Tennessee Valley Authority, Post Office Box 2000, Spring City, TN 37381-2000

February 9, 2010

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U.S. Nuclear Regulatory Commission
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Watts Bar Nuclear Plant, Unit 2
NRC Docket No. 50-391

Subject: WATTS BAR NUCLEAR PLANT (WBN) UNIT 2 – PROBABILISTIC RISK ASSESSMENT INDIVIDUAL PLANT EXAMINATION SUMMARY REPORT

Reference: 1. TVA letter dated March 20, 2008, "Watts Bar Nuclear Plant (WBN) – Unit 2 – Generic Communications for Unit 2 – Restructured Tables"

The purpose of this letter is to provide the results of the Individual Plant Evaluation (IPE) for WBN Unit 2. Enclosure 1 provides the IPE summary report. In Reference 1, TVA committed to complete the evaluation for WBN Unit 2. The IPE was performed in accordance with the applicable portions of ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Applications," and Regulatory Guide 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities."

The total core damage frequency for WBN Unit 2 is 3.28×10^{-5} per reactor-year. The large early release frequency for WBN Unit 2 is 2.62×10^{-6} per reactor-year. Each of these values is substantially below the NRC guideline values of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

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Enclosure 2 provides a list of commitments made in this submittal. I declare under penalty of perjury that the foregoing is true and correct. Executed on the 9th day of February, 2010.

If you have any questions, please contact me at (423) 365-2351.

Sincerely,

M. Bajestani

Masoud Bajestani
Watts Bar Unit 2 Vice President

Enclosures:

1. IPE Summary Report
2. List of commitments

cc (Enclosures):

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Enclosure 1

IPE Summary Report

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT UNIT 1
Probabilistic Risk Assessment

WBN Unit 2 IPE Summary Report

February 2010

Table of Contents

1. INTRODUCTION

This report is the IPE summary report for Revision 0 to the Unit 2 Probabilistic Risk Assessment (PRA) for Watts Bar Nuclear Plant (WBN). The WBN PRA model has been developed as a dual unit model using EPRI's CAFTA software and both units are represented under a single top event in the fault tree. The WBN Unit 1 PRA was previously developed using Riskman® and was converted to CAFTA as a part of the process for developing the dual unit PRA model and the model was developed as a full Level 2 model. The Level 1 model for Internal Events also includes an internal flooding analysis. This report summarizes the key portions of the Unit 2 PRA model and provides the calculated core damage frequency (CDF) and Large Early Release (LERF) frequency.

The Unit 2 PRA model was developed based on the as-built, as-operated configuration of Unit 1 with a freeze date of April 1, 2008. The Unit 2 design had not been finalized as of this date and this PRA assumed that the Unit 2 as-built, as-operated configuration will be the same as Unit 1. Prior to Unit 2 start-up, it will be confirmed that the Unit 2 PRA model matches the as-built, as-operated plant. One future modification was incorporated into the Unit 2 model and identified as an assumption. The present analysis assumes that that the carbon steel used for the Raw Cooling Water (RCW) and High Pressure Fire Protection (HPFP) piping in Auxiliary Building rooms Elevation 757.0-A2, Elevation 757.0-A5, Elevation 757.0-A9, Elevation 757.0-A17, Elevation 757.0-A21, Elevation 757.0-A24, Elevation 772.0-A7, Elevation 772.0-A8, Elevation 772.0-A9 and Elevation 772.0-A10 piping will be replaced with stainless steel piping.

Systems shared between the two units such as electric power, component cooling, essential raw cooling water, and plant compressed air systems were modeled to support dual unit operation. The WBN Unit 1 and Unit 2 models share a single plant-specific database and failure rates, unavailabilities and initiating event frequencies have been updated to account for Unit 1 plant-specific data through April 1, 2008.

The success criteria results are based on a MAAP 4.0.5 bounding case for Watts Bar Units 1 and 2. The bounding cases use the steam generator (SG) that is installed in Unit 2 (Original Steam Generator – model D3) and a thermal power rating of 3459 MWt .

The total core damage frequency computed for Watts Bar Nuclear Plant Unit 1 is 3.69E-05 per reactor-year and Unit 2 CDF is 3.28E-05 per reactor-year. These values were quantified using a truncation limit of 1.0E-12.

The large early release frequency (LERF) computed for Watts Bar Nuclear Plant Unit 1 is 2.69E-06 per reactor-year and Unit 2 is 2.62E-06 per reactor-year. These values were quantified using a truncation limit of 1.0E-12.

Each of these values is substantially below the NRC guideline values of 10^{-4} for CDF and 10^{-5} for LERF.

2. BACKGROUND AND OBJECTIVES

This report documents the work performed by the Tennessee Valley Authority (TVA) in accordance with the U.S. Nuclear Regulatory Commission (NRC) Generic letter 88-20 which requested each utility to perform an individual plant examination (IPE). The PRA performed to meet the requirements of the IPE has also been developed to meet the requirements of ASME-RA-Sb-2005 (Reference 5) and Regulatory Guide 1.200 (Reference 6). The model development also consisted of a revision of the Unit 1 PRA which converted the model from a Riskman® large event tree model to a CAFTA large fault tree model and updated the model based on the current plant design and operation. System fault trees and the integrated logic model were developed using CAFTA. The previous systemic event trees were replaced by functional event trees which are also based on current plant operating and emergency procedures. The internal flooding analysis was upgraded in accordance with NUREG-6850. The LERF analysis was performed in accordance with current industry guidance. The human error probability evaluation was upgraded using the EPRI HRA Calculator tool and the generic prior data is now based on NUREG-6928. All of these changes are categorized as model upgrades per the ASME PRA standard (Reference 5) which require a new peer review.

The Watts Bar Nuclear Plant is located in Rhea County, Tennessee, approximately 50 miles northeast of Chattanooga and 31 miles north-northeast of TVA's Sequoyah Nuclear Plant. The plant is on the west shore of Chickamauga Lake on the Tennessee River. The plant consists of two units, Unit 1 with a rated Reactor power level of 3,459 MWt which accounts for the new steam generators and Unit 2 which is currently under construction with a rated Reactor power level of 3,411 MWt.

Both Unit 1 and 2 are four-loop Pressurized Water Reactors (PWR) Nuclear Steam Supply System (NSSS) furnished by Westinghouse Electric Corporation. Major structures at Watts Bar include two reactor buildings with ice condenser containments, a turbine building, an auxiliary building, a control building, a service and office building, two diesel buildings, an intake pumping station, and two natural draft cooling towers.

A detailed description on the plant site, facilities, and safety criteria is documented in the Watts Bar Final Safety Analysis Report.

2.1. Development of the WBN Unit 2 PRA Model

The Individual Plant Examination (IPE) submittal document for Unit 1 was completed in 1992. The Unit 1 model has undergone several revisions since the IPE was completed. The first update (Revision 1) to the IPE was performed in 1995 to incorporate numerous plant design changes, procedure upgrades, and training enhancements that had either been made since the initial IPE or had not been modeled in the initial effort. Revision 1 included the latest changes and included less conservatism as compared to the original IPE to represent a more realistic model. This effort represented a comprehensive review and update of the Level 1 Probabilistic Risk Assessment (PRA).

The second update (Revision 2) to the IPE was performed in 1997 to incorporate changes made to the plant design as a result of the Severe Accident Mitigation Design Alternatives (SAMDA) performed for WBN Unit 1.

The third update to the IPE was performed to incorporate data collected by the Maintenance Rule program and review the plant model against plant operation to more accurately reflect actual plant conditions. This update was also used in the submittal of the WBN TS change to request an extension of the diesel generator Completion Time from 72 hours to 14 days. This update was reviewed by an industry Peer Certification team. Revision 4 to the PSA was performed to modify the model in order to supply the information required by the Mitigating Systems Performance Indicator (MSPI) Program. This revision of the model resolved WOG PEER review Findings and Observations (F&O's) that were determined to impact MSPI; updated the model to current plant design; updated the initiating event data based on the latest plant-specific and industry data; incorporated the latest maintenance rule data into the database; and incorporated comments on the systems analyses by the WBN system engineers. Also, changes were made to the model to permit calculation of Fussel-Vessely importance values of certain maintenance alignments in support of the MSPI program.

The insights developed from the Unit 1 model were incorporated into the development of the Unit 2 PRA model. A PWROG peer review of the dual unit model was conducted from November 9, to November 13, 2009. Of the 326 supporting requirements covered in the ASME standard the peer review team judged 9 SRs as not applicable to WBN, 272 SRs were judged as met Category I/II or greater, 19 met Category I and 26 SRs were judged as not met. In addition, 112 Findings and Observations (F&O's) were identified. Disposition of these F&Os is currently in progress. The F&Os and their resolution status are included as Appendix A to this report. The overall conclusions of the peer review team regarding the WBN PRA are as follows:

- The overall model structure is robust and well developed, but needs refinement
- Documentation is very thorough, detailed , and well organized such that comparison with the standard is facilitated
- The processes and tools utilized for the WBN PRA are at the state of the art technology and generally consistent with Capability Category II
- The PRA maintenance and update program includes all necessary processes and does a very good job of tracking pending changes, and
- The qualitative assessment of sources of modeling uncertainty for the Level 1 model is very comprehensive and well documented to support future applications.

2.2. Summary of Objectives for Unit 2 PRA

Consistent with the original IPE, this PRA has been performed in accordance with the U.S. Nuclear Regulatory Commission (NRC) Generic Letter No. 88-20 (Reference 4), which requested each utility to perform an individual plant examination in order to:

(1) develop an appreciation of severe accident behavior

(2) understand the most likely severe accident sequences that could occur at its plant

(3) gain a more quantitative understanding of the overall frequencies of core damage and fission product releases, and

(4) if necessary, reduce the overall frequencies of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents

In addition to meeting the IPE objectives set forth in Generic Letter 88-20, TVA's overall objectives for the IPE update were to:

- Develop a plant-specific PRA model for both units at Watts Bar Nuclear plant (WBN) based on current plant design and using EPRI's PRA software programs such as CAFTA.
- Develop and apply databases using the latest WBN Unit 1 plant-specific and industry data for initiating events, component failures rates, maintenance unavailabilities, common cause failure parameters and human error rates
- Develop point estimate and uncertainty distribution results and identify and understand the key sources of uncertainty
- Determine the underlying risk controlling factors in support of the evaluation of potential safety improvements

The scope of the update included:

(1) the Level 1 PRA in which the accident sequences are developed sufficiently to define and quantify core damage event sequences and included an update to the thermal hydraulic

analyses for both Level 1 and Level 2.

(2) The Level 2 PRA model which quantified the containment response.

3. MODEL DEVELOPMENT

A series of Notebooks was developed to document every aspect of the PRA Model. These notebooks are designed to capture the most of the Capability Category II requirements of ASME RA-Sb-2005, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications* (Reference 5).

The following is a list of the Notebooks developed for WBN Dual Unit Model Revision 0 of the WBN PRA with the applicable section to the ASME in parenthesis:

- Initiating Events (IE)
- Accident Sequence Analysis (AS)
- Success Criteria (SC)
- Systems Analysis Summary Notebook and Individual system notebooks (RHR, SI, etc)(SY)
- Electric Power Recovery Notebook
- LOOP Frequency Notebook(IE)
- Human Reliability Analysis(HR)
- Thermal Hydraulic Analysis(SC)
- Data Analysis (DA)
- Interfacing Systems LOCA (ISLOCA)
- Internal Flooding (IF)
- Quantification (QU)
- Level 2 Analysis (LE)
- Sensitivity and Uncertainty (UNC)

The following sections summarize the purpose of each notebook and provide a discussion of important issues/findings relevant to the WBN Dual Unit Model Revision 0 PRA. The LOOP frequency notebook is discussed as a part of the Initiating Event Notebook, the thermal hydraulic analysis is discussed as part of the Success Criteria Notebook and the Electric Power Recovery analysis is discussed in the Systems Analysis section.

3.1. Initiating Events (IE)

An initiating event notebook was prepared to provide a discussion of the methodology used to develop the plant specific initiating events database and meet the requirements of Regulatory Guide 1.200.

The internal Initiating Events that challenge normal plant operation and require successful mitigation to prevent core damage are identified below in Table 3-1. For the WBN PRA the initial condition for an initiating event is defined as full-power operation, the plant transient condition will result in a reactor trip or turbine trip and challenge the safety systems. In less sudden transients, such as controlled power reductions that do not induce trips, there is a high probability that plant operators will affect an orderly plant shutdown without the safety system actuation. Orderly or controlled shutdowns, such as Technical Specification required shutdowns, were not considered initiating events since they do not challenge the plant safety systems. The initiating events in the WBN PRA model and how their frequencies were developed may be discussed in five general categories 1)initiating event frequencies that were derived from industry data or plant specific Failure Modes and Effects Analysis (FMEAs); 2)initiating event frequencies that were calculated using systems analyses; 3)initiating event frequencies that were calculated based on interfacing systems Loss of Coolant Accident (ISLOCA) analysis; 4)initiating event frequencies analyzed for Loss of Offsite Power (LOOP) events, 5)initiating events derived from an analysis of Internal Flooding. Failure of the reactor to trip automatically, called anticipated transient without scram (ATWS), is considered in the PRA model in the course of developing plant response scenarios. Therefore, ATWS events are not defined as a separate initiating event category.

The first group of initiating events was derived using several sources 1)comparison with categories from previous PRAs and other industry studies, 2)failure modes and effects analysis (FMEA) of the plant systems, 3) review of the Final Safety Analysis Report (FSAR), and 4)discussions with plant operators about specific postulated events. In addition to the initiating events modeled in Revision 4 of the WBN PRA a review of initiating event data from industry sources was performed including the NUREG/CR-3862 , WASH-1400 , NUREG/CR-2300, the Indian Point Probabilistic Safety Study, the Diablo Canyon PRA, South Texas Project PSA, the PLG Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants, NUREG/CR-5750, and NUREG/CR-6928. The initiating event frequencies used come primarily from NUREG/CR-5750, and NUREG/CR-6928. Plant-specific data was included in the calculation of these frequencies using a Bayesian updating process. The plant specific data is based on the evaluation and categorization of Licensee Event Reports (LERs) for WBN Unit 1 during the time period January 1, 2003 through March 31, 2008.

Support system initiating events were determined from reviewing the same sources as the initiating events described above, however; their frequency was calculated using plant specific system analyses. The plant Specific Support System analysis was performed consistent with the EPRI Support System Initiating Event Guideline, TR-1013490. Support system initiating fault trees were developed for all systems except for Total Loss of Plant Compressed Air. The IE frequency for Plant Compressed Air was a point estimate based on historical operating experience. To address the dependency a support system IE has on mitigation equipment the support system Initiating event trees were "OR-ed" with the post-initiators support system tops. All initiating system analysis fault trees were reviewed to insure the incorporation of passive failure, and potential cause failures that could lead to a support system initiating event. Initiating event basic events were given an 8760 hour mission time. System alignments were considered and flags were added into the model to reflect different system configurations. Standby failures are given a mission time of 24 hours, average repair time, or the allowed outage time.

Events and LOOP categories described in EPRI TR10109192, "Losses of Off-Site Power at U.S. Nuclear Power Plants – Through 2008," dated May 2009 were reviewed. The WBN PRA evaluation of LOOP events established separate LOOP frequencies for plant-centered, grid related and weather related events. This LOOP initiating event frequency representation was used so as to be consistent with the LOOP recovery curves development. The reason for segregating LOOP events is to facilitate an accurate estimate of efforts to restore voltage to the safety busses from the power-grid. The recovery likelihood is primarily a function of LOOP duration (which tends to differ among the LOOP event categories) and the personnel chiefly responsible for the tasks need to restore voltage to the safety busses. The WBN LOOP frequency assessment is based on data from the time period from January 1, 2000 to December 31, 2007. Selection of this period excludes older generic and plant specific data, much of which is no longer considered applicable. LOOP data is identified from a number of sources including EPRI reports, NUREGs and LERs. This data was reviewed to determine the applicability of the events to WBN. Events such as LOOPs caused by hurricanes which do not affect WBN due to its inland location were excluded. WBN plant specific data was considered, however no plant specific LOOP events have occurred at WBN during the time frame of interest. At WBN the two lines credited as offsite power line are 161kV lines originating from different switchyards and grid than the 500kV switchyard. The 500kV switchyard is connected to the generators.

At WBN the interfacing systems consist of both high-pressure and low-pressure piping connected by interface valves. The high-pressure piping is designed to operate at the normal reactor operating pressure of 2235 psig which is approximately 2250 psia, whereas the low-pressure piping is designed to operate at some pressure less than 2250 psia. The high-pressure piping is isolated from the low-pressure piping through a combination of check valves and isolation valves. A combination of valve failures and operator errors could overpressurize the low-pressure system. If the low-pressure system integrity is breached, a LOCA event will occur. The interfacing LOCA frequency is plant unique since it depends on plant characteristics such as piping configuration, plant operating procedures regarding testing of isolation valves during plant operation, etc. Therefore, a plant specific evaluation was used to estimate the frequencies used in the WBN PRA The WBN PRA analysis uses NSAC-154 , "ISLOCA Evaluation Guidelines", and NUREG/CR-5102 , "Interfacing Systems LOCA: Pressurized Water Reactors", as guidance for developing the ISLOCA event trees, success criteria, failure probabilities, and fault trees. Four classes of ISLOCA events are defined in this analysis. The criteria for defining these classes are; a) contained release, after failure of the pressure isolation valves (PIVs), through the relief valves or low-pressure system rupture, and b) release inside or outside containment. The descriptions of these four classes are as follows:

i. Small LOCA inside containment - In these scenarios the leak through the PIVs is within the capacity of the relief valve and they relieve to a tank inside containment.

ii. Small LOCA outside containment - In these scenarios the relief valves discharge to a tank outside containment or a pump seal leaks.

iii. Overpressurization/LOCA inside containment - In these scenarios either the leak through the PIVs is beyond the capacity of the relief valves or the relief valve fails to open. This results in overpressurization of the low-pressure piping and potential break in the low-pressure system. The break occurs inside containment.

iv. Overpressurization/LOCA outside containment - In these scenarios either the leak through the PIVs is beyond the capacity of the relief valves or the relief valve fails to open. This results in overpressurization of the low-pressure piping and potential break in the low-pressure system. The break occurs outside containment.

Potential pathways were screened based upon the criteria in ISLOCA NUREG CR/5102. Fault trees were created for this analysis using the CAFTA software tool. Twelve ISLOCA IEs were identified for Units 1 and Unit 2. Several of these identified ISLOCA IEs are the result of a mitigation system line rupture where the system is credited in the PRA model. To address the dependency that each ISLOCA IE could have on a mitigation system, a given ISLOCA IE fault tree was added into the corresponding system fault tree. This was done to ensure that the injection path/mitigation equipment was not credited during a given ISLOCA IE.

The WBN PRA upgraded the flooding analysis previously performed for the Unit 1 PRA to be consistent with the draft EPRI Guidelines for Internal Flooding Probabilistic Risk Assessment (IFPRAs) and the Internal Flooding portions of the joint ANS/ASME PRA standard and Regulatory Guide 1.200. The purpose of the flooding analysis is to identify all significant potential flood sources which can produce risk significant event sequences in the PRA. This included an assessment of the flood initiating event frequencies. Pressure boundary failure of piping or other passive, non-piping components, and inadvertent or spurious system or component actuations (e.g., maintenance-induced activities) could lead to localized or global flooding causing failures that affect plant safety. Flood-induced impacts on Structures, Systems and Components (SSCs) important to safety were evaluated to identify:

- Water sources within the plant that could create adverse conditions and affect the plant mitigating equipment are identified.
- The spray/flood scenarios that contribute significantly to Core Damage Frequency (CDF) or Large Early Release Frequency (LERF) so they could be quantified.

Internal flooding events differ from other internal initiating events in several ways. These differences are described below:

- Flooding events are often the result of passive component pressure boundary failure, inadvertent system actuations (for example, Fire Protection water system sprinkler-caused spraying/flooding), or maintenance-induced flooding (for example, heat exchanger cleaning).
- Internal flooding events may simultaneously impact multiple structures, redundant systems, and components at a plant. Mitigation of the event may therefore require a combination of plant system responses and manual interventions not considered in the accident sequence models for other internal event initiators.
- The evaluation of recovery actions from internal flooding events requires detailed consideration of unique challenges in detecting an impending flood and responding to it in a timely manner. Depending on spill rate, certain plant areas may not be accessible, hence, further complicating timely gathering of diagnostic information by plant personnel. Furthermore, risk of electrocution is another complicating factor in the assessment and evaluation of flooding response.

The internal flooding hazard has several characteristics that influence the identification, quantification, and treatment of the initiators. The following characteristics were included in the WBN development of flooding initiating event frequencies:

- The plant specific routing of piping
- Flood and spray events including the impact of submergence, jet impingement, spray, pipe whip, humidity, condensation, temperature and electrocution concerns
- The operating crew response to a flood initiator including challenges by diagnostic difficulties; communications difficulties between equipment operators and Main Control Room operators; difficulty in implemented internal flood response procedures that may be less well developed than other procedures

The primary source of rupture data used to calculate passive failures of piping in the WBN PRA is the 2006 EPRI report on pipe rupture frequencies. An exception for the initiating event frequency calculation is made for the main steam system. Pipe failure frequencies for this system were not explicitly included in the system catalogue present in EPRI-TR-1013141. Therefore, the generic pipe break frequencies for this system were extracted from EPRI-TR-102266(

Table 3-1
Initiating Events

Initiating Event	Description	IE Frequency (per reactor-year)	Impacted Unit
CPEX	Core Power Excursion	7.27E-03	1 & 2
ELOCA	Excessive LOCA	1.00E-07	1 & 2
EXMFW	Excessive Main Feedwater	3.95E-02	1 & 2
IMSIV	Inadvertent Closure of all MSIVs	1.53E-02	1 & 2
ISI	Inadvertent Safety Injection	1.03E-02	1 & 2
LLOCA	Large Break LOCA	1.33E-06	1 & 2
LOCV	Loss of Condenser Vacuum	6.53E-02	1 & 2
LRCP	Loss of 1 or More RCS/Primary Flow	2.89E-02	1 & 2
MLOCA	Medium Break LOCA	1.44E-05	1 & 2
MSIV	Inadvertent Closure of One MSIV	1.97E-02	1 & 2
MSVO	Steam Generator PORV Fails Open	8.55E-04	1 & 2
PLMFW	Partial Loss of Main Feedwater	1.46E-01	1 & 2
RTIE	Reactor Trip	2.85E-01	1 & 2
SGTR	Steam Generator Tube Rupture	3.54E-03	1 & 2
SLBIC	Steam Line Break Inside Containment	1.00E-03	1 & 2
SLBOC	Steam Line Break Outside Containment	1.00E-02	1 & 2
SLOCAL	Stuck Open Safety/Relief Valve	2.88E-03	1 & 2
SLOCAN	Small LOCA Non-Isolable	5.20E-04	1 & 2
SLOCAV	Very Small LOCA Non-Isolable	3.88E-03	1 & 2
TLMFW	Total Loss of Main Feedwater	7.01E-02	1 & 2
TLPCA	Total Loss of Plant Compressed Air	9.81E-03	1 & 2

TTIE	Turbine Trip	2.32E-01	1 & 2
1CCSA	LOSS OF CCS TRAIN A INITIATING EVENT UNIT 1	8.03E-03	1
2CCSA	LOSS OF CCS TRAIN A INITIATING EVENT UNIT 2	7.71E-03	2
U1_CCSTL	CCS TOTAL INITIATING EVENT UNIT 1	5.20E-06	1
U2_CCSTL	CCS TOTAL INITIATING EVENT UNIT 2	5.14E-06	2
ERCW1B	PARTIAL LOSS OF ERCW TRAIN 1B-B	3.56E-03	1
ERCW2A	PARTIAL LOSS OF ERCW TRAIN 2A-A	3.54E-03	2
ERCWTL	TOTAL LOSS OF ERCW	1.40E-05	1 & 2
U1_LDAAC	LOSS OF 120 VAC VITAL BOARD 1-I	4.63E-02	1
U1_LDBAC	LOSS OF 120 VAC VITAL BOARD 1-II	4.63E-02	1
U1_LDCAC	LOSS OF 120 VAC VITAL BOARD 1-III	4.63E-02	1
U1_LDDAC	LOSS OF 120 VAC VITAL BOARD 1-IV	4.63E-02	1
U2_LDAAC	LOSS OF 120 VAC VITAL BOARD 2-I	5.09E-02	2
U2_LDBAC	LOSS OF 120 VAC VITAL BOARD 2-II	5.09E-02	2
U2_LDCAC	LOSS OF 120 VAC VITAL BOARD 2-III	5.09E-02	2
U2_LDDAC	LOSS OF 120 VAC VITAL BOARD 2-IV	5.09E-02	2
LVBB1	LOSS OF BATTERY BOARD I	2.04E-02	1
LVBB2	LOSS OF BATTERY BOARD II	2.04E-02	1
LVBB3	LOSS OF BATTERY BOARD III	2.04E-02	2
LVBB4	LOSS OF BATTERY BOARD IV	2.04E-02	2
%0FLAFW1	Flood event induced by Unit 1 AFW line break in room 692.0-A1, 713.0-A1, 737.0-A1 or 737.0-A3	4.18E-06	1 & 2
%0FLAFW2	Flood event induced by Unit 2 AFW line break in room 692.0-A1, 713.0-A1, 737.0-A1 or 737.0-A12	4.19E-06	1 & 2
%0FLAFW1692A6	Flood event induced by AFW line break in room 692.0-A6	4.21E-07	1 & 2
%0FLAFW1692A7	Flood event induced by AFW line break in room 692.0-A7	9.96E-08	1 & 2

%0FLAFW2692A25	Flood event induced by AFW line break in room 692.0-A25	8.79E-08	1 & 2
%0FLAFW2692A26	Flood event induced by AFW line break in room 692.0-A26	2.86E-07	1 & 2
%0FLAFW713A6	Flood event induced by AFW line break in room 713.0-A6	1.95E-07	1 & 2
%0FLAFW713A19	Flood event induced by AFW line break in room 713.0-A19	1.94E-07	1 & 2
%0FLAFW737A5	Flood event induced by AFW line break in room 737.0-A5	1.28E-06	1 & 2
%0FLAFW737A9	Flood event induced by AFW line break in room 737.0-A9	1.29E-06	1 & 2
%0FLHPFPABF	Flood event induced by HPFP in the common areas of the Auxiliary Building (multiple elevations)	5.49E-04	1 & 2
%0FLCRDM1F	Flood event induced by HPFP or RCW line breaks in room 782.0-A1	6.46E-05	1 & 2
%0FLCRDM2F	Flood event induced by HPFP or RCW line breaks in room 782.0-A3	6.51E-05	1 & 2
%0FLHPFPAB772A7	Flood event induced by break of HPFP line in room 772.0-A7	4.05E-08	1 & 2
%0FLHPFPAB772A10	Flood event induced by break of HPFP line in room 772.0-A10	3.94E-08	1 & 2
%0FLHPFPAB757A2	Flood event induced by break of HPFP line in room 757.0-A2	1.44E-07	1 & 2
%0FLHPFPAB757A5	Flood event induced by break of HPFP line in room 757.0-A5	5.44E-08	1 & 2
%0FLHPFPAB757A21	Flood event induced by break of HPFP line in room 757.0-A21	4.21E-08	1 & 2
%0FLHPFPAB757A24	Flood event induced by break of HPFP line in room 757.0-A24	1.07E-07	1 & 2
%0FLHPFP737A5F	Flood event induced by HPFP line break in room 737.0-A5	2.60E-06	1 & 2
%0FLHPFP737A9F	Flood event induced by HPFP line break in room 737.0-A9	1.14E-06	1 & 2
%0FLHPFPAB713A68F	Flood event induced by HPFP line break in room 713.0-A6 or 713.0-A8	1.18E-05	1 & 2
%0FLHPFPAB713A1921F	Flood event induced by HPFP line break in room 713.0-A19 or 713.0-A21	1.21E-05	1 & 2
%0FLHPFP692A7F	Flood event induced by a HPFP line break in room 692.0-A7	2.14E-06	1 & 2
%0FLHPFP692A25F	Flood event induced by a HPFP line break in room 692.0-A25	3.99E-06	1 & 2
%0FLHPFPCB	Flood event induced by a HPFP line break in the Control Building	1.06E-05	1 & 2
%0FLHPFPIPS	Flood event induced by a HPFP or RCW line break in room 711.0-E1	2.89E-04	1 & 2
%0FLDWSAB	Flood event induced by DWS in the common areas of the Auxiliary Building (multiple elevations)	9.36E-06	1 & 2

%0FLDWS713A6	Flood event induced by DWS line break in room 713.0-A6	4.07E-07	1 & 2
%0FLDWS713A19	Flood event induced by DWS line break in room 713.0-A19	6.88E-07	1 & 2
%1FLCCS	Flood event induced by CCS line break (Train A)	2.14E-05	1
%2FLCCS	Flood event induced by CCS line break (Train B)	2.09E-05	2
%1FLCCS1AB692A7	Flood event induced by CCS line break in room 692.0-A7	1.34E-06	1
%2FLCCS2AB692A25	Flood event induced by CCS line break in room 692.0-A15	1.05E-06	2
%1FLCCS757A13	Flood event induced by CCS line break in room 757.0-A13 (Surge tank A)	3.01E-07	1
%2FLCCS757A13	Flood event induced by CCS line break in room 757.0-A13 (Surge tank B)	3.01E-07	2
%1FLCCS713A28	Flood event induced by unisolated break in CCS line in room 713.0-A28	1.21E-06	1
%2FLCCS713A29	Flood event induced by unisolated break in CCS line in room 713.0-A29	1.21E-06	2
%1FLCCS737A5	Flood event induced by CCS line break in room 737.0-A5	2.18E-05	1
%2FLCCS737A9	Flood event induced by CCS line break in room 737.0-A9	2.15E-05	2
%0FLRCWABF	Flood event induced by RCW in the common areas of the Auxiliary Building (multiple elevations)	3.42E-04	1 & 2
%0FLRCWABMF	Major flood event induced by RCW in the common areas of the Auxiliary Building (multiple elevations)	3.94E-05	1 & 2
%0FLRCW772A8	Flood event induced by rupture of RCW line in room 772.0-A8	1.06E-06	1 & 2
%0FLRCW772A9	Flood event induced by rupture of RCW line in room 772.0-A9	1.06E-06	1 & 2
%0FLRCW757A9	Flood event induced by rupture of RCW line in room 757.0-A9	1.27E-07	1 & 2
%0FLRCW757A17	Flood event induced by rupture of RCW line in room 757.0-A17	1.27E-07	1 & 2
%0FLRCW737A5F	Flood event induced by rupture of RCW lines in room 737.0-A5	4.36E-05	1 & 2
%0FLRCW737A5MF	Major flood event induced by rupture of RCW lines in room 737.0-A5	5.07E-06	1 & 2
%0FLRCW737A9F	Flood event induced by rupture of RCW lines in room 737.0-A9	5.55E-05	1 & 2
%0FLERCWAB676F-1A	Flood event induced by unisolated ERCW break at elevation 676' of Auxiliary Building (ESF room cooling train 1A)	1.29E-04	1 & 2
%0FLERCWAB676F-1B	Flood event induced by unisolated ERCW break at elevation 676' of Auxiliary Building (ESF room cooling train 1B)	1.31E-04	1 & 2

%0FLERCWAB676F-2A	Flood event induced by unisolated ERCW break at elevation 676' of Auxiliary Building (ESF room cooling train 2A)	1.29E-04	1 & 2
%0FLERCWAB676F-2B	Flood event induced by unisolated ERCW break at elevation 676' of Auxiliary Building (ESF room cooling train 2B)	1.31E-04	1 & 2
%0FLERCWAB676MF-1A	Major flood event induced by unisolated ERCW break in room 676.0-A1 (ESF room cooling train 1A)	5.88E-06	1 & 2
%0FLERCWAB676MF-1B	Major flood event induced by unisolated ERCW break in room 676.0-A1 (ESF room cooling train 1B)	5.88E-06	1 & 2
%0FLERCWAB676MF-2A	Major flood event induced by unisolated ERCW break in room 676.0-A1 (ESF room cooling train 2A)	5.88E-06	1 & 2
%0FLERCWAB676MF-2B	Major flood event induced by unisolated ERCW break in room 676.0-A1 (ESF room cooling train 2B)	5.88E-06	1 & 2
%0FLERCWDISAF	Flood event induced by ERCW line break: discharge header A	1.59E-03	1 & 2
%0FLERCWDISAMF	Major flood event induced by ERCW line break: discharge header A	1.27E-04	1 & 2
%0FLERCW692A6F	Flood event induced by ERCW line break: discharge header A (AFW TD pump room)	4.41E-06	1 & 2
%0FLERCW692A6MF	Major flood event induced by ERCW line break: discharge header A (AFW TD pump room)	2.51E-07	1 & 2
%0FLERCWDISBF	Flood event induced by ERCW line break: discharge header B	1.87E-03	1 & 2
%0FLERCWDISBMF	Major flood event induced by ERCW line break: discharge header B	1.27E-04	1 & 2
%0FLERCW692A26F	Flood event induced by ERCW line break: discharge header B (AFW TD pump room)	5.40E-04	1 & 2
%0FLERCW692A26MF	Major flood event induced by ERCW line break: discharge header B (AFW TD pump room)	3.52E-05	1 & 2
%0FLERCW692A7	Flood event induced by unisolated ERCW break in one supply header in room 692.0-A7.	7.34E-04	1 & 2
%0FLERCW1AESFRCF	Flood event induced by unisolated ERCW break associated with ESF room cooling train 1A	1.10E-04	1 & 2
%0FLERCW1AESFRCMF	Major flood event induced by unisolated ERCW break associated with ESF room cooling train 1A	9.40E-07	1 & 2

%0FLERCW1BESFRCF	Flood event induced by unisolated ERCW break associated with ESF room cooling train 1B	1.15E-04	1 & 2
%0FLERCW1BESFRCMF	Major flood event induced by unisolated ERCW break associated with ESF room cooling train 1B	4.54E-07	1 & 2
%0FLERCW2AESFRCF	Flood event induced by unisolated ERCW break associated with ESF room cooling train 2A	1.31E-04	1 & 2
%0FLERCW2AESFRCMF	Major flood event induced by unisolated ERCW break associated with ESF room cooling train 2A	1.02E-06	1 & 2
%0FLERCW2BESFRCF	Flood event induced by unisolated ERCW break associated with ESF room cooling train 2B	1.85E-04	1 & 2
%0FLERCW2BESFRCMF	Major flood event induced by unisolated ERCW break associated with ESF room cooling train 2B	1.24E-06	1 & 2
%0FLERCW692A25	Flood event induced by unisolated ERCW break in one supply header in room 692.0-A25	5.75E-04	1 & 2
%0FLERCW713A6	Flood event induced by unisolated ERCW break in one supply header in room 713.0-A6	5.28E-04	1 & 2
%0FLERCW713A19	Flood event induced by unisolated ERCW break in one supply header in room 713.0-A19	4.85E-04	1 & 2
%0FLERCW713A28	Flood event induced by unisolated ERCW break in one supply header in room 713.0-A28	1.17E-04	1 & 2
%0FLERCW713A29	Flood event induced by unisolated ERCW break in one supply header in room 713.0-A29	3.18E-05	1 & 2
%0FLERCW737A5	Flood event induced by unisolated ERCW break in one supply header in room 737.0-A5	2.62E-04	1 & 2
%0FLERCW737A9	Flood event induced by unisolated ERCW break in one supply header in room 737.0-A9	1.50E-04	1 & 2
%0FLERCWCB	Flood event induced by ERCW line break in Control Building	2.29E-04	1 & 2
%0FLERCWIPSA	Flood event in ERCW Strainer room A	1.60E-04	1 & 2
%0FLERCWIPSB	Flood event in ERCW Strainer room B	1.60E-04	1 & 2

%0FLRWST1AB676	Flood event induced by unisolated line break from RWST 1 at elevation 676' of Auxiliary Building	4.12E-05	1 & 2
%0FLRWST2AB676	Flood event induced by unisolated line break from RWST 2 at elevation 676' of Auxiliary Building	4.17E-05	1 & 2
%0FLRWST1AB692A1	Flood event induced by rupture of RWST 1 header in room 692.0-A1	1.66E-05	1 & 2
%0FLRWST2AB692A1	Flood event induced by rupture of RWST 2 header in room 692.0-A1	1.66E-05	1 & 2
%0FLRWST1692A7	Flood event induced by break in the lines from RWST 1 in room 692.0-A7	3.41E-06	1 & 2
%0FLRWST1692A8	Flood event induced by break in the lines from RWST 1 in rooms 692.0-A8 or 713.0-A7	2.14E-05	1 & 2
%0FLRWST1SIS	Flood event induced by SIS line break in any of the Unit 1 SIS pump rooms	7.59E-07	1 & 2
%0FLRWST2SIS	Flood event induced by SIS line break in any of the Unit 2 SIS pump rooms	1.13E-06	1 & 2
%0FLRWST2692A24	Flood event induced by break in the lines from RWST 2 in rooms 692.0-A24 or 713.0-A20	2.12E-05	1 & 2
%0FLRWST2692A25	Flood event induced by break in the lines from RWST 2 in room 692.0-A25	2.72E-06	1 & 2
%0FLRWST1713HX	Flood event induced by a rupture of the lines from RWST1 in any of the Unit 1 HX rooms at elevation 713'	6.70E-06	1 & 2
%0FLRWST2713HX	Flood event induced by a rupture of the lines from RWST2 in any of the Unit 2 HX rooms at elevation 713'	6.70E-06	1 & 2
%0FLRWST1713A28	Flood event induced by break in the lines from RWST 1 in room 713.0-A28	3.71E-05	1 & 2
%0FLRWST2713A29	Flood event induced by break in the lines from RWST 2 in room 713.0-A29	3.71E-05	1 & 2
%0FLCVCS1713A6	Flood event induced by CVCS break in room 713.0-A6	1.26E-06	1 & 2
%0FLCVCS1713A0	Flood event induced by CVCS break in area 713.0-A0 (Unit 1)	3.78E-06	1 & 2
%0FLCVCS1PITS	Flood event induced by Unit 1 CVCS break in sealed pits	4.76E-06	1 & 2
%0FLCVCS2713A19	Flood event induced by CVCS break in room 713.0-A19	1.26E-06	1 & 2
%0FLCVCS2713A0	Flood event induced by CVCS break in area 713.0-A0 (Unit 2)	3.78E-06	1 & 2
%0FLCVCS2PITS	Flood event induced by Unit 2 CVCS break in sealed pits	4.76E-06	1 & 2
%0FLCVCS1692A9	Flood event induced by CVCS break in room 692.0-A9	6.93E-07	1 & 2

%0FLCVCS1692A10	Flood event induced by CVCS break in room 692.0-A10	5.52E-07	1 & 2
%0FLCVCS2692A22	Flood event induced by CVCS break in room 692.0-A22	6.93E-07	1 & 2
%0FLCVCS2692A23	Flood event induced by CVCS break in room 692.0-A23	5.52E-07	1 & 2
%0FLTBMF	Major flood in the Turbine Building	8.58E-03	1 & 2
%0FLTBCST1MF	Major flood in the Turbine Building involving line break from CST1	7.11E-04	1 & 2
%0FLTBCST2MF	Major flood in the Turbine Building involving line break from CST2	2.11E-06	1 & 2
%1FLTBSPRAY1-A-B	Spray event on Unit 1 6.9kV boards A and B	1.24E-04	1
%1FLTBSPRAY1-B-C	Spray event on Unit 1 6.9kV boards B and C	1.24E-04	1
%1FLTBSPRAY1-C-D	Spray event on Unit 1 6.9kV boards C and D	1.24E-04	1
%0FLTBSPRAY1-A-D	Spray event on 6.9kV board 1D and 2A	1.24E-04	1 & 2
%2FLTBSPRAY1-A-B	Spray event on Unit 2 6.9kV boards A and B	1.24E-04	2
%2FLTBSPRAY1-B-C	Spray event on Unit 2 6.9kV boards B and C	1.24E-04	2
%2FLTBSPRAY1-C-D	Spray event on Unit 2 6.9kV boards C and D	1.24E-04	2
%1FLTBSPRAY2A	Spray event on U1 board 203A (480V TB)	3.49E-05	1
%1FLTBSPRAY2B	Spray event on U1 board 203B (480V TB)	1.67E-04	1
%2FLTBSPRAY2B	Spray event on U2 board 203B (480V TB)	1.67E-04	2
%0FLTBSPRAY3	Spray event on common board 205 B	3.40E-05	1 & 2
%0FLTBSPRAY4	Spray event on air compressor D and sequencer	6.80E-05	1 & 2
%0FLTBSPRAY5	Spray event on dryers	5.80E-05	1 & 2
%1FLTBSPRAY6	Spray event on distribution board WBN-0-DPL -239-0001	5.59E-05	1
%1FLRTIE	Spray event on MG sets – Unit 1	5.42E-04	1
%2FLRTIE	Spray event on MG sets – Unit 2	5.47E-04	2
%1FLHELBAFW	HELB scenario induced by MSS supply to AFW line break – Unit 1	7.10E-06	1
%2FLHELBAFW	HELB scenario induced by MSS supply to AFW line break – Unit 2	1.32E-05	2
%0FLHELB01A	HELB scenario induced by CVCS line break in room 713.0-A28	2.04E-05	1 & 2

%0FLHELB01B	HELB scenario induced by CVCS line break in room 713.0-A29	2.04E-05	1 & 2
%0FLHELB02A	HELB scenario induced by CVCS line break in room 737.0-A7	1.73E-07	1 & 2
%0FLHELB02B	HELB scenario induced by CVCS line break in room 737.0-A8	1.73E-07	1 & 2
ISL-IEX15	ISLOCA CVCS LETDOWN PENETRATION X-15	4.37E-10	1 & 2
ISL-IEX107	ISLOCA RHR Supply Penetration X-107	1.38E-07	1 & 2
ISL-IEX20A-OVPR	ISLOCA RHR COLD LEG INJECTION FROM PUMP B Penetration X-20A (OVERPRESSURE)	1.74E-08	1 & 2
ISL-IEX20B-OVPR	ISLOCA RHR COLD LEG INJECTION FROM PUMP A Penetration X-20B (OVERPRESSURE)	1.74E-08	1 & 2
ISL-IEX17-OVPR	ISLOCA RHR HOT LEG PENETRATION X-17 (OVERPRESSURE)	1.78E-10	1 & 2
ISL-IEX33-OVPR	ISLOCA SI COLD LEG Penetration X-33 (OVERPRESSURE)	3.03E-09	1 & 2
ISL-IEX21-OVPR	ISLOCA SAFETY INJECTION HOT LEG B PENETRATION X-21 (OVERPRESSURE)	2.12E-12	1 & 2
ISL-IEX32-OVPR	ISLOCA SAFETY INJECTION HOT LEG A PENETRATION X-32 (OVERPRESSURE)	2.12E-12	1 & 2
ISL-IERWSTRHR-LL	LARGE BREAK ISL (> 6 INCHES) - FAILURE OF RWST **2**	1.56E-08	1 & 2
ISL-IERHRPMPSEALSL	SMALL BREAK ISL (<OR=2 INCHES)-BOTH RHR TRAINS FAIL - PUMP SEAL FAILURE **3**	9.19E-06	1 & 2
ISL-IESIPMPSEAL-A	ISLOCA - SI TRAIN A FAILS DUE TO PUMP SEAL FAILURE **3**	1.30E-08	1 & 2
ISL-IESIPMPSEAL-B	ISLOCA - SI TRAIN B FAILS DUE TO PUMP SEAL FAILURE **3**	1.30E-08	1 & 2
%0LOSP-GR	Grid-related Loss of Offsite Power	1.01E-02	1 & 2
%0LOSP-PC	Plant-Centered Loss of Offsite Power	8.12E-03	1 & 2
%0LOSP-WI	Weather-induced Loss of Offsite Power	2.03E-03	1 & 2

The importance of initiating events was examined by determining the contributions of core damage sequences grouped by initiating event. Table 3-2 and Figure 3-1 show the contribution toward the total core damage frequency for each initiating event modeled in the WBN PRA for Unit 2.

Table 3-2
Initiating Event Group Contributions to Core Damage Frequency

CDF (per reactor-year)	**Percent Contribution**	**Initiating Event**
7.20E-06	22.00%	Loss of Offsite Power (Grid Related)
6.13E-06	18.70%	Loss of Offsite Power (Plant Centered)
5.02E-06	15.30%	Total Loss of ERCW
3.73E-06	11.20%	Flood
3.04E-06	9.40%	Loss of 120V AC Vital Instrument Board
1.69E-06	5.20%	Loss of Offsite Power (Weather Induced)
1.46E-06	4.30%	Loss of Battery Boards
1.11E-06	3.40%	Total Loss of Component Cooling System Unit 1
9.49E-07	2.80%	Secondary Side Break Outside Containment
6.43E-07	2.00%	Small LOCA Stuck Open Safety Relief Valve
3.27E-07	1.00%	Turbine Trip
2.36E-07	0.70%	Reactor Trip
2.03E-07	0.60%	Partial Loss of Main Feedwater
9.64E-08	0.40%	Small LOCA
1.80E-07	0.40%	Medium LOCA
9.40E-08	0.40%	Secondary Side Break Inside Containement
1.40E-07	0.40%	Total Loss of Plant Compressed Air
1.00E-07	0.30%	Excessive LOCA (Vessel Rupture)
9.23E-08	0.30%	Loss of Condenser Vacuum
9.92E-08	0.30%	Total Loss of Main Feedwater
5.90E-08	0.20%	Loss of Component Cooling System Train 2A
2.18E-08	0.10%	Inadvertent Closure of 1 MSIV
4.45E-08	0.10%	Excessive Main Feedwater
2.18E-08	0.10%	Inadvertent Closure of all MSIVs
3.24E-08	0.10%	Loss of Primary Flow
1.26E-08	0.00%	Partial Loss of ERCW Unit 1
1.14E-08	0.00%	Large LOCA
1.11E-08	0.00%	Inadvertent Safety Injection
8.10E-09	0.00%	Interfacing Systems LOCA
7.72E-09	0.00%	Core Power Excursion
2.66E-09	0.00%	Very Small LOCA Initiating Event
2.34E-09	0.00%	Steam Generator Tube Rupture

Table 3-2 Initiating Event Group Contributions to Core Damage Frequency		
CDF (per reactor-year)	**Percent Contribution**	**Initiating Event**
7.68E-10	0.00%	Steam Generator PORV Fails Open
3.28E-05		**TOTAL**

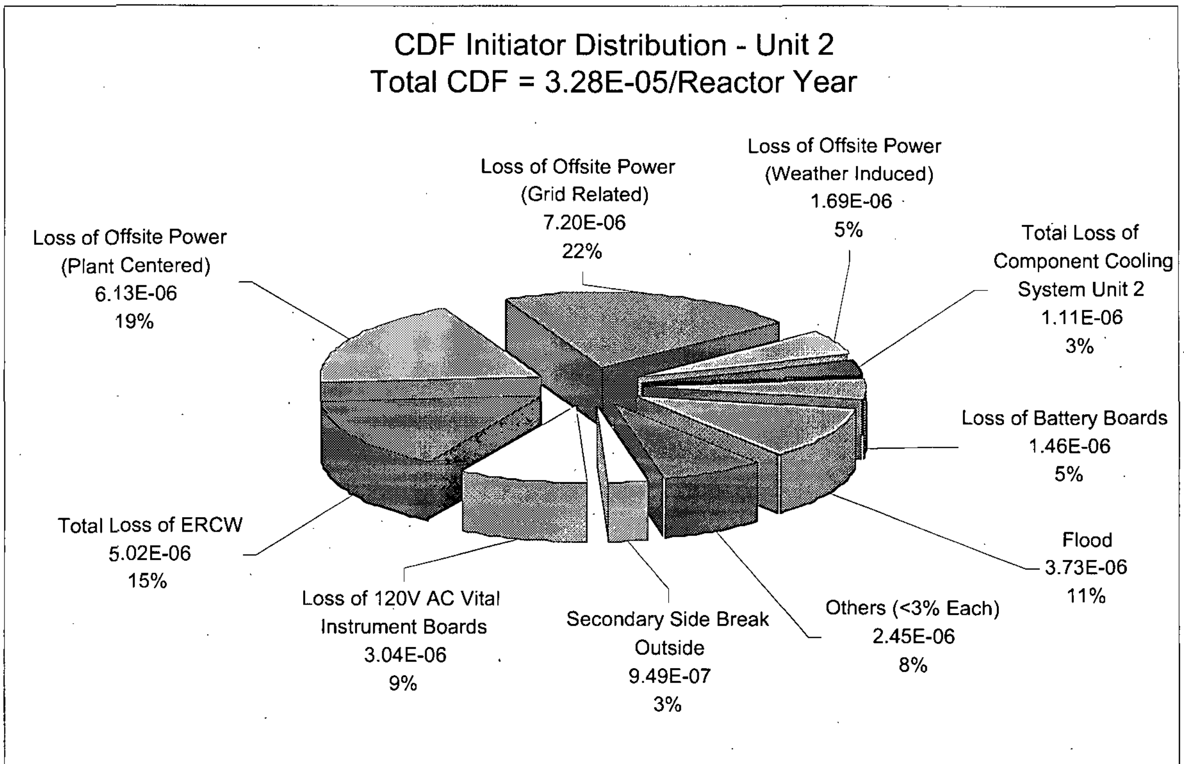


Figure 3-1. Initiating Event Group Contributions to Core Damage Frequency

Loss of offsite power sequences contribute 45.9% to the total CDF. The LOOP sequences include grid related, plant centered, and weather induced LOOPs. Grid related LOOP contributes 22%, plant centered LOOP contributes 18.7%, and a 5.2% contribution comes from Weather Induced LOOP.

Loss of Essential Raw Cooling Water (ERCW) events contribute 15.3% of the total CDF. These scenarios are typified by induced RCP seal failures caused by loss of seal cooling.

Internal floods account for 11.2% of CDF. The most important sources of internal floods are associated with a rupture or major flow diversion in one ERCW train combined with failure of the other train. Many of these sequences are effectively a total loss of ERCW. ERCW is an important support system since it provides the ultimate heat sink for reactor coolant pump (RCP) seal cooling and ECCS pump cooling. Thus, a complete loss of ERCW results in an RCP seal LOCA with inadequate coolant makeup capability. Other important sources of internal floods are associated with a rupture of the RCW line in the fifth vital battery and board room and HEPA filter plenum room.

Loss of 120V AC Vital Instrument Board accounts for 9.4% of the total CDF. Failure of a train of Vital Instrument Power can initiate a plant transient, and with an independent failure of the opposite train, leads to challenging the operators to perform required manual actions.

Loss of Battery Boards accounts for 4.3% of the total CDF. The loss of battery boards affects safety related equipment.

Loss of component cooling water events account for 3.4% of the total CDF. These scenarios are typified by induced RCP seal failures caused by loss of seal cooling.

The general class of other LOCAs accounts for approximately 3% of the total CDF. This class includes the following specific initiating events: small isolable LOCAs, medium LOCAs, large LOCAs and excess LOCA (i.e. reactor vessel rupture). Interfacing system LOCAs are also included in this category. However, should they lead to core damage, these initiators are significant because of their potential for a large release path to bypass the containment. The LOCA class of events is primarily characterized by failure of the emergency core cooling systems (ECCS) in recirculation. These failures are due to either operator errors in aligning for recirculation or hardware failures in the recirculation systems.

All other initiating events individually contribute less than 3% of the total CDF and can be seen in Table 3-2.

3.2. Accident Sequence Analysis (AS)

As part of the WBN PRA an accident sequence notebook was developed that describes the event tree models developed to analyze accident sequences from an initiating event to a safe stable state or core damage. The accident sequence analysis models, chronologically (to the extent practical), the different possible progressions of events (i.e., accident sequences) that can occur from the start of the initiating event to either successful mitigation or core damage. The accident sequences account for the systems that are used (and available) and operator actions performed to mitigate the initiator based on the defined success criteria and plant operating procedures (e.g., plant emergency and abnormal operating procedures) and training. The availability of a system includes consideration of the functional, phenomenological, and operational dependencies and interfaces between the various systems and operator actions during the course of the accident progression.

All initiating events were grouped into classes that could be evaluated collectively. For each functional group of initiating events, an event tree model was developed that defines the possible plant responses, mitigating system functions, and operator actions that determine the event sequence progression. During risk model development, existing safety analyses are reviewed, and selected thermal hydraulic analyses are performed to establish realistic success criteria for the mitigating systems and operator actions that are modeled in the PRA. A comprehensive set of plant damage states were defined to account for important conditions that may affect containment response and possible offsite releases after a severe core damage event. These plant damage states provide the interface between the Level 1 PRA models and the Level 2 PRA models. Licensed operators were interviewed as part of this process to ensure realistic conditions were modeled.

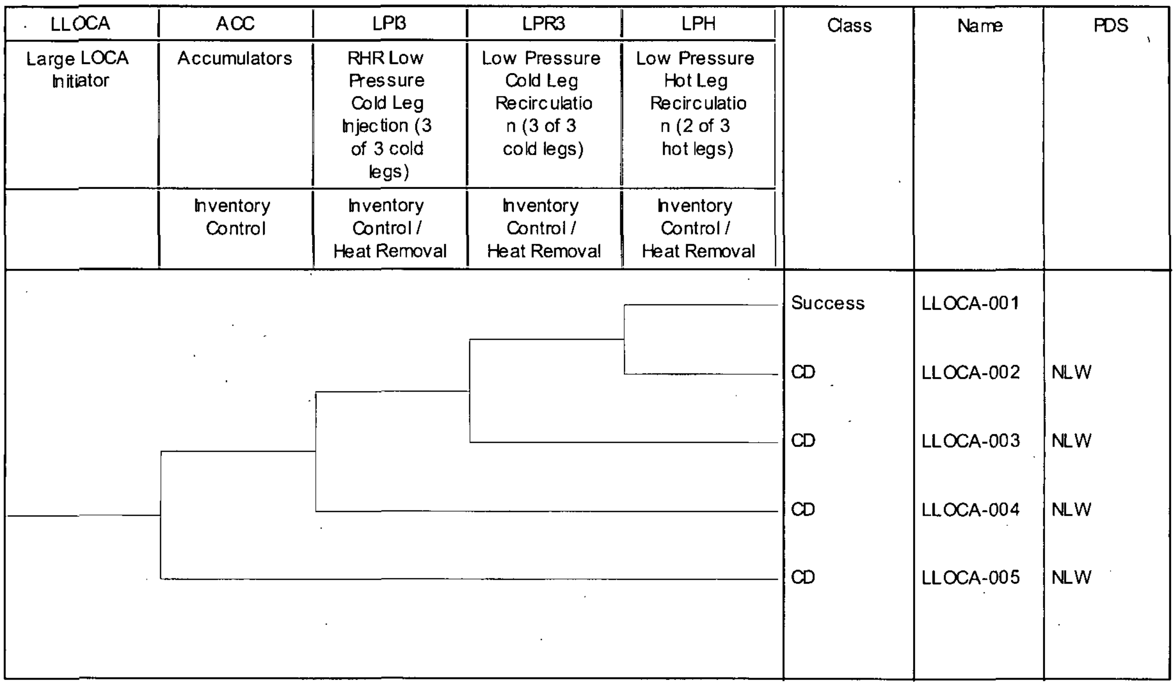
A total of ten event trees were developed for the WBN PRA:

General Transient (GTRAN)
Large Break LOCA (LLOCA)

Medium Break LOCA (MLOCA)
Small Break LOCA (SLOCA)
Very Small Break LOCA (SLOCAV)
Secondary Side Break Inside Containment (SSBI)
Secondary Side Break Outside Containment (SSBO)
Steam Generator Tube Rupture (SGTR)
Anticipated Transient without SCRAM (ATWS)
Interfacing System LOCA (ISLOCA)

The Large LOCA event tree is shown as an example below:

Figure 3-2 Event Tree LLOCA



The mapping of initiating events to event trees is provided in the following table:

Table 3.2-3: Watts Bar Initiating Event Linkage to Accident Sequence Event Trees

Group	Category	Initiator Designator	Core Damage Event Tree
Loss of Coolant	1. Excessive LOCA (reactor vessel failure)	ELOCA	NONE
	2. Large LOCA (> 6-inch diameter)	LLOCA	LLOCA
	3. Medium LOCA (≥ 2 to ≤ 6-inch diameter)	MLOCA	MLOCA
	4. Small LOCA (nonisolable)	SLOCAN	SLOCA
	5. Small LOCA (isolable)	SLOCAI	SLOCA
	6. Very Small LOCA (nonisolable)	SLOCAV	SLOCAV
	7. Steam Generator Tube Rupture	SGTR	SGTR
	8. Interfacing Systems LOCA — Large and Medium	XI	ISLOCA
	9. Interfacing Systems LOCA — Small	XS	ISLOCA
Transients	10. Reactor Trips	RTIE	GTRAN
	11. Core Power Excursion	CPEX	GTRAN
	12. Turbine Trip	TTIE	GTRAN
	13. Inadvertent Safety Injection	ISI	GTRAN
	14. Total Loss of All Main Feedwater	TLMFW	GTRAN
	15. Partial Loss of Main Feedwater	PLMFW	GTRAN
	16. Loss of Condenser Vacuum	LOCV	GTRAN
	17. Excessive Feedwater	EXMFW	GTRAN
	18. Inadvertent Closure of One MSIV	MSIV	GTRAN
	19. Inadvertent Closure of All MSIVs	IMSIV	GTRAN
	20. Loss of Primary Flow	LRCP	GTRAN
	21. Steam Line Break Outside Containment	SLBOC	SSBO
	22. Steam Line Break Inside Containment	SLBIC	SSBI
	23. Inadvertent Opening of Main Steam Relief Valves	MSVO	SSBO
Loss of Support Initiating Events	24. Loss of Offsite Power	LOSP-GR LOSP-PC LOSP-WI	GTRAN
	25. Loss of 1-I Vital AC Instrument Board	LDAAC	GTRAN
	26. Loss of 1-II Vital AC Instrument Board	LDBAC	GTRAN
	27. Loss of 1-III Vital AC Instrument Board	LDCAC	GTRAN
	28. Loss of 1-IV Vital AC Instrument Board	LDDAC	GTRAN
	29. Loss of Vital Battery Board I	LVBB1	GTRAN
	30. Loss of Vital Battery Board II	LVBB2	GTRAN
	31. Total Loss of CCS	CCSTL	GTRAN
	32. Loss of CCS Train A	CCSA	GTRAN
	33. Total Loss of ERCW	ERCWTL	GTRAN
	34. Loss of ERCW Train A	ERCWA	GTRAN
	35. Loss of ERCW Train B	ERCWB	GTRAN
Internal Flooding Events	Internal flooding initiating events are defined in the Internal Flooding Notebook. All internal flood initiating events are analyzed using the GTRAN event tree.		GTRAN

Plant damage states are the end states of the level 1 accident sequences. They define the condition of systems and containment that affect the level 2 analysis. The PDSs can be considered the entry conditions for the Level 2 analysis described in the Level 2 Analysis Notebook. The PDSs are shown in Table 3-4. The PDS nomenclature is made up of 3 characters. The first designates whether the sequence is a Containment Bypass or not, and is designated with an N or B. Steam Generator Tube Ruptures or Interfacing System LOCA (ISLOCA) bypass the containment. The second character designates the RCS pressure at the time of core damage as follows; L for pressures below the accumulator setpoint (about 600 psi) or H for high RCS pressure. In general medium and large LOCAs result in low RCS pressure and transients with failure to cooldown and depressurize result in high RCS pressure. The third character designates the steam generator condition with D if the Steam Generator (SG) is dry and no Auxiliary Feedwater (AFW) flow to the SG, or W if the SG is removing decay heat.

