry Requirements</u>

- Location of Failure; this is a free-format memo field describing the location of a flaw (e.g., line or weld number, or using a P&ID reference).
- Plant Location; a roll-down menu defines the different options.
- Method of Detection; a roll-down menu defines the different options.
- Method of Fabrication; a free-format text field.
- Apparent Cause; a roll-down menu defines the different options. Normally this field has already been filled in.
- Underlying Cause – 1; a roll-down menu defines possible contributing factors.
- Underlying Cause – 2; a roll-down menu defines possible contributing factors.
- Root Cause Analysis; a free-format memo field. This field should include any relevant information on the cause-consequence relationship and should be supplemental to the Event Narrative in Form 1.

- Comments; a free-format memo field. It is intended for any other, relevant information that is not captured by other database fields.

A.4 Database Accessibility

PIPExp is a proprietary database whereas the OPDE database is restricted. The full OPDE database is available to participating organizations that supply data. An unrestricted version of OPDE ('OPDE-Light') is available to interested parties upon request to respective National Coordinator (the U.S. representative in the project is the Nuclear Regulatory Agency, Office of Nuclear Regulatory Research). OPDE-Light does not include any proprietary information or any information that enables the identification of plant name.

APPENDIX B PIPE FAILURE RATES & RUPTURE FREQUENCIES APPLICABLE TO NON-CODE PIPING SYSTEMS

Table B-1 FWC Piping Failure Rates & Rupture Frequencies

Case	**Description**	**Uncertainty Distribution**			
		Mean [1/ft.yr]	**5th Percentile**	**Median**	**95th Percentile**
KNPP01	EBS1 - FWC Pipe Failure Rate - with post 1988 data	3.19E-06	1.99E-06	2.97E-06	5.92E-06
	EBS1 - FWC Pipe Rupture - with post 1988 data	6.72E-08	3.80E-08	6.19E-08	1.24E-07
KNPP02	EBS2 - FWC Pipe Failure Rate - with data through 1988	3.56E-06	2.21E-06	3.31E-06	6.59E-06
	EBS2 - FWC Pipe Rupture - with post 1988 data	1.09E-07	6.25E-08	1.01E-07	2.00E-07
KNPP03	EBS1 - FWC Pipe Failure Rate - with data through 1988	2.78E-05	1.73E-05	2.60E-05	5.20E-05
	EBS1 - FWC Pipe Rupture - with data through 1988	5.85E-07	3.39E-07	5.41E-07	1.09E-06
KNPP04	EBS2 - FWC Pipe Failure Rate - with data through 1988	3.98E-05	2.49E-05	3.73E-05	7.45E-05
	EBS2 - FWC Pipe Rupture - with data through 1988	1.22E-06	7.27E-07	1.14E-06	2.28E-06
KNPP05	EBS1 - FWC Pipe Failure Rate – with FAC events screened out	9.21E-07	5.23E-07	8.45E-07	1.70E-06
	EBS1 - FWC Pipe Rupture – with FAC events screened out	8.60E-09	3.64E-09	7.66E-09	1.68E-08
KNPP06	EBS2 - FWC Pipe Failure Rate – with FAC events screened out	8.29E-07	4.41E-07	7.56E-07	1.52E-06
	EBS2 - FWC Pipe Rupture – with FAC events screened out	6.35E-09	2.40E-09	5.55E-09	1.30E-08

Table B-2 Steam Extraction Piping Failure Rates & Rupture Frequencies

Case	**Description**	**Uncertainty Distribution**			
		Mean [1/ft.yr]	**5th Percentile**	**Median**	**95th Percentile**
KNPP07	EBS1 – Steam Extraction Pipe Failure Rate with post 1988 data	3.40E-06	1.65E-06	3.06E-06	6.41E-06
	EBS1 – Steam Extraction Pipe Rupture with post 1988 data	7.71E-08	3.19E-08	6.37E-08	1.39E-07
KNPP08	EBS2 – Steam Extraction Pipe Failure Rate with post 1988 data	2.58E-06	1.00E-06	2.23E-06	5.31E-06
	EBS2 – Steam Extraction Pipe Rupture with post 1988 data	7.93E-08	2.89E-08	6.75E-08	1.68E-07
KNPP09	EBS1 – Steam Extraction Pipe Failure Rate with data through 1988	3.32E-04	2.06E-04	3.10E-04	6.17E-04
	EBS1 – Steam Extraction Pipe Rupture with data through 1988	6.99E-06	4.03E-06	6.45E-06	1.28E-05
KNPP10	EBS2 – Steam Extraction Pipe Failure Rate with data through 1988	4.86E-04	3.03E-04	4.55E-04	9.05E-04
	EBS2 – Steam Extraction Pipe Rupture with data through 1988	1.49E-05	8.78E-06	1.38E-05	2.73E-05
KNPP11	EBS1 – Steam Extraction Pipe Failure Rate – with FAC events screened out	1.93E-07	1.32E-08	9.36E-08	6.73E-07
	EBS1 – Steam Extraction Pipe Rupture – with FAC events screened out	1.80E-09	1.10E-10	8.25E-10	6.45E-09
KNPP12	EBS2 – Steam Extraction Pipe Failure Rate – with FAC events screened out	2.68E-07	1.64E-08	1.23E-07	9.58E-07
	EBS2 – Steam Extraction Pipe Rupture – with FAC events screened out	2.07E-09	1.05E-10	8.81E-10	7.63E-09

Table B-3 LP Steam Piping Failure Rates & Rupture Frequencies

Case	Description	Uncertainty Distribution			
		Mean [1/ft.yr]	5th Percentile	Median	95th Percentile
KNPP13	EBS1 – LP Steam Piping Failure Rate - with post 1988 data	1.33E-05	7.89E-06	1.23E-05	2.42E-05
	EBS1 – LP Steam Piping Rupture - with post 1988 data	2.80E-07	1.47E-07	2.56E-07	5.14E-07
KNPP14	EBS2 – LP Steam Piping Failure Rate - with post 1988 data	1.07E-05	5.82E-06	9.75E-06	1.96E-05
	EBS2 – LP Steam Piping Rupture - with post 1988 data	3.29E-07	1.64E-07	2.97E-07	6.13E-07
KNPP15	EBS1 – LP Steam Piping Failure Rate - with data through 1988	7.15E-05	4.45E-05	6.66E-05	1.33E-04
	EBS1 – LP Steam Piping Rupture – with data through 1988	1.51E-06	8.49E-07	1.39E-06	2.77E-06
KNPP16	EBS2 – LP Steam Piping Failure Rate - with data through 1988	9.09E-05	5.66E-05	8.45E-05	1.68E-04
	EBS2 – LP Steam Piping Rupture – with data through 1988	2.79E-06	1.60E-06	2.57E-06	5.07E-06
KNPP17	EBS1 – LP Steam Piping Failure Rate – with FAC events screened out	2.25E-07	1.47E-08	1.07E-07	7.87E-07
	EBS1 – LP Steam Piping Rupture – with FAC events screened out	2.10E-09	1.19E-10	9.44E-10	7.48E-09
KNPP18	EBS2 – LP Steam Piping Failure Rate – with FAC events screened out	9.22E-07	1.78E-07	6.58E-07	2.52E-06
	EBS2 – LP Steam Piping Rupture – with FAC events screened out	7.05E-09	1.11E-09	4.76E-09	2.04E-08

Table B-4 HP Steam Piping Failure Rates & Rupture Frequencies

Case	Description	Uncertainty Distribution			
		Mean [1/ft.yr]	5th Percentile	Median	95th Percentile
KNPP19	EBS1 – HP Steam Piping Failure Rate - with post 1988 data	3.25E-06	1.62E-06	2.94E-06	6.01E-06
	EBS1 – HP Steam Piping Rupture – with post 1988 data	3.03E-08	1.16E-08	2.64E-08	6.28E-08
KNPP20	EBS2 – HP Steam Piping Failure Rate - with post 1988 data	1.16E-06	3.33E-07	9.37E-07	2.75E-06
	EBS2 – HP Steam Piping Rupture – with post 1988 data	8.90E-09	2.01E-09	6.78E-09	2.26E-08
KNPP21	EBS1 – HP Steam Piping Failure Rate - with data through 1988	1.60E-05	9.34E-06	1.47E-05	2.94E-05
	EBS1 – HP Steam Piping Rupture – with data through 1988	1.49E-07	6.40E-08	1.34E-07	2.90E-07
KNPP22	EBS2 – HP Steam Piping Failure Rate - with data through 1988	2.50E-05	1.47E-05	2.30E-05	4.60E-05
	EBS2 – HP Steam Piping Rupture – with data through 1988	1.91E-07	7.72E-08	1.70E-07	3.78E-07
KNPP23	EBS1 – HP Steam Piping Failure Rate – with FAC events screened out	1.74E-07	1.23E-08	8.44E-08	5.93E-07
	EBS1 – HP Steam Piping Rupture – with FAC events screened out	1.64E-09	9.98E-11	7.52E-10	5.71E-09
KNPP24	EBS2 – HP Steam Piping Failure Rate – with FAC events screened out	2.36E-07	1.53E-08	1.12E-07	8.29E-07
	EBS2 – HP Steam Piping Rupture – with FAC events screened out	1.80E-09	9.99E-11	8.01E-10	6.49E-09

Appendix B

Flood Area Definition for Turbine Building Basement

Flood Area Definition for Turbine Building Basement

Prepared by: R. SHARPE VIA EMAIL
Signature Print Name

12/01/05
Date

Reviewed by: R.J. DREMEL VIA EMAIL
Signature Print Name

12/01/05
Date

1.0 PURPOSE

Internal floods are defined as those floods that result from the failure, incorrect operation (including errors in maintenance), or incorrect alignment of components within the plant. Accident sequences initiated by internal floods can be a significant contributor to risk because of the potential of the event to impair, simultaneously, multiple components required for accident mitigation. The overall objective of the internal flooding analysis is to determine the contribution of accident sequences initiated by such flooding events to core damage and individual accident class frequencies.

An internal flooding PRA requires that areas of the plant be identified that contain equipment needed to mitigate accidents and that are subject to flooding effects. Areas are defined as separate for flooding purposes where physical boundaries are present that prevent propagation of a flood source in one area from causing damage to equipment in another area. For each flood area, flooding sources are identified in the area that have the potential to damage equipment within the area or that have the potential to propagate from the area to another area and damage equipment needed for accident mitigation. Propagation paths are identified and defined between flood areas. The flooding walkdowns confirmed the boundaries between flood areas and identified barriers in the boundaries that separate the flood areas. This information can be used to identify areas of the plant that can be designated as separate, independent flood areas.

In order to streamline the accident sequence analysis it is beneficial to limit the analysis to only those sequences that will contribute to flooding risk. Such risk-significant sequences are identified through the application of a screening process. Each flood area is evaluated against important factors including the existence of flooding initiators, the existence of safety-significant equipment, and ability of a flooding initiator to cause a reactor trip to determine which flood areas are worthy of additional analysis and quantification.

This document designates independent flood areas that will be analyzed in further detail through accident sequence analysis and initiating event frequency analysis. Development of independent flood areas will support the high level requirements identified in the Flood Area Definition Guideline [GUIDE01]. This document also applies a screening analysis to all the defined flood areas to focus the accident sequence analysis on risk-significant scenarios. Application of a screening process will support some of the high level requirements identified in the Accident Sequence Analysis Guideline [GUIDE02].

2.0 METHODOLOGY

2.1 Flood Area Definition

A flood area addresses physical boundaries that impact the propagation of water and the potential to damage equipment. The subject water originates from a pipe break in the flood area and is categorized as a submergence event or a spray event, thus flood areas are defined differently for these two events. A submergence event is defined as a pipe break with sufficient flow rate to overwhelm the flood area's flood mitigation equipment, accumulate in such a manner as to damage equipment, and has the potential to propagate laterally in an amount significant enough to damage additional equipment. A spray event is a pipe break with a flow rate within the capacity of the flood area's mitigation equipment (especially floor drains) that can damage equipment from direct spray, but cannot propagate laterally. Thus, a pipe break in a spray flood area is expected to result in direct damage to equipment in the area and to result in the vast majority of that water exiting the flood area via the floor drains and floor openings. Any water that propagates laterally from a spray flood area will be of insignificant quantity and will not cause equipment damage in adjoining areas. Each flood area is analyzed for both spray events and submergence events.

Lateral propagation for submergence events occurs due to lack or failure of barriers separating the areas. Typical barrier failures are of doors, but could include water that flows through open penetrations through walls, water that flows over protective curbs and weirs, water that backflows through drain lines, and structural failure of gypsum walls. Normally closed access doors are able to withstand some amount of force due to accumulated water, however when the water level reaches a critical depth the door is expected to fail. Thus, water will propagate laterally through such failed doors. Since a failure is required for this lateral propagation to occur, these lateral zones are not included in the flood area definition. Only zones with open communication are considered in defining flood areas.

2.2 Assumptions

The following assumptions were utilized in the definition of flood areas:

1. Leakage under and around doors is the only form of drainage inside Safeguards Alley since a pipe break in the Turbine Hall will fill the Turbine Building sump and subsequently fill the drain lines connecting the sump to the floor drains in Safeguards Alley. Thus, the floor drains will not be able to remove water from Safeguards Alley. Gaps under doors are documented as part of the GOTHIC input. [CALC02]

2. KPS access doors inside the Turbine Building generally can withstand a water height of 4 feet when the water is pushing the door open and 5 feet when the water is pushing the door closed. Exceptions to this include doors 243 and 244 which can withstand 3 feet 3 inches when water is opening the door and 4 feet 9 inches when water is closing the door, and door 8 which can withstand 3 feet 9 inches when water is opening the door. [CALC01]

3. All junction boxes are gasketed and not vulnerable to spray unless otherwise noted in the Walkdown Sheets [FLOOD01].

4. Flood-induced failure of motor-operated valves (MOVs) involves the valve operator's loss of

function, but does not involve the MOV changing position. The MOV is expected to remain in the same position, however any new change in position will require manual action to turn the handwheel.

5. Flood-induced failure of air-operated valves (AOVs) involves the valve operator's loss of function, but would also involve the AOV failing to its fail-safe position.

6. Sealed penetrations are assumed to pass no fluid. Penetrations make use of various types of sealant including grout and elastomer. Grout behaves similar to concrete and is basically impervious to water.

7. Cable insulation is not subject to failure from submergence or spray.

8. Walls and trench barriers are assumed to remain intact throughout a flooding event with the exception of the firewall separating flood areas TU-95A and TU-95B-1. This gypsum wall was analyzed and determined to be structurally capable of withstanding only approximately 3 feet of water. [CALC01]

9. The probability of rupture of encapsulated high-energy lines is insignificant as both the inner pipe and the surrounding guard pipe must fail.

10. Environmentally qualified (EQ) components are assumed to be able to perform their safety functions when exposed to spray conditions and high heat and humidity due to a pipe break. For example, the solenoid operators for feedwater valves in the feedwater valve room, by design, perform their safety functions during high-energy line break (HELB) events.

11. Lines that are not normally pressurized or charged such as drain lines and dry fire protection piping are not considered as credible flood or spray sources.

12. Flooding in containment is not considered in this analysis. This is a subset of the Loss of Coolant Initiating Event (LOCA) in the Internal Events PRA.

13. Rupture of seismic Class I tanks (e.g., concrete reinforced refueling water storage tank) is not considered credible in this analysis.

14. Failure of a fire protection deluge valve is not analyzed as a potential initiator in this analysis. The flow rate of a single deluge valve is insufficient to cause flooding concerns in Safeguards Alley. The simultaneous failure of multiple deluge valves has an insignificantly small probability.

3.0 DESCRIPTION OF FLOOD AREAS IN TURBINE BUILDING BASEMENT

Flood areas were defined in a previous analysis [FLOOD01]. Walkdowns of the various flood areas were performed earlier and are documented on walkdown forms [Appendix C of FLOOD01]. For each flood area these walkdowns recorded information that included resident equipment, flood sources, and barriers (including doors). The information from these walkdowns combined with the information obtained from the general arrangement drawings [DWG01] provides the basis for the following flood area descriptions.

This analysis is concerned only with flood events that originate in the Turbine Building and then propagate to Safeguards Alley. Therefore, flood areas associated with the Battery Rooms on the mezzanine level of the Turbine Building and the Turbine Oil Storage Area in the Turbine Building basement as well as flood areas in the Auxiliary Building are disregarded as flooding areas of interest in this analysis.

Figure 1 identifies the various flood areas and their important features.

TU-22-1

Description - Flood Zone TU-22-1 comprises all of the general areas of the Turbine Building including the Operating Deck on the 626'-0" elevation, the Mezzanine Floor on the 606'-0" elevation, and the Basement on the 586'-0" elevation. It also includes the rooms in the south end of the Auxiliary Building basement from the waste neutralizer tank (room 17B) west to the Reactor Building Support Ring (room 11B) since these rooms are not part of the radiological area and communicate openly with the Turbine Building basement. Additionally, the shop area, working material storage area, and steam generator blowdown area in the south end of the 606'-0" elevation of the Auxiliary Building are also included in this flood zone since these rooms communicate openly with the other Auxiliary Building rooms in Flood zone TU-22-1 and since these areas are not part of the radiologically controlled area. Table 1 contains a complete listing of the fire zones and room numbers that comprise each flood zone. On the 626'-0" elevation the zone is bounded on the north by the Technical Support Center and exterior walls, on the south by the Transformer Area and exterior walls, on the west by the Auxiliary Building, and on the east by the Administration Building and exterior walls. On the 606'-0" elevation the zone is bounded on the north by zones TU-97 and TU-98, the Technical Support Center, exterior walls, the Containment Building, and zone AX-32-1, on the south by the Transformer Area, exterior walls, and zones AX-33 and AX-39, on the west by the Auxiliary Building and Containment Building, and on the east by the Administration Building and exterior walls. Zones TU-94, TU-95A, TU-95B-1, TU-95B-2, TU-95C, TU-96, TU-97, and TU-98 lie beneath zone TU-22-1 and zone TU-96 and the Turbine Building roof lie above.

All wall and ceiling penetrations are sealed. The floor of this zone is the basement floor and is finished concrete. The north wall has a normally-closed door (120) on the 626'-0" elevation of the Auxiliary Building leading to a stairwell, normally-closed doors (47 and 48) leading to zones TU-97 and TU-98, normally-closed doors (46 and 280) on the 606'-0' elevation of the Turbine

Building leading outdoors, normally-closed doors (11, 15, and 16) on the 586'-0" elevation of the Auxiliary Building leading to zones AX-20B, AX-21-1, and AX-23A-1, normally-closed door 401 leading to zone TU-94, and normally-closed doors (4 and 6) leading to zone TU-95B-1. The south wall has a normally-closed door (117) on the 626'-0" elevation leading to the Control Room, a normally-closed roll-up door (42) on the 606'-0" elevation leading to the outdoors, normally-closed doors (70 and 74) leading to zones AX-39 and AX-33 on the 606'-0" elevation, and no doors on the 586'-0" elevation. The east wall has a normally-closed door (109) on the 626'-0" elevation and a normally-closed door (39) on the 606'-0" elevation leading to the Administrative Building, and no doors on the 586'-0" elevation. The west wall has normally-closed doors (118, 133, and 161) leading to zones AX-32-1 and AX-37 on the 626'-0" elevation, normally-closed doors (41, 44, and 49) leading to zones AX-32-1 and AX-30 on the 606'-0" elevation of the Auxiliary Building, a normally-open door (68) leading to the dosimetry offices on the 606'-0" elevation, a normally-closed door (75) in the Electric Shop leading outdoors, and a normally-closed door (7) on the 586'-0" elevation leading to zone TU-96. The east wall has a normally-closed door (109) on the 626'-0" elevation leading to the Administration Building, normally-closed doors (39 and 40) on the 606'-0" elevation leading to the Administration Building and outdoors, and no doors on the 586'-0" elevation.

The major PRA equipment in zone TU-22-1 includes the feedwater pumps (1A and 1B), the condensate pumps (1A and 1B), MCC 45-F, and the Redundant Overspeed Trip System Cabinet. The Internal Flood Walkdown Form [Appendix C of FLOOD01] for zone TU-22-1 contains a complete listing of the flood-susceptible PRA equipment in this zone.

Potential flood sources in this zone include fire protection piping, feedwater piping, service water piping, main steam piping, and circulating water piping which are the primary flood sources and represent both a flooding hazard and a spray hazard.

Flood mitigation is present in this zone in the form of floor grating, open stairways and sump pumps.

Analysis – Water from a pipe break in TU-22-1 will readily propagate to the basement level. The effects of a spray source in any part of the zone are limited to equipment in the vicinity of the spray source. Water is likely to splash onto equipment on lower levels as it passes through the floor grating. Accumulation is possible in the basement level (586'-0") of the zone.

As water from any pipe break in zone TU-22-1 makes its way into the Turbine Building basement, it will eventually fill the Turbine Building Sump. The sump contains two pumps with design capacities of < 100 gpm each. The level switch for the Turbine Building sump pump control is a mechanically alternating device. A high water level (30") starts one pump. A return to low level (12") stops the pump. A subsequent high level starts the alternate pump. [SYSTEM01]

If a high-high water level (34.5") is reached, the level switch starts the second pump. Both pumps continue to run until an intermediate level cutoff point, 19", is reached. At this point, the level switch turns off the leading (first) pump. The lagging (second) pump continues to run until the low-level setpoint, 12", is reached. [SYSTEM01]

The Turbine Building sump contains Level Switch LA-16666 that actuates Control Room Alarm 47033P when a high-high-high water level setpoint, 34.5", is reached. [SYSTEM01]

Thus, only pipe breaks of greater than 200 gpm, the combined discharge capacity of the Turbine Building sump pumps, are of concern for zone TU-22-1.

The first indication of such a break would be a Turbine Building sump high level alarm in the control room. The procedure for abnormal operation of the miscellaneous drains and sumps instructs the operator to dispatch someone to investigate the source of the alarm. If the source of leakage is from a break in the Circulating Water System, the operator is instructed to trip the circulating water pumps, trip the reactor, and perform a shutdown using emergency operating procedure E-0.

The effectiveness of such operator actions is dependent on the size of the pipe break. A small pipe break would likely afford the operator the time to perform the actions necessary to protect vital equipment in the Turbine Building basement. A large break would result in significant accumulation in the Turbine Building basement and could challenge the flood protection features in place to protect equipment located in adjacent zones. Water level in areas TU-94, TU-95B-1, TU-95B-2, and TU-95C would closely mirror the water level in TU-22-1 due to leakage under doors 4, 6, and 401 and due to flow through the drain lines that connect Safeguards Alley and the Turbine Building sump (these lines do not have check valves). Since drainage in these areas will be disabled due to the water in the Turbine Building, water will begin to accumulate in these rooms and begin to propagate to zones TU-90, TU-92, and TU-95A due to leakage under doors 3, 263, and 268. Power to the motor loads on the 4 kV buses in TU-90 and TU-92 will fail when the water level reaches 4 inches, submerging the lockout relays and tripping the breakers associated with the motor loads.

Summary – Pipe breaks in zone TU-22-1 can result in both equipment spray and submergence. For spray events TU-22-1 becomes a flood area by itself since only equipment in TU-22-1 is susceptible to damage from direct spray originating in zone TU-22-1. However, water from such a spray event can result in the splashing of equipment in other elevations of the zone. For submergence events, zone TU-22-1 combines with all the zones in Safeguards Alley due to leakage under the doors associated with these rooms. When the water level in the Turbine Building sump reaches the high-high setpoint (approximately 34.5 inches above the sump floor) an annunciator sounds in the control room. Power to the motor loads on the 4 kV buses is expected to fail at 4 inches of water (although power will still be available to the 480 V buses), the 480 V buses will then fail at 11 inches of water, and the turbine-driven auxiliary feedwater (TDAFW) pump will fail to start at 9 inches of water and fail to continue running at approximately 18 inches of water.

Zone TU-22-1 is a relatively large room such that any water from a pipe break is expected to spray only equipment in zone TU-22-1 that is in close proximity to the pipe break.

For a pipe break in zone TU-22-1, equipment in zones TU-22-1, TU-90, TU-92, TU-94, TU-95A, TU-95B-1, TU-95B-2 and TU-95C can be vulnerable and could be at risk.

TU-90

Description - Flood Zone TU-90 is Diesel Generator Room 1A on the 586'-0" elevation. The zone is bounded on the north by an exterior wall, on the south by zone TU-92, on the east by an exterior wall and the pipe tunnel leading to the Screenhouse, and on the west by zones TU-94 and TU-95A. The Administrative Building lies above zone TU-90 and exterior soil lies below. All penetrations in zone TU-90 are sealed. The south wall has a normally-closed access door (2) leading to a Screenhouse pipe tunnel and the west wall has a normally-closed access door (136) leading to zone TU-95A.

The major PRA equipment in zone TU-90A includes Diesel Generator 1A, 4 kV Switchgear Bus 5, and MCC 52A. The Internal Flood Walkdown Form (Appendix C of FLOOD01] for zone TU-90 contains a complete listing of the flood-susceptible PRA equipment in this zone.

Potential flood sources in this zone include service water piping and fire protection piping which represent both a flooding hazard and a spray hazard.

Flood mitigation is present in this zone in the form of a trench which is sealed to prevent flow from traveling to zones TU-94 and TU-95A, but is open via a 4-inch pipe to the pipe tunnel leading to the Screenhouse. Floor drains will transfer water to the Turbine Building sump.

Analysis – Water from a pipe break in TU-90 will easily propagate to the pipe tunnel leading to the Screenhouse through an open 4-inch pipe that connects the two areas in the existing trench. Floor drains will divert water to the Turbine Building sump. Thus, only pipe breaks that exceed the capability of the floor drains are a concern for accumulation in zone TU-90. Equipment damage from spray sources in TU-90 is limited to the equipment residing in that zone.

Zone TU-90 is equipped with two normally closed doors that initially prevent lateral propagation of water from pipe breaks beyond the capacity of the floor drains. Door 2 (double door with a 1/64" gap) opens outwardly to the pipe tunnel leading to the Screenhouse and door 136 (double door with a 1/8" gap) opens inwardly from zone TU-95A. Initially water would flow through the floor drains to the Turbine Building sump and flow through the open 4-inch pipe to the Screenhouse sump. However, given the limited capacity of the floor drains and the 4-inch pipe in TU-90, neither the Turbine Building sump nor the Screenhouse sump will reach a level high enough to initiate a control room alarm. While water is flowing through the floor drains it will also be leaking under door 2 to the pipe tunnel that leads to the Screenhouse. Once the seiche hump is overcome in the pipe tunnel, water leaking under the door will also flow to the Screenhouse sump. When the water level inside TU-90 reaches a critical height, both doors are expected to fail allowing water to freely propagate to zone TU-95A and the pipe tunnel leading to the Screenhouse. A significant flow of water through a pipe break would be required for any accumulation of water in TU-90.

The Turbine Building Sump contains two pumps with design capacities of < 100 gpm each. The level switch for the Turbine Building sump pump control is a mechanically alternating device. A high water level (30") starts one pump. A return to low level (12") stops the pump. A subsequent high level starts the alternate pump. [SYSTEM01]

If a high-high water level (34.5") is reached, the level switch starts the second pump. Both pumps continue to run until an intermediate level cutoff point, 19", is reached. At this point, the level switch turns off the leading (first) pump. The lagging (second) pump continues to run until the low-level setpoint, 12", is reached. [SYSTEM01]

The Turbine Building sump contains Level Switch LA-16666 that actuates Control Room Alarm 47033P when a high-high-high water level setpoint, 34.5", is reached. [SYSTEM01]

The Screenhouse sump contains two pumps with design capacities of < 100 gpm each. The level switch for the Screenhouse sump pump control is a mechanically alternating device. A high water level (30") starts one pump. A return to low level (12") stops the pump. A subsequent high level starts the alternate pump.

If a high-high water level (34.5") is reached, the level switch starts the second pump. Both pumps continue to run until an intermediate level cutoff point, 19", is reached. At this point, the level switch turns off the leading (first) pump. The lagging (second) pump continues to run until the low-level setpoint, 12", is reached.

The Screenhouse sump contains Level Switch LA-16669 that actuates Control Room Alarm 47033P when a high-high-high water level setpoint, 34.5", is reached [SYSTEM01].

The operator's first indication of a pipe break inside TU-90 will be the high Screenhouse sump level alarm in the control room once the water level inside TU-90 rises high enough to fail door 2 which will allow water to flow freely to the Screenhouse. The only other possible indication of a pipe break would be equipment failure that forces an operator to investigate locally.

Summary – Pipe breaks in zone TU-90 can result in both equipment spray and submergence. For spray events TU-90 becomes a flood area by itself since only equipment in TU-90 is susceptible to damage from direct spray originating in zone TU-90. For submergence events, zone TU-90 combines with zone TU-95A and the pipe tunnel leading to the Screenhouse due to leakage under the doors.

Zone TU-90 is a relatively small room such that any water from a pipe break is expected to spray all the equipment in zone TU-90.

For a pipe break in zone TU-90, equipment in the zone zones TU-90, TU-95A, and the pipe tunnel leading to the Screenhouse can be vulnerable and could be at risk.

TU-92

Description - Flood Zone TU-92 is Diesel Generator Room 1B on the 586'-0" elevation. The zone is bounded on the north by zones TU-90 and the pipe tunnel leading to the Screenhouse, on the south by an exterior wall, on the east by an exterior wall and the pipe tunnel leading to the Screenhouse, and on the west by zones TU-94 and TU-22-1. The Administrative Building lies above and exterior soil lies below.

All penetrations in zone TU-92 are sealed. The north wall has a normally-closed access door (1) leading to a service water piping tunnel that leads to the Screenhouse and the west wall has a normally-closed access door (3) leading to zone TU-94.

The major PRA equipment in zone TU-92 includes Diesel Generator 1B, 4 kV Switchgear Bus 6, and MCC 62A. The Internal Flood Walkdown Form [Appendix C of FLOOD01] for zone TU-92 contains a complete listing of the flood-susceptible PRA equipment in this zone.

Potential flood sources in this zone include service water piping and fire protection piping which represent both a flooding hazard and a spray hazard.

Flood mitigation is present in this zone in the form of floor drains. (A six-inch curb that ran east and west just north of all the equipment protected the equipment from water originating from outside the room until late 2004, however it has since been removed.)

Analysis – Water from a pipe break in TU-92 will not easily propagate elsewhere since all the penetrations are sealed. Floor drains will divert water to the Turbine Building sump. Thus, only significant pipe breaks are a concern for accumulation in zone TU-92. Equipment damage from spray sources in TU-92 is limited to the equipment residing in that zone.

Zone TU-92 is equipped with two normally closed doors that initially prevent lateral propagation of water from pipe breaks beyond the capacity of the floor drains. Door 1 (double door with a 1/64" gap) opens outwardly to the pipe tunnel leading to the Screenhouse and door 3 (double door with a 1/64" gap) opens inwardly from zone TU-94. Initially water would flow through the floor drains to the Turbine Building sump, but given the limited capacity of the floor drains in TU-

92, the Turbine Building sump will not reach a level high enough to initiate a control room alarm. While water is flowing through the floor drains it will also be leaking under door 1 to the pipe tunnel that leads to the Screenhouse. Once the seiche hump is overcome in the pipe tunnel, water leaking under the door will flow to the Screenhouse sump. When the water level inside TU-92 reaches a critical height, both doors are expected to fail allowing water to freely propagate to zone TU-94 and the pipe tunnel leading to the Screenhouse.

The Turbine Building Sump contains two pumps with design capacities of < 100 gpm each. The level switch for the Turbine Building sump pump control is a mechanically alternating device. A high water level (30") starts one pump. A return to low level (12") stops the pump. A subsequent high level starts the alternate pump. [SYSTEM01]

If a high-high water level (34.5") is reached, the level switch starts the second pump. Both pumps continue to run until an intermediate level cutoff point, 19", is reached. At this point, the level switch turns off the leading (first) pump. The lagging (second) pump continues to run until the low-level setpoint, 12", is reached. [SYSTEM01]

The Turbine Building sump contains Level Switch LA-16666 that actuates Control Room Alarm 47033P when a high-high-high water level setpoint, 34.5", is reached. [SYSTEM01]

The Screenhouse sump contains two pumps with design capacities of < 100 gpm each. The level switch for the Screenhouse sump pump control is a mechanically alternating device. A high water level (30") starts one pump. A return to low level (12") stops the pump. A subsequent high level starts the alternate pump.

If a high-high water level is reached, the level switch starts the second pump. Both pumps continue to run until an intermediate level cutoff point, 19", is reached. At this point, the level switch turns off the leading (first) pump. The lagging (second) pump continues to run until the low-level setpoint, 12", is reached.

The Screenhouse sump contains Level Switch LA-16669 that actuates Control Room Alarm 47033P when a high-high-high water level setpoint, 34.5", is reached [SYSTEM01].

The operator's first indication of a pipe break inside TU-92 will be the high Screenhouse sump level alarm in the control room once the water level inside TU-92 rises high enough to fail door 1 which will allow water to flow freely to the Screenhouse. The only other possible indication of a pipe break would be equipment failure that forces an operator to investigate locally.