Table 3.2-4: Plant Damage States

PDS	Description
NHD	Containment is not bypassed. There is no or small leakage from the RCS and it is at a high pressure at the time of core damage. There is no feedwater or auxiliary feedwater to the steam generators and the steam generators are dry at the time of core damage.
NLD	Containment is not bypassed. There is a medium or large LOCA from the RCS and it is at low or atmospheric pressure at the time of core damage. There is no feedwater or auxiliary feedwater to the steam generators and the steam generators are dry at the time of core damage.
NHW	Containment is not bypassed. There is no or small leakage from the RCS and it is at a high pressure at the time of core damage. Feedwater or auxiliary feedwater is being supplied to the steam generators and the steam generator water level is at nominal level at the time of core damage.
NLW	Containment is not bypassed. There is a medium or large LOCA from the RCS and it is at low pressure at the time of core damage. Feedwater or auxiliary feedwater is being supplied to the steam generators and the steam generator water level is at nominal level at the time of core damage.
BHD	The containment is bypassed at the time of core damage (i.e., Steam Generator Tube Rupture (SGTR) or ISLOCA). There is no or small leakage from the RCS and it is at high or intermediate pressure (above the accumulator setpoint) at the time of core damage. There is no feedwater or auxiliary feedwater to the steam generators and the steam generators are dry at the time of core damage.

PDS	Description
BLD	The containment is bypassed at the time of core damage (i.e., SGTR or ISLOCA). There is a large leakage from the RCS or the RCS has been depressurized and it is at a low pressure at the time of core damage. There is no feedwater or auxiliary feedwater to the steam generators and the steam generators are dry at the time of core damage.
BHW	The containment is bypassed at the time of core damage (i.e., SGTR or ISLOCA). There is no or small leakage from the RCS and it is at high/intermediate pressure at the time of core damage. Feedwater or auxiliary feedwater is being supplied to the steam generators and the steam generators are at nominal level at the time of core damage.
BLW	The containment is bypassed at the time of core damage (i.e., SGTR or ISLOCA). There is large leakage from the RCS or the RCS has been depressurized and it is at a low pressure at the time of core damage. Feedwater or auxiliary feedwater is being supplied to the steam generators and the steam generators are at nominal level at the time of core damage.

In the CAFTA model, top logic was developed to allow event tree sequences to be quantified separately. The results of the sequence quantifications for internal event (without internal flooding) CDF are shown in Table 3-5. The truncations shown are used only for the individual sequence. Some sequences required a lower truncation to produce results. Sequence cutset results include removal of mutually exclusive events and recovery factors applied to appropriate cutsets after quantification resulting in individual cutset values below the truncation limit. All Unit 1 sequences were evaluated during the initial quantification to review and check the model logic and consistency with systems and success criteria. The event tree logic structures which show the accident sequences are presented in the WBN Accident Sequence Notebook. The sequences described here are for Unit 1. The Unit 2 logic model is based on the Unit 1 model therefore the sequences which are large contributors to core damage are similar. The following are descriptions of the accident sequences from the event trees for sequences that cumulatively contribute more than 95% of the internal event (without internal flooding) CDF.

<u>SLOCA-024</u>

Sequence SLOCA-024 which contributes 27.0% of the internal event CDF is described below.

Sequence SLOCA-024 is a small LOCA with failure of both CVCS and SI to inject to the cold leg. The auxiliary feedwater system removes decay heat and the operators initiate a rapid cooldown and depressurization to RHR injection conditions. When the RCS reaches low pressure, the RHR pumps fail to provide cold leg injection leading to core damage.

The primary initiating events that lead to this sequence are Total Loss of ERCW (40.0%), Partial Loss of ERCW (21.5%), Loss of Offsite Power initiators LOSP-GR (10.2%), LOSP-PC (9.5%), LOSP-WI (2.8%) and Loss of Component Cooling (10.5%).

The small LOCA in this sequence is initiated by a 182 gpm/pump RCP seal LOCA resulting from a station blackout condition, or loss of ERCW or CCS events with consequential seal injection and thermal barrier cooling failures. The station blackout condition or loss of ERCW also fails the RHR system.

GTRAN-015

The next highest sequence is GTRAN-015 which contributes 26.6% of the internal event CDF.

Sequence GTRAN-015 is a plant transient initiating event with successful reactor trip and the pressurizer PORVs are either not challenged or closed following steam or water relief. Failure of secondary heat removal via the auxiliary feedwater system and failure of high pressure injection for bleed and feed operation fails all heat removal functions.

The primary initiating events that lead to this sequence are Loss of Offsite Power initiators LOSP-GR (43.6%), LOSP-PC (37.0%), LOSP-WI (10.3%) and Total Loss of ERCW (7.2%).

This sequence is dominated by station blackout cutsets initiated by a loss of offsite power with independent failures of the onsite electrical power system. Station blackout conditions result in failure of the motor driven auxiliary feedwater pumps and the high pressure injection pumps. Recovery of the event by recovery of offsite power or by manual control of the turbine driven auxiliary feedwater pump is unsuccessful.

GTRAN-008

The third highest sequence is GTRAN-008 which contributes 10.1% of the internal event CDF.

Sequence GTRAN-008 is a plant transient initiating event with successful reactor trip and the pressurizer PORVs are either not challenged or closed following steam or water relief. Secondary heat removal via the steam generators is successful; however a long term supply of auxiliary feedwater is unsuccessful. High pressure injection fails.

The primary initiating events that lead to this sequence are Loss of Offsite Power initiators LOSP-GR (47.3%), LOSP-PC (38.1%), LOSP-WI (9.4%) and Total Loss of ERCW (5.2%).

This sequence is dominated by station blackout cutsets initiated by a loss of offsite power with independent failures of the onsite electrical power system. Station blackout conditions result in failure of the motor driven auxiliary feedwater pumps and the high pressure injection pumps. The turbine driven auxiliary feedwater pump successfully removes decay heat until the condensate storage tank is exhausted. Swapover to ERCW is unsuccessful due to the loss of ERCW pumps upon the blackout. During a station blackout the CST can be refilled by the diesel driven fire protection pump. Core damage occurs in this sequence when the diesel driven pump fails or the operators fail to align it to the CST.

GTRAN-011

The fourth highest sequence is GTRAN-011 which contributes 8.9% of the internal event CDF.

Sequence GTRAN-011 is a plant transient initiating event with successful reactor trip and the pressurizer PORVs are either not challenged or closed following steam or water relief. The charging pumps continue to inject to the RCS, however core damage occurs when secondary heat removal via the steam generators fails and implementation of feed and bleed cooling is unsuccessful.

This sequence is dominated by initiating events: loss of a 120V vital instrument board (56.0%), loss of a vital battery board (14.8%) or loss of offsite power (14.7%). Auxiliary feedwater fails due to independent and consequential failures of electric power with failures of instrument power. Auxiliary feedwater also fails during transient initiated events due to pump maintenance alignments with pre-initiator flow path isolation errors.

SLOCAV-015

The fifth highest sequence is SLOCAV-015 which contributes 8.5% of the internal event CDF.

Sequence SLOCA-015 is a very small LOCA with failure of both CVCS and SI. Reactor trip is successful but the auxiliary feedwater system fails to remove decay heat resulting in core damage.

The primary initiating events that lead to this sequence are Loss of Offsite Power initiators LOSP-GR (36.1%), LOSP-PC (34.1%), LOSP-WI (10.6%) and Total Loss of ERCW (18.0%).

Similar to sequence SLOCA-024, this sequence is initiated by a RCP seal LOCA resulting from a station blackout condition, or loss of ERCW events with consequential seal injection and thermal barrier cooling failures, however the RCP seal leakage rate is only 21 gpm/pump.

Recovery of the event by recovery of offsite power or by manual control of the turbine driven auxiliary feedwater pump is unsuccessful.

GTRAN-010

The sixth highest sequence is GTRAN-010 which contributes 6.1% of the internal event CDF.

Sequence GTRAN-010 is a plant transient initiating event with successful reactor trip and the pressurizer PORVs are either not challenged or closed following steam or water relief. Secondary heat removal via the steam generators fails. The operators implement bleed and feed cooling using the charging pumps; however recirculation from the sump fails when high pressure recirculation is attempted.

Similar to sequence GTRAN-011 this sequence is dominated by initiating events: loss of a vital battery board (34.2%), loss of a 120V vital instrument board (33.0%) or loss of offsite power (15.8%). Auxiliary feedwater fails due to independent and consequential failures of electric power with failures of instrument power. Auxiliary feedwater also fails during transient initiated events due to pump maintenance alignments with pre-initiator flow path isolation errors.

High pressure recirculation failures are caused by instrumentation failures, failures of the RHR pumps or failure of the operator to manually align recirculation.

Since the CVCS system is successful, feed and bleed cooling failure is dominated by operator action failures.

SSBO-010

The seventh highest sequence is SSBO-010 which contributes 1.7% of the internal event CDF.

Sequence SSBO-010 is a secondary side break with successful high pressure injection via CVCS and successful AFW initiation. The operators identify and isolate the faulted steam generator, and cooldown using the intact steam generators. High pressure injection continues, and when SI is not terminated bleed and feed is established through the open PORV, however high pressure recirculation fails when the swapover to the containment sump is attempted.

High pressure recirculation failures are caused by failure of the operator to manually align recirculation, operator failure to restart the RHR pumps for HP recirculation, failures of the sump valves, or failure of RHR pumps due to room ventilation failures.

SLOCA-007

The eighth highest sequence is SLOCA-007 which contributes 1.7% of the internal event CDF.

Sequence SLOCA-007 is a small LOCA due to a small RCS cold leg break or a stuck open safety relief valve, with successful cold leg injection via the CVCS pumps. The Auxiliary Feedwater system provides flow to the steam generators to remove decay heat. When the

RWST reaches the switchover setpoint, the RHR/CVCS pumps fail to transfer to high pressure cold leg recirculation from the containment sump. The operators fail to cooldown and depressurize the RCS. The operators then also fail to refill the RWST to remain on high pressure injection, using the containment spray test line.

<u>SLOCAV-008</u>

The ninth highest sequence is SLOCAV-008 which contributes 1.4% of the internal event CDF.

Sequence SLOCAV-008 is a very small LOCA. The operators trip the reactor and cooldown using AFW but cannot continue steam generator cooling once the CST empties. Bleed and feed cooling is attempted, however both the CVCS pumps and the SI pumps fail resulting in core damage.

The very small LOCA is initiated by station blackout events (60.8%) or total loss of ERCW events (37.3) resulting in a 21 gpm/pump RCP seal LOCA. The initiating events also cause failure of the ECCS pumps.

<u>MLOCA-013</u>

The tenth highest sequence is MLOCA-013 which contributes 1.3% of the internal event CDF.

Sequence MLOCA-013 is a medium LOCA with failure of the CVCS and the SI pumps to inject. The operators use the AFW system to rapidly cooldown and depressurize to allow low pressure injection. Failure of the RHR system to provide LPI leads to core damage.

<u>SSBO-007</u>

The eleventh highest sequence is SSBO-007 which contributes 0.7% of the internal event CDF.

Sequence SSBO-007 is a secondary side break with successful high pressure injection via CVCS and successful AFW initiation. The operators identify and isolate the faulted steam generator, cooldown using the intact steam generators and terminate high pressure injection. The PORV fails to reclose when high pressure injection is stopped resulting in a loss of RCS inventory. The operators re-initiate high pressure injection and establish bleed and feed through the open PORV, however high pressure recirculation fails when the swapover to the containment sump is attempted.

<u>SLOCA-025</u>

The twelfth highest sequence is SLOCA-025 which contributes 0.7% of the internal event CDF.

Sequence SLOCA-025 is a small LOCA with failure of high pressure cold leg injection. The Auxiliary Feedwater system removes decay heat; however the operators can not cooldown to RHR conditions. Failure of high pressure and low pressure injection leads to core damage.

<u>ATWS-008</u>

The thirteenth highest sequence is ATWS-008 which contributes 0.6% of the internal event CDF.

Sequence ATWS-008 is transient event with failure of automatic reactor trip. AMSAC successfully initiates 50% of the auxiliary feedwater flow. RCS pressure boundary failure and subsequent core damage occurs due to insufficient PORV capability to mitigate the RCS pressure increase.

This sequence is dominated by the loss of a 120V vital instrument board (76.2%) or loss of a vital battery board (16.7%). The loss of vital 120V AC or 125 DC trains results in consequential failure to start of one train of Auxiliary Feedwater. The Unfavorable Exposure Time (UET) values determine the probability that pressure relief is insufficient.

Table 3-5
CDF Sequence Results

Sequence	Truncation	Number of Cutsets	Frequency (per reactor-year)
U1_ATWS-003	1.00E-16	336	6.29E-14
U1_ATWS-004	1.00E-12	2642	1.99E-07
U1_ATWS-007	1.00E-16	467	1.36E-13
U1_ATWS-008	1.00E-12	3348	2.30E-07
U1_ATWS-009	1.00E-12	7431	8.74E-08
U1_ATWS-010	1.00E-12	9245	8.22E-08
U1_ATWS-013	1.00E-18	1307	9.82E-15
U1_ATWS-014	1.00E-12	888	3.99E-08
U1_ATWS-017	1.00E-17	341	1.10E-14
U1_ATWS-018	1.00E-12	1028	3.00E-08
U1_ATWS-019	1.00E-12	1602	6.67E-09
U1_ATWS-020	1.00E-12	1745	6.27E-09
U1_GTRAN-003	1.00E-12	2162	1.58E-08
U1_GTRAN-004	1.00E-10	9	1.30E-09
U1_GTRAN-006	1.00E-14	2198	1.26E-10
U1_GTRAN-007	1.00E-12	1063	6.00E-09
U1_GTRAN-008	1.00E-11	10779	3.65E-06
U1_GTRAN-010	1.00E-13	200843	2.18E-06
U1_GTRAN-011	1.00E-12	24585	3.21E-06
U1_GTRAN-013	1.00E-13	3912	3.13E-09
U1_GTRAN-014	1.00E-12	18915	2.05E-07
U1_GTRAN-015	1.00E-11	39845	9.60E-06
U1_ISLM-003	1.00E-20	84	1.43E-14
U1_ISLM-004	1.00E-18	0	0
U1_ISLM-006	1.00E-17	1602	5.06E-13
U1_ISLM-007	1.00E-16	452	1.19E-12
U1_ISLM-009	1.00E-17	104	8.09E-09
U1_ISLM-012	1.00E-20	0	0
U1_ISLM-013	1.00E-18	0	0
U1_ISLM-015	1.00E-18	0	0
U1_ISLM-016	1.00E-20	8816	3.84E-16
U1_ISLM-018	1.00E-21	0	0
U1_ISLM-019	1.00E-18	211888	9.27E-11
U1_LLOCA-002	1.00E-12	212	8.30E-09
U1_LLOCA-003	1.00E-12	32	1.59E-09
U1_LLOCA-004	1.00E-12	160	1.81E-09
U1_LLOCA-005	1.00E-12	36	1.24E-10
U1_MLOCA-003	1.00E-12	619	4.09E-08
U1_MLOCA-004	1.00E-12	115	1.60E-07
U1_MLOCA-005	1.00E-15	229	1.72E-12
U1_MLOCA-008	1.00E-13	3227	2.75E-09
U1_MLOCA-009	1.00E-14	1830	1.56E-09
U1_MLOCA-010	1.00E-16	12251	7.81E-12

Table 3-5
CDF Sequence Results

Sequence	Truncation	Number of Cutsets	Frequency (per reactor-year)
U1_MLOCA-012	1.00E-15	2916	2.02E-11
U1_MLOCA-013	1.00E-12	9776	4.67E-07
U1_MLOCA-014	1.00E-12	1860	9.05E-09
U1_MLOCA-015	1.00E-12	6262	8.70E-09
U1_SGTR-003	1.00E-15	16	3.70E-14
U1_SGTR-006	1.00E-17	16	2.48E-16
U1_SGTR-009	1.00E-17	8	1.25E-16
U1_SGTR-012	1.00E-17	2	1.24E-14
U1_SGTR-014	1.00E-16	9260	6.15E-12
U1_SGTR-015	1.00E-13	220	4.34E-10
U1_SGTR-018	1.00E-19	36	7.95E-18
U1_SGTR-021	1.00E-20	0	0
U1_SGTR-024	1.00E-20	0	0
U1_SGTR-027	1.00E-20	25	1.64E-16
U1_SGTR-029	1.00E-18	3060	1.55E-14
U1_SGTR-030	1.00E-16	224	4.53E-14
U1_SGTR-033	1.00E-16	188	6.11E-11
U1_SGTR-034	1.00E-12	8	1.07E-10
U1_SGTR-035	1.00E-12	220	2.14E-09
U1_SGTR-036	1.00E-13	384	3.26E-08
U1_SLOCA-002	1.00E-15	1778	6.00E-11
U1_SLOCA-004	1.00E-16	705	4.93E-13
U1_SLOCA-005	1.00E-12	3942	2.24E-07
U1_SLOCA-007	1.00E-12	374	5.98E-07
U1_SLOCA-009	1.00E-14	845	2.09E-10
U1_SLOCA-010	1.00E-14	1483	1.23E-09
U1_SLOCA-012	1.00E-14	1942	2.71E-10
U1_SLOCA-014	1.00E-16	4187	3.87E-12
U1_SLOCA-015	1.00E-12	4986	5.91E-08
U1_SLOCA-017	1.00E-14	1541	4.85E-10
U1_SLOCA-019	1.00E-14	15	2.53E-13
U1_SLOCA-020	1.00E-16	514294	4.93E-09
U1_SLOCA-022	1.00E-16	6720	5.88E-12
U1_SLOCA-023	1.00E-13	641	3.06E-10
U1_SLOCA-024	1.00E-10	5052	9.73E-06
U1_SLOCA-025	1.00E-13	22593	2.42E-07
U1_SLOCA-026	1.00E-12	30285	2.03E-07
U1_SLOCA-17A	1.00E-15	892	3.10E-12
U1_SLOCAV-003	1.00E-14	544	2.01E-11
U1_SLOCAV-004	1.00E-15	1223	1.34E-11
U1_SLOCAV-006	1.00E-16	3630	9.50E-13
U1_SLOCAV-007	1.00E-13	2531	4.11E-09
U1_SLOCAV-008	1.00E-12	21691	4.90E-07
U1_SLOCAV-010	1.00E-13	873	8.04E-10

Table 3-5
CDF Sequence Results

Sequence	Truncation	Number of Cutsets	Frequency (per reactor-year)
U1_SLOCAV-011	1.00E-14	6115	2.24E-09
U1_SLOCAV-013	1.00E-14	524	1.76E-11
U1_SLOCAV-014	1.00E-14	176232	7.13E-08
U1_SLOCAV-015	1.00E-12	101393	3.06E-06
U1_SSBI-003	1.00E-18	0	0
U1_SSBI-004	1.00E-13	148	2.88E-10
U1_SSBI-005	1.00E-15	2292	4.12E-11
U1_SSBI-007	1.00E-12	528	2.37E-08
U1_SSBI-008	1.00E-12	456	5.80E-09
U1_SSBI-010	1.00E-12	328	6.26E-08
U1_SSBI-013	1.00E-14	44	1.17E-12
U1_SSBI-014	1.00E-15	444	5.82E-12
U1_SSBI-016	1.00E-15	3605	1.63E-10
U1_SSBI-017	1.00E-14	484	1.42E-10
U1_SSBI-020	1.00E-17	1128	4.20E-14
U1_SSBI-021	1.00E-17	3276	2.03E-13
U1_SSBI-022	1.00E-18	832	5.20E-15
U1_SSBI-024	1.00E-17	35928	1.07E-11
U1_SSBI-025	1.00E-14	12	2.26E-13
U1_SSBI-027	1.00E-15	1844	2.59E-11
U1_SSBI-030	1.00E-19	0	0
U1_SSBI-031	1.00E-19	35340	3.28E-12
U1_SSBI-033	1.00E-18	10340	7.01E-14
U1_SSBI-034	1.00E-17	3916	2.11E-13
U1_SSBI-036	1.00E-18	1132	1.80E-11
U1_SSBI-037	1.00E-18	467998	1.19E-09
U1_SSBI-038	1.00E-15	184	2.42E-11
U1_SSBO-003	1.00E-16	8	1.25E-15
U1_SSBO-004	1.00E-16	28	1.49E-14
U1_SSBO-005	1.00E-17	40	7.53E-16
U1_SSBO-007	1.00E-15	37024	2.54E-07
U1_SSBO-008	1.00E-12	924	5.95E-08
U1_SSBO-010	1.00E-12	880	6.29E-07
U1_SSBO-013	1.00E-18	16	3.15E-17
U1_SSBO-014	1.00E-17	4	5.50E-17
U1_SSBO-016	1.00E-16	65557	1.88E-09
U1_SSBO-017	1.00E-19	0	0
U1_SSBO-020	1.00E-19	0	0
U1_SSBO-021	1.00E-19	4	4.73E-19
U1_SSBO-022	1.00E-19	0	0
U1_SSBO-024	1.00E-17	133744	1.13E-10
U1_SSBO-025	1.00E-17	5884	8.81E-12
U1_SSBO-027	1.00E-19	0	0
U1_SSBO-030	1.00E-19	0	0

Table 3-5
CDF Sequence Results

Sequence	Truncation	Number of Cutsets	Frequency (per reactor-year)
U1_SSBO-031	1.00E-19	0	0
U1_SSBO-033	1.00E-16	1763	8.50E-13
U1_SSBO-034	1.00E-17	20698	4.04E-12
U1_SSBO-036	1.00E-18	72	1.54E-16
U1_SSBO-037	1.00E-17	508559	1.26E-08
U1_SSBO-038	1.00E-18	134789	2.51E-10

No single core damage sequence was found to dominate the total frequency of core damage. In fact, the largest individual core damage sequence contributes 3.02E-6 per reactor year to the total CDF (approximately 9.2% of the total CDF). As is typical with linked event tree PRAs, a large number of sequences make up the total CDF. Table 3-6 provides information on the distribution of core damage sequences across the frequency range.

Table 3-6
Breakdown of Core Damage Sequences in Each Frequency Range

Frequency Range (events per year)	Number of Sequences	Percentage of CDF
$> 10^{-6}$	3	15.7%
10^{-7} to 10^{-6}	19	11.9%
10^{-8} to 10^{-7}	278	18.8%
10^{-9} to 10^{-8}	3771	31.3%
10^{-10} to 10^{-9}	16128	15.1%
10^{-11} to 10^{-10}	56932	5.5%
10^{-12} to 10^{-11}	184855	1.6%

3.3. Success Criteria (SC)

A success criteria analysis was performed to determine the minimum requirements for each function (and ultimately the systems used to perform the functions) to prevent core damage (or to mitigate a release) given an initiating event. The requirements defining the success criteria are based on engineering analyses that represent the design and operation of the WBN. It was assumed for this analysis that Unit 2 would operate similarly to Unit 1. For a function to be successful, the criteria are dependent on the initiator and the conditions created by the initiator. The computer codes used to perform the analysis for developing the success criteria such as MAAP4.0.7 are validated and verified for both technical integrity and suitability to assess plant conditions for the reactor pressure, temperature, and flow range of interest, and that they accurately analyze the phenomena of interest. Calculations were performed by personnel who are qualified to perform the types of analyses of interest and are well trained in the use of the codes.

The objectives of the success criteria element are to define the plant-specific measures of success and failure that support the other technical elements of the PRA in such a way that overall success criteria are defined to determine the core damage frequency and large early release frequency for each unit. Success criteria are defined for critical safety functions, supporting systems, structures, components and operator actions necessary to support accident sequence development.

During risk model development, existing safety analyses were reviewed, and specific thermal hydraulic analyses were performed to establish realistic success criteria for the mitigating systems and operator actions that are modeled in the PRA. In some cases, conservative success criteria were used to simplify the models or their supporting analyses when the degree of conservatism was determined not to have an important impact on the overall PRA results.

Functional success criteria are developed from the overall plant success criteria in order to provide insights regarding the use of individual system success criteria. The functions that have been identified as important safety functions for accident prevention or mitigation in the Level I PRA are the following:

- Reactivity Control
- Inventory Control
- Heat removal

The success criteria for these major plant safety functions with respect to each initiating event were determined through thermal hydraulic analysis. These success criteria are presented in the Success Criteria Notebook of the WBN PRA. An example of the success criteria is presented below:

LLOCA is successful with 3 of 3 accumulators dumping to the intact legs, 1 of 2 RHR pumps providing low pressure ECCS injection to 3 of 3 intact legs, 1 of 2 RHR pumps providing low pressure ECCS recirculation and successful transfer to hot leg recirculation with 1 of 2 RHR pumps providing flow to 2 of 4 legs and 1 of 2 SI pumps to 2 of 4 legs.

The thermal hydraulic analyses were primarily performed using the MAAP code. The following is the definition for core damage applied for the Watts Bar MAAP analysis:

> For success criteria quantifications using MAAP, core damage is defined to occur when the maximum reactor core temperature (parameter TCRHOT) is greater than 1800°F, when the center temperature monitor (parameter TCLN (13)) is greater than 1800°F, or temperature of gas in the upper plenum of the core (parameter TGUP) is greater than 1200°F, whichever value is reached at the shortest time after reactor trip. TCRHOT was found to be the limiting case. In most scenarios, it reaches 1800°F before the other core damage temperature parameters (TCLN (13) and TGUP) reach their damage state.

Other definitions used were:

1. Safe, stable state: RCS conditions are controllable at or near desired values. No operator actions or additional equipment is required to maintain these RCS conditions.

2. Mission Time: The time period that a system or component is required to operate in order to successfully perform its function.

   Success is defined as the plant reaching a stable plant condition within 24 hours after the initiating event. If core damage is not reached within 24 hours, the MAAP quantification is aborted, and the end state is "success".

   If the plant does not reach a safe stable state, the analysis may be continued past 24 hours. A safe, stable state is when the RCS conditions are controllable at or near desired values. No operator actions or additional equipment is required to maintain these RCS conditions.

There are three sizes of RCP seal LOCAs considered in this analysis: 21 gpm, 182 gpm and 480 gpm per pump. The 76 gpm per pump leakage has been subsumed into the 182 gpm leakage for simplicity because it has a very small probability of occurrence. The 21 gpm seal LOCA per pump has been classified as a SLOCAV; it is less than the normal charging function (100 gpm total). The 182 gpm seal LOCA per pump has been classified as a SLOCA; it is greater than the normal charging function but decay heat cannot be removed entirely by break flow and auxiliary feedwater is required for success. The 480 gpm seal LOCA per pump has been classified as a MLOCA. Although its equivalent break size is less than the lower bound MLOCA 2 inch criteria (actually 0.4464 in^2 per pump), the 480 gpm seal LOCA meets the MLOCA requirement of not requiring AFW for accident mitigation.

3.4. Systems Analysis (SY)

The systems analysis task assesses the likelihood that a system will fail to meet its functional success criteria defined by the plant response event tree model top events. The method by which this is accomplished is fault tree analysis. All systems that are required for accident mitigation and those systems supporting accident mitigating systems have been re-analyzed as part of the conversion from RISKMAN to CAFTA. An individual System model and Notebook was developed for each of the systems modeled below:

- Auxiliary Feedwater System
- Component Cooling Water System
- Condensate and Feedwater System
- Containment Spray System
- Containment Systems
- Chemical and Volume Control System
- Electric Power System
- Electric Power Recovery
- Essential Raw Cooling Water System
- Engineered Safety Features Actuation System
- Main Steam System
- Plant Compressed Air System
- PORVs and Safety Valves
- Reactor Coolant Pump Seals and Thermal Barrier
- Reactor Protection System
- Residual Heat Removal System
- Safety Injection System
- Steam Generator Isolation System

The System Analysis notebooks contain all information and documentation necessary to provide a single source of reference regarding individual system treatment to facilitate future updates. Plant information was reviewed to define system components and boundaries, dependencies on other systems, instrumentation and control requirements, test and maintenance requirements and practices, operating limitations, component operability and design limits, procedures for the operation of the system during normal and accident conditions, and system configuration during normal and accident conditions. This information was also used to determine component exposure times, interfaces and dependencies, and to identify environmental issues.

System failures modeled may result from independent or common cause equipment hardware failures, human error, or from combinations of equipment failure, human errors, maintenance actions, and testing activities. Specific system failures may affect the availability of other systems (e.g., support system failures and, in limited cases, frontline systems), or they may directly affect the ability to mitigate the consequences of transient events; e.g., frontline system failures. In addition, analyses were reviewed to define HVAC requirements. Flow paths that can divert the system fluid from the intended destination were reviewed to determine if they needed to be modeled as a failure mode.

Active and passive failures affecting system operability were included in the system model. Active failures typically affect pumps, valves, relays and air compressors. Passive failures typically affect heat exchangers, valves not required to change position and tanks. In general normally operating equipment basic event unavailabilities are calculated using the failure rate multiplied by the mission time (CAFTA method 1), and standby equipment basic event unavailabiliies are calculated using the failure rate times the exposure time divided by 2 (CAFTA method 2). Demand failures of equipment which must start or change state are calculated using the failures per demand value multiplied by the number of demands required. Unless otherwise stated in system notebooks a mission time of 24 hours was used in the model.

The repair of hardware failures is not modeled in the WBN PRA except where the probability of repair is justified through an adequate analysis or examination of data.

The manual actions required to bypass failed systems or recover malfunctioning equipment are modeled as described in the Human Reliability Analysis Notebook. Recovery actions were only modeled if they are procedure directed (i.e., Emergency Operating Procedures [EOP]) and can be accomplished in the time available. The recovery of offsite power is modeled as described in the Electric Power Recovery Notebook.

The possibility of latent (pre-initiating event) errors resulting in mispositioning/ misalignment of valves for standby systems or trains was considered. The possibility of common cause miscalibration of equipment affecting multiple systems or trains was also considered

A common cause group of components has a significant likelihood of experiencing one or more common cause events affecting two or more components in that group. As stated above common cause failures are included in the model. The following is a list of considerations that were used in the development of common cause groups:

1. Same design
2. Same hardware
3. Same function
4. Same installation, maintenance
5. Same procedures
6. Same system/component interface
7. Same environment

Identical, non-diverse, and active components that are used to provide redundancy, were considered for assignment to common cause groups in the PRA model.

Table 3-7 is an example list from the SI system of the common cause failure types that have the CCF factors explicitly calculated:

Table 3-7- Example of Common Cause Failure Types	
System	Common Cause Failure Type
High Pressure Safety Injection System	• Motor-driven pumps fail to start • Motor-driven pumps fail to run • Motor-operated valves fail to open • Motor-operated valves fail to close • Motor-operated valves fail to remain closed • Air-operated valves fail to close • Check valves fail to open • Check valves fail to close Check valves fail to remain closed

To support the development of the system analyses, each system modeled in the PRA was walked down by a group of PRA analysts to evaluate:

1. Component location and operational status;
2. Susceptibility to flooding and spray;
3. Environmental considerations such as heat sources, ventilation, and steam/humidity sources;
4. Considerations for manual operation; and
5. Physical characteristics of the room/area

Any plant design changes made to unit 1 since the last PRA model update were incorporated into the system models. It was assumed that these design changes will also be made on Unit 2. Within the PRA model, the basic events were identified to include the WBN unique identifiers to support future applications such as online risk management. The simplified drawings included in the notebooks for each system were re-drawn to match the current plant configuration and reference the revision of the corresponding WBN drawings reviewed. All components included in the PRA models are represented in the simplified drawings. Any non-modeled components represented in the drawing are annotated as such. The simplified drawings are included as Appendix C.

Each system notebook was reviewed by the responsible System Engineer(s) at WBN. A PRA analyst also interviewed the respective System Engineers. The purpose of the interview was to:

- Ensure system modeling in the PRA is consistent with the as-built, as-operated plant
- Ensure potential initiating events have not been overlooked
- Ensure system operating experience is properly considered and documented in the PRA

All of the success paths for the SBO depend on recovery of at least one emergency AC power Bus. The timing of the recovery is the success variable. The following is a brief description of the functional success criteria for the SBO:

Reactivity Control is provided by control rod insertion with sufficient shutdown margin to maintain subcriticality.

Inventory Control until power can be recovered is provided by RCS pressure reduction to minimize RCS inventory loss through PORVs and the RCP seals.

Heat Removal is provided by turbine-driven AFW flow to the SGs.

In SBO sequences, there is one unique success path: auxiliary feedwater, AC power recovery, either charging pump or SI, as well as long term heat removal using either auxiliary feedwater to the unaffected SGs or via residual heat removal (RHR).

Factors that influence the time available to restore AC power include:

The availability of 125V DC power (i.e., battery lifetime)

The length of time to core damage due to reactor coolant pump seal leakage

The length of time of Power Operated Relief Valve (PORV) discharge following a loss of all on-site AC power. Coolant inventory loss out the PORV would occur during station blackout after the time of steam generator dryout for sequences in which the steam driven auxiliary feedwater pump fails.

The key to having LOOP cutsets avoid core damage is the off-site power non-recovery probability (OSPNR probability). The likelihood of restoring AC power from off-site is largely a matter of time. The time available, in turn, depends on the reliability of equipment needed to operate in the absence of off-site AC power, e.g., diesel generators, turbine-driven auxiliary feedwater. The amount of time each mitigating feature operates is characterized by a time-dependent probability distribution function and the total amount of time gained from temporary success of on-site systems. The OSPNRs are best estimated by convoluting the probability density functions (pdfs) of the system components that operate after event initiation with the pdfs constructed to represent the likely duration of the LOOP. The convolution process is described in the Electric Power Recovery Notebook.

An importance analysis of plant system and component failure modes to the total CDF was also performed. The system importance is measured in terms of the reduction of the total CDF that could be achieved if the hardware never failed (i.e. "Risk Reduction Worth") and presented in Table 3-8.

Table 3-8 Important Systems			
System	**System Number**	**RRW**	**Percent Contribution to CDF**
Ventilating	30	1.21	17.63%
Standby Diesel Generator	82	1.18	14.96%
Main & Auxiliary Feedwater	3	1.10	8.75%
125V DC Vital Power	236	1.05	4.79%
Control Air	32	1.03	2.69%
Essential Raw Cooling Water	67	1.03	2.60%
6.9KV Shutdown Power & Load Shed Logic	211	1.02	2.17%
Reactor Protection	99	1.02	1.77%
Reactor Coolant	68	1.02	1.57%
120V AC Vital Power	235	1.01	0.57%

3.5. Human Reliability (HR)

The purpose of the Human Reliability Analysis (HRA) is to identify human interactions that could play a role in the accident sequences, and to provide an estimate of the probabilities for failure events corresponding to these interactions. The HRA for the WBN PRA was completely re-evaluated as part of the dual unit model. For the Unit 2 model it was assumed that operating procedures and operator training for Unit 2 would be similar to Unit 1.

The following four categories of human error probabilities (HEPs) were developed to support the WBN PRA.

Category 1 – Pre-Initiator: The first category accounts for latent human interactions that occur during routine activities including testing, maintenance, or calibration activities that are performed before the initiating event. The pre-initiator actions could impact the availability of required equipment to mitigate an accident. Category 1 corresponds to type A, pre-initiating event interactions in EPRI TR-100259.

Category 2 – Initiator and Post-Initiator: The next category includes actions that can cause an initiating event and actions required in response to an initiating event. Category 2 corresponds to type B, initiating event-related interactions and type C, post-initiating event interactions in EPRI TR-100259.

Category 3 – Recovery: The third category includes recovery actions that can be taken to restore functions, systems or components. For the WBN PRA, recoveries following a Loss of Offsite Power (LOOP) initiating event are credited. One recovery is documented in this Notebook. The HFE is HAOSBF, Locally operate TD AFW pump after battery depletion. A screening value is assigned to this HFE in Appendix B of this HRA Notebook.

Category 4 – Flooding: Actions taken in response to flooding initiating events comprise the final category. These actions are a special type of post-initiator action and are taken in response to a flooding initiator.

All HEPs were generated by entering information gathered from procedures, plant walkdowns and operator interviews into the EPRI HRA Calculator (verified and validated). The pre-initiator actions were evaluated using the Technique for Human Error Rate Prediction (THERP). This was done based on the analyst's understanding of the physical plant configuration and also taking into account the results of interviews conducted with Auxiliary Unit Operators (AUOs), Electrical and

Mechanical Maintenance personnel and Maintenance Planning personnel. These interviews were conducted for each of the pre-initiator HFEs in the WBN PRA. The initiator, post-initiator, and flooding actions were evaluated using the Cause Based Decision Tree Method (CBDTM)/THERP. Operations personnel were interviewed to assess cues and indications available to the operator, the timing, applicable procedures and operator training requirements, level of stress, location of specific operator actions, accessibility of actuation equipment during accident conditions, and number operators required for specific tasks.

New industry methods and philosophy for human reliability analysis were incorporated into the WBN HRA including addressing and documenting dependency between actions.

The results from the HRA calculator are input into the WBN PRA model.

An importance analysis was performed to determine contributions from sequences grouped by the occurrence of specific operator actions. The importance measure used here is Risk Reduction Worth. Table 3-9 summarizes the important operator action failures that individually contribute at least 0.5% to the total CDF. The percent contribution to total CDF is calculated as follows:

PCCDF = [1 - 1/RRW] x 100%
or
PCCDF = [F-V] x 100%

Where:
PCCDF = Percent Contribution to Core Damage Frequency
F-V = Fussell-Vesely

Table 3-9
Important Operator Actions

HRA Name	Description	RRW	Percent Contribution to Core Damage
HAOSBF	Steam generator feed with manual level control fails	1.073	6.78%
HAFR1	Restore AFW control following initiator and loss of air	1.049	4.70%
WHEMDA_1	Motor Driven AFW Pump Train A Isolation Test Error	1.042	4.04%
WHEMDA_2	Motor Driven AFW Pump Train B Isolation Test Error	1.041	3.90%
HCCSR4	Align & Initiate Alternate Cooling to 1A-A CCP, 1B-B failed	1.032	3.13%
HAAF3	Align HPFP to provide makeup after CST depletion (SBO)	1.031	3.00%
HAOB2	Establish RCS Bleed and Feed cooling given no CCPS running	1.017	1.70%
WHEAFW	Turbine Driven AFW Isolation Test Error	1.017	1.67%
HAAEIE	Start standby ERCW pump - operating pump fails - normal ops	1.01	1.00%
FLAB4F	Isolate break in HPFP line (supplied by RCW - HPFP diesel pump does not start)	1.01	0.98%
SSIOP	Terminate Safety Injection to prevent PORV water challenge	1.009	0.86%
HARR1	Align high pressure recirculation, given auto swapover works	1.007	0.70%

Table 3-9 Important Operator Actions			
HRA Name	Description	RRW	Percent Contribution to Core Damage
HART1	Manually trip reactor, given SSPS fails	1.005	0.54%

The following sensitivity analyses were performed and documented in the Sensitivity and Uncertainty Notebook:

1. The CBDTM/THERP methodology was used to evaluate all post-initiator human actions. For some actions with short time frames, the HCR/ORE/THERP method resulted in higher HEPs. The impact on core damage frequency (CDF) of using the later values was evaluated.
2 The impact on CDF due to large changes in the HEPs was determined by recalculating CDF with all HEPs set at the 5^{th} and 95^{th} percentiles of the 90 percent confidence intervals.

3.6. Data Analysis (DA)

The Data Analysis Notebook provides a listing of all variables used in the WBN PRA. The content of the notebook is largely based on information provided in NUREG/CR-6928 with the incorporation of plant specific data. The plant specific data is based on the evaluation and categorization of information compiled for WBN Unit 1 during the time period January 1, 2003 through March 31, 2008. The compiled information includes failure Cause Determination Evaluation (CDE) reports, unavailability due to test and maintenance, component demands, and component run times. The end date for the data window coincides with the start date of Cycle 9 at WBN.

The types of variables included in the WBN PRA are component failure data, maintenance data, operator action failure data, common cause data, exposure times, initiating event data, and internal flooding data. The derivations of the component failure data, maintenance data, and common cause data are included in the Data Analysis Notebook. The distributions have been Bayesian updated using Maintenance Rule Data.

The remainder of the data is documented in separate notebooks as shown below:

Data Type	Documentation
Operator Action Failure Data	Human Reliability Analysis Notebook
Exposure Times	Individual System Notebooks
Initiating Event Data	Initiating Events Notebook
Internal Flooding Data	Internal Flooding Notebook

The EPRI R&R workstation uses a master database file to store probability data used during the final logic model development and quantification processes. The master database file used by CAFTA is typically given an ".rr" filename extension. It consists of the following database tables:

- BE: Stores basic event names. Typically, basic events are composed of a component type, a failure mode, a system, and component number codes. The System Summary Notebook contains more information about the basic event naming convention. Events stored in this table can also take the form of being initiating events, Human Reliability Analysis events, and logic flags. Component boundaries are defined in the Data Analysis Notebook.
- GT: Stores fault tree gate information.
- TC: Stores probability "type code" information. Type codes are variables that are used to assign probability data to basic events. They ensure consistency in assigning data to basic events.

A master database file was created by merging the individual databases created during the system analysis phase into one BE table and by merging the master TC table from the database documented in the Data Analysis Notebook . The system databases are documented in their associated system notebooks.

It should be noted that a complete and single source type code database was used to develop and quantify the system-level models. In this manner, the development and control of type codes can be traced to the Data Analysis Notebook.

Added to the TC and BE tables were the HRA events and HEPs from the HRA notebook, and the initiating events and initiating event frequencies developed in IE notebook. When combined, the basic event failure probabilities, initiating event frequencies, and HRA events make up the probability data used in the final quantification model.

As stated previously, failure probability data integrated into the final master database file comes from different sources. A summary of the methodology used to develop each follows:

1) <u>Initiating Event Frequencies</u>

   Prior data was collected from generic data sources. Plant specific data was also collected from Watts Bar License Event Reports for the period January 1, 2003 to March 31, 2008. Bayesian updating guidelines were then used to determine which prior data was to be Bayesian updated to get posterior data to develop final IEFs. In addition to the IEFs developed from generic data sources, plant specific system analyses were performed to derive IEFs.

2) <u>System Analysis Failure Probability Data</u>

   The data analysis is comprised of determining required parameters, component failure parameters, component and equipment unavailability, and common cause failure data. The required parameters for the data analysis include component failure modes and component boundaries modeled in the WBN PRA. The basic event failure probabilities are calculated by multiplying hourly rates by the mission time, by multiplying demand failure rates by the number of demands required during the mission time or by multiplying the standby failure rate by one-half the exposure time. The mission times, exposure times and demands are identified in Appendix C-2 of all system notebooks. Generic failure parameters were identified and plant specific component failures were collected for the type codes in the WBN PRA. The same Bayesian updating rules used in IE analysis was used in Data Analysis. The Multiple Greek Letter factors were also developed in the data analysis and included in the master type code table. MGL factors are assigned to each CCF group and CAFTA automatically calculates the probability of the CCF basic event using equations based on the group size and the number of failed components.

3) <u>Human Failure Events and Human Error Probabilities</u>

Four categories of HEPs were developed to support the PRA including pre-initiator, initiator and post-initiator, recovery, and flooding. Operator interviews were conducted to gain insights from the operators based on their knowledge and experience. HEPs were generated by entering information from procedures, plant walkdowns and operator interviews into the EPRI HRA Calculator.

4) <u>Common Cause Failure Data</u>

Components of similar manufacture and functions are subject to CCF. Common cause failure can result in failure of a system when identical, non-diverse, and active components are used to provide redundancy. Failure of two or more components in a common cause group can occur if they are of the same design, perform the same function, share the same installation and maintenance procedures, and are located in the same location or environment. The CCF groups for each system are identified during system analysis using the methodology described in NUREG/CR-5485. The system analysis notebooks provide a complete listing of the CCF groups as well as the components that are part of the group. If the component is subject to multiple failure modes, a CCF group is created for each failure mode. After the CCF groups are identified, the CAFTA CCF tool was used to create the basic events that represent combinations of common cause failures for each group. CAFTA does not determine the applicable error factor for the CCF basic event probability. This is addressed in the uncertainty and sensitivity analysis section of the WBN Quantification notebook.

The Multiple Greek Letter (MGL) method which is described in NUREG/CR-5485 was then used to determine the probability of each common cause basic event. In order to do this, MGL factors are assigned to each CCF group and CAFTA automatically calculates the probability of the CCF basic events using equations based on the group size and the number of failed components. The MGL factors used in the WBN PRA were derived from the values in WCAP-16672-P. These values are applicable to components in Pressurized Water Reactors (PWRs) with Nuclear and Steam Supply Systems (NSSS) designed by either Westinghouse or Combustion Engineering

3.7. <u>Internal Flooding (IF)</u>

The Internal Flooding Notebook contains an evaluation to identify flood -induced vulnerabilities at WBN. The analysis was performed to determine the frequency and consequences of internal flooding events and upgrades the WBN Unit 1 Internal Flooding Probabilistic Risk Assessment (IF-PRA) to be consistent with the draft EPRI guidance, "Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment", and the Internal Flooding portions of the joint ANS/ASME PRA standard and Regulatory Guide 1.200.

The purpose of the IF-PRA is to identify all significant potential flood sources which can produce risk significant event sequences in the PRA. This involves the characterization of flood-induced event sequences, an assessment of the flood initiating event frequencies, and an impact assessment so that the resulting flood propagation pathways are identified and account for plant specific spatial dependencies. Pressure boundary failure of piping or other passive, non-piping components, and inadvertent or spurious system or component actuations (e.g., maintenance-induced activities) could lead to localized or global flooding causing failures that affect plant safety. An objective of the IF-PRA is to evaluate flood-induced impacts on Structures, Systems and Components (SSCs) important to safety in such a way that:

- Water sources within the plant that could create adverse conditions and affect the plant mitigating equipment are identified.
- The spray/flood scenarios that contribute significantly to Core Damage Frequency (CDF) or Large Early Release Frequency (LERF) are identified and quantified.

At each level of the flood hazard evaluation different types of passive component pressure boundary failures were considered including the following categories of loss-of-fluid events. These include sprays, floods, major floods and High Energy Line Breaks (HELB) events.

The following key tasks were completed as part of the IF-PRA:

1. **Identify Flood Areas and SSCs** - During this task the independent flood areas of the plant and the SSCs located within these areas were identified. A flood area is defined as a physically separate area that can be considered independent of other areas in terms of the potential for internal flood effects and flood propagation.

   The major structures at WBN are two Reactor Buildings, a shared Turbine Building, a shared Auxiliary Building, a shared Control Building, a shared Service and Office Building, two Diesel Generator Buildings (one hosting the four active diesel generators and one hosting the abandoned in place fifth diesel generator), an Intake Pumping Station, a shared Condenser Circulating Water (CCW) pumping station, a shared Makeup Water Treatment Plant, a Condensate Demineralizer Waste Evaporator building and two natural draft Cooling Towers.

   Components that have been retained as vulnerable to flooding-related effects have been associated with the correspondent spatial information in terms of room number and elevation. A summary of all components modeled in the PRA and their physical location is contained in the Internal Flooding Notebook Appendices A and B.

2. **Identify Flood Sources** - During this task the potential flood sources in each flood area of the plant and their associated flooding mechanisms were identified. A list of plant fluid systems was identified using the information provided in plant documents; location of flood sources was defined through a review of the MELB analysis, which provides a comprehensive catalogue of potential flood sources in key location of the plants. Additionally, mechanical drawings were reviewed to confirm presence of fluid systems in specific areas. IF-PRA specific walkdowns were then performed to confirm the information collected, was as described. The following table provides a summary of the flooding sources analyzed.

Table 3-10: Summary of Flood Sources Analyzed in IF-PRA

System #	System	Operating/ Standby	Inventory (gal)	Comments	Reactor trip/shutdown
1	Main Steam and Ancillary Steam Systems	Operating		Treated in baseline PRA as SLBOC	Steam Line break outside containment will induce a reactor trip.
2/3	Main Feedwater and Condensate systems	Operating		Treated in baseline PRA as TLMFW	Secondary line breaks outside containment will induce a reactor trip with loss of main feedwater.

Table 3-10: Summary of Flood Sources Analyzed in IF-PRA

System #	System	Operating/ Standby	Inventory (gal)	Comments	Reactor trip/shutdown
3a	Auxiliary Feedwater System	Standby	395,000		Break in the Auxiliary Feedwater line will not induce an automatic reactor trip. LCO-3.7.6 allows for up to 7 days to restore CST level above 200,000 gal, provided that ERCW is available as a backup.
24	Raw Cooling Water System	Operating	Infinite	RCW flow rate is bounded by CCW flow rate for flood in Turbine Building.	Complete loss of RCW can induce turbine and reactor trip. If stator outlet temperature reaches 90°C (194°F) for 45 seconds with unit load greater than 15%, the turbine will trip. If the turbine trips above P-9 (50% power), the reactor will trip)
26	High Pressure Fire Protection System	Standby pressurized	Infinite		No reactor trip or emergency shutdown is expected following a loss of fire protection system.
27	Condenser Circulating Water System	Operating	Infinite	Includes piping and expansion joints (2 for water box)	Loss of Condenser Circulating Water System is expected to induce an initiating event.
29	Potable Water Distribution System	Operating	Infinite		Loss of the potable water system will not result in reactor trip or immediate shutdown need.
44	Building Heating System	Operating			Loss of the Building heating system will not induce any reactor trip or immediate shutdown.
59	Demineralized Water System	Operating	500,000	Small bore pipes	Loss of the Demineralized Water System will not induce any reactor trip or immediate shutdown.
62	Chemical Volume and Control System	Operating		Letdown orifices limit flow rate to 75 gpm in the section upstream of the VCT.	Loss of any portion of the purification/charging portion of the CVCS, including seal injection, is expected to induce an initiating event. Loss of the portion of the CVCS outside the purification/charging loop is not expected to induce any plant trip or forced shutdown.

Table 3-10: Summary of Flood Sources Analyzed in IF-PRA

System #	System	Operating/ Standby	Inventory (gal)	Comments	Reactor trip/shutdown
63	Safety Injection System	Standby	380,000		Loss of one train of Safety Injection System is not expected to induce a forced shutdown. LCO 3.5.2 allows for up to 72 hours to re-align the unavailable train of Emergency Core Cooling System (ECCS). A break in the SIS line that has the potential for significant depletion of the RWST will result into entering TechSpec 3.5.4, which will require immediate action. Immediate shutdown of the reactor is postulated.
67	Essential Raw Cooling Water System	Operating	Infinite		Loss of either one train of ERCW would result in a reactor trip consistent with IE %1PLERCW, %2PLERCW and %0TLERCW.
70	Component Cooling System	Operating	500,000	Limited auto-makeup from Demineralized Water Storage Tank.	Loss of either one train of CCS would result in a reactor trip.
72	Containment Spray System	Standby	380,000		Loss of one train of Containment Spray System is not expected to induce a forced shutdown. LCO 3.6.6 allows for up to 72 hours to re-align the unavailable train of CSS. A break in the CSS line that has the potential for significant depletion of the RWST will result into entering TechSpec 3.5.4, which will require immediate action. Immediate shutdown of the reactor is postulated.
74	Residual Heat Removal System	Standby	380,000		Loss of one train of RHR is not expected to induce a forced shutdown. LCO 3.5.2 allows for up to 72 hours to re-align the unavailable train of ECCS. A break in the RHR line that has the potential for significant depletion of the RWST will result into entering TechSpec 3.5.4, which will require immediate action. Immediate shutdown of the reactor is postulated.
78	Spent Fuel Pool Cooling System	Operating	53,300		Loss of the SFPCS is not expected to induce reactor trip or immediate shutdown.

3. **Plant Walkdowns** - Plant walkdowns were performed and the walkdowns included identification/confirmation of:

   - Flood sources (piping, tanks, etc.)
   - Potential targets (PRA-related equipment, electrical cabinets and other components)
   - Barriers to flood propagation (curbs, doors, fire dampers, etc.)
   - Floor drains

   Walkdowns addressed the Auxiliary Building, the Control Building, the Turbine Building, the Diesel Building, the Intake Pumping Station and the Water treatment Plant. Operating limitation and RADCON limitation restricted access to a number of rooms in the Auxiliary Building; Unit 2 rooms were accessed instead of Unit 1 rooms in case of operating restriction.

4. **Qualitative Screening Assessment** -This task performed a screening evaluation of all areas of the plant based on criteria that consider three aspects of flood area importance in IF-PRA: 1) the sources of flooding; 2) the flood propagation pathways; and 3) the consequences of flooding in terms of flood initiating events and the impacts on SSCs that are needed to prevent core damage and large early release in response to the internal flooding initiating events. The screening criteria used are consistent with Section 3.4 of the EPRI guideline.

5. **Characterize Flood Scenarios** - This task developed the potential flooding scenarios for each flooding source not screened out previously by identifying the flood source and mode (e.g., spray or flood), the propagation paths of the fluid and the affected SSCs. The effects of spraying, local or global flooding on plant operability and safety and the manual and automatic responses to an impending or imminent flood event were considered.

6. **Flood Initiating Events Analysis** - This task identified the flooding-induced initiating events and estimated their frequencies. The majority of initiating events involve some form of passive component failure, but maintenance-induced and other human error-induced events were also considered. The IF-PRA developed for WBN is limited to the at-power operating mode and therefore includes initiating events in which reactor trip is induced by a flood or spray from the at-power state; nevertheless, even when the flood does not directly cause an initiating event, if there is a need for immediate plant shutdown from the plant operating state, then the plant shutdown event constitutes the initiating event.

7. **Flood Consequence Analysis** - For each IF initiating event identified in Task 6 a flood consequence analysis was performed. The purpose of this is to evaluate the impact on equipment, including failures by submergence and spray. Consequences of a flooding event can be characterized as direct or indirect effects. A direct effect represents the functional impact on the fluid system that experiences the breach in pressure boundary. A direct effect can be manifested as a total or partial loss of system function. An indirect effect represents the impact resulting from the release of water or high energy fluid (i.e., steam) on components located in the originating flood area and along the entire associated propagation path. An indirect effect results in flood-induced failure of components because of spray or submergence. Table 3-11 provides a summary of this analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
Flood events induced by MFW/AFW (excluded TB)			
%0FLAFW1	Flood event induced by Unit 1 AFW line break in room 692.0-A1, 713.0-A1, 737.0-A1 or 737.0-A3	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). AFW system impacted by CST-A unavailability (modeled through mapping to BE TKPRP1TANK00200229). ERCW backup supply to AFW-1 is not possible due to the break in the AFW line (modeled by mapping to spuriouse closure of valves 1-67-923A and 1-67-924B through basic events HORXC1ISV_0670923A and HORXC1ISV_0670924B).	Captured through the propagation path analysis.
%0FLAFW2	Flood event induced by Unit 2 AFW line break in room 692.0-A1, 713.0-A1, 737.0-A1 or 737.0-A12	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). AFW system impacted by CST-B unavailability (modeled through mapping to BE TKPRP2TANK00200232). ERCW backup supply to AFW-2 is not possible due to the break in the AFW line (modeled by mapping to spuriouse closure of valves 2-67-923A and 2-67-924B through basic events HORXC2ISV_0670923A and HORXC2ISV_0670924B).	Captured through the propagation path analysis.
%0FLAFW1692 A6	Flood event induced by AFW line break in room 692.0-A6	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). AFW system impacted by CST-A unavailability (modeled through mapping to BE TKPRP1TANK00200229). ERCW backup supply to AFW-1 is not possible due to the break in the AFW line (modeled by mapping to spuriouse closure of valves 1-67-923A and 1-67-924B through basic events HORXC1ISV_0670923A and HORXC1ISV_0670924B).	Captured through the propagation path analysis.
%0FLAFW1692 A7	Flood event induced by AFW line break in room 692.0-A7	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). AFW system impacted by CST-A unavailability (modeled through mapping to BE TKPRP1TANK00200229). ERCW backup supply to AFW-1 is not possible due to the break in the AFW line (modeled by mapping to spuriouse closure of valves 1-67-923A and 1-67-924B through basic events HORXC1ISV_0670923A and HORXC1ISV_0670924B).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLAFW2692 A25	Flood event induced by AFW line break in room 692.0-A25	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). AFW system impacted by CST-B unavailability (modeled through mapping to BE TKPRP2TANK00200232). ERCW backup supply to AFW-2 is not possible due to the break in the AFW line (modeled by mapping to spuriouse closure of valves 2-67-923A and 2-67-924B through basic events HORXC2ISV_0670923A and HORXC2ISV_0670924B).	Captured through the propagation path analysis.
%0FLAFW2692 A26	Flood event induced by AFW line break in room 692.0-A26	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). AFW system impacted by CST-B unavailability (modeled through mapping to BE TKPRP2TANK00200232). ERCW backup supply to AFW-2 is not possible due to the break in the AFW line (modeled by mapping to spuriouse closure of valves 2-67-923A and 2-67-924B through basic events HORXC2ISV_0670923A and HORXC2ISV_0670924B).	Captured through the propagation path analysis.
%0FLAFW713A 6	Flood event induced by AFW line break in room 713.0-A6	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). AFW system impacted by CST-A unavailability (modeled through mapping to BE TKPRP1TANK00200229). ERCW backup supply to AFW-1 is not possible due to the break in the AFW line (modeled by mapping to spuriouse closure of valves 1-67-923A and 1-67-924B through basic events HORXC1ISV_0670923A and HORXC1ISV_0670924B).	Captured through the propagation path analysis.
%0FLAFW713A 19	Flood event induced by AFW line break in room 713.0-A19	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). AFW system impacted by CST-B unavailability (modeled through mapping to BE TKPRP2TANK00200232). ERCW backup supply to AFW-2 is not possible due to the break in the AFW line (modeled by mapping to spuriouse closure of valves 2-67-923A and 2-67-924B through basic events HORXC2ISV_0670923A and HORXC2ISV_0670924B).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLAFW737A5	Flood event induced by AFW line break in room 737.0-A5	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). AFW system impacted by CST-A unavailability (modeled through mapping to BE TKPRP1TANK00200229). ERCW backup supply to AFW-1 is not possible due to the break in the AFW line (modeled by mapping to spuriouse closure of valves 1-67-923A and 1-67-924B through basic events HORXC1ISV_0670923A and HORXC1ISV_0670924B).	Captured through the propagation path analysis.
%0FLAFW737A9	Flood event induced by AFW line break in room 737.0-A9	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). AFW system impacted by CST-B unavailability (modeled through mapping to BE TKPRP2TANK00200232). ERCW backup supply to AFW-2 is not possible due to the break in the AFW line (modeled by mapping to spuriouse closure of valves 2-67-923A and 2-67-924B through basic events HORXC2ISV_0670923A and HORXC2ISV_0670924B).	Captured through the propagation path analysis.
Flood events induced by HPFP			
%0FLHPFPABF	Flood event induced by HPFP in the common areas of the Auxiliary Building (multiple elevations)	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.
%0FLCRDM1F	Flood event induced by HPFP or RCW line breaks in room 782.0-A1	Unit 1 trip on spray effect induced on the Unit 1 MG set equipment (modeled through mapping to initiator %1RTIE). Unit 2 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %2RTIE).	Captured through the propagation path analysis.
%0FLCRDM2F	Flood event induced by HPFP or RCW line breaks in room 782.0-A3	Unit 2 trip on spray effect induced on the Unit 2 MG set equipment (modeled through mapping to initiator %2RTIE). Unit 1 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %1RTIE).	Captured through the propagation path analysis.
%0FLHPFPAB772A7	Flood event induced by break of HPFP line in room 772.0-A7	Expected dual unit trip due to loss of electric power. This is modeled by mapping to loss of 125V DC Vital Battery boards (U1_LVBB1, U1_LVBB2, U2_LVBB3 and U2_LBB4).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLHPFPAB772A10	Flood event induced by break of HPFP line in room 772.0-A10	Expected dual unit trip due to loss of electric power. This is modeled by mapping to loss of 125V DC Vital Battery boards (U1_LVBB1, U1_LVBB2, U2_LVBB3 and U2_LBB4).	Captured through the propagation path analysis.
%0FLHPFPAB757A2	Flood event induced by break of HPFP line in room 757.0-A2	Expected dual unit trip due to loss of electric power. This is modeled by mapping to loss of 125V DC Vital Battery boards (U1_LVBB1, U1_LVBB2, U2_LVBB3 and U2_LBB4).	Captured through the propagation path analysis.
%0FLHPFPAB757A5	Flood event induced by break of HPFP line in room 757.0-A5	Expected dual unit trip due to loss of electric power. This is modeled by mapping to loss of 125V DC Vital Battery boards (U1_LVBB1, U1_LVBB2, U2_LVBB3 and U2_LBB4).	Captured through the propagation path analysis.
%0FLHPFPAB757A21	Flood event induced by break of HPFP line in room 757.0-A21	Expected dual unit trip due to loss of electric power. This is modeled by mapping to loss of 125V DC Vital Battery boards (U1_LVBB1, U1_LVBB2, U2_LVBB3 and U2_LBB4).	Captured through the propagation path analysis.
%0FLHPFPAB757A24	Flood event induced by break of HPFP line in room 757.0-A24	Expected dual unit trip due to loss of electric power. This is modeled by mapping to loss of 125V DC Vital Battery boards (U1_LVBB1, U1_LVBB2, U2_LVBB3 and U2_LBB4).	Captured through the propagation path analysis.
%0FLHPFP737A5F	Flood event induced by HPFP line break in room 737.0-A5	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.
%0FLHPFP737A9F	Flood event induced by HPFP line break in room 737.0-A9	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.
%0FLHPFPAB713A68F	Flood event induced by HPFP line break in room 713.0-A6 or 713.0-A8	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.
%0FLHPFPAB713A1921F	Flood event induced by HPFP line break in room 713.0-A19 or 713.0-A21	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.
%0FLHPFP692A7F	Flood event induced by a HPFP line break in room 692.0-A7	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLHPFP692 A25F	Flood event induced by a HPFP line break in room 692.0-A25	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.
%0FLHPFPCB	Flood event induced by a HPFP line break in the Control Building	Flooding of lower level of Control Building will impact the electrical boards associated with BOP; this is expected to induce a dual unit plant trip (modeled through mapping to initiators %1LOCV and %2LOCV).	Captured through the propagation path analysis.
%0FLHPFPIPS	Flood event induced by a HPFP or RCW line break in room 711.0-E1	HPFP spray or flood events and RCW spray or flood events will have the same effects in room 711.0-E1. It is assumed that RCW controller and electrical equipment will be immediately affected inducing a complete loss of RCW system with consequential turbine/reactor trip (modeled through mapping to initiators %1TTIE and %2TTIE).	Captured through the propagation path analysis.
Flood events induced by DWS			
%0FLDWSAB	Flood event induced by DWS in the common areas of the Auxiliary Building (multiple elevations)	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of DWS storage tank modeled through mapping to basic event TKURP0TANK95900030).	Captured through the propagation path analysis.
%0FLDWS713 A6	Flood event induced by DWS line break in room 713.0-A6	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of DWS storage tank modeled through mapping to basic event TKURP0TANK95900030).	Captured through the propagation path analysis.
%0FLDWS713 A19	Flood event induced by DWS line break in room 713.0-A19	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of DWS storage tank modeled through mapping to basic event TKURP0TANK95900030).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
Flood events induced by CCS			
%1FLCCS	Flood event induced by major CCS line break (Train A)	A break in the Train A CCS will result in a Unit 1 initiator (modeled through mapping to gate U1_CCSA1).	Even though the AFW TD pump has a feature that allows manual operation without any electrical support, the Unit 1 turbine driven AFW pump is considered to be not available due to spray effects on the pump controller cabinet. In addition to the indirect effects captured through the propagation path analysis, the above mentioned indirect effect is modeled through mapping to BE PTSFR1PMP_003001AS and PTSF11PMP_003001AS.
%2FLCCS	Flood event induced by CCS line break (Train B)	A break in the Train B CCS will result in a Unit 2 initiator (modeled through mapping to gate U2_CCSA2).	Captured through the propagation path analysis. Because of the asymmetry in the routing of the CCS piping (i.e., concentrated within the unit 1 side), CCS is not routed on top of the control cabinet for the Unit 2 AFW TD pump outside room 692.0-A26.
%1FLCCS1AB692A7	Flood event induced by CCS line break in room 692.0-A7	A break in the CCS in room 692.0-A7 has the potential to result in an immediate loss of Unit 1 thermal barrier cooling from the thermal barrier booster pumps. Unit 1 is expected to be immediately tripped following this event (modeled through mapping to gate U1_CCSA1). No initiator expected for Unit 2.	Captured through the propagation path analysis.
%2FLCCS2AB692A25	Flood event induced by CCS line break in room 692.0-A15	A break in the CCS in room 692.0-A25 has the potential to result in an immediate loss of Unit 2 thermal barrier cooling from the thermal barrier booster pumps. Unit 2 is expected to be immediately tripped following this event (modeled through mapping to gate U2_CCSA2). No initiator expected for Unit 1.	Captured through the propagation path analysis.
%1FLCCS757A13	Flood event induced by CCS line break in room 757.0-A13 (Surge tank A)	A break in the Train A CCS will result in a Unit 1 initiator (modeled through mapping to gate U1_CCSA1). No expected initiator for Unit 2.	Captured through the propagation path analysis.
%2FLCCS757A13	Flood event induced by CCS line break in room 757.0-A13 (Surge tank B)	A break in the Train B CCS will result in a Unit 2 initiator (modeled through mapping to gate U2_CCSA2). No expected initiator for Unit 1.	Captured through the propagation path analysis.
%1FLCCS713A28	Flood event induced by unisolated break in CCS line in room 713.0-A28	A break in the CCS line in room 713.0-A28 will result in a Unit 1 initiator due to the loss of the excess letdown heat exchanger (modeled through mapping to gate U1_CCSA1). No expected initiator for Unit 2.	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%2FLCCS713A29	Flood event induced by unisolated break in CCS line in room 713.0-A29	A break in the CCS line in room 713.0-A29 will result in a Unit 2 initiator due to the loss of the excess letdown heat exchanger (modeled through mapping to gate U2_CCSA2). No expected initiator for Unit 1.	Captured through the propagation path analysis.
%1FLCCS737A5	Flood event induced by CCS line break in room 737.0-A5	A break in the CCS in room 737.0-A5 has the potential to result in an immediate loss of Unit 1 thermal barrier cooling from the thermal barrier booster pumps. Unit 1 is expected to be immediately tripped following this event (modeled through mapping to gate U1_CCSA1). No expected initiator for Unit 2.	Captured through the propagation path analysis.
%2FLCCS737A9	Flood event induced by CCS line break in room 737.0-A9	A break in the CCS in room 737.0-A9 has the potential to result in an immediate loss of Unit 2 thermal barrier cooling from the thermal barrier booster pumps. Unit 2 is expected to be immediately tripped following this event (modeled through mapping to gate U2_CCSA2). No expected initiator for Unit 1.	Captured through the propagation path analysis.
Flood events induced by RCW			
%0FLRCWABF	Flood event induced by RCW in the common areas of the Auxiliary Building (multiple elevations)	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.
%0FLRCWABMF	Major flood event induced by RCW in the common areas of the Auxiliary Building (multiple elevations)	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.
%0FLRCW772A8	Flood event induced by rupture of RCW line in room 772.0-A8	Expected dual unit trip due to loss of electric power. This is modeled by mapping to loss of 125V DC Vital Battery boards (U1_LVBB1, U1_LVBB2, U2_LVBB3 and U2_LBB4).	Captured through the propagation path analysis.
%0FLRCW772A9	Flood event induced by rupture of RCW line in room 772.0-A9	Expected dual unit trip due to loss of electric power. This is modeled by mapping to loss of 125V DC Vital Battery boards (U1_LVBB1, U1_LVBB2, U2_LVBB3 and U2_LBB4).	Captured through the propagation path analysis.
%0FLRCW757A9	Flood event induced by rupture of RCW line in room 757.0-A9	Expected dual unit trip due to loss of electric power. This is modeled by mapping to loss of 125V DC Vital Battery boards (U1_LVBB1, U1_LVBB2, U2_LVBB3 and U2_LBB4).	Captured through the propagation path analysis.
%0FLRCW757A17	Flood event induced by rupture of RCW line in room 757.0-A17	Expected dual unit trip due to loss of electric power. This is modeled by mapping to loss of 125V DC Vital Battery boards (U1_LVBB1, U1_LVBB2, U2_LVBB3 and U2_LBB4).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLRCW737 A5F	Flood event induced by rupture of RCW lines in room 737.0-A5	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.
%0FLRCW737 A5MF	Major flood event induced by rupture of RCW lines in room 737.0-A5	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.
%0FLRCW737 A9F	Flood event induced by rupture of RCW lines in room 737.0-A9	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE).	Captured through the propagation path analysis.
Flood events induced by ERCW			
%0FLERCWAB 676F-1A	Flood event induced by unisolated ERCW break at elevation 676' of Auxiliary Building (ESF room cooling train 1A)	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 1A for ESF room cooling modeled through mapping to spurious closure of valve 1-FCV-67-127 (Basic event HORXC1FCV_06700127).	Captured through the propagation path analysis.
%0FLERCWAB 676F-1B	Flood event induced by unisolated ERCW break at elevation 676' of Auxiliary Building (ESF room cooling train 1B)	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 1B for ESF room cooling modeled through mapping to spurious closure of valve 1-FCV-67-128 (Basic event HORXC1FCV_06700128).	Captured through the propagation path analysis.
%0FLERCWAB 676F-2A	Flood event induced by unisolated ERCW break at elevation 676' of Auxiliary Building (ESF room cooling train 2A)	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 2A for ESF room cooling modeled through mapping to spurious closure of valve 2-FCV-67-127 (Basic event HORXC2FCV_06700127).	Captured through the propagation path analysis.
%0FLERCWAB 676F-2B	Flood event induced by unisolated ERCW break at elevation 676' of Auxiliary Building (ESF room cooling train 2B)	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 2B for ESF room cooling modeled through mapping to spurious closure of valve 2-FCV-67-128 (Basic event HORXC2FCV_06700128).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLERCWAB 676MF-1A	Major flood event induced by unisolated ERCW break in room 676.0-A1 (ESF room cooling train 1A)	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 1A for ESF room cooling modeled through mapping to spurious closure of valve 1-FCV-67-127 (Basic event HORXC1FCV_06700127).	Captured through the propagation path analysis.
%0FLERCWAB 676MF-1B	Major flood event induced by unisolated ERCW break in room 676.0-A1 (ESF room cooling train 1B)	Expected Unit 1 initiator is the loss of ERCW supply header 1B while Unit 2 initiator is a forced shutdown (for LCO 3.0.3) (modeled through mapping to U1_ABBEX and %2RTIE). Unavailability of train 1B for ESF room cooling modeled through mapping to spurious closure of valve 1-FCV-67-128 (Basic event HORXC1FCV_06700128).	Captured through the propagation path analysis.
%0FLERCWAB 676MF-2A	Major flood event induced by unisolated ERCW break in room 676.0-A1 (ESF room cooling train 2A)	Expected Unit 2 initiator is the loss of ERCW supply header 2A while Unit 1 initiator is a forced shutdown (for LCO 3.0.3) (modeled through mapping to U2_AABEX and %1RTIE). Unavailability of train 2A for ESF room cooling modeled through mapping to spurious closure of valve 2-FCV-67-127 (Basic event HORXC2FCV_06700127).	Captured through the propagation path analysis.
%0FLERCWAB 676MF-2B	Major flood event induced by unisolated ERCW break in room 676.0-A1 (ESF room cooling train 2B)	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 2B for ESF room cooling modeled through mapping to spurious closure of valve 2-FCV-67-128 (Basic event HORXC2FCV_06700128).	Captured through the propagation path analysis.
%0FLERCWDI SAF	Flood event induced by ERCW line break: discharge header A	Break in ERCW discharge header will induce dual unit reactor trip according to AOI-13, Section 3.6 (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of the discharge header will make ERCW backup supply to AFW. This is modeled by mapping to spurious closure of valves 1-67-923A and 2-67-923A (through BE HORXC1ISV_0670923A and HORXC2ISV_0670923A).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLERCWDI SAMF	Major flood event induced by ERCW line break: discharge header A	Break in ERCW discharge header will induce dual unit reactor trip according to AOI-13, Section 3.6 (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of the discharge header will make ERCW backup supply to AFW. This is modeled by mapping to spurious closure of valves 1-67-923A and 2-67-923A (through BE HORXC1ISV_0670923A and HORXC2ISV_0670923A).	Captured through the propagation path analysis.
%0FLERCW69 2A6F	Flood event induced by ERCW line break: discharge header A (AFW TD pump room)	Break in ERCW discharge header will induce dual unit reactor trip according to AOI-13, Section 3.6 (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of the discharge header will make ERCW backup supply to AFW. This is modeled by mapping to spurious closure of valves 1-67-923A and 2-67-923A (through BE HORXC1ISV_0670923A and HORXC2ISV_0670923A).	Captured through the propagation path analysis.
%0FLERCW69 2A6MF	Major flood event induced by ERCW line break: discharge header A (AFW TD pump room)	Break in ERCW discharge header will induce dual unit reactor trip according to AOI-13, Section 3.6 (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of the discharge header will make ERCW backup supply to AFW. This is modeled by mapping to spurious closure of valves 1-67-923A and 2-67-923A (through BE HORXC1ISV_0670923A and HORXC2ISV_0670923A).	Captured through the propagation path analysis.
%0FLERCWDI SBF	Flood event induced by ERCW line break: discharge header B	Break in ERCW discharge header will induce dual unit reactor trip according to AOI-13, Section 3.6 (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of the discharge header will make ERCW backup supply to AFW. This is modeled by mapping to spurious closure of valves 1-67-924B and 2-67-924B (through BE HORXC1ISV_0670924B and HORXC2ISV_0670924B).	Captured through the propagation path analysis.
%0FLERCWDI SBMF	Major flood event induced by ERCW line break: discharge header B	Break in ERCW discharge header will induce dual unit reactor trip according to AOI-13, Section 3.6 (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of the discharge header will make ERCW backup supply to AFW. This is modeled by mapping to spurious closure of valves 1-67-923A and 2-67-923A (through BE HORXC1ISV_0670923A and HORXC2ISV_0670923A).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLERCW692A26F	Flood event induced by ERCW line break: discharge header B (AFW TD pump room)	Break in ERCW discharge header will induce dual unit reactor trip according to AOI-13, Section 3.6 (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of the discharge header will make ERCW backup supply to AFW. This is modeled by mapping to spurious closure of valves 1-67-924B and 2-67-924B (through BE HORXC1ISV_0670924B and HORXC2ISV_0670924B).	Captured through the propagation path analysis.
%0FLERCW692A26MF	Major flood event induced by ERCW line break: discharge header B (AFW TD pump room)	Break in ERCW discharge header will induce dual unit reactor trip according to AOI-13, Section 3.6 (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of the discharge header will make ERCW backup supply to AFW. This is modeled by mapping to spurious closure of valves 1-67-923A and 2-67-923A (through BE HORXC1ISV_0670923A and HORXC2ISV_0670923A).	Captured through the propagation path analysis.
%0FLERCW692A7	Flood event induced by unisolated ERCW break in one supply header in room 692.0-A7	Unit 1 trip on partial loss of ERCW due to loss of supply header 1B (modeled through mapping to U1_ABBEX). Unit 2 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %2RTIE).	Captured through the propagation path analysis.
%0FLERCW1AESFRCF	Flood event induced by unisolated ERCW break associated with ESF room cooling train 1A	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 1A for ESF room cooling modeled through mapping to spurious closure of valves 1-FCV-67-127 (Basic event HORXC1FCV_06700127).	Captured through the propagation path analysis.
%0FLERCW1AESFRCMF	Major flood event induced by unisolated ERCW break associated with ESF room cooling train 1A	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 1A for ESF room cooling modeled through mapping to spurious closure of valves 1-FCV-67-127 (Basic event HORXC1FCV_06700127).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLERCW1B ESFRCF	Flood event induced by unisolated ERCW break associated with ESF room cooling train 1B	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 1B for ESF room cooling modeled through mapping to spurious closure of valves 1-FCV-67-128 (Basic event HORXC1FCV_06700128).	Captured through the propagation path analysis.
%0FLERCW1B ESFRCMF	Major flood event induced by unisolated ERCW break associated with ESF room cooling train 1B	Unit 1 initiator is the loss of ERCW supply header due to the isolation (modeled through mapping to U1_ABBEX); Unit 2 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %2RTIE). Unavailability of train 1B for ESF room cooling modeled through mapping to spurious closure of valves 1-FCV-67-128 (Basic event HORXC1FCV_06700128).	Captured through the propagation path analysis.
%0FLERCW2A ESFRCF	Flood event induced by unisolated ERCW break associated with ESF room cooling train 2A	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 2A for ESF room cooling modeled through mapping to spurious closure of valves 2-FCV-67-127 (Basic event HORXC2FCV_06700127).	Captured through the propagation path analysis.
%0FLERCW2A ESFRCMF	Major flood event induced by unisolated ERCW break associated with ESF room cooling train 2A	Unit 2 initiator is the loss of ERCW supply header due to the isolation (modeled through mapping to U2_AABEX); Unit 1 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %1RTIE). Unavailability of train 2A for ESF room cooling modeled through mapping to spurious closure of valves 2-FCV-67-127 (Basic event HORXC2FCV_06700127).	Captured through the propagation path analysis.
%0FLERCW2B ESFRCF	Flood event induced by unisolated ERCW break associated with ESF room cooling train 2B	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 2B for ESF room cooling modeled through mapping to spurious closure of valves 2-FCV-67-128 (Basic event HORXC2FCV_06700128).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLERCW2B ESFRCMF	Major flood event induced by unisolated ERCW break associated with ESF room cooling train 2B	Common initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Unavailability of train 2B for ESF room cooling modeled through mapping to spurious closure of valves 2-FCV-67-128 (Basic event HORXC2FCV_06700128).	Captured through the propagation path analysis.
%0FLERCW69 2A25	Flood event induced by unisolated ERCW break in one supply header in room 692.0-A25	Unit 2 trip on partial loss of ERCW due to loss of supply header 2A (modeled through mapping to U2_AABEX). Unit 1 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %1RTIE).	Captured through the propagation path analysis.
%0FLERCW71 3A6	Flood event induced by unisolated ERCW break in one supply header in room 713.0-A6	Unit 1 trip on partial loss of ERCW due to loss of supply header 1B (modeled through mapping to U1_ABBEX). Unit 2 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %2RTIE).	Captured through the propagation path analysis.
%0FLERCW71 3A19	Flood event induced by unisolated ERCW break in one supply header in room 713.0-A19	Unit 2 trip on partial loss of ERCW due to loss of supply header 2A (modeled through mapping to U2_AABEX). Unit 1 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %1RTIE).	Captured through the propagation path analysis.
%0FLERCW71 3A28	Flood event induced by unisolated ERCW break in one supply header in room 713.0-A28	Unit 1 trip on partial loss of ERCW due to loss of supply header 1B (modeled through mapping to U1_ABBEX). Unit 2 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %2RTIE).	Captured through the propagation path analysis.
%0FLERCW71 3A29	Flood event induced by unisolated ERCW break in one supply header in room 713.0-A29	Unit 2 trip on partial loss of ERCW due to loss of supply header 2A (modeled through mapping to U2_AABEX). Unit 1 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %1RTIE).	Captured through the propagation path analysis.
%0FLERCW73 7A5	Flood event induced by unisolated ERCW break in one supply header in room 737.0-A5	Unit 1 trip on partial loss of ERCW due to loss of supply header 1B (modeled through mapping to U1_ABBEX). Unit 2 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %2RTIE).	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLERCW737A9	Flood event induced by unisolated ERCW break in one supply header in room 737.0-A9	Unit 2 trip on partial loss of ERCW due to loss of supply header 2A (modeled through mapping to U2_AABEX). Unit 1 initiator is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiator %1RTIE).	Captured through the propagation path analysis.
%0FLERCWCB	Flood event induced by ERCW line break in Contriol Building	Flooding of lower level of Control Building will impact the electrical boards associated with BOP; this is expected to induce a dual unit plant trip (modeled through mapping to initiators %1LOCV and %2LOCV).	Captured through the propagation path analysis.
%0FLERCWIPSA	Flood event in ERCW Strainer room A	An ERCW line break in the Train A strainer room will induce a loss of trains 1A and 2A of ERCW (modeled through mapping to CE and EE) thus inducing a manual shutdown (modeled through mapping to %1RTIE and %2RTIE).	Captured through the propagation path analysis.
%0FLERCWIPSB	Flood event in ERCW Strainer room B	An ERCW line break in the Train B strainer room will induce a loss of trains 1B and 2B of ERCW (modeled through mapping to DE and FE) thus inducing a manual shutdown (modeled through mapping to %1RTIE and %2RTIE).	Captured through the propagation path analysis.
Flood events involving RWSTs			
%0FLRWST1AB676	Flood event induced by unisolated line break from RWST 1 at elevation 676' of Auxiliary Building	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLRWST2AB676	Flood event induced by unisolated line break from RWST 2 at elevation 676' of Auxiliary Building	Unit 2 trip on low RWST level (for LCO 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP2TANK06300046.	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLRWST1A B692A1	Flood event induced by rupture of RWST 1 header in room 692.0-A1	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLRWST2A B692A1	Flood event induced by rupture of RWST 2 header in room 692.0-A1	Unit 2 trip on low RWST level (for LCO 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP1TANK06300046	Captured through the propagation path analysis.
%0FLRWST16 92A7	Flood event induced by break in the lines from RWST 1 in room 692.0-A7	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLRWST16 92A8	Flood event induced by break in the lines from RWST 1 in rooms 692.0-A8 or 713.0-A7	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLRWST1SI S	Flood event induced by SIS line break in any of the Unit 1 SIS pump rooms	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLRWST2SIS	Flood event induced by SIS line break in any of the Unit 2 SIS pump rooms	Unit 2 trip on low RWST level (for LCO 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLRWST2692A24	Flood event induced by break in the lines from RWST 2 in rooms 692.0-A24 or 713.0-A20	Unit 2 trip on low RWST level (for LCO 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLRWST2692A25	Flood event induced by break in the lines from RWST 2 in room 692.0-A25	Unit 2 trip on low RWST level (for LCO 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLRWST1713HX	Flood event induced by a rupture of the lines from RWST1 in any of the Unit 1 HX rooms at elevation 713'	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLRWST2713HX	Flood event induced by a rupture of the lines from RWST2 in any of the Unit 2 HX rooms at elevation 713'	Unit 2 trip on low RWST level (for Tech.Spec 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLRWST1713A28	Flood event induced by break in the lines from RWST 1 in room 713.0-A28	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLRWST2713A29	Flood event induced by break in the lines from RWST 2 in room 713.0-A29	Unit 2 trip on low RWST level (for Tech.Spec 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLCVCS1713A6	Flood event induced by CVCS break in room 713.0-A6	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLCVCS1713A0	Flood event induced by CVCS break in area 713.0-A0 (Unit 1)	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLCVCS1PITS	Flood event induced by Unit 1 CVCS break in sealed pits	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLCVCS2713A19	Flood event induced by CVCS break in room 713.0-A19	Unit 2 trip on low RWST level (for LCO 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP2TANK06300046.	Captured through the propagation path analysis.
%0FLCVCS2713A0	Flood event induced by CVCS break in area 713.0-A0 (Unit 2)	Unit 2 trip on low RWST level (for LCO 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP2TANK06300046.	Captured through the propagation path analysis.
%0FLCVCS2PITS	Flood event induced by Unit 2 CVCS break in sealed pits	Unit 2 trip on low RWST level (for LCO 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP2TANK06300046.	Captured through the propagation path analysis.
%0FLCVCS1692A9	Flood event induced by CVCS break in room 692.0-A9	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.
%0FLCVCS1692A10	Flood event induced by CVCS break in room 692.0-A10	Unit 1 trip on low RWST level (for LCO 3.5.4). Unit 2 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 1 will induce unavailability of Unit 1 SIS and charging pumps. Unit 2 SIS and charging pumps not affected. Unavailability of RWST-1 modeled through mapping to basic event TKURP1TANK06300046.	Captured through the propagation path analysis.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLCVCS269 2A22	Flood event induced by CVCS break in room 692.0-A22	Unit 2 trip on low RWST level (for LCO 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP2TANK06300046.	Captured through the propagation path analysis.
%0FLCVCS269 2A23	Flood event induced by CVCS break in room 692.0-A23	Unit 2 trip on low RWST level (for LCO 3.5.4). Unit 1 trip is a forced shutdown (for LCO 3.0.3) due to loss of both trains of RHR and CSS (modeled through mapping to initiators %1RTIE and %2RTIE). Depleted RWST for Unit 2 will induce unavailability of Unit 2 SIS and charging pumps. Unit 1 SIS and charging pumps not affected. Unavailability of RWST-2 modeled through mapping to basic event TKURP2TANK06300046.	Captured through the propagation path analysis.
Flood scenarios in the Turbine Building			
%0FLTBMF	Major flood in the Turbine Building	The impacted unit will experience a loss of condenser and subsequent turbine and reactor trip. Submersion of equipment in the lower elevation of the Turbine Building will result in loss of hotwell pumps and subsequent loss of condenser vacuum also in the other unit. Initiating events are modeled through mapping to %1LOCV and %2LOCV.	Captured through the propagation path analysis.
%0FLTBCST1 MF	Major flood in the Turbine Building involving line break from CST1.	Submersion of equipment in the lower elevation of the Turbine Building will result in loss of hotwell pumps and subsequent loss of condenser vacuum also in the other unit. Initiating events are modeled through mapping to %1LOCV and %2LOCV. Unit 1 AFW functionally impacted by loss of CST; modleed through mapping to TKPRP1TANK00200229.	Captured through the propagation path analysis.
%0FLTBCST2 MF	Major flood in the Turbine Building involving line break from CST2.	Submersion of equipment in the lower elevation of the Turbine Building will result in loss of hotwell pumps and subsequent loss of condenser vacuum also in the other unit. Initiating events are modeled through mapping to %1LOCV and %2LOCV. Unit 1 AFW functionally impacted by loss of CST; modleed through mapping to TKPRP2TANK00200232.	Captured through the propagation path analysis.
Spray only events			

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%1FLTBSPRA Y1-A-B	Spray event on Unit 1 6.9kV boards A and B	Among other loads, the Unit 1 6.9kV boards A and B potentially impact the Unit 1 hotwell pumps, which can potentially induce a loss of condenser vacuum for Unit 1. This is modeled through mapping to initiator %1LOCV.	PRA components impacted by this spray event are the breakers located on the 6.9kV boards 1A and 1B. See the IF Flooding Notebook for specific UNIDs.
%1FLTBSPRA Y1-B-C	Spray event on Unit 1 6.9kV boards B and C	Among other loads, the Unit 1 6.9kV boards B and C potentially impact the Unit 1 hotwell, pumps which can potentially induce a loss of condenser vacuum for Unit 1. This is modeled through mapping to initiator %1LOCV.	PRA components impacted by this spray event are the breakers located on the 6.9kV boards 1B and 1C, namely. See the IF Flooding Notebook for specific UNIDs.
%1FLTBSPRA Y1-C-D	Spray event on Unit 1 6.9kV boards C and D	Among other loads, the Unit 1 6.9kV boards C and D potentially impact the Unit 1 hotwell pumps, which can potentially induce a loss of condenser vacuum for Unit 1. This is modeled through mapping to initiator %1LOCV.	PRA components impacted by this spray event are the breakers located on the 6.9kV boards 1C and 1D. See the IF Flooding Notebook for specific UNIDs.
%0FLTBSPRA Y1-A-D	Spray event on 6.9kV board 1D and 2A	Among other loads, the 6.9kV boards potentially impact the hotwell pumps, which can potentially induce a loss of condenser vacuum for both units. This is modeled through mapping to initiator %1LOCV and %2LOCV.	PRA components impacted by this spray event are the breakers located on the 6.9kV boards 1D and 2A. See the IF Flooding Notebook for specific UNIDs.
%2FLTBSPRA Y1-A-B	Spray event on Unit 2 6.9kV boards A and B	Among other loads, the Unit 1 6.9kV boards A and B potentially impact the Unit 1 hotwell pumps, which can potentially induce a loss of condenser vacuum for Unit 1. This is modeled through mapping to initiator %2LOCV.	PRA components impacted by this spray event are the breakers located on the 6.9kV boards 2A and 2B. See the IF Flooding Notebook for specific UNIDs.
%2FLTBSPRA Y1-B-C	Spray event on Unit 2 6.9kV boards B and C	Among other loads, the Unit 1 6.9kV boards B and C potentially impact the Unit 1 hotwell pumps, which can potentially induce a loss of condenser vacuum for Unit 1. This is modeled through mapping to initiator %2LOCV.	PRA components impacted by this spray event are the breakers located on the 6.9kV boards 2B and 2C. See the IF Flooding Notebook for specific UNIDs.
%2FLTBSPRA Y1-C-D	Spray event on Unit 2 6.9kV boards C and D	Among other loads, the Unit 1 6.9kV boards C and D potentially impact the Unit 1 hotwell pumps, which can potentially induce a loss of condenser vacuum for Unit 1. This is modeled through mapping to initiator %2LOCV.	PRA components impacted by this spray event are the breakers located on the 6.9kV boards 2C and 2D. See the IF Flooding Notebook for specific UNIDs.
%1FLTBSPRA Y2A	Spray event on U1 board 203A (480V TB)	Unit 1 Reactor Trip Initiator expected since control rod MG set breakers are on the board. This is modeled through mapping to %1RTIE.	PRA components impacted by this spray events are the breaker located on the 1-203A board. See the IF Flooding Notebook for specific UNIDs.
%1FLTBSPRA Y2B	Spray event on U1 board 203B (480V TB)	Unit 1 Reactor Trip Initiator expected since control rod MG set breakers are on the board. This is modeled through mapping to %1RTIE.	PRA components impacted by this spray events are the breaker located on the 1-203B board. See the IF Flooding Notebook for specific UNIDs.
%2FLTBSPRA Y2B	Spray event on U2 board 203B (480V TB)	Unit 2 Reactor Trip Initiator expected since control rod MG set breakers are on the board. This is modeled through mapping to %2RTIE.	PRA components impacted by this spray events are the breaker located on the 2-203A board. See the IF Flooding Notebook for specific UNIDs.

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLTBSPRA Y3	Spray event on common board 205 B	A potential dual unit Turbine Trip can be expected in case of a spray event on the common turbine building board. Modeled through mapping to %1TTIE and %2TTIE.	PRA components impacted by this spray event are the breaker located on the 205-B board. See the IF Flooding Notebook for specific UNIDs.
%0FLTBSPRA Y4	Spray event on air compressor D and sequencer	Spray on air compressor sequencer will induce loss of compressed air. Modeled through mapping to %0TLPCA.	Air compressor D and the sequencer are impacted: WBN-0-COMP-032-4900 and WBN-0-PIC -032-0125.
%0FLTBSPRA Y5	Spray event on dryers	Spray on air compressor dryers will induce loss of compressed air. Modeled through mapping to %0TLPCA.	Dryers WBN-0-DRYR-032-0010, WBN-0-DRYR-032-0015 and WBN-0-DRYR-032-0156 are impacted. the IF Flooding Notebook for relevant BE.
%1FLTBSPRA Y6	Spray event on distribution board WBN-0-DPL -239-0001	Reactor trip potential indicated near the cabinet (orange sign can be seen in walkdown picture WBN-IF-WDP-050). Modeled through mapping to %1RTIE.	PRA components impacted by this spray event are the breaker located on the 239-0001 Board. See the IF Flooding Notebook for specific UNIDs.
%1FLRTIE	Spray event on MG sets – Unit 1	A spray event on the Unit 1 MG set or control rod drive mechanism control is expected to induce a spurious reactor trip (modeled through mapping to %1RTIE).	No indirect effects.
%2FLRTIE	Spray event on MG sets – Unit 2	A spray event on the Unit 2 MG set or control rod drive mechanism control is expected to induce a spurious reactor trip (modeled through mapping to %2RTIE).	No indirect effects.
Flood/HELB events			
%1FLHELBAF W	HELB scenario induced by MSS supply to AFW line break. Unit 1	A reactor trip is assumed for Unit 1. Modeled through mapping to %1RTIE. Unavailability of TD AFW pump modeled through mapping to PTSF11PMP_003001AS).	In additiono to the indirect effects captured through the propagation path analysis, the indirect effects resulting from the HELB harsh environment are captured in the IF Flooding Notebook.
%2FLHELBAF W	HELB scenario induced by MSS supply to AFW line break, Unit 2	A reactor trip is assumed for Unit 2. Modeled through mapping to %2RTIE. Unavailability of TD AFW pump modeled through mapping to PTSF12PMP_003001AS).	In addition to the indirect effects captured through the propagation path analysis, the indirect effects resulting from the HELB harsh environment are captured in the IF Flooding Notebook.
%0FLHELB01A	HELB scenario induced by CVCS line break in room 713.0-A28	There are temperature sensors in the vicinity of the CVCS lines that isolate the lines due to high temperature in the vicinity of the lines. These temperature sensors were put in as part of the EQ program specifically for isolating CVCS line breaks to limit EQ temperatures. Manual reactor trip assumed for Unit 1, similar to a response to a LOCA outside containment (modeled through mapping to %1RTIE).	In addition to the indirect effects captured through the propagation path analysis, the indirect effects resulting from the HELB harsh environment are captured in the IF Flooding Notebook..

Table 3-3: Direct/Functional and Indirect Effects of Flooding events

IE	Description	Direct effect	Indirect effects
%0FLHELB01B	HELB scenario induced by CVCS line break in room 713.0-A29	There are temperature sensors in the vicinity of the CVCS lines that isolate the lines due to high temperature in the vicinity of the lines. These temperature sensors were put in as part of the EQ program specifically for isolating CVCS line breaks to limit EQ temperatures. Manual reactor trip assumed for Unit 2, similar to a response to a LOCA outside containment (modeled through mapping to %2RTIE).	In addition to the indirect effects captured through the propagation path analysis, the indirect effects resulting from the HELB harsh environment are captured in the IF Flooding Notebook.
%0FLHELB02A	HELB scenario induced by CVCS line break in room 737.0-A7	There are temperature sensors in the vicinity of the CVCS lines that isolate the lines due to high temperature in the vicinity of the lines. These temperature sensors were put in as part of the EQ program specifically for isolating CVCS line breaks to limit EQ temperatures. Manual reactor trip assumed for Unit 1, similar to a response to a LOCA outside containment (modeled through mapping to %1RTIE).	In addition to the indirect effects captured through the propagation path analysis, the indirect effects resulting from the HELB harsh environment are captured in the IF Flooding Notebook.
%0FLHELB02B	HELB scenario induced by CVCS line break in room 737.0-A8	There are temperature sensors in the vicinity of the CVCS lines that isolate the lines due to high temperature in the vicinity of the lines. These temperature sensors were put in as part of the EQ program specifically for isolating CVCS line breaks to limit EQ temperatures. Manual reactor trip assumed for Unit 2, similar to a response to a LOCA outside containment (modeled through mapping to %2RTIE).	In addition to the indirect effects captured through the propagation path analysis, the indirect effects resulting from the HELB harsh environment are captured in .the IF Flooding Notebook.

8 **Evaluate Flood Mitigation Strategies -** For each IF initiating event identified in Task 6 and consistent with equipment degradation identified in Task 7, flood mitigation strategies are developed. This evaluation consists of Human Reliability Analysis (HRA) of actions taken by Main Control Room (MCR) operators as well as by auxiliary operators out in the plant to terminate the flood and secure the plant. The evaluation includes considerations of equipment access restrictions, risk of electrocution, additional workload and stress and uncertainty in event progression. Recovery actions are defined as operator actions that have the ability to terminate the flood impacts and propagation and include evaluation of available times and identification of existing flood alarms and procedures.

9 **PRA Modeling of Flood Scenarios -** This task includes the finalization of flood scenario development by modifying existing system fault trees and completing IF accident sequence models and the performance of evaluations by examining potential propagation paths, giving credit for appropriate flood mitigation systems and operator actions and identifying susceptible SSCs that are included in the PRA model. For the WBN flooding analysis, the XINIT tool was used to insert the flood scenario initiators into the internal events PRA model. This tool was originally developed for inserting initiators or modifying external initiators and related events into a PRA model. Because of its general applicability, XINIT was used to modify the PRA model in order to integrate and perform the quantification for the flooding sequences simultaneously. Among other functions, XINIT has the capability to perform the following functions to a PRA model which are pertinent to a flooding analysis:

- Insert new initiating events
- Insert initiator-specific human failure events and/or mutually exclusive logic
- Insert initiator-specific recovery events

10 **PRA Quantification of Flood Scenarios**

The purpose of this task is to perform a quantification of flooding-induced accident sequences. This task includes the performance of quantitative screening analysis to manage a potentially large number of scenarios and locations that have not been screened out previously. Another key purpose of this task is to develop IF-PRA results and insights, and perform uncertainty and sensitivity analysis.

Table 3-12 summarizes the results of the Base Case WBN PRA quantification of CDF and LERF due to internal floods. The internal flooding quantification was performed to account for the flood scenario initiating event frequencies and basic event probabilities for the associated mitigating systems. The resulting CDF and LERF are presented as best estimates.

Table 3-12: Base Case Best Estimate Results

Unit	CDF			LERF		
	Total	**Flood only**	**%**	**Total**	**Flood only**	**%**
Unit 1	3.69E-05	4.72E-06	13%	2.69E-06	4.58E-07	17%
Unit 2	3.28E-05	3.73E-06	11%	2.62E-06	4.51E-07	17%

As can be seen in Figure 3-1, the contribution of internal flooding event is approximately 11.2% of the total core damage frequency. Figure 3-2 provides a breakdown of CDF due to internal floods in terms of flood-specific initiators. The symmetric behavior between the two units can be observed in the fact that while the flood events associated with the non-RCA portion of the Auxiliary Building are common between the two units, as well as the events associated with other common areas of the Auxiliary Building or the Turbine Building, flood events significantly impacting Unit 2 are associated with loss of ERCW header 2A in rooms 692.0-A26 or 737.0-A9. The descriptions of the flooding initiators in Figure 3-2 can be found in Table 3-1.

Two sets of sensitivity cases were run on the WBN IF-PRA. The first set of sensitivity cases (i.e., risk-management cases) focused on evaluating alternative design/procedural changes that would significantly impact (i.e., reduce) the flood-related CDF and LERF. The second set of sensitivity cases was designed to address epistemic uncertainties identified in the development of the WBN IF PRA.

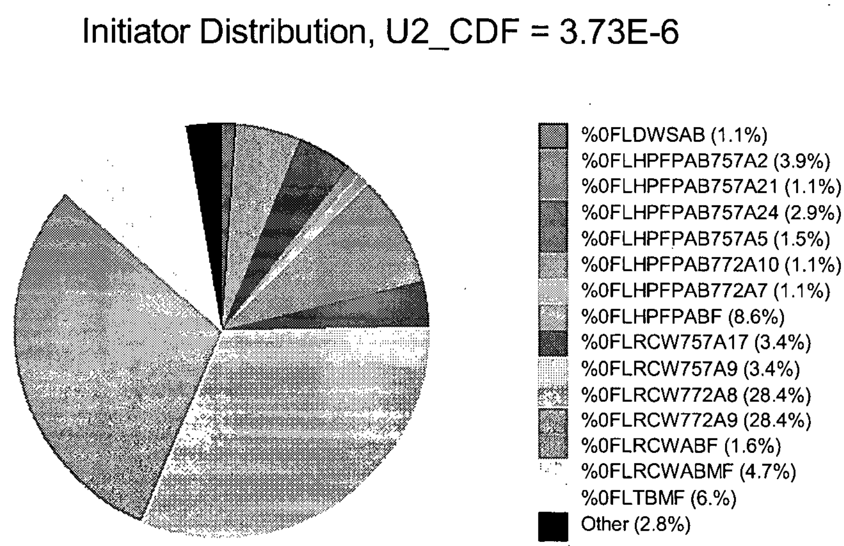


Figure 3-2: Unit 2 Flood-Initiator CDF Breakdown

3.8. Quantification (QU)

Quantification of the PRA model occurred at various levels:

- System level – to generate results for each system top event or support system model
- Accident sequence level – to generate results for each accident sequence
- CDF level – to generate overall results at the consequence category

System level quantification was done for each top event and each support system module. If a system contained a support system, then the transfer event for the support system was set to 0.1 so that the model would quantify and generate results. Other events such as HRA events and flags that were part of the system model were set to a representative value so that the model would generate results. System level quantification was performed prior to completion of all data analysis. The purpose of the system level cutset review was to confirm that the system models reflect the current design and operation. The use of screening data values is appropriate for this level of system review.

Accident sequence level quantification was done for each accident sequence. The truncation level for each sequence was set so that a minimal level set of results was obtained in all cases.

A CDF or consequence level quantification was performed with the complete integrated model that included all of the merged elements, such as system level models, house/flag events, HRA events, mutually exclusive combinations and initiating event models.

CDF cutset reviews were performed with the cutset from the CDF level quantification. A cutset review was also performed for each initiating event grouping (e.g., LLOCA, GTRAN, and SGTR).

Various techniques were used to perform cutset reviews:

- Qualitatively or intuitively to ensure that a cutset makes physical sense
- Quantitatively to ensure that the cutset probability or frequency is correct
- CAFTA's Browser tool was used to trace cutsets through the event sequence and the system models to ensure model accuracy
- Reviews to break circular logic were performed.

Important systems and components identified through the cutset review include:

- ERCW - The loss of ERCW either as an initiating event or as a consequential failure leads to the loss of ECCS equipment.
- 6.9kV and 480V Shutdown Power - Flooding events which impact the shutdown boards induce a station blackout condition without recovery.
- 120V AC Vital Instrument Power - Failure of a train of Vital Instrument Power can initiate a plant transient, and with an independent failure of the opposite train, leads to challenging the operators to perform required manual actions.
- Auxiliary Feedwater - Tests or maintenance which can result in both the motor driven and turbine driven pumps to be inoperable challenge the operators to establish feed and bleed operation.

Quantification of the WBN PRA model was performed using the following EPRI R&R workstation software suite of programs.

- CAFTA for Windows 5.4 – logic model development program
- PRAQuant 5.1 – sequence level quantification control program
- FTREX 1.4.0.1 – cutset generation program
- QRecover 2.5 – cutset recovery and modification program

- Uncert 3.0 (Beta) – computes probability density distributions

Often two events may appear in a cutset that could not occur simultaneously. To address mutually exclusive events, combinations are identified and the non-applicable cutsets are removed. Two separate methods for removing mutually exclusive cutsets were used during the quantification. A fault tree addresses mutually exclusive system level combinations, and inconsistent common cause combinations generated from the CAFTA common cause tool. A text file was also generated to address cross-system test and maintenance combinations.

The MUX top event was developed from the system fault tree files and then merged into the linked fault tree model with an AND-NOT gate (see gates U1_CDF or U2_CDF in Figure 3.2-1). The MUX fault tree is contained in the plant-level fault tree model.

Recovery actions were applied to the cutset results by a recovery rule file during post processing of the quantification results. A recovery fault tree was generated and is called on from the RecruleCDF.txt file during the post-quantification processing stage using the QRECOVER software. Two types of recoveries are credited in the Level 1 WBN model. The first type of recovery is for LOOP. The LOOP recovery factors are documented in the LOOP Non-Recovery Probabilities Notebook. The second type of credited recovery is for successful SG level control during a SBO using the Turbine Driven AFW pump. This human action for SG level control during a SBO (HAOSBF) is described in further detail in the HRA Notebook.

The total core damage frequency computed for Watts Bar Nuclear Plant Unit 1 is 3.69E-05 per reactor-year and Unit 2 CDF is 3.28E-05 per reactor-year. These values were quantified using a truncation limit of 1.0E-12.

The top 100 CDF cutsets for Unit 2 are shown in Appendix B. The top 100 cutsets contain 37.3% of the total Unit 2 CDF. The cutsets were reviewed extensively to identify any inconsistencies in logic. Discussions of the results by initiating events, accident sequence and flooding were discussed in previous sections of this report

To establish the appropriate truncation limit for the plant-level CDF calculation, quantifications were performed with different truncation levels and the ensuing results were recorded. Table 3-13 contains a summary of truncation values, the CDF, and the percent change for the given value. For final quantification of the plant-level CDF model, a cutoff of 1.0E-12 was chosen. This value is over 6 orders of magnitude lower than CDF, moreover, the CDF calculation with a truncation value of 1.0E-13 yields only a ~3% change in CDF. The ~3% change is less than the recommended industry standard (See Supporting Requirement QU-B3 of the ASME PRA Standard,) for measuring percent change in CDF when establishing a truncation level.

Figure 3-3 plots the truncation value and the CDF result. This figure gives a graphical indication of CDF convergence as the truncation value is lowered.

Table 3-13
Model Convergence on Truncation Value

Truncation*	1.0E-07	1.0E-08	1.0E-09	1.0E-10	1.0E-11	1.0E-12	1.0E-13
U1 Total CDF*	7.30E-06	1.33E-05	2.52E-05	3.15E-05	3.42E-05	3.69E-05	3.80E-05
U1 Percent CDF Change	-	82.19%	89.45%	25.01%	8.57%	7.99%	2.83%
U2 Total CDF*	7.30E-06	1.18E-05	2.22E-05	2.79E-05	3.02E-05	3.28E-05	3.37E-05
U2 Percent CDF Change	-	61.64%	88.28%	25.39%	8.25%	8.67%	2.84%

* (per reactor-year)

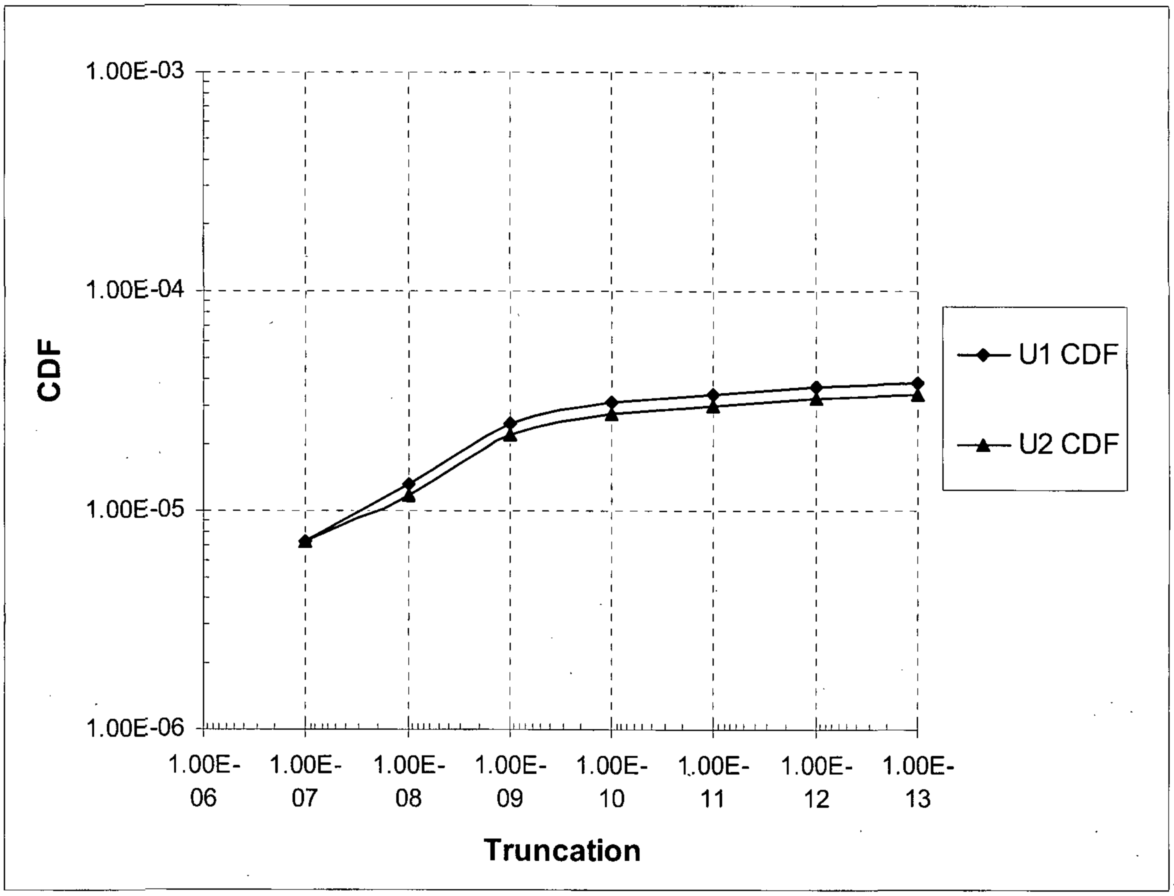


Figure 3-3: Truncation Analysis

Standard industry risk importance measures were computed for various types of events. CAFTA contains a report feature that computes these measures. The plant-level cutset file was used, in conjunction with CAFTA, to generate the measures. The plant-level cutset file includes internal IEs and IF IEs. Table 3-14 lists the event risk importance measures in descending order for Fussell-Vesely (FV), and also lists the Risk Achievement Worth (RAW), and Risk Reduction Worth (RRW).

Table 3-14: Unit 2 Basic Event FV Importance >0.5%

Event Name	Probability	F-V	RRW	RAW	Description
PTSF12PMP_003001AS	2.43E-02	7.67E-02	1.083	4.08	PUMP FAILS TO START AND RUN FOR 1 HOUR WBN-1-3-1AS
DGGFD_FP	5.46E-03	5.48E-02	1.058	10.99	Diesel Generator fails to start or during first hour of operation (Portable Fire Protection Pump)
DGGFR2GEN_0822A-A	1.46E-02	4.11E-02	1.043	3.77	DIESEL GENERATOR FAILS TO RUN AFTER FIRST HOUR
DGGFR2GEN_0822B-B	1.46E-02	4.11E-02	1.043	3.77	DG 2B-B FAILS FAILS TO RUN (WBN-2-GEN -082-0002B -B)
INVFR12NV_2353-F_IE	4.63E-02	3.88E-02	1.04	1.8	INVERTER 1-III FAILS DURING OPERATION (1-FU-235-0003/F1-F)
INVFR2INV_2354-G_IE	4.63E-02	3.88E-02	1.04	1.8	INVERTER 1-IV FAILS DURING OPERATION
FNSFD2FAN_030462	9.13E-03	3.70E-02	1.038	5.01	BOARD ROOM EXHAUST FAN FAILS TO START OR RUN FIRST HOUR
FNSFD2FAN_030460	9.13E-03	3.59E-02	1.037	4.89	BOARD ROOM EXHAUST FAN FAILS TO START OR RUN FIRST HOUR
DGGFD2GEN_0822B-B	6.88E-03	2.78E-02	1.029	5.01	DIESEL GENERATOR FAILS TO START AND RUN FIRST HOUR (WBN-2-GEN -082-0002B -B)
DGGFD2GEN_0822A-A	6.88E-03	2.70E-02	1.028	4.89	DIESEL GENERATOR 2A-A FAILS TO START AND RUN FIRST HOUR
DGGFR1GEN_0821B-B	1.46E-02	2.54E-02	1.026	2.71	DG 1B-B FAILS TO RUN
DGGFR1GEN_0821A-A	1.46E-02	2.46E-02	1.025	2.66	DG 1A-A FAILS TO RUN
DGGFR	2.28E-03	2.28E-02	1.023	10.99	Diesel Generator fails to run after first hour (Portable Fire Protection Pump)
BATFR0BAT_2364-G_IE	1.63E-02	1.86E-02	1.019	2.12	BATT IV FAILS DURING OPERATION (0-BAT-236-3-F)
FNSFD1FAN_030461	9.13E-03	1.78E-02	1.018	2.93	BOARD ROOM EXHAUST FAN FAILS TO START OR RUN FIRST HOUR
FNSFD1FAN_030459	9.13E-03	1.74E-02	1.018	2.89	BOARD ROOM EXHAUST FAN FAILS TO START OR RUN FIRST HOUR
FNSFD2FAN_03000214	9.13E-03	1.51E-02	1.015	2.64	DC EMERG EXHAUST FAN FAILS TO START AND RUN FOR 1ST HOUR WBN-2-30-214
BATFR0BAT_2363-F_IE	1.63E-02	1.34E-02	1.014	1.81	BATT IV FAILS DURING OPERATION (0-BAT-236-3-F)
DGGFD1GEN_0821B-B	6.88E-03	1.34E-02	1.014	2.93	DG 1B-B FAILS TO START AND RUN FIRST HOUR
SEQFD2B-B	3.33E-03	1.34E-02	1.014	5.01	SEQUENCER 2B-B FAILS (Unknown UNID)

Table 3-14: Unit 2 Basic Event FV Importance >0.5%

Event Name	Probability	F-V	RRW	RAW	Description
DGGFD1GEN_0821A-A	6.88E-03	1.31E-02	1.013	2.89	DG 1A-A FAILS TO START AND RUN FIRST HOUR
SEQFD2A-A	3.33E-03	1.30E-02	1.013	4.89	SEQUENCER 2A-A FAILS (Unknown UNID)
FNSFD2FAN_030450	9.13E-03	1.15E-02	1.012	2.25	EXHAUST FAN FAILS TO START OR RUN FIRST HOUR
FNSFD2FAN_030454	9.13E-03	1.15E-02	1.012	2.25	EXHAUST FAN FAILS TO START OR RUN FIRST HOUR
FNSFD2FAN_030448	9.13E-03	1.12E-02	1.011	2.22	EXHAUST FAN FAILS TO START OR RUN FIRST HOUR
FNSFD2FAN_030452	9.13E-03	1.12E-02	1.011	2.22	EXHAUST FAN FAILS TO START OR RUN FIRST HOUR
CMPSR0COMP03200086	6.29E-02	1.09E-02	1.011	1.16	COMPRESSOR B-B FAILS TO RUN WBN-0-32-86
POEFR0PMP_06700028IE	2.97E-02	1.05E-02	1.011	1.34	ERCW PUMP A-A FAILS TO RUNINITIATING EVENT WBN-0-67-28
POEFR0PMP_06700036IE	2.97E-02	1.05E-02	1.011	1.34	ERCW PUMP C-A FAILS TO RUN INITIATING EVENT WBN-0-67-36
POEFR0PMP_06700047IE	2.97E-02	1.05E-02	1.011	1.34	ERCW PUMP E-B FAILS TO RUN CC 1/4 INITIATING EVENT WBN-0-67-E-B
POEFR0PMP_06700055IE	2.97E-02	1.05E-02	1.011	1.34	ERCW PUMP G-B FAILS TO RUNINITIATING EVENT WBN-0-67-55
CBKFO2BKR_2111828/16-B	2.55E-03	1.04E-02	1.01	5.05	6.9kV SDBD BREAKER 1828 FAILS TO OPEN
CBKFO2BKR_2111816/16-A	2.55E-03	1.03E-02	1.01	5.03	6.9kV SDBD BREAKER 1816 FAILS TO OPEN
CMPSR0COMP03200060	6.29E-02	1.03E-02	1.01	1.15	COMPRESSOR A-A FAILS TO RUN WBN-0-32-60
FNSFR2FAN_03000214	2.66E-03	7.55E-03	1.008	3.83	DC EMERGENCY EXHAUST FAN FAILS TO RUN AFTER 1ST HOUR WBN-2-30-214
PTSFR2PMP_003001AS	1.76E-03	7.45E-03	1.008	5.22	PUMP FAILS AFTER 1 HOUR WBN-2-3-1AS
SGDCF2SGD_099A517B	1.77E-03	7.41E-03	1.007	5.18	WBN-2-99-A517-B Safeguard Driver Card Fails
SGDCF2SGD_099A517A	1.77E-03	7.40E-03	1.007	5.17	WBN-2-99-A517-A Safeguard Driver Card Fails
FNSFR2FAN_030460	2.66E-03	7.39E-03	1.007	3.76	EXHAUST FAN 2-FAN-30-460 FAILS TO RUN
FNSFR2FAN_030462	2.66E-03	7.37E-03	1.007	3.76	EXHAUST FAN 2-FAN-30-462 FAILS TO RUN
SEQFD1B-B	3.33E-03	6.44E-03	1.006	2.93	SEQUENCER 1B-B FAILS (Unknown UNID)
SEQFD1A-A	3.33E-03	6.30E-03	1.006	2.89	SEQUENCER 1A-A FAILS (Unknown UNID)
INVFR2INV_2351-D_IE	4.63E-02	5.26E-03	1.005	1.11	INVERTER 1-I FAILS DURING OPERATION (WBN-1-INV -235-0001 -D)
INVFR2INV_2352-E_IE	4.63E-02	5.24E-03	1.005	1.11	INVERTER 1-II FAILS DURING OPERATION
FNSFD1FAN_030449	9.13E-03	5.23E-03	1.005	1.57	EXHAUST FAN FAILS TO START OR RUN FIRST HOUR
FNSFD1FAN_030453	9.13E-03	5.23E-03	1.005	1.57	EXHAUST FAN FAILS TO START OR RUN FIRST HOUR

Table 3-14: Unit 2 Basic Event FV Importance >0.5%

Event Name	Probability	F-V	RRW	RAW	Description
FNSFD1FAN_030447	9.13E-03	5.15E-03	1.005	1.56	EXHAUST FAN FAILS TO START OR RUN FIRST HOUR
FNSFD1FAN_030451	9.13E-03	5.15E-03	1.005	1.56	EXHAUST FAN FAILS TO START OR RUN FIRST HOUR

Limitations in the PRA model and the quantification process need to be reviewed and assessed with regard to possible impact on applications of the PRA model. The WBN Sensitivity and Uncertainty Notebook provides a detailed listing and discussion of the uncertainties and assumptions in the level 1 and level 2 model. The Internal Flooding Notebook discusses the uncertainties and assumptions in the internal flooding model. The uncertainties considered include parameter, model, and completeness. The parameter uncertainty relates to the computation of parameter values for initiating event frequencies, component failure probabilities, and human error probabilities. The model uncertainty relates to the assumptions made in the analysis and the models. The completeness uncertainty relates to the contributions to risk that have been excluded from the model.