Summary – Pipe breaks in zone TU-92 can result in both equipment spray and submergence. For spray events TU-92 becomes a flood area by itself since only equipment in TU-92 is susceptible to damage from direct spray originating in zone TU-92. For submergence events, zone TU-92 combines with zone TU-94 and the pipe tunnel leading to the Screenhouse due to leakage under the doors.

Zone TU-92 is a relatively small room such that any water from a pipe break is expected to spray all the equipment in zone TU-92.

For a pipe break in zone TU-92, equipment in the zone zones TU-92, TU-94, and the pipe tunnel leading to the Screenhouse can be vulnerable and could be at risk.

TU-94

Description - Flood Zone TU-94 is the CO2 Storage Tank Room 1B on the 586'-0" elevation. The zone is bounded on the north by zone TU-95A, on the south by zone TU-22-1, on the east by zones TU-90 and TU-92, and on the west by zone TU-22-1. Zone TU-22-1 lies above and exterior soil lies below.

All penetrations in zone TU-94 are sealed. The north wall has a normally-closed access door (5) leading to zone TU-95A, the south wall has a normally-closed access door (401) leading to zone TU-22-1, and the east wall has a normally-closed access door (3) leading to zone TU-92.

The major PRA equipment in zone TU-94 includes Station and Instrument Air Compressor 1A. The Internal Flood Walkdown Form for zone TU-94 contains a complete listing of the flood-susceptible PRA equipment in this zone.

Potential flood sources in this zone include service water piping and fire protection piping which represent both a flooding hazard and a spray hazard.

Flood mitigation is present in this zone in the form of a floor drain in a trench that is sealed at the boundary of zone TU-90.

Analysis – Water from a pipe break in TU-94 will not easily propagate elsewhere since all the penetrations are sealed. Floor drains will divert water to the Turbine Building sump. Thus, only significant pipe breaks are a concern for accumulation in zone TU-94. Equipment damage from spray sources in TU-94 is limited to the equipment residing in that zone.

Zone TU-94 is equipped with three normally closed doors that initially prevent lateral propagation of water from pipe breaks beyond the capacity of the floor drains. Door 3 (double door with a 1/64" gap) opens outwardly to zone TU-92, door 5 (double door with 1/64" gap) opens inwardly from zone TU-95A, and door 401 (double door with 7/8" gap) opens outwardly to zone TU-22-1. Initially water would simply leak under doors 3 and 5 to flood areas TU-92 and TU-95A, respectively. Water will also flow to the Turbine Building sump via the floor drains. When the water level inside TU-94 reaches a critical height, doors 3 and 401 are expected to fail allowing water to freely propagate to zones TU-92 and TU-22-1, respectively.

The first indication of such a break would be a Turbine Building sump high level alarm in the control room if the flow via the floor drain is sufficiently high to fill the sump. The procedure for abnormal operation of the miscellaneous drains and sumps instructs the operator to dispatch someone to investigate the source of the alarm, regardless of which sump fills first. The only other possible indication of a pipe break would be equipment failure that forces an operator to investigate locally.

Summary – Pipe breaks in zone TU-94 can result in both equipment spray and submergence. For spray events TU-94 becomes a flood area by itself since only equipment in TU-94 is susceptible to damage from direct spray originating in zone TU-94. For submergence events, zone TU-94 combines with zones TU-22-1, TU-95A, and TU-92 due to leakage under the associated doors.

Zone TU-94 is a relatively small room such that any water from a pipe break is expected to spray all the equipment in zone TU-94.

For a pipe break in zone TU-94, equipment in the zone zones TU-94, TU-22-1 and TU-95A can be vulnerable and could be at risk.

TU-95A

Description - Flood Zone TU-95A is the 480 V Switchgear Bus 1-51 and 1-52 Room on the 586'-0" elevation. The zone is bounded on the north by an exterior wall and the Technical Support Center, on the south by zones TU-22-1 and TU-94, on the east by zone TU-90, and on the west by zone TU-95B-1. Zone TU-22-1 lies above and exterior soil lies below.

All penetrations in zone TU-95A are sealed. The south wall has normally-closed access doors (5, 263 and 268) leading to zones TU-94 and TU-95B-1 and the east wall has a normally closed door (136) leading to zone TU-90.

The major PRA equipment in zone TU-95A includes Station and Instrument Air Compressor 1C, and 480 V Switchgear Buses 51 and 52. The Internal Flood Walkdown Form [Appendix C of FLOOD01] for zone TU-95A contains a complete listing of the flood-susceptible PRA equipment in this zone.

Potential flood sources in this zone include service water piping and fire protection piping which represent both a flooding hazard and a spray hazard.

Flood mitigation is present in this zone in the form of a trench that communicates with zone TU-90 and contains a floor drain leading to the Turbine Building sump.

Analysis – Water from a pipe break in TU-95A will easily propagate to zone TU-90 via an open 4-inch pipe under door 136. Floor drains will divert water to the Turbine Building sump. Thus, only significant pipe breaks are a concern for accumulation in zone TU-95A. Equipment damage from spray sources in TU-95A is limited to the equipment residing in that zone.

Zone TU-95A is equipped with three normally closed doors that initially prevent lateral propagation of water from pipe breaks beyond the capacity of the floor drains. Door 5 (double door with 1/64" gap) opens outwardly to zone TU-94, door 136 (double door with 1/8" gap) opens outwardly to zone TU-90, door 263 (double door with 3/16" gap) opens outwardly to zone TU-95B-1, and door 268 (single door) opens outwardly to zone TU-95B-1. Additionally, a firewall constructed of gypsum board separates TU-95A and TU-95B-1. Initially water would simply leak under doors to the various adjoining zones and flow to the Turbine Building sump via the floor drains. When the water level inside TU-95A reaches a critical height, the firewall is expected to fail structurally allowing water to freely propagate to zones TU-95B-1.

The first indication of such a break would likely be from investigation of failed equipment since free flow to either the Screenhouse sump or the Turbine Building sump does not occur until water level accumulates to several feet and doors and gypsum wall begin to fail.

Summary – Pipe breaks in zone TU-95A can result in both equipment spray and submergence. For spray events TU-95A becomes a flood area by itself since only equipment in TU-95A is susceptible to damage from direct spray originating in zone TU-95A. For submergence events, zone TU-95A combines with zones TU-90, TU-94, and TU-95B-1 due to leakage under the associated doors and an open pipe that allows communication between TU-95A and TU-90.

Zone TU-95A is a relatively small room such that any water from a pipe break is expected to spray all the equipment in zone TU-95A.

For a pipe break in zone TU-95A, equipment in zones TU-94, TU-95B-1, TU-90, and TU-95A can be vulnerable and could be at risk.

TU-95B-1

Description - Flood Zone TU-95B-1 consists of the 480 V Switchgear Bus 61 and 62 Room and the Auxiliary Feedwater Pump 1B Room on the 586'-0" elevation. These two rooms are connected via an open trench such that any water in one room will travel freely to the other, thus they are combined to form a single flood area for submergence issues. The zone is bounded on the north by the Technical Support Center, on the south by zone TU-22-1, on the east by zones TU-95A, TU-95B-2, and TU-95C, and on the west by zone AX-23B-1. Zone TU-22-1 lies above and exterior soil lies below.

All penetrations in zone TU-95B-1 are sealed. The south wall has normally-closed access doors

(4 and 6) leading to zone TU-22-1, the north wall has normally-closed access doors (268, 263, 262, and 261) leading to zones TU-95A and TU-95C, the west wall has a normally-closed access door (244) leading to zone TU-95B-2 and a normally-closed access door (8) leading to the Auxiliary Building, and the east wall has a normally-closed door (243) leading to zone TU-95B-2.

The major PRA equipment in zone TU-95B-1 includes Station and Instrument Air Compressor 1B, Motor Driven Auxiliary Feedwater Pump B, and 480 V Switchgear Buses 1-61 and 1-62. The Internal Flood Walkdown Form [Appendix C of FLOOD01] for zone TU-95B-1 contains a complete listing of the flood-susceptible PRA equipment in this zone.

Potential flood sources in this zone include service water piping, CST piping, main steam piping, and fire protection piping which represent both a flooding hazard and a spray hazard.

Flood mitigation is present in this zone in the form of a trench that is sealed at the boundary of zone TU-95A and zone TU-95B-1. The trench contains a floor drain leading to the Turbine Building sump.

Analysis – Water from a pipe break in TU-95B-1 will not easily propagate elsewhere since all the penetrations are sealed. Floor drains will divert water to the Turbine Building sump. Thus, only significant pipe breaks are a concern for accumulation in zone TU-95B-1. Equipment damage from spray sources in TU-95B-1 is limited to the equipment residing in that zone unless it is a prolonged spray. A prolonged spray (greater than 90 minutes) in the western half of the area would probably degrade the gypsum board that comprises area TU-95C to the point that the auxiliary feedwater components housed inside TU-95C would be damaged.

Zone TU-95B-1 is equipped with seven normally closed doors that initially prevent lateral propagation of water from pipe breaks beyond the capacity of the floor drains. Door 4 (double door with a 1/8" gap) opens outwardly to zone TU-22-1, door 6 (double door with 1/4" gap) opens outwardly to zone TU-22-1, door 243 (single door with 1/32" gap) opens outwardly to TU-95B-2, door 244 (single door with 1/32" gap) opens outwardly to TU-95B-2, door 261 (single door with 3/16" gap) opens inwardly from TU-95C, door 262 (double door with 3/16" gap) opens inwardly from TU-95C, door 263 (double door with 3/16" gap) opens inwardly from zone TU-95A, and door 268 (single door) opens inwardly from zone TU-95A. Initially water would simply leak under doors to flood areas TU-95A, TU-95B-2, and TU-95C as well as flow to the Turbine Building sump via the floor drains. When the water level inside TU-95B-1 reaches a critical height, doors and gypsum walls are expected to fail allowing water to freely propagate to adjoining areas. As doors fail water will propagate to TU-22-1 where it will fill the Turbine Building sump.

The Turbine Building Sump contains two pumps with design capacities of < 100 gpm each. The level switch for the Turbine Building sump pump control is a mechanically alternating device. A high water level (30") starts one pump. A return to low level (12") stops the pump. A subsequent high level starts the alternate pump. [SYSTEM01]

If a high-high water level (34.5") is reached, the level switch starts the second pump. Both pumps continue to run until an intermediate level cutoff point, 19", is reached. At this point, the level switch turns off the leading (first) pump. The lagging (second) pump continues to run until the low-level setpoint, 12", is reached. [SYSTEM01]

The Turbine Building sump contains Level Switch LA-16666 that actuates Control Room Alarm 47033P when a high-high-high water level setpoint, 34.5", is reached. [SYSTEM01]

The first indication of such a break would be a Turbine Building sump high level alarm in the control room. The procedure for abnormal operation of the miscellaneous drains and sumps instructs the operator to dispatch someone to investigate the source of the alarm. The only other possible indication of a pipe break would be equipment failure that forces an operator to investigate locally.

Summary – Pipe breaks in zone TU-95B-1 can result in both equipment spray and submergence. For spray events TU-95B-1 becomes a flood area by itself since only equipment in TU-95B-1 is susceptible to damage from direct spray originating in zone TU-95B-1. For submergence events, zone TU-95B-1 combines with zones TU-22-1, TU-95B-2, TU-95C, and TU-95A due to leakage under the associated doors.

Zone TU-95B-1 is separated into two distinct sections by zone TU-95B-2. Each of these sections is a relatively small area such that any water from a pipe break is expected to spray all the equipment in that area of zone TU-95B-1.

For a pipe break in zone TU-95B-1, equipment in zones TU-95B-1, TU-22-1, TU-95B-2, TU-95C, and TU-95A can be vulnerable and could be at risk.

TU-95B-2

Description - Flood Zone TU-95B-2 is the Turbine Driven Auxiliary Feedwater Pump Room on the 586'-0" elevation. The zone is bounded on the north by an exterior wall and the Technical Support Center, on the south by zone TU-22-1, on the east by zones TU-95B-1 and TU-95C, and on the west by zone TU-95B-1. Zone TU-95B-2 makes use of a false ceiling for HELB purposes and Zone TU-95B-1 actually lies above. Exterior soil lies below.

All penetrations in zone TU-95B-2 are sealed. The east wall has a normally closed access door (244) leading to zone TU-95B-1 and the west wall has a normally closed door (243) leading to zone TU-95B-1. The south wall has a normally closed blowout panel that opens to zone TU-22-1.

The major PRA equipment in zone TU-95B-2 includes the Turbine Driven Auxiliary Feedwater Pump. The Internal Flood Walkdown Form [Appendix C of FLOOD01] for zone TU-95B-2 contains a complete listing of the flood-susceptible PRA equipment in this zone.

Potential flood sources in this zone include service water piping, CST piping, and main steam piping which represent both a flooding hazard and a spray hazard.

Flood mitigation is present in this zone in the form of a covered trench that communicates with zone TU-95B-1. A floor drain approximately 4 inches above the ground also communicates with this trench.

Analysis – Water from a pipe break in TU-95B-2 will not easily propagate elsewhere since all the penetrations are sealed. Floor drains will divert water to the Turbine Building sump. Thus, only significant pipe breaks are a concern for accumulation in zone TU-95B-2. Equipment damage from spray sources in TU-95B-2 is limited to the equipment residing in that zone.

Zone TU-95B-2 is equipped with two normally closed doors that initially prevent lateral propagation of water from pipe breaks beyond the capacity of the floor drains. Door 243 (single door with 1/32" gap) opens inwardly from TU-95B-1 and door 244 (single door with 1/32" gap) opens inwardly from TU-95B-1. Initially water would simply leak under doors to the various adjoining zones and flow to the Turbine Building sump via the floor drains. When the water level inside TU-95B-2 reaches a critical height, one of two things will occur. Either the blowout panel will fail allowing water to propagate to TU-22-1 and subsequently to the Turbine Building sump or both doors will fail allowing water to freely propagate to zone TU-95B-1. When the water level inside TU-95B-1 reaches a critical height, doors are expected to fail allowing water to freely propagate to TU-22-1 where it will fill the Turbine Building sump. In either case water will reach the Turbine Building sump.

The Turbine Building Sump contains two pumps with design capacities of < 100 gpm each. The level switch for the Turbine Building sump pump control is a mechanically alternating device. A high water level (30") starts one pump. A return to low level (12") stops the pump. A subsequent high level starts the alternate pump. [SYSTEM01]

If a high-high water level (34.5") is reached, the level switch starts the second pump. Both pumps continue to run until an intermediate level cutoff point, 19", is reached. At this point, the level switch turns off the leading (first) pump. The lagging (second) pump continues to run until the low-level setpoint, 12", is reached. [SYSTEM01]

The Turbine Building sump contains Level Switch LA-16666 that actuates Control Room Alarm 47033P when a high-high-high water level setpoint, 34.5", is reached. [SYSTEM01]

The first indication of such a break would be a Turbine Building sump high level alarm in the control room. The procedure for abnormal operation of the miscellaneous drains and sumps instructs the operator to dispatch someone to investigate the source of the alarm. The only other possible indication of a pipe break would be equipment failure that forces an operator to investigate locally.

Summary – Pipe breaks in zone TU-95B-2 can result in both equipment spray and submergence. For spray events TU-95B-2 becomes a flood area by itself since only equipment in TU-95B-2 is susceptible to damage from direct spray originating in zone TU-95B-2. For submergence events, zone TU-95B-2 combines with zone TU-95B-1 due to door leakage.

Zone TU-95B-2 is a relatively small room such that any water from a pipe break is expected to spray all the equipment in zone TU-95B-2.

For a pipe break in zone TU-95B-2, equipment in zones TU-95B-1 and TU-95B-2 can be vulnerable and could be at risk.

TU-95C

Description - Flood Zone TU-95C is the Motor Driven Auxiliary Feedwater Pump 1A Room on the 586'-0" elevation. The zone is bounded on the north by the Technical Support Center, on the south by zone TU-95B-1, on the east by zone TU-95B-2, and on the west by zone TU-95B-1. Zone TU-22-1 lies above and exterior soil lies below.

All penetrations in zone TU-95C are sealed. The south wall has normally-closed access doors (261 and 262) leading to zone TU-95B-1. The south and west walls are constructed of simple drywall and are expected to initially survive a spray event, but prolonged exposure to water will result in failure of the walls.

The major PRA equipment in zone TU-95C includes Motor Driven Auxiliary Feedwater Pump 1A. The Internal Flood Walkdown Form [Appendix C of FLOOD01] for zone TU-95C contains a complete listing of the flood-susceptible PRA equipment in this zone.

Potential flood sources in this zone include service water piping, CST piping, and main steam piping which represent both a flooding hazard and a spray hazard.

Flood mitigation is present in this zone in the form of a floor drain approximately 4 inches above the ground that communicates with the trench in zone TU-95B-1.

Analysis – Water from a pipe break in TU-95C will not initially propagate elsewhere since all the penetrations are sealed. Floor drains will divert water to the Turbine Building sump. Thus, only significant pipe breaks are a concern for accumulation in zone TU-95C. Equipment damage from spray sources in TU-95C is initially limited to the equipment residing in that zone.

However, a sustained pipe break could eventually spray equipment in the western half of TU-95B-1 since the west and south walls of TU-95C are constructed of gypsum that is not expected to survive a sustained spray of water.

Zone TU-95C is equipped with two normally closed doors that initially prevent lateral propagation of water from pipe breaks beyond the capacity of the floor drains. Door 261 (single door with 3/16" gap) opens outwardly to TU-95B-1 and door 262 (double door with 3/16" gap) opens outwardly to TU-95B-1. Initially water would simply leak under doors to the various adjoining zones and flow to the Turbine Building sump via the floor drains. However, since the west and south walls of TU-95C are constructed of drywall, any sustained exposure to water is expected to result in failure of walls and open communication with TU-95B-1. Regardless of the failure mechanism, water will propagate to TU-95B-1.

When the water level inside TU-95B-1 reaches a critical height, doors are expected to fail allowing water to freely propagate to TU-22-1 where it will fill the Turbine Building sump.

The Turbine Building Sump contains two pumps with design capacities of < 100 gpm each. The level switch for the Turbine Building sump pump control is a mechanically alternating device. A high water level (30") starts one pump. A return to low level (12") stops the pump. A subsequent high level starts the alternate pump. [SYSTEM01]

If a high-high water level (34.5") is reached, the level switch starts the second pump. Both pumps continue to run until an intermediate level cutoff point, 19", is reached. At this point, the level switch turns off the leading (first) pump. The lagging (second) pump continues to run until the low-level setpoint, 12", is reached. [SYSTEM01]

The Turbine Building sump contains Level Switch LA-16666 that actuates Control Room Alarm 47033P when a high-high-high water level setpoint, 34.5", is reached. [SYSTEM01]

The first indication of such a break would be a Turbine Building sump high level alarm in the control room. The procedure for abnormal operation of the miscellaneous drains and sumps instructs the operator to dispatch someone to investigate the source of the alarm. The only other possible indication of a pipe break would be equipment failure that forces an operator to investigate locally.

Summary – Pipe breaks in zone TU-95C can result in both equipment spray and submergence. For spray events TU-95C combines with TU-95B-1 to become a flood area since the drywall construction of the TU-95C walls cannot withstand sustained exposure to water spray. For submergence events, zone TU-95C combines with TU-95B-1 to become a flood area due to door leakage and eventual door failure or gypsum wall failure.

Zone TU-95C is a relatively small room such that any water from a pipe break is expected to spray all the equipment in zones TU-95C and TU-95B-1.

For a pipe break in zone TU-95C, equipment in zones TU-95C and TU-95B-1 can be vulnerable and could be at risk.

4.0 REFERENCES

[DWG01] Drawings

a. A-203 Rev. AW, General Arrangement Turbine and Administration Building Basement Floor

b. A-204 Rev. BC, General Arrangement Reactor and Auxiliary Building Basement Floor

c. A-205 Rev. AM, General Arrangement Turbine and Administration Building Mezzanine Floor

d. A-206 Rev. BS, General Arrangement Reactor and Auxiliary Building Mezzanine Floor

e. A-207 Rev. U, General Arrangement Turbine and Administration Building Operating Floor

f. A-208 Rev. BL, General Arrangement Reactor and Auxiliary Building

g. A-209 Rev. Y, General Arrangement Reactor and Auxiliary Building Miscellaneous Floor Plans

h. A-212 Rev. Y, General Arrangement Miscellaneous Plans and Sections

i. A-213 Rev. Y, General Arrangement Screenhouse and Circulating Water Discharge

[CALC01] “Evaluation of Various Doors and Walls for HELB Flooding – Turbine Building Basement,” Revision 1, Scientel Wireless, LLC.

[CALC02] Attachment “KNPP Design Input Document.pdf” to email from Ling Yu Song (MPR) to Dale Franson (NMC) and Paul Miller (NMC), April 8, 2005.

[FLOOD01] Kewaunee Nuclear Plant Internal Flooding Analysis – Qualitative Screening Assessment and Flood Frequency Development, SCIENTECH, LLC.

[GUIDE01] Dominion Probabilistic Risk Assessment Manual, Part II, Chapter G, Section 2, *Flood Area Definition*, Revision 0.

[GUIDE02] Dominion Probabilistic Risk Assessment Manual, Part II, Chapter G, Section 4, *Accident Sequence Analysis*, Revision 0.

[NOTEBOOK01] KPS Internal Flooding Accident Sequence Analysis Notebook, Rev. 0

[PROC01] KNPP Operating Procedure A-MDS-30, Rev. P, “Miscellaneous Drains and Sumps (MDS) Abnormal Operation”

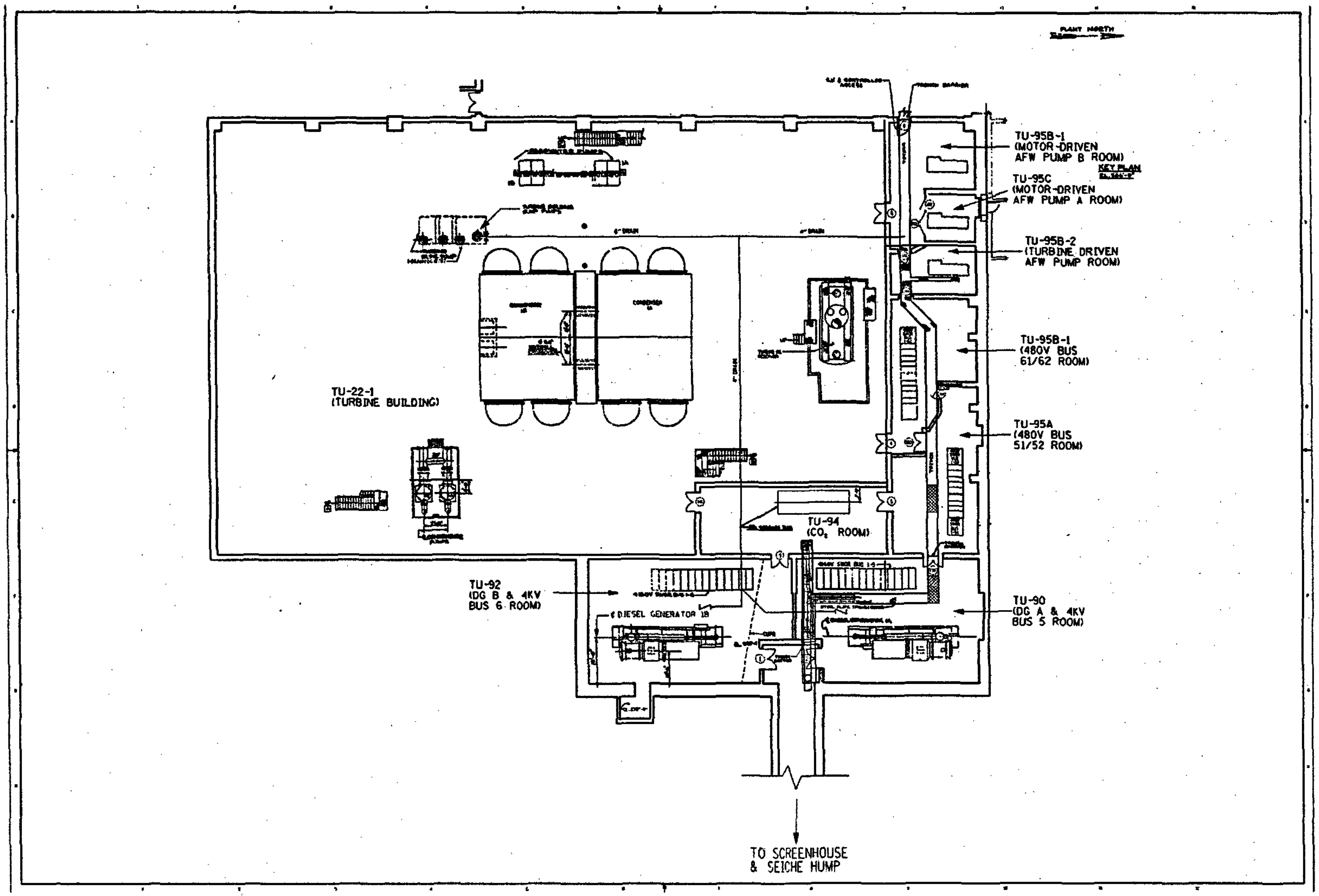
[STANDARD01] ASME RA-S-2002, “Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,” 2002

[SYSTEM01] KPS System Description 30, Rev. 2, “Miscellaneous Drains and Sumps (MDS)”

Table 1 – Flood Area Descriptions

Flood Zone	Room Number	Room Description
TU-22-1	6B	Turbine Building - (Condenser) Basement Floor
	120	Turbine Building - Mezzanine Floor
	121	Turbine Building - Mezzanine Floor
	122	Turbine Building - Mezzanine Floor
	123	Turbine Building - Mezzanine Floor
	124	Turbine Building - Mezzanine Floor
	125	Turbine Building - Mezzanine Floor
	126	Turbine Building - Mezzanine Floor
	127	Turbine Building - Mezzanine Floor
	128	Turbine Building - Mezzanine Floor
	220A	Turbine Building - Operating Floor
	10B	Elevator B Machine Room
	11B	Corridor and Ramps
	17B	Waste Tank Area
	144	Welding Shop
	147	Corridor
	149	Main Shop and Corridor (147)
	150	Working Material Storage Area
	154	Shop Office
	155	Electric Shop
	234	Cation, Brine, and Mixed Beds - Water Treatment Area
	234A	SG Boric Acid Area
TU-90	2B	Diesel Generator A Room
	25B	Diesel Generator A Fuel Oil Day Tank Room
TU-92	3B	Diesel Generator B Room
	24B	Diesel Generator B Fuel Oil Day Tank Room
TU-94	4B	CO_2 Storage Room
TU-95A	5B	480V Swgr Bus 1-51 and 1-52 Room
TU-95B-1	5B-1	480V Swgr Bus 1-61 and 1-62 Room
	5B-3	Aux FW Pump B Room
TU-95B-2	5B-4	Turbine Driven Aux FW Pump Room
TU-95C	5B-2	Aux FW Pump A Room

Figure 1 – Turbine Building Basement/Safeguards Alley Arrangement



Appendix C

Fault Tree Analysis

Fault Tree Analysis

Owner's Acceptance: Thomas G Hook THOMAS G HOOK
Signature Print Name

12/01/05
Date

Kewaunee Power Station

Fault Tree Analysis for Turbine Building Floods

Effective Date: November 2005

R. SHARPE VIA EMAIL — 12/01/05
Prepared By: J. R. Sharpe — Date

D. JONES VIA EMAIL — 12/01/05
Reviewed By: D. Jones — Date

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INTERNAL FLOODING – Fault Tree Analysis for Turbine Building Floods

1.0 PURPOSE

The purpose of this notebook is to document the WinNUPRA model that was developed to analyze flooding scenarios originating from pipe breaks in the Turbine Building before February 2005.

The following information is identified, correlated, and developed as part of this analysis:

- Fault trees developed to support event tree analysis
- Basic event data used to support the flooding model
- Human error probabilities (HEPs) used to support the flooding model

2.0 MODEL SCOPE

This notebook documents the models that were developed for evaluating internal flooding sequences due to pipe breaks in the Turbine Building before February 2005.

3.0 UNIT DIFFERENCES

Kewaunee Power Station is a single unit site so there are no unit differences.

4.0 RISK MONITOR CONSIDERATIONS

The risk monitor used at KPS is the Safety Monitor. The Safety Monitor was not modified to reflect this analysis.

5.0 MODEL DEVELOPMENT

5.1 FAULT TREES

The existing system fault trees for the KPS internal events PRA [NB01] comprise the majority of the Turbine Building Flood model. Two new fault trees were developed to support this analysis; AFM.LGC and FLOODING.LGC are described below in Sections 5.1.1 and 5.1.2. Fault tree AFM.LGC contains the logic associated with Auxiliary Feedwater (AFW) failures and fault tree FLOODING.LGC was developed to accommodate new initiating events and new human actions specifically related to Turbine Building flooding. Of the existing fault trees from the internal events PRA, only those for DC power were modified, as described in Section 5.1.3.

The human error probabilities (HEPs) used in the analysis are documented in Attachment 1. The bases for the HEPs from a review of procedures (e.g., cues) and training materials is provided in Attachment 2. A summary of a simulator exercise performed to determine timing for operator actions in the feedwater line break scenario with actuation of all the fire sprinklers in the turbine building is provided in Attachment 3.

5.1.1 Fault Tree AFM

Fault Tree AFM is presented as Figure 1. This fault tree contains the logic associated with failure of the Turbine Driven AFW pump and Motor Driven AFW Pump B (MDAFP B) to deliver flow to the steam generators. The logic in AFM is simply copied from Fault Tree AFW in the Internal Events PRA [NB01] and rearranged for use in this flooding analysis. No new analysis was performed in the development of fault tree AFM. Top Event AFS (as defined in the Accident Sequence Analysis, Appendix D) uses gate GAFM302 to model the failure of MDAFP B to start. Top Event AFR uses gate GAFM700 to model the failure of MDAFP B to run and provide flow to Steam Generator B. Top Event AFT uses gate GAFM1002 to model the failure of the TDAFP to start and run.

5.1.2 Fault Tree FLOODING

Fault Tree FLOODING is presented as Figure 2. This fault tree contains the logic used to model the initiating events used for Turbine Building floods and the HEPs associated with the isolation of pipe breaks and the operation of mitigating equipment. In some cases the hardware failure basic events are also included.

5.1.3 DC Power Fault Tree Modifications

The DC power fault trees were modified to include basic event 16-BATCLG--F-HE, which represents operator action to establish battery room cooling. This event applies to flooding scenarios where the 480 V buses have failed, thereby causing failure of normal battery room cooling. After the Battery Room A/B Exhaust Flow Low annunciator activates in the control room, the operator is directed to use the fire equipment to ventilate the Battery Rooms. The air trunks and fans are then rigged to supply battery room cooling.

Figure 3 shows the placement of new event 16-BATCLG--F-HE in fault tree BRA104, at grid location "2-3". The same event is similarly placed in the following DC power fault trees:

BRA104B
BRA104T
BRA105
BRA105T
BRA113
BRA127
BRB104
BRB104B
BRB104T
BRB105
BRB105T
BRB114
BRB127
BRC103
BRC103T
BRD103
BRD103T
BRD115

5.2 HUMAN ERROR PROBABILITIES

Human error probabilities (HEPs) were developed using the same methodology used in the existing PRA [NB02]. This section briefly describes each HEP developed as part of the analysis of Turbine Building floods. The detailed analyses of these HEPs are documented as attachments to this report. Table 1 lists all of the new human actions and their values that were developed in support of the flooding analysis.

5.2.1 04-CW-TRIP-F-HE – Detection and Isolation of a 58,000-gpm Circulating Water Break before Failing Both 480 V Buses

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Large Circulating Water break in the Turbine Building. This basic event represents the failure of the operator to trip the Circulating Water pumps in time to prevent the eventual failure of the 480 V buses.

A large rupture of an inlet condenser expansion joint in the Turbine Building (TU-22-1) could propagate through the open drain lines and under doors to Safeguards Alley (TU-90, TU-92, TU-95A, TU-95B-1, TU-95B-2 and TU-95C). Areas TU-95A and TU-95B-1 contain the train A and B 480 VAC buses which could be failed due to propagation of a break in TU-22-1.

Indication of this type of break would be provided by a reactor trip due to low condenser vacuum and a Miscellaneous Sump Level High alarm in the control room.

Propagation to Safeguards Alley will begin when the Turbine Building sump begins to fill since the open drain lines from Safeguards Alley directly communicate with this sump. Additionally, when water begins to accumulate on the floor water will begin to leak under doors 4, 6, and 401 into Safeguards Alley. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 3 minutes to prevent eventual loss of the 480 VAC buses in Safeguards Alley.

Thus, 3 minutes would be available to trip the Circulating Water pumps following an expansion joint rupture to prevent the eventual failure of the 480 V buses. Based on simulator observations and operator interviews at least 9 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 04-CW-TRIP-F-HE and the human error probability (HEP) is 1.0 since sufficient time does not exist to perform the isolation.

5.2.2 04-CWSTP13-F-HE – Detection and Isolation of a 14,000-gpm Circulating Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Moderate Circulating Water break in the Turbine Building. This basic event represents the failure of the operator to trip the Circulating Water pumps in time to prevent the eventual isolation of the 4 kVAC Bus 5 motor loads due to the automatic tripping of the associated circuit breakers.