The parameter uncertainty is addressed by assigning error factors and distributions to the PRA input. The WBN PRA model quantification process propagates uncertainties through the PRA to define CDF distributions. For parameters that are important to the application and have large associated uncertainties, sensitivities should be completed to determine if the decision-making process for applications is impacted by large uncertainties.

Model uncertainties are primarily related to assumptions and are discussed in detail in the Sensitivity and Uncertainty Notebook for the level 1 and level 2 PRA models, except internal flooding. Assumptions can be characterized as conservatively biased, optimistic, realistic, or unknown. The Sensitivity and Uncertainty Notebook address the following areas: grid reliability and LOOP, support state initiating events, initiating event frequencies, accident sequences, system modeling, equipment survivability and HVAC, human reliability analysis, data and common cause failure, success criteria and thermal hydraulic analysis, general areas, and LERF model. Tables are included in each section that provides detailed information on each element for these areas. Each uncertainty is characterized with regard to model impact, alternatives, and possible sensitivities. Appendix A of the Sensitivity and Uncertainty Notebook includes a listing of all assumptions and provides a characterization of each in terms of impacted areas, assumption type (conservative, bias, realistic, simplifying, optimistic, completeness), and impact level (low medium, high). Assumptions and uncertainties related to internal flooding are discussed in the Internal Flooding Notebook. The epistemic uncertainties related to flood scenarios, flood initiating event analysis, and the HRA contribution in the flood mitigation evaluation are characterized.

The quantification process by itself offers a few limitations that can impact applications. Those specifically to be addressed include the following:

- Truncation or cutoff limit - depending on the application and importance of the component or system of interest, a lower truncation limit could be necessary to ensure relevant cutsets are captured in the quantification process.
- Component and system recoveries added to cutsets - the applicability of any recoveries added to the model need to be assessed relative to the application. For example, in determining a conditional CDF value for a Tech Spec application, a specific component is made unavailable. In this situation recoveries should not be added to this specific component.
- Simplifications made at the time of the quantification to facilitate model assembly and quantification - any simplifications made during the quantification process need to be

1) reviewed and determined to be applicable, 2) characterized for impact on the results, or 3) modified.

- Mutually exclusive events that may not have been addressed in the base model - additional mutually exclusive events may become apparent for application specific quantifications. The cutsets need to be reviewed for these events.
- Dependencies between human actions - lower level cutsets may contain multiple operator actions that were not addressed in the base model. The cutsets need to be reviewed to identify dependencies between operator actions that may have been missed in the base quantification.
- Changes to the plant since the freeze date for development of the PRA model - any changes made to the as-built, as- (to be) operated plant need to be considered. Appendix B in the Internal Flooding Notebook contains a list of components (basic events) not assessed in the internal flooding analysis that could impact the internal flooding PRA model.
- Flag settings – flag settings were developed to model a specific plant/system alignment or configuration. These need to be reviewed for applicability to the application.

The limitations in the quantification process that can impact applications are dependent on the specific application. The limitations need to be considered in light of each application. For example, an application that exercises only one part of the PRA model will need to identify the uncertainties and assumptions related to that specific part of the model. Other applications that involve a broader use of the model will need to consider additional limitations.

The UNCERT code was used to propagate uncertainty for cutsets based on a 1.0E-12 truncation. The state-of-knowledge correlation was addressed using the Monte Carlo sampling method. The uncertainty analysis was performed using 1,000 samples and using the default random seed. The following figure and table display the results for Unit 2 CDF.

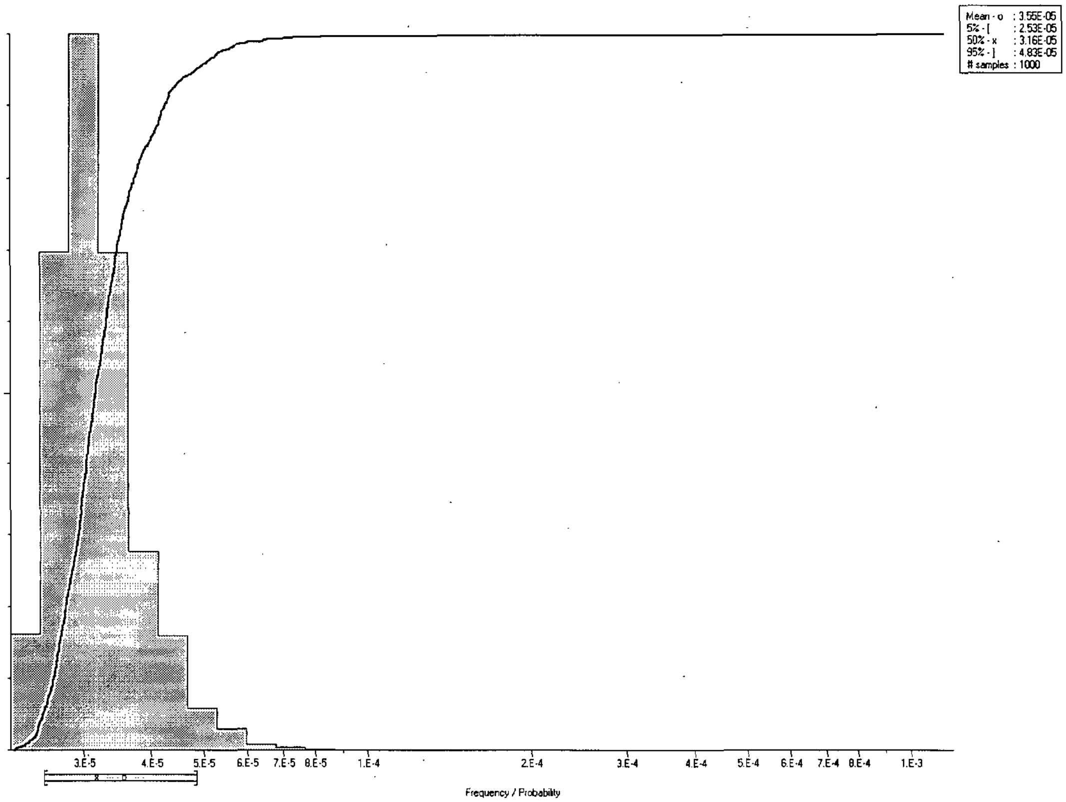


Figure 3-4: Unit 2 CDF Uncertainty Plot

Table 3-15: Unit 2 CDF Uncertainty Results			
Mean	5%	Median	95%
3.55E-05	2.53E-05	3.16E-05	4.83E-05

3.9. Large Early Release Frequency (LE)

The LERF Analysis Notebook documents the containment interface event trees and the LERF event trees used in the WBN Dual Unit Model Revision 0.

As a part of the analysis the previous MAAP4.0.4 parameter file was modified as described in the Thermal Hydraulic Analysis Notebook. It used an 8-node containment model; the upgraded MAAP4.0.7 parameter file uses a 23-node containment model. The increased nodalization is required due to the compartmentalized nature of the WBN ice condenser containment buildings. The MAAP4.0.7 parameter file includes the following:

1) The MAAP4.0.7 parameter file is designed via the ice condenser parameters to include and address ice bed bypass.

2) The MAAP4.0.7 containment model represents a realistic representation of the containment. It contains the free volume, heat sinks, and communication paths between areas.

The MAAP4.0.7 containment and ice condenser models have increased nodalization and compartmentalization to monitor hydrogen pocketing and concentration issues in an ice condenser containment. The MAAP4.0.7 ice condenser doors were benchmarked against numerous scaled experiments including: Waltz Mill experiments, the Pacific Northwest Laboratory experiments, and the Containment Systems Test Facility. The SGs in Watts Bar Unit 2 (model D3) are assumed to be the bounding model for both units. The model D3 SG tube alloy, Alloy 600 is less resistant to creep rupture than the tube alloy of the model 68AXP (Thermally Treated Alloy 690). Although the SG tube thickness could affect creep rupture, the tube thickness is the same for the SG models. Therefore, the model D3 SG is the bounding case for the Level 2 Analysis.

The method used to develop the Watts Bar Level 2 model is based on enhancements to NUREG/CR-6595 and includes realistic quantification of containment threats resulting from high pressure failure of the reactor vessel and hydrogen deflagrations / detonations and additional detail on the treatment of Interfacing System LOCA (ISLOCA) and induced steam generator tube rupture (I-SGTR). Two Containment Event Trees (CET) for Station Blackout (SBO) and Non-SBO events were developed based on NUREG/CR-6595.

The release category for each accident progression's endstate (i.e., Large Early Release Frequency (LERF), Small Early Release Frequency (SERF), LATE, INTACT, etc.) was developed.

The Level 2 event trees were converted into fault trees and additional logic was incorporated to model all necessary plant specific features and to ensure accurate quantifications. There are eighteen event tree questions associated with SBO and non-SBO Level 2 event trees. However, most of the questions are applicable to both SBO and non-SBO events. There are ten event tree questions that involve Level 1 requirements.

The following questions involve Level 1, system requirements or recovery actions:

- Question 1: SBO or Non-SBO
- Question 2: Containment Bypassed
- Question 3: Containment Isolated
- Question 4: Break Size
- Question 5: Feedwater Available to SG
- Question 10: Core Damage Stopped Prior to Vessel Failure
- Question 11: Availabillity of Air Return Fan System
- Question 12: Igniters Available
- Question 16: Containment Heat Removal
- Question 18: Large Early Release

The following questions do not have an endstate:

- Question 1: SBO or Non-SBO
- Question 4: Break Size
- Question 5: Feedwater Available to SG
- Question 7: RCS Depressurization (Early)
- Question 9: RCS Depressurization (Late)
- Question 10: Core Damage Stopped Prior to Vessel Failure
- Question 11: Availabillity of Air Return Fan System

- Question 12: Igniters Available

The following questions have an endstate associated with the response of the question:

- Question 2: Containment Bypassed
- Question 3: Containment Isolated
- Question 6: Pressure Induced SG Tube Rupture
- Question 8: Thermally Induced SG Tube Rupture
- Question 13: Hydrogen Detonation
- Question 14:Direct Containment Heating
- Question 15:Containment Failure (Early)
- Question 16: Containment Heat Removal
- Question 17: Basemat Melt-Through
- Question 18: Large Early Release

There is one endstate of an intact containment assessed in this analysis:

INTACT – an intact containment with no release to the environment

This endstate assesses an intact containment with no releases to the environment.

There are five endstates of large releases assessed in this analysis:

1. BLERF – LER via bypass of the containment

   This endstate assesses bypasses of containment which have a release to the environment. The bypass LERF is given its own category because its releases are much larger than those from LLERF and HLERF. A bypass release does not have an opportunity to undergo scrubbing within the containment. However, the SGTR tube rupture cases may have an opportunity for scrubbing.

2. ILERF – LER via failure of isolation of containment

   This endstate is part of the Level 1 analysis. It assesses failures of containment isolation which will lead to a release to the environment. The isolation failure LERF is given its own category because its releases are much larger than those from LLERF and HLERF. A containment isolation failure release may have the opportunity to undergo scrubbing via the containment sprays. Large isolation failures are considered if the line sizes are greater than or equal to 2 inches.

3. LLERF – LER which occurs during low pressure sequences

   This endstate is determined from large early releases which have a low RCS pressure. The low RCS pressure does not affect the LERF release. LERFs were divided into low pressure and high pressure for ease of modeling.

4. HLERF – LER which occurs during high pressure sequences

   This endstate is determined from large early releases which have a high RCS pressure. The high RCS pressure does not affect the LERF release. LERFs were divided into high pressure and low pressure for ease of modeling.

5. LATE – late release which releases radionuclides into the environment

   This endstate is determined from releases that do not have the potential for early fatalities.

There are three endstates of small releases assessed in this analysis:

6. BSERF – SER via bypass of the containment

   This endstate assesses bypasses of containment which have a release to the environment. The bypass SERF is given its own category because its releases are much smaller than LERFs. However, a bypass release does not have an opportunity to undergo scrubbing within the containment.

7. ISERF – SER via failure of isolation of containment

   This endstate is part of the Level 1 analysis. It assesses failures of containment isolation which will lead to a release to the environment. This isolation failure SERF is given its own category because it has a release which cannot be classified in any other endstate. Small isolation failures are considered if the line sizes are less than 2 inches.

8. SERF – SER via recovery of AC power

   This endstate represents small early releases that occur due to the fission product scrubbing once AC power is recovered. This endstate is only credited in the SBO tree with power recovery and a "not VB" answer to Core Damage Stopped Prior to Vessel Failure.

Table 3-16 shows the breakdown of the various phenomena and other inputs that contribute to LERF. Figure 3-5**Figure 3-** shows the LERF containment failure mode distribution with respect to the total LERF frequency.

Table 3-16
Comparison of LERF Phenomena Contributors

LERF Type	Frequency (per year)	% LERF Contribution
ISLOCA & SGTR	8.69E-09	0.33%
Containment Isolation Failure	4.20E-08	1.61%
Non-SBO CFE (e.g., H2 Burns & EVSE)	9.72E-07	37.13%
TI-SGTR	8.21E-07	31.36%
DCH	1.42E-07	5.43%
PI-SGTR	9.90E-08	3.78%
Hydrogen Detonation	6.09E-09	0.23%
SBO CFE (e.g., H2 Burns & EVSE)	5.27E-07	20.14%

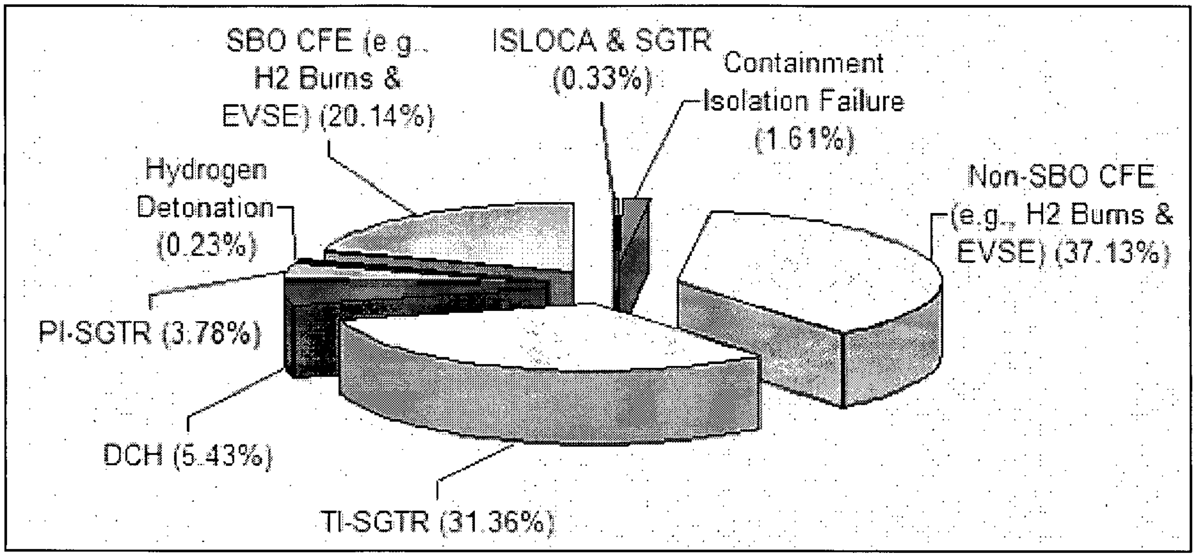


Figure 3-5. Initiating Event Group Contributions to Large Early Release Frequency

Several sensitivities were run using the Level II model and are described in the Level 2 notebook. One of the sensitivities included the hydrogen igniters. As a result GSI-189, WBN Unit 1 has committed to voluntarily enhance the capability of the containment hydrogen igniters. The WBN enhancements includes procuring one trailed mounted diesel generator that can be connected to the plant power system to provide back-up to either train of hydrogen igniters. The generators and cables were procured as commercial grade and will be maintained in accordance with the vendor recommendations. Procedures have been developed for using this equipment. It is expected that the enhanced power supply will also be committed for Unit 2.

During SBO conditions, one train of igniters can also receive power from a dedicated diesel generator. This diesel generator can only be connected to a single unit at a time and can be connected in a timely fashion.

The use of a trailer mounted diesel generator was not credited in the modeling of the HMS. However, the expected modeling change concerning the availability of power to the HMS (crediting the trailer mounted SG) is expected to decrease the LERF frequency.

3.10. Maintenance & Update/Configuration Control (MU)

The TVA process for controlling updates to the PRA is documented in TVA procedure SPP-9.11, "The Probabilistic Risk Assessment Program" and Nuclear Engineering Department Procedure (NEDP)-26, "Probabilistic Risk Assessment". SPP-9.11 covers the management of PRA application, periodic updates and interdepartmental PRA documentation. This procedure provides definitions for PRA model update, PRA model application, and PRA evaluation. This procedure also defines responsibilities of other departments such as operations and system engineering for review of the PRA.

NEDP-26 describes the process used by the PRA staff to perform applications, model updates and PRA model maintenance and review. The terms PRA upgrade and maintenance are defined in the procedures using the definitions provided in the ASME standard. The procedure requires that updates should be completed at least once every other fuel cycle (for the lead unit at multi-unit sites) or sooner if estimated cumulative impact of plant configuration changes exceeds +10%

of CDF. Changes in PRA inputs or discovery of new information shall be evaluated to determine whether such information warrants PRA update. Changes that do not meet the threshold for immediate update are tracked.

PRA updates shall follow the guidelines established by the ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications for a minimum of a Category II assessment. This procedure also defines the requirements for PRA documentation of the model of record and PRA applications. The MOR is composed of the 1) PRA computer model and supporting documentation, 2) MAAP model and supporting documentation, and 3) other Supporting Computer Evaluations (e.g., UNCERT, SYSIMP, EPRI HRA Calculator, etc). The purpose of the PRA MOR is to provide a prescriptive method for quality, configuration, and documentation control. PRA applications and evaluations are referenced to a MOR and therefore the pedigree of PRA applications and evaluations is traceable and verifiable.

After September 2008 all PRA notebooks modified will be converted to desirable calculations. The NEDP-2 calculation process requires calculations to be prepared and independently checked and approved. NEDP-26 also specifies the requirements for independent review and periodic self assessments of the model.

4.0 RESULTS

4.1 Overview of Results

This section presents the results of the Unit 2 Probabilistic Risk Assessment (PRA). The plant sequence model includes the responses of all support and frontline systems that are important for determining the core damage frequency (CDF) and the large early release frequency (LERF). The model results include contributions from internal initiating events and internal floods.

The total core damage frequency computed for Watts Bar Nuclear Plant Unit 1 is 3.69E-05 per reactor-year and Unit 2 CDF is 3.28E-05 per reactor-year. These values were quantified using a truncation limit of 1.0E-12.

The large early release frequency (LERF) computed for Watts Bar Nuclear Plant Unit 1 is 2.69E-06 per reactor-year and Unit 2 is 2.62E-06 per reactor-year. These values were quantified using a truncation limit of 1.0E-12.

Table 4-1: Unit 2 CDF Uncertainty Results

Mean	5%	Median	95%
3.55E-05	2.53E-05	3.16E-05	4.83E-05

The results from the current plant model quantification may be examined in numerous ways. One way to examine the results is by initiating event category. Figure 3-1 shows the frequency of core damage attributable to sequences grouped by initiating events. The most important initiators are related to the Loss of Offsite Power (LOOP). As a group the grid related, plant centered, and weather related LOOPs contribute 45.9% of the total CDF. The loss of ERCW events contribute a little more than 15% of the CDF followed by internal flooding with 11%. The complete loss of ERCW results in a RCP seal LOCA with inadequate coolant makeup capability.

Individual sequences that lead to core damage were discussed in Section 3.2 of this report. The highest frequency damage sequence begins with the total loss of ERCW due to a common cause event resulting in the loss of all 8 ERCW pumps. The total loss of ERCW causes the failure of

ECCS pumps due to the loss of cooling to the ECCS pump room cooling and loss of a heat sink to the component cooling water heat exchangers. The loss of ERCW and consequential loss of CCS induce an RCP seal LOCA with no injection or recirculation capability. The containment spray pumps are also unavailable due to a loss of lube oil and heat exchanger cooling.

The hydrogen igniters are backed up by an external power source. This alternative power arrangement is not modeled in the PSA nor is it discussed in the IPE report. The additional protection would actually reduce the risk of hydrogen detonation from that derived by the PRA model.

4.2 Application of Generic Letter Screening Criteria

The U.S. Nuclear Regulatory Commission (NRC) sequence reporting requirements for the purpose of fulfilling the individual plant examination requirements are discussed in Reference 9. The NRC sequence-reporting guidance states that the total number of most significant sequences to be reported should not exceed 100. The accident analysis is also to be limited to sequences initiated from power operation and from hot standby; events that are initiated from cold shutdown or during refueling are specifically excluded. Events that are both initiated from power operation and from hot standby are included in the model and therefore are considered for inclusion in the list of key sequences reported. The NRC reporting guidelines specify that the mean frequency be reported for each sequence.

Appendix B presents a narrative listing of the 100 highest frequency sequences contributing to the total CDF. This list accounts for sequences whose individual frequency is greater than about 2.4 x 10^{-8} per reactor-year. The sum of all sequences with frequencies greater than 1 x 10-8 (300 sequences) contributes 46% to the total CDF.

The front-end analysis for Watts Bar Nuclear Plant includes consideration of containment bypass events from SGTRs and interfacing system LOCA initiators. The highest frequency core damage sequences from these initiators are also listed in Appendix B.

A back-end analysis was performed as a part of this revision 2 update and is documented in Section 3.9 of this IPE Summary Report.

<u>4.3. VULNERABILITY SCREENING</u>

The results of PSA analysis were also reviewed to identify any potential vulnerabilities. The criteria adopted for identifying vulnerabilities was an exceedance of safety goals in the EPRI PSA Applications Guide. The PSA Applications guide lists a number of safety goals by NRC and the ACRS over the years, among these are:

- Core Damage Frequency < 1 x 10^{-4} / reactor year
- Early Release Frequency < 1 x 10^{-5} / reactor year

A vulnerability may also be identified if a common function, system, operator action, or other common element can be identified which contributes substantially to the total frequency. More than one vulnerability may then be identified. Alternatively, none may be identified if the frequency is well balanced and made up of many different and individually small contributions. Identified vulnerabilities are then to be evaluated for availability of cost effective enhancements.

The occurrence of a vulnerability is therefore based on the total CDF or the early release frequency. If a vulnerability exists, then the specific plant design or operating feature defined as the vulnerability is that which contributes in a substantial way to the frequency criteria being exceeded. To be unique to Watts Bar, the vulnerability must be either a contributor not seen in PRAs for other plants or one that makes a disproportionately high frequency contribution.

The CDF computed for Watts Bar Unit 2 is 3.28 x 10^{-5} /reactor year which is less than the NRC safety goal of 1 x 10^{-4} /reactor year. The LERF frequency is 2.62E-06 for Unit 2 which is below the NRC's LERF Safety Goal of 1.00E-05 per reactor year. The CDF of 3.28 x 10^{-5} /reactor year also favorably compares to the CDFs computed for similar plants, therefore, no particular vulnerabilities with respect to core damage frequency and large early release frequency were identified.

Various plant improvements were evaluated as a part of the Sever Accident Management Alternatives analysis for Unit 2. The SAMA analysis was submitted to the NRC via letter dated January 27, 2009. The results of the SAMA analysis identified four potentially cost beneficial procedural changes and one potentially cost beneficial training enhancement. In this report TVA committed to implement four SAMAs. One of these SMAs to enhance the procedure for controlling temporary alterations to reduce fire risk from temporary cables would not reduce the risk for internal events, but does reduce the fire risk. The following SAMAs would provide a risk reduction to the internal events CDF and LERF:

- SAMA 4: Review station blackout procedures for improvements in DC load shedding
- SAMA 45: Enhance procedural guidance for the use of cross-tied component cooling or service water pumps
- SAMA 156: Enhance procedural guidance for the use of ERCW for RCP thermal barrier cooling.

As discussed previously in this report the most important initiators are related to the Loss of Offsite Power (LOOP). As a group the grid related, plant centered, and weather related LOOPs contribute 45.9% of the total CDF. SAMA 4 to review station blackout procedures for improvements in DC load shedding would allow the DC vital batteries to last longer in the event of a station blackout and potentially reduce the CDF and LERF due to station blackout events. The loss of ERCW events contribute a little more than 15% of the CDF and SAMAs 45 and 156 have the potential to help mitigate losses of ERCW.

5.0 CONCLUSIONS

This model is a complete upgrade from the previous Watts Bar PRA model. The quantification method was changed from the linked event tree (RISKMAN) approach, to the linked fault tree (CAFTA) approach. System fault trees and the integrated logic model were developed using CAFTA. These models are based on current plant references. The previous systemic event trees were replaced by functional event trees which are also based on current plant operating and emergency procedures. The internal flooding analysis was upgraded in accordance with NUREG-6850. The LERF analysis was performed in accordance with current industry guidance. The human error probability evaluation was upgraded using the EPRI HRA Calculator tool and the generic prior data is now based on NUREG-6928. All of these changes are categorized as model upgrades per the ASME PRA standard (Reference 5) which require a new peer review.

The total core damage frequency computed for Watts Bar Nuclear Plant Unit 2 is 3.28E-05 per reactor-year using a 1E-12 truncation limit. This value is below the NRCs CDF Safety Goal of 1.00E-04 per reactor year. The resultant LERF frequency is 2.62E-06 for Unit 2 using a 1E-12 truncation limit. The large, early release frequency assessed in the Level 2 analysis is below the NRCs LERF Safety Goal of 1.00E-05 per reactor year.

6.REFERENCES

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15. Nuclear Power Group (NPG) Standard Department Procedure NEDP-26, "Probabilistic Risk Assessment (PRA), R1, May 29, 2009

16. Watts Bar Nuclear Plant (WBN) Unit 2 - Final Supplemental Environmental Impact Statement - Severe Accident Mitigation Alternatives, T02 090127 001, dated January 27, 2009.

Appendix A: Status of Findings and Observations

Findings

Finding and Observation ID 1-4

F&O Details

Appropriate actuation signals from RPS and ESFAS are modeled. However, the actuation signals from the DG load sequencers are not modeled for each load.

It appears that the loading relays were treated as being in the boundary of the pump. However, this is not consistent with the boundary definitions in NUREG/CR-6928 or Data Notebook MDN-000-999-2008-0145 Table 4.1 2.

(This F&O originated from SR SY-B10)

Basis for Significance
Failure to model the actuation signal following LOSP may cause some dependencies to be missed.

Possible Resolution
Explicitly model actuation logic from the DG load sequencers for each controlled load.

Supporting Requirement	Requirement Met?
SY-B10	☑

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 1-5

F&O Details

Two issues were noted with the modeling of the DC support system:

1. Battery depletion is modeled as an EQU gate with all LOSP initiating events as inputs. This effectively fails all batteries at time 0 following an LOSP, meaning that SBO sequences do not credit delayed failure of the TDAFW pump. Combinations of LOSP and failure of the TDAFW pump may also not be represented.

2. The modeling of the battery boards (e.g., BE BUSFROBD__2363-F) should be at a higher level in the model to ensure it reflects loss of power from both the battery and the battery charger.

(This F&O originated from SR SY-B11)

Basis for Significance
Correct modeling of the battery depletion following LOSP is needed to support recovery analysis and ensure accurate results.

Possible Resolution
1. Add a basic event with a probability of 1.0 to represent battery depletion ANDed with the LOSP initiating events. This provides a basic event in the cutsets that can be used as an indication of delayed TDAFW failure.

2. Revise the modeling of the battery boards to ensure the correct impact is captured.

Supporting Requirement	Requirement Met?
SY-B11	☑

Resolution

Item 1: The CAFTA model was updated to address this item.

Item 2: Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 1-6

F&O Details

MDN-000-999-2008-0145 Section 5.3 documents the Bayesian update process used for WBN. Both mean and EF values are produced for each type code.

However, it was noted that uncertainty interval data was not entered into the WSBN2.RR file and that extraneous information from previous versions of the database were being applied to the factor (demands or exposure time) field of the BE table.

(This F&O originated from SR DA-D3)

Basis for Significance
Incorrect entry of uncertainty intervals in the CAFTA database will result in incorrect output from the UNCERT program.

Possible Resolution
Review the WSBN2.RR file to ensure appropriate uncertainty interval information is entered for each type code and that the uncertainty interval information in the basic event table is removed where it is not applicable.

Supporting Requirement	Requirement Met?
DA-D3	☑

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 1-7

F&O Details

Three problems were noted related to assignment of basic event parameter estimates:

CCF failure probabilities generated by the CAFTA CCF tool do not match hand calculations for some events. For example, hand calculation of the appropriate BE value for BE U0-CCS-PCO-FR2-CCF-IE_ALL produces a value of 7.34E 04/year instead of the value of 2.98E-06/year generated by the CCF tool. (See also F&O 4-7 on SRs IE-C9, IE-C10, and IE-C15)

Several basic events for the AFW system were assigned to incorrect type codes. Basic events PTSFR1PMP_003001AS, PMAF11PMP_00300118, and PMAF11PMP_00300128 were assigned to type codes PTSFR and PMAFR when they should have been assigned type codes PTSF1 and PMAF1. A spot check of the WSBN2.RR file revealed no similar instances for other systems.

Basic event PTSFR1PMP_003001AS is assigned a mission time of 1 hour. It would seem that the mission time for the pump should be at least 4 hours consistent with the battery or 24 hours if the charger is available.

(This F&O originated from SR DA-D1)

Basis for Significance
Underestimation of basic event values will bias the results and may mask important failures.

Possible Resolution
1. Evaluate the results generated by the CCF tool, particularly for annualized events used in initiating event fault trees, to ensure that it is calculating accurate BE values.
2. Correct the type code assignments for the AFW pump failure to start basic events.
3. Evaluate basic event PTSFR1PMP_003001AS to ensure the correct mission time is assigned.

Supporting Requirement	Requirement Met?
DA-D1	☑

Resolution

Turbine-driven AFW pump basic event corrected resolution of other items is in progress.

Findings

Finding and Observation ID 1-8

F&O Details

The division of the ERCW pumps into separate common-cause groups for running and standby pumps is not consistent with current industry practice. Some common-cause failure modes are shared between normally running and standby pumps and should be captured.

In addition, division of the AFW pumps into separate groups by driver type may ignore common mode failures affecting the pumps such as steam binding due to discharge check valve backleakage.

Basis for Significance
Division of common-cause groups for the ERCW and AFW pumps into separate groups may underestimate the impact of common-cause failures.

Possible Resolution
Develop a common group for running and standby ERCW pumps and apply adjustments to the MGL factors to account for shared characteristics between normally running and standby pumps.

Add a common-cause factor to account for potential CCF modes between the AFW pumps that are independent of the type of driver used.

Supporting Requirement	Requirement Met?
SY-B3	☑

Resolution

Resolution in progress. This F&O was determined to also be applicable to CCS.

Finding and Observation ID 2-11

F&O Details

Calculation MDN-000-999-2008-0153 provides details of T/H calculations for timing of cues and time windows. Operator interviews were also used to estimate timing, but no simulations were used to verify operator capability and timing estimates in the accident scenario.

(This F&O originated from SR HR-G4)

Basis for Significance
Criteria met for time windows, cues etc., but operator interviews about the time it takes to do the action is only a secondary way of addressing the "operator time."

Possible Resolution
Use training simulations and simulator training records to validate crew response times for key sequences. Also, document insights from the operator interviews as part of the HRA.

Supporting Requirement	Requirement Met?
HR-E4	☑
HR-G4	☑
HR-G5	☑

Resolution

Resolution in progress.

Findings

Finding and Observation ID 2-12

F&O Details

MDN-000-999-2008-0144 The only system level recovery action input to the model is for recovery of LOOP. Error recovery as part of the HEP calculation is addressed within the HRA calculator, This does not address component, system, or sequence recovery.

(This F&O originated from SR HR-H1)

Basis for Significance
Recovery actions are needed to make the study more realistic.

Possible Resolution
Document a review of the key cutsets in each scenario bin for potential recovery actions. This can be done as part of the dependency assessment.

Supporting Requirement	Requirement Met?
HR-H1	☑

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 2-28

F&O Details

MDN-000-999-2008-0144 Appendix F addresses identification of dependencies. The criteria are met since the analysts followed common practice. However, the stated rule for application of a lower limit (1E-5) on the combined HEP was not applied in the Qrecover File.

Basis for Significance
Some of the combined operator action probabilities are below the threshold specified in the notebook.

Possible Resolution
Redefine the lower threshold for combined HEPs to a value of 1.0E-06 and ensure the combined HEP values are consistent with this threshold. The basis for the lower limit could be that some of the PSFs are global in nature and apply as a sum rather than a product.

For any combinations which are retained with a value lower than the specified threshold, a justification should be provided.

Supporting Requirement	Requirement Met?
HR-D5	☑
HR-G7	☑
QU-C1	☑
QU-C2	☑

Resolution

Resolution in progress.

Findings

Finding and Observation ID 2-29

F&O Details

A reasonableness check is not documented for pre-initiators.

(This F&O originated from SR HR-D7)

Basis for Significance
Criteria met, but some cases of high HEPs were found.

Possible Resolution
Review the details of use of procedures to define the exact details of the human error. For example, WHEMDA/WHEAFW appear to quantify errors at two points in the procedure which is illogical. Using just the last failure to restore step has a 10% reduction on the current CDF. Also during WHESDB the current model does not include local manual operation of TD AFW pump as a recovery action for Loss of 6.9Kv panel and WHESDB sequences.

Supporting Requirement	Requirement Met?
HR-D7	☑

Resolution

Resolution in progress.

Finding and Observation ID 2-3

F&O Details

MDN-000-999-2008-0145 Section 5.2.1 only specifies that "failures that would not have impacted any PRA success criteria" were determined to be not applicable. There is no detailed discussion of what types of failures are encompassed by that statement.

The Maintenance Rule database dispositions failures as functional failures consistent with the PRA definition of functional failure. However, review of plant specific data in Appendix B is not conclusive on the process for separating the events as independent or common-cause (e.g., additional descriptors should be used to list how the components should be treated). Also, screening rules should be stated for failure events left out and retained for processing to ensure that consistent decisions are made.

Examples of incorrect screening were identified for CDE #s 723 (unavailability with no actual failure), 650 & 651 (single unavailability event counted as two start failures), 790 & 791 (unavailability counted as failures, CDE considered these as a single continuing event even though they occurred on separate days).

(This F&O originated from SR DA-C4)

Basis for Significance
Criterion is not met

Possible Resolution
Recommend enhancing Section 5.2.1 by:
(a) explaining how failures that would not have impacted any PRA success criteria are determined to be not applicable.
(b) When using the Maintenance Rule database descriptions of failures, provide a process for screening, binning, or subsuming to match the PRA definition of functional failure. This also should include a process for identification of dependent events.
(c) Include both screened and unscreened failure events in the data analysis notebook. This would clearly document the bases for screening and retaining events in the failure count for each type code.
(d) Correct the noted examples of incorrect event screening and review the failure events for other cases of incorrect screening.

Supporting Requirement	Requirement Met?
DA-C4	☐
DA-C5	☑

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 2-30

F&O Details

MDN-000-999-2008-0144 provides good documentation of what was done in the main body of this calculation and its appendices with specific operator action details shown in Appendix B. Some documentation improvements are needed.

(This F&O originated from SR HR-I2)

Basis for Significance
Criterion for process is met.

Possible Resolution
Review the cutsets for key manual recoveries (e.g., manual operation of the AFW turbine, if this can be accomplished under some scenarios such as an electrical bus failure make sure that the DC buses provide enough power for manual alignments).

Supporting Requirement	Requirement Met?
HR-I2	☑

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 2-6

F&O Details

MDN-000-999-2008-0145 specifies that equipment demand data comes from the WBN "DatAware" system. This appears to consist of computerized logging data with no identification of whether demands come from post-maintenance testing. No adjustment of the data to account for post-maintenance demands is apparent. However, the Sensitivity and Uncertainty Analysis notebook includes an evaluation to assess the impact of this. Recommend discussion of rules used to screen and count special cases.

(This F&O originated from SR DA-C6)

Basis for Significance
The plant specific data gathering process depends on a computer system that is not fully explained in the PRA documentation.

Possible Resolution
Recommend documentation of the process used to screen and count data from the DatAware system.

Recommend that for specific high importance components the meaning of the key test point be provided so that the data can be appropriately applied to the PRA model elements. For example, if the DG demand and run-hour data is based on a computer point that records rpm greater than a certain value, additional data for breaker closure and loading is needed to support the PRA modeled parameter of "Diesel starts and runs for one hour."

Supporting Requirement	**Requirement Met?**
DA-C6	☐

Resolution

Resolution in progress.

Findings

Finding and Observation ID 3-1

F&O Details

The convergence analysis for CDF was performed, see Section 5.5 of the Quantification Notebook. However, the convergence analysis for LERF was not performed. The truncation level for both CDF and LERF is set at 1E-12.

(This F&O originated from SR QU-B3)

Basis for Significance
The convergence analysis for LERF should be performed to justify the same truncation limit used for both CDF and LERF.

Possible Resolution
Perform the convergence analysis for LERF.

Supporting Requirement	Requirement Met?
QU-B3	☐
LE-E4	☐

Resolution

Resolution in progress.

Finding and Observation ID 3-10

F&O Details

Section 5.0 of the Quantification Notebook provides a high level discussion of the quantification results, but the PRA Summary report was not available at the time of the peer review.

(This F&O originated from SR QU-F3)

Basis for Significance
Need to provide a detailed discussion of the results (including both CDF and LERF) and risk insights based on the current model of record.

Possible Resolution
Prepare the PRA Summary report.

Supporting Requirement	Requirement Met?
QU-F3	☑

Resolution

Resolution in progress.

Finding and Observation ID 3-13

F&O Details

Section 6.0 of MDN-000-999-2008-0141 was reviewed. The discussion of the top events should be expanded to include the description as to how each top event is modeled in the logic models. The discussion for LOSP and SBO sequences are not included in MDN-000-999-2008-0141, and should be either discussed in this document or provide a clear reference to the document where it is discussed. Appendix A should be revised to the latest ASME Standards.

(This F&O originated from SR AS-C1)

Basis for Significance
Even though the technical elements are met, the documentation needs some improvements.

Possible Resolution
Provide clear discussions of the treatment of the RCP Seal LOCA, LOSP/SBO, and ATWS sequences in the AS notebook or provide clear links to other support documents where the treatment of these transfers are discussed.

In addition, the sequence level operator actions should be included in the sequence descriptions as well as the dependencies between these operator actions at the sequence level.

Supporting Requirement	Requirement Met?
AS-C1	☐

Resolution

The Accident Sequence Notebook was revised to add additional information specified by this F&O.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 3-15

F&O Details

There are some significant cutsets that do not look reasonable or need further review to ensure that there are properly modeled by accounting for key mitigation SSC(s) (e.g., it does not appear that LOSP sequences leading to SBO are crediting operation of the turbine-driven AFW pump).

In addition, the cutsets should be reviewed for consistency between the model and plant operations, in order to ensure that the model reflects the as-built and as-operated plant. For example, cutsets 72, 86 and 95 contain pre-initiator HEPs WHEAFW and WHEMDA representing test isolation errors for both the motor-driven AFW pump and turbine-driven AFW pump. This should be inconsistent with plant operations in that there is usually some verification of operability of the redundant source prior to entering a test which makes a system train unavailable.

Basis for Significance
See the description section.

Possible Resolution
Correct the modeling issues identified in other F&Os and re-quantify the results. A new review should be performed on the resulting cutsets focusing not just on the validity of the cutsets which are present, but also looking for cutsets that would be expected and are missing (e.g., SBO and failure of the turbine-driven AFW pump to start and potential recovery actions that could lessen the impact of low order cutsets (e.g., cutsets 1, 2, 16, 19, 20, and 26 which are single-order cutsets). It is recommended that the cutset review team include someone who was not involved in the model development but is familiar with other PWR models.

Supporting Requirement	Requirement Met?
QU-D1	☐
QU-D2	☐

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 3-17

F&O Details

There is no quantitative definition used for significant basic event, significant cutset, and significant accident sequence found in Section 5.0 of the Quantification Notebook.
In addition, there is no quantitative definition used for significant accident progression sequence found in the LE notebook.

(This F&O originated from SR QU-F66)

Basis for Significance
The definitions are not found in the applicable documents.

Possible Resolution
Document the definitions consistent with ASME/ANS Standard, Section 1-2.

Supporting Requirement	Requirement Met?
QU-F6	☐
LE-G6	☐

Resolution

Resolution in progress.

Findings

Finding and Observation ID 3-18

F&O Details

Section 5.0 of the Quantification Notebook does not address the limitations in the quantification process that would impact applications. For example, the use of HRADEP* in the recovery process may have significant impact on the a(4) assessments and other risk applications. In addition, use of a global recovery rule such as 'U1_L2F_SBOFLAG -U1_L2-SBO' may have impact on the a(4) assessments, which needs to be verified to show that there is no significant errors introduced.

(This F&O originated from SR QU-F5)

Basis for Significance
See the description section.

Possible Resolution
The limitations associated with the WBN PRA model, the results (including CDF/LERF and importance measures), and the insights should be clearly defined in the conclusion section of the Quantification Notebook.

Supporting Requirement	Requirement Met?
QU-F5	☐

Resolution

Section 8.1 was added to the Accident Sequence Analysis to address this F&O.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 3-20

F&O Details

The subsection for each event tree in Section 6.4 of MDN-000-999-2008-0141 provides a discussion of the initiating event mapping to each event tree, including the transfers from other event trees which are included in the fault tree model. A specific discussion of each specific transferred initiator from another event tree should be included in each section for MLOCA, SLOCA, SLOCAV and ATWS. For example, Table 6.1-1 of the AS notebook does not include an ATWS event tree, since the event tree is only used with the initiators transferred from other event trees.
Further discussion on the event tree transfers for ATWS and RCP Seal LOCA are included in Section 3.4.3 of the Quantification Notebook (MDN-000-999-2008-0147).

(This F&O originated from SR AS-A11)

Basis for Significance
The transfers between the event trees should be clearly understood and documented.

Possible Resolution
Ensure the logic model reflects the transfers as intended and provide clear documentation of the transfers in the AS notebook.

Supporting Requirement	Requirement Met?
AS-A11	☑

Resolution

Resolution in progress.

Findings

Finding and Observation ID 3-3

F&O Details

The system successes are not included in the CDF quantification.

This F&O originated from SR QU-B6)

Basis for Significance
The one-top fault tree model does not include the system successes at the accident sequence level, nor is any justification provided as to why this is OK.

Possible Resolution
Either include the system successes in the one-top model or provide a justification for not including the system successes by comparing the cutsets from the CDF one-top model to the individual accident sequence cutsets quantified with the system successes incorporated.

Supporting Requirement	**Requirement Met?**
QU-B6	☐

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 3-6

F&O Details

Section 5.8 of the Quantification Notebook provides a result of the parametric uncertainty analysis. The analysis does not include the uncertainty parameters for the CCF events and ISLOCA events. In addition, the HRADEP* recovery events found in the recovery files are not treated properly in the parametric uncertainty analysis.

(This F&O originated from SR QU-A3)

Basis for Significance
The parametric uncertainty assessment is only a partial assessment. The assessment needs to properly account for the CCF events, ISLOCA events and HRA events in the parametric uncertainty assessment, or provide a State-Of-Knowledge Correlation assessment to show that the results are not impacted significantly.

Possible Resolution
Either include the CCF events, ISLOCA events and HRA events properly in the parametric uncertainty assessment, or provide a State-Of-Knowledge Correlation assessment to show that the results are not impacted significantly. The concern with uncertainty assessment of the CCF events is that uncertainty parameters are not defined for the MGL factors. Therefore, the uncertainty analysis only propagates the uncertainty parameters of the independent failures to the CCF events. Consideration should be given to adopting the Alpha method (which does allow definition of uncertainty parameters for each factor) or performance of additional sensitivity analysis to assess the correlated uncertainty of the CCF events.

Supporting Requirement	Requirement Met?
QU-A3	☑
QU-E3	☑

Resolution

Resolution in progress.

Finding and Observation ID 3-7

F&O Details

Tables 5.7.3-1 and 5.7.3-2 list the important operator actions, but these are not the complete list, since the events replaced by HRADEP* events are not included in the table.

The recovery file for application of HEP dependency contains HRADEP* recovery events that replace several individual operator actions with a single dependent event that creates several problems, such as assessing the importance of the individual operator actions, parametric uncertainty assessment, sequence level dependence analysis, etc.

(This F&O originated from SR QU-D66)

Basis for Significance
Use of HRADEP* recovery rules in the recovery file is introducing several problems, see the description section.

Possible Resolution
Revise the recovery rule to append the dependent events using the "Replace Events" command, instead of replacing the individual operator actions from the quantified model results using the "ChangeEvents" command. Otherwise, perform sensitivity analyses to ensure that the importance of the operator actions and their contribution to parametric uncertainty is fully understood.

Supporting Requirement	Requirement Met?
QU-C1	☑
QU-D6	☑

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 3-8

F&O Details

Section 5.4 of the Quantification Notebook provides a comparison to similar plants. However, the comparison is provided only for total CDF values. The comparison does not identify the causes for significant differences.
In addition, the WBN PRA results are not compared with the previous results for the WBN PRA Riskman model.

(This F&O originated from SR QU-D4)

Basis for Significance
See description section.

Possible Resolution
Provide a result of comparison as to why the significant differences exists, if any. Comparison of the results at the initiating event level and comparison of risk-significant SSCs and HEPs would facilitate the identification of plant-specific differences and may aid identification of results that are not logical.
Additionally, provide a comparison of results (even if at the qualitative level) between the new linked fault tree model and the old support state model for WBN.

Supporting Requirement	**Requirement Met?**
QU-D4	☑

Resolution

Resolution in progress.

Finding and Observation ID 3-9

F&O Details

Section 5.7 of the Quantification Notebook provides listings of the importances by various groupings. The tables are just the listing from CAFTA at the basic event level. Tables 5.7.3-1 and 5.7.3-2 list the important operator actions, but these are not the complete list, since the events replaced by HRADEP* events are not included in the table.

(This F&O originated from SR QU-D6)

Basis for Significance
The importance list should be generated for the SSCs by grouping the basic events as appropriate. The operator actions should be also grouped by HEPs that represent the same actions with respect to the accident scenarios.

Possible Resolution
Provide a listing of SSC importances and the operator action importances by grouping them appropriately. In addition, the importance should be discussed to ensure that the risk insights are properly understood and documented.

The grouping should specifically include consideration of SSCs where different basic event names are used for mitigating system and initiating event fault trees to ensure the total SSC importance is captured.

Supporting Requirement	Requirement Met?
QU-D6	☑
QU-D7	☐

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 4-11

F&O Details

No requirements exist for maintaining control of computer codes used to support PRA per the process described in SPP-2.6.

(This F&O originated from SR MU-E1)

Basis for Significance
Computer codes used to support PRA quantification should have some level of software controls placed on them.

Possible Resolution
Per SPP-2.6, Computer Software Control, Appendix B, revise the Application Software Category for PRA software from E to C. Then implement the software documentation requirements as shown in Appendix C for Category C.

Supporting Requirement	Requirement Met?
QU-B1	☑
MU-E1	☑

Resolution

Resolution in progress.

Findings

Finding and Observation ID 4-14

F&O Details

Table 4.2 does not appear to contain every normally operating plant system. It is not clear what selection process was used for evaluation of the systems listed and why a complete listing of normally operating systems was not used. Not using a complete listing of normally operating systems could result in missing some initiating events.

(This F&O originated from SR IE-A5)

Basis for Significance
Incomplete evaluation to assess the possibility of an initiating event occurring due to a failure of the system.

Possible Resolution
Perform a systematic evaluation of each normally operating system in the plant.

Supporting Requirement	Requirement Met?
IE-A5	☑

Resolution

The Initiating Event Analysis Notebook was revised to address this F&O.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 4-3

F&O Details

The use of General Transient initiating event data from NUREG/CR-6928 improperly allocates the total frequency to the sub categories. The IEF calculations for General Transient in Section 5.3.13 rely on the fraction of total events from Table 5-5 (1987- March 2008) multiplied by the General Transient IEF of NUREG/CR-6928 Section D.2.23. The NUREG IEF value is based on 228 General Transient events between 1998-2002.

Basis for Significance
Improper partitioning of General Transients in the calculation of initiating event frequencies due to using more events than went into the calculation of the initiating event itself.

Possible Resolution
Recalculate the initiating event frequencies for General Transients based on the proper number of events. Data sources are available to do this calculation.

Supporting Requirement	Requirement Met?
IE-C1	☑

Resolution

Resolution in progress.

Finding and Observation ID 4-7

F&O Details

The treatment of common-cause failure to run in the initiator fault trees is not based on an annualized value (8760 hours) but is based on the value calculated for the mitigation model which uses a 24 hour mission time (IE-C9, IE-C10, IE-C15).

Basis for Significance
Calculation inaccuracy for CCF values in initiator fault trees.

Possible Resolution
Recheck all CCF values used in initiator fault tree models and ensure that an adjusted annualized value is being used. If not, re-calculate the CCF values. Use EPRI TR-1016741 vs. TR-1013490.

Supporting Requirement	Requirement Met?
IE-C9	☐
IE-C10	☑
IE-C15	☑

Resolution

The ERCW, CCS, and EP notebooks were reviewed to identify instances to which this F&O is applicable. No instances in the EP notebook were identified. The F&O was determined to be applicable to the ERCW and CCS initiator models.

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 5-1

F&O Details

The mission time used for room heatup calculations (MDN-000-999-2008-0143, Appendix B,WBNOSG4-242, 200, and 197) was optimistically justified.

(This F&O originated from SR SC-A5)

Basis for Significance
According to WBNOSG4-197, 200 and 242, the mission time for mitigation was verified based on simplified calculations and optimistic engineering judgment. Because the component cooling relies on HVAC, the results of room heatup calculation affects the ability of components to function without room cooling.

Possible Resolution
Based on room heatup calculation results, judge whether the safe and stable condition is met and the basis of the judgment should be presented explicitly.

Supporting Requirement	Requirement Met?
SC-A5	☑

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 5-12

F&O Details

The current analysis does not provide a detailed assessment with regard to how various initiating events and systems impact LERF. For example, the relative contribution to LERF from each PDS was not presented.

(This F&O originated from SR LE-F1)

Basis for Significance
To meet CC II, a quantitative evaluation of the relative contribution to LERF from each PDS is required.

Possible Resolution
Perform a quantitative evaluation of the relative contribution to LERF from each plant damage state.

Supporting Requirement	**Requirement Met?**
LE-F1	☑
LE-G3	☑

Resolution

Resolution in progress.

Findings

Finding and Observation ID 5-13

F&O Details

The condition of the SG after core damage was not correctly linked to the Plant Damage States. Sequences with LTHR failure should be grouped into a PDS for DRY SGs and no scrubbing should be credited without proper justification.

For example, in Table 9-3 sequences ATWS-003, ATWS-007, ATWS-013, ATWS-017, GTRAN-003, GTRAN-004, GTRAN-006, GTRAN-007, SLOCAV-003, SLOCAV-004, SLOCAV-006, and SLOCAV-007 are on the failure path of LTHR, but are designated as SG Wet.

In addition, sequences LLOCA-002, LLOCA-003, LLOCA-004, and LLOCA-005 are designated as SG Wet, but AFW is not assured for these sequences because it is not addressed in the event tree. Although it may be valid to assume that even without AFW, the SGs would not dry out due to lack of heat transport to the SGs following a LLOCA event, the justification for this designation should be provided.

(This F&O originated from SR LE-A5)

Basis for Significance
Failure of LTHR means failure of AFW injection after the CST is depleted. Thus the SG will be eventually dry.

Possible Resolution
Regroup sequences with failure of LTHR from WET SG to DRY SG plant damage states.
Describe the rationale for crediting scrubbing of fission products with LTHR failure.
Add an assumption discussing the rationale for designating the LLOCA sequences as SG Wet.

Supporting Requirement	Requirement Met?
LE-A5	☐
LE-C13	☑

Resolution

The CAFTA model, the Accident Sequence Notebook, and the Level 2 Notebook were revised to address this F&O.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 5-15

F&O Details

The criteria to group sequences into the SERF end state was not clearly presented.

Basis for Significance
The definition of SERF was presented in MDN-000-999-2008-0148. However, the scrubbing effect in the RPV or SG was not described in the definition. The basis for grouping containment accident sequences like SERF-003, 004, etc. into SERF should be presented.

Possible Resolution
Provide a criteria for grouping sequences into the SERF end state and document the basis for the applied criteria.

Supporting Requirement	Requirement Met?
LE-C1	☑

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 5-8

F&O Details

The operator action failure probabilities considered in the LERF analysis were not correctly estimated. After core damage, the operation steps in the SAMGs would be much different from the steps in the EOPs before core damage.

(This F&O originated from SR LE-C2)

Basis for Significance
HAPRZ is a key operator action to prevent high pressure accident scenarios. HAPRZ was estimated to be 4.4e-04 while a similar operator action for the level 1 analysis, HAOB1, was estimated to be 1.6e-02.

Possible Resolution
Describe more specifically how the HEP for action HAPRZ was calculated and how the calculation accounted for conditions after core damage.

Supporting Requirement	Requirement Met?
LE-C2	☑
LE-C7	☑
LE-C9	☑
LE-E1	☑

Resolution

Resolution in progress.

Findings

Finding and Observation ID 7-1

F&O Details

A propagation assessment is developed for zone to zone propagation. It is not provided at a flood source level, but does provide a bounding path assessment.

(This F&O originated from SR IFSN-A1)

Basis for Significance
SR IFSN-A1 indicates that each flooding source should be assessed for propagation. The approach in this study provided a zone-to-zone general propagation assessment regardless of the source. This finding also relates to other elements that require source-specific assessments with regard to propagation, mitigation and timing. The overall assessment does provide the basis for such a detailed assessment, but the information is possibly too coarsely grouped as a result of compounding conservative simplifications. This conservatism can bias the assessment rank order.

Possible Resolution
Utilize the existing information to provide a flow rate and accumulation study for each source in each assessed area.

Supporting Requirement	Requirement Met?
IFSN-A1	☑
IFSN-A2	☐
IFSN-A3	☐
IFSN-A9	☑
IFSN-A10	☑

Resolution

To be resolved in future revision.

Finding and Observation ID 7-10

F&O Details

The analysis in Section 5.4.1 includes an assessment that evaluates existing human actions. From a cursory review, the main impact seems to be an exclusion of non-MCR actions given a flood event. There appears to be little if any adjustment to the other actions that are performed in the MCR.

(This F&O originated from SR IFQU-A6)

Basis for Significance
The information in Table 5-15 lists the existing operator actions and defines an impact. No changes are listed for MCR events and those not in the MCR are typically considered to be infeasible. The text indicates that "All actions solely performed from the Main Control Room (MCR) are also expected not to be physically impacted by the flood event." This seems to be in contrast to the SR requirement to adjust PSFs to address additional stress and the work environment following a flood event. This is particularly of interest for events that could include damaged systems such as starting a CCP (HACV2) which could increase flooding rates or results in failure of standby equipment.

Possible Resolution
Develop a more detailed assessment of why no change would be anticipated for actions or perform a PSF evaluation concentrating on those events that could compound the event (fail equipment due to lack of cooling for instance).

Supporting Requirement	Requirement Met?
IFQU-A6	☐

Resolution

To be resolved in future revision.

Finding and Observation ID 7-11

F&O Details

At the time of the analysis, Unit 2 was still under construction. Assumptions made regarding the as-built status of Unit 2 need to be verified and the model updated as necessary to reflect the final design.

(This F&O originated from SR IFPP-A4)

Basis for Significance
Flooding requires detailed knowledge of the plant layout and spatial considerations that can only be confirmed once the final design is installed. New equipment or control systems could alter current assumptions and must be confirmed to ensure fidelity of the model.

Possible Resolution
Commit to performing a confirmatory as-built walkdown for Unit 2.

Supporting Requirement	Requirement Met?
IFPP-A4	☑

Resolution

To be resolved in future revision.

Finding and Observation ID 7-12

F&O Details

Pressure and temperature of each flood source is identified and documented. However, a characterization of the breach, flow rate, and capacity of each source are not clearly documented. Typically a generic value taken from MDN-000-999-2008-0146, Reference 312 is utilized. It is not believed that MDN-000-999-2008-0146, Reference 312 flow rates were intended to be utilized but rather provided a bound on expected flow rate.

(This F&O originated from SR IFSO-A5)

Basis for Significance
The flow rate and source capacity are important when performing the grouping of flood sources to ensure that the grouped event is representative of the range of possible sources and that the dependent faults are consistent.

Possible Resolution
Document the source and the expected flow rate to provide a timing to reach critical heights for sources such that the grouping process is documented and traceable.

Supporting Requirement	Requirement Met?
IFSO-A5	☐

Resolution

To be resolved in future revision.

Finding and Observation ID 7-14

F&O Details

The flooding sources are documented along with their progression to the plant.

However, to identify flood timing and other factors it would be helpful to list the line size and flow rates for the zones for each source. This is mostly available from the walkdown documentation but would provide a more traceable assessment for use in future applications.

Basis for Significance
Enhancement of the documentation is needed to provide a more traceable assessment for use in future applications.

Possible Resolution
Transfer the walkdown size information to the source assessment for each flood source and area.

Supporting Requirement	Requirement Met?
IFSO-B1	☑

Resolution

To be resolved in future revision.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 7-15

F&O Details

A sensitivity study related to the consequences of spray was performed. Variability of sources (such as forced flow rates) were not addressed and were not considered in the assessment.

Basis for Significance
The assessment did not provide detailed flow rates for floods involving normally running systems. It is possible that systems could be in alternative alignments such that the base flow rate would be different. Additionally, it is possible that the operators would trip or load additional pumping capacity that would increase or decrease flow. No assessments are provided.

Possible Resolution
Include assumptions related to flow in addition to source volumes and provide basis for any alternative alignments. Provide a qualitative assessment of uncertainty.

Supporting Requirement	Requirement Met?
IFSO-B3	☑

Resolution

To be resolved in future revision.

Finding and Observation ID 7-16

F&O Details

The potential source equipment located in the flood areas are well defined. However, plant internal and external sources of flooding or in-leakage from other flood areas are not well defined. Further, the statement is made that: "The limitation of the source identification to piping greater than 3" is a recognized source of epistemic uncertainty associated with the Source Identification phase. As described in Assumption 16, this approach is not expected to significantly underestimate the probability of occurrence of a flood event since small bore pipe are likely only capable of inducing spray scenarios due to the limited flow rate that can be expected. Spray events have been investigated on a component-by-component basis during the second walkdown (see Appendix A) independently from the pipe size of the piping around recognized potential targets. This would minimize the impact of this epistemic uncertainty." It is not clear however, that areas with piping on the order of 3" or less were retained by the selection process such that a flooding or spray event would be identified if the only source(s) were smaller than 3".

(This F&O originated from SR IFSO-A1)

Basis for Significance
Assumption #16 indicates a screening criterion of 2" or less. The text indicates that in this case 3" was used and then the basis is assumption #16. This appears to be inconsistent.

Possible Resolution
To support other SRs and F&Os, remove screening criterion based on size.

Supporting Requirement	**Requirement Met?**
IFSO-A1	☐

Resolution

To be resolved in future revision.

Findings

Finding and Observation ID 7-19

F&O Details

The containment challenges were considered based on plant-specific analysis and applicable generic information.

However, the analysis specifies that the 480 gpm/pump seal LOCA is a low-pressure (medium LOCA) scenario which implies that DCH is not a concern. This is at odds with several similar assessments and it is not clear that the pressure cutoff can be met for this sequence class.

(This F&O originated from SR LE-B2)

Basis for Significance
It is not clear that the pressure cutoff to justify that DCH is not a concern can be met for this sequence class.

Possible Resolution
Reclassify sequences with the 480 gpm/pump seal LOCA as high-pressure sequences or provide a plant-specific assessment to show that the pressure cutoff for DCH is supported.

Supporting Requirement	Requirement Met?
LE-B2	☑

Resolution

Resolution in progress.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 7-20

F&O Details

The containment event tree presented necessary logic needed to provide a realistic estimation of the significant accident progression sequences. Depressurization of RCS, operation of hydrogen igniters, etc. were considered and beneficial failure of PZR PORV stuck open was considered, with technical bases.

(This F&O originated from SR LE-C4)

Basis for Significance
For SGTR it is possible to account for cycling SG SRV versus stuck open SG SRV which can allow for a significant fraction of SGTR events to be removed from LERF.

Possible Resolution
Credit holdup of fission products as a result of SG SRVs cycling following SGTR.

Supporting Requirement	Requirement Met?
LE-C4	☑

Resolution

Resolution in progress.

Finding and Observation ID 7-21

F&O Details

The range factors are developed for the flood initiating events, however there is no propagation through the model.

(This F&O originated from SR IFEV-B3)

Basis for Significance
The current analysis does include uncertainty estimates for the flood initiating events. However, the impact and resultant uncertainty associated with combining the different flooding sources, each with an associated range factor, with regard to the overall study uncertainty is not addressed. Additionally, the sensitivity of assumptions related to propagation and flow rates with regard to consequential failures should be addressed to ensure that the impact of such simplifications on the overall results are known.

Possible Resolution
Perform a statistical uncertainty assessment for the results and provide additional sensitivity studies assuming various combinations of assumptions related to initiating event grouping and consequences.

Supporting Requirement	**Requirement Met?**
IFEV-B3	☐

Resolution

To be resolved in future revision.

Finding and Observation ID 7-22

F&O Details

The secondary side isolation of a ruptured SG was modeled in the SGTR event tree (top event SL). After core damage, there was no consideration of the secondary side isolation capability in the accident progression sequences.

(This F&O originated from SR LE-D5)

Basis for Significance
A cycling SRV allows for the SG to be maintained at a higher pressure which tends to increase holdup time prior to release to the environment and to reduce the rate of release such that the overall source term is lower than for cases with a stuck open SG SRV on the faulted steam generator. Prior analyses have indicated that the resulting reduction is sufficient to reduce the source term from large to small.

Possible Resolution
The analysis of the SGTR sequences should include credit not only for the ability to maintain covered tubes, but also the impact of the SG SRV cycling instead of failing open. This would provide a sizeable reduction in the release and may result in the reclassification of some LERF sequences to SERF.

Supporting Requirement	Requirement Met?
LE-D5	☑

Resolution

Resolution in progress.

Findings

Finding and Observation ID 7-3

F&O Details

The spatial assessment was provided but critical depths were not applied based on a realistic assessment of component fragility.

(This F&O originated from SR IFSN-A5)

Basis for Significance
The assessment for failing SSCs is very conservative in that it assumes all components within an area are considered failed on the occurrence of either a flood or a spray event within the area. Only limited credit for elevation differences is provided and additional mitigation time could be defined given a more rigorous assessment. As an example, the 6.9Kv boards are considered failed when water is essentially present in the associated room. However, the presence of ventilation slats at the bottom of the boards up to approximately 30" would tend to indicate that components inside the cabinet would not be impacted prior to a flood of this depth. Further, there are ventilation dampers that would dewater the area when the level reached approximately 24" which again would provide time for identification and mitigation. Another example is the assumption that a spray will fail AFW TDP control panels. The panels are vented but the vents are sparse and completely covered/shielded from downward spray. It does not seem likely that a spray event would impact the cabinet unless a very specific pattern was defined. This also allows for zone of influence split fractions that can limit sequences and lower overall frequency.

Possible Resolution
Utilize the existing information supplemented by additional walkdowns to assess critical component heights based on realistic criteria.

Supporting Requirement	Requirement Met?
IFSN-A5	☑
IFSN-A10	☑

Resolution

To be resolved in future revision.

Findings

Finding and Observation ID 7-4

F&O Details

Table 4-57 identifies potential flooding sources in zones that would not lead to immediate trip, but screening appears to be in most cases related to size. The justification is based on an assumption that a lack of frequency data is available, although the cited reference does include failure data for smaller size piping.

(This F&O originated from SR IFSN-A12)

Basis for Significance
The current SR lists potential methods for screening but does not provide size as a means for exclusion. The WBN study indicates under assumption #16 that: "Breaks in small bore pipes were only considered if the size was within those for which pipe break probability is provided in Reference 314 or if it is expected that the break would result in a plant trip or immediate shutdown. This assumption results in focusing the analysis mainly on piping greater than 2" in diameter." In Table 4-57 several sources are screened based on "Line size below size cutoff (see Assumption #16)".

Possible Resolution
The sources solely screened on size should be reconsidered and the frequency data provided in the referenced document should be applied.

Supporting Requirement	Requirement Met?
IFSO-A1	☐
IFSO-A4	☑
IFSN-A12	☐
IFSN-A15	☐

Resolution

To be resolved in future revision.

Finding and Observation ID 7-5

F&O Details

The area flood initiating event assessment does combine the various pipes found in an area into a single frequency. However, in some cases there is no basis to ensure that different systems would result in same consequences. As per IE-3B: 'DO NOT SUBSUME scenarios into a group unless (1) the impacts are comparable to or less than those of the remaining events in that group AND (2) it is demonstrated that such grouping does not impact significant accident sequences.' It is not clear that timing or recovery actions would not be different.

Basis for Significance
There are several sources, such as fire water and cooling water, that are found in several areas. These events may have different impacts on other safety equipment and could alter success criteria when examining operation for the length of the mission. Also, the flow rates could be limited in a source specific assessment and would have different potentials for recovery. Further, the isolation actions would be different.

Possible Resolution
Assess the events on a source-specific basis using the available information collected from the walkdown.

Supporting Requirement	Requirement Met?
IFSN-A10	☑
IFEV-A2	☑
IFEV-A5	☐

Resolution

To be resolved in future revision.

Findings

Finding and Observation ID 7-7

F&O Details

The current assessment does not provide a rigorous propagation of uncertainty characteristics through the model. Sensitivity cases are provided for several elements, but there is no concise listing of the uncertainty characteristics based on either qualitative or quantitative measure. Major assumptions are listed, but inferred assumptions related to grouping of piping within a zone are not provided.

(This F&O originated from SR IFQU-B3)

Basis for Significance
The internal flooding notebook contains several sensitivity studies that examine specific aspects of the assessment, but there is very little discussion on qualitative factors that could drive uncertainty, how uncertainties related to flood volumes and flow rates (pumps being terminated) would influence timing and thereby the potential for mitigation. The grouping of the sources is also not discussed.

Possible Resolution
Provide thorough documentation of the sources of uncertainty and characterization of the impact of each item on the results of the analysis. This should be similar in scope to the discussion of uncertainty in the Sensitivity and Uncertainty Notebook for other analysis areas.

Supporting Requirement	Requirement Met?
IFQU-B3	☐

Resolution

To be resolved in future revision.

Findings

Finding and Observation ID 7-8

F&O Details

The results are listed at the total CDF level and some important contributors listed. However, there is no discussion of the flooding event tree, event sequences, timing or how flooding might influence LERF. It would also seem reasonable to expect additional results to be presented involving risk ranking of flooding sources, areas, operator mitigation activities and other parameters relevant to flooding.

(This F&O originated from SR IFQU-B1)

Basis for Significance
There is no discussion of the development of the event tree for the flooding event. There is also no description of the internal flooding accident sequences or a discussion of how the flooding analysis was propagated within the LERF assessment.

Possible Resolution
Provide a more complete explanation for the flooding assessment in MDN-000-999-2008-0146 consistent with the level provided for the other internal events. This should include:
(a) A description of the flooding event tree, event sequences, timing and how flooding might influence LERF, and
(b) Risk ranking of flooding sources, areas, operator mitigation activities and other parameters relevant to flooding.

Supporting Requirement	Requirement Met?
IFQU-A10	☑
IFQU-B1	☐

Resolution

To be resolved in future revision.

Watts Bar PRA Model F&O Resolution

Findings

Finding and Observation ID 7-9

F&O Details

HRA events related to isolation and/or mitigation were evaluated in the HRA notebook. They were, however, considered on a somewhat generic basis (not specific to the break but rather the system). This may result in an inappropriate value if the actions defined for the general event do not match with the actual actions for the specific event.

(This F&O originated from SR IFQU-A5)

Basis for Significance
The HRA evaluation for flooding mitigation is based on a high level assessment on the basis that there were sufficiently many sequences that detailed assessment was impractical. If it is assured that no alternative actions are more plausible based on operator input, then this is not inappropriate. An alternative would be to work a top-down approach addressing the controlling events and addressing those in detail. This would be more consistent with the SR related to source-specific assessment.

Possible Resolution
Perform a top-down assessment to ensure that the highest recovered sequences are consistent with the plant expectations for action.

Supporting Requirement	Requirement Met?
IFSN-A9	☑
IFQU-A5	☑

Resolution

To be resolved in future revision.

Appendix B: Top 100 CDF Cutsets

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
1	3.02E-06	1.00E+00	%0TLERCW	Total Loss of ERCW	Total loss of ERCW due to a common cause event resulting in the loss of all 8 ERCW pumps. The loss of ERCW and consequential loss of CCS induced seal LOCA with no injection.
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		6.32E-05	U0_ERCW_PMP_ FR_CCF_IE_ALL	CCF OF ALL ERCW PUMPS FAILS TO RUN IE	
2	1.06E-06	1.06E-06	%0FLRCW772A8	Flood event induced by rupture of RCW line in room 772.0-A8	Raw water pipe in the 5th vital battery room ruptures. The 5th vital battery is rendered unavailable due to spray. Upon propagation to the 480V board rooms, water impacts Unit 1 and Unit 2 inverters on El 772. Shorting the inverters will cause a plant trip. Further water propagation down to the 480V shutdown boards on El 757 will induce a station blackout with no recovery. Manual operation of the turbine driven AFW pump was not credited for flood sequences.

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
3	1.06E-06	1.06E-06	%0FLRCW772A9	Flood event induced by rupture of RCW line in room 772.0-A9	Raw water pipe in the HEPA filter room on El 772 ruptures. Upon propagation to the 480V board rooms, water impacts Unit 1 and Unit 2 inverters on El 772. Shorting the inverters will cause a plant trip. Further water propagation down to the 480V shutdown boards on El 757 will induce a station blackout with no recovery. Manual operation of the turbine driven AFW pump was not credited for flood sequences.
4	8.55E-07	1.00E+00	%2CCS	Total Loss of Component Cooling System Unit 2	Total loss of component cooling due to a common cause event of all the CCS pumps failing to run. Loss of CCS fails thermal barrier cooling. Failure of CCS fails the Train A and Train B ECCS pump cooling. Operator fails to align ERCW cooling to Train A Charging Pump. Loss of thermal barrier cooling and loss of RCP seal injection induced seal LOCA with no injection available.
		6.50E-02	HCCSR4	Align & Initiate Alternate Cooling to 1A-A CCP, 1B-B failed	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		2.75E-04	U0-CCS-PCO-FR-CCF-IE-ALL	CCF of CCS PUMPS FAIL TO RUN, CCS HX PLUGGS, & CCS HX EXCESSIVE LEAKAGE/RUPTURE	
5	3.19E-07	1.00E+00	%0TLERCW	Total Loss of ERCW	Total loss of ERCW due to common cause failure of all the ERCW strainers with a failure of the strainer backwash. Induced seal LOCA with no injection.
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		4.21E-04	U0_ERCW_MTR_FP_CCF_ALL	CCF of all components in group 'U0_ERCW_MTR_FP_CCF'	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		1.58E-02	U0_ERCW_STR_PL_CCF_IE_ALL	CCF of all components in group 'U0_ERCW_STR_PL_CCF_IE'	
6	2.85E-07	2.88E-03	%2SLOCAL	Small LOCA Stuck Open Safety Relief Valve	A small LOCA due to an inadvertent stuck open pressurizer safety relief valve. The operator fails to align high pressure recirculation. Cooldown to LPI conditions fails. The operators fail to realign the containment spray pumps to the sump to refill the RWST.
		9.90E-05	HRADEP-POST-128		
7	2.11E-07	1.00E+00	%0TLERCW	Total Loss of ERCW	Total loss of ERCW event due to a common cause failure of all 8 ERCW pumps to run. Total loss of ERCW fails Train A and Train B ECCS pumps. Total loss of ERCW also failing cooling to the Plant Compressed Air and ACAS compressors. Loss of all air and failure of the operator to locally control LCVs fails the available TDAFW pump. This is a GTRAN event with loss of all injection and AFW.
		3.70E-03	HAFR1	Restore AFW control following initiator and loss of air	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		6.32E-05	U0_ERCW_PMP_FR_CCF_IE_ALL	CCF OF ALL ERCW PUMPS FAILS TO RUN IE	
8	1.83E-07	1.00E+00	%2LDDAC	Loss of 120V AC Vital Instrument Board IV	Inverter 2-IV fails during operation causing a loss of 120V AC Vital Instrument Board IV. Loss of 120V Vital Instrument Power causes a reactor trip on closure of the
		4.63E-02	INVFR2INV_2354-G_IE	INVERTER 2-IV FAILS DURING OPERATION	
		1.83E-03	MTM_2SSPS_TRAINA	SSPS TRAIN A UNAVAILABLE DUE MAINTENANCE	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	MSIV. SSPS Train A is in maintenance and failure of the 120V Vital Instrument Power 1-IV fails SSPS Train B. Operator fails to perform cooldown with MFW following AFW failure. Operator fails to manually start AFW, and the operator fails to establish feed and bleed.
		2.40E-03	HRADEP-POST-220		
9	1.77E-07	1.00E+00	%2LDCAC	Loss of 120V AC Vital Instrument Board III	Inverter 2-III fails during operation causing a loss of 120V AC Vital Instrument Board III. Loss of 120V Vital Instrument Power causes a reactor trip on closure of the MSIV. SSPS Train B fails due to a driver card failure and failure of the 120V Vital Instrument Power 2-III fails SSPS Train A. Operator fails to perform cooldown with MFW following AFW failure. Operator fails to manually start AFW, and the operator fails to establish feed and bleed.
		4.63E-02	INVFR12NV_2353-F_IE	INVERTER 2-III FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		1.77E-03	SGDCF2SGD_099A517B	WBN-2-99-A517-B Safeguard Driver Card Fails	
		2.40E-03	HRADEP-POST-220		
					Inverter 2-IV fails during operation causing a loss of 120V AC Vital Instrument Board IV. Loss of 120V Vital Instrument Power causes a reactor trip on closure of the MSIV. SSPS Train A fails due to a driver card failure and
10	1.77E-07	1.00E+00	%2LDDAC	Loss of 120V AC Vital Instrument Board IV	
		4.63E-02	INVFR2INV_2354-G_IE	INVERTER 2-IV FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		1.77E-03	SGDCF2SGD_099A517A	WBN-2-99-A517-A Safeguard Driver Card Fails	

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
		2.40E-03	HRADEP-POST-220		failure of the 120V Vital Instrument Power 2-IV fails SSPS Train B. Operator fails to manually start AFW, and the operator fails to perform cooldown with MFW following AFW failure. Operator fails to establish feed and bleed.
11	1.73E-07	1.00E+00	%2LDCAC	Loss of 120V AC Vital Instrument Board III	Inverter 2-III fails during operation causing a loss of 120V AC Vital Instrument Board III. Loss of 120V Vital Instrument Power causes a reactor trip on closure of the MSIV. SSPS Train B is out of service due to testing and failure of the 120V Vital Instrument Power 2-III fails SSPS Train A. Operator fails to manually start AFW, and the operator fails to perform cooldown with MFW following AFW failure. Operator fails to establish feed and bleed.
		4.63E-02	INVFR12NV_2353-F_IE	INVERTER 2-III FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		1.73E-03	TTM_2SSPS_TRAINB	SSPS TRAIN B UNAVAILABLE DUE TEST	
		2.40E-03	HRADEP-POST-220		
					Inverter 1-IV fails during operation causing a loss of 120V AC Vital Instrument Board IV. Loss of 120V Vital Instrument Power causes a reactor trip on closure of the
12	1.73E-07	1.00E+00	%2LDDAC	Loss of 120V AC Vital Instrument Board IV	
		4.63E-02	INVFR2INV_2354-G_IE	INVERTER 2-IV FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		1.73E-03	TTM_2SSPS_TRAINA	SSPS TRAIN A UNAVAILABLE DUE TEST	MSIV. SSPS Train A is out of service due to testing and failure of the 120V Vital Instrument Power 2-IV fails SSPS Train B. Operator fails to manually start AFW, and the operator fails to perform cooldown with MFW following AFW failure. Operator fails to establish feed and bleed.
		2.40E-03	HRADEP-POST-220		
13	1.72E-07	1.00E+00	%2LDCAC	Loss of 120V AC Vital Instrument Board III	Inverter 2-III fails during operation causing a loss of 120V AC Vital Instrument Board III. Loss of 120V Vital Instrument Power causes a reactor trip on closure of the MSIV. SSPS Train B is in maintenance and failure of the 120V Vital Instrument Power 2-III fails SSPS Train A. Operator fails to manually start AFW, and the operator fails to perform cooldown with MFW following AFW failure. Operator fails to establish feed and bleed.
		4.63E-02	INVFR12NV_2353-F_IE	INVERTER 2-III FAILS DURING OPERATION	
		1.72E-03	MTM_2SSPS_TRAINB	SSPS TRAIN B UNAVAILABLE DUE MAINTENANCE	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.40E-03	HRADEP-POST-220		
14	1.71E-07	1.00E+00	%0TLERCW	Total Loss of ERCW	Total loss of ERCW event due to a common cause event of all 8 ERCW pumps failing to run. Total loss of ERCW fails both Trains of ECCS. Operator fails
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		6.32E-05	U0_ERCW_PMP_FR_CCF_IE_ALL	CCF OF ALL ERCW PUMPS FAILS TO RUN IE	

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
		3.00E-03	HRADEP-POST-206		to makeup CST using the demineralized water pumps and operator fails to align HPFP to provide makeup after CST depletion. This is a GTRAN event with failure of all ECCS and failure of long term heat removal using the AFW system.
15	1.44E-07	1.44E-07	%0FLHPFPAB757A2	Flood event induced by break of HPFP line in room 757.0-A2	High pressure fire protection piping to a hose station in the 6.9kV & 480V shutdown board room A on El 757 ruptures. Trip of ERCW and charging pumps will follow the loss of the boards. Upon further propagation to the 6.9kV & 480V shutdown board room B this event results in a station blackout with no recovery. Manual operation of the turbine driven AFW pump was not credited for flood sequences.
16	1.43E-07	1.00E+00	%0TLERCW	Total Loss of ERCW	Total loss of ERCW due to a common cause event resulting in the loss of all 8 ERCW pumps. The loss of ERCW and consequential loss of CCS induced seal LOCA with no injection.
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.50E-03	RCPSEAL480	RCP SEAL 480GPM	
		6.32E-05	U0_ERCW_PMP_FR_CCF_IE_ALL	CCF OF ALL ERCW PUMPS FAILS TO RUN IE	
17	1.28E-07	1.00E+00	%0TLERCW	Total Loss of ERCW	Total loss of ERCW due to common cause plugging of the traveling screens. Failure to
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		6.73E-04	U0_ERCW_TS_PL_CCF_IE_ALL	CCF of all components in group 'U0_ERCW_TS_PL_CCF_IE'	clear the traveling screens before plant trip. The loss of ERCW causes loss of all ECCS pumps. The loss of ERCW also causes loss of plant air and the operators then fail to manually control AFW.
		2.10E-04	HRADEP-POST-171		
18	1.27E-07	1.27E-07	%0FLRCW757A17	Flood event induced by rupture of RCW line in room 757.0-A17	Raw water pipe in the Unit 2 side personal and equipment access room on El 757 ruptures. Upon propagation to the 6.9kV & 480V shutdown board room B ERCW and charging pumps will trip following the loss of the boards. Upon further propagation to the 6.9kV & 480V shutdown board room A this event results in a station blackout with no recovery. Manual operation of the turbine driven AFW pump was not credited for flood sequences.

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
19	1.27E-07	1.27E-07	%0FLRCW757A9	Flood event induced by rupture of RCW line in room 757.0-A9	Raw water pipe in the Unit 1 side personal and equipment access room on El 757 ruptures. Upon propagation to the 6.9kV & 480V shutdown board room A ERCW and charging pumps will trip following the loss of the boards. Upon further propagation to the 6.9kV & 480V shutdown board room B this event results in a station blackout with no recovery. Manual operation of the turbine driven AFW pump was not credited for flood sequences.
20	1.22E-07	1.00E+00	%0TLERCW	Total Loss of ERCW	Total loss of ERCW due to common cause plugging of the traveling screens with a failure of operators to clear the screens before a plant trip. Loss of ERCW causes failure of all ECCS pumps. Induced seal LOCA with no injection.
		3.80E-03	DHAERCWS	Operators fail to clear ERCW screens before plant trip	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		6.73E-04	U0_ERCW_TS_PL_CCF_IE_ALL	CCF of all components in group 'U0_ERCW_TS_PL_CCF_IE'	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
21	1.07E-07	1.07E-07	%0FLHPFPAB757A24	Flood event induced by break of HPFP line in room 757.0-A24	High pressure fire protection piping to a hose station in the 6.9kV & 480V shutdown board room B on El 757 ruptures. Trip of ERCW and charging pumps will follow the loss of the boards. Upon further propagation to the 6.9kV & 480V shutdown board room A this event results in a station blackout with no recovery. Manual operation of the turbine driven AFW pump was not credited for flood sequences.
22	1.00E-07	1.00E-07	%2EX	EXCESSIVE LOCA (VESSEL RUPTURE)	Vessel rupture leads directly to core damage.
23	8.93E-08	2.88E-03	%2SLOCAL	Small LOCA Stuck Open Safety Relief Valve	Small LOCA due to an inadvertent stuck open safety relief valve. The operator fails to align high pressure recirculation. Cooldown to LPI conditions fails. The operators fail to realign the containment spray pump to the sump to refill the RWST.
		3.10E-05	HRADEP-POST-193		
24	7.75E-08	1.00E+00	%0TLERCW	Total Loss of ERCW	This cutset is an overly conservative treatment of the initiating event since both pump initiating failures are multiplied by 8760. This cutset could be
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.97E-02	POEFR0PMP_06700028IE	ERCW PUMP A-A FAILS TO RUNINITIATING EVENT WBN-0-67-28	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		2.97E-02	POEFR0PMP_06700047IE	ERCW PUMP E-B FAILS TO RUN CC 1/4 INITIATING EVENT WBN-0-67-E-B	addressed via a mutually exclusive event because there are cutsets with the 24 hour mission time in the mitigating portion of the model. This sequence is a loss of ERCW with operator failure to start the standby pump and failure of control of AFW due to the loss of plant air.
		9.70E-05	HRADEP-POST-180		
25	7.75E-08	1.00E+00	%0TLERCW	Total Loss of ERCW	See cutset 24
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.97E-02	POEFR0PMP_06700036IE	ERCW PUMP C-A FAILS TO RUN INITIATING EVENT WBN-0-67-36	
		2.97E-02	POEFR0PMP_06700047IE	ERCW PUMP E-B FAILS TO RUN CC 1/4 INITIATING EVENT WBN-0-67-E-B	
		9.70E-05	HRADEP-POST-180		
26	7.75E-08	1.00E+00	%0TLERCW	Total Loss of ERCW	See cutset 24
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.97E-02	POEFR0PMP_06700028IE	ERCW PUMP A-A FAILS TO RUNINITIATING EVENT WBN-0-67-28	
		2.97E-02	POEFR0PMP_06700055IE	ERCW PUMP G-B FAILS TO RUNINITIATING EVENT WBN-0-67-55	
		9.70E-05	HRADEP-POST-180		
27	7.75E-08	1.00E+00	%0TLERCW	Total Loss of ERCW	See cutset 24
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		2.97E-02	POEFR0PMP_06700036IE	ERCW PUMP C-A FAILS TO RUN INITIATING EVENT WBN-0-67-36	
		2.97E-02	POEFR0PMP_06700055IE	ERCW PUMP G-B FAILS TO RUNINITIATING EVENT WBN-0-67-55	
		9.70E-05	HRADEP-POST-180		
28	7.62E-08	1.00E+00	%0TLERCW	Total Loss of ERCW	This cutset is an overly conservative treatment of the initiating event since both pump initiating failures are multiplied by 8760. This cutset could be addressed via a mutually exclusive event because there are cutsets with the 24 hour mission time in the mitigating portion of the model. This sequence is a total loss of ERCW with failure of starting the standby pump leading to a seal LOCA with no injection available.
		1.80E-03	HAAEIE	Start standby ERCW pump - operating pump fails - normal ops	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.97E-02	POEFR0PMP_06700028IE	ERCW PUMP A-A FAILS TO RUNINITIATING EVENT WBN-0-67-28	
		2.97E-02	POEFR0PMP_06700047IE	ERCW PUMP E-B FAILS TO RUN CC 1/4 INITIATING EVENT WBN-0-67-E-B	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
29	7.62E-08	1.00E+00	%0TLERCW	Total Loss of ERCW	See cutset 28
		1.80E-03	HAAEIE	Start standby ERCW pump - operating pump fails - normal ops	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.97E-02	POEFR0PMP_06700036IE	ERCW PUMP C-A FAILS TO RUN INITIATING EVENT WBN-0-67-36	
		2.97E-02	POEFR0PMP_06700047IE	ERCW PUMP E-B FAILS TO RUN CC 1/4 INITIATING EVENT WBN-0-67-E-B	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
30	7.62E-08	1.00E+00	%0TLERCW	Total Loss of ERCW	See cutset 28
		1.80E-03	HAAEIE	Start standby ERCW pump - operating pump fails - normal ops	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.97E-02	POEFR0PMP_06700028IE	ERCW PUMP A-A FAILS TO RUNINITIATING EVENT WBN-0-67-28	
		2.97E-02	POEFR0PMP_06700055IE	ERCW PUMP G-B FAILS TO RUNINITIATING EVENT WBN-0-67-55	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
31	7.62E-08	1.00E+00	%0TLERCW	Total Loss of ERCW	See cutset 28
		1.80E-03	HAAEIE	Start standby ERCW pump - operating pump fails - normal ops	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.97E-02	POEFR0PMP_06700036IE	ERCW PUMP C-A FAILS TO RUN INITIATING EVENT WBN-0-67-36	
		2.97E-02	POEFR0PMP_06700055IE	ERCW PUMP G-B FAILS TO RUNINITIATING EVENT WBN-0-67-55	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
32	6.37E-08	2.85E-01	%2RTIE	Reactor Trip	A reactor trip occurs with common cause failure of both A and B safeguard drivers cards. AFW auto start fails due to the SSPS failure and the operators fail to initiate AFW. Operator fails to cooldown with MFW. Failure of feed and bleed due to operator action.
		9.32E-05	U2_ESF_SGD_CF_517_CCF_1_2	CCF of two components: SGDCF2SGD_099A517A & SGDCF2SGD_099A517B	
		2.40E-03	HRADEP-POST-220		

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
33	6.25E-08	2.50E-03	%2SSBO-1	SECONDARY BREAK OUTSIDE CONTAINMENT SG 1	A secondary side break occurs outside containment. The operators fail to terminate SI resulting in filling the pressurizer and opening the PORV, and operator then fails to align high pressure recirculation.
		2.50E-05	HRADEP-POST-309		
34	6.25E-08	2.50E-03	%2SSBO-2	SECONDARY BREAK OUTSIDE CONTAINMENT SG 2	See cutset 33
		2.50E-05	HRADEP-POST-309		
35	6.25E-08	2.50E-03	%2SSBO-3	SECONDARY BREAK OUTSIDE CONTAINMENT SG 3	See cutset 33
		2.50E-05	HRADEP-POST-309		
36	6.25E-08	2.50E-03	%2SSBO-4	SECONDARY BREAK OUTSIDE CONTAINMENT SG 4	See cutset 33
		2.50E-05	HRADEP-POST-309		
37	5.86E-08	2.85E-01	%2RTIE	Reactor Trip	Consequential seal LOCA due to loss of all ERCW following a reactor trip with no injection.
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		3.88E-06	U0_ERCW_PMP_FR_CCF_ALL	CCF of all components in group 'U0_ERCW_PMP_FR_CCF'	

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
38	5.44E-08	5.44E-08	%0FLHPFPAB757A5	Flood event induced by break of HPFP line in room 757.0-A5	High pressure fire protection pipe to a hose station in the 480V shutdown board room 1B on El 757 ruptures. Trip of ERCW and charging pumps follow the loss of the boards. Following propagation to the other 6.9kV and 480V shutdown board rooms this event induces a station blackout with no recovery. Manual operation of the turbine driven AFW pump was not credited.
39	5.27E-08	5.49E-04	%0FLHPFPABF	Flood event induced by HPFP in the common areas of the Auxiliary Building (multi	High pressure fire protection pipe in the common areas of the Auxiliary Building ruptures. The pipe rupture frequency includes all the piping in the common area and is assumed to occur on the refueling deck, thus disabling Train 1B air dryers due to spray effect. This
		7.39E-04	AOCFC0PCV_03300004	AOV FAILS TO CLOSE ON DEMAND WBN-0-33-4	
		1.30E-01	FLAB4F		

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		1.00E+00	HAFR1_FL		will render compressed air to both Turbine Building and Auxiliary Building compressors unavailable through Train B. Random failure of AOV 33-4 fails both the ACAS and the CAS on Train A. Local control of the AFW pump is not credited due to the flood. The flood is not isolated before the passive sump in the Auxiliary Building overflows and floods El 676, thus disabling both trains of RHR pumps.
40	5.18E-08	2.32E-01	%2TTIE	Turbine Trip	A turbine trip occurs with common cause failure of A and B safeguard driver cards. AFW auto start fails due to the SSPS failure and the operators fail to initiate AFW. Operator fails to cooldown with MWF. Failure of feed and bleed due to operator action.
		9.32E-05	U2_ESF_SGD_CF_517_CCF_1_2	CCF of two components: SGDCF2SGD_099A517A & SGDCF2SGD_099A517B	
		2.40E-03	HRADEP-POST-220		
41	4.97E-08	1.00E+00	%2CCS	Total Loss of Component Cooling System Unit 2	Total loss of CCS due to a common cause event of all CCS pumps failing to run. Thermal barrier cooling fails due to the loss of all CCS. Operator is able to align ERCW
		3.78E-03	MTM_2PMP_0620108A	WBN-2-PMP-062-0108-A CCP 1A-A IN MAINTENANCE	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		2.75E-04	U0-CCS-PCO-FR-CCF-IE-ALL	CCF of CCS PUMPS FAIL TO RUN, CCS HX PLUGGS, & CCS HX EXCESSIVE LEAKAGE/RUPTURE	cooling to the CCP 2A-A pump but it is unavailable due to maintenance. Loss of thermal barrier cooling and loss of RCP seal injection induced seal LOCA with no injection available.
42	4.76E-08	2.32E-01	%2TTIE	Turbine Trip	Turbine trip with a common cause failure of all the ERCW pumps leading to a consequential seal LOCA with no injection.
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		3.88E-06	U0_ERCW_PMP_FR_CCF_ALL	CCF of all components in group 'U0_ERCW_PMP_FR_CCF'	
43	4.61E-08	2.88E-03	%2SLOCAL	Small LOCA Stuck Open Safety Relief Valve	A small LOCA results from an inadvertent stuck open pressurizer safety relief valve. Operator inadvertently resets SI. The operator fails to manually swapover to recirculation and fails to cooldown to LPI conditions. The operator then fails to align the containment spray pumps to refill the RWST.
		1.60E-05	HRADEP-POST-192		

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
44	4.21E-08	4.21E-08	%0FLHPFPAB757A21	Flood event induced by break of HPFP line in room 757.0-A21	High pressure fire protection pipe to a hose station in the 480V shutdown board room 2A on El 757 ruptures. Trip of ERCW and charging pumps follow the loss of the boards. Following propagation to the other 6.9kV and 480V shutdown board rooms this event induces a station blackout with no recovery. Manual operation of the turbine driven AFW pump was not credited for flood sequences.
45	4.05E-08	4.05E-08	%0FLHPFPAB772A7	Flood event induced by break of HPFP line in room 772.0-A7	A high pressure fire protection pipe ruptures in the mechanical equipment room on Unit 1 side at El 772. After propagation to the 480V board rooms water will impact Unit 1 and Unit 2 inverters on El 772. Shorting the inverters will cause a plant trip. Further water propagation down to the 480V shutdown boards on El 757 will induce a station blackout with no recovery. Manual operation of the turbine driven AFW pump was not credited for flood sequences.
46	4.04E-08	1.00E+00	%2CCS	Total Loss of Component Cooling System Unit 2	Total loss of CCS due to a common cause event of all

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
		6.50E-02	HCCSR4	Align & Initiate Alternate Cooling to 1A-A CCP, 1B-B failed	CCS pumps failing to run. Thermal barrier cooling fails due to the loss of all CCS. Operator fails to align ERCW cooling to the CCP 2A-A pump. Loss of thermal barrier cooling and loss of RCP seal injection induced seal LOCA with no injection available.
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.50E-03	RCPSEAL480	RCP SEAL 480GPM	
		2.75E-04	U0-CCS-PCO-FR-CCF-IE-ALL	CCF of CCS PUMPS FAIL TO RUN, CCS HX PLUGGS, & CCS HX EXCESSIVE LEAKAGE/RUPTURE	
47	3.96E-08	3.60E-06	%2MLOCA-CL1	MLOCA ON COLD LEG 1	A medium LOCA occurs with failure to align high pressure recirculation and the operator fails to cooldown to LPR conditions.
		1.10E-02	HRADEP-POST-289		
48	3.96E-08	3.60E-06	%2MLOCA-CL2	MLOCA ON COLD LEG 2	See cutset 47
		1.10E-02	HRADEP-POST-289		
49	3.96E-08	3.60E-06	%2MLOCA-CL3	MLOCA ON COLD LEG 3	See cutset 47
		1.10E-02	HRADEP-POST-289		
50	3.96E-08	3.60E-06	%2MLOCA-CL4	MLOCA ON COLD LEG 4	See cutset 47
		1.10E-02	HRADEP-POST-289		

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
51	3.94E-08	1.00E+00	%2LVBB3	Loss of Battery Board 3	Loss of battery board III IE due to failure of the battery to operate. Loss of battery board III fails the MDAFW pump A, TDAFW pump fails to start, and MDAFW pump B fails due to an isolation pre-initiator. This results in a total loss of all AFW. Loss of the battery board also fails the ability to cooldown on MFW. Operator fails to establish feed and bleed. This is a GTRAN event with a loss of all AFW, MFW, and Bleed and Feed.
		1.63E-02	BATFR0BAT_2363-F_IE	BATT IV FAILS DURING OPERATION (0-BAT-236-3-F)	
		1.60E-02	HAOB2	Establish RCS Bleed and Feed cooling given no CCPS running	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.43E-02	PTSF12PMP_003001AS	PUMP FAILS TO START AND RUN FOR 1 HOUR WBN-1-3-1AS	
		6.90E-03	WHEMDA_2	Motor Driven AFW Pump Train B Isolation Test Error	
52	3.94E-08	1.00E+00	%2LVBB4	Loss of Battery Board 4	Loss of battery board 4 IE due to failure of the battery to operate. Loss of battery board 4 fails the MDAFW pump B, TDAFW pump fails to start, and MDAFW pump A fails due to isolation pre-initiator. This results in a total loss of all AFW. Loss of the battery board also fails the ability to cooldown on MFW. Operator fails to establish feed and bleed. This is a GTRAN event with a loss of all AFW, MFW, and Bleed and Feed.
		1.63E-02	BATFR0BAT_2364-G_IE	BATT IV FAILS DURING OPERATION (0-BAT-236-3-F)	
		1.60E-02	HAOB2	Establish RCS Bleed and Feed cooling given no CCPS running	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.43E-02	PTSF12PMP_003001AS	PUMP FAILS TO START AND RUN FOR 1 HOUR WBN-1-3-1AS	
		6.90E-03	WHEMDA_1	Motor Driven AFW Pump Train A Isolation Test Error	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
53	3.94E-08	3.94E-08	%0FLHPFPAB772A10	Flood event induced by break of HPFP line in room 772.0-A10	HPFP line to hose station in room 772.0-A10 ruptures. Upon propagation to the 480V board rooms, water impacts Unit 1 and Unit 2 inverters on El 772. Shorting the inverters will cause a plant trip. Further water propagation down to the 480V shutdown boards on El 757 will induce a station blackout with no recovery. Manual operation of the turbine driven AFW pump was not credited for flood sequences.
54	3.78E-08	1.00E+00	%2LDCAC	Loss of 120V AC Vital Instrument Board III	Loss of 120V AC instrument board III due to an IE of inverter 2-III failing during operation. Loss of this board fails the MDAFW pump A, TDAFW pump fails to start, and MDAFW pump B fails due to an isolation pre-initiator. This results in a total loss of all AFW. Operator fails to cooldown with MFW and operator fails to establish feed and bleed. This is a GTRAN event with a loss of all AFW, MFW and Bleed and Feed.
		4.63E-02	INVFR12NV_2353-F_IE	INVERTER 2-III FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.43E-02	PTSF12PMP_003001AS	PUMP FAILS TO START AND RUN FOR 1 HOUR WBN-1-3-1AS	
		6.90E-03	WHEMDA_2	Motor Driven AFW Pump Train B Isolation Test Error	
		5.40E-03	HRADEP-POST-218		
55	3.78E-08	1.00E+00	%2LDDAC	Loss of 120V AC Vital Instrument Board IV	Loss of 120V AC instrument board IV due to an IE of inverter 2-IV failing during operation. Loss of this board fails the MDAFW pump B,
		4.63E-02	INVFR2INV_2354-G_IE	INVERTER 2-IV FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
		2.43E-02	PTSF12PMP_003001AS	PUMP FAILS TO START AND RUN FOR 1 HOUR WBN-1-3-1AS	TDAFW pump fails to start, and MDAFW pump A fails due to an isolation pre-initiator. This results in a total loss of all AFW. Operator fails to cooldown with MFW and operator fails to establish feed and bleed. This is a GTRAN event with a loss of all AFW, MFW and Bleed and Feed.
		6.90E-03	WHEMDA_1	Motor Driven AFW Pump Train A Isolation Test Error	
		5.40E-03	HRADEP-POST-218		
56	3.61E-08	1.01E-02	%0LOSP-GR	Loss of Offsite Power (Grid Related)	Grid related LOSP event with DG 2B-B failing to run and DG 2A-A unavailable due to maintenance. TDAFW is running until battery depletion. Operator fails to control SG level manually after battery depletion and recovery of LOSP fails.
		1.46E-02	DGGFR2GEN_0822B-B	DG 2B-B FAILS FAILS TO RUN (WBN-2-GEN -082-0002B -B)	
		1.00E+00	FL-BATDEP	Battery Depleted FLAG	
		2.00E-01	HAOSBF	Steam generator feed with manual level control fails	
		1.51E-02	MTM_2GEN_0822A-A	DIESEL 2A-A MAINTENANCE	
		8.10E-02	XSBO05	Recovery Sequence 5 (One EDG Fails to Start and One Fails to Run) GR	
57	3.51E-08	1.00E+00	%0TLERCW	Total Loss of ERCW	Total loss of ERCW due to a common cause event all ERCW strainer plugging. The IE resulted from the ERCW strainer plugging, then the auto backwash fails due to a common cause event of motor
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		4.64E-05	U0_ERCW_FCV_FO_CCF_ALL	CCF of all components in group 'U0_ERCW_FCV_FO_CCF'	

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
		1.58E-02	U0_ERCW_STR_PL_CCF_IE_ALL	CCF of all components in group 'U0_ERCW_STR_PL_CCF_IE'	backwash due to a check valve failing to open. Total loss of ERCW fails both trains of ECCS. Loss of thermal barrier cooling and loss of RCP seal injection induced seal LOCA with no injection available.
58	3.51E-08	1.00E+00	%2CCS	Total Loss of Component Cooling System Unit 2	Total loss of CCS event due to a common cause failure of all CCS pumps to run. Operator successfully aligned ERCW cooling to the 2A CCP but pump fails due to a loss of room cooling. Loss of thermal barrier cooling and loss of RCP seal injection induced seal LOCA with no injection available.
		2.66E-03	FNSFR2FAN_03000183	CCP A ROOM COOLER FAN FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		2.75E-04	U0-CCS-PCO-FR-CCF-IE-ALL	CCF of CCS PUMPS FAIL TO RUN, CCS HX PLUGGS, & CCS HX EXCESSIVE LEAKAGE/RUPTURE	
59	3.43E-08	8.12E-03	%0LOSP-PC	Loss of Offsite Power (Plant Centered)	Plant centered LOSP event with DG 2B-B failing to run and DG 2A-A unavailable due to maintenance. TDAFW is running until battery depletion. Operator fails to control SG level manual after battery depletion and recovery of LOSP fails.
		1.46E-02	DGGFR2GEN_0822B-B	DG 2B-B FAILS FAILS TO RUN (WBN-2-GEN -082-0002B -B)	
		1.00E+00	FL-BATDEP	Battery Depleted FLAG	
		2.00E-01	HAOSBF	Steam generator feed with manual level control fails	
		1.51E-02	MTM_2GEN_0822A-A	DIESEL 2A-A MAINTENANCE	
		9.56E-02	XSBO04	Recovery Sequence 5 (One EDG Fails to Start and One Fails to Run) PC	
60	3.36E-08	1.00E+00	%2LDDAC	Loss of 120V AC Vital Instrument Board IV	Loss of 120V AC Board IV IE due to a failure of an inverter to operate resulting in a reactor trip. Loss of this board fails
		4.63E-02	INVFR2INV_2354-G_IE	INVERTER 1-IV FAILS DURING OPERATION	

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
		1.83E-03	MTM_2SSPS_TRAINA	SSPS TRAIN A UNAVAILABLE DUE MAINTENANCE	MDAFW pump B. MDAFW pump A fails to start due to SSPS Train A maintenance unavailability. TDAFW fails to start due to the loss of 120VAC Board IV (failing ESFAS Train B) and the SSPS Train A in maintenance. The operator fails to manually start AFW. Operator also fails to restore MFW and establish bleed and feed cooling. GTRAN event with failure of AFW, MFW, and bleed and feed cooling.
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		4.40E-04	HRADEP-POST-238		
61	3.34E-08	9.81E-03	%0TLPCA	Total Loss of Plant Compressed Air	Loss of plant compressed air IE with both ACAS compressors failing to run. Operator fails to restore AFW control following initiator and loss of air and fails to establish RCS Bleed and Feed cooling, GTRAN event with failure of AFW, MFW, and Bleed and Feed. The common cause failure of the ACAS compressors to run and the start failures are lower than the independent failure of both to run. Common cause assessment and failure rates were reviewed.
		6.29E-02	CMPSR0COMP03200060	COMPRESSOR A-A FAILS TO RUN WBN-0-32-60	
		6.29E-02	CMPSR0COMP03200086	COMPRESSOR B-B FAILS TO RUN WBN-0-32-86	
		8.60E-04	HRADEP-POST-221		
62	3.26E-08	1.46E-01	%2PLMFW	Partial Loss of Main Feedwater	GTRAN IE due to a partial loss of MFW. All AFW fails to start

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		9.32E-05	U2_ESF_SGD_CF_517_CCF_1_2	CCF of two components: SGDCF2SGD_099A517A & SGDCF2SGD_099A517B	due to the common cause failure of both trains' Safeguard Driver Cards. The operator fails to manually start AFW. Operator fails to perform cooldown with MFW after successful recovery and operator fails to establish Bleed and Feed cooling. GTRAN event with no AFW, MFW, and Bleed and Feed cooling.
		2.40E-03	HRADEP-POST-220		
63	3.25E-08	1.00E+00	%2LDCAC	Loss of 120V AC Vital Instrument Board III	Loss of 120V AC instrument board III due to a failure of an inverter to operate. Loss of instrument board IIII fails MDAFW Train A. Train B MDAFW fails due to loss of the B train Safeguard Driver Card to start the pump and the operator failing to manually start AFW. TDAFW pump fails due to the loss of the 120V AC instrument board III and the loss of Train B of ESFAS fails to auto start and operator fails to manually start the pump. The operator also fails to restore MFW and establish RCS Bleed and Feed cooling.
		4.63E-02	INVFR12NV_2353-F_IE	INVERTER 1-III FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		1.77E-03	SGDCF2SGD_099A517B	WBN-2-99-A517-B Safeguard Driver Card Fails	
		4.40E-04	HRADEP-POST-238		
64	3.25E-08	1.00E+00	%2LDDAC	Loss of 120V AC Vital Instrument Board IV	Loss of 120V AC instrument board IV due to a failure of an inverter to operate. Loss of instrument board IV fails MDAFW Train A. Train A
		4.63E-02	INVFR2INV_2354-G_IE	INVERTER 1-IV FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		1.77E-03	SGDCF2SGD_099A517A	WBN-2-99-A517-A Safeguard Driver Card Fails	MDAFW fails due to loss of the A Train Safeguard Driver Card to start the pump and the operator failing to manually start AFW. TDAFW fails to start on the loss of the 120V AC instrument board IV and the loss of Train B of ESFAS to auto start and operator fails to manual start the pump. The operator also fails to restore MFW and establish RCS Bleed and Feed cooling.
		4.40E-04	HRADEP-POST-238		
65	3.25E-08	1.00E+00	%2CCS	Total Loss of Component Cooling System Unit 2	Total loss of CCS IE due to a common cause failure of all CCS pumps to run. Thermal barrier cooling is lost as a result of the failure of both trains of CCS. RCP seal injection fails due to a relief valve failing to open after the Phase A isolation signal. Loss of thermal barrier cooling and loss of RCP seal injection induced seal LOCA with no injection available.
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		2.47E-03	RLVFO2RFV_0620636	WBN-1-RFV-062-0636-S RELIEF VALVE FAILS TO OPEN	
		2.75E-04	U0-CCS-PCO-FR-CCF-IE-ALL	CCF of CCS PUMPS FAIL TO RUN, CCS HX PLUGGS, & CCS HX EXCESSIVE LEAKAGE/RUPTURE	
66	3.25E-08	2.50E-03	%2SSBO-1	SECONDARY BREAK OUTSIDE CONTAINMENT SG 1	Secondary side break IE in SG 1. After IE there is successful

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		1.30E-05	HRADEP-POST-305		SI then the operator fails to terminate SI. Failure of auto swapover occurs due to two operator errors as a result of inadvertently Reset SI Signal and failure to recover from auto swapover failure.
67	3.25E-08	2.50E-03	%2SSBO-2	SECONDARY BREAK OUTSIDE CONTAINMENT SG 2	
		1.30E-05	HRADEP-POST-305		See cutset 66
68	3.25E-08	2.50E-03	%2SSBO-3	SECONDARY BREAK OUTSIDE CONTAINMENT SG 3	
		1.30E-05	HRADEP-POST-305		See cutset 66
69	3.25E-08	2.50E-03	%2SSBO-4	SECONDARY BREAK OUTSIDE CONTAINMENT SG 4	
		1.30E-05	HRADEP-POST-305		See cutset 66
70	3.18E-08	1.00E+00	%2LDCAC	Loss of 120V AC Vital Instrument Board III	Loss of 120V AC instrument board III due to a failure of an inverter to operate. Loss of instrument board III fails MDAFW Train A. Train B MDAFW fails to start due to
		4.63E-02	INVFR12NV_2353-F_IE	INVERTER 2-III FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		1.73E-03	TTM_2SSPS_TRAINB	SSPS TRAIN B UNAVAILABLE DUE TEST	loss of the B train due to unavailability of ESFAS in test and the operator fails to manually start AFW. TDAFW pump fails due to the loss of the 120V AC instrument board III and the loss of Train B of ESFAS fails to auto start and operator fails to manually start the pump. The operator also fails to restore MFW and establish RCS Bleed and Feed cooling.
		4.40E-04	HRADEP-POST-238		
					Loss of 120V AC instrument board IV due to a failure of an inverter to operate. Loss of instrument board IV fails MDAFW Train B. Train A MDAFW fails to start due to loss of the A train due to unavailability of ESFAS in test and the operator failing to manually start AFW. TDAFW pump fails due to the loss of the 120V AC instrument board IV and the loss of Train A of ESFAS fails to auto start and operator fails to manual start the pump. The operator also fails to restore MFW and establish RCS Bleed and Feed cooling.
71	3.18E-08	1.00E+00	%2LDDAC	Loss of 120V AC Vital Instrument Board IV	
		4.63E-02	INVFR2INV_2354-G_IE	INVERTER 1-IV FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		1.73E-03	TTM_2SSPS_TRAINA	SSPS TRAIN A UNAVAILABLE DUE TEST	
		4.40E-04	HRADEP-POST-238		

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
72	3.16E-08	1.00E+00	%2LDCAC	Loss of 120V AC Vital Instrument Board III	Loss of 120V AC instrument board III due to a failure of an inverter to operate. Loss of instrument board III fails MDAFW Train A. Train B MDAFW fails to start due to loss of the A train due to unavailability of ESFAS due to maintenance and the operator failing to manually start AFW. TDAFW pump fails due to the loss of the 120V AC instrument board III and the loss of Train B of ESFAS fails to auto start and operator fails to manual start the pump. The operator also fails to restore MFW and establish RCS Bleed and Feed cooling.
		4.63E-02	INVFR12NV_2353-F_IE	INVERTER 1-III FAILS DURING OPERATION	
		1.72E-03	MTM_2SSPS_TRAINB	SSPS TRAIN B UNAVAILABLE DUE MAINTENANCE	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		4.40E-04	HRADEP-POST-238		
73	3.14E-08	5.49E-04	%0FLHPFPABF	Flood event induced by HPFP in the common areas of the Auxiliary Building (multi	High pressure fire protection pipe in the common areas of the Auxiliary Building ruptures. The pipe rupture frequency includes all the piping in the common area and is assumed to occur on the refueling deck, thus disabling Train 1B air dryers due to spray effect. This will render compressed air from both Turbine Building and Auxiliary Building compressors unavailable through Train B. The flood also fails the ACAS
		1.00E+00	FL_SPARE_250_CHGR_NOT_A	SPARE CHARGER NOT ALIGNED FOR A TRAIN	
		1.30E-01	FLAB4F		
		1.00E+00	HAFR1_FL		
		2.20E-03	MTM_0CHGR2391	250VDC CHARGER 1 MAINTENANCE	
		2.00E-01	SUMMER	SUMMER SEASON	
		1.00E+00	U1_250BATTDEP	Unit 1 250V Battery Life Depleted	

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
					compressor A due to flood in the Auxiliary building. The flood is not isolated before the passive sump in the Auxiliary Building overflows and floods El 676, thus disabling both trains of RHR pumps. Accessibility limitations due to the flood event results in not crediting restoration of Restore AFW control following initiator and loss of air All of the Control Air (PD) Compressors fail due to the loss of the support from the 480V Unit Board 1A that supplies power to the temporary ventilation fans used during the summer to cool the compressors. The 480V Unit Board 1A fails due to its dependency on the Battery Board 1. Battery Board 1 is unavailable to maintenance on the 250VDC CHARGER 1 and the spare charger not aligned to that train. No operator action is taken to locally control the AFW LCVs resulting in a failure of all AFW. It is recommended that the Maintenance & Spare Charger flag be removed or the spare charger be modeled and flag

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		1.00E+00	U1_250BATTDEP	Unit 1 250V Battery Life Depleted	update. This cutset does not match operating practice.
74	3.14E-08	5.49E-04	%0FLHPFPABF	Flood event induced by HPFP in the common areas of the Auxiliary Building (multi	High pressure fire protection pipe in the common areas of the Auxiliary Building ruptures. The pipe rupture frequency includes all the piping in the common area and is assumed to occur on the refueling deck, thus disabling Train 1B air dryers due to spray effect. This
		1.00E+00	FL_SPARE_250_CHGR_NOT_B	SPARE CHARGER NOT ALIGNED FOR B TRAIN	
		1.30E-01	FLAB4F		
		1.00E+00	HAFR1_FL		
		2.20E-03	MTM_0CHGR2392	250VDC CHARGER 2 MAINTENANCE	
		2.00E-01	SUMMER	SUMMER SEASON	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
					will render compressed air from both Turbine Building and Auxiliary Building compressors unavailable through Train B. The flood also fails the ACAS compressor A due to flood in the Auxiliary building. The flood is not isolated before the passive sump in the Auxiliary Building overflows and floods El 676, thus disabling both trains of RHR pumps. Accessibility limitations due to the flood event results in not crediting restoration of Restore AFW control following initiator and loss of air All of the Control Air (PD) Compressors fail due to the loss of the support from the 480V Unit Board 1A that supplies power to the temporary ventilation fails during the summer to cool the compressors. The 480V Unit Board 1A fails due to its dependency on the Battery Board 2. Battery Board 1 is unavailable to maintenance on the 250VDC CHARGER 2 and the spare charger not aligned to that train. No operator action is taken to locally control the AFW LCVs resulting in a failure of all AFW.