A rupture of an outlet condenser expansion joint in the Turbine Building (TU-22-1) could propagate through the open drain lines and under doors to Safeguards Alley (TU-90, TU-92, TU-95A, TU-95B-1, TU-95B-2 and TU-95C). Area TU-90 contains kVAC Bus 5 which could be failed due to propagation of a break in TU-22-1.

Indication of this type of break would be provided by a Miscellaneous Sump Level High alarm in the control room.

Propagation to Safeguards Alley will begin when the Turbine Building sump begins to fill since the open drain lines from Safeguards Alley directly communicate with this sump. Additionally, when water begins to accumulate on the floor water will begin to leak under doors 4, 6, and 401 into Safeguards Alley. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 13 minutes to prevent eventual isolation of 4 kV Bus 5 motor loads.

Thus, 13 minutes would be available to trip the Circulating Water pumps following an outlet expansion joint rupture to prevent the eventual isolation of 4 kV Bus 5 motor loads due to the automatic tripping of the associated circuit breakers. Based on simulator observations and operator interviews at least 9 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 04-CWSTP13-F-HE and the human error probability (HEP) is 2.6E-01.

5.2.3 04-CWSTP19-F-HE – Detection and Isolation of a 14,000-gpm Circulating Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Moderate Circulating Water break in the Turbine Building. This basic event represents the failure of the operator to trip the Circulating Water pumps in time to prevent the eventual failure of the Turbine Driven AFW pump auxiliary lube oil pump.

This event is identical to the one described in section 5.2.2 except that the failure of interest is the Turbine Driven AFW pump auxiliary lube oil pump at 9 inches of water. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 19 minutes to prevent eventual loss of the ability to start the TDAFP.

Based on simulator observations and operator interviews at least 9 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 04-CWSTP19-F-HE and the human error probability (HEP) is 1.2E-01.

5.2.4 04-CWSTP22-F-HE – Detection and Isolation of a 14,000-gpm Circulating Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Moderate Circulating Water break in the Turbine Building. This basic event represents the failure of the operator to trip the Circulating Water pumps in time to prevent submergence failure of 480 VAC Buses 61 and 62 and the eventual isolation of the 4 kV Bus 6 motor loads due to the automatic tripping of the associated circuit breakers. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 22 minutes to accomplish these objectives.

Based on simulator observations and operator interviews at least 9 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 04-CWSTP22-F-HE and the human error probability (HEP) is 1.2E-01.

5.2.5 04-CWSTP25-F-HE – Detection and Isolation of a 14,000-gpm Circulating Water Break before Failure of the Motor Driven AFW Pumps and a Water Level of 18 Inches in the Turbine Building

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Moderate Circulating Water break in the Turbine Building. This basic event represents the failure of the operator to trip the Circulating Water pumps in time to prevent the submergence failure of the motor driven AFW pumps and prevent the water level from reaching 18 inches in the Turbine Building. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 25 minutes to accomplish these objectives.

Based on simulator observations and operator interviews at least 9 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 04-CWSTP25-F-HE and the human error probability (HEP) is 1.2E-01.

5.2.6 02-SW4A-B29F-HE – Detection and Isolation of a Service Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Large Service Water break in the Turbine Building. This basic event represents the failure of the operator to close MOVs SW-4A and SW-4B in time to prevent the eventual isolation of the 4 kVAC Bus 5 motor loads due to the automatic tripping of the associated circuit breakers.
A large Service Water pipe break in the Turbine Building (TU-22-1) could propagate through the

open drain lines and under doors to Safeguards Alley (TU-90, TU-92, TU-95A, TU-95B-1, TU-95B-2 and TU-95C). Area TU-90 contains kVAC Bus 5 which could be failed due to propagation of a break in TU-22-1.

Indication of this type of break would be provided by a Miscellaneous Sump Level High alarm in the control room.

Propagation to Safeguards Alley will begin when the Turbine Building sump begins to fill since the open drain lines from Safeguards Alley directly communicate with this sump. Additionally, when water begins to accumulate on the floor water will begin to leak under doors 4, 6, and 401 into Safeguards Alley. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 29 minutes to prevent eventual isolation of 4 kV Bus 5 motor loads.

Thus, 29 minutes would be available to close MOV SW-4A or SW-4B (only one will be open normally) following a Service Water pipe break to prevent eventual isolation of 4 kV Bus 5 motor loads due to the automatic tripping of the associated circuit breakers.. Based on simulator observations and operator interviews about 13 minutes are required to diagnose the cause of the high sump level alarm, decide the course of action, and execute the isolation. The basic event ID for this HEP is 02-SW4A-B29F-HE and the human error probability (HEP) is 2.0E-02.

5.2.7 02-SW4A-B45F-HE – Detection and Isolation of a Service Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Large Service Water break in the Turbine Building. This basic event represents the failure of the operator to close MOVs SW-4A and SW-4B in time to prevent the eventual failure of the Turbine Driven AFW pump auxiliary lube oil pump.

This event is identical to the one described in Section 5.2.6 except that the failure of interest is the Turbine Driven AFW pump auxiliary lube oil pump at 9 inches of water. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 45 minutes to prevent eventual loss of the ability to start the TDAFP.

Based on simulator observations and operator interviews about 13 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 02-SW4A-B45F-HE and the human error probability (HEP) is 2.0E-02.

5.2.8 02-SW4A-B51F-HE – Detection and Isolation of a Large Service Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Large Service Water break in the Turbine Building. This basic event represents the failure of the operator to close MOVs SW-4A and SW-4B in time to prevent submergence failure of 480 VAC Buses 61 and 62 and the eventual isolation of the 4 kV Bus 6 motor loads due to the automatic tripping of the associated circuit breakers. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 51 minutes to accomplish these objectives.

Based on simulator observations and operator interviews at least 13 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 02-SW4A-B51F-HE and the human error probability (HEP) is 2.0E-02.

5.2.9 02-SW4A-B60F-HE – Detection and Isolation of a Large Service Water Break before Failure of the Motor Driven AFW Pumps

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Large Service Water break in the Turbine Building. This basic event represents the failure of the operator to close MOVs SW-4A and SW-4B in time to prevent the eventual submergence failure or the MDAFPs at 13 inches.

This event is identical to the one described in Section 5.2.6 except that the result of interest is submergence of the motor driven AFW pumps. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 60 minutes to prevent the submergence failure of the motor driven AFW pumps.

Based on simulator observations and operator interviews at least 13 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 02-SW4A-B60F-HE and the human error probability (HEP) is 2.0E-02.

5.2.10 02-SW4A-B66F-HE – Detection and Isolation of a Large Service Water Break before Water Level Reaches 18 Inches in the Turbine Building

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Large Service Water break in the Turbine Building. This basic event represents the failure of the operator to close MOVs SW-4A and SW-4B in time to prevent the water level from reaching 18 inches in the Turbine Building.

This event is identical to the one described in Section 5.2.6 except that the result of interest is 18 inches of water in the Turbine Building. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 66 minutes to prevent 18 inches of water in the Turbine Building.

Based on simulator observations and operator interviews at least 13 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 02-SW4A-B66F-HE and the human error probability (HEP) is 2.0E-02.

5.2.11 08-FPISO29-F-HE – Detection and Isolation of a Fire Protection Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Large Fire Protection Water break in the Turbine Building. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump or by securing the power to the pumps in time to prevent the eventual isolation of the 4 kVAC Bus 5 motor loads due to the automatic tripping of the associated circuit breakers.

A large Fire Protection Water pipe break in the Turbine Building (TU-22-1) could propagate through the open drain lines and under doors to Safeguards Alley (TU-90, TU-92, TU-95A, TU-95B-1, TU-95B-2 and TU-95C). Area TU-90 contains kVAC Bus 5 which could be failed due to propagation of a break in TU-22-1.

Indication of this type of break would be provided by the Fire Pump Abnormal alarm in the control room and a Miscellaneous Sump Level High alarm in the control room.

Propagation to Safeguards Alley will begin when the Turbine Building sump begins to fill since the open drain lines from Safeguards Alley directly communicate with this sump. Additionally, when water begins to accumulate on the floor water will begin to leak under doors 4, 6, and 401 into Safeguards Alley. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 29 minutes to prevent eventual isolation of the 4 kV Bus 5 motor loads.

Thus, 29 minutes would be available to close the Fire pump discharge manual valves or isolate power to the Fire pumps following a Fire Protection Water pipe break to prevent eventual isolation of the 4 kV Bus 5 motor loads due to the automatic tripping of the associated circuit breakers. Based on simulator observations and operator interviews about 32 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 08-FPISO29-F-HE and the human error probability (HEP) is 1.0.

5.2.12 08-FPISO45-F-HE – Detection and Isolation of a Fire Protection Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Large Fire Protection Water break in the Turbine Building. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump or by securing the power to the pumps in time to prevent the eventual failure of the Turbine Driven AFW pump auxiliary lube oil pump.

This event is identical to the one described in Section 5.2.11 except that the failure of interest is the Turbine Driven AFW pump auxiliary lube oil pump at 9 inches of water. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 45 minutes to prevent eventual loss of the ability to start the TDAFP.

Based on simulator observations and operator interviews about 32 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 08-FPISO45-F-HE and the human error probability (HEP) is 6.6E-02.

5.2.13 08-FPISO56-F-HE – Detection and Isolation of a Fire Protection Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Large Fire Protection Water break in the Turbine Building. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump or by securing the power to the pumps in time to prevent submergence failure of 480 VAC Buses 61 and 62 and the eventual isolation of the 4 kV Bus 6 motor loads due to the automatic tripping of the associated circuit breakers. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 56 minutes to accomplish these objectives.

Based on simulator observations and operator interviews nearly 32 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 08-FPISO56-F-HE and the human error probability (HEP) is 2.4E-02.

5.2.14 08-FPISO68-F-HE – Detection and Isolation of a Fire Protection Water Break before Failure of the Motor Driven AFW Pumps

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Fire Protection Water break in the Turbine Building. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump or by securing the power to the pumps in time to prevent the submergence failure of the motor driven AFW pumps.

This event is identical to the one described in Section 5.2.11 except that the result of interest is submergence of the motor driven AFW pumps. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 68 minutes to prevent the submergence failure of the motor driven AFW pumps.

Based on simulator observations and operator interviews at least 32 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 08-FPISO68-F-HE and the human error probability (HEP) is 1.6E-02.

5.2.15 08-ISO-FS18F-HE – Detection and Isolation of a Large Flood due to a Feedwater Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Feedwater break resulting in a large Fire Protection System discharge in the Turbine Building. A Feedwater line break in the Turbine Building will spill the contents of the hotwell onto the Turbine Building floor and result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent the eventual isolation of the 4 kVAC Bus 5 motor loads due to the automatic tripping of the associated circuit breakers. A Feedwater pipe break in the Turbine Building (TU-22-1) would set off multiple fire sprinklers in addition to pumping the hotwell inventory into the Turbine Building. This water could propagate through the open drain lines and under doors to Safeguards Alley (TU-90, TU-92, TU-95A, TU-95B-1, TU-95B-2 and TU-95C). Area TU-90 contains kVAC Bus 5 which could be failed due to propagation of a break in TU-22-1. This event analyzes a Feedwater pipe break resulting in a 6000-gpm discharge of the Fire Protection system.

Indication of this type of break would be provided by the Fire Pump Abnormal alarm in the control room and a Miscellaneous Sump Level High alarm in the control room.

Propagation to Safeguards Alley will begin when the Turbine Building sump begins to fill since the open drain lines from Safeguards Alley directly communicate with this sump. Additionally, when water begins to accumulate on the floor water will begin to leak under doors 4, 6, and 401 into Safeguards Alley. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 18 minutes to prevent eventual isolation of the 4 kV Bus 5 motor loads.

Thus, 18 minutes would be available to isolate the sprinklers following a Feedwater pipe break to prevent eventual isolation of the 4 kV Bus 5 motor loads due to the automatic tripping of the associated circuit breakers. In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. The basic event ID for this HEP is 08-ISO-FS18F-HE and the human error probability (HEP) is 1.0 since sufficient time does not exist to isolate flow from the Fire pumps.

5.2.16 08-ISO-FS33F-HE – Detection and Isolation of a Large Flood due to a Feedwater Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Feedwater break resulting in a large Fire Protection System discharge in the Turbine Building. A Feedwater line break in the Turbine Building will spill the contents of the hotwell onto the Turbine Building floor and result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent the eventual failure of the Turbine Driven AFW pump auxiliary lube oil pump.

This event is identical to the one described in Section 5.2.15 except that the failure of interest is the Turbine Driven AFW pump auxiliary lube oil pump at 9 inches of water. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 33 minutes to prevent eventual failure of the ability to start the TDAFP.

Thus, 33 minutes would be available to isolate the sprinkler flow following a Feedwater pipe break to prevent the eventual failure of the Turbine Driven AFW pump auxiliary lube oil pump. In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. The basic event ID for this HEP is 08-ISO-FS33F-HE and the human error probability (HEP) is 4.4E-01.

5.2.17 08-ISO-FS40F-HE – Detection and Isolation of a Large Flood due to a Feedwater Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Feedwater break resulting in a large Fire Protection System discharge in the Turbine Building. A Feedwater line break in the Turbine Building will spill the contents of the hotwell onto the Turbine Building floor and result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent submergence failure of 480 VAC Buses 61 and 62 and the eventual isolation of the 4 kV Bus 6 motor loads due to the automatic tripping of the associated circuit breakers. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 40 minutes to accomplish these objectives.

Thus, 40 minutes would be available to isolate the sprinkler flow following a Feedwater pipe break to prevent eventual failure of 480VAC Buses 61 and 62, and 4 kVAC Bus 6. In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. The basic event ID for this HEP is 08-ISO-FS40F-HE and the human error probability (HEP) is 1.3E-01.

5.2.18 08-ISO-FS54F-HE – Detection and Isolation of a Large Flood due to a Feedwater Break before Failure of the Motor Driven AFW Pumps

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Feedwater break resulting in a large Fire Protection System discharge in the Turbine Building. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent the eventual submergence failure of the motor driven AFW pumps.

This event is identical to the one described in Section 5.2.15 except that the result of interest is submergence of the motor driven AFW pumps. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 54 minutes to prevent the submergence failure of the motor driven AFW pumps.

In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. The basic event ID for this HEP is 08-ISO-FS54F-HE and the human error probability (HEP) is 3.0E-02.

5.2.19 08-ISO-FS55F-HE – Detection and Isolation of a Medium Flood due to a Feedwater Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Feedwater break resulting in a moderate Fire Protection System discharge in the Turbine Building. A Feedwater line break in the Turbine Building will spill the contents of the hotwell onto the Turbine Building floor and result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent the eventual isolation of the 4 kVAC Bus 5 motor loads due to the automatic tripping of the associated circuit breakers.

A Feedwater pipe break in the Turbine Building (TU-22-1) would set off multiple fire sprinklers in addition to pumping the hotwell inventory into the Turbine Building. This water could propagate through the open drain lines and under doors to Safeguards Alley (TU-90, TU-92, TU-95A, TU-95B-1, TU-95B-2 and TU-95C). Area TU-90 contains kVAC Bus 5 which could be failed due to propagation of a break in TU-22-1. This event analyzes a Feedwater pipe break resulting in a 2000-gpm discharge of the Fire Protection system.

Indication of this type of break would be provided by the Fire Pump Abnormal alarm in the control room and a Miscellaneous Sump Level High alarm in the control room.

Propagation to Safeguards Alley will begin when the Turbine Building sump begins to fill since the open drain lines from Safeguards Alley directly communicate with this sump. Additionally, when water begins to accumulate on the floor water will begin to leak under doors 4, 6, and 401 into Safeguards Alley. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 55 minutes to prevent eventual isolation of the 4 kV Bus 5 motor loads.

Thus, 55 minutes would be available to isolate the sprinkler flow following a Feedwater pipe break to prevent eventual isolation of the 4 kV Bus 5 motor loads due to the automatic tripping of the associated circuit breakers. In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. The basic event ID for this HEP is 08-ISO-FS55F-HE and the human error probability (HEP) is 3.0E-02.

5.2.20 08-ISO-FS97F-HE – Detection and Isolation of a Medium Flood due to a Feedwater Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Feedwater break resulting in a moderate Fire Protection System discharge in the Turbine Building. A Feedwater line break in the Turbine Building will spill the contents of the hotwell onto the Turbine Building floor and result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent the eventual failure of the Turbine Driven AFW pump auxiliary lube oil pump.

This event is identical to the one described in Section 5.2.19 except that the failure of interest is the Turbine Driven AFW pump auxiliary lube oil pump at 9 inches of water. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 97 minutes to prevent eventual failure of the ability to start the TDAFP.

In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. . The basic event ID for this HEP is 08-ISO-FS97F-HE and the human error probability (HEP) is 3.0E-02.

5.2.21 08-ISO-FS2HF-HE – Detection and Isolation of a Medium Flood due to a Feedwater Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6

Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Feedwater break resulting in a moderate Fire Protection System discharge in the Turbine Building. A Feedwater line break in the Turbine Building will spill the contents of the hotwell onto the Turbine Building floor and result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent submergence failure of 480 VAC Buses 61 and 62 and the eventual isolation of the 4 kV Bus 6 motor loads due to the automatic tripping of the associated circuit breakers. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 2 hours to accomplish these objectives.

In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. . The basic event ID for this HEP is 08-ISO-FS2HF-HE and the human error probability (HEP) is 1.7E-02.

5.2.22 08-FPSISO29F-HE – Detection and Isolation of a Large Flood due to a Steamline Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Steamline break resulting in a large Fire Protection System discharge in the Turbine Building. A Steamline break in the Turbine Building will result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent the eventual isolation of the 4 kVAC Bus 5 motor loads due to the automatic tripping of the associated circuit breakers.

A Steamline pipe break in the Turbine Building (TU-22-1) would set off multiple fire sprinklers in the Turbine Building. This water could propagate through the open drain lines and under doors to Safeguards Alley (TU-90, TU-92, TU-95A, TU-95B-1, TU-95B-2 and TU-95C). Area TU-90 contains kVAC Bus 5 which could be failed due to propagation of a break in TU-22-1. This event analyzes a Steamline break resulting in a 6000-gpm discharge of the Fire Protection system.

Indication of this type of break would be provided by the Fire Pump Abnormal alarm in the control room and a Miscellaneous Sump Level High alarm in the control room.

Propagation to Safeguards Alley will begin when the Turbine Building sump begins to fill since the open drain lines from Safeguards Alley directly communicate with this sump. Additionally, when water begins to accumulate on the floor water will begin to leak under doors 4, 6, and 401 into Safeguards Alley. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 29 minutes to prevent eventual isolation of 4 kV Bus 5 motor loads.

Thus, 32 minutes would be available to isolate the sprinkler flow following a Steamline pipe break to prevent eventual isolation of 4 kV Bus 5 motor loads due to the automatic tripping of the associated circuit breakers. In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. . The basic event ID for this HEP is 08-FPSISO29F-HE and the human error probability (HEP) is 1.0.

5.2.23 08-FPSISO45F-HE – Detection and Isolation of a Large Flood due to a Steamline Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Steamline break resulting in a large Fire Protection System discharge in the Turbine Building. A Steamline break in the Turbine Building will result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent the eventual failure of the Turbine Driven AFW pump auxiliary lube oil pump.

This event is identical to the one described in Section 5.2.22 except that the failure of interest is the Turbine Driven AFW pump auxiliary lube oil pump at 9 inches of water. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 45 minutes to prevent eventual failure of the ability to start the TDAFP.

In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. The basic event ID for this HEP is 08-FPSISO45F-HE and the human error probability (HEP) is 6.6E-02.

5.2.24 08-FPSISO56F-HE – Detection and Isolation of a Large Flood due to a Steamline Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Steamline break resulting in a large Fire Protection System discharge in the Turbine Building. A Steamline break in the Turbine Building will result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent submergence failure of 480 VAC Buses 61 and 62 and the eventual isolation of the 4 kV Bus 6 motor loads due to the automatic tripping of the associated circuit breakers. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 56 minutes to accomplish these objectives.

In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews, the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. The basic event ID for this HEP is 08-FPSISO56F-HE and the human error probability (HEP) is 3.0E-02.

5.2.25 08-FPSISO68-F-HE – Detection and Isolation of a Large Flood due to a Steamline Break before Failure of the Motor Driven AFW Pumps

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Steamline break resulting in a large Fire Protection System discharge in the Turbine Building. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent the submergence failure of the motor driven AFW pumps.

This event is identical to the one described in Section 5.2.22 except that the result of interest is submergence of the motor driven AFW pumps. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 68 minutes to prevent the submergence failure of the motor driven AFW pumps.

In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews, the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing.. The basic event ID for this HEP is 08-FPSISO68-F-HE and the human error probability (HEP) is 3.0E-02.

5.2.26 08-FPSISO1CF-HE – Detection and Isolation of a Medium Flood due to a Steamline Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Steamline break resulting in a moderate Fire Protection System discharge in the Turbine Building. A Steamline break in the Turbine Building will result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent the eventual isolation of the 4 kVAC Bus 5 motor loads due to the automatic tripping of the associated circuit breakers.

A Steamline pipe break in the Turbine Building (TU-22-1) would set off multiple fire sprinklers in the Turbine Building. This water could propagate through the open drain lines and under doors to Safeguards Alley (TU-90, TU-92, TU-95A, TU-95B-1, TU-95B-2 and TU-95C). Area TU-90 contains kVAC Bus 5 which could be failed due to propagation of a break in TU-22-1. This event analyzes a Steamline break resulting in a 2000-gpm discharge of the Fire Protection system.

Indication of this type of break would be provided by the Fire Pump Abnormal alarm in the control room and a Miscellaneous Sump Level High alarm in the control room.

Propagation to Safeguards Alley will begin when the Turbine Building sump begins to fill since the open drain lines from Safeguards Alley directly communicate with this sump. Additionally, when water begins to accumulate on the floor water will begin to leak under doors 4, 6, and 401 into Safeguards Alley. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 100 minutes to prevent eventual isolation of the 4 kV Bus 5 motor loads.

Thus, 100 minutes would be available to isolate the sprinkler flow following a Steamline pipe break to prevent eventual isolation of the 4 kV Bus 5 motor loads due to the automatic tripping of the associated circuit breakers. In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews, the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. The basic event ID for this HEP is 08-FPSISO1CF-HE and the human error probability (HEP) is 3.0E-02.

5.2.27 08-FPSISO2CF-HE – Detection and Isolation of a Medium Flood due to a Steamline Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Steamline break resulting in a moderate Fire Protection System discharge in the Turbine Building. A Steamline break in the Turbine Building will result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent the eventual failure of the Turbine Driven AFW pump auxiliary lube oil pump.

This event is identical to the one described in Section 5.2.26 except that the failure of interest is the Turbine Driven AFW pump auxiliary lube oil pump at 9 inches of water. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 150 minutes to prevent eventual failure of the ability to start the TDAFP.

In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews, the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. The basic event ID for this HEP is 08-FPSISO2CF-HE and the human error probability (HEP) is 3.0E-02.

5.2.28 08-FPSISO3CF-HE – Detection and Isolation of a Medium Flood due to a Steamline Break

before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Steamline break resulting in a moderate Fire Protection System discharge in the Turbine Building. A Steamline break in the Turbine Building will result in an elevated building temperature that actuates multiple fire sprinklers. This basic event represents the failure of the operator to isolate flow from the Fire Pumps either by closing the manual discharge isolation valve on each pump, securing the power to the pumps, or closing the manual isolation valves for the sprinklers in time to prevent the eventual submergence failure of 480 VAC Buses 61 and 62 and the eventual isolation of the 4 kV Bus 6 motor loads due to the automatic tripping of the associated circuit breakers. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 170 minutes to accomplish these objectives.

In order to isolate the sprinklers the operators must receive the initial signal, decide the course of action, and execute isolation of the sprinklers, which first requires closing manual valves located on the mezzanine of the turbine building to isolate the four main sprinkler headers, and later close the supply valves to isolate the basement deluge systems. Based on simulator observations and operator interviews, the four main sprinkler headers would be isolated in 32 minutes. As described in Appendix D, GOTHIC analyses [CALC01] show that isolating flow from the basement deluge systems can be delayed for an additional 60 minutes without changing the overall accident sequence timing. The basic event ID for this HEP is 08-FPSISO3CF-HE and the human error probability (HEP) is 3.0E-02.

5.2.29 04-CWSTP29-F-HE – Detection and Isolation of a Small Circulating Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Small Circulating Water break in the Turbine Building. This basic event represents the failure of the operator to trip the Circulating Water pumps in time to prevent the eventual isolation of the 4 kVAC Bus 5 motor loads due to the automatic tripping of the associated circuit breakers.

A small rupture of an inlet or outlet condenser expansion joint in the Turbine Building (TU-22-1) could propagate through the open drain lines and under doors to Safeguards Alley (TU-90, TU-92, TU-95A, TU-95B-1, TU-95B-2 and TU-95C). Area TU-90 contains kVAC Bus 5 which could be failed due to propagation of a break in TU-22-1.

Indication of this type of break would be provided by a Miscellaneous Sump Level High alarm in the control room.

Propagation to Safeguards Alley will begin when the Turbine Building sump begins to fill since the open drain lines from Safeguards Alley directly communicate with this sump. Additionally, when water begins to accumulate on the floor water will begin to leak under doors 4, 6, and 401 into Safeguards Alley. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 29 minutes to prevent eventual isolation of the 4 kV Bus 5 motor loads.

Thus, 29 minutes would be available to trip the Circulating Water pumps following a Small Circulating Water break to prevent eventual isolation of 4 kV Bus 5 motor loads due to the automatic tripping of the associated circuit breakers.. Based on simulator observations and operator interviews about 10 minutes are required to diagnose the cause of the high sump level alarm, decide the course of action, and execute the isolation. The basic event ID for this HEP is 04-CWSTP29-F-HE and the human error probability (HEP) is 4.3E-02.

5.2.30 04-CWSTP45-F-HE – Detection and Isolation of a Small Circulating Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Small Circulating Water break in the Turbine Building. This basic event represents the failure of the operator to trip the Circulating Water pumps in time to prevent the eventual failure of the Turbine Driven AFW pump auxiliary lube oil pump.

This event is identical to the one described in Section 5.2.29 except that the failure of interest is the Turbine Driven AFW pump auxiliary lube oil pump at 9 inches of water. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 45 minutes to prevent eventual loss of the ability to start the TDAFP.

Based on simulator observations and operator interviews about 10 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 04-CWSTP45-F-HE and the human error probability (HEP) is 1.7E-02.

5.2.31 04-CWSTP51-F-HE – Detection and Isolation of a Small Circulating Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Small Circulating Water break in the Turbine Building. This basic event represents the failure of the operator to trip the Circulating Water pumps in time to prevent submergence failure of 480 VAC Buses 61 and 62 and the eventual isolation of the 4 kV Bus 6 motor loads due to the automatic tripping of the associated circuit breakers. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 51 minutes to accomplish these objectives.

Based on simulator observations and operator interviews at least 10 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 04-CWSTP51-F-HE and the human error probability (HEP) is 1.4E-02.

5.2.32 04-CWSTP60-F-HE – Detection and Isolation of a Small Circulating Water Break before Failure of the Motor Driven AFW Pumps

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Small Circulating Water break in the Turbine Building. This basic event represents the failure of the operator to trip the Circulating Water pumps in time to prevent submergence failure of the MDAFPs at 13 inches.

This event is identical to the one described in Section 5.2.29 except that the result of interest is submergence of the motor driven AFW pumps. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 60 minutes to prevent the submergence failure of the motor driven AFW pumps.

Based on simulator observations and operator interviews at least 10 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 04-CWSTP60-F-HE and the human error probability (HEP) is 1.2E-02.

5.2.33 04-CWSTP66-F-HE – Detection and Isolation of a Small Circulating Water Break before Water Level Reaches 18 Inches in the Turbine Building

The analysis of this HEP is documented in Attachment 1. This basic event applies only to a Small Circulating Water break in the Turbine Building. This basic event represents the failure of the operator to trip the Circulating Water pumps in time to prevent the water level from reaching 18 inches in the Turbine Building.

This event is identical to the one described in Section 5.2.29 except that the result of interest is 18 inches of water in the Turbine Building. Based on GOTHIC analysis [CALC01] the operator must isolate the break within 66 minutes to prevent 18 inches of water in the Turbine Building.

Based on simulator observations and operator interviews at least 10 minutes is required to receive the initial signal, decide the course of action, and execute the isolation. The basic event ID for this HEP is 04-CWSTP66-F-HE and the human error probability (HEP) is 1.2E-02.

5.2.34 16-BATCLG--F-HE – Establish Battery Room Cooling

The analysis of this HEP is documented in Attachment 1. This basic event applies to flooding scenarios where the 480 V buses have failed, thereby causing failure of normal battery room cooling. After the Battery Room A/B Exhaust Flow Low annunciator activates in the control room, the operator is directed to use the fire equipment to ventilate the Battery Rooms. The air trunks and fans are then rigged to supply battery room cooling.

The operator must execute the action within 180 minutes to prevent excessive Battery Room heatup. Based on simulator observations and operator interviews about 77 minutes is required to receive the initial signal, decide the course of action, and execute the action. The basic event ID for this HEP is 16-BATCLG--F-HE and the human error probability (HEP) is 7.9E-02.

5.2.35 27A-ORR----F-HE – Failure to Throttle SI Flow to Conserve RWST Inventory

The analysis of this HEP is documented in Attachment 1. This basic event applies to flooding scenarios where secondary cooldown has failed and the remaining SI pump is available. If secondary cooldown fails, the flow rate through existing RCP Seal LOCA is expected to worsen. The operator would attempt to replace the lost RCS inventory using the available SI pump. Since high-pressure recirculation is unavailable due to the failure of the CCW pump power supplies, the operator must conserve the RWST inventory. This is done by manually throttling the SI pump discharge flow.

The operator must execute the action within 67 minutes to extend the time the RWST is available. Based on simulator observations and operator interviews about 58 minutes is required to receive the initial signal, decide the course of action, and execute the action. The basic event ID for this HEP is 27A-ORR----F-HE and the human error probability (HEP) is 5.0E-03.

5.2.36 05B-BYALOP-F-HE – Failure to Bypass AFW Auxiliary Lube Oil Pressure Interlock

The analysis of this HEP is documented in Attachment 1. This basic event applies to flooding scenarios where the water level in the AFW pump area has risen to 9 inches and the operator needs to start an AFW pump. If the auxiliary lube oil pump is failed due to submergence, then the associated AFW pump will not start due to a lube oil pressure interlock. This basic event addresses the bypass of this interlock to allow starting of the AFW pump.

The operator must execute the action within approximately 4 hours to restart an AFW pump. Based on simulator observations and operator interviews about 3.5 hours is required to receive the initial signal, decide the course of action, and execute the action. The basic event ID for this HEP is 05B-BYALOP-F-HE and the human error probability (HEP) is 4.4E-01.

5.2.37 06-NOINDAFWF-HE – Failure to Feed Steam Generator Without Level Indication

The analysis of this HEP is documented in Attachment 1. This basic event applies to flooding scenarios where power to the instrument bus is failed and AFW operation is required to maintain steam generator level. The operator must then provide makeup to the steam generators without instrument power to allow monitoring of steam generator level and prevent overfilling the steam generator and failing the TDAFP.

The basic event ID for this HEP is 06-NOINDAFWF-HE and the human error probability (HEP) is 6.4E-01.

5.2.38 06--OC2----F-HE – Failure to Perform RCS Cooldown Using Natural Circulation

The analysis of this HEP is documented in Attachment 1. This basic event applies to flooding scenarios where RXCP seal cooling systems, i.e., charging and CCW, are not failed by the flooding event, but fail randomly shortly into the event. For this event, the operators must cooldown and depressurize the RCS per ES-0.2.

The operator must execute the action within approximately 6 hours to perform cooldown. Based on simulator observations and operator interviews about 3 hours is required to receive the initial signal, decide the course of action, and execute the action. The basic event ID for this HEP is 06--OC2----F-HE and the human error probability (HEP) is 7.4E-02.

5.2.39 06--OC6----F-HE – Failure to Perform RCS Cooldown with Boration

The analysis of this HEP is documented in Attachment 1. This basic event applies to flooding scenarios where 480 VAC power is lost to all charging and CCW pumps. In these scenarios, a RXCP seal LOCA is assumed to occur and the operators would cooldown and depressurize the RCS per ES-1.2.

The operator must execute the action within approximately 200 minutes to perform cooldown. Based on simulator observations and operator interviews about 65 minutes is required to receive the initial signal, decide the course of action, and execute the action. The basic event ID for this HEP is 06--OC6----F-HE and the human error probability (HEP) is 9.2E-02.

5.2.40 05B-MDPTD36F-HE – Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (36 Minutes)

The analysis of this HEP is documented in Attachment 1. This basic event applies to the Moderate Circulating Water pipe break. After the Motor Driven AFW pumps have failed, the operator must start the Turbine Driven AFW pump within 36 minutes of the initial pipe break to avoid submergence of the auxiliary lube oil pump and subsequent failure of the Turbine Driven AFW pump to start due to a lube oil pressure interlock. [CALC01]

Based on simulator observations and operator interviews about 18 minutes is required to receive the initial signal, decide the course of action, and execute the action. The basic event ID for this

HEP is 05B-MDPTD36F-HE and the human error probability (HEP) is 4.7E-01.