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
		1.00E+00	U1_250BATTDEP	Unit 1 250V Battery Life Depleted	It is recommended that the Maintenance & Spare Charger flag be removed or the spare charger be modeled and flag added to flag file in a future update. This cutset does not match operating practice
75	3.03E-08	1.00E+00	%2LVBB4	Loss of Battery Board 2	Loss of battery board 2 IE due to failure of the battery to operate. Loss of battery board 2 fails the CCP 2B-B and the CCS Train 2B-B pump. Operator fails to align alternate cooling to the 2A-A CCP pump from ERCW. CCP A fails due to unavailability due to maintenance. Loss of thermal barrier cooling and loss of RCP seal injection induced seal LOCA. It is recommended that all start failures of the 2B-B CCS pump be added to the MUX file for events when 2A-A CCS pump is in maintenance.
		1.63E-02	BATFR0BAT_2364-G_IE	BATT IV FAILS DURING OPERATION (0-BAT-236-3-F)	
		6.50E-02	HCCSR4	Align & Initiate Alternate Cooling to 1A-A CCP, 1B-B failed	
		5.97E-04	MTM_2PMP_0700059	WBN-2-PMP-070-0059 CCS PUMP 2A-A IN MAINTENANCE	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
76	3.00E-08	1.01E-02	%0LOSP-GR	Loss of Offsite Power (Grid Related)	Grid related LOSP event with all DGs failed due to a common cause event of all board room exhaust fans failing to start. Successful TDAFW until battery depletion. There are two failed recoveries taken on
		1.00E+00	FL-BATDEP	Battery Depleted FLAG	
		2.00E-01	HAOSBF	Steam generator feed with manual level control fails	
		1.62E-04	U0_EPS_VDG_FAN_FD2_CCF_ALL	CCF of all components in group 'U0_EPS_VDG_FAN_FD2_CCF'	

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
		9.18E-02	XSBO14	Recovery Sequence 7 (Common Cause of DG to Start) GR	this cutset. One is the operator fails to manually control SG level after battery depletion and the second is the LOSP recovery.
77	2.99E-08	1.46E-01	%2PLMFW	Partial Loss of Main Feedwater	Partial loss of MFW event with a common cause failure of all ERCW pumps failing to run after the IE. Loss of thermal barrier cooling and loss of RCP seal injection induced seal LOCA with no injection available.
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		3.88E-06	U0_ERCW_PMP_FR_CCF_ALL	CCF of all components in group 'U0_ERCW_PMP_FR_CCF'	
78	2.94E-08	8.12E-03	%0LOSP-PC	Loss of Offsite Power (Plant Centered)	Plant Centered LOSP event with all EDG fail due to a common cause event of all board room exhaust fan failing to start. Successful TDAFW until battery depletion. There are two recoveries taken on this cutset. One is the operator fails to manual control SG level after battery depletion and the second is the LOSP recovery.
		1.00E+00	FL-BATDEP	Battery Depleted FLAG	
		2.00E-01	HAOSBF	Steam generator feed with manual level control fails	
		1.62E-04	U0_EPS_VDG_FAN_FD2_CCF_ALL	CCF of all components in group 'U0_EPS_VDG_FAN_FD2_CCF'	
		1.12E-01	XSBO13	Recovery Sequence 7 (Common Cause of DG to Start) PC	
79	2.92E-08	8.58E-03	%0FLTBMF	Major flood in the Turbine Building	A major flood event is initiated in the Turbine Building due to a break of the condenser expansion joints; all equipment underneath EL 711 of the Turbine Building is expected to
		6.29E-02	CMPSR0COMP03200060	COMPRESSOR A-A FAILS TO RUN WBN-0-32-60	
		6.29E-02	CMPSR0COMP03200086	COMPRESSOR B-B FAILS TO RUN WBN-0-32-86	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		8.60E-04	HRADEP-POST-221		be lost. Flood is isolated before propagation to the Control Building. All the control air compressors fail due to the major flood in the turbine building. ACAS compressors A-A and B-B fail to run resulting in a total loss of air event. Operator fails to restore AFW control following loss of air. Operator also fails to establish RCS Bleed and Feed cooling.
80	2.91E-08	3.94E-05	%0FLRCWABMF	Major flood event induced by RCW in the common areas of the Auxiliary Building (	High pressure fire protection pipe in the common areas of the Auxiliary Building ruptures. The pipe rupture frequency
		7.39E-04	AOCFC0PCV_03300004	AOV FAILS TO CLOSE ON DEMAND WBN-0-33-4	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		1.00E+00	HAFR1_FL		includes all the piping in the common area and is assumed to occur on the refueling deck, thus disabling train 1B air dryers due to spray effect. This will render compressed air from both Turbine Building and Auxiliary Building compressors unavailable through train B. ACAS Train A fails due to the flood impact on the A-A compressor. All Control Air is lost due to an AOV fails to close. Accessibility limitations due to the flood event results in not crediting restoration of Restore AFW control following initiator and loss of air so all AFW is lost. Long term heat removal is lost due to a failure of recirculation. Both RHR pumps fail due to flood.
81	2.88E-08	2.88E-03	%2SLOCAL	Small LOCA Stuck Open Safety Relief Valve	Small LOCA IE with an operator error to inadvertently

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		1.00E-05	HRADEP-POST-127		Reset SI Signal preventing auto swapover. Operator fails to manually recover from auto swapover failure. After high head recirculation fails operator fails to depressurize/cooldown to low pressure injection. The operator then fails to align the containment spray to sump preventing refill of the RWST using the containment spray pumps.
82	2.88E-08	2.88E-03	%2SLOCAL	Small LOCA Stuck Open Safety Relief Valve	Small LOCA IE due to a stuck open Safety Relief value. After successful injection all sump strainers plug, preventing successful recirculation.
		1.00E-05	SMPPS2STN_SUMP2	SUMP SUCTION STRAINERS PLUGGED (GENERAL)	
83	2.86E-08	1.00E+00	%2LDCAC	Loss of 120V AC Vital Instrument Board III	Loss of 120V AC instrument board III IE due to a failure of the 2-III inverter followed by a failure of the 2-IV inverter failing to operate. All AFW fails due to the loss of the 2-III & 2-IV instrument boards. Operator fails to perform cooldown with MFW and fails to establish RCS Bleed and Feed cooling.
		4.63E-02	INVFR12NV_2353-F_IE	INVERTER 2-III FAILS DURING OPERATION	
		1.27E-04	INVFR2INV_2354-G	Inverter 2-IV Fails During Operation	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.40E-03	HRADEP-POST-218		
84	2.86E-08	1.00E+00	%2LDDAC	Loss of 120V AC Vital Instrument Board IV	Loss of 120V AC instrument board IV IE due to a failure of the 2-IV inverter followed by a failure of the 2-III inverter failing to operate. All AFW fails due to the loss of the 2-III & 2-IV
		1.27E-04	INVFR2INV_2353-F	Inverter 2-III Fails During Operation	
		4.63E-02	INVFR2INV_2354-G_IE	INVERTER 2-IV FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	

Top 100 CDF Cutsets					
#	Cutset Prob.	Event Prob.	Event	Event Description	Cutset Description
		5.40E-03	HRADEP-POST-218		instrument boards. Operator fails to perform cooldown with MFW and fails to establish RCS Bleed and Feed cooling.
85	2.79E-08	2.88E-03	%2SLOCAL	Small LOCA Stuck Open Safety Relief Valve	Small LOCA IE due to a stuck open safety relief valve. Operator fails to align high pressure recirculation after swappover. After high head recirculation fails operator fails to depressurize/cooldown to low pressure injection. Refilling the RWST using the containment spray pumps fails due to a pre-initiator isolating the flow path.
		1.70E-02	WHECSA	RWST Purification Flow Interference - (Containment Spray Diversion Path)	
		5.70E-04	HRADEP-POST-290		
86	2.79E-08	2.85E-01	%2RTIE	Reactor Trip	Reactor trip followed by a common cause failure of all the ERCW traveling screens to plug. Loss of thermal barrier cooling and loss of RCP seal injection induced seal LOCA.
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		1.84E-06	U0_ERCW_TS_PL_CCF_ALL	CCF of all components in group 'U0_ERCW_TS_PL_CCF'	
87	2.74E-08	9.81E-03	%0TLPCA	Total Loss of Plant Compressed Air	Loss of plant compressed air IE with both ACAS compressors failing to run due to common cause. Operator fails to restore AFW control following initiator and fails to establish RCS Bleed and Feed cooling. GTRAN event with failure of AFW, MFW, and Bleed and Feed.
		3.25E-03	U0_032_ACAS_CMP_FR_CCF_1_2	CCF of two components: CMPSR0COMP03200060 & CMPSR0COMP03200086	
		8.60E-04	HRADEP-POST-221		

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
88	2.63E-08	1.00E+00	%2CCS	Total Loss of Component Cooling System Unit 2	Total loss of component cooling due to a common cause event of all the CCS pumps failing to run. Loss of CCS fails thermal barrier cooling. Operator is unable to align ERCW cooling to A Charging Pump since the CCP 2A room cooling is in maintenance. Loss of thermal barrier cooling and loss of RCP seal injection induced seal LOCA with no injection available.
		2.00E-03	MTM_2FAN_03000183	CCP A ROOM COOLER FAN IN MAINTENANCE	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		2.75E-04	U0-CCS-PCO-FR-CCF-IE-ALL	CCF of CCS PUMPS FAIL TO RUN, CCS HX PLUGS, & CCS HX EXCESSIVE LEAKAGE/RUPTURE	
89	2.63E-08	1.00E+00	%2CCS	Total Loss of Component Cooling System Unit 2	Total loss of component cooling due to a common cause event of all the CCS pumps failing to run. Loss of CCS fails thermal barrier cooling. Operator is unable to align ERCW cooling to A Charging Pump since the CCP 2A room cooling is in test. Loss of thermal barrier cooling and loss of RCP seal injection induced seal LOCA with no injection available.
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		5.30E-02	RCPSEAL182	RCP SEAL 182 GPM	
		2.00E-03	TTM_2FAN_03000183	CCP A ROOM COOLER FAN IN TEST	
		2.75E-04	U0-CCS-PCO-FR-CCF-IE-ALL	CCF of CCS PUMPS FAIL TO RUN, CCS HX PLUGGS, & CCS HX EXCESSIVE LEAKAGE/RUPTURE	
90	2.63E-08	1.01E-02	%0LOSP-GR	Loss of Offsite Power (Grid Related)	Grid related LOSP followed by a common cause event for all the board room exhaust fans failing to start resulting in unavailability of all DGs. TDAFW pump fails to start and LOSP fails to be recovered.
		2.43E-02	PTSF12PMP_003001AS	PUMP FAILS TO START AND RUN FOR 1 HOUR WBN-1-3-1AS	
		1.62E-04	U0_EPS_VDG_FAN_FD2_CCF_ALL	CCF of all components in group 'U0_EPS_VDG_FAN_FD2_CCF'	
		6.62E-01	XSBO17	Recovery Sequence 8 (Common Cause of DG to Start AND TDAWF Fails to Start) GR	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
91	2.56E-08	1.01E-02	%0LOSP-GR	Loss of Offsite Power (Grid Related)	Grid related LOSP event with DG 2A-A failing to run and DG 2B-2B unavailable due to maintenance. TDAFW is running until battery depletion. Operator fails to control SG level manual after battery depletion and recovery of LOSP fails.
		1.46E-02	DGGFR2GEN_0822A-A	DIESEL GENERATOR FAILS TO RUN AFTER FIRST HOUR	
		1.00E+00	FL-BATDEP	Battery Depleted FLAG	
		2.00E-01	HAOSBF	Steam generator feed with manual level control fails	
		1.07E-02	MTM_2GEN_0822B-B	DIESEL 2B-B MAINTENANCE	
		8.10E-02	XSBO05	Recovery Sequence 5 (One EDG Fails to Start and One Fails to Run) GR	
92	2.56E-08	1.01E-02	%0LOSP-GR	Loss of Offsite Power (Grid Related)	Grid related LOSP event with DG 2B-B failing due to a board room exhaust fan failing to start and DG 2A-2A unavailable due to maintenance. TDAFW is running until battery depletion. Operator fails to control SG level manual after battery depletion and recovery of LOSP fails.
		1.00E+00	FL-BATDEP	Battery Depleted FLAG	
		9.13E-03	FNSFD2FAN_030462	BOARD ROOM EXHAUST FAN FAILS TO START OR RUN FIRST HOUR	
		2.00E-01	HAOSBF	Steam generator feed with manual level control fails	
		1.51E-02	MTM_2GEN_0822A-A	DIESEL 2A-A MAINTENANCE	
		9.18E-02	XSBO02	Recovery Sequence 1 (DG A and B FAILS TO START) GR	
93	2.54E-08	1.00E+00	%2LDCAC	Loss of 120V AC Vital Instrument Board III	Loss of 120V AC instrument board III IE due to a failure of the 2-III inverter followed by a failure of the 2-IV inverter failing to operate. All AFW fails due to the loss of the 2-III & 2-IV instrument boards. Operator fails to perform cooldown with MFW and fails to recover from auto swapover failure.
		4.63E-02	INVFR12NV_2353-F_IE	INVERTER 2-III FAILS DURING OPERATION	
		1.27E-04	INVFR2INV_2354-G	Inverter 2-IV Fails During Operation	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		4.80E-03	HRADEP-POST-278		

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
94	2.54E-08	1.00E+00	%2LDDAC	Loss of 120V AC Vital Instrument Board IV	Loss of 120V AC instrument board IV IE due to a failure of the 2-IV inverter followed by a failure of the 2-III inverter failing to operate. All AFW fails due to the loss of the 2-III & 2-IV instrument boards. Operator fails to perform cooldown with MFW and fails to recover from auto swapover failure.
		1.27E-04	INVFR2INV_2353-F	Inverter 2-III Fails During Operation	
		4.63E-02	INVFR2INV_2354-G_IE	INVERTER 2-IV FAILS DURING OPERATION	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		4.80E-03	HRADEP-POST-278		
95	2.51E-08	8.12E-03	%0LOSP-PC	Loss of Offsite Power (Plant Centered)	Plant Centered LOSP event with DG 2B-B failing due to a board room exhaust fan failing to start and DG 2A-2A unavailable due to maintenance. TDAFW is running until battery depletion. Operator fails to control SG level manual after battery depletion and recovery of LOSP fails.
		1.00E+00	FL-BATDEP	Battery Depleted FLAG	
		9.13E-03	FNSFD2FAN_030462	BOARD ROOM EXHAUST FAN FAILS TO START OR RUN FIRST HOUR	
		2.00E-01	HAOSBF	Steam generator feed with manual level control fails	
		1.51E-02	MTM_2GEN_0822A-A	DIESEL 2A-A MAINTENANCE	
		1.12E-01	XSBO01	Recovery Sequence 1 (DG A and B FAILS TO START) PC	
96	2.50E-08	1.01E-02	%0LOSP-GR	Loss of Offsite Power (Grid Related)	Grid related LOSP with a common cause failure of DG 1A-A and 2B-B failing to run. MDAFW pump 2A-A fails due to a pre-initiator isolation flow. TDAFW is running until battery
		1.00E+00	FL-BATDEP	Battery Depleted FLAG	
		3.59E-04	U0_EPS_GA_GEN_FR_CCF_1_4	CCF of two components: DGGFR1GEN_0821A-A & DGGFR2GEN_0822B-B	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
		6.90E-03	WHEMDA_1	Motor Driven AFW Pump Train A Isolation Test Error	depletion. No recovery is taken for this cutset. ECCS is unavailable due to a loss of ERCW. Since only 1 ERCW pump can be run off of one DG both A train and B train does not meet the 2 out of 4 pumps per train running for success.
97	2.50E-08	1.01E-02	%0LOSP-GR	Loss of Offsite Power (Grid Related)	Grid related LOSP with a common cause failure of DG 1B-B and 2A-A failing to run. MDAFW pump 2B-B fails due to a pre-initiator isolation flow. TDAFW is running until battery depletion. No recovery is taken for this cutset. ECCS is unavailable due to a loss of ERCW. Since only 1 ERCW pump can be run off of one DG both A train and B train does not meet the 2 out of 4 pumps per train running for success.
		1.00E+00	FL-BATDEP	Battery Depleted FLAG	
		3.59E-04	U0_EPS_GA_GEN_FR_CCF_2_3	CCF of two components: DGGFR1GEN_0821B-B & DGGFR2GEN_0822A-A	
		6.90E-03	WHEMDA_2	Motor Driven AFW Pump Train B Isolation Test Error	
98	2.43E-08	8.12E-03	%0LOSP-PC	Loss of Offsite Power (Plant Centered)	Plant centered LOSP event with DG 2A-A failing to run and DG 2B-2B unavailable due to maintenance. TDAFW is running until battery depletion. Operator fails to control SG level manually after battery depletion and recovery of LOSP fails.
		1.46E-02	DGGFR2GEN_0822A-A	DIESEL GENERATOR FAILS TO RUN AFTER FIRST HOUR	
		1.00E+00	FL-BATDEP	Battery Depleted FLAG	
		2.00E-01	HAOSBF	Steam generator feed with manual level control fails	
		1.07E-02	MTM_2GEN_0822B-B	DIESEL 2B-B MAINTENANCE	
		9.56E-02	XSBO04	Recovery Sequence 5 (One EDG Fails to Start and One Fails to Run) PC	

Top 100 CDF Cutsets					
#	**Cutset Prob.**	**Event Prob.**	**Event**	**Event Description**	**Cutset Description**
99	2.41E-08	1.00E+00	%2LVBB3	Loss of Battery Board 3	Loss of battery board 3 IE due to a loss of the III battery to operate. Loss of the battery board 3 fails MDAFW Pump 2A-A. MDAFW Pump 2B-B is unavailable due to maintenance and TDAFW fails to start. Operator fails to establish Bleed and Feed cooling.
		1.63E-02	BATFR0BAT_2363-F_IE	BATT III FAILS DURING OPERATION (0-BAT-236-3-F)	
		1.60E-02	HAOB2	Establish RCS Bleed and Feed cooling given no CCPS running	
		4.22E-03	MTM_2PMP00300128	PUMP WBN-2-3-128-B IN MAINTENANCE	
		9.03E-01	PAF	PLANT AVAILABILITY FACTOR	
		2.43E-02	PTSF12PMP_003001AS	PUMP FAILS TO START AND RUN FOR 1 HOUR WBN-1-3-1AS	
100	2.40E-08	8.58E-03	%0FLTBMF	Major flood in the Turbine Building	A major flood event is initiated in the Turbine Building due to a break of the condenser expansion joints; all equipment underneath EL 711 of the Turbine Building is expected to be lost. Flood is isolated before propagation to the Control Building. All the control air compressors fail due to the major flood in the turbine building. ACAS compressors A-A and B-B fails to run (common cause) resulting in a total loss of air event. Operator fails to restore AFW control following loss of air. Operator also fails to establish RCS Bleed and Feed cooling.
		3.25E-03	U0_032_ACAS_CMP_FR_CCF_1_2	CCF of two components: CMPSR0COMP03200060 & CMPSR0COMP03200086	
		8.60E-04	HRADEP-POST-221		

Appendix C - Simplified Drawings

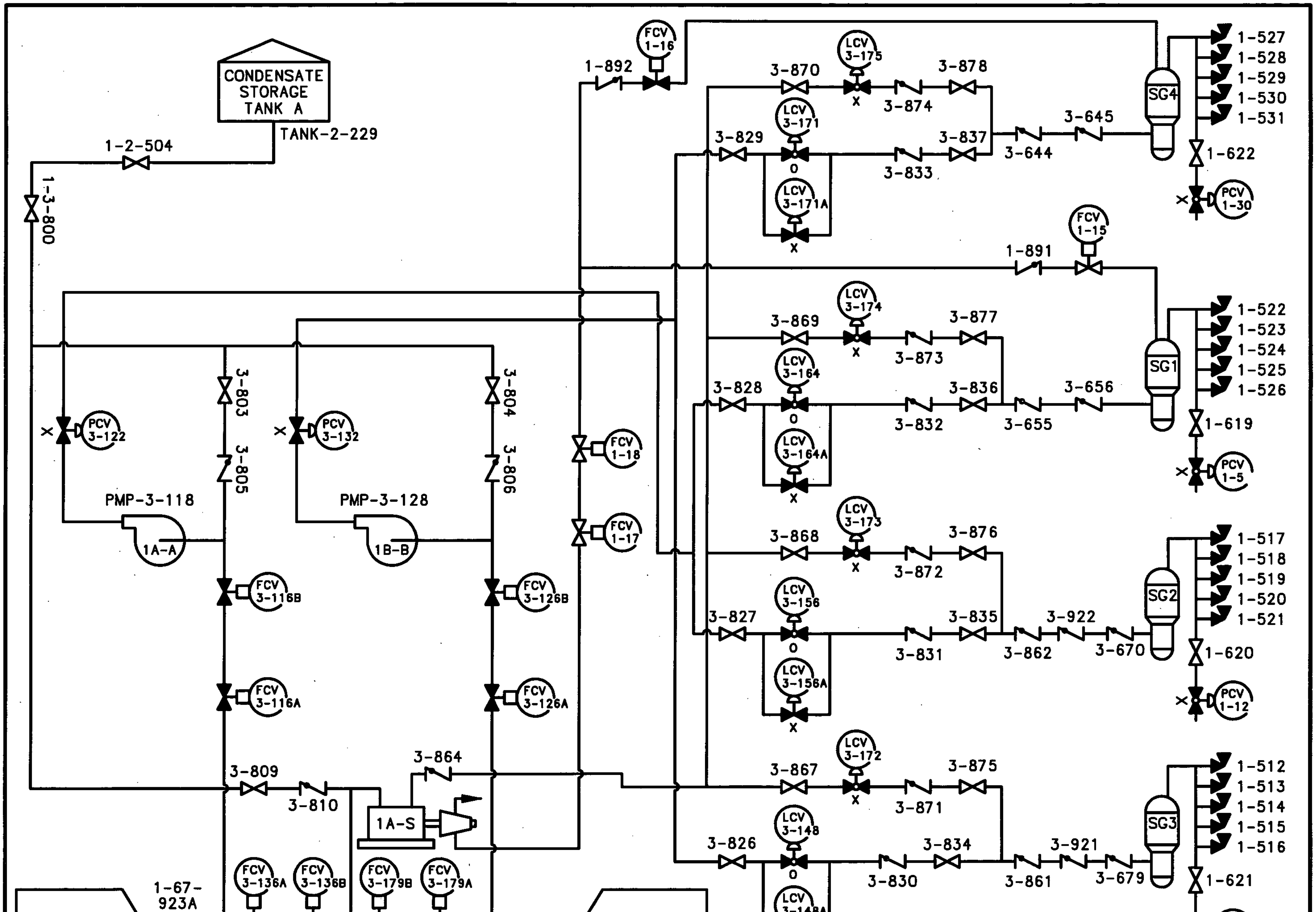


FIGURE 1
AUXILIARY FEEDWATER SYSTEM

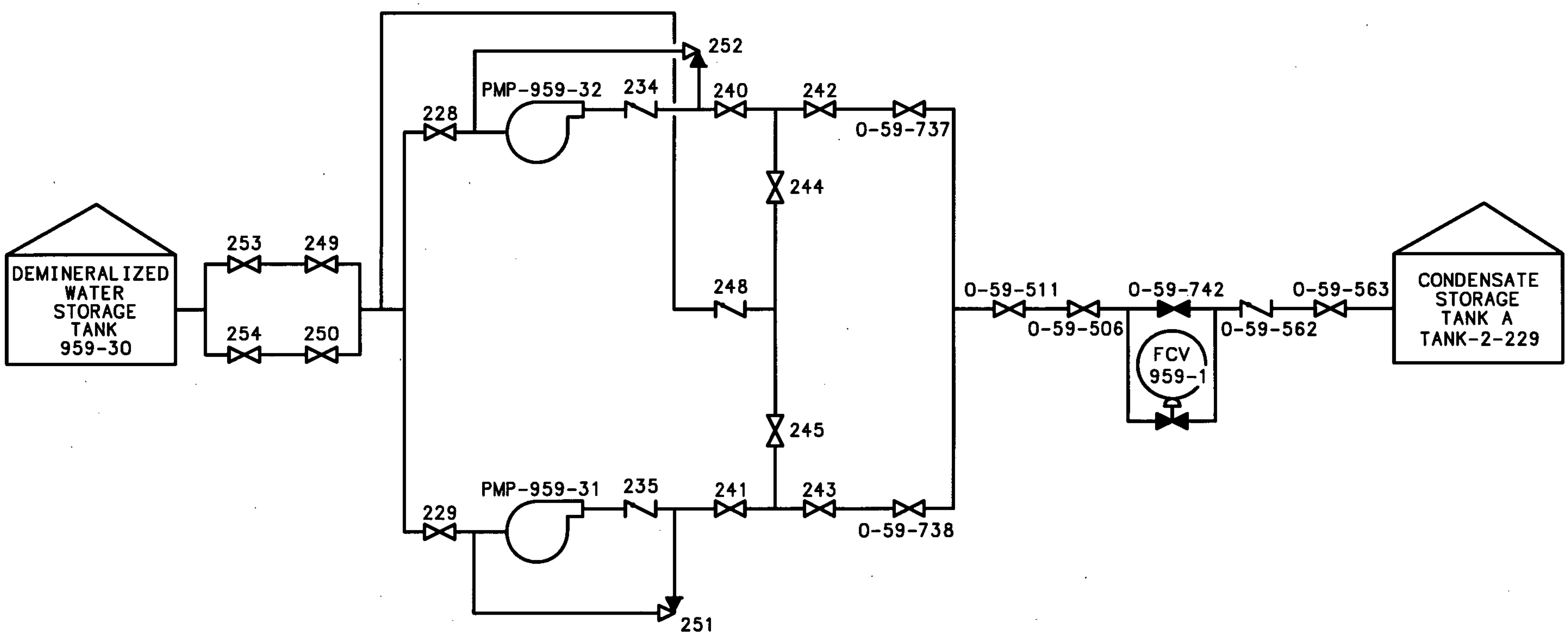


FIGURE 2
CONDENSATE STORAGE MAKEUP SYSTEM

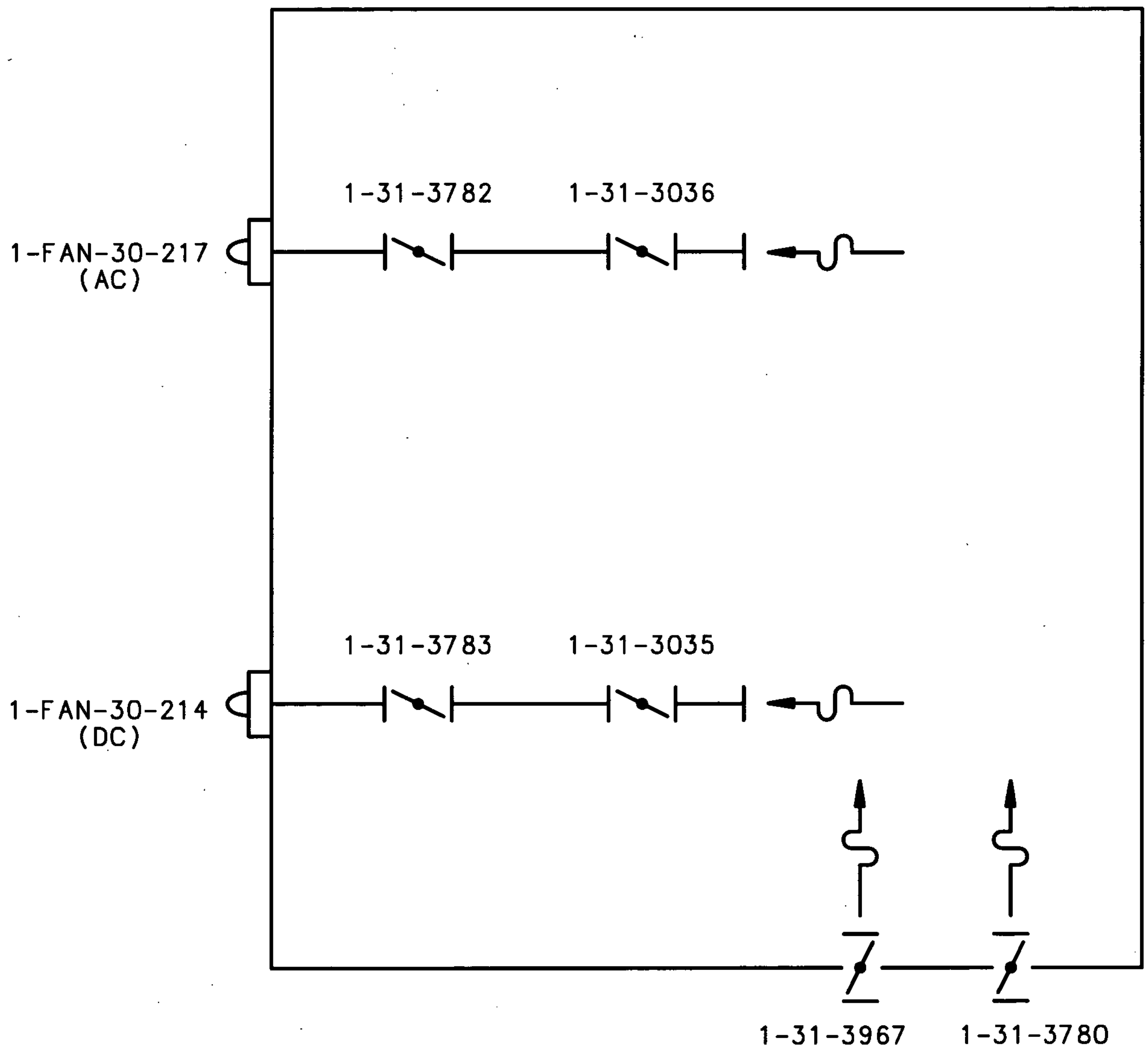


FIGURE 3
TURBINE DRIVEN PUMP FANS & DAMPERS

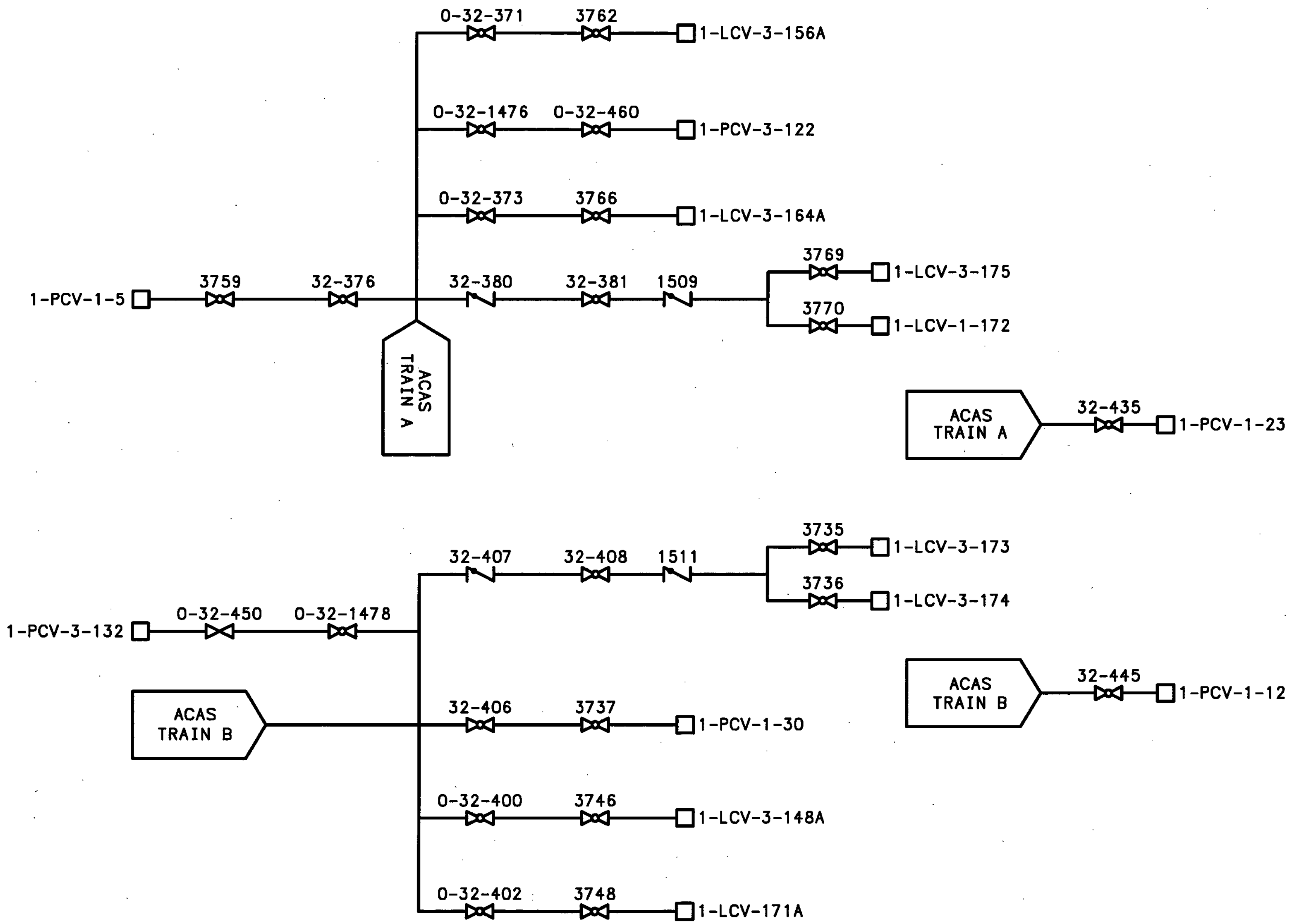


FIGURE 4
AIR SUPPLY TO AOV'S

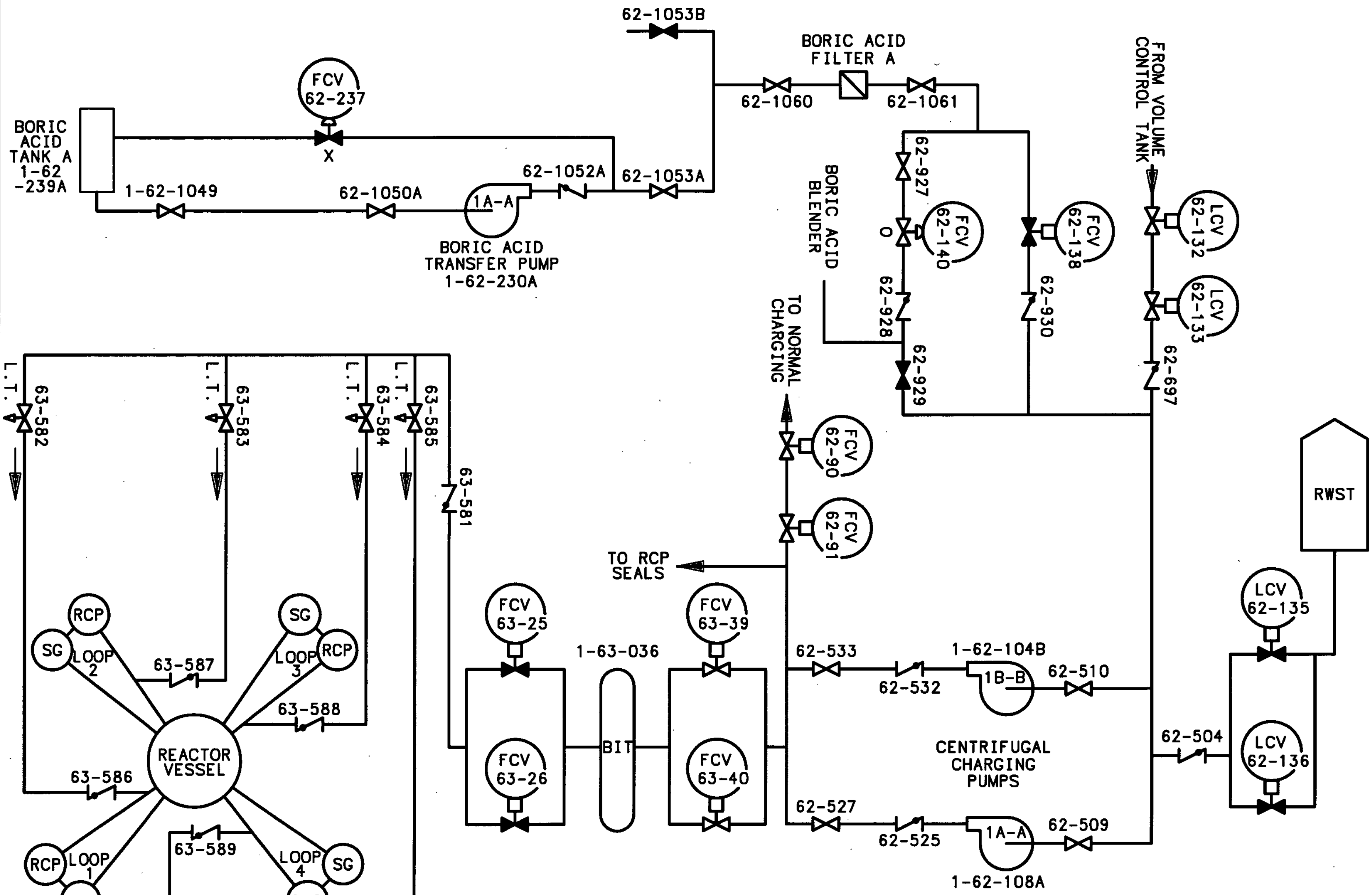


FIGURE 1
CHEMICAL AND VOLUME CONTROL SYSTEM
(NORMAL ALIGNMENT)

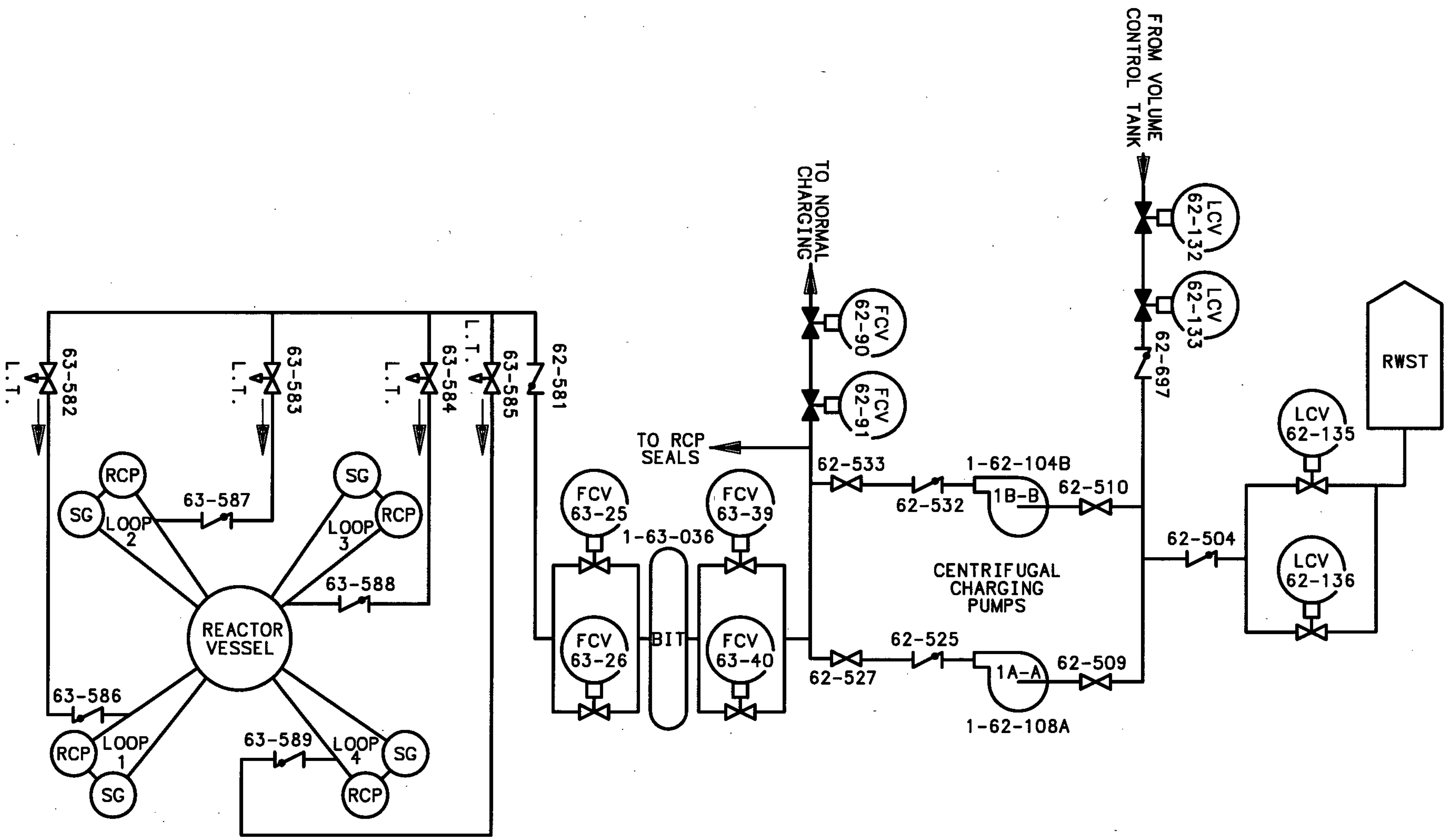


FIGURE 2
CHEMICAL AND VOLUME CONTROL SYSTEM
(COLD LEG INJECTION ALIGNMENT)

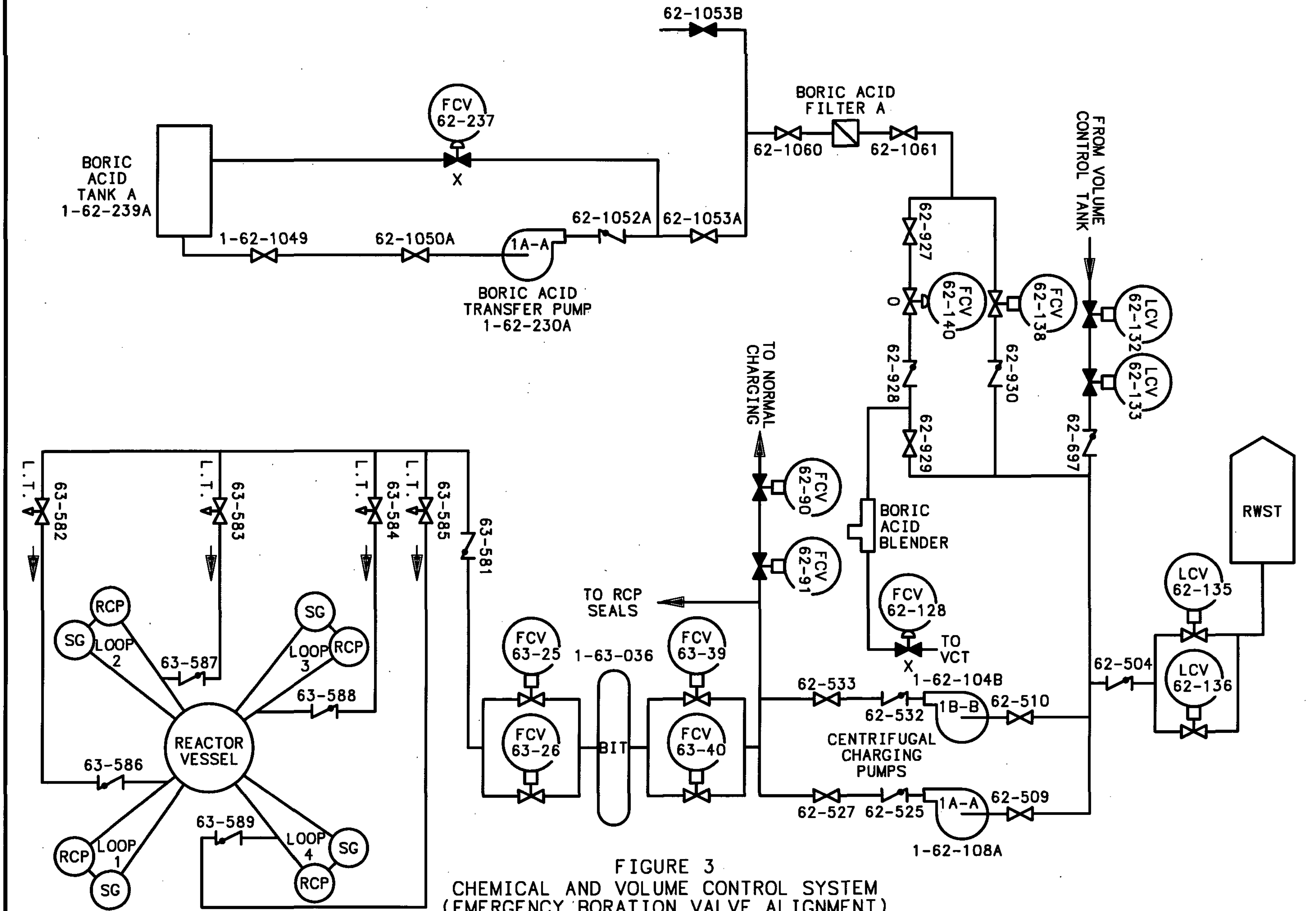


FIGURE 3
CHEMICAL AND VOLUME CONTROL SYSTEM
(EMERGENCY BORATION VALVE ALIGNMENT)

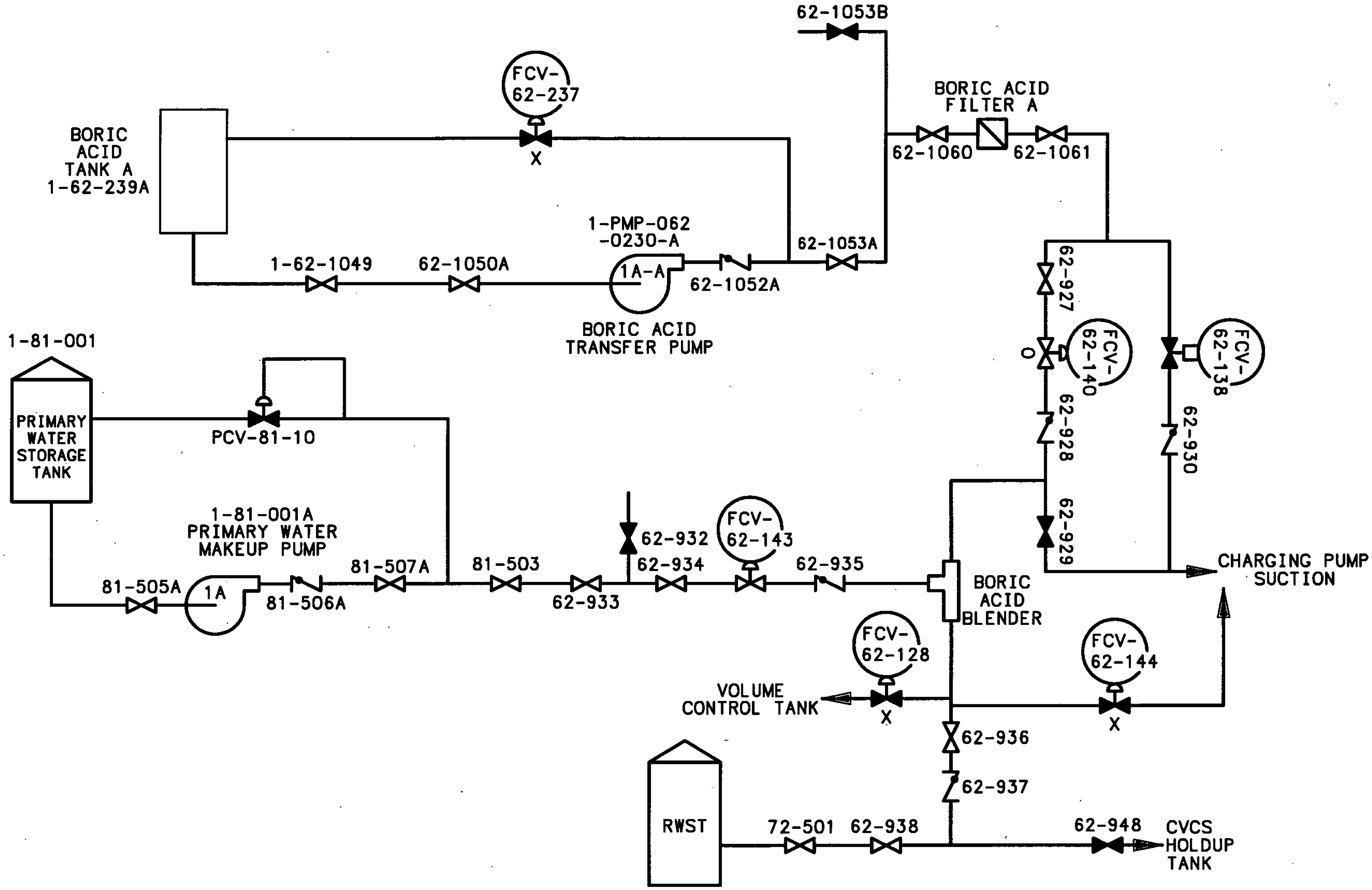


FIGURE 4A
CHEMICAL AND VOLUME CONTROL SYSTEM
(MAKEUP TO RWST FOLLOWING SGTR)

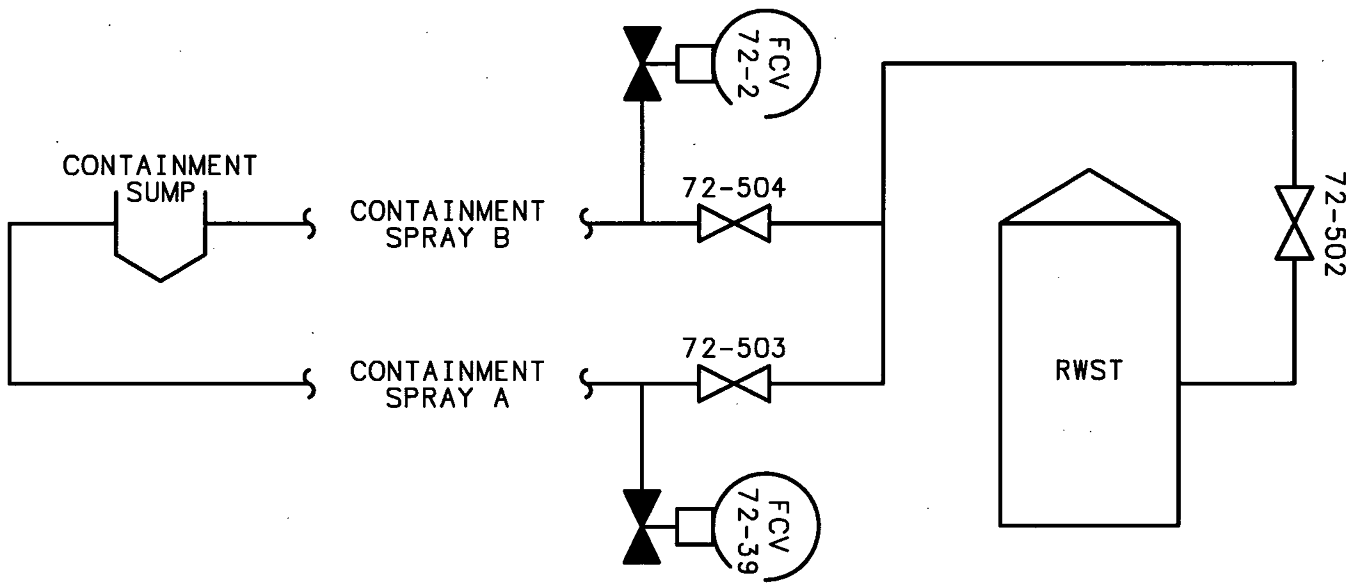


FIGURE 4B
CHEMICAL AND VOLUME CONTROL SYSTEM
(MAKEUP TO RWST FOLLOWING SLOCA)

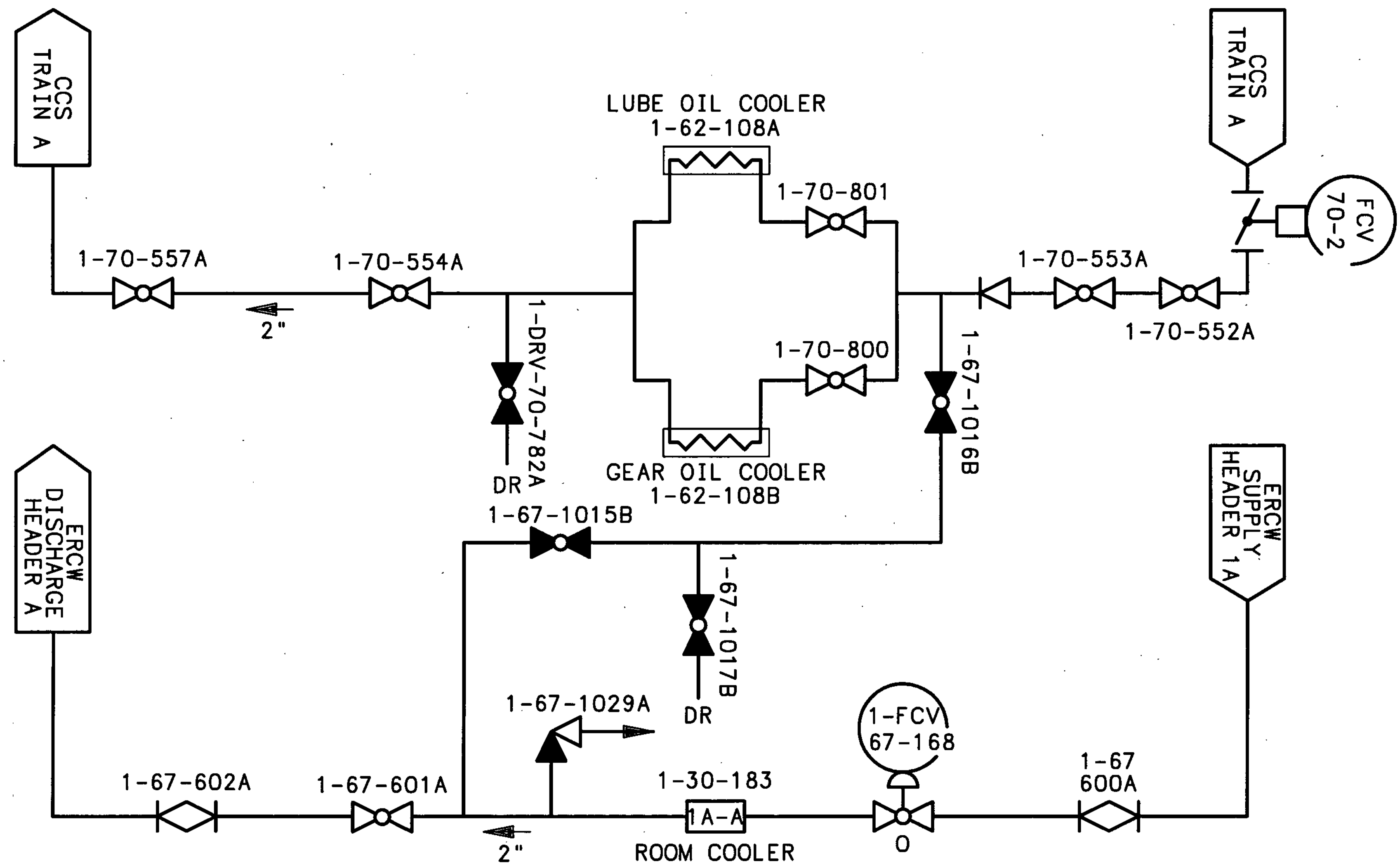


FIGURE 5
CHEMICAL AND VOLUME CONTROL SYSTEM
(CCP 1A-A LUBE AND GEAR OIL AND ROOM COOLERS)

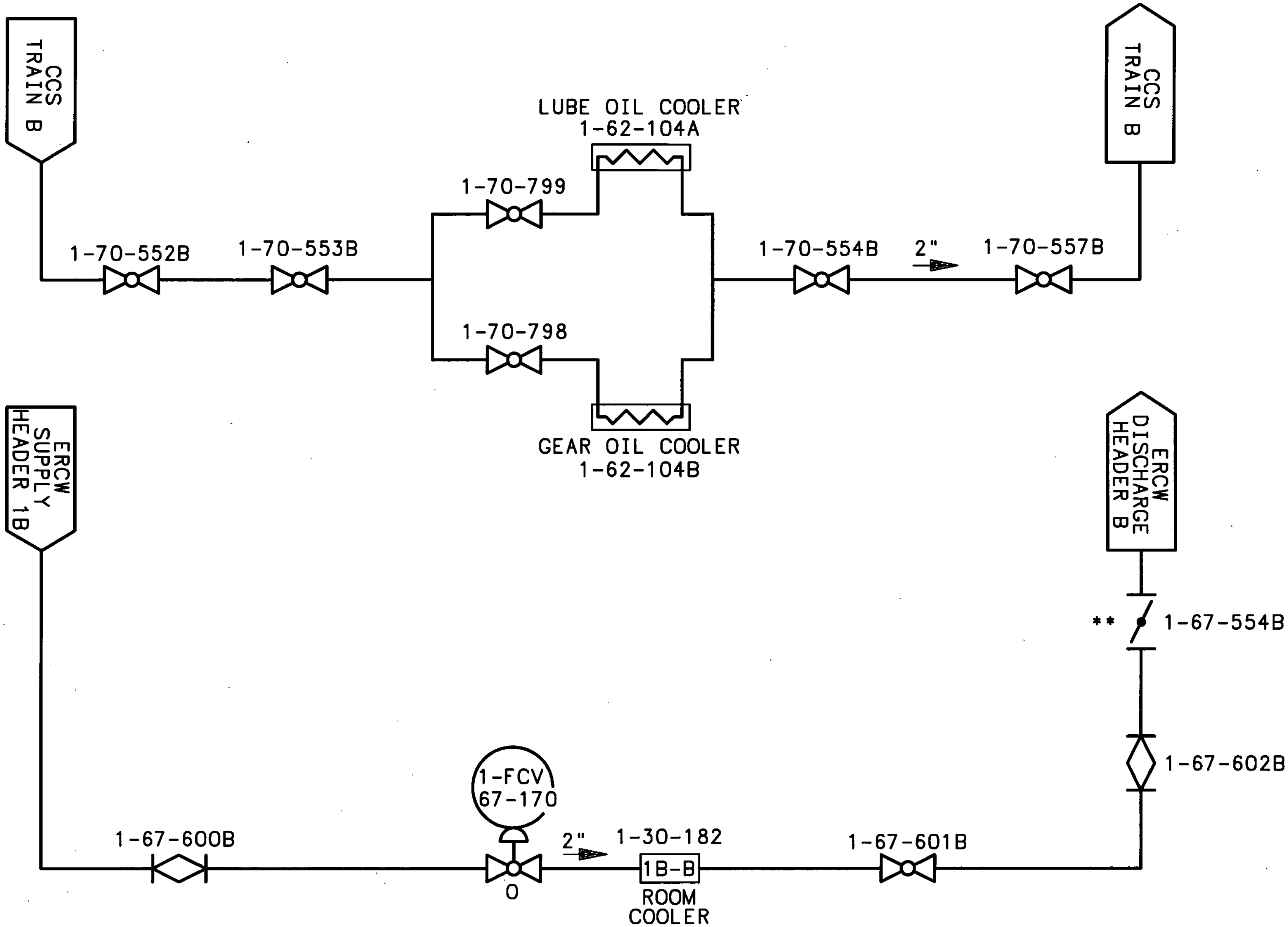


FIGURE 6
CHEMICAL AND VOLUME CONTROL SYSTEM
(CCP 1B-B LUBE AND GEAR OIL AND ROOM COOLERS)

** COMMON TO CVCS, SI, RHR, AND CS

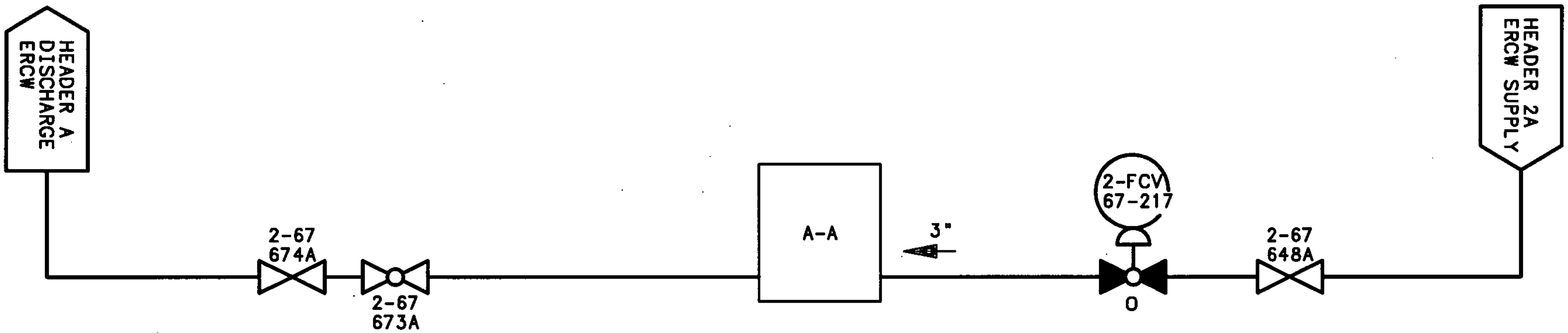


FIGURE 7
CHEMICAL AND VOLUME CONTROL SYSTEM
BORIC ACID TRANSFER PUMP & AFW PUMP SPACE COOLING (NORMAL ALIGNMENT)

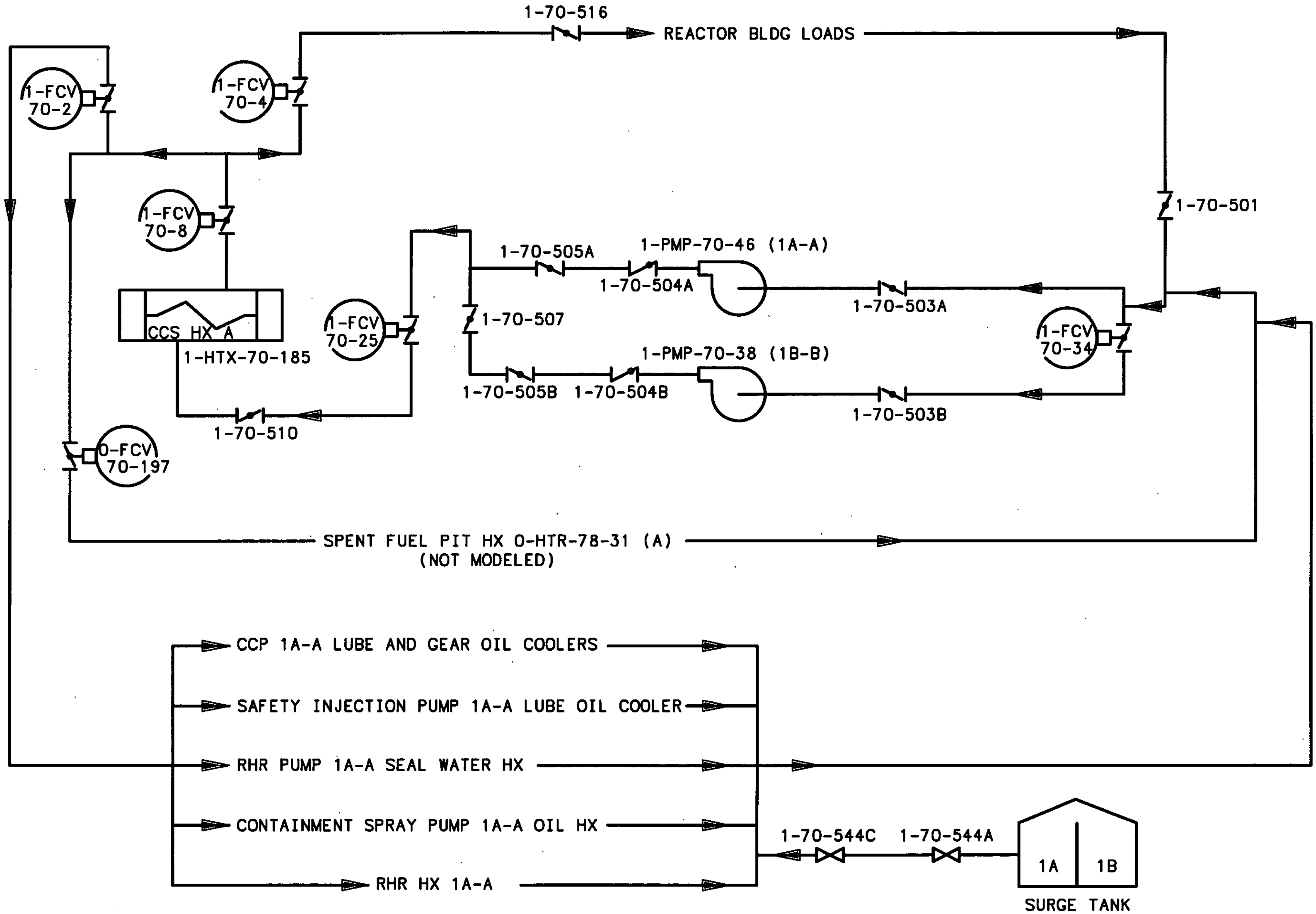


FIGURE 1
COMPONENT COOLING SYSTEM
TRAIN 1A

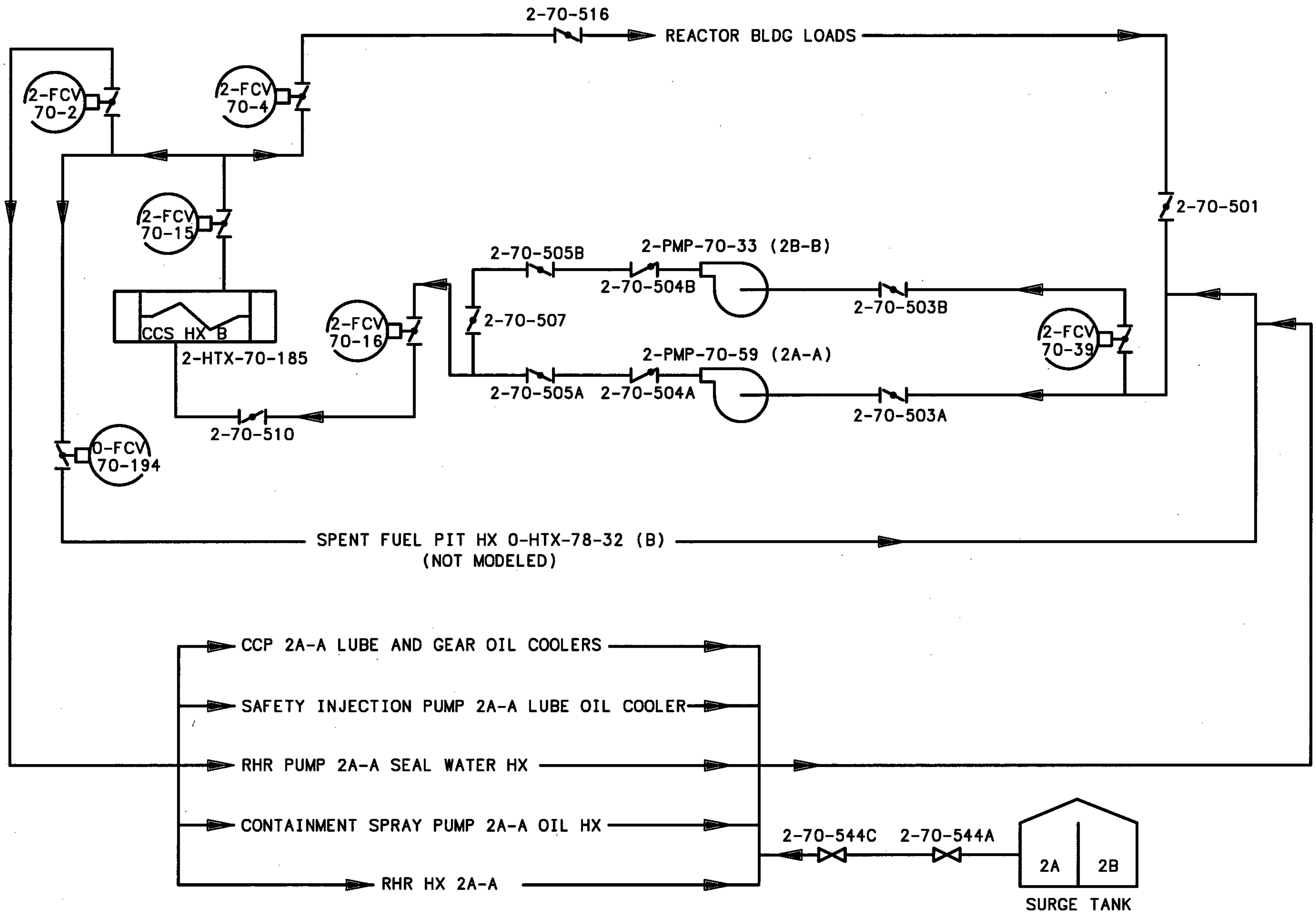


FIGURE 2
COMPONENT COOLING SYSTEM
TRAIN 2A

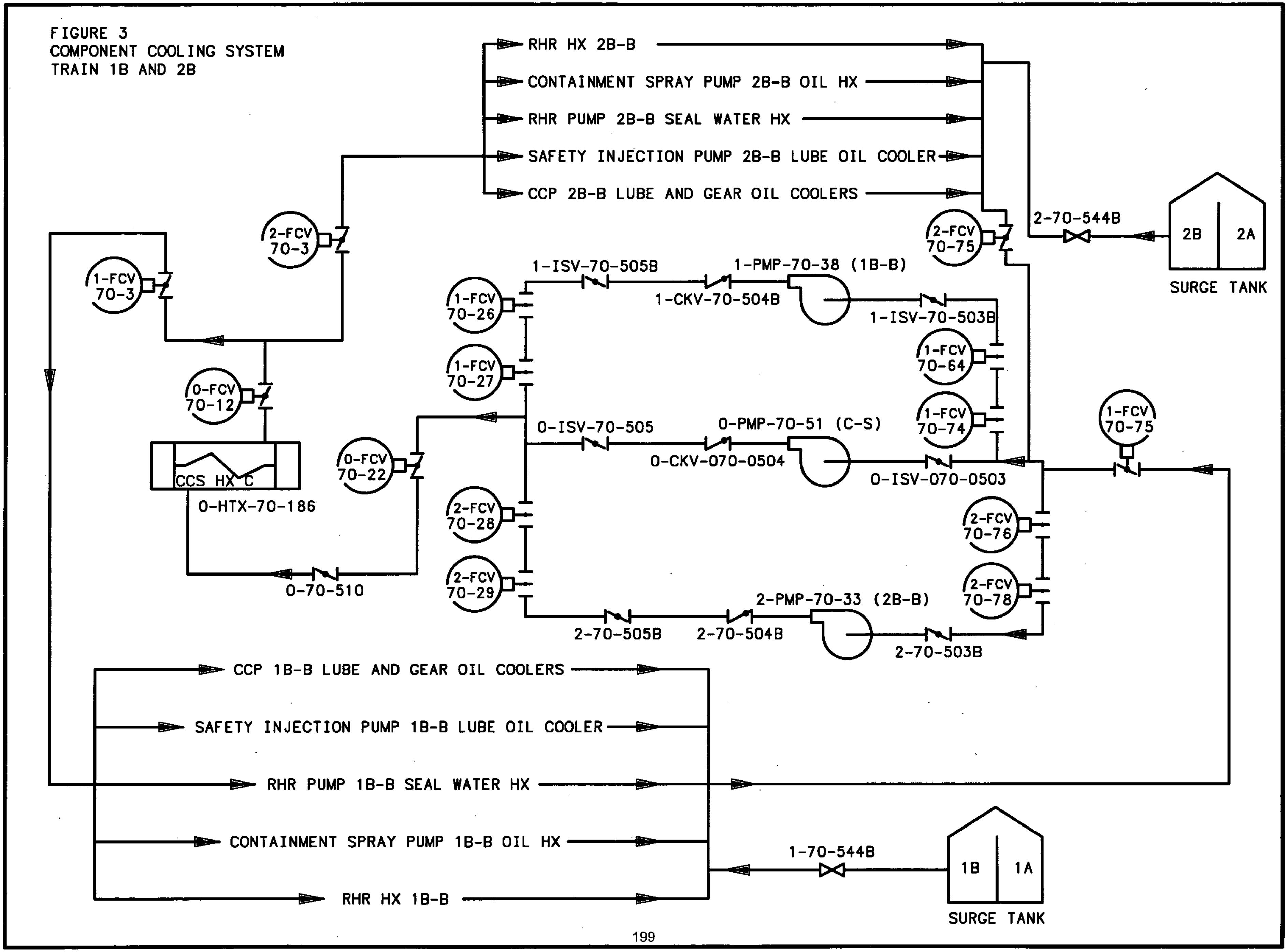


FIGURE 3
COMPONENT COOLING SYSTEM
TRAIN 1B AND 2B
RHR HX 2B-B
CONTAINMENT SPRAY PUMP 2B-B OIL HX
RHR PUMP 2B-B SEAL WATER HX
SAFETY INJECTION PUMP 2B-B LUBE OIL COOLER
CCP 2B-B LUBE AND GEAR OIL COOLERS
2-FCV 70-3
1-FCV 70-3
2-FCV 70-75
2-70-544B
2B
2A
SURGE TANK
1-ISV-70-505B
1-PMP-70-38 (1B-B)
1-FCV 70-26
1-CKV-70-504B
1-ISV-70-503B
1-FCV 70-64
1-FCV 70-27
0-FCV 70-12
1-FCV 70-74
1-FCV 70-75
0-ISV-70-505
0-PMP-70-51 (C-S)
0-CKV-070-0504
0-ISV-070-0503
0-FCV 70-22
CCS HX C
0-HTX-70-186
2-FCV 70-28
2-FCV 70-76
2-FCV 70-29
2-FCV 70-78
0-70-510
2-PMP-70-33 (2B-B)
2-70-505B
2-70-504B
2-70-503B
CCP 1B-B LUBE AND GEAR OIL COOLERS
SAFETY INJECTION PUMP 1B-B LUBE OIL COOLER
RHR PUMP 1B-B SEAL WATER HX
CONTAINMENT SPRAY PUMP 1B-B OIL HX
1-70-544B
RHR HX 1B-B
1B
1A
SURGE TANK

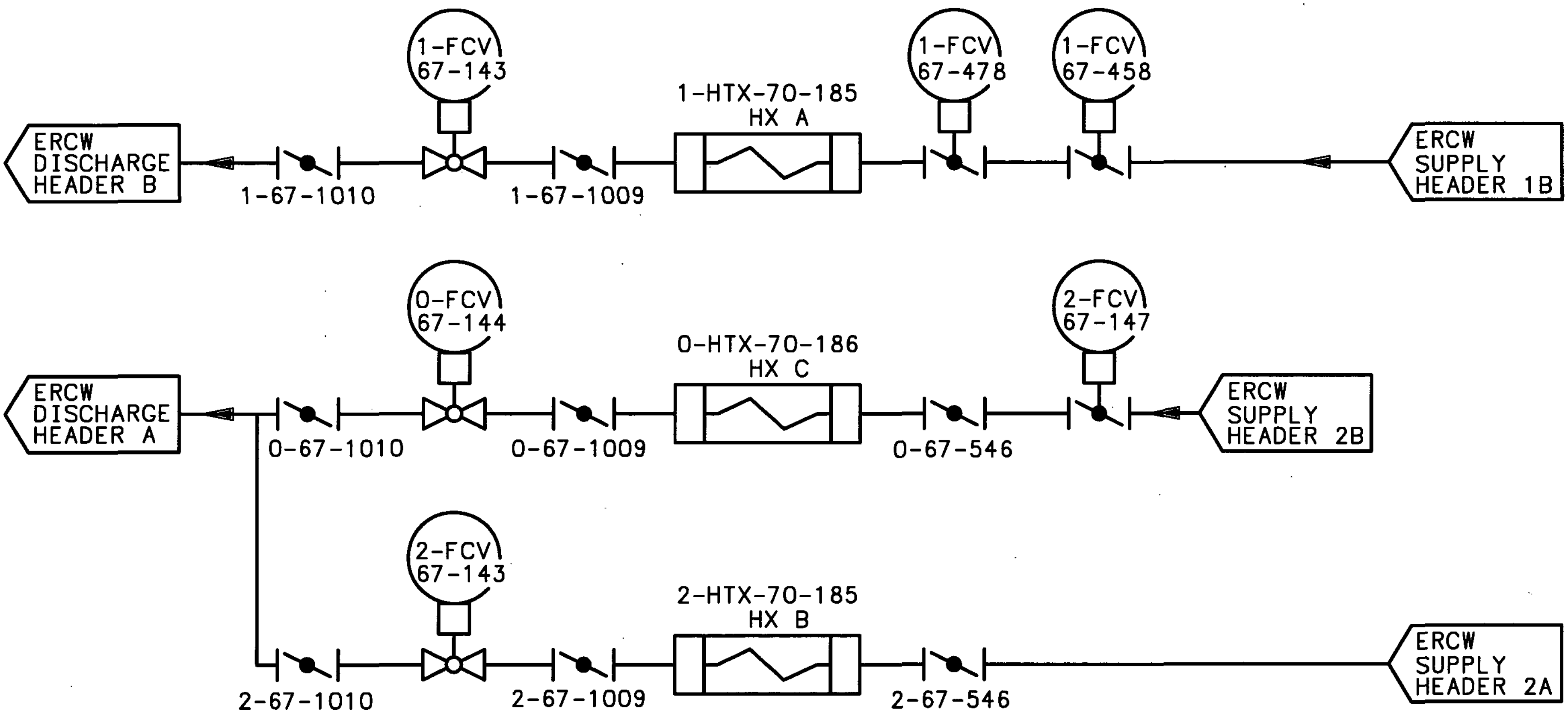


FIGURE 4
COMPONENT COOLING SYSTEM
ESSENTIAL RAW
COOLING WATER SUPPLY

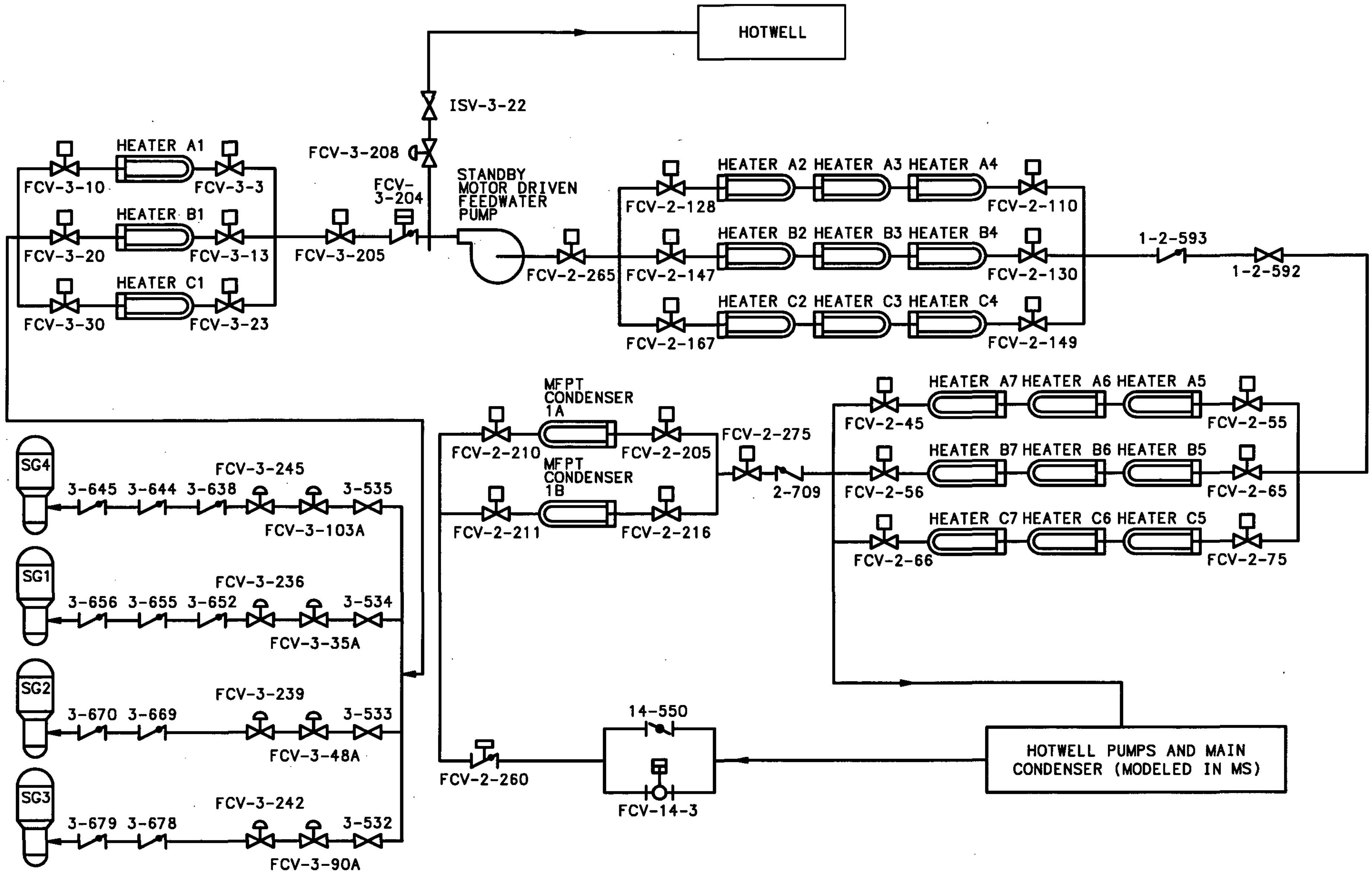
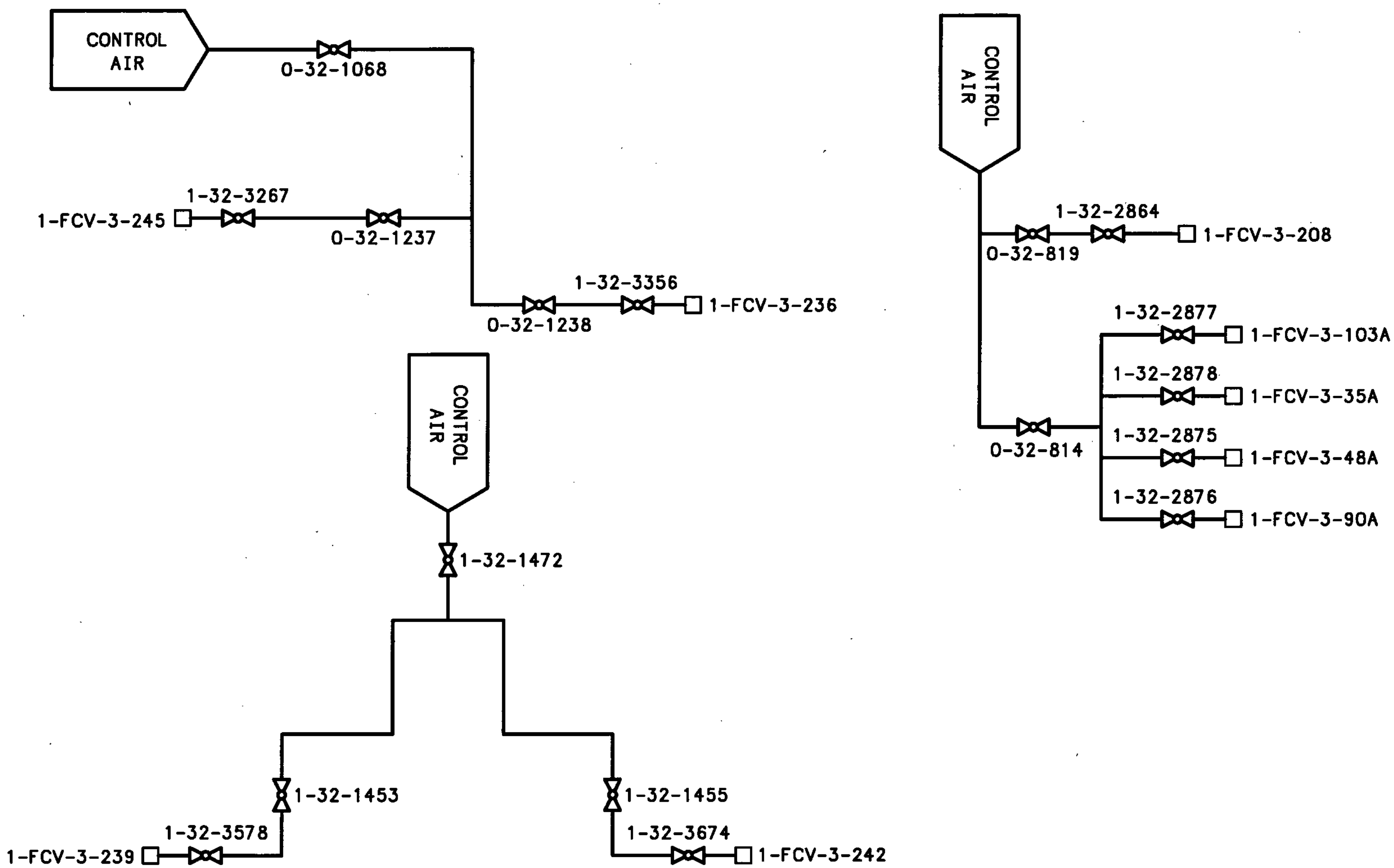
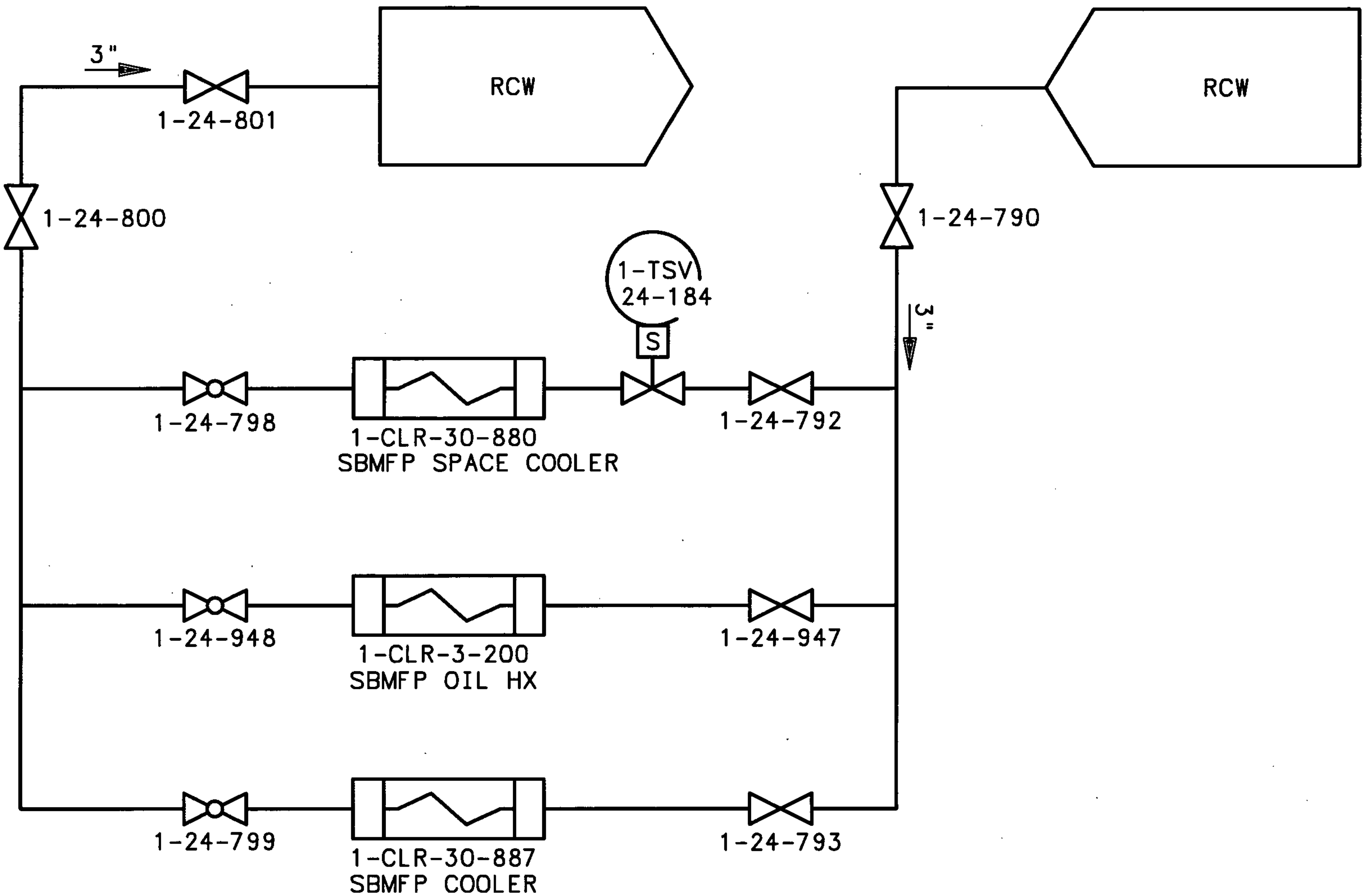


FIGURE 1
SIMPLIFIED DRAWING OF THE CONDENSATE AND FEEDWATER SYSTEM



REFERENCE DRAWINGS:
1-47W848-6
1-47W848-9
1-47W848-12

FIGURE 2
CONTROL AIR FOR MFW VALVES



REFERENCE DRAWING:
1-47W844-3

FIGURE 3
RAW COOLING WATER FOR SBMFP COOLING

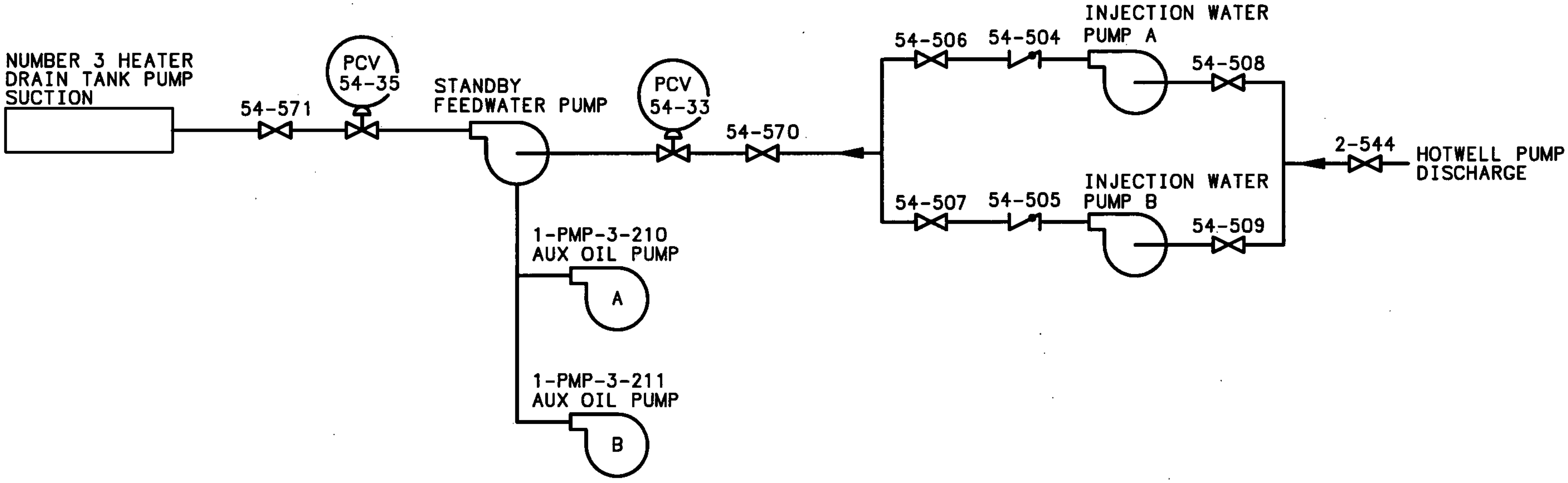


FIGURE 4
SBFWP SEAL INJECTION AND AUX OIL PUMPS

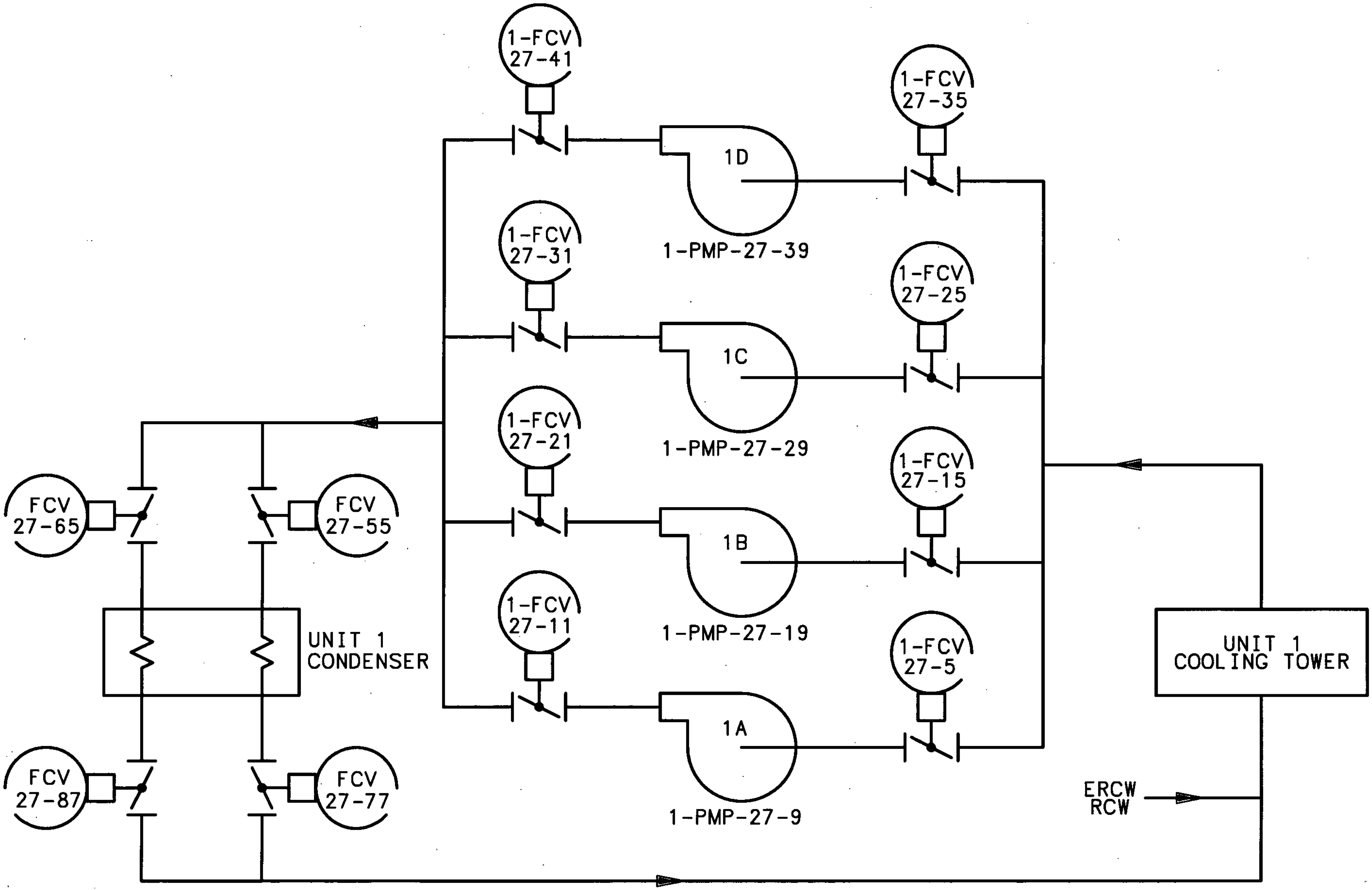


FIGURE 1
CONDENSER CIRCULATING WATER

REFERENCE DRAWING:
1-47W831-1

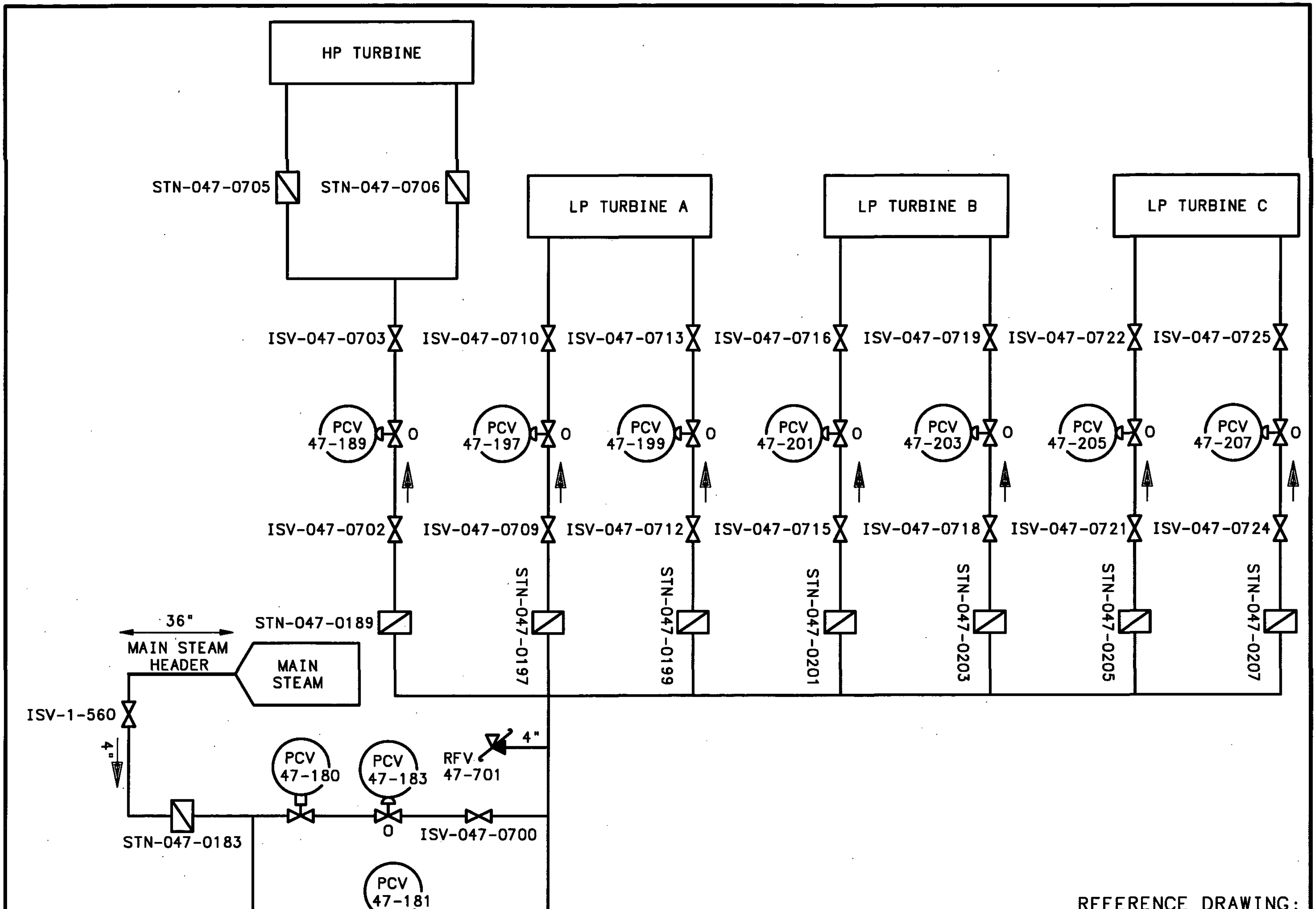


FIGURE 2
GLAND SEAL STEAM

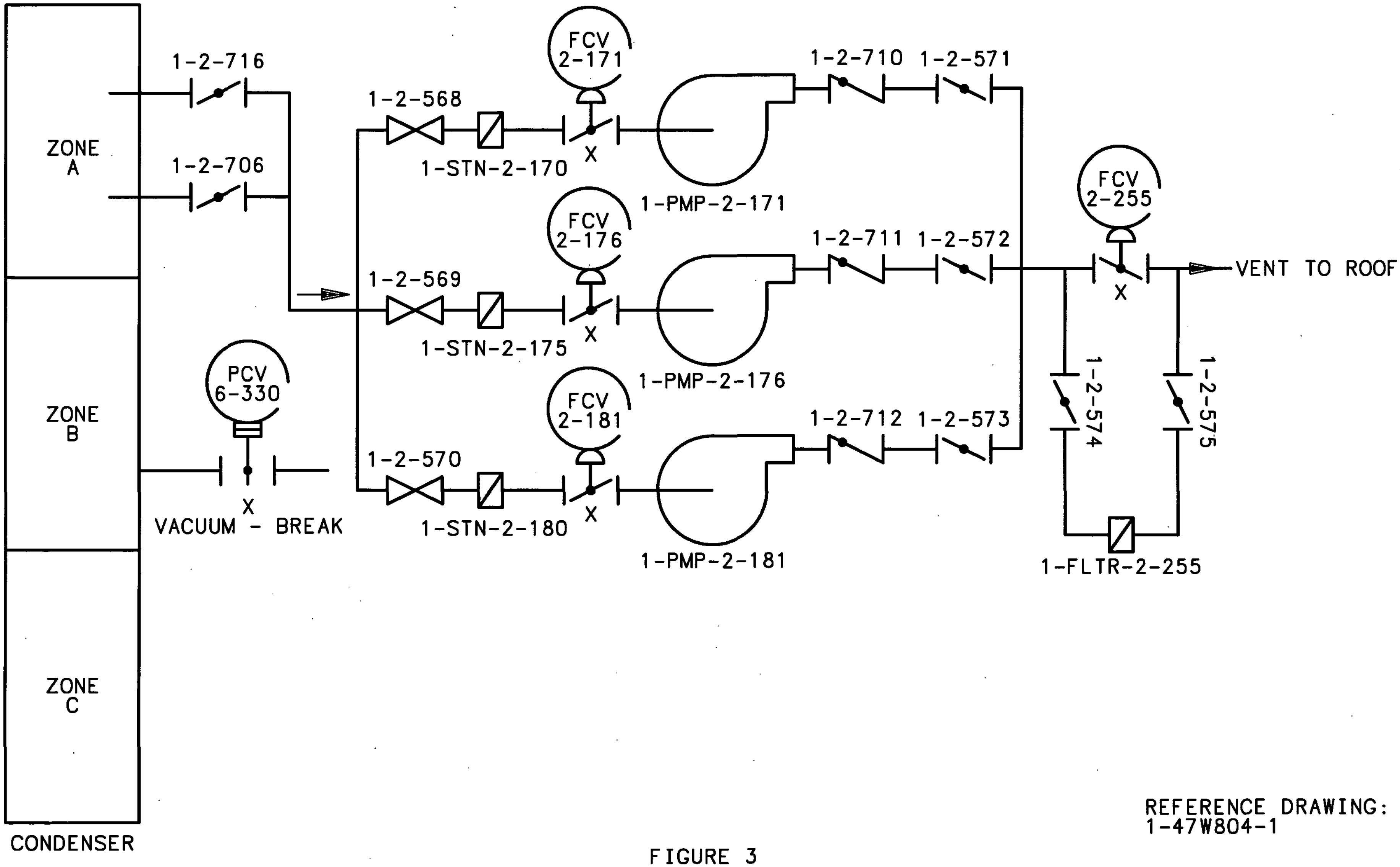


FIGURE 3
CONDENSER VACUUM PUMPS

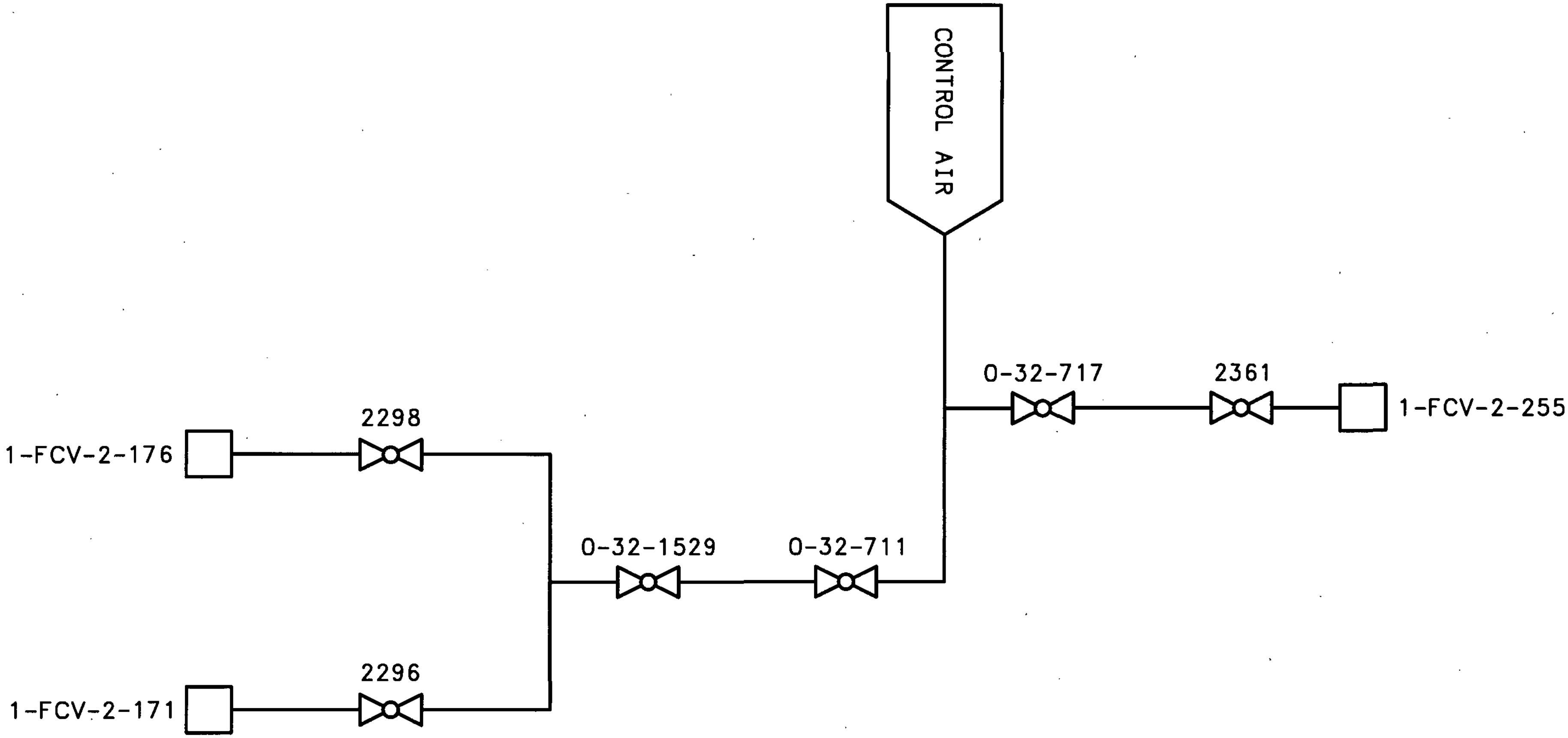
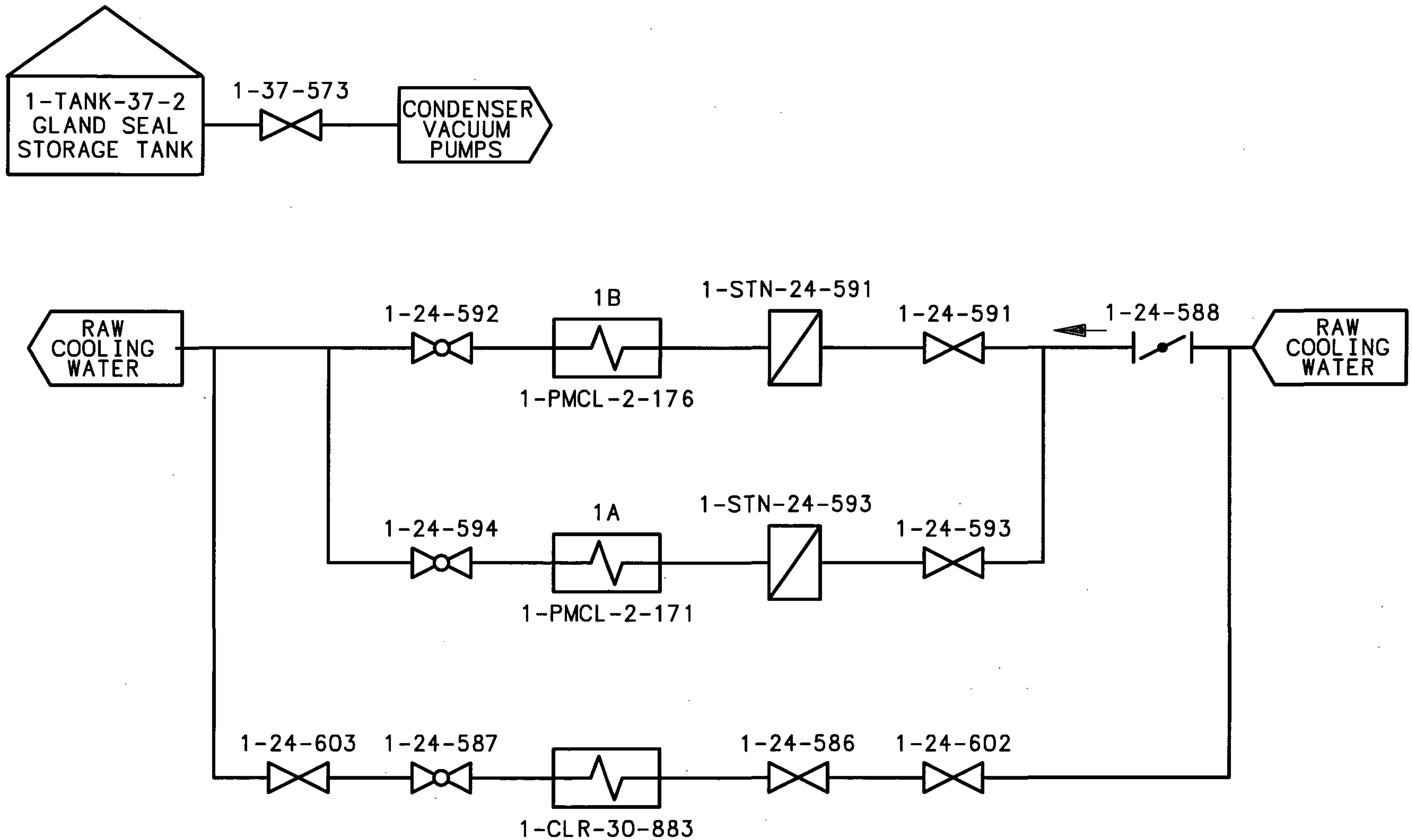


FIGURE 4
CONTROL AIR TO VACUUM PUMP SUCTION FCVs

REFERENCE DRAWING:
1-47W848-11



REFERENCE DRAWING:
1-47W841-1
1-47W844-2
1-47W844-3

FIGURE 5
CONDENSER VACUUM PUMP COOLING, GLAND SEAL WATER

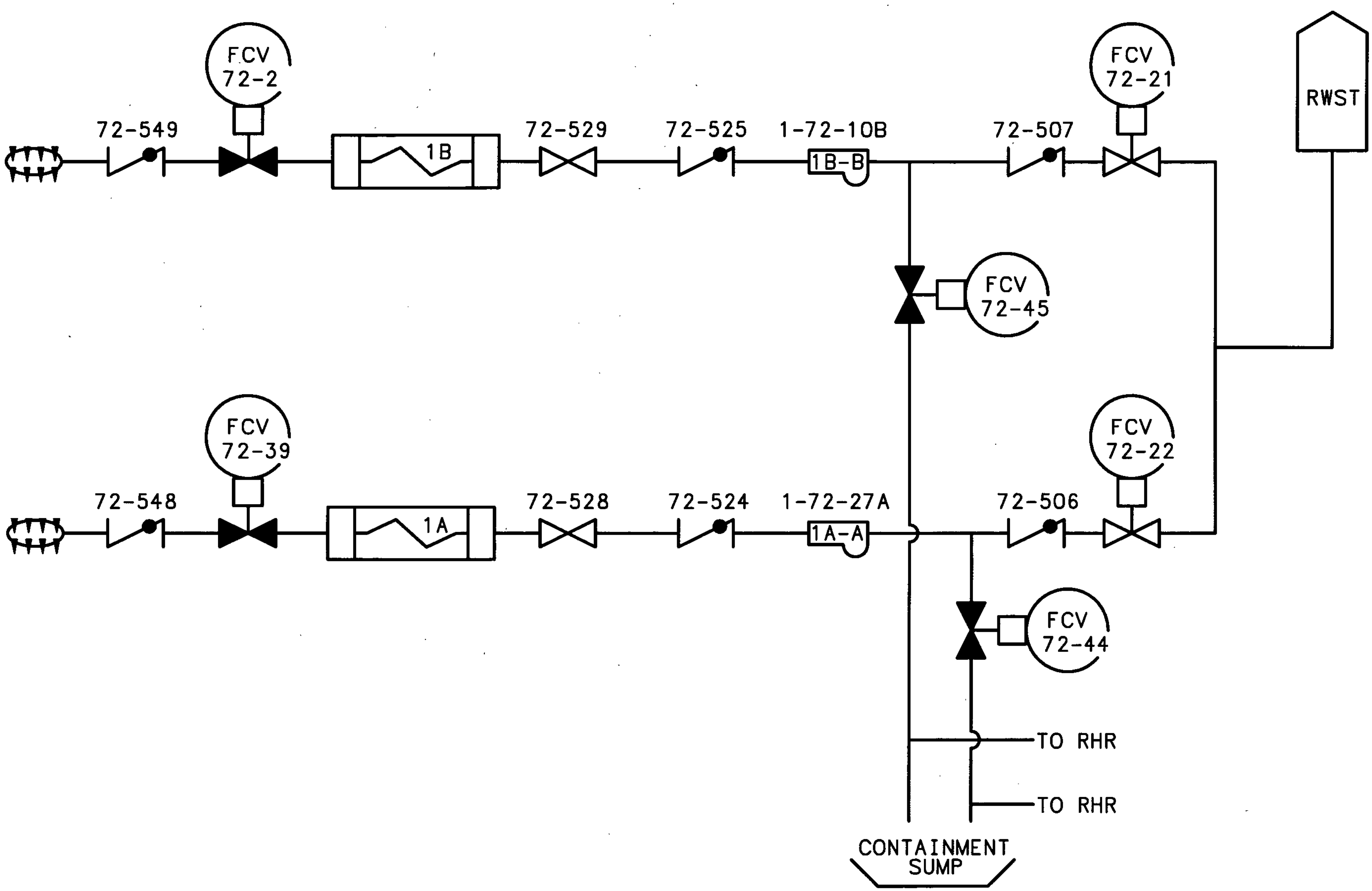
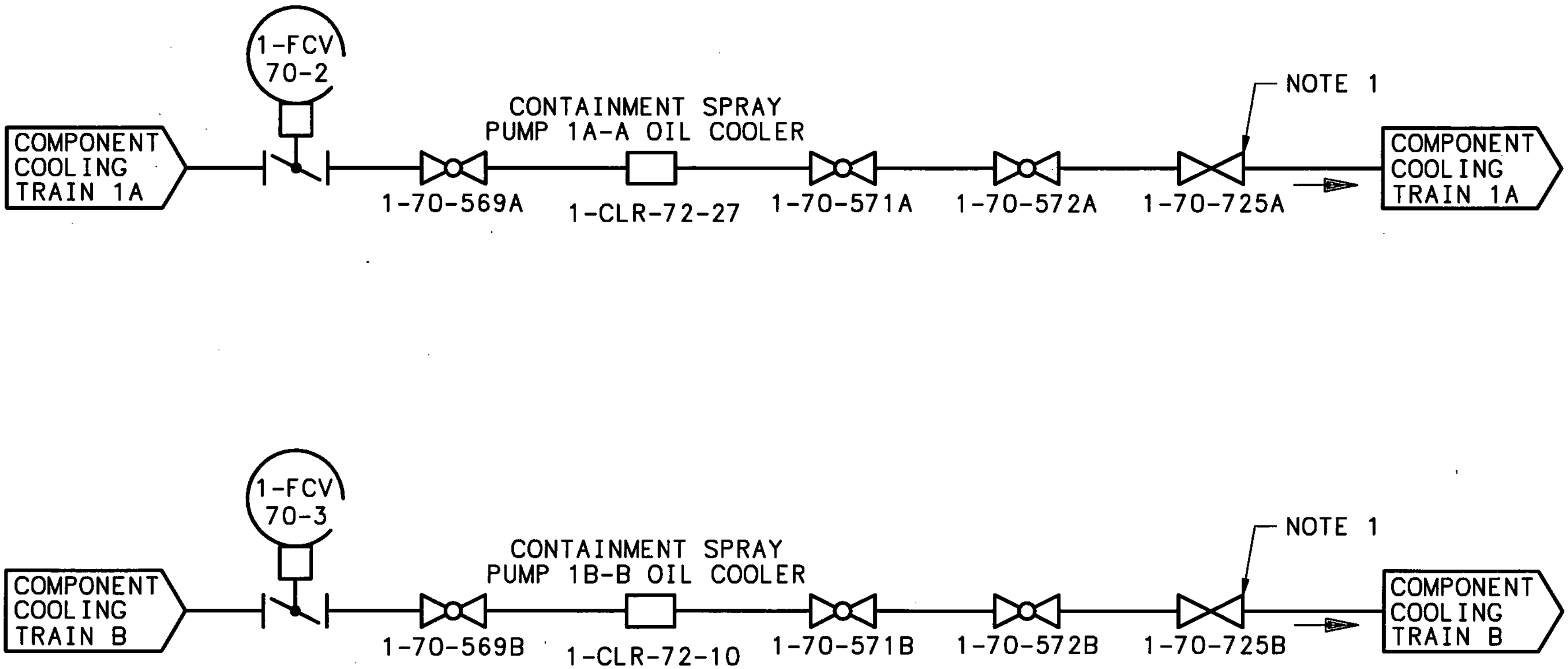