5.2.41 05B-MDPTD49F-HE – Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (49 Minutes)

The analysis of this HEP is documented in Attachment 1. This basic event applies to the Feedwater pipe break with a large Fire Protection sprinkler discharge to the turbine building. After the Motor Driven AFW pumps have failed, the operator must start the Turbine Driven AFW pump within 49 minutes of the initial pipe break to avoid submergence of the auxiliary lube oil pump and subsequent failure of the Turbine Driven AFW pump to start due to a lube oil pressure interlock. [CALC01]

Based on simulator observations and operator interviews about 18 minutes is required to receive the initial signal, decide the course of action, and execute the action. The basic event ID for this HEP is 05B-MDPTD49F-HE and the human error probability (HEP) is 4.7E-01.

5.2.42 05B-MDPTD61F-HE – Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (61 Minutes)

The analysis of this HEP is documented in Attachment 1. This basic event applies to the Large Service Water, Fire Protection Water, and Steamline pipe breaks with a large Fire Protection sprinkler discharge. After the Motor Driven AFW pumps have failed, the operator must start the Turbine Driven AFW pump within 61 minutes of the initial pipe break to avoid submergence of the auxiliary lube oil pump and subsequent failure of the Turbine Driven AFW pump to start due to a lube oil pressure interlock. [CALC01]

Based on simulator observations and operator interviews about 18 minutes is required to receive the initial signal, decide the course of action, and execute the action. The basic event ID for this HEP is 05B-MDPTD61F-HE and the human error probability (HEP) is 3.1E-01.

5.2.43 05B-MDPTD1CF-HE – Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (109 Minutes)

The analysis of this HEP is documented in Attachment 1. This basic event applies to the Feedwater pipe break with a moderate Fire Protection sprinkler discharge to the turbine building. After the Motor Driven AFW pumps have failed, the operator must start the Turbine Driven AFW pump within 109 minutes of the initial pipe break to avoid submergence of the auxiliary lube oil pump and subsequent failure of the Turbine Driven AFW pump to start due to a lube oil pressure interlock. [CALC01]

Based on simulator observations and operator interviews about 18 minutes is required to receive the initial signal, decide the course of action, and execute the action. The basic event ID for this HEP is 05B-MDPTD1CF-HE and the human error probability (HEP) is 1.6E-01.

5.2.44 05B-MDPTD2HF-HE – Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (2.5 Hours)

The analysis of this HEP is documented in Attachment 1. This basic event applies to the Steamline pipe break with a moderate Fire Protection sprinkler discharge to the turbine building. After the Motor Driven AFW pumps have failed, the operator must start the Turbine Driven AFW pump within 2.5 hours of the initial pipe break to avoid submergence of the auxiliary lube oil pump and subsequent failure of the Turbine Driven AFW pump to start due to a lube oil pressure interlock. [CALC01]

Based on simulator observations and operator interviews about 18 minutes is required to receive the initial signal, decide the course of action, and execute the action. The basic event ID for this HEP is 05B-MDPTD2HF-HE and the human error probability (HEP) is 4.1E-02.

5.2.45 86-INSTRRCRF-HE – Failure to Recover AFW Control

The analysis of this HEP is documented in Attachment 1. This basic event applies to all scenarios where AFW flow exists and at least one train of safety-related AC power is available. Under these conditions the operator is instructed to control AFW flow using the AFW pump discharge valves to adjust the flow rate to the steam generators. If these valves cannot be controlled remotely from the control room, then the operator has approximately 11 hours to operate them manually from the pump room. [CALC01] .

Based on simulator observations and operator interviews about 9.5 hours is required to receive the initial signal, decide the course of action, and execute the action. The basic event ID for this HEP is 86-INSTRRCRF-HE and the human error probability (HEP) is 1.8E-02.

5.3 DATA

The KNPP.BED database was used for the flooding analysis. The Turbine Building flooding initiators and the HEPs discussed in Section 5.2 were added to KNPP.BED, along with the basic events modeled in fault tree FLOODING.LGC. One other new basic event was also added to the database:

> Basic Event 05B-FRACTDP-OFF represents the fraction of time that the operator is expected to trip the Turbine Driven AFW pump early in a flooding event given that both Motor Driven AFW pumps have successfully started.

No other basic events were added.

Table 1 lists all of the new operator actions and their HEPs that were developed in support of the flooding analysis.

Table 2 lists all the new basic events (and their values) that were added to KNPP.BED in support of the flooding analysis.

6.0 MODEL EVALUATION (EQUATIONS)

Each fault tree used to represent an event tree top event in this analysis is quantified various times under different initial conditions. Each of these fault tree quantifications produces an equation. The equation's location on the event tree (i.e., which previous top events have succeeded and failed) dictates the initial conditions used to quantify the fault tree and develop that unique equation. Such initial conditions are modeled by setting to TRUE the failure rates of equipment that is known to be unavailable due to the flooding event. The same fault tree is then quantified with different initial conditions to yield different equations.

With the exception of Top Events AFZ and AFX (which are described in more detail in the following subsection), this analysis generally develops two unique equations for each top event. The initial conditions for the first equation consist of the flood-induced failures of equipment in the Turbine Building as well as the bottom row of breakers on Buses 51/52 and 61/62. This represents the flood-induced equipment failures that occur very early in the event. For example, when analyzing the Large Feedwater scenario (WI06B) the equation for Top Event AFR using these initial conditions is named AFRWI06B .

The initial conditions for the second equation simply build on those of the first equation. In addition to the equipment failures of the first equation, the second equation adds the flood-induced failures of 480 VAC Buses 61 and 62. Thus, the second equation is quantified assuming the failure of 480 VAC Train B safety-related power. For example, when analyzing the Large Feedwater scenario (WI06B) the equation for Top Event AFR using these initial conditions is named AFRWI064.

6.1 EVALUATION OF TOP EVENT AFZ

Instead of quantifying the same fault tree various times to develop different equations, the equations associated with Top Event AFZ use multiple fault trees that are quantified a single time. This is due to the complexity added by various human actions and the potential of equipment to already be running. Only the equations associated with a Large Feedwater Break are described here (e.g., AFZ-AWIB). The descriptions of the equations associated with all other initiators (e.g., AFZ-ACXB, AFZ-ASIB, AFZ-ATIB, etc.) are identical except for the name of the initiator and the timing associated with the model.

6.1.1 AFZ-AWIB

This equation is used to model failure of the operator to provide AFW flow for decay heat removal using the turbine-driven AFW pump. The equation is used to model operator action-related failures only. Hardware-related failures of the turbine-driven AFW train are evaluated by other equations in the event tree.

This equation is used for sequences with the following conditions:

- The water volume released to the turbine building will result in opening the bottom row of breakers on 480 VAC Buses 51, 52, 61, and 62.
- The break was isolated before the volume of water released in the turbine building would cause water level in the AFW pump rooms to submerge the TDAFP auxiliary lube oil pump.
- The break was isolated before the volume of water released in the turbine building would cause water level in the B-train 4kVAC room to reach a level that would fail 4 kVAC Bus 6 motor loads or 480 VAC Buses 61/62.
- The water volume released to the turbine building will result in failure of 4kVAC Bus 5 motor loads.
- The B-train motor-driven AFW pump successfully started.
- The B-train motor-driven AFW pump failed to run. By definition of the sequences where equation AFZ-AW1B is used, the mission time for the AFW pump is 24 hours. On average, the pump is assumed to run halfway through the mission time or 12 hours.

For the sequences where equation AFZ-AW1B is used, several potential success paths exist. First, the operators may have recognized that the flooding event could threaten the motor-driven AFW pumps and would maintain the turbine-driven AFW pump running throughout the event. Second, if the turbine-driven AFW pump were secured, then restart would merely require that the operators take the control switch from pull-to-lock. Then, even if the operators did not start the pump, it would automatically start on a low-low steam generator level signal.

6.1.2 AFZ-BWIB

This equation is used to model failure of the operator to provide AFW flow for decay heat removal using the turbine-driven AFW pump. The equation is used to model operator action-related failures only. Hardware-related failures of the turbine-driven AFW train are evaluated by other equations in the event tree.

This equation is used for sequences with the following conditions:

- The water volume released to the turbine building will result in opening the bottom row of breakers on 480 VAC Buses 51, 52, 61, and 62.
- The water volume released to the turbine building will result in failure of 4kVAC Bus 5 motor loads.
- The water volume released to the turbine building would cause water level in the AFW pump rooms to submerge the TDAFP auxiliary lube oil pump.
- The break was isolated before the volume of water released in the turbine building would cause water level in the B-train 4kVAC room to reach a level that would fail 4 kVAC Bus 6 motor loads or 480 VAC Buses 61/62.
- The B-train motor-driven AFW pump successfully started.
- The B-train motor-driven AFW pump failed to run. By definition of the sequences where equation AFZ-BWIB is used, the mission time for the AFW pump is 24 hours. On average, the pump is assumed to run halfway through the mission time or 12 hours.

For the sequences where equation AFZ-BWIB is used, multiple potential success paths exist. First, the operators could have maintained the turbine-driven AFW pump running throughout the event. The pump would be maintained running if either motor-driven AFW pump failed or if the operators recognized that the flooding event could threaten the motor-driven AFW pumps and would want the added reliability of the third AFW pump. Second, even if the turbine-driven AFW pump was secured early in the event, then the operators could recognize that the rising water levels would soon threaten the motor-driven AFW pumps and may restart the turbine-driven AFW pump. By definition of the sequences where equation AFZ-BWIB is used, water level will reach a level that will submerge the turbine-driven AFW pump auxiliary lube oil pump. Therefore, for the operators to successfully start the pump from the control room, action must be taken before 49 minutes. Otherwise, the auxiliary lube oil pump would be submerged, thereby preventing the turbine-driven AFW pump from starting.

If the turbine-driven pump is not started within 49 minutes after flood initiation and then maintained running, then the pump could be started if the low oil pressure interlock is bypassed. Bypass of the low oil pressure interlock may be directed by personnel manning the technical support center and would need to be completed before water level in either of the steam generators dropped to less than 5-percent wide range, the point that bleed and feed cooling would be initiated.

Given the definition of the sequences where equation AFZ-BWIB is used, the B-train motor-driven AFW pump started, but failed to run. Since, on average, the pump is assumed to fail one-half way through the mission time, or 12 hours, steam generator water level would be at or near normal level when flow from the motor-driven AFW pump is lost. Previous analyses have shown that about three hours are required for water level in the steam generators to decrease from nominal to 5-percent wide range. Therefore, three hours would be available to bypass the interlock.

6.1.3 AFZ-BWI4

This equation is used to model failure of the operator to provide AFW flow for decay heat removal using the turbine-driven AFW pump. The equation is used to model operator action-related failures only. Hardware-related failures of the turbine-driven AFW train are evaluated by other equations in the event trees.

This equation is used for sequences with the following conditions:

- The water volume released to the turbine building will result in opening the bottom row of breakers on 480 VAC Buses 51, 52, 61, and 62.
- The water volume released to the turbine building will result in failure of 4kVAC Bus 5 motor loads.
- The water volume released to the turbine building would cause water level in the AFW pump rooms to submerge the TDAFP auxiliary lube oil pump.
- The water volume released to the turbine building would cause water level in safeguards alley to submerge 480 VAC Buses 61 and 62.
- The water volume released to the turbine building would cause water level in the B-train 4kVAC room to reach a level that would fail 4 kVAC Bus 6 motor loads.
- The break was isolated before the volume of water released in the turbine building would cause water level in the safeguards alley to fail the turbine-driven AFW pump if it was already running.

For the sequences where equation AFZ-BWI4 is used, the only method available for long-term decay heat removal is the turbine-driven AFW pump. Although the B-train motor-driven AFW pump may start and provide flow, the pump will be lost when water level on 4 kVAC Bus 6 reaches the level at which bus failure is expected. Therefore, no credit is taken for operation of the B-train motor-driven AFW pump.

Multiple potential success paths exist for the conditions where equation AFZ-BWI4 is used. First, the operators could have maintained the turbine-driven AFW pump running throughout the event. The pump would be maintained running if either motor-driven AFW pump failed or if the operators recognized that the flooding event could threaten the motor-driven AFW pumps and would want the added reliability of the third AFW pump. Second, even if the turbine-driven AFW pump was secured early in the event, then the operators could recognize that the rising water levels would soon threaten the motor-driven AFW pumps and may restart the turbine-driven AFW pump. By definition of the sequences where equation AFZ-BWIB is used, water level will reach a level that will submerge the turbine-driven AFW pump auxiliary lube oil pump. Therefore, for the operators to successfully start the pump from the control room, action must be taken within 49 minutes. Otherwise, the auxiliary lube oil pump would be submerged, thereby preventing the turbine-driven AFW pump from starting.

If the turbine-driven pump is not started within 49 minutes after flood initiation and then maintained running, then the pump could be started if the low oil pressure interlock is bypassed. Bypass of the low oil pressure interlock may be directed by personnel manning the technical support center and would need to be completed before water level in either of the steam generators dropped to less than 5-percent wide range, the point that bleed and feed cooling would be initiated.

6.2 EVALUATION OF TOP EVENT AFX

Instead of quantifying the same fault tree various times to develop different equations, the equations associated with Top Event AFX use multiple fault trees that are quantified a single time. This is due to the complexity added by various human actions and the potential of equipment to already be running. Only the equations associated with a Large Feedwater Break are described here (e.g., AFX-1WIB). The descriptions of the equations associated with all other initiators (e.g., AFX-1CXB, AFX-1SIB, AFX-1TIB, etc.) are identical except for the name of the initiator and the timing associated with the model.

6.2.1 AFX-1WIB

This equation is used to model failure of the operator to control AFW flow to maintain level in the steam generators.

This equation is used for sequences with the following conditions:

- The water volume released to the turbine building will result in opening the bottom row of breakers on 480 VAC Buses 51, 52, 61, and 62.
- The break was isolated before the volume of water released in the turbine building would cause water level in the AFW pump rooms to submerge the TDAFP auxiliary lube oil pump.
- The break was isolated before the volume of water released in the turbine building would cause water level in the B-train 4kVAC room to reach a level that would fail 4 kVAC Bus 6 motor loads or 480 VAC Buses 61/62.
- The water volume released to the turbine building will result in failure of 4kVAC Bus 5 motor loads.
- The B-train motor-driven AFW pump successfully started.
- The B-train motor-driven AFW pump can successfully run for 24 hours.

By definition, all equipment in the turbine building basement is assumed failed by the initiating event. Therefore, service air compressors are lost. Also by definition of the sequences where equation AFX-1WIB is used, the B-train electrical safety buses (4kVAC and 480 VAC) are available. Because the B-train 480 VAC safety buses are available, the B-train instrument air compressor, C1B, is potentially available, as are the alternate power supplies to the 120 VAC instrument inverters. In addition, the A-train 480 VAC safety buses would be available so instrument air compressor C1C is potentially available.

Even though the B-train 480 VAC safety buses are available, the bottom row of breakers on the 480 VAC buses will have opened. When these breakers open, several loads that impact the flooding accident sequence progression are lost. These loads include the power supply to the associated train battery room fan cooling units, battery chargers, standby power supplies for 120 VAC instrument inverters, battery room exhaust fans, and auxiliary lube oil pumps for the motor-driven AFW pumps.

Given the conditions described above, success of the AFX-1WIB equation can be achieved by several means. First the B-train motor-driven AFW pump could be maintained running and flow controlled using AFW-2B. If air and power are available, then flow can be controlled from the control room. Air could be supplied from instrument air compressors C1B or C1C and power is provided from panel BRD-115, which is supplied with power from either battery BRD-101 or MCC-62C. These power sources can be backed up by DC distribution cabinet BRC-102 via either battery BRC-101 or MCC-46C. Given the redundancy and diversity of these four power supplies, explicit consideration of their failure is assumed to be insignificant and need not be modeled. If air is not available, then AFW-2B can be operated locally.

Second, if the operators secure the B-train AFW pump, then the turbine-driven AFW pump can be used. If the turbine-driven AFW pump was maintained running, then no additional actions are required. If the turbine-driven AFW pump was secured, then taking the control switch from pull-to-lock would restart the pump when level reached the low-low setpoint. Once the turbine-driven AFW pump is running, flow can be controlled using valves AFW-10A/B. For either of these options, either a long-term source of DC power must be provided for instrumentation or steam generator level must be controlled following a loss of all level indication. Lastly, if the motor-driven AFW pump 1B has been secured, then the low oil pressure interlock can be bypassed to allow starting the motor-driven AFW pumps without the auxiliary lube oil pumps.

Provision of a long-term source of DC power can be ensured by multiple means for sequences involving equation AFX-1WIB. First, the 120 VAC instrument inverters can be aligned to their alternate power source. Since 480 VAC Buses 61 and 62 are available, the alternate sources are available. Evaluations have shown that if the instrument inverters are removed from battery BRB-101, then the battery can supply needed DC loads for well in excess of 24 hours. Alignment of the instrument inverters to their alternate power source also ensures that steam generator level indication is available in the control room even if the battery fails or is depleted. If needed, an alternate power source can be aligned to the DC buses. Alternatives include installation of an alternate power source to the existing battery charger or installation of a spare battery charger with power from an alternative source.

Given the availability of equipment for sequences where equation AFX-1WIB is used and the multiple success paths that are available, it is likely that many hours would be available for the operators to initiate the actions. Therefore, time would not be critical to completing any of the actions and explicit evaluation of timing is not necessary.

6.2.2 AFX-2WIB

This equation is used to model failure of the operator to control AFW flow to maintain level in the steam generators.

This equation name is used for sequences with the following conditions:

- The water volume released to the turbine building will result in opening the bottom row of breakers on 480 VAC Buses 51, 52, 61, and 62.
- The break was isolated before the volume of water released in the turbine building would cause water level in the B-train 4kVAC room to reach a level that would fail 4 kVAC Bus 6 motor loads or 480 VAC Buses 61/62.
- The water volume released to the turbine building will result in failure of 4kVAC Bus 5 motor loads.
- The B-train motor-driven AFW pump successfully started.
- The B-train motor-driven AFW pump failed to run. By definition of the sequences where equation AFX-2WIB is used, the mission time for the AFW pump is 24 hours. On average, the pump is assumed to fail to run halfway through the mission time or 12 hours.
- The turbine-driven AFW pump has been started and can successfully operate for 24 hours.

By definition, all equipment in the turbine building basement is assumed failed by the initiating event. Therefore, service air compressors are lost. Also by definition of the sequences where equation AFX-2WIB is used, the B-train electrical safety buses (4kVAC and 480 VAC) are available. Because the B-train 480 VAC safety buses are available, the B-train instrument air compressor, C1B, is potentially available, as are the alternate power supplies to the 120 VAC instrument inverters. In addition, the A-train 480 VAC safety buses would be available so instrument air compressor C1C is potentially available.

Even though the B-train 480 VAC safety buses are available, the bottom row of breakers on the 480 VAC buses will have opened. When these breakers open, several loads that impact the flooding accident sequence progression are lost. These loads include the power supply to the associated train battery room fan cooling units, battery chargers, standby power supplies for 120

VAC instrument inverters, battery room exhaust fans, and auxiliary lube oil pumps for the motor-driven AFW pumps.

Given the conditions described above, success of the AFX-2WIB equation can be achieved by controlling AFW flow using valves AFW-10A/B. If a long-term source of DC power is available, then steam generator level can be controlled from the control room. Provision of a long-term source of DC power can be ensured by multiple means for sequences involving equation AFX-2WIB. First, the 120 VAC instrument inverters can be aligned to their alternate power source. Since 480 VAC Buses 61 and 62 are available, the alternate sources are available. Evaluations have shown that if the instrument inverters are removed from battery BRB-101, then the battery can supply needed DC loads for well in excess of 24 hours. Alignment of the instrument inverters to their alternate power source also ensures that steam generator level indication is available in the control room even if the battery fails or is depleted. If needed, an alternate power source can be aligned to the DC buses. Alternatives include installation of an alternate power source to the existing battery charger or installation of a spare battery charger with power from an alternative source.

Given the availability of equipment for sequences where equation AFX-2WIB is used and the multiple success paths that are available, it is likely that many hours would be available for the operators to initiate the actions. Therefore, time would not be critical to completing any of the actions and explicit evaluation of timing is not necessary.

6.2.3 AFX-1AWI

This equation is used to model failure of the operator to control AFW flow to maintain level in the steam generators.

This equation is used for sequences with the following conditions:

- The water volume released to the turbine building will result in opening the bottom row of breakers on 480 VAC Buses 51, 52, 61, and 62.
- The water volume released to the turbine building will result in submergence of the TDAFP auxiliary lube oil pump.
- The break was isolated before the volume of water released in the turbine building would cause water level in the B-train 4kVAC room to reach a level that would fail 4 kVAC Bus 6 motor loads or 480 VAC Buses 61/62.
- The water volume released to the turbine building will result in failure of 4kVAC Bus 5 motor loads.
- The B-train motor-driven AFW pump successfully started.
- The B-train motor-driven AFW pump can successfully run for 24 hours.

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By definition, all equipment in the turbine building basement is assumed failed by the initiating event. Therefore, service air compressors are lost. Also by definition of the sequences where equation AFX-1AWI is used, the B-train electrical safety buses (4kVAC and 480 VAC) are available. Because the B-train 480 VAC safety buses are available, the B-train instrument air compressor, C1B, is potentially available, as are the alternate power supplies to the 120 VAC instrument inverters. In addition, the A-train 480 VAC safety buses would be available so instrument air compressor C1C is potentially available.

Even though the B-train 480 VAC safety buses are available, the bottom row of breakers on the 480 VAC buses will have opened. When these breakers open, several loads that impact the flooding accident sequence progression are lost. These loads include the power supply to the associated train battery room fan cooling units, battery chargers, standby power supplies for 120 VAC instrument inverters, battery room exhaust fans, and auxiliary lube oil pumps for the motor-driven AFW pumps.

Given the conditions described above, success of the AFX-1AWI equation can be achieved by several means. First the B-train motor-driven AFW pump could be maintained running and flow controlled using AFW-2B. If air and power are available, then flow can be controlled from the control room. Air could be supplied from instrument air compressors C1B or C1C and power is provided from panel BRD-115, which is supplied with power from either battery BRD-101 or MCC-62C. These power sources can be backed up by DC distribution cabinet BRC-102 via either battery BRC-101 or MCC-46C. Given the redundancy and diversity of these four power supplies, explicit consideration of their failure is assumed to be insignificant and need not be modeled. If air is not available, then AFW-2B can be operated locally.

Second, if the operators secure the B-train AFW pump, then the turbine-driven AFW pump can be used. By definition of the sequences where equation AFX-1AWI is used, water level will reach a level that will submerge the turbine-driven AFW pump auxiliary lube oil pump. Therefore, for the operators to successfully start the pump from the control room, action must be taken within 49 minutes. Otherwise, the auxiliary lube oil pump would be submerged, thereby preventing the turbine-driven AFW pump from starting.

If the turbine-driven pump is not started within 49 minutes after flood initiation and then maintained running, then the pump could be started if the low oil pressure interlock is bypassed. Bypass of the low oil pressure interlock may be directed by personnel manning the technical support center and would need to be completed before water level in either of the steam generators dropped to less than 5-percent wide range, the point that bleed and feed cooling would be initiated.

For either of these options, either a long-term source of DC power must be provided for instrumentation or steam generator level must be controlled following a loss of all level indication.

Provision of a long-term source of DC power can be ensured by multiple means for sequences involving equation AFX-1AWI. First, the 120 VAC instrument inverters can be aligned to their alternate power source. Since 480 VAC Buses 61 and 62 are available, the alternate sources are available. Evaluations have shown that if the instrument inverters are removed from battery BRB-101, then the battery can supply needed DC loads for well in excess of 24 hours. Alignment of the instrument inverters to their alternate power source also ensures that steam generator level indication is available in the control room even if the battery fails or is depleted. If needed, an alternate power source can be aligned to the DC buses. Alternatives include installation of an alternate power source to the existing battery charger or installation of a spare battery charger with power from an alternative source.

6.2.4 AFX-2WI4

This equation is used to model failure of the operator to control AFW flow to maintain level in the steam generators.

This equation is used for sequences with the following conditions:

- The water volume released to the turbine building will result in opening the bottom row of breakers on 480 VAC Buses 51, 52, 61, and 62.
- The water volume released to the turbine building will result in failure of 4kVAC Bus 5 motor loads.
- The water volume released to the turbine building would cause water level in the AFW pump rooms to submerge the TDAFP auxiliary lube oil pump.
- The water volume released to the turbine building would cause water level in safeguards alley to submerge 480 VAC Buses 61 and 62.
- The water volume released to the turbine building would cause water level in the B-train 4kVAC room to reach a level that would fail 4 kVAC Bus 6 motor loads.
- The break was isolated before the volume of water released in the turbine building would cause water level in the safeguards alley to fail the turbine-driven AFW pump if it was already running.
- The turbine-driven AFW pump has been started and can successfully operate for 24 hours.

By definition of the sequences where equation AFX-2WI4 is used, all AC power will be lost. In addition, all DC power may eventually be lost because of the loss of power to the battery chargers.

Success of the AFX-2WI4 equation requires that the operators control flow using valves AFW-10A/B. In addition, either a long-term source of DC power must be provided for instrumentation or steam generator level must be controlled following a loss of all level indication.

7.0 REFERENCES

[CALC01] Calculation 0064-0515-LYS-01, Evaluation of Flooding Levels for Various PRA Cases, Revision 2, MPR Associates, Inc.

[NB01] KPS Internal Events PRA, Volumes 2 through 4.

[NB02] KPS Internal Events PRA, Section 4.15, "Human Reliability Analysis."

Table 1
Summary of KPS Turbine Building Flood Human Actions

Basic Event ID	Basic Event Description	HEP
02-SW4A-B29F-HE	Detection and Isolation of a Service Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	2.0E-02
02-SW4A-B45F-HE	Detection and Isolation of a Service Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	2.0E-02
02-SW4A-B51F-HE	Detection and Isolation of a Large Service Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	2.0E-02
02-SW4A-B60F-HE	Detection and Isolation of a Large Service Water Break before Failure of the Motor Driven AFW Pumps	2.0E-02
02-SW4A-B66F-HE	Detection and Isolation of a Large Service Water Break before Water Level Reaches 18 Inches in the Turbine Building	2.0E-02
04-CWSTP13-F-HE	Detection and Isolation of a 14,000-gpm Circulating Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	2.6E-01
04-CWSTP19-F-HE	Detection and Isolation of a 14,000-gpm Circulating Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	1.2E-01
04-CWSTP22-F-HE	Detection and Isolation of a 14,000-gpm Circulating Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	1.2E-01
04-CWSTP25-F-HE	Detection and Isolation of a 14,000-gpm Circulating Water Break before Water Level Reaches 18 Inches in the Turbine Building	1.2E-01
04-CWSTP29-F-HE	Detection and Isolation of a Small Circulating Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	4.3E-02

Table 1
Summary of KPS Turbine Building Flood Human Actions

Basic Event ID	Basic Event Description	HEP
04-CWSTP45-F-HE	Detection and Isolation of a Small Circulating Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	1.7E-02
04-CWSTP51-F-HE	Detection and Isolation of a Small Circulating Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	1.4E-02
04-CWSTP60-F-HE	Detection and Isolation of a Small Circulating Water Break before Failure of the Motor Driven AFW Pumps	1.2E-02
04-CWSTP66-F-HE	Detection and Isolation of a Small Circulating Water Break before Water Level Reaches 18 Inches in the Turbine Building	1.2E-02
04-CW-TRIP-F-HE	Detection and Isolation of a 58,000-gpm Circulating Water Break before Failing Both 480 V Buses	1.0E+00
05B-BYALOP-F-HE	Failure to Bypass AFW Auxiliary Lube Oil Pressure Interlock	4.4E-01
05B-MDPTD1CF-HE	Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (108 Minutes)	1.6E-01
05B-MDPTD2HF-HE	Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (2 Hours)	4.1E-02
05B-MDPTD36F-HE	Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (36 Minutes)	4.7E-01
05B-MDPTD49F-HE	Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (49 Minutes)	4.7E-01
05B-MDPTD61F-HE	Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (61 Minutes)	3.1E-01
06-NOINDAFWF-HE	Failure to Feed Steam Generator Without Level Indication	6.4E-01
06--OC2----F-HE	Failure to Perform RCS Cooldown Using Natural Circulation	7.4E-02

Table 1
Summary of KPS Turbine Building Flood Human Actions

Basic Event ID	Basic Event Description	HEP
06--OC6----F-HE	Failure to Perform RCS Cooldown with Boration	9.2E-02
08-FPISO29-F-HE	Detection and Isolation of a Fire Protection Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	1.0E+00
08-FPISO45-F-HE	Detection and Isolation of a Fire Protection Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	6.6E-02
08-FPISO56-F-HE	Detection and Isolation of a Fire Protection Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	2.4E-02
08-FPISO68-F-HE	Detection and Isolation of a Fire Protection Water Break before Failure of the Motor Driven AFW Pumps	1.6E-02
08-FPSISO1CF-HE	Detection and Isolation of a Medium Flood due to a Steamline Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	3.0E-02
08-FPSISO29F-HE	Detection and Isolation of a Large Flood due to a Steamline Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	1.0
08-FPSISO2CF-HE	Detection and Isolation of a Medium Flood due to a Steamline Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	3.0E-02
08-FPSISO3CF-HE	Detection and Isolation of a Medium Flood due to a Steamline Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	3.0E-02
08-FPSISO45F-HE	Detection and Isolation of a Large Flood due to a Steamline Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	6.6E-02
08-FPSISO56F-HE	Detection and Isolation of a Large Flood due to a Steamline Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	3.0E-02

Table 1
Summary of KPS Turbine Building Flood Human Actions

Basic Event ID	Basic Event Description	HEP
08-FPSISO68F-HE	Detection and Isolation of a Large Flood due to a Steamline Break before Failure of the Motor Driven AFW Pumps	3.0E-02
08-ISO-FS18F-HE	Detection and Isolation of a Large Flood due to a Feedwater Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	1.0E+00
08-ISO-FS2HF-HE	Detection and Isolation of a Medium Flood due to a Feedwater Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	1.7E-02
08-ISO-FS33F-HE	Detection and Isolation of a Large Flood due to a Feedwater Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	4.4E-01
08-ISO-FS40F-HE	Detection and Isolation of a Large Flood due to a Feedwater Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	1.3E-01
08-ISO-FS54F-HE	Detection and Isolation of a Large Flood due to a Feedwater Break before Failure of the Motor Driven AFW Pumps	3.0E-02
08-ISO-FS55F-HE	Detection and Isolation of a Medium Flood due to a Feedwater Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	3.0E-02
08-ISO-FS97F-HE	Detection and Isolation of a Medium Flood due to a Feedwater Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	3.0E-02
16-BATCLG--F-HE	Establish Battery Room Cooling	7.9E-02
27A-ORR----F-HE	Failure to Throttle SI Flow to Conserve RWST Inventory	5.0E-03
86-INSTRRCRF-HE	Failure to Recover AFW Control	1.8E-02

Table 2: Basic Events Added to KNPP.BED

Basic Event ID	Basic Event Description	Point Estimate
02-SW4A-B29F-HE	Detection and Isolation of a Service Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	2.0E-02
02-SW4A-B45F-HE	Detection and Isolation of a Service Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	2.0E-02
02-SW4A-B51F-HE	Detection and Isolation of a Large Service Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	2.0E-02
02-SW4A-B60F-HE	Detection and Isolation of a Large Service Water Break before Failure of the Motor Driven AFW Pumps	2.0E-02
02-SW4A-B66F-HE	Detection and Isolation of a Large Service Water Break before Water Level Reaches 18 Inches in the Turbine Building	2.0E-02
04-CWSTP13-F-HE	Detection and Isolation of a 14,000-gpm Circulating Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	2.6E-01
04-CWSTP19-F-HE	Detection and Isolation of a 14,000-gpm Circulating Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	1.2E-01
04-CWSTP22-F-HE	Detection and Isolation of a 14,000-gpm Circulating Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	1.2E-01
04-CWSTP25-F-HE	Detection and Isolation of a 14,000-gpm Circulating Water Break before Water Level Reaches 18 Inches in the Turbine Building	1.2E-01
04-CWSTP29-F-HE	Detection and Isolation of a Small Circulating Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	4.3E-02
04-CWSTP45-F-HE	Detection and Isolation of a Small Circulating Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	1.7E-02

Table 2: Basic Events Added to KNPP.BED

Basic Event ID	Basic Event Description	Point Estimate
04-CWSTP51-F-HE	Detection and Isolation of a Small Circulating Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	1.4E-02
04-CWSTP60-F-HE	Detection and Isolation of a Small Circulating Water Break before Failure of the Motor Driven AFW Pumps	1.2E-02
04-CWSTP66-F-HE	Detection and Isolation of a Small Circulating Water Break before Water Level Reaches 18 Inches in the Turbine Building	1.2E-02
04-CW-TRIP-F-HE	Detection and Isolation of a 58,000-gpm Circulating Water Break before Failing Both 480 V Buses	1.0E+00
05B-BYALOP-F-HE	Failure to Bypass AFW Auxiliary Lube Oil Pressure Interlock	4.4E-01
05B-MDPTD1CF-HE	Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (108 Minutes)	1.6E-01
05B-MDPTD2HF-HE	Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (2 Hours)	4.1E-02
05B-MDPTD36F-HE	Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (36 Minutes)	4.7E-01
05B-MDPTD49F-HE	Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (49 Minutes)	4.7E-01
05B-MDPTD61F-HE	Failure to Start Turbine Driven AFW Pump Before Loss of Motor Driven AFW Pump (61 Minutes)	3.1E-01
06-NOINDAFWF-HE	Failure to Feed Steam Generator Without Level Indication	6.4E-01
06--OC2----F-HE	Failure to Perform RCS Cooldown Using Natural Circulation	7.4E-02
06--OC6----F-HE	Failure to Perform RCS Cooldown with Boration	9.2E-02

Table 2: Basic Events Added to KNPP.BED

Basic Event ID	Basic Event Description	Point Estimate
08-FPISO29-F-HE	Detection and Isolation of a Fire Protection Water Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	1.0E+00
08-FPISO45-F-HE	Detection and Isolation of a Fire Protection Water Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	6.6E-02
08-FPISO56-F-HE	Detection and Isolation of a Fire Protection Water Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	2.4E-02
08-FPISO68-F-HE	Detection and Isolation of a Fire Protection Water Break before Failure of the Motor Driven AFW Pumps	1.6E-02
08-FPSISO1CF-HE	Detection and Isolation of a Medium Flood due to a Steamline Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	3.0E-02
08-FPSISO29F-HE	Detection and Isolation of a Large Flood due to a Steamline Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	1.0
08-FPSISO2CF-HE	Detection and Isolation of a Medium Flood due to a Steamline Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	3.0E-02
08-FPSISO3CF-HE	Detection and Isolation of a Medium Flood due to a Steamline Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	3.0E-02
08-FPSISO45F-HE	Detection and Isolation of a Large Flood due to a Steamline Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	6.6E-02
08-FPSISO56F-HE	Detection and Isolation of a Large Flood due to a Steamline Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	3.0E-02
08-FPSISO68F-HE	Detection and Isolation of a Large Flood due to a Steamline Break before Failure of the Motor Driven AFW Pumps	3.0E-02
08-ISO-FS18F-HE	Detection and Isolation of a Large Flood due to a Feedwater Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	1.0E+00

Table 2: Basic Events Added to KNPP.BED

Basic Event ID	Basic Event Description	Point Estimate
08-ISO-FS2HF-HE	Detection and Isolation of a Medium Flood due to a Feedwater Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	1.7E-02
08-ISO-FS33F-HE	Detection and Isolation of a Large Flood due to a Feedwater Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	4.4E-01
08-ISO-FS40F-HE	Detection and Isolation of a Large Flood due to a Feedwater Break before Failing 480 VAC Buses 61 and 62 and Eventual Isolation of the 4 kVAC Bus 6 Motor Loads	1.3E-01
08-ISO-FS54F-HE	Detection and Isolation of a Large Flood due to a Feedwater Break before Failure of the Motor Driven AFW Pumps	3.0E-02
08-ISO-FS55F-HE	Detection and Isolation of a Medium Flood due to a Feedwater Break before Eventual Isolation of the 4 kVAC Bus 5 Motor Loads	3.0E-02
08-ISO-FS97F-HE	Detection and Isolation of a Medium Flood due to a Feedwater Break before Failing the Turbine Driven AFW Pump Auxiliary Lube Oil Pump	3.0E-02
16-BATCLG--F-HE	Establish Battery Room Cooling	7.9E-02
27A-ORR----F-HE	Failure to Throttle SI Flow to Conserve RWST Inventory	5.0E-03
86-INSTRRCRF-HE	Failure to Recover AFW Control	1.8E-02
04-CW-MDAFPAMHE	Operator Fails to Control MDAFP Med CW Break AC Avail	1.00E-01
04-CW-TRIP-F-HE	FAIL TO ISOL LRG CIRC WTR BRK BEFORE FAILURE OF 480V BUS	1.00E+00
05B-FRACTDP-OFF	Prob of Conditions Where TDAFP Is Secured	9.00E-01
CX06-ISOL-A	Fail to Isolate Before Failure of any Buses CW Mod	5.00E-01
CX06-ISOL-B	Fail to Isolate Before Failure of AFWP CW Mod	5.00E-01
CX06-ISOL-C	Fail to Start MDAFP CW Moderate	5.00E-01
CX06-ISOL-D	Fail to Start MDAFP CW Moderate	5.00E-01

Table 2: Basic Events Added to KNPP.BED

Basic Event ID	Basic Event Description	Point Estimate
FI06-ISOL-A	Fail to Isolate Before Failure of any Buses FP Large	5.00E-01
FI06-ISOL-B	Fail to Isolate Before Failure of AFWP FP Large	5.00E-01
FI06-ISOL-C	Fail to Start MDAFP FP Large	5.00E-01
IE-CI06B	LARGE CIRC WTR LINE BREAK IN TURB BLDG BASEMENT	4.76E-05
IE-CX06B	MEDIUM CIRC WTR LINE BREAK IN TURB BLDG BASEMENT	4.76E-05
IE-CY06B	SMALL CIRC WTR LINE BREAK IN TURB BLDG BASEMENT	7.34E-05
IE-FI06B	LARGE FIRE PROTECT LINE BREAK IN TURB BLDG BASEMENT	1.05E-04
IE-SI06B	LARGE SERVICE WTR LINE BREAK IN TURB BLDG BASEMENT	3.22E-05
IE-TI06B	STEAMLINE BRK IN TURB BLDG CAUSES LARGE FIRE PROT	9.00E-03
IE-TX06B	STEAMLINE BRK IN TURB BLDG CAUSES MEDIUM FIRE PROT	9.00E-03
IE-WI06B	LARGE FEEDWATER BREAK IN TURBINE BLDG BASEMENT	9.41E-04
IE-WX06B	MEDIUM FEEDWATER BREAK IN TURBINE BLDG BASEMENT	9.41E-04
SI06-ISOL-A	Fail to Isolate Before Failure of any Buses SW Large	1.00E+00
SI06-ISOL-B	Fail to Isolate Before Failure of AFWP SW Large	5.00E-01
SI06-ISOL-C	Fail to Start MDAFP SW Large	5.00E-01
SI06-ISOL-D	FAIL ISOLATION BEFORE 18 INCHES ON TDAFP SW LARGE	5.00E-01
SL21-CD	RCP SEAL LOCA GREATER THAN 21 GPM NO RCS COOLDOWN	2.00E-01
SL21-NO-CD	RCP SEAL LOCA GREATER THAN 21 GPM NO RCS COOLDOWN	6.00E-01
TI06-ISOL-A	Fail to Isolate Before Failure of any Buses STM Large	5.00E-01

Table 2: Basic Events Added to KNPP.BED

Basic Event ID	Basic Event Description	Point Estimate
TI06-ISOL-B	Fail to Isolate Before Failure of AFWP STM Large	5.00E-01
TI06-ISOL-C	Fail to Start MDAFP STM Large	5.00E-01
TX06-ISOL-A	Fail to Isolate Before Failure of any Buses STM Mod	5.00E-01
TX06-ISOL-B	Fail to Isolate Before Failure of AFWP STM Mod	5.00E-01
TX06-ISOL-C	Fail to Start MDAFP STM Moderate	5.00E-01
WI06-ISOL-A	Fail to Isolate Before Failure of any Buses FW Large	1.00E+00
WI06-ISOL-B	Fail to Isolate Before Failure of AFWP FW Large	5.00E-01
WI06-ISOL-C	Fail to Start MDAFP FW Large	5.00E-01
WX06-ISOL-A	Fail to Isolate Before Failure of any Buses FW Mod	5.00E-01
WX06-ISOL-B	Fail to Isolate Before Failure of AFWP FW Mod	5.00E-01
WX06-ISOL-C	Fail to Start MDAFP FW Moderate	5.00E-01

INTERNAL FLOODING – Fault Tree Analysis for Turbine Building Floods

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Figure 1 – Fault Tree AFM.LGC

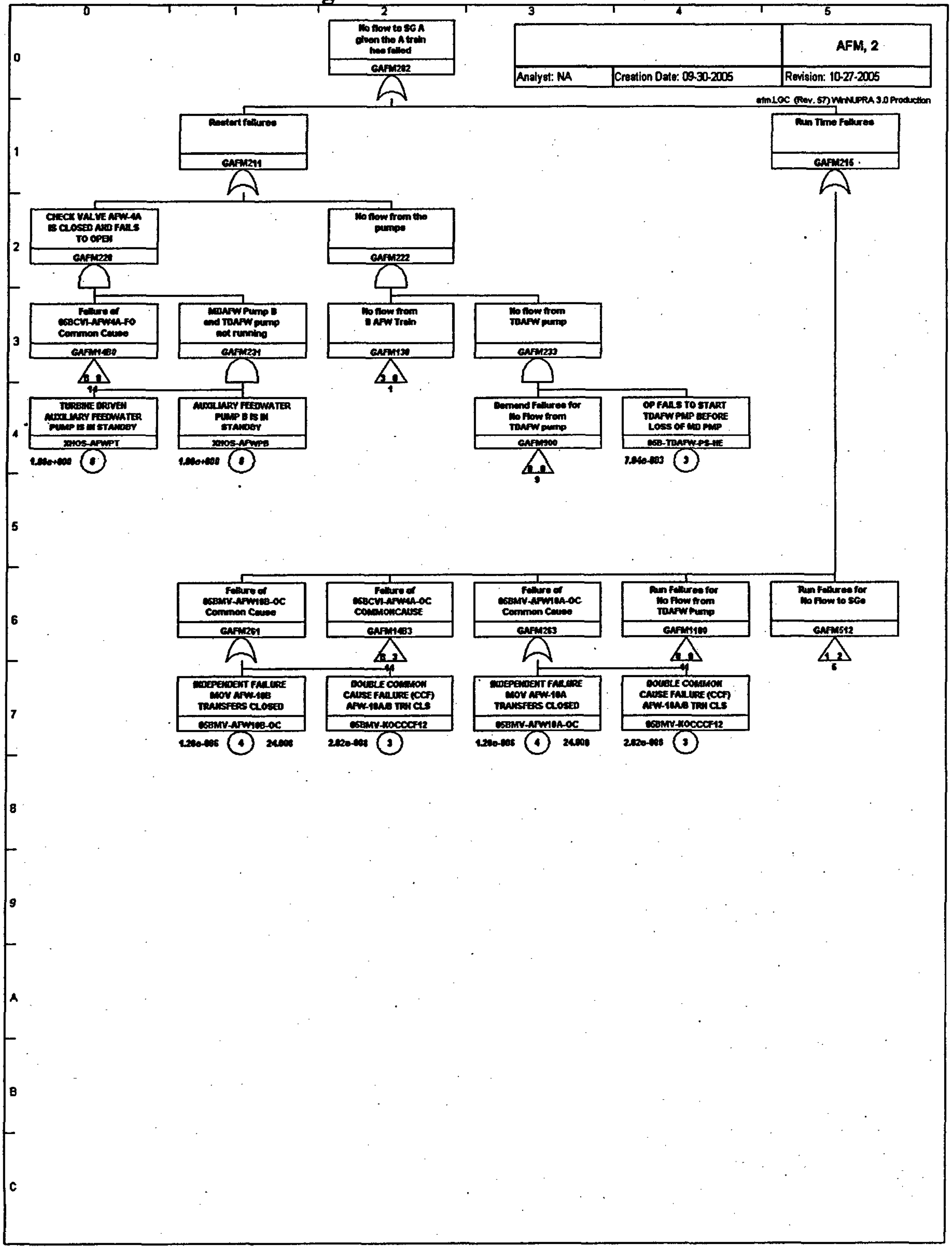

Figure 1 – Fault Tree AFM.LGC (continued)

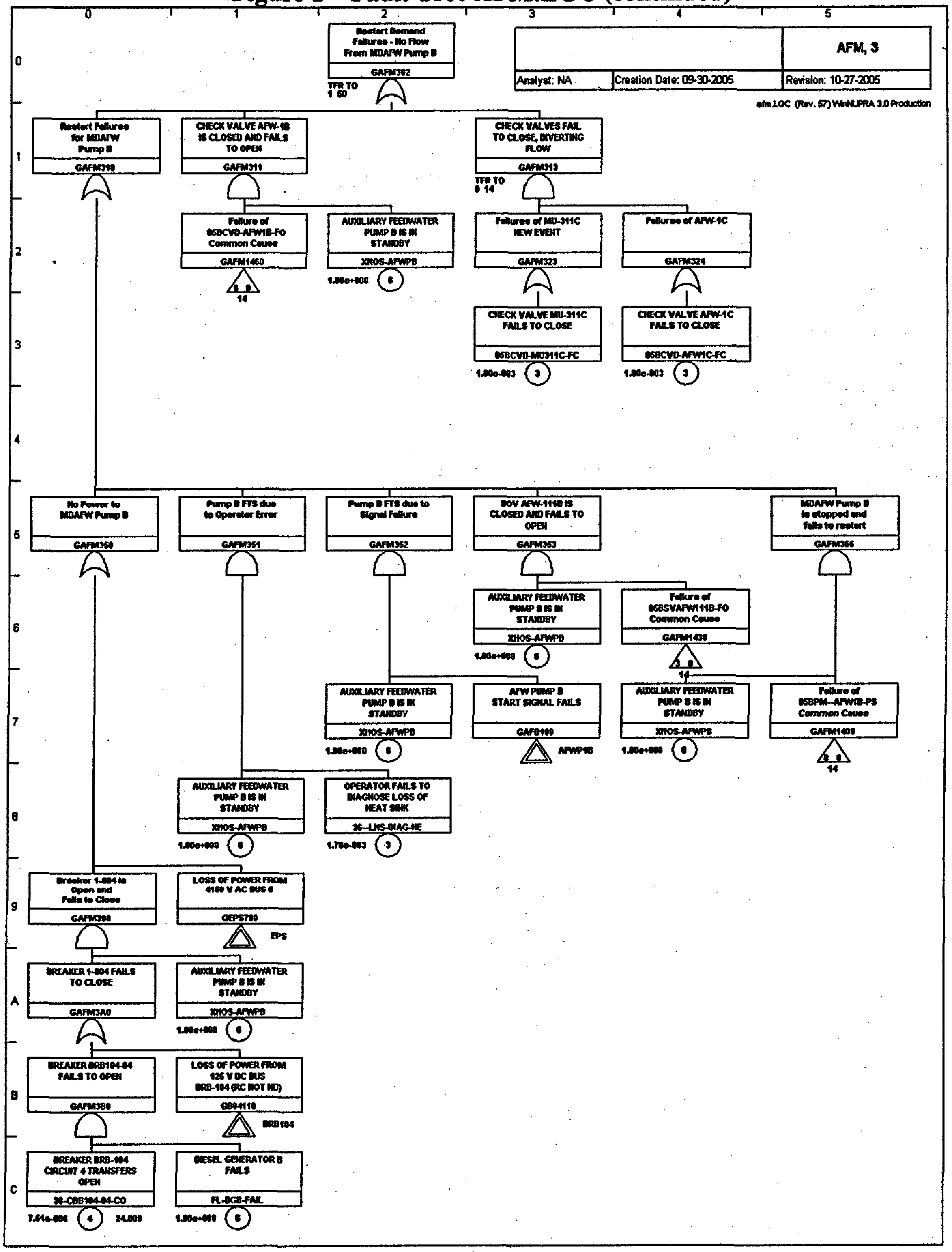

Figure 1 – Fault Tree AFM.LGC (continued)

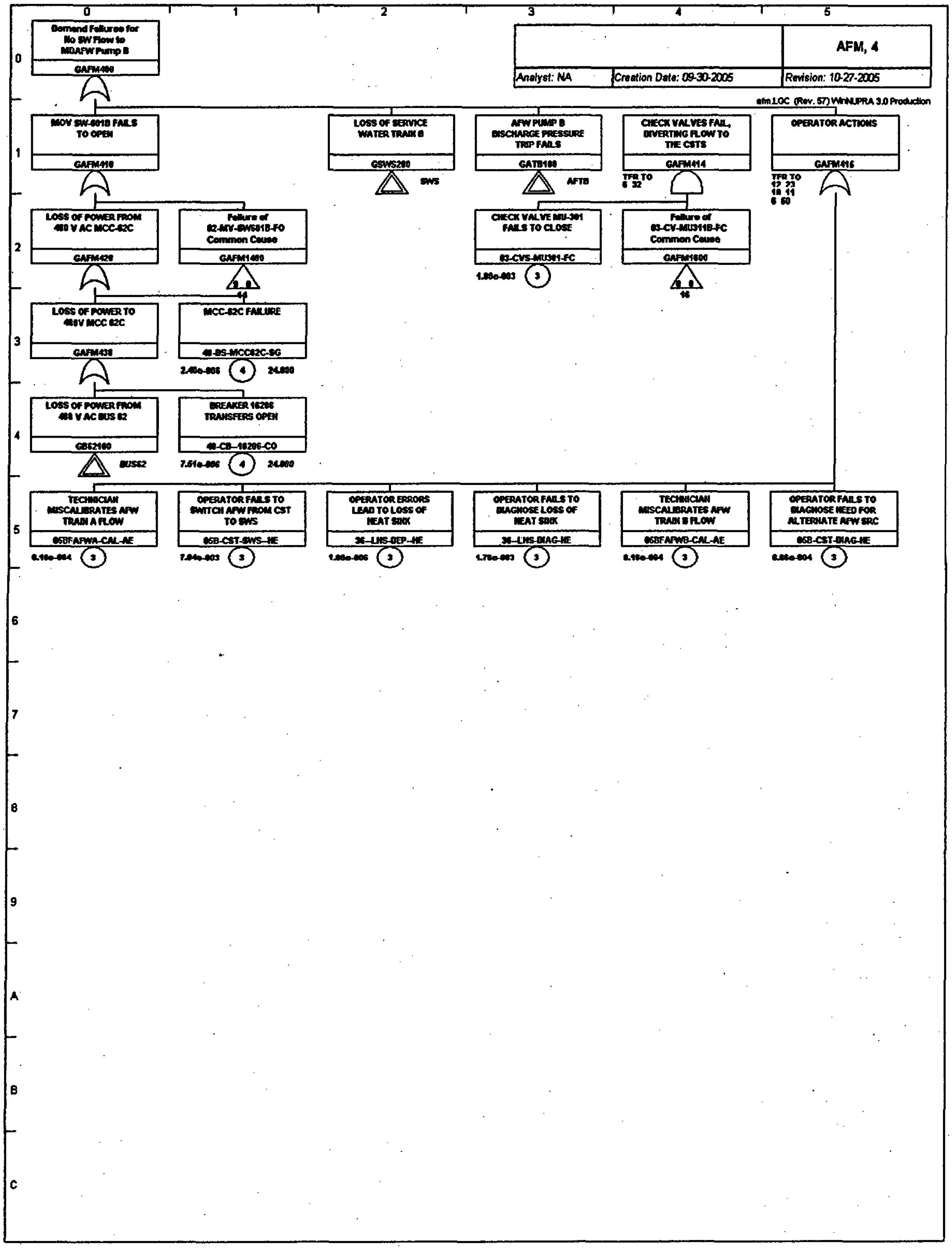

Figure 1 – Fault Tree AFM.LGC (continued)

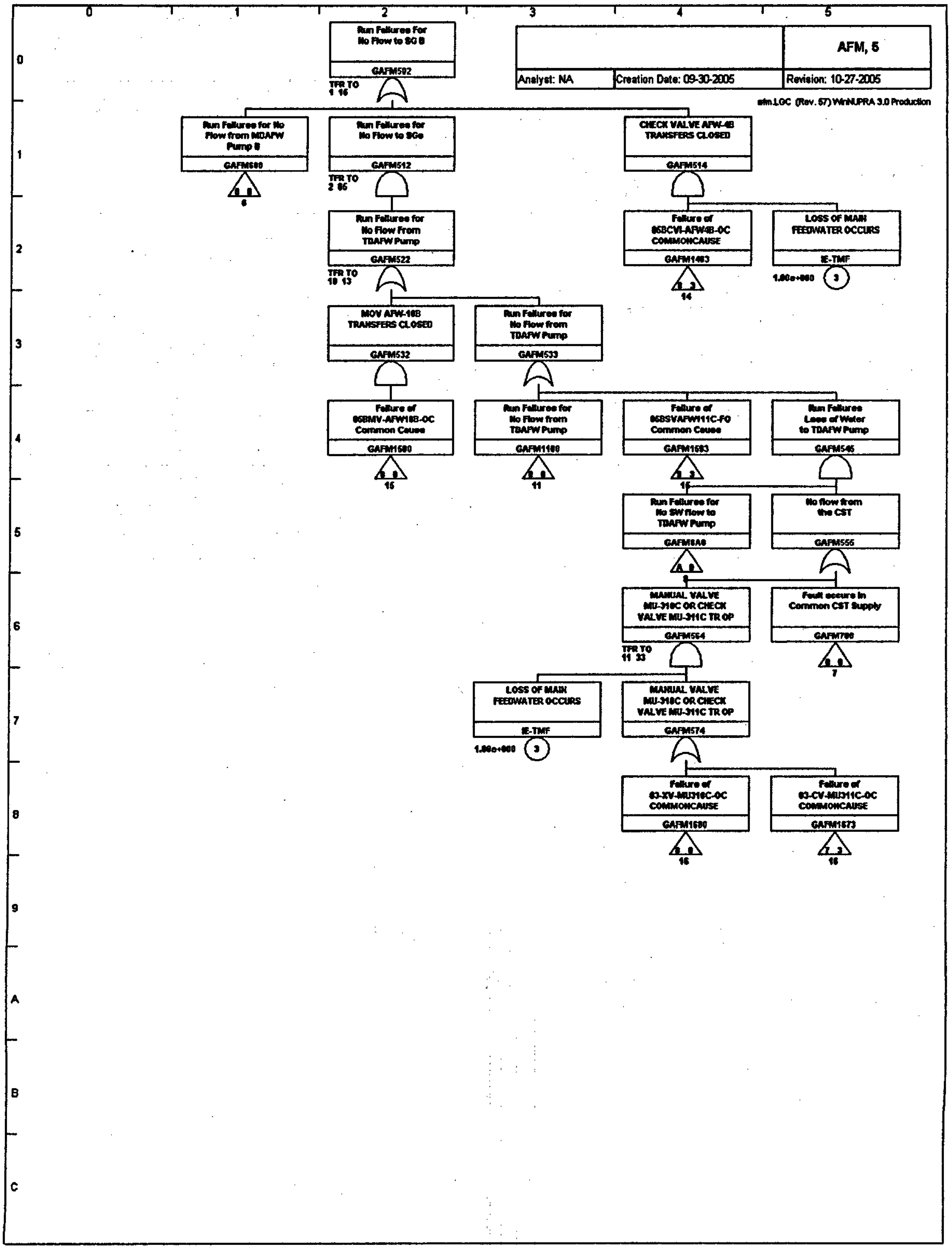

Figure 1 – Fault Tree AFM.LGC (continued)

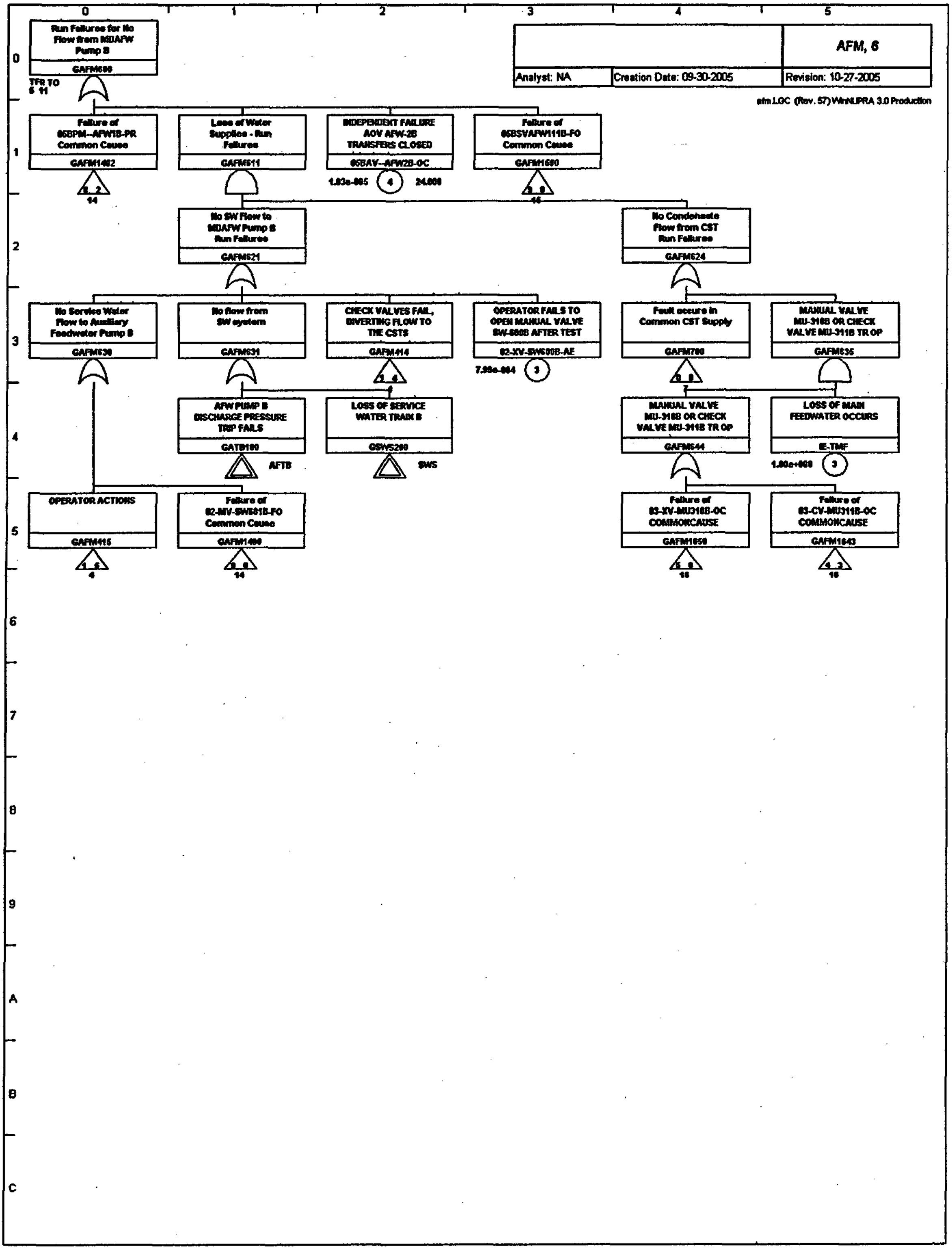

Figure 1 – Fault Tree AFM.LGC (continued)

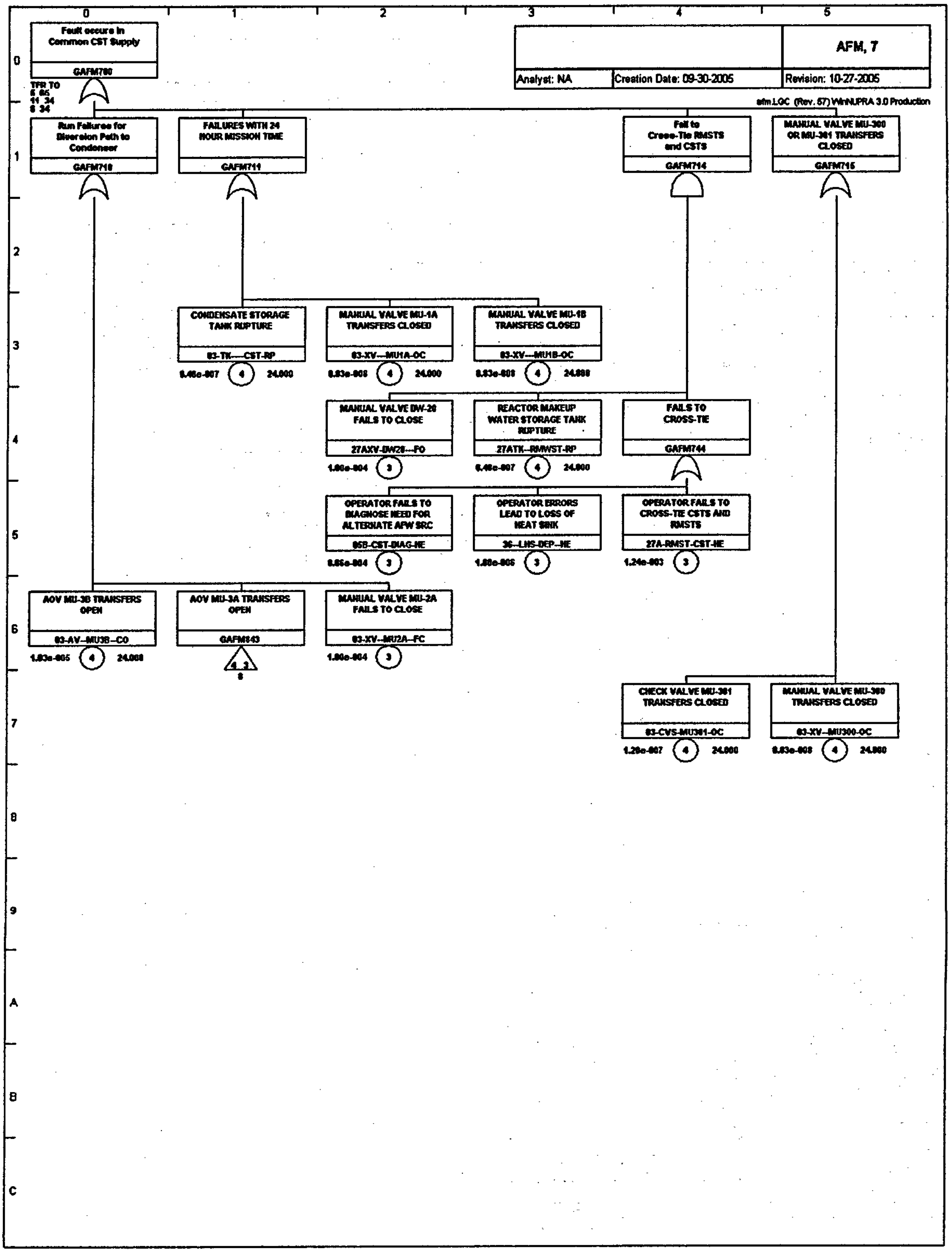

Figure 1 – Fault Tree AFM.LGC (continued)

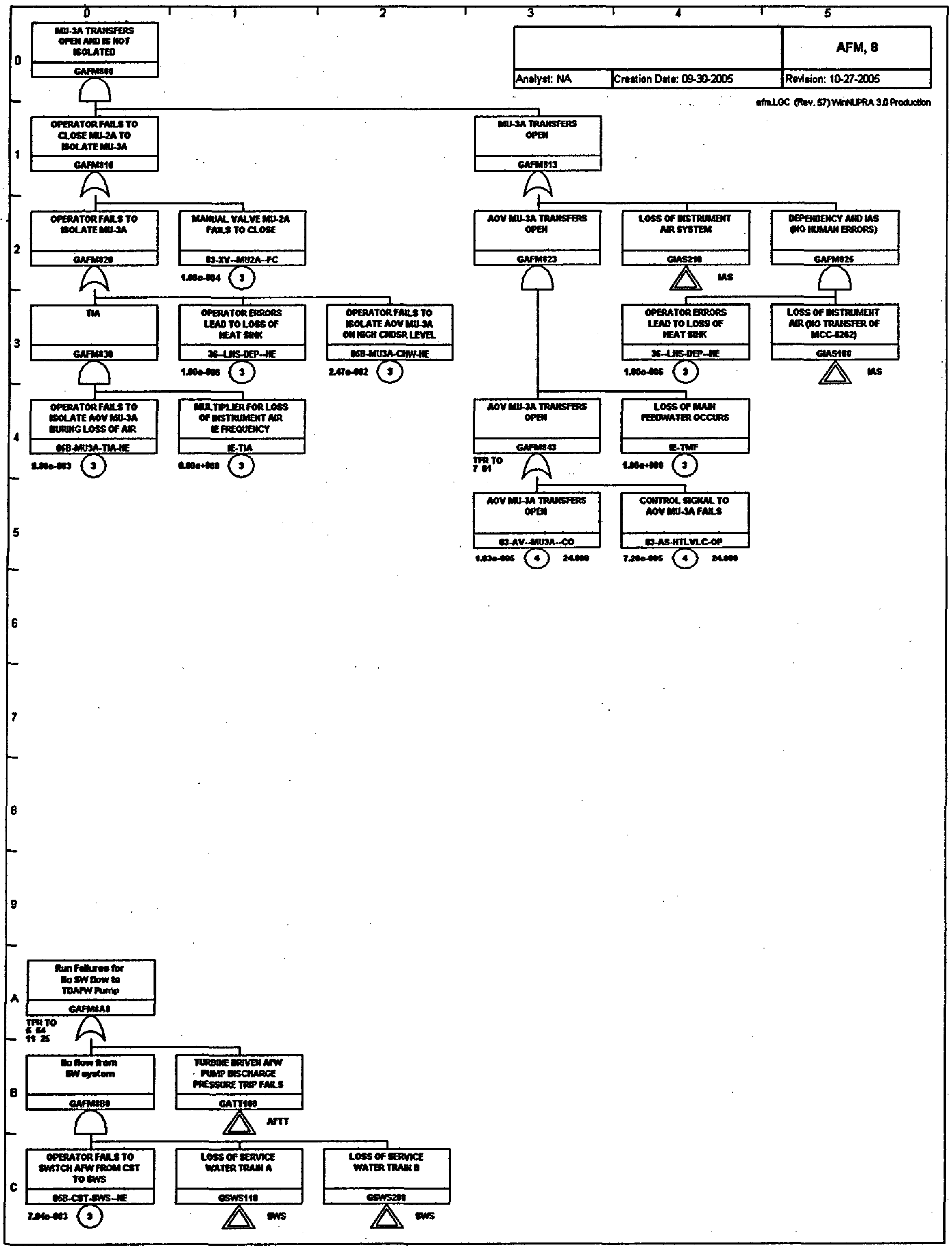

Figure 1 – Fault Tree AFM.LGC (continued)

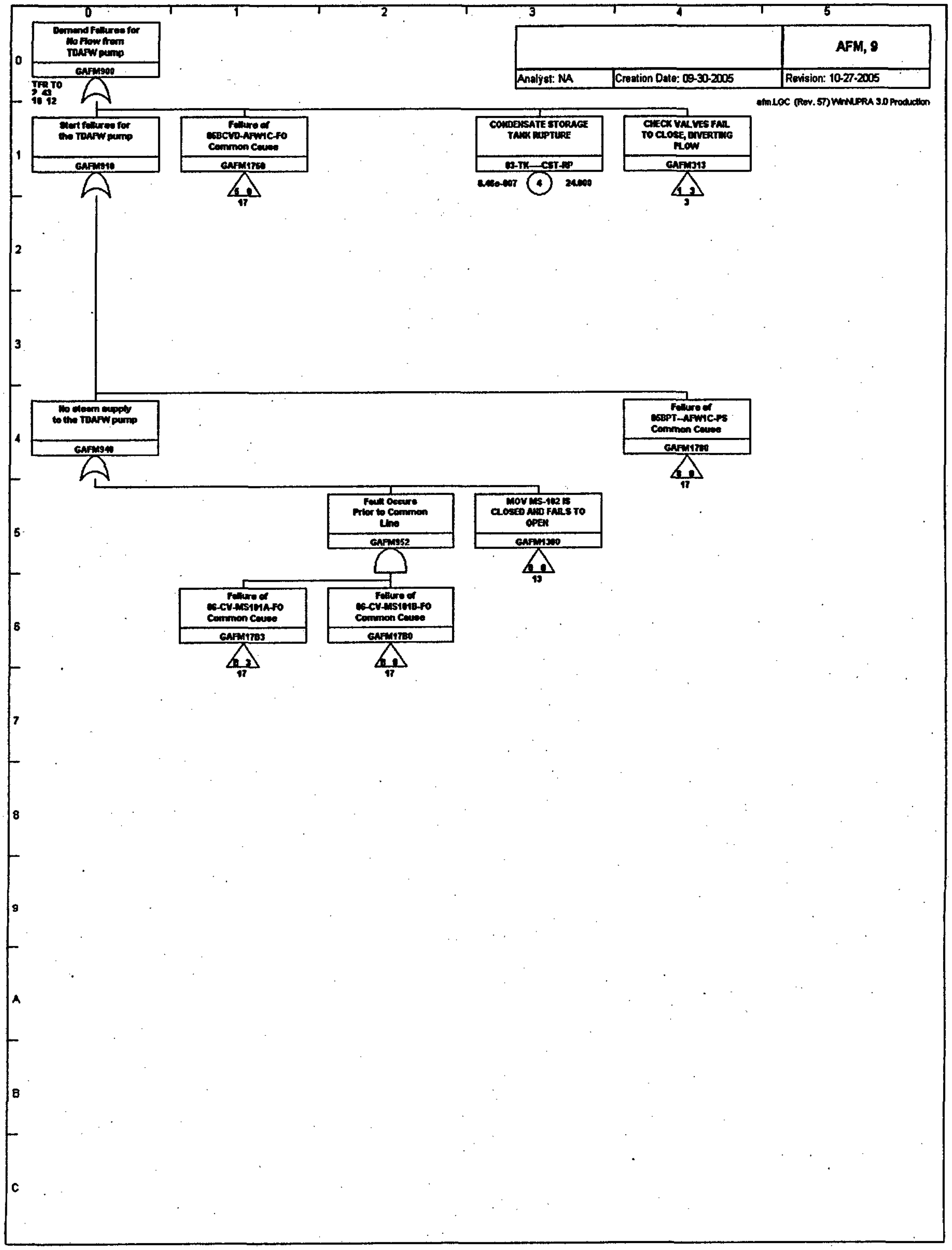

Figure 1 – Fault Tree AFM.LGC (continued)

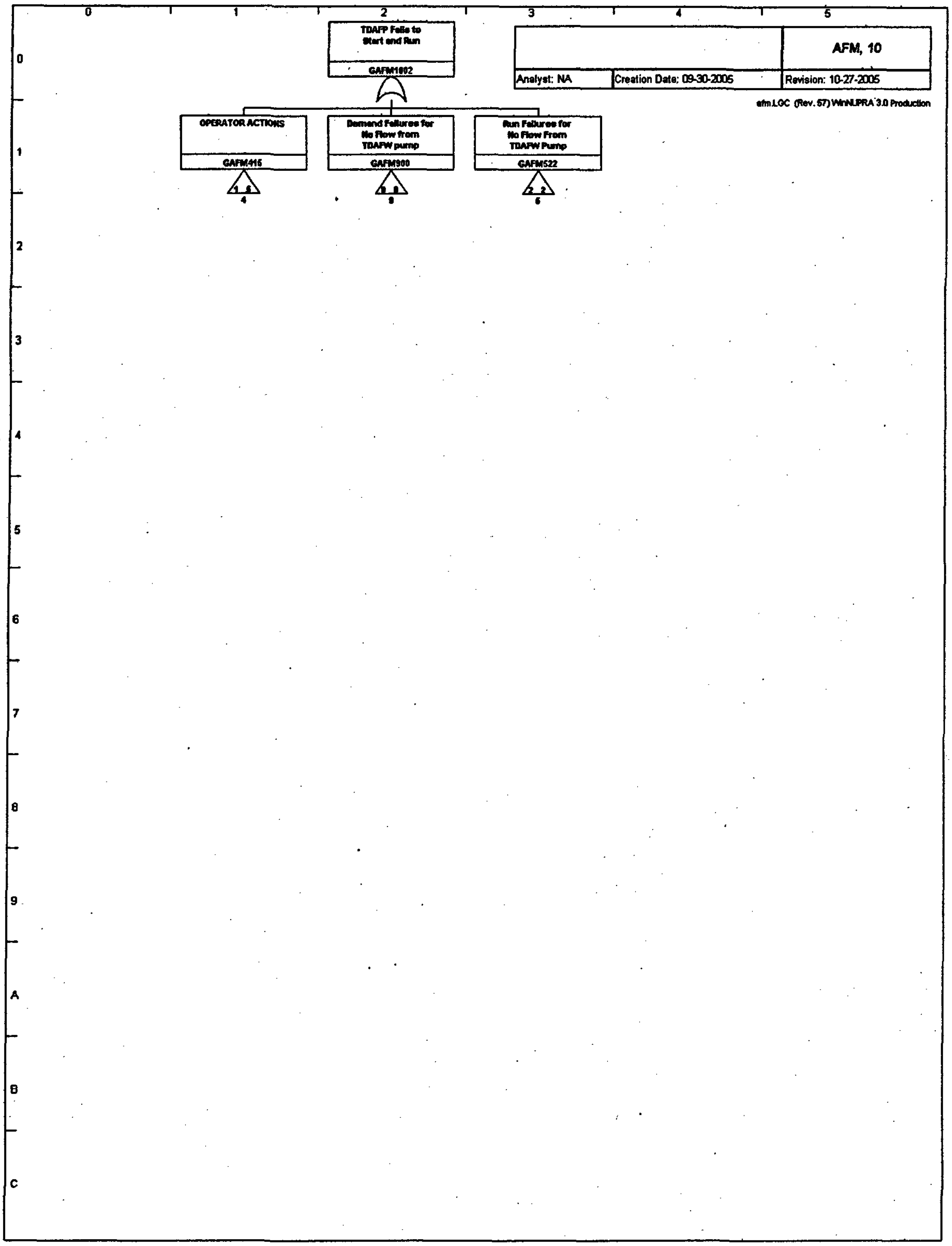

Figure 1 – Fault Tree AFM.LGC (continued)

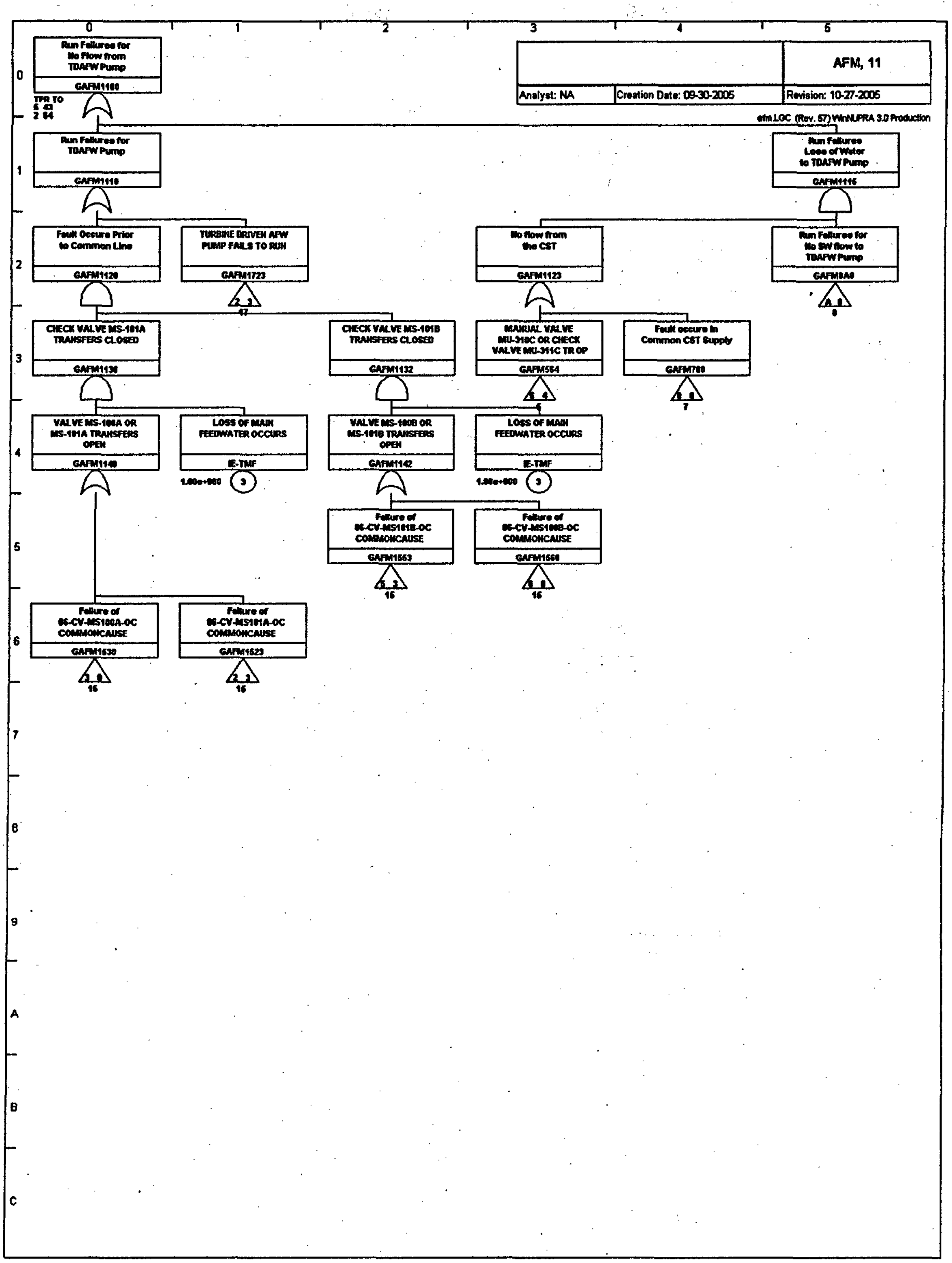

Figure 1 – Fault Tree AFM.LGC (continued)

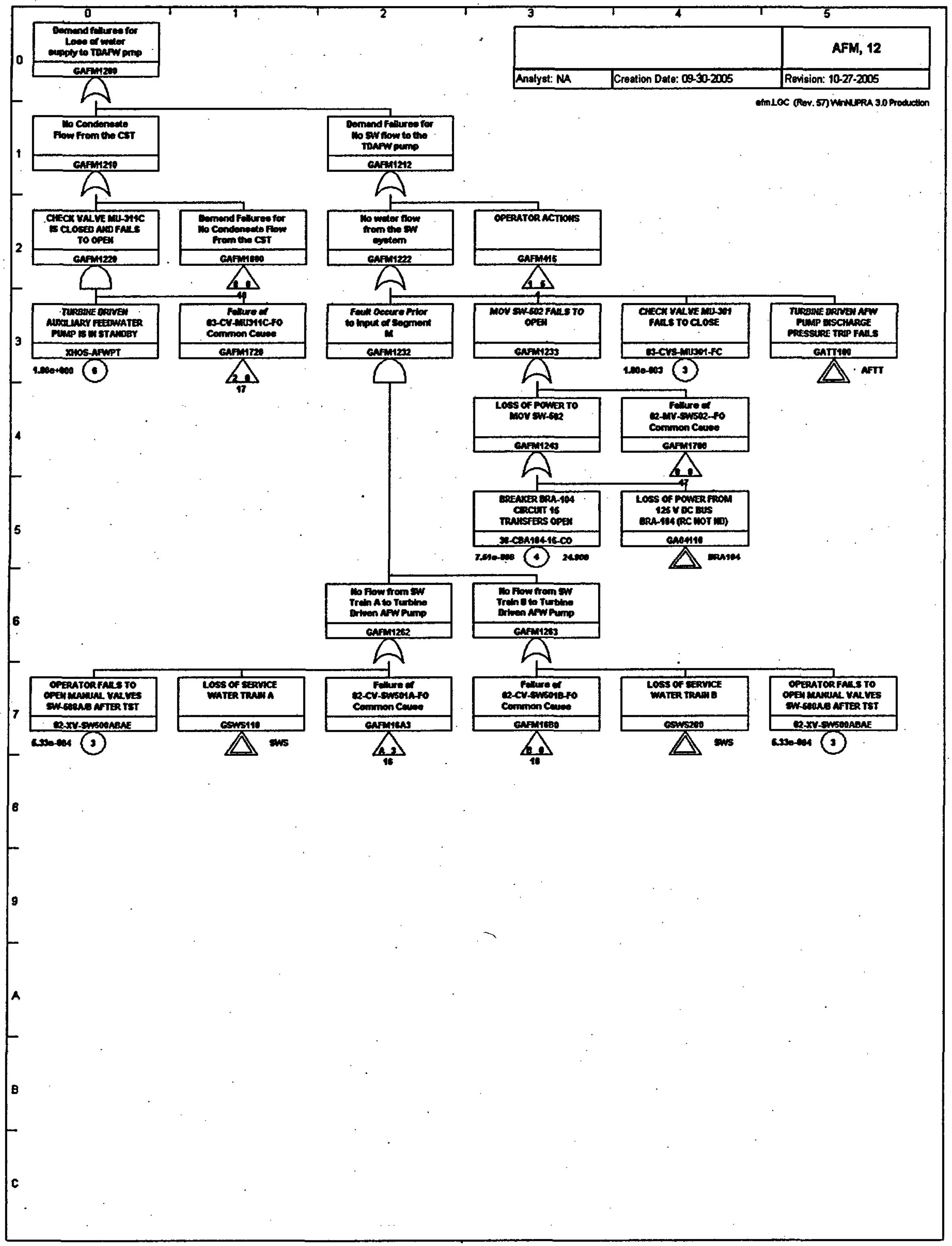

Figure 1 – Fault Tree AFM.LGC (continued)

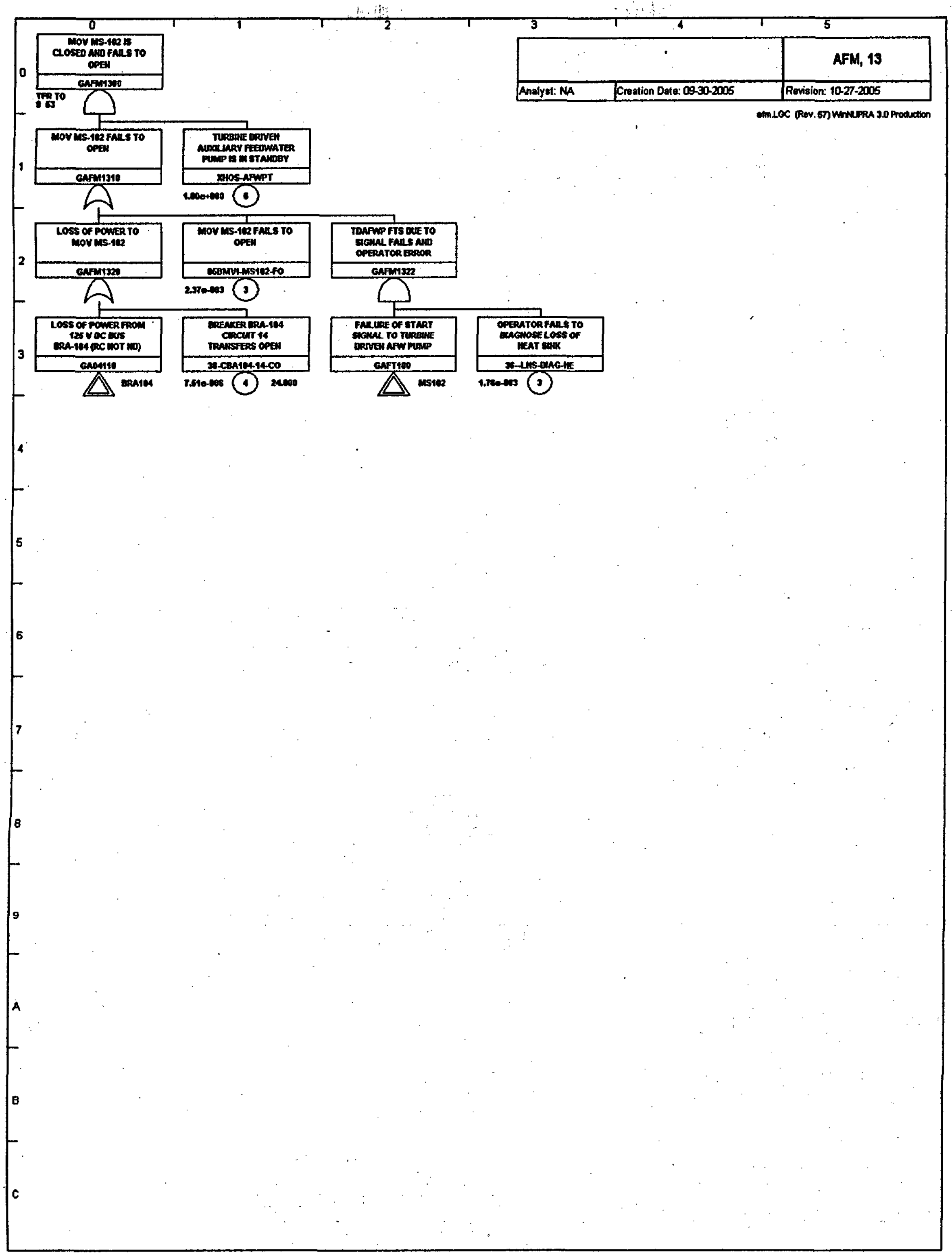

Figure 1 – Fault Tree AFM.LGC (continued)

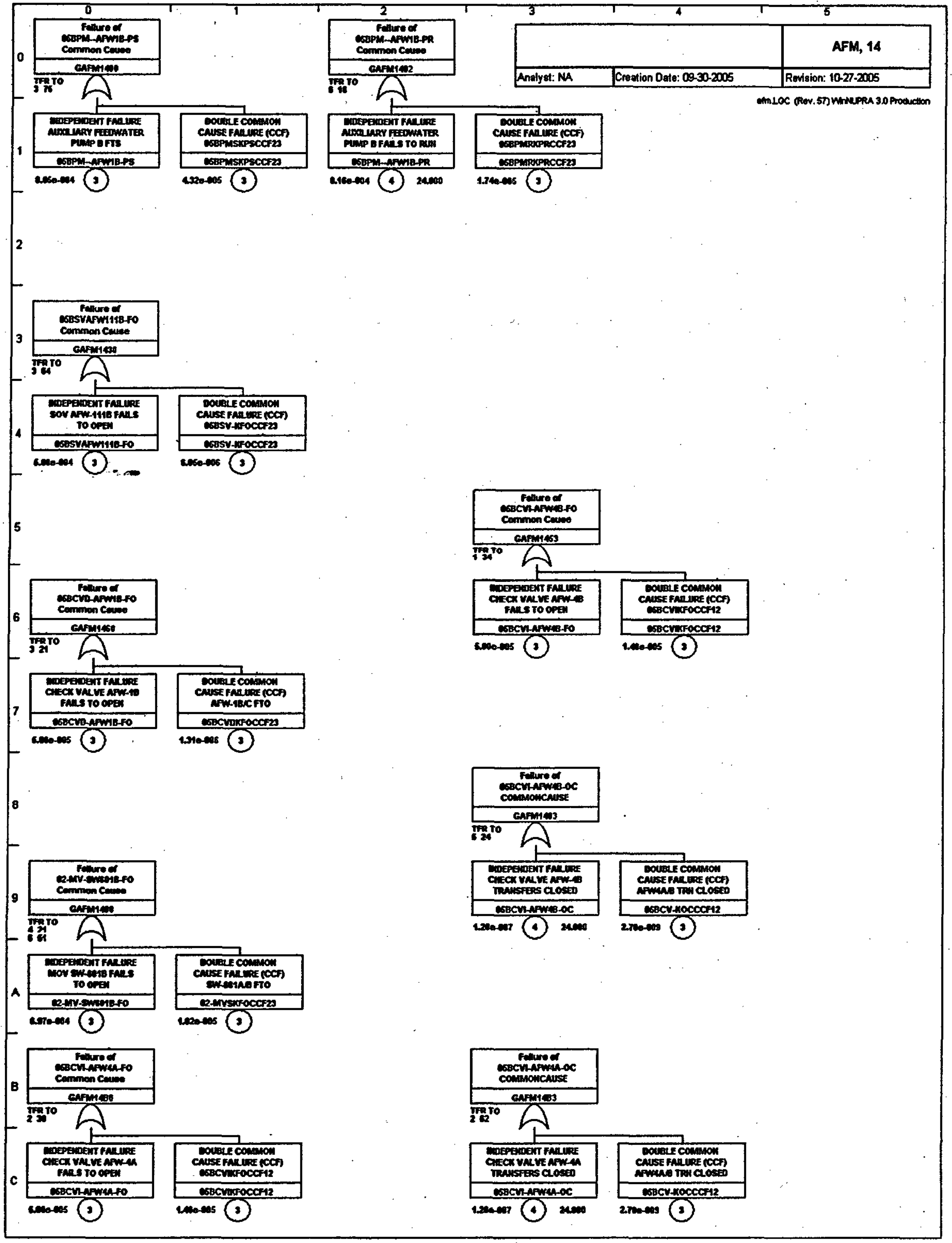

Figure 1 – Fault Tree AFM.LGC (continued)

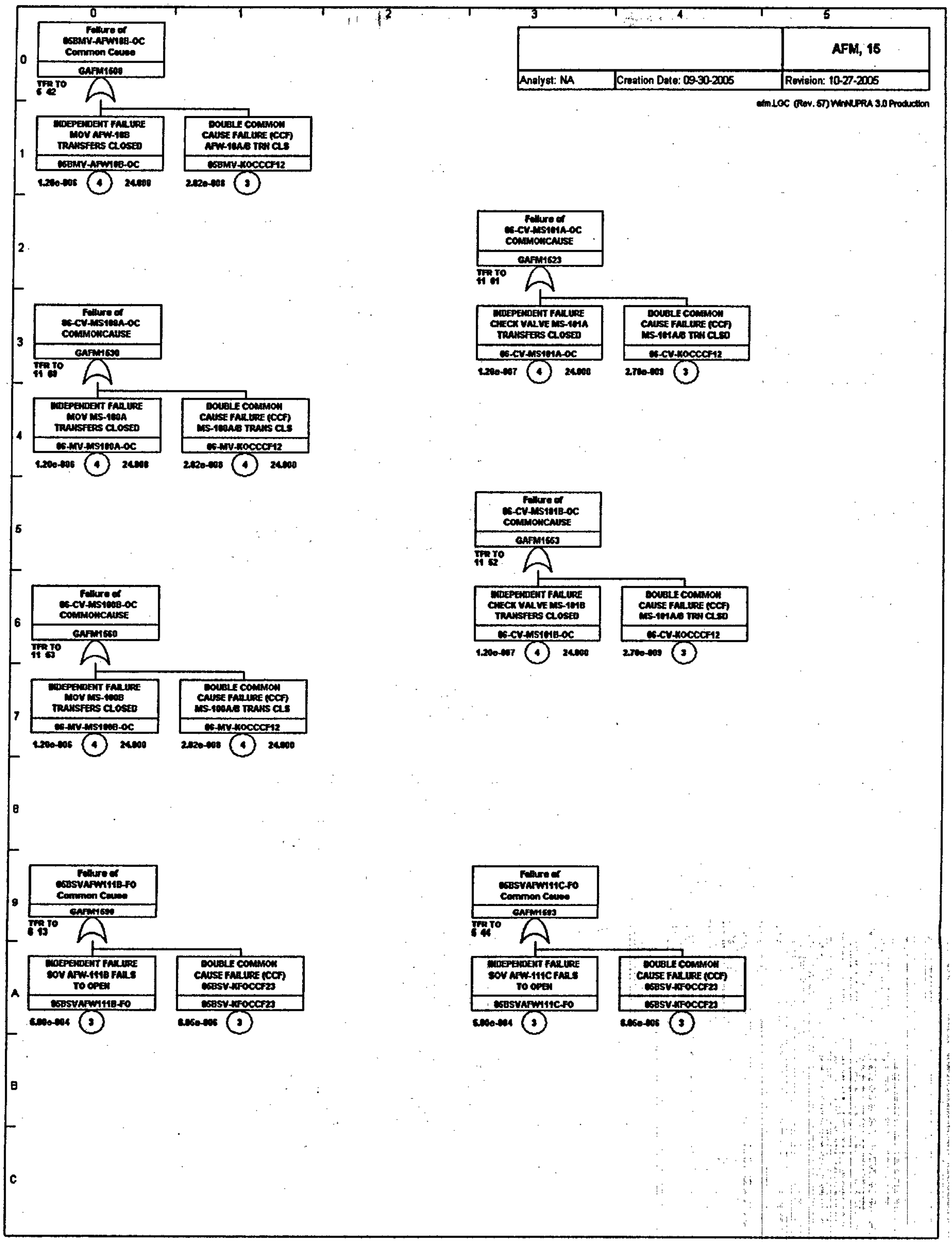

Figure 1 – Fault Tree AFM.LGC (continued)

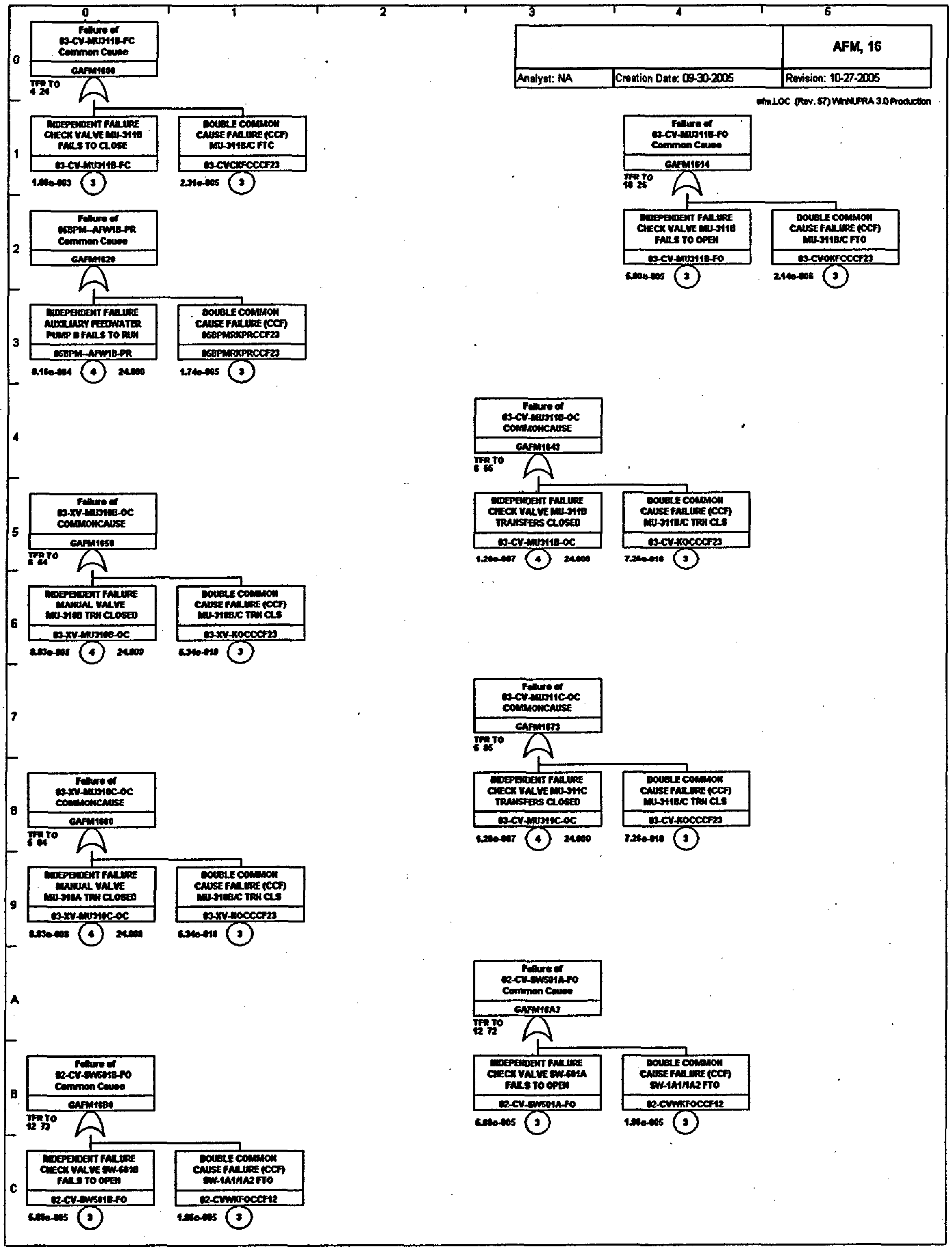

Figure 1 – Fault Tree AFM.LGC (continued)

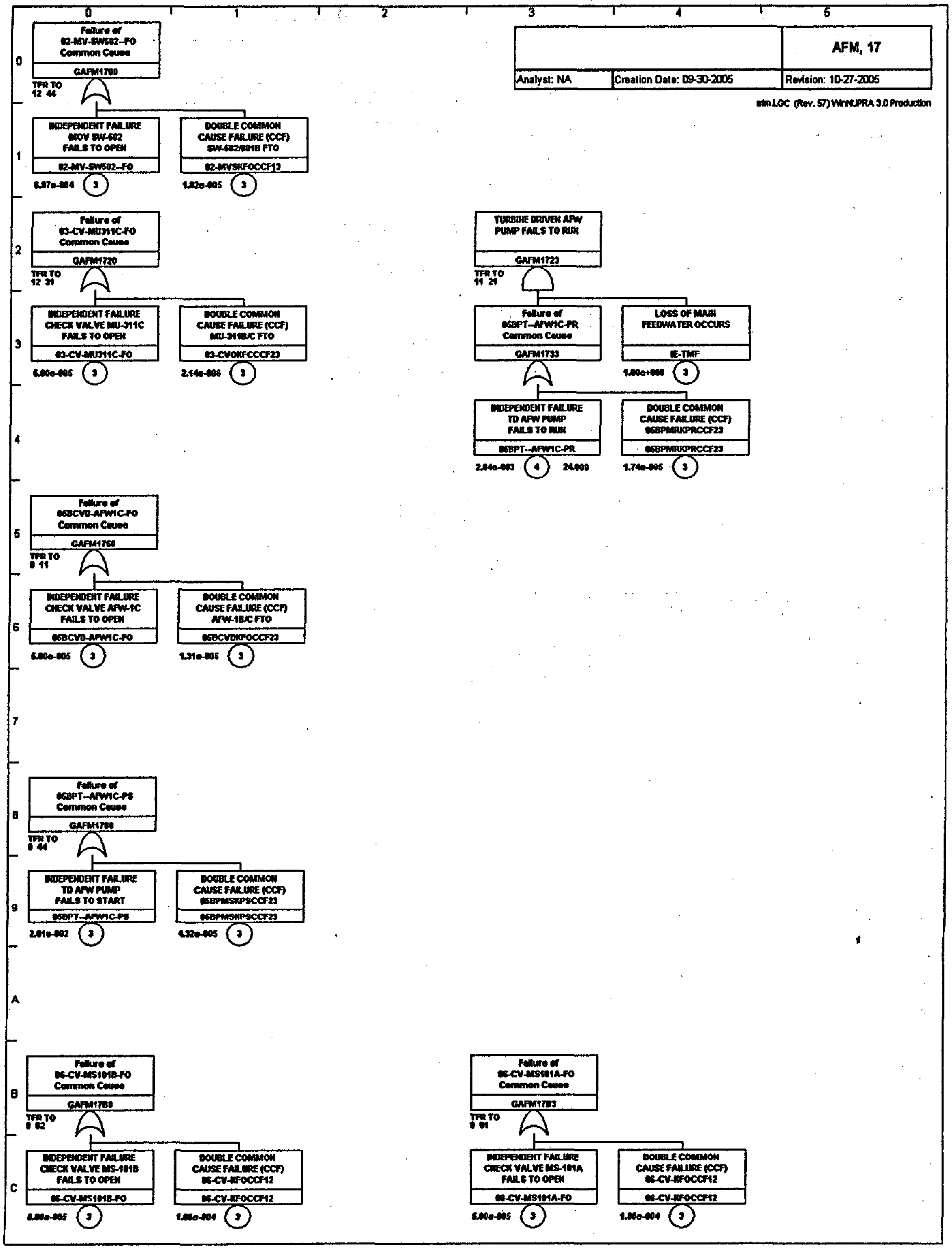

Figure 1 – Fault Tree AFM.LGC (continued)

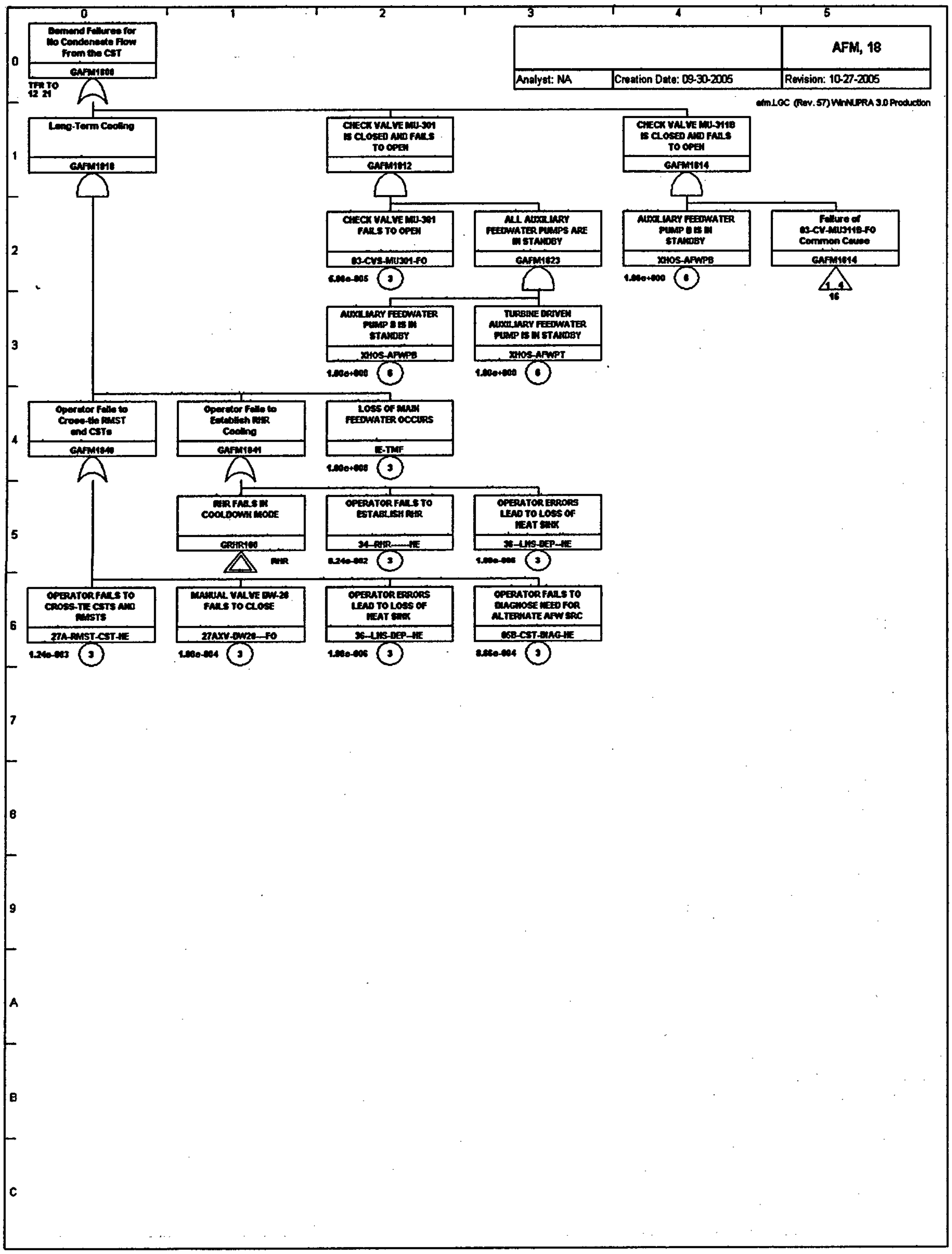

Figure 1 – Fault Tree AFM.LGC (continued)

INTERNAL FLOODING – Fault Tree Analysis for Turbine Building Floods

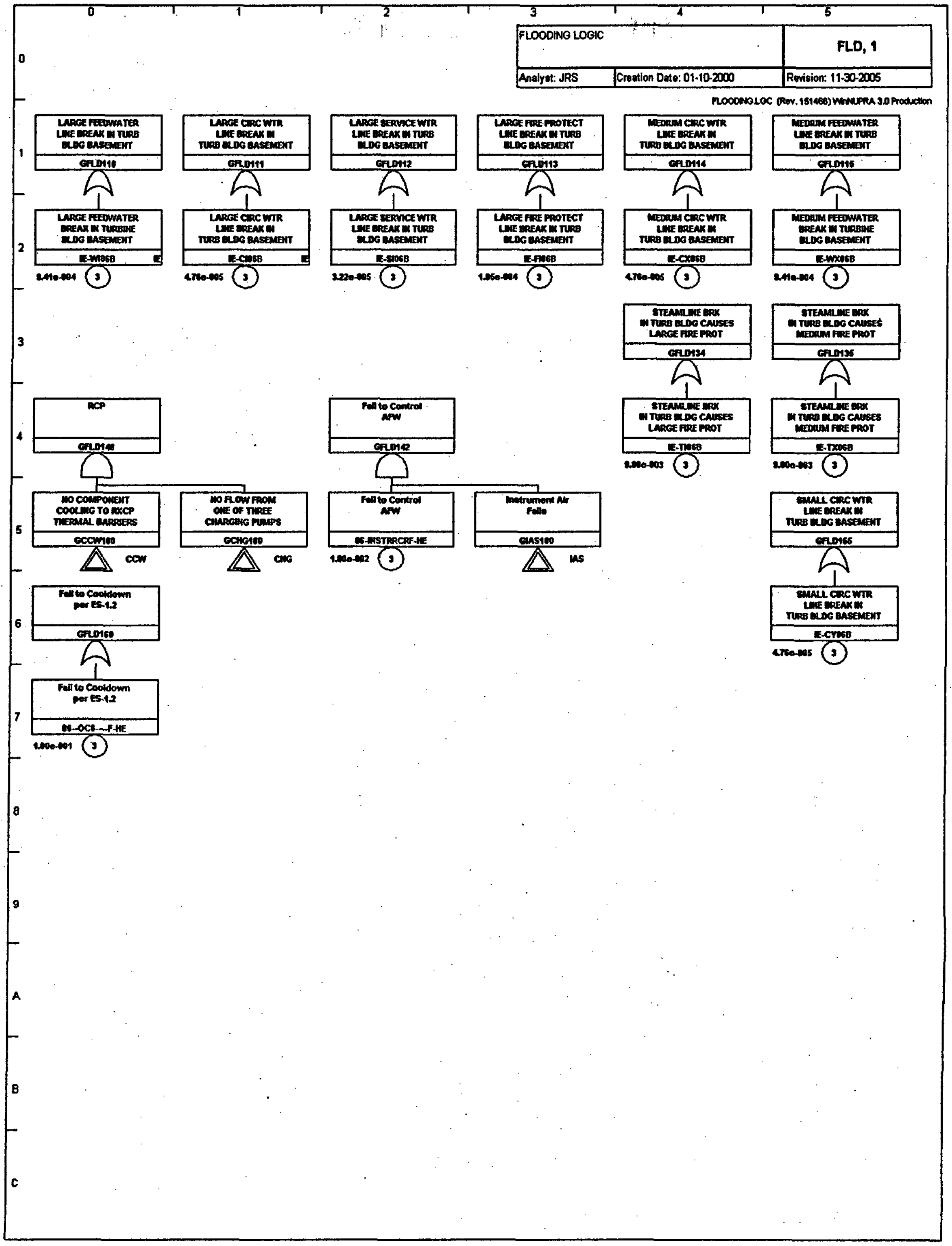

Figure 2 – Fault Tree FLOODING.LGC

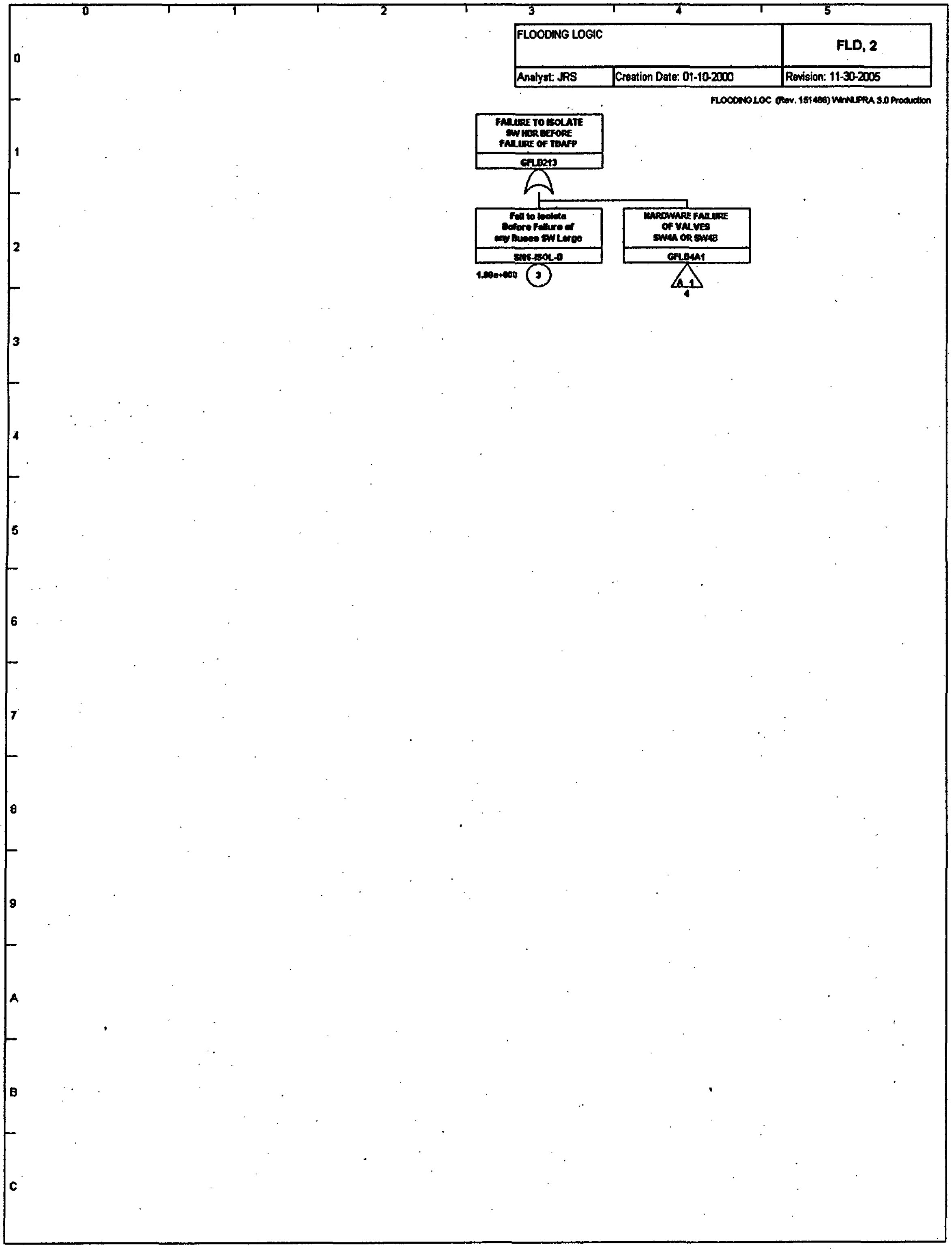

Figure 2 – Fault Tree FLOODING.LGC (continued)

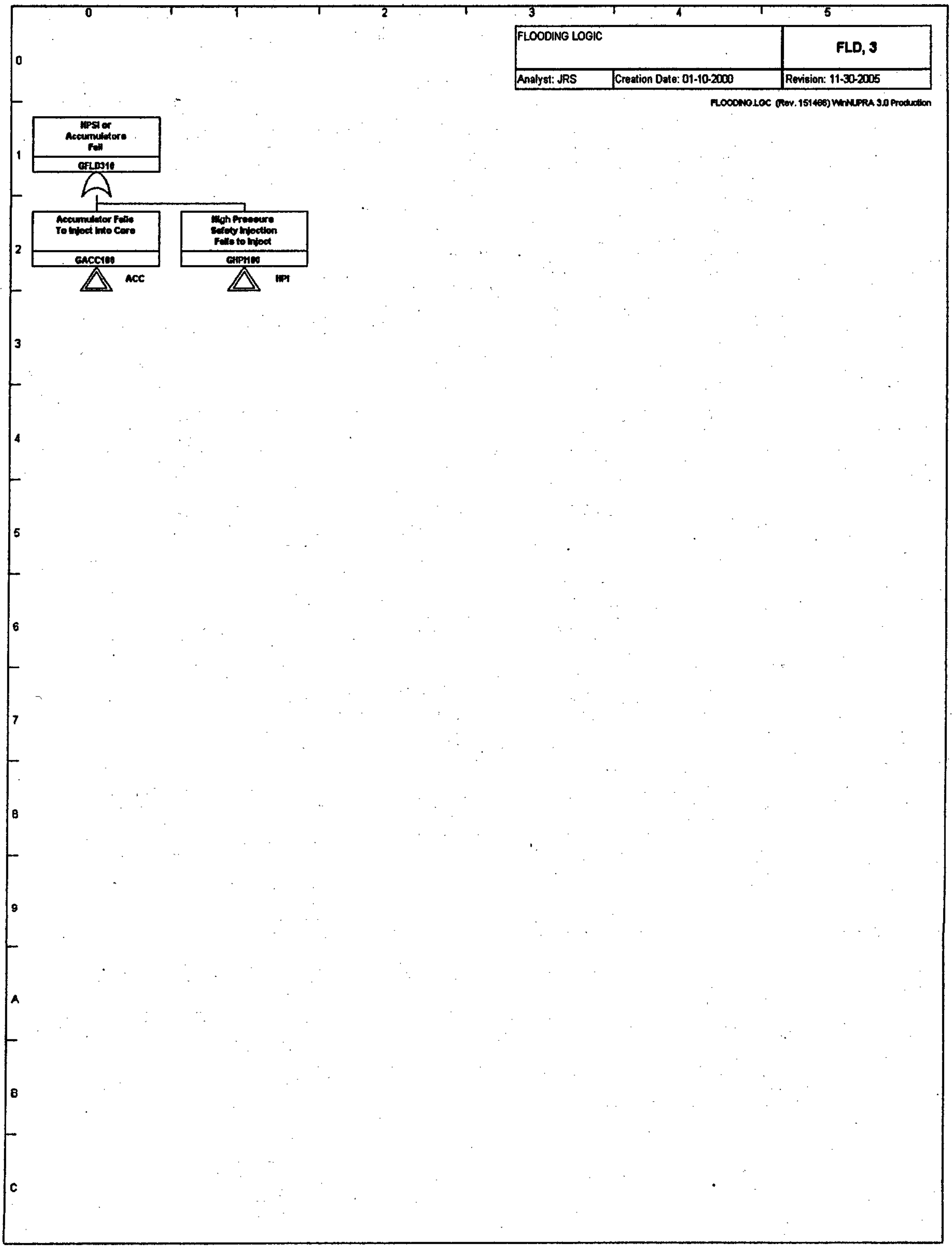

Figure 2 – Fault Tree FLOODING.LGC (continued)

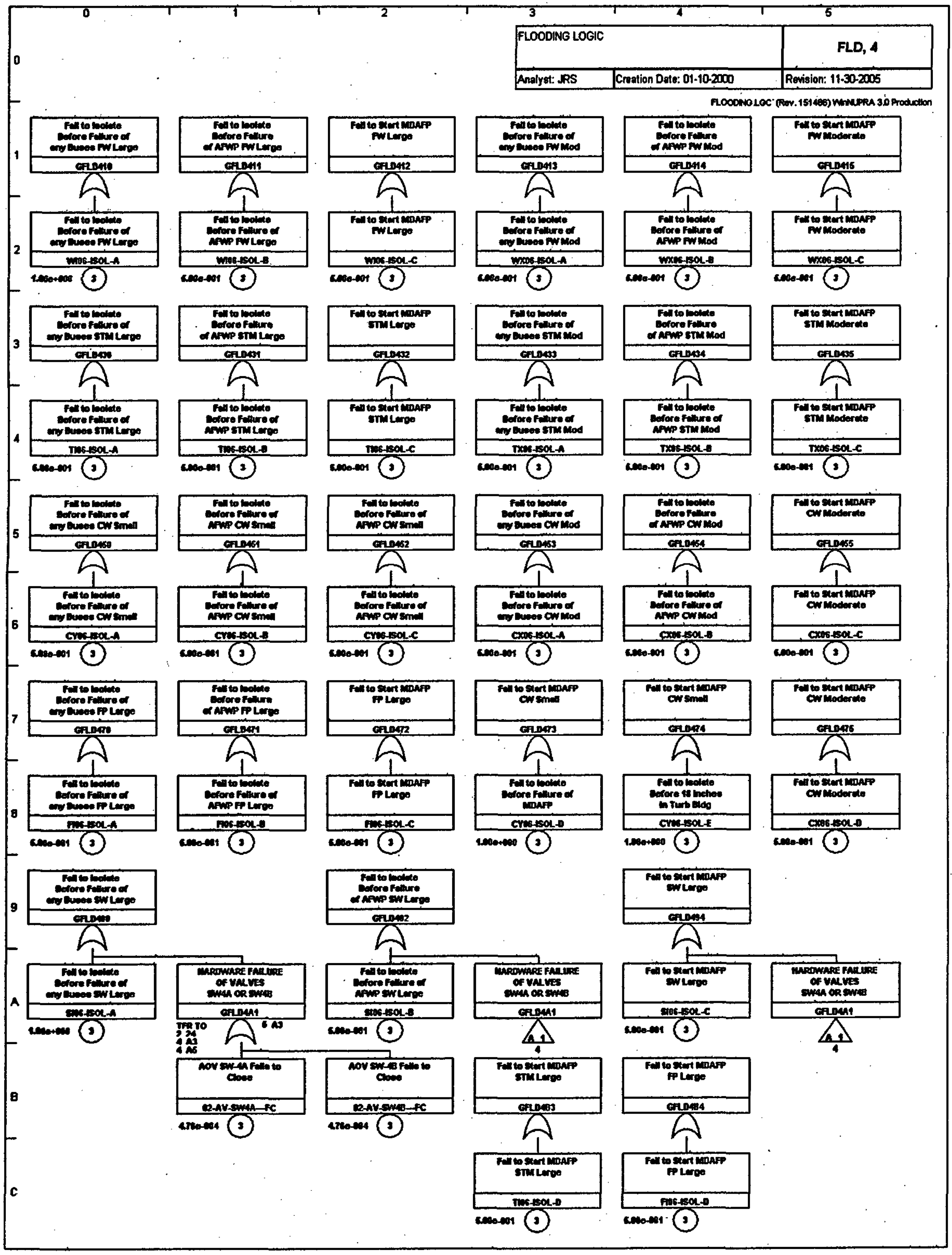

Figure 2 – Fault Tree FLOODING.LGC (continued)

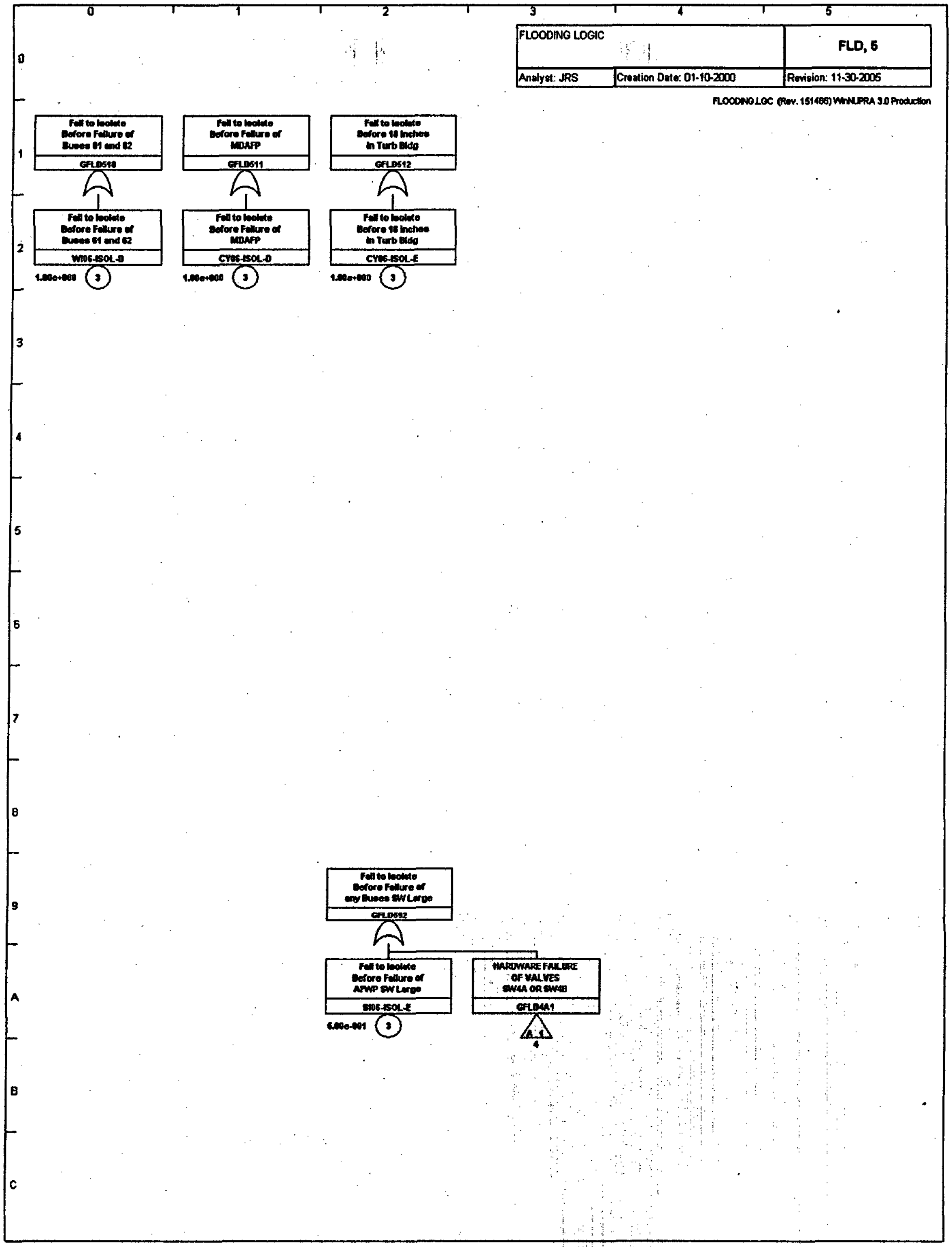

Figure 2 – Fault Tree FLOODING.LGC (continued)

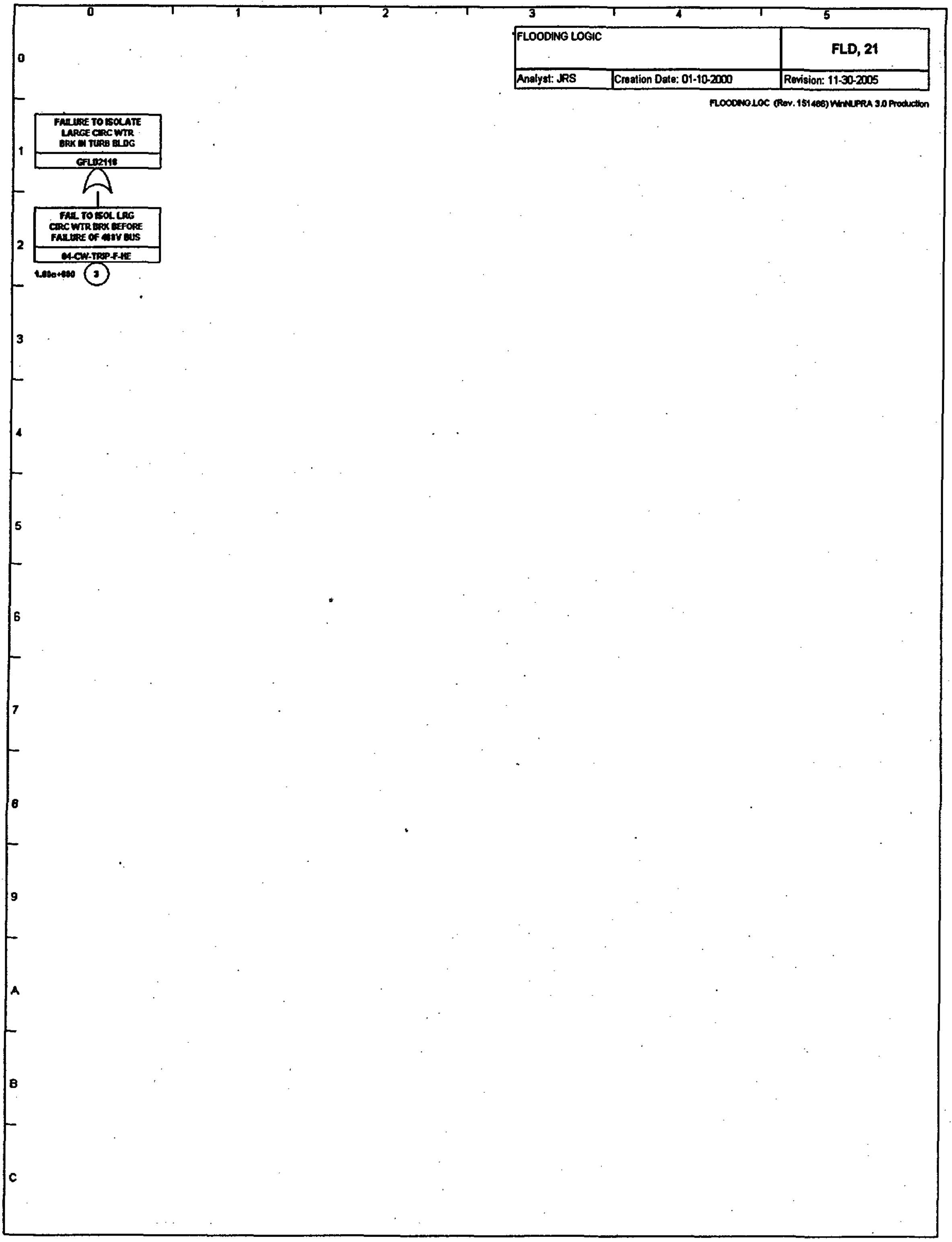

Figure 2 – Fault Tree FLOODING.LGC (continued)

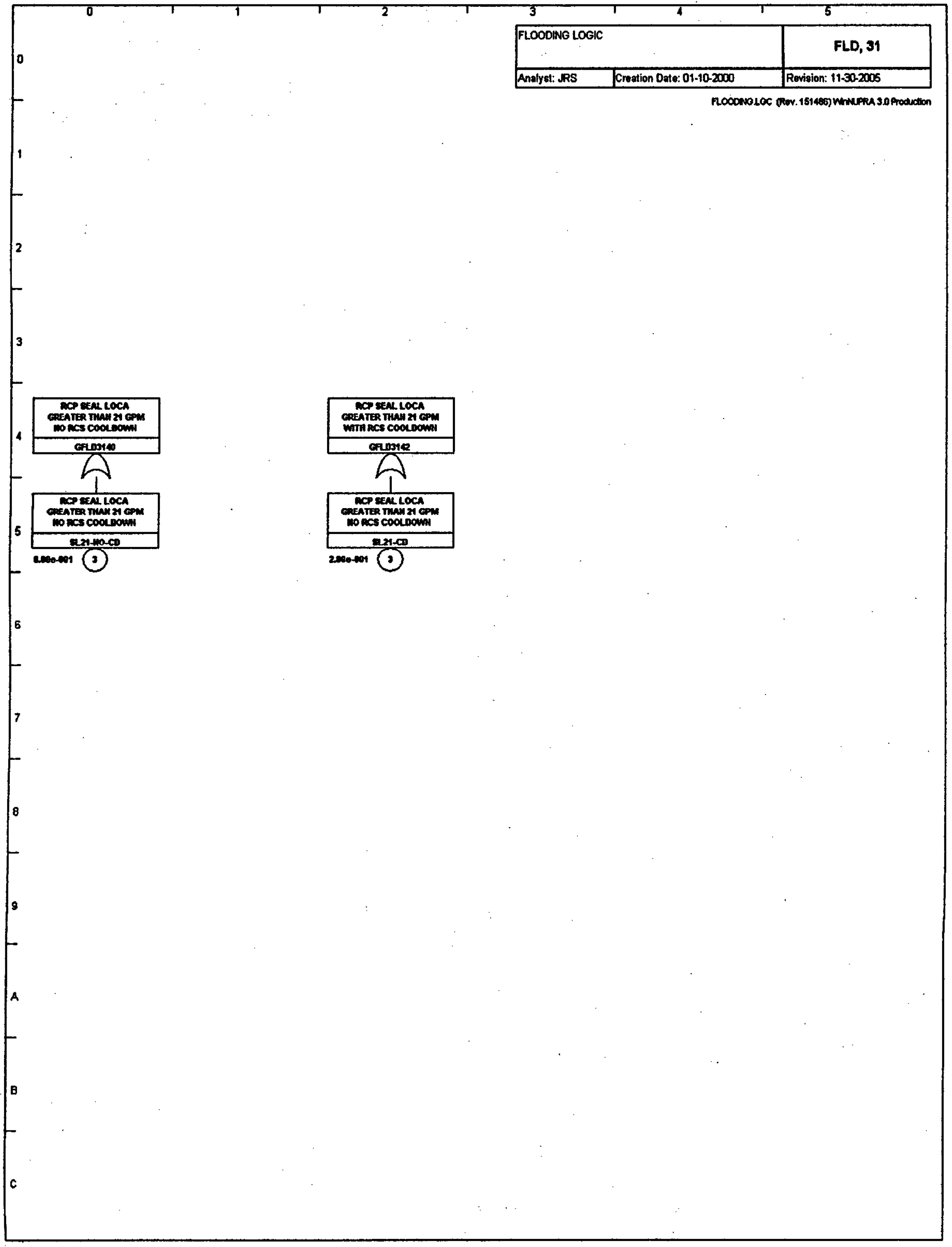

Figure 2 – Fault Tree FLOODING.LGC (continued)

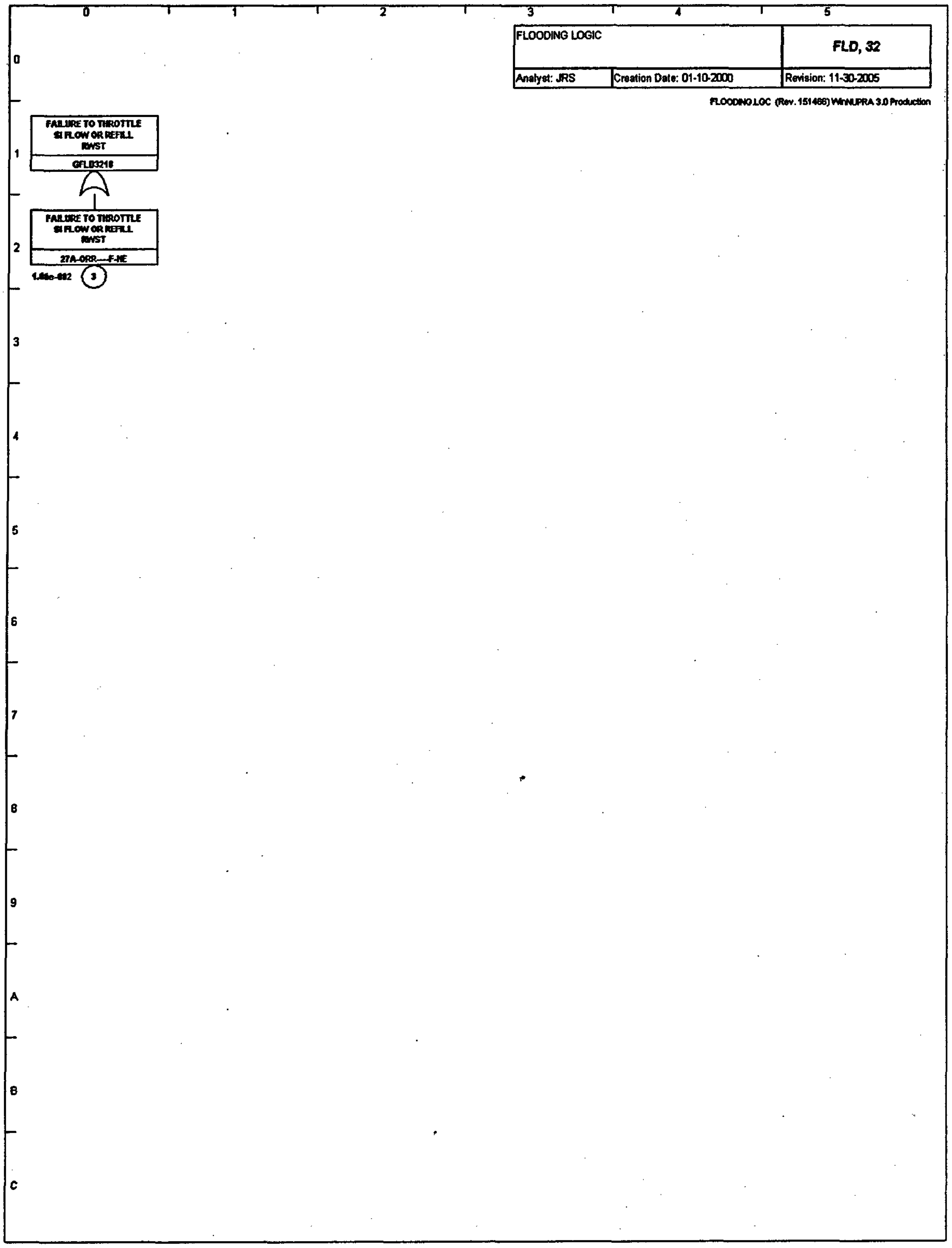

Figure 2 – Fault Tree FLOODING.LGC (continued)

INTERNAL FLOODING – Fault Tree Analysis for Turbine Building Floods

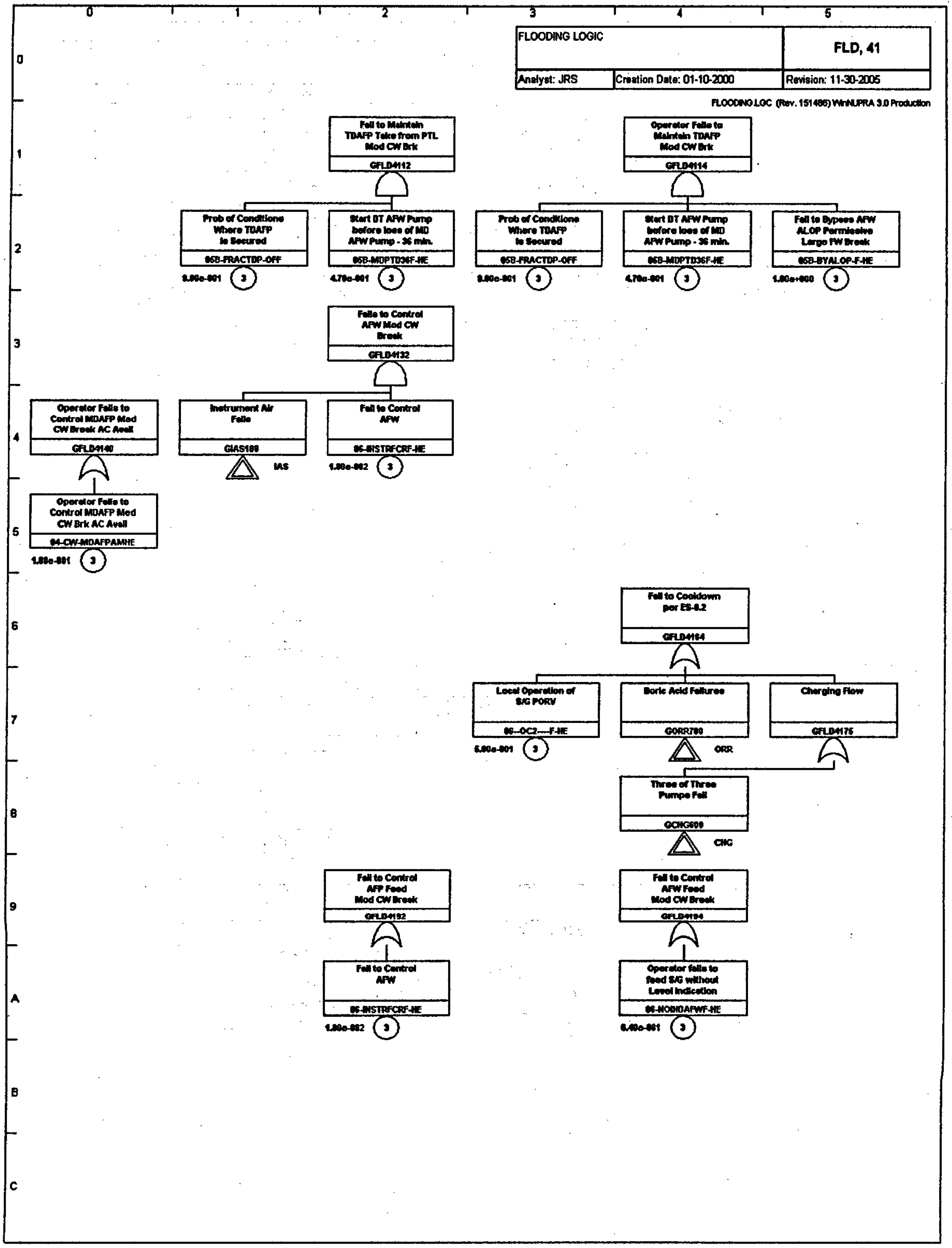

Figure 2 – Fault Tree FLOODING.LGC (continued)

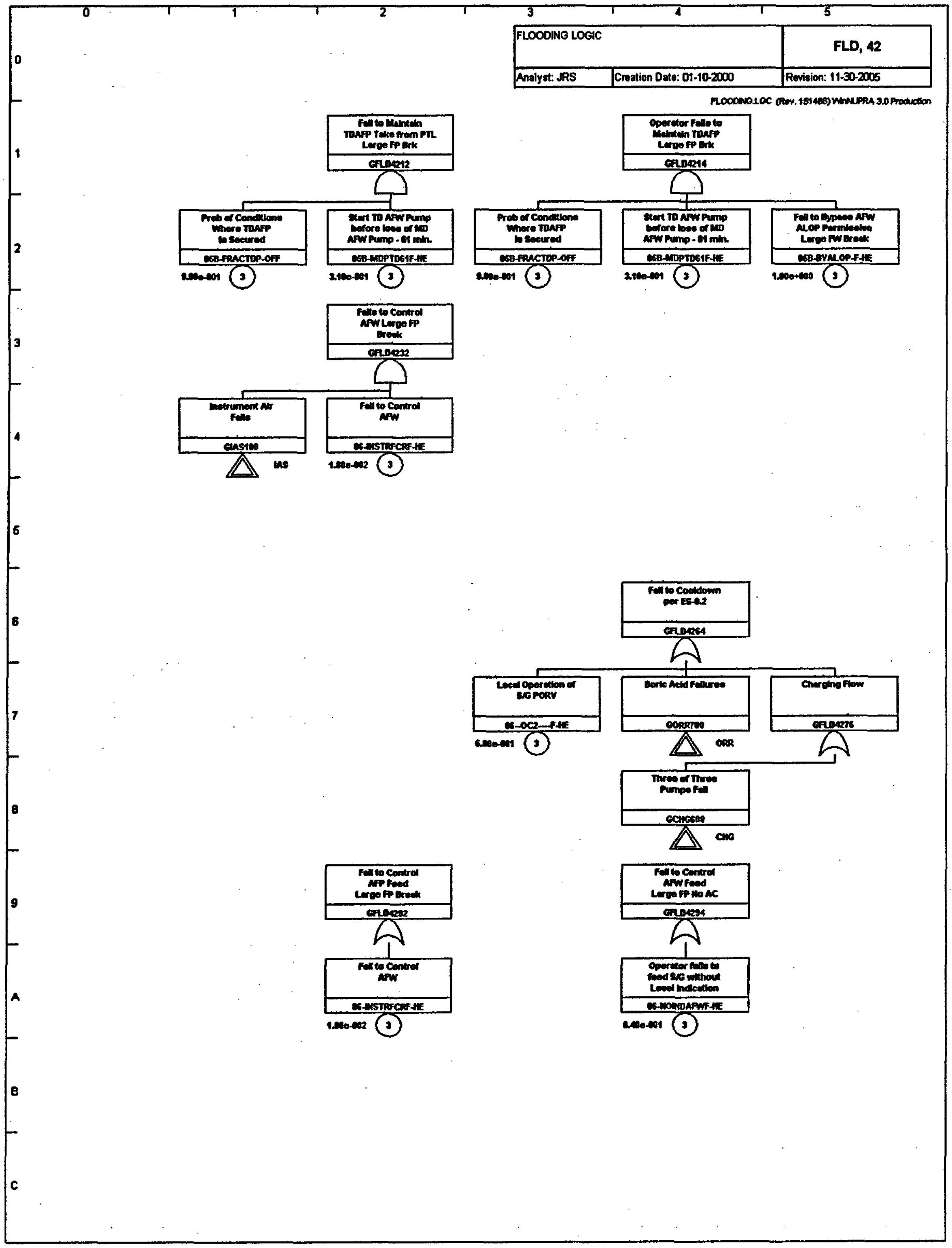

Figure 2 – Fault Tree FLOODING.LGC (continued)

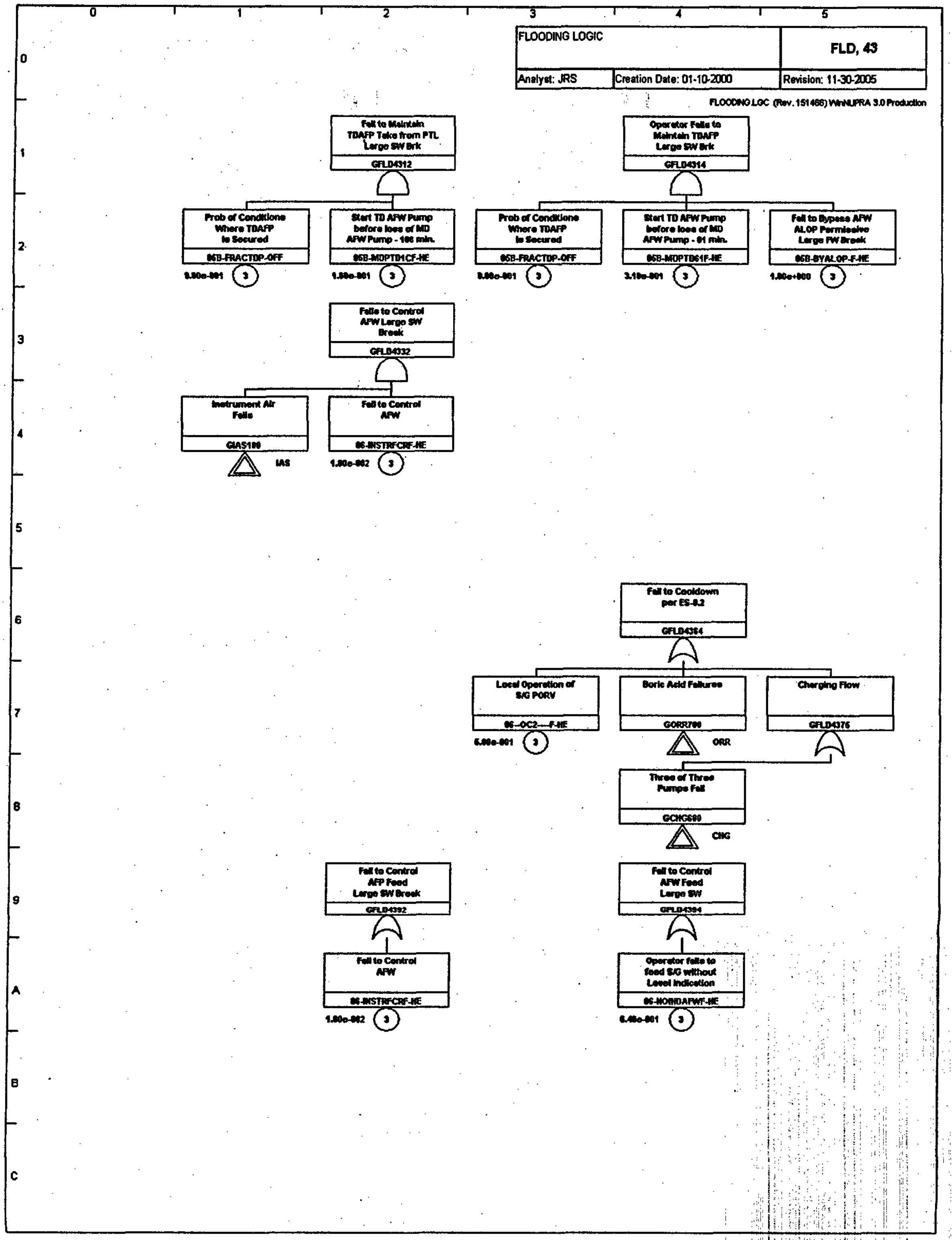

Figure 2 – Fault Tree FLOODING.LGC (continued)

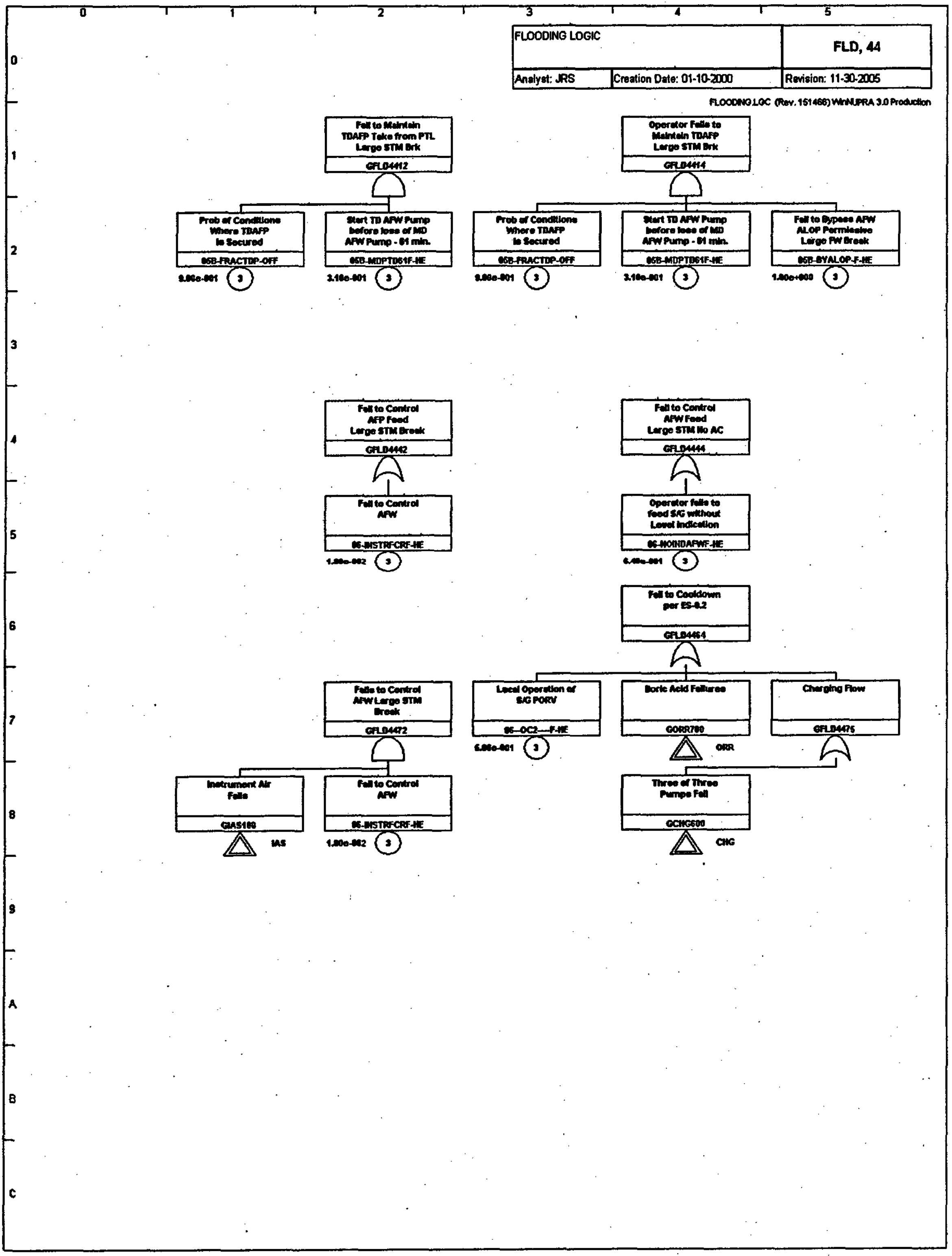

Figure 2 – Fault Tree FLOODING.LGC (continued)

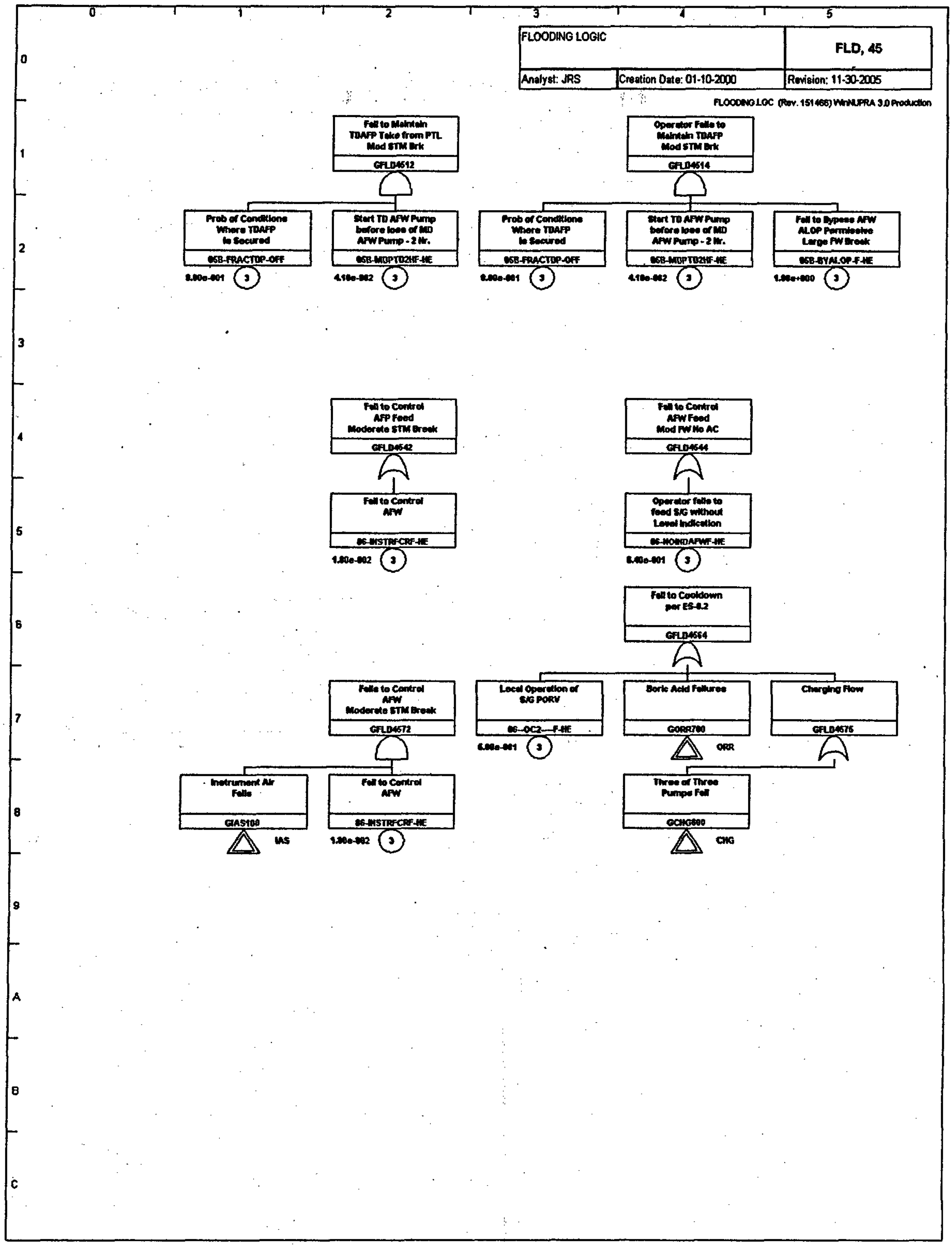

Figure 2 – Fault Tree FLOODING.LGC (continued)

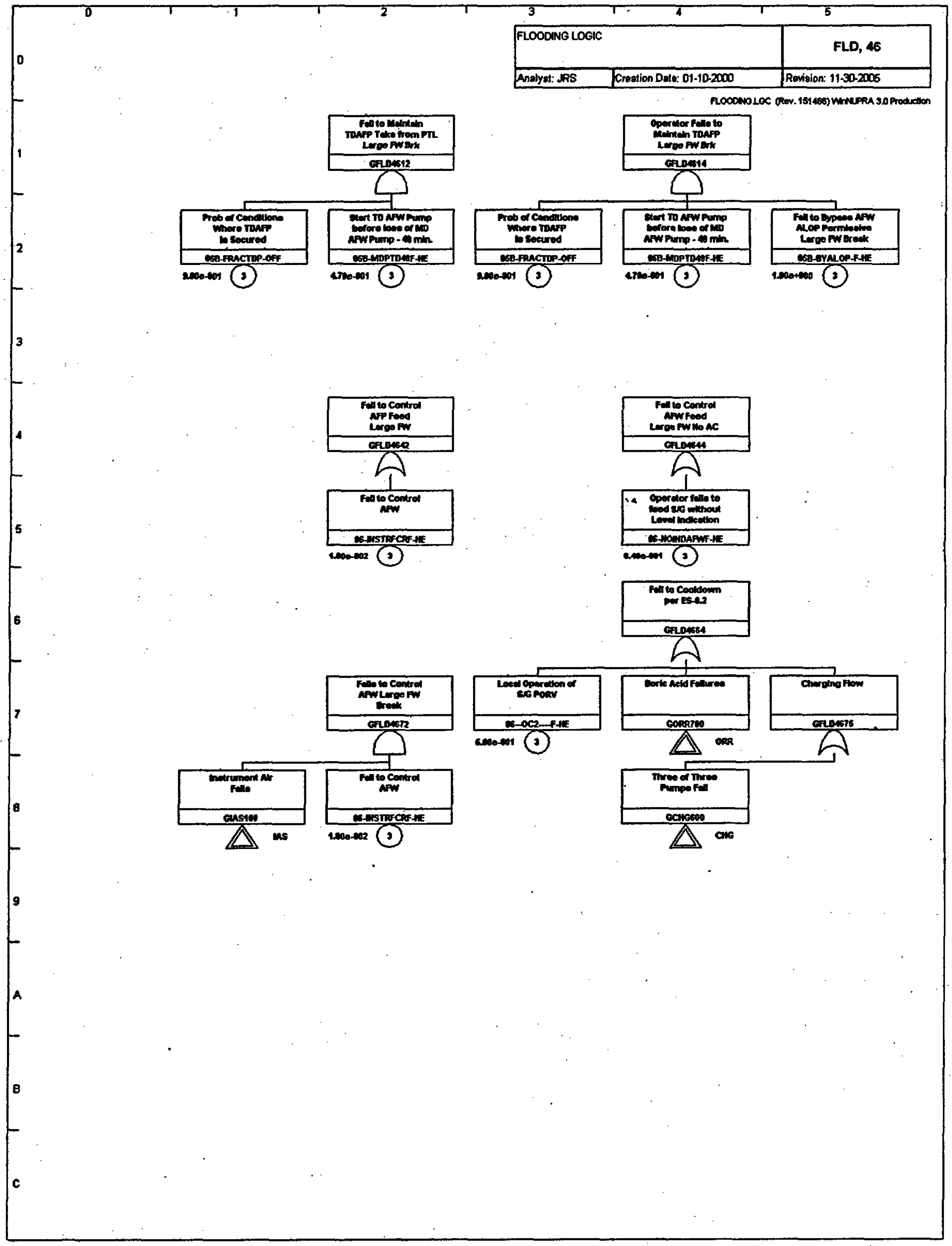

Figure 2 – Fault Tree FLOODING.LGC (continued)

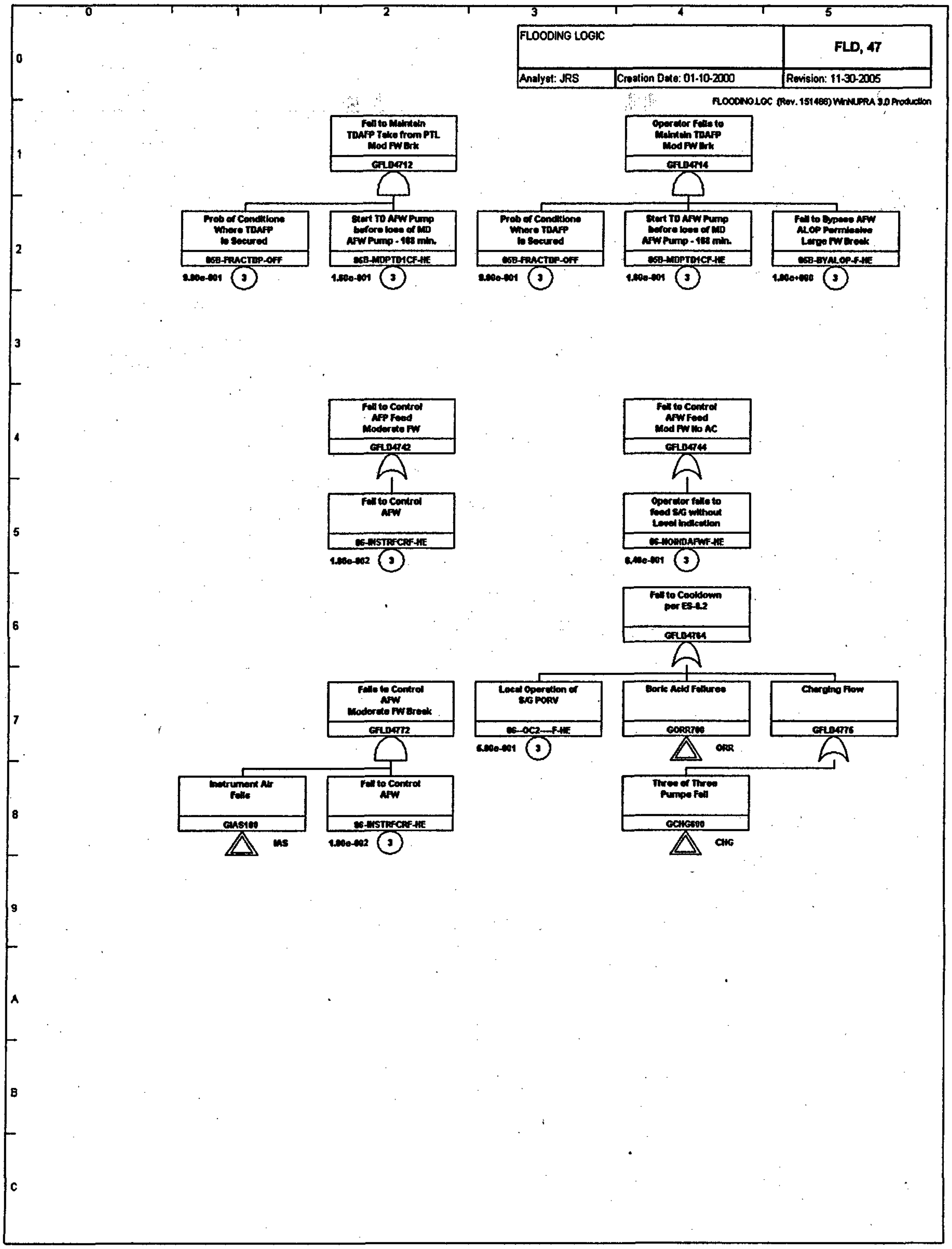

Figure 2 – Fault Tree FLOODING.LGC (continued)

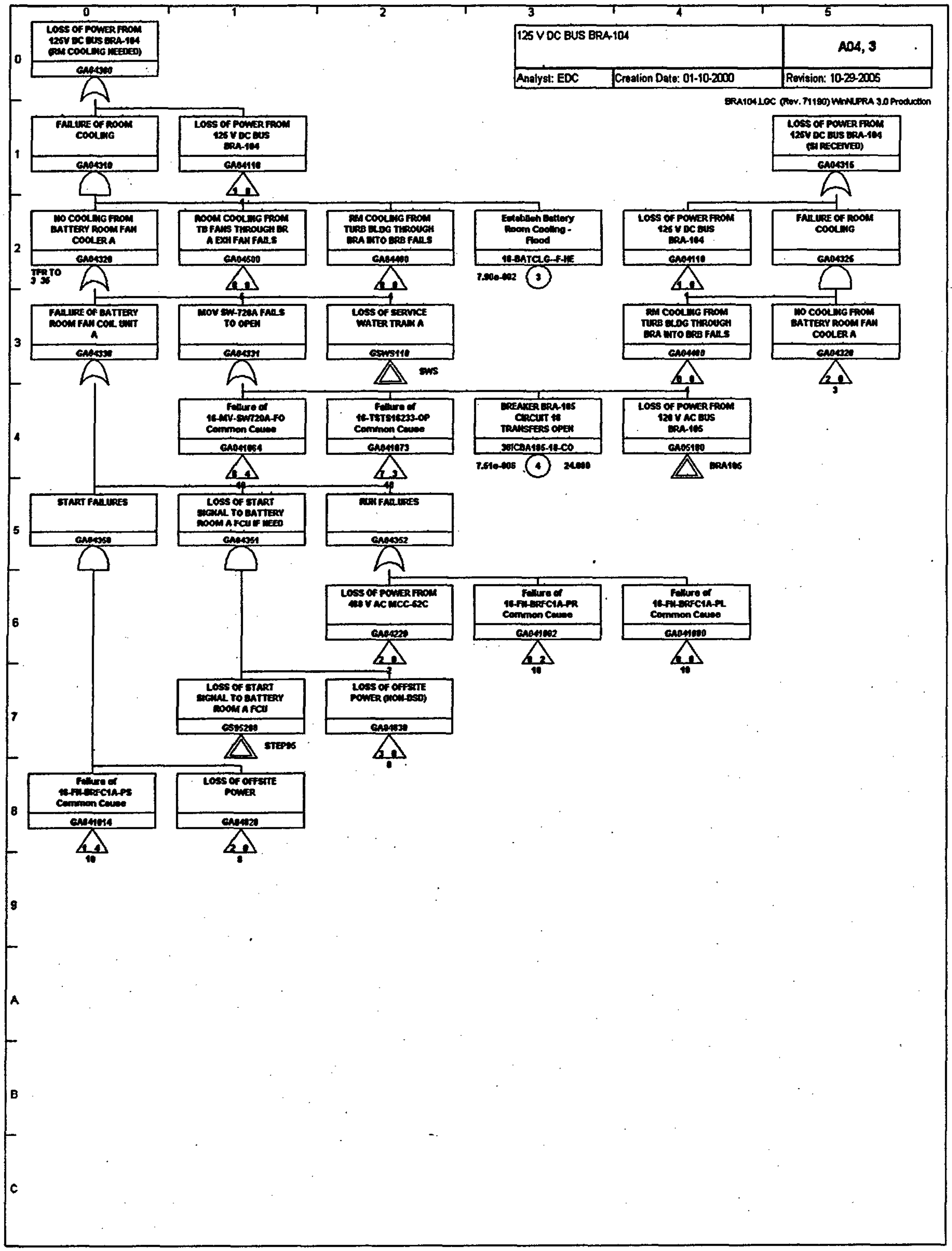

Figure 3 – Placement of New Basic Event 16-BATCLG--F-HE