FIGURE 1
CONTAINMENT SPRAY SYSTEM

REFERENCE DRAWINGS:
47W845-2
47W845-4
47W812-1



NOTE 1: COMMON TO CS AND RHR

FIGURE 2
CONTAINMENT SPRAY SYSTEM
PUMP OIL COOLING

REFERENCE DRAWINGS:
1-47W859-1
1-47W859-4

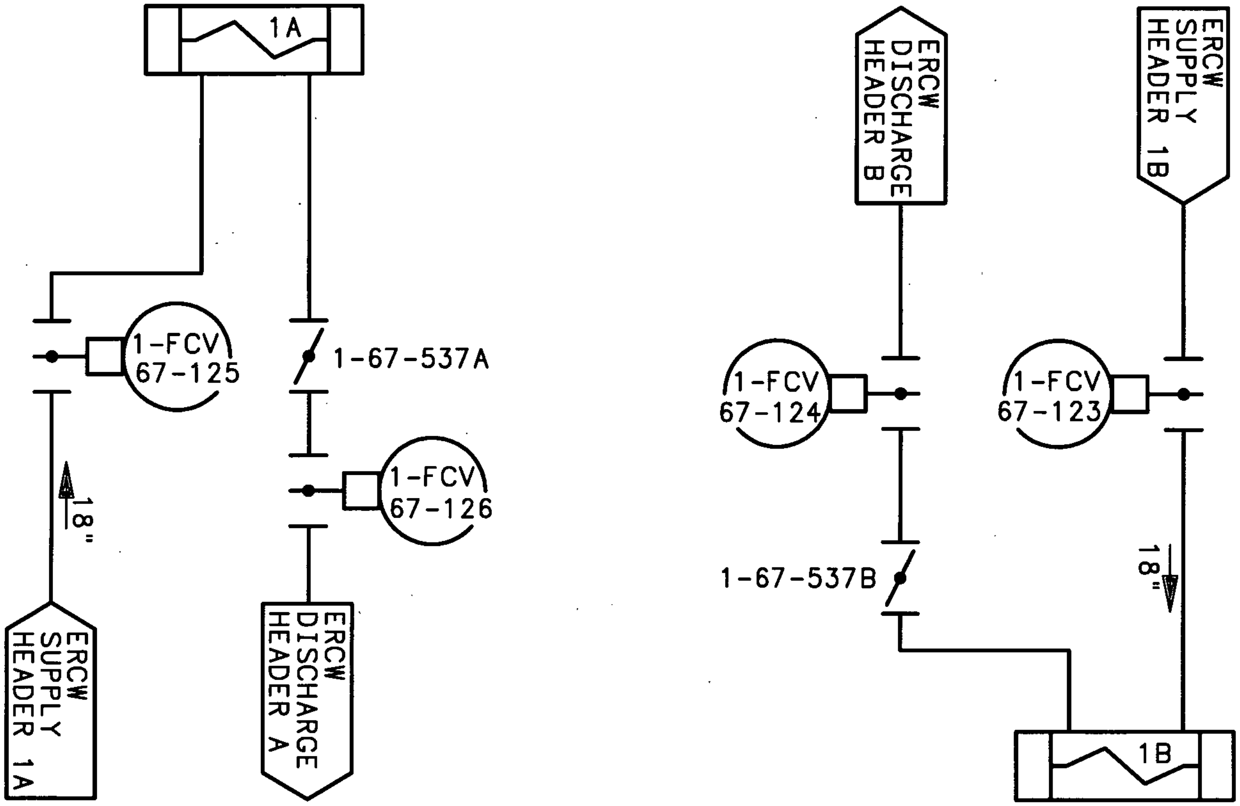
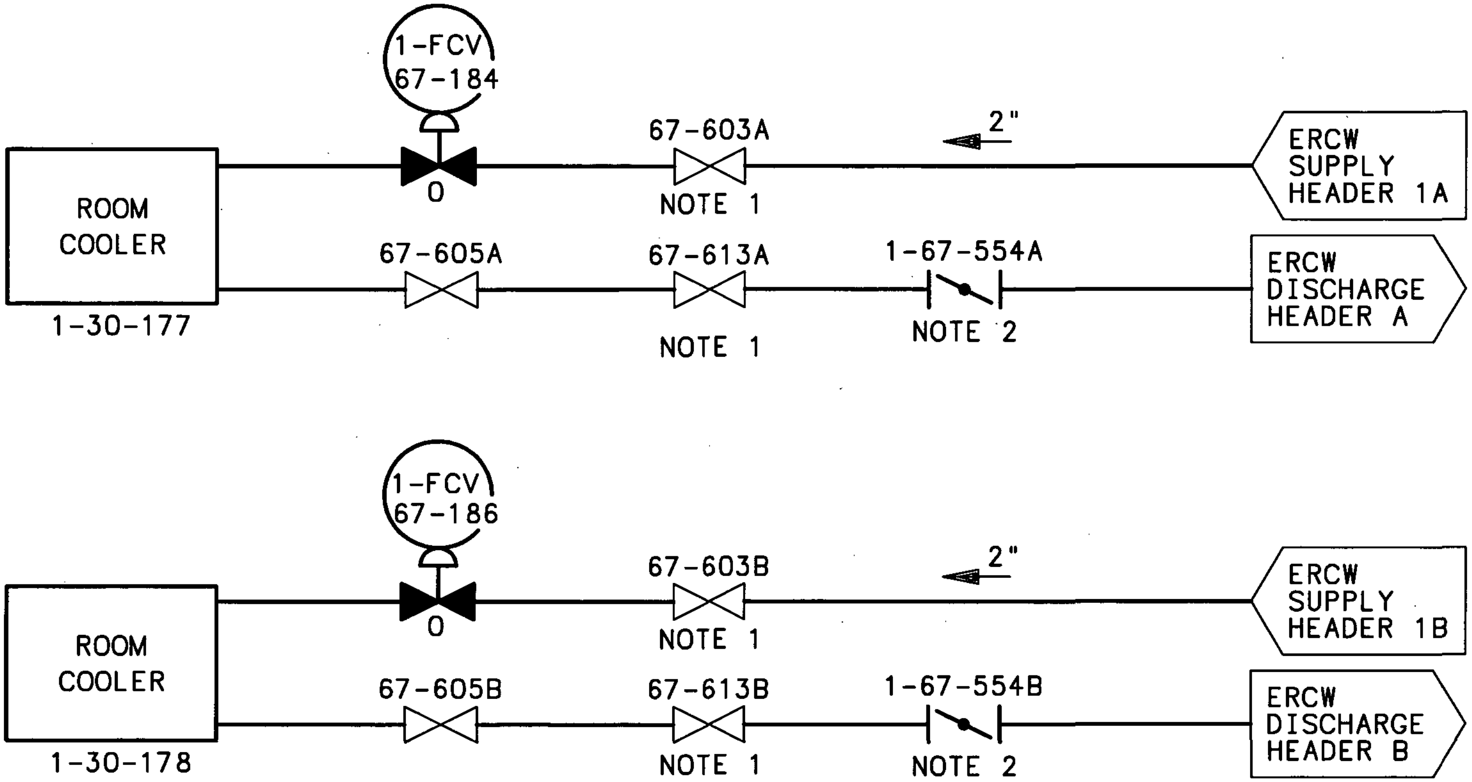


FIGURE 3
CONTAINMENT SPRAY SYSTEM
HEAT EXCHANGER COOLING



NOTE 1 COMMON TO CS, SI, AND RHR

NOTE 2 COMMON TO CVCS, SI, RHR, AND CS

FIGURE 4
CONTAINMENT SPRAY SYSTEM
PUMP ROOM COOLING

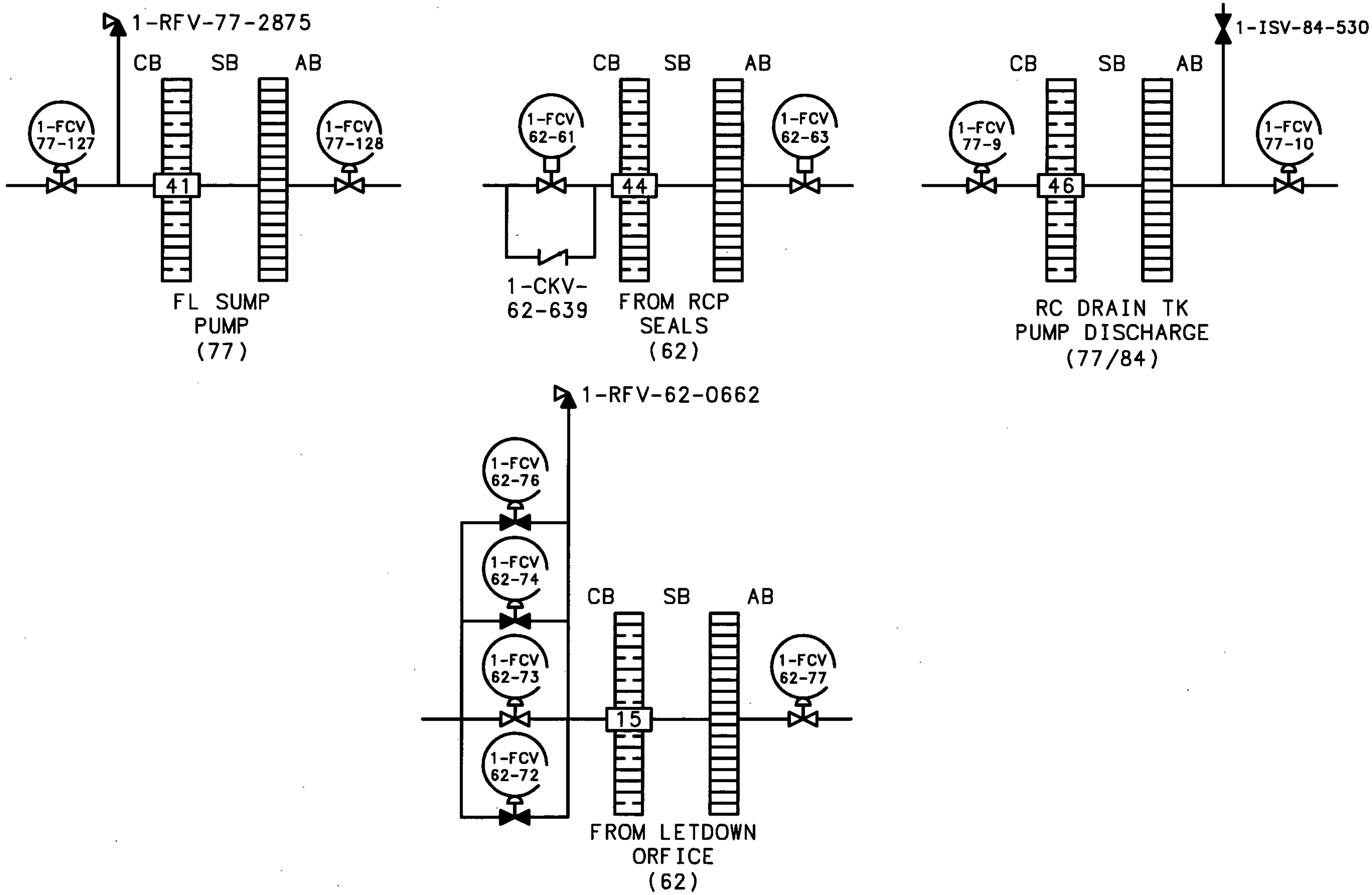


FIGURE 1 (SHEET 1 OF 2)
CONTAINMENT SYSTEMS TOP EVENT CI_LARGE
SIMPLIFIED DIAGRAM

REFERENCE DRAWINGS:
TABLE 6.2.4-1
FSAR

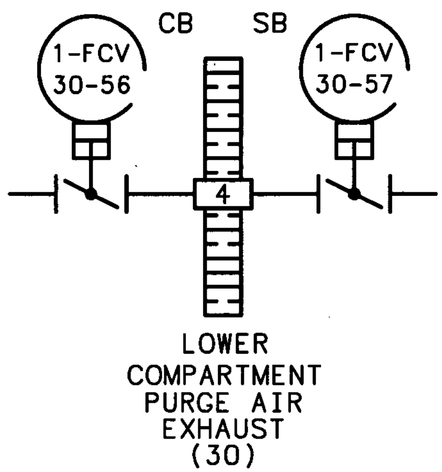
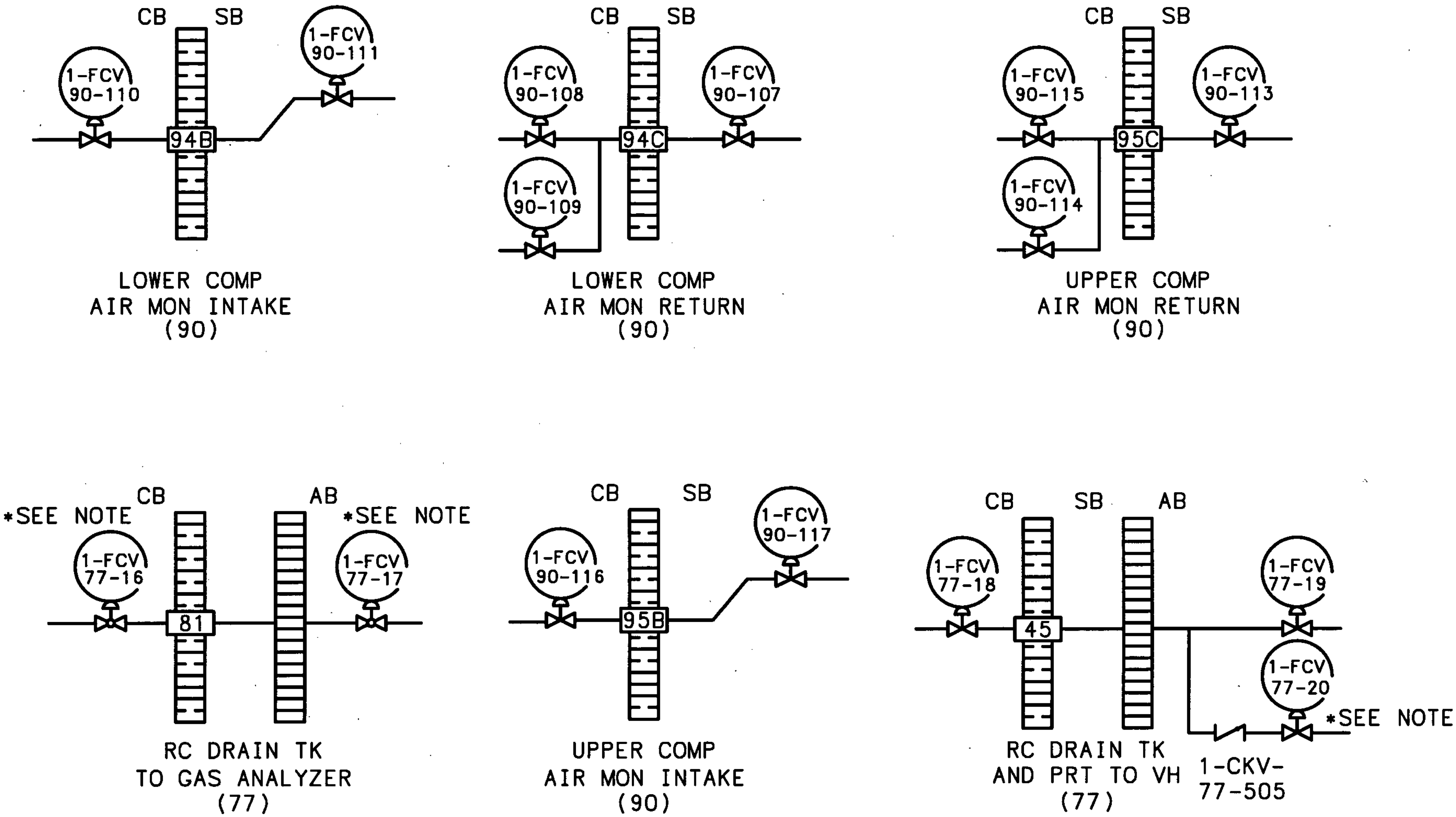


FIGURE 1 (SHEET 2 OF 2)
CONTAINMENT SYSTEMS TOP EVENT CI_LARGE
SIMPLIFIED DIAGRAM

REFERENCE DRAWINGS:
TABLE 6.2.4-1
FSAR



NOTE: POSITION VARIES

REFERENCE DRAWINGS:
TABLE 6.2.4-1
FSAR

FIGURE 2
CONTAINMENT SYSTEMS TOP EVENT CI_SMALL
SIMPLIFIED DIAGRAM

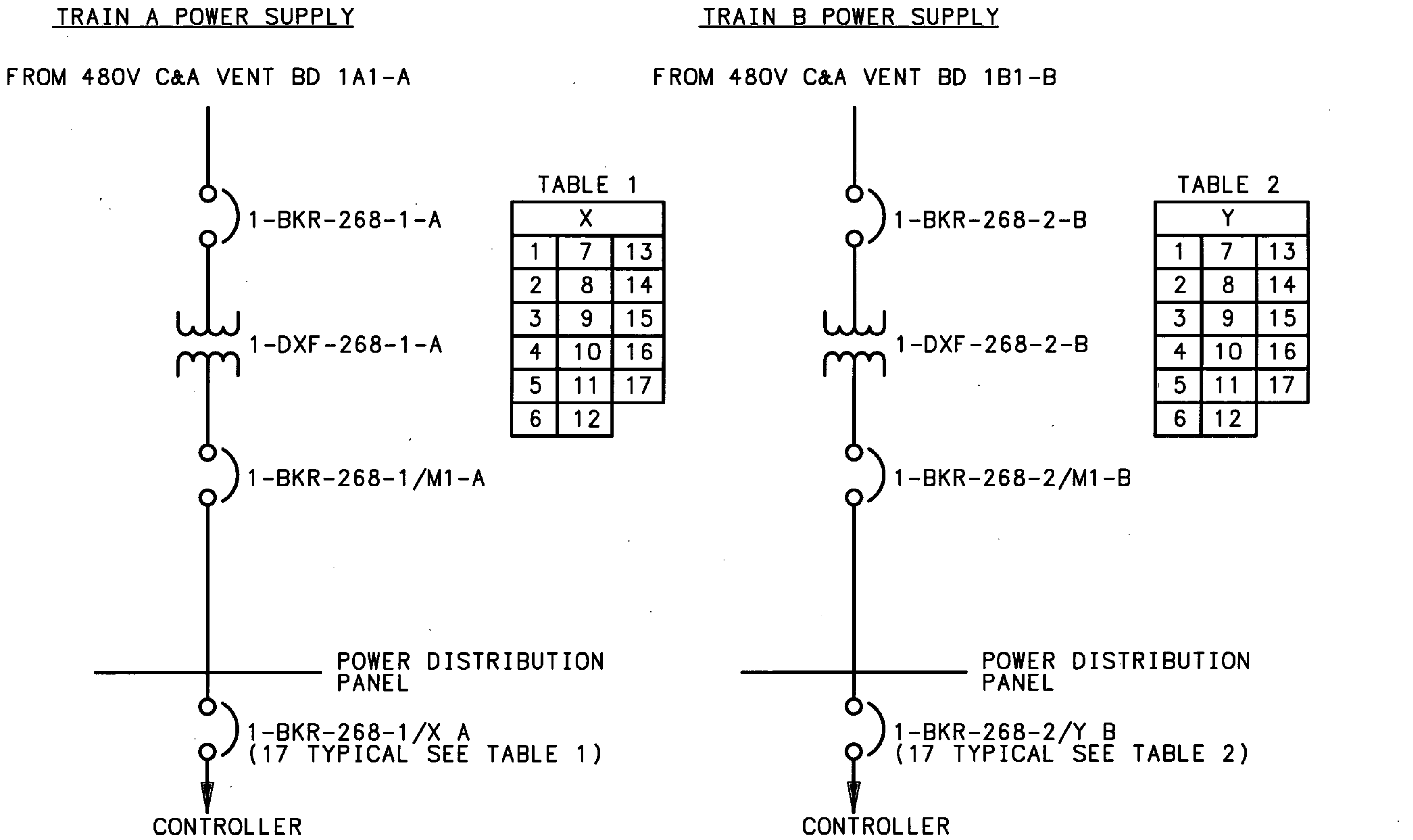


FIGURE 3
CONTAINMENT SYSTEMS
TOP EVENT HH - HYDROGEN IGNITERS
SIMPLIFIED DIAGRAM

REFERENCE DRAWINGS:
1-45W1770-8
1-45W1772-8
1-45W756-2
1-45W756-6

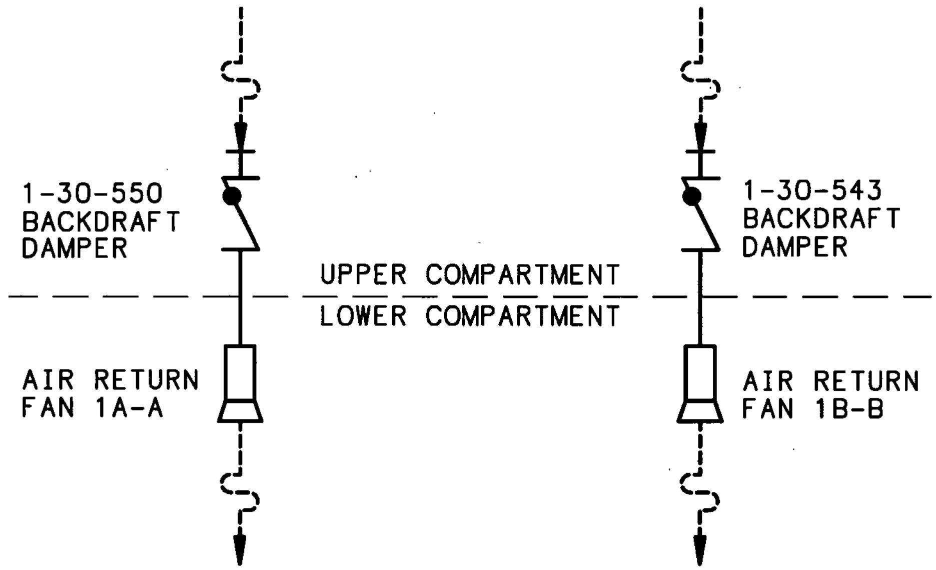
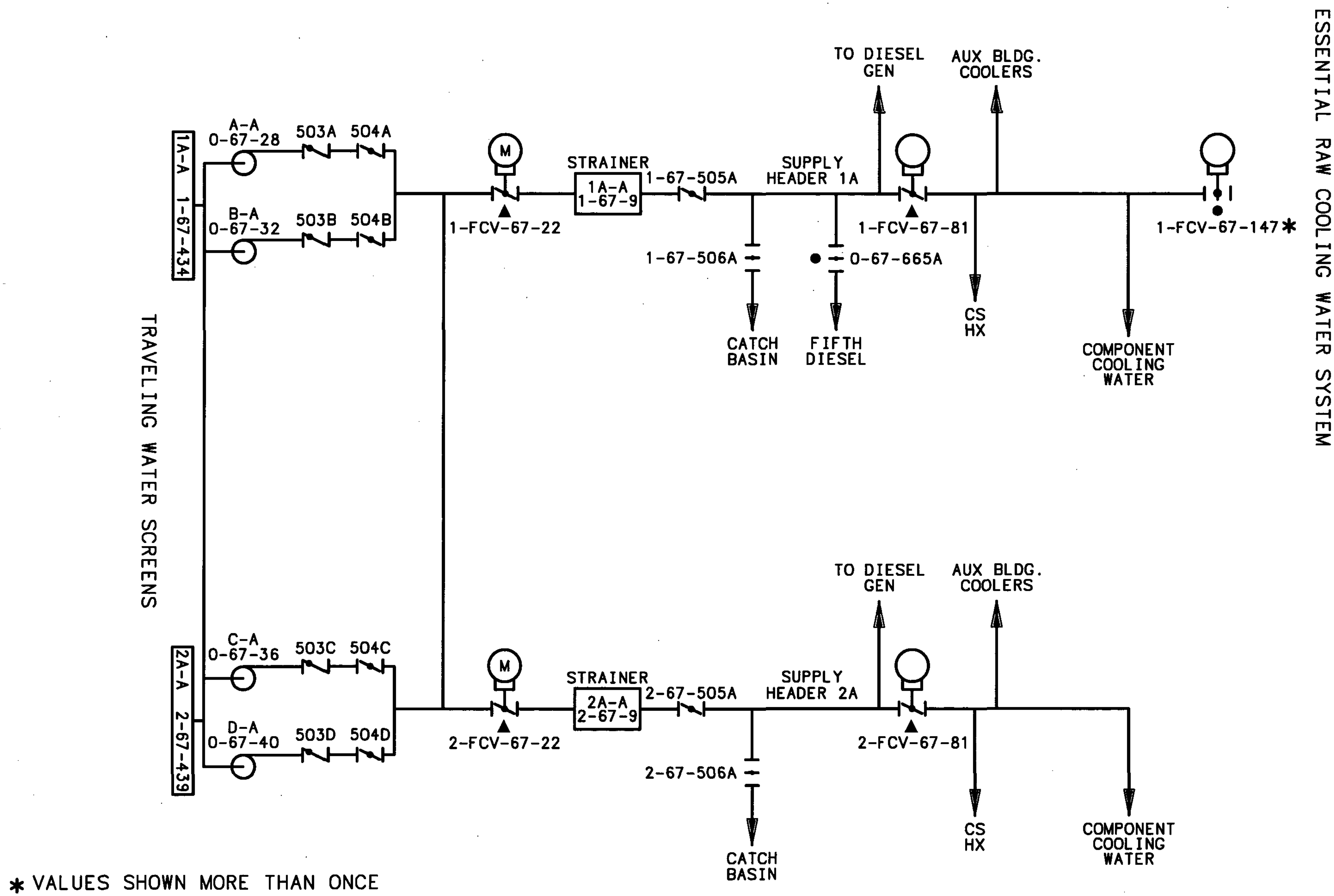


FIGURE 4
CONTAINMENT SYSTEMS TOP EVENT AR
SIMPLIFIED DIAGRAM

REFERENCE DRAWINGS:
1-47W866-1



* VALUES SHOWN MORE THAN ONCE
▲ NORMALLY OPEN WITH POWER REMOVED
● NORMALLY CLOSED WITH POWER REMOVED

FIGURE 1
TRAIN A SUPPLY HEADERS

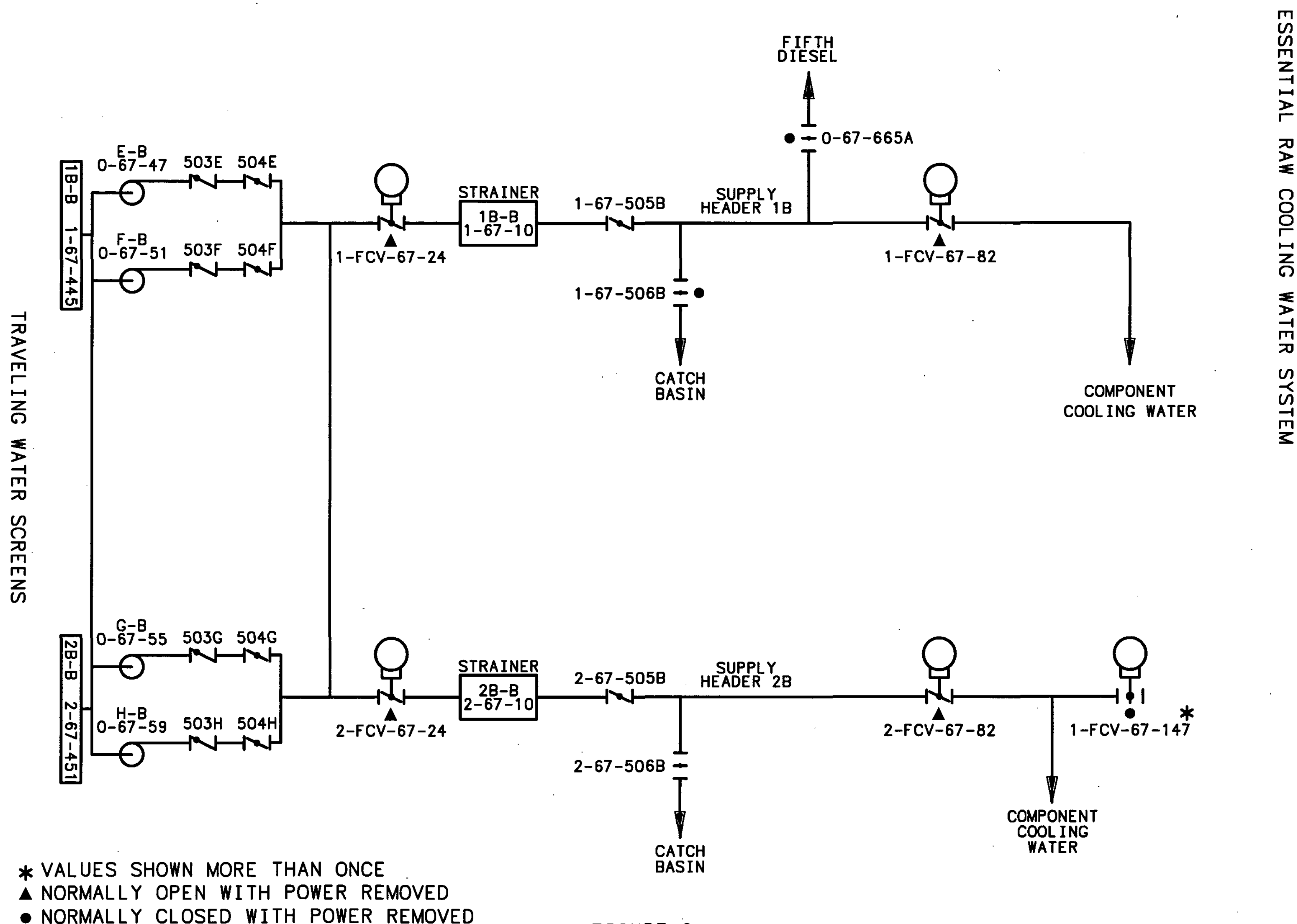


FIGURE 2
TRAIN B SUPPLY HEADERS

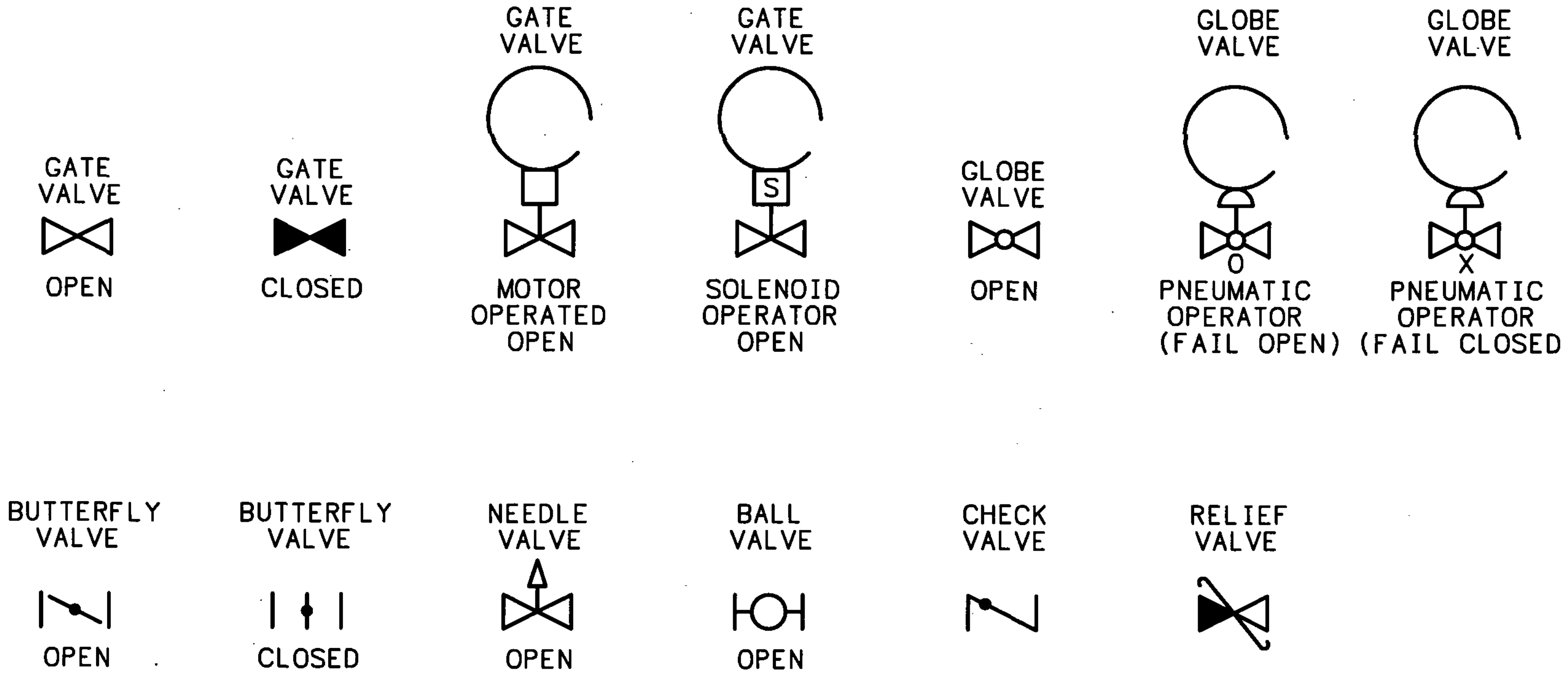


FIGURE 1
SIMPLIFIED DRAWING VALVE SYMBOLS

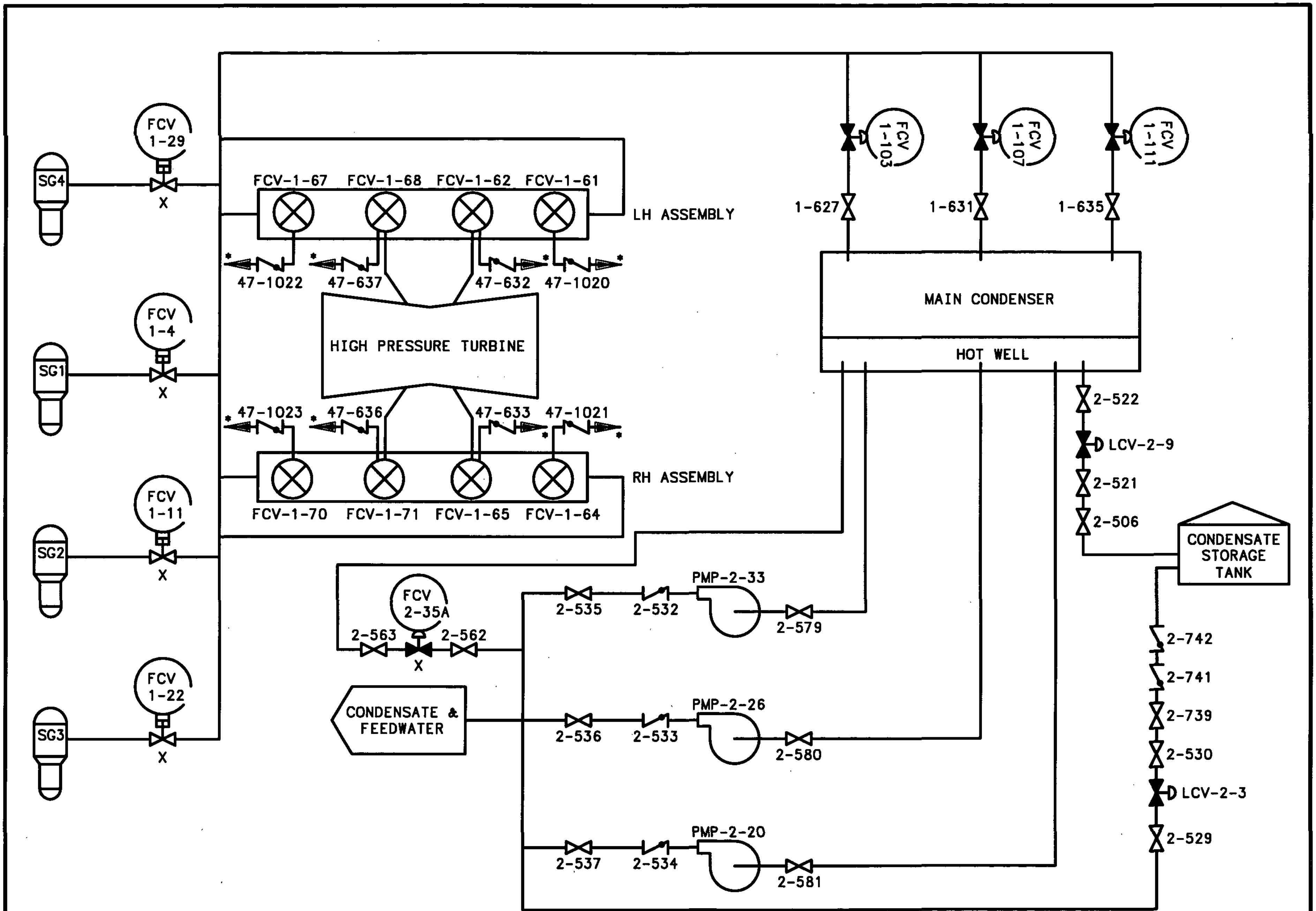


FIGURE 1
MAIN STEAM

*TO EHC RESEROIR

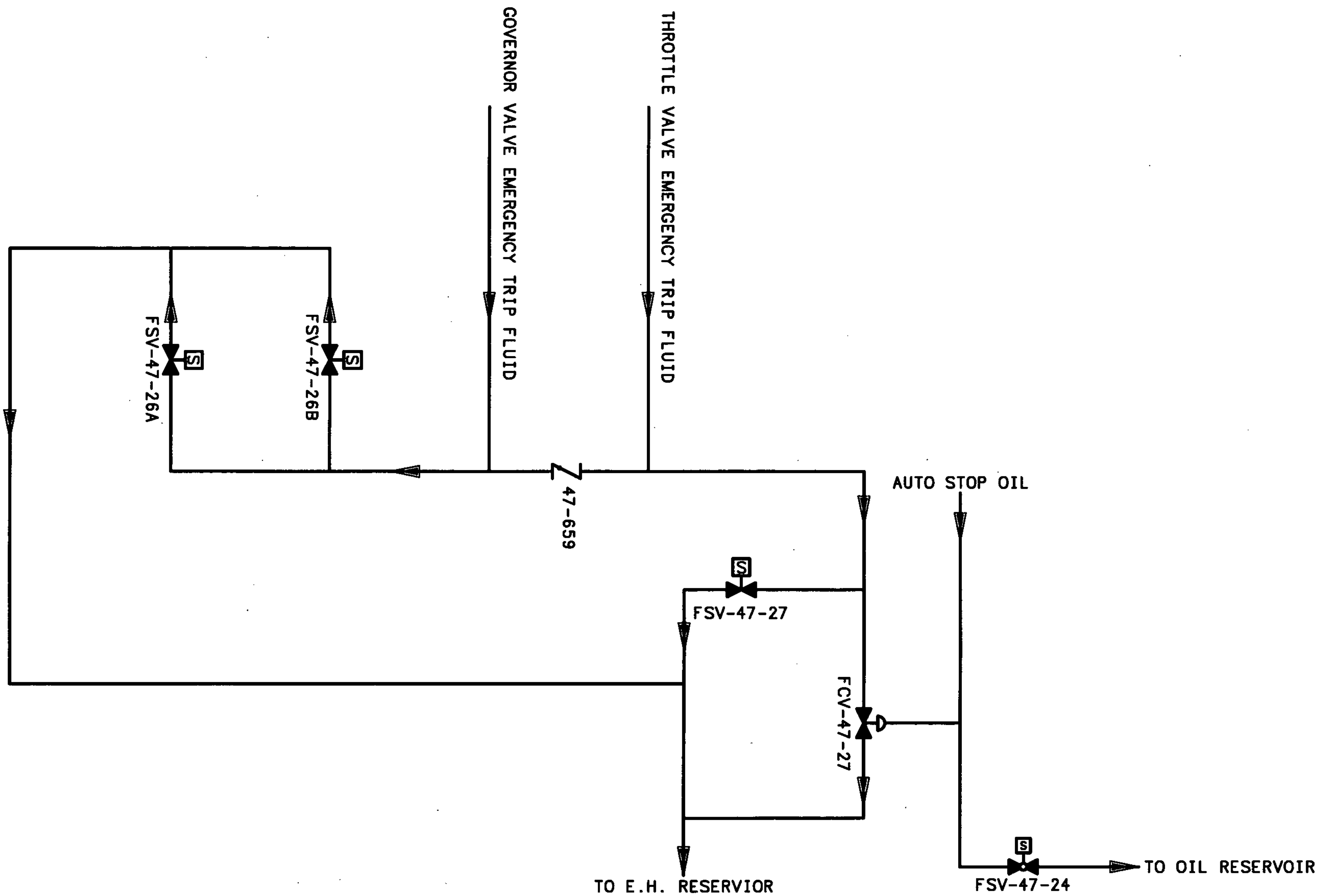


FIGURE 2
EHC FLOW DIAGRAM

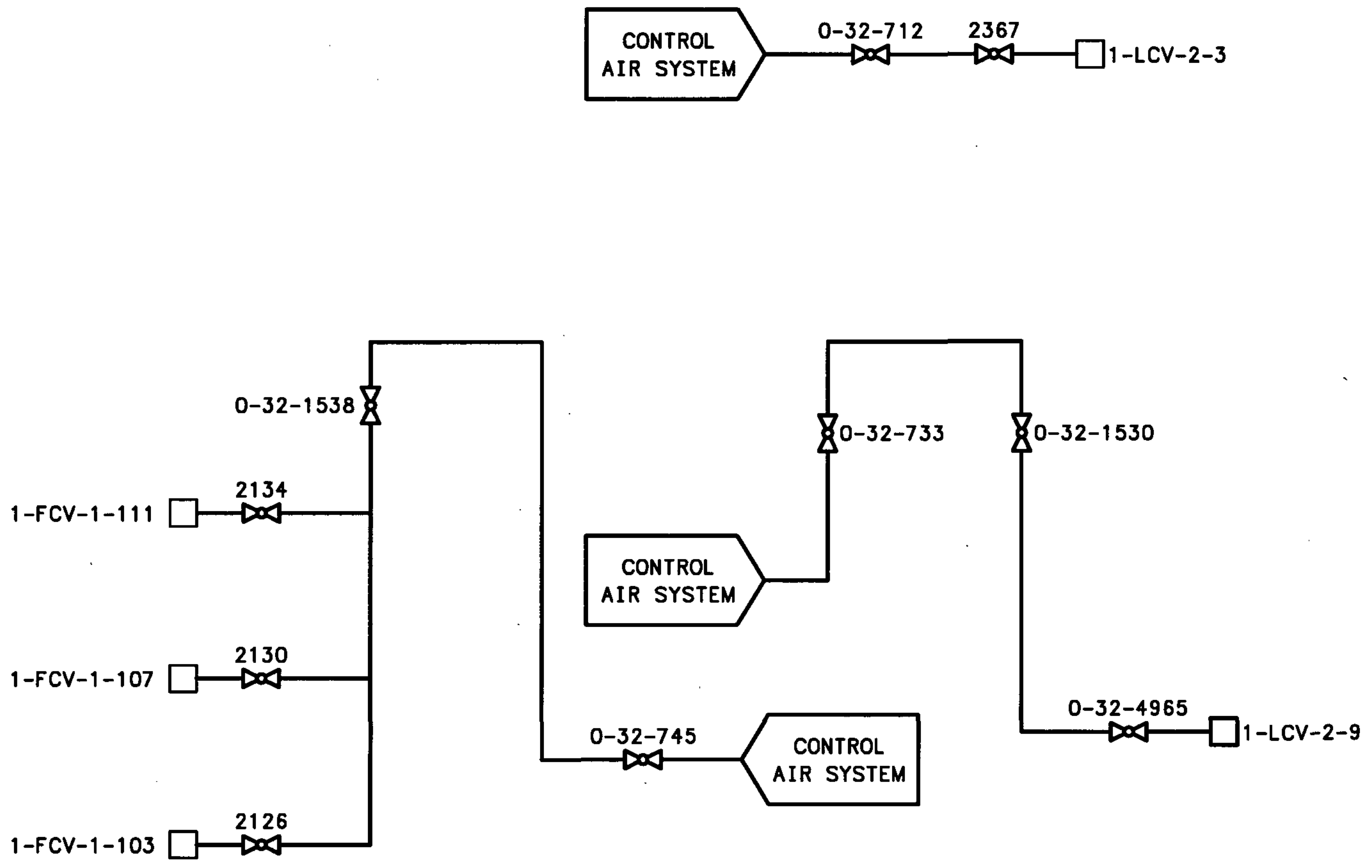
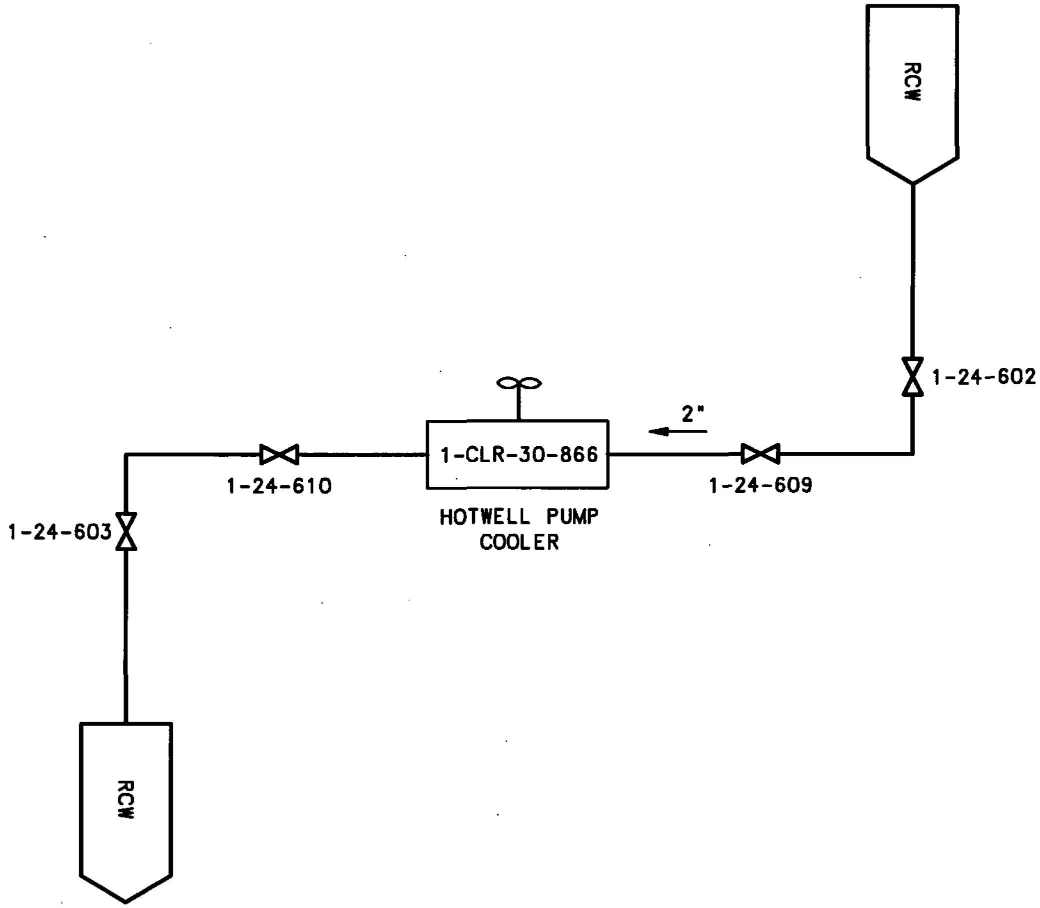


FIGURE 3
AIR SUPPLY VALVES



REFERENCE DRAWING:
47W844-2

FIGURE 4
HOTWELL PUMP COOLING

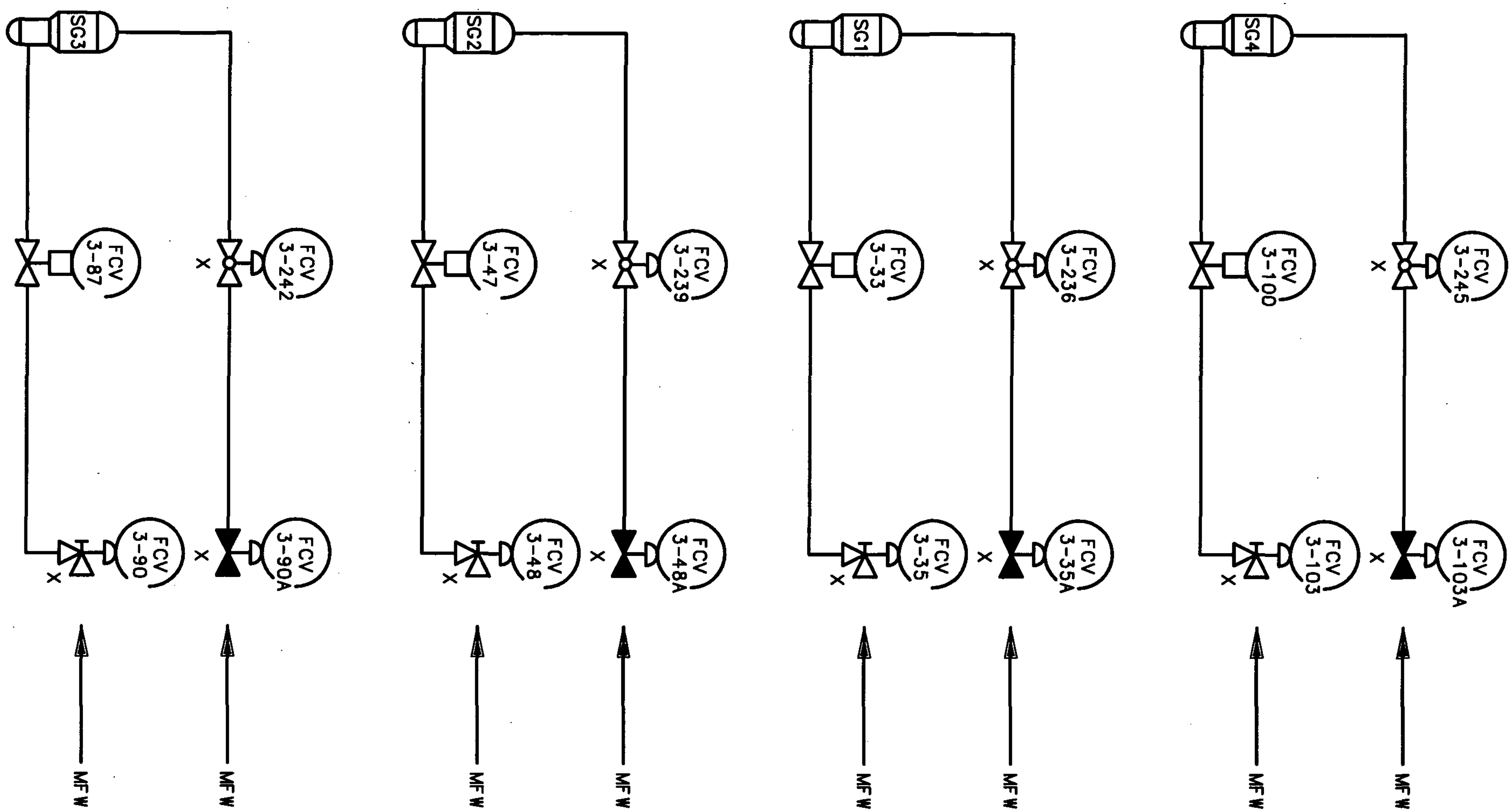


FIGURE 5
MFW ISOLATION

REFERENCE DRAWING:
1-47W803-1

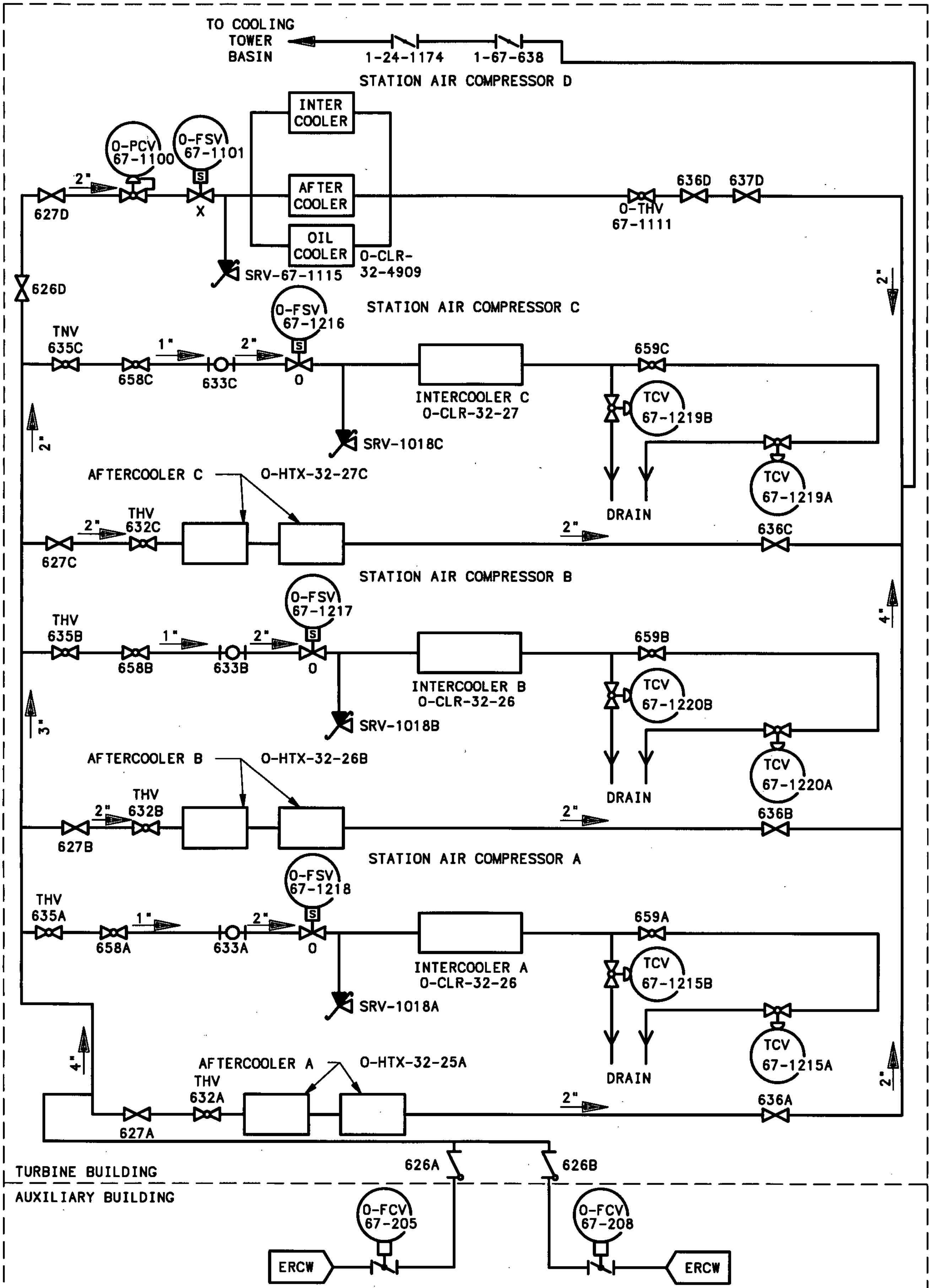


FIGURE 1
CAS COMPRESSOR ERCW COOLING SUPPLY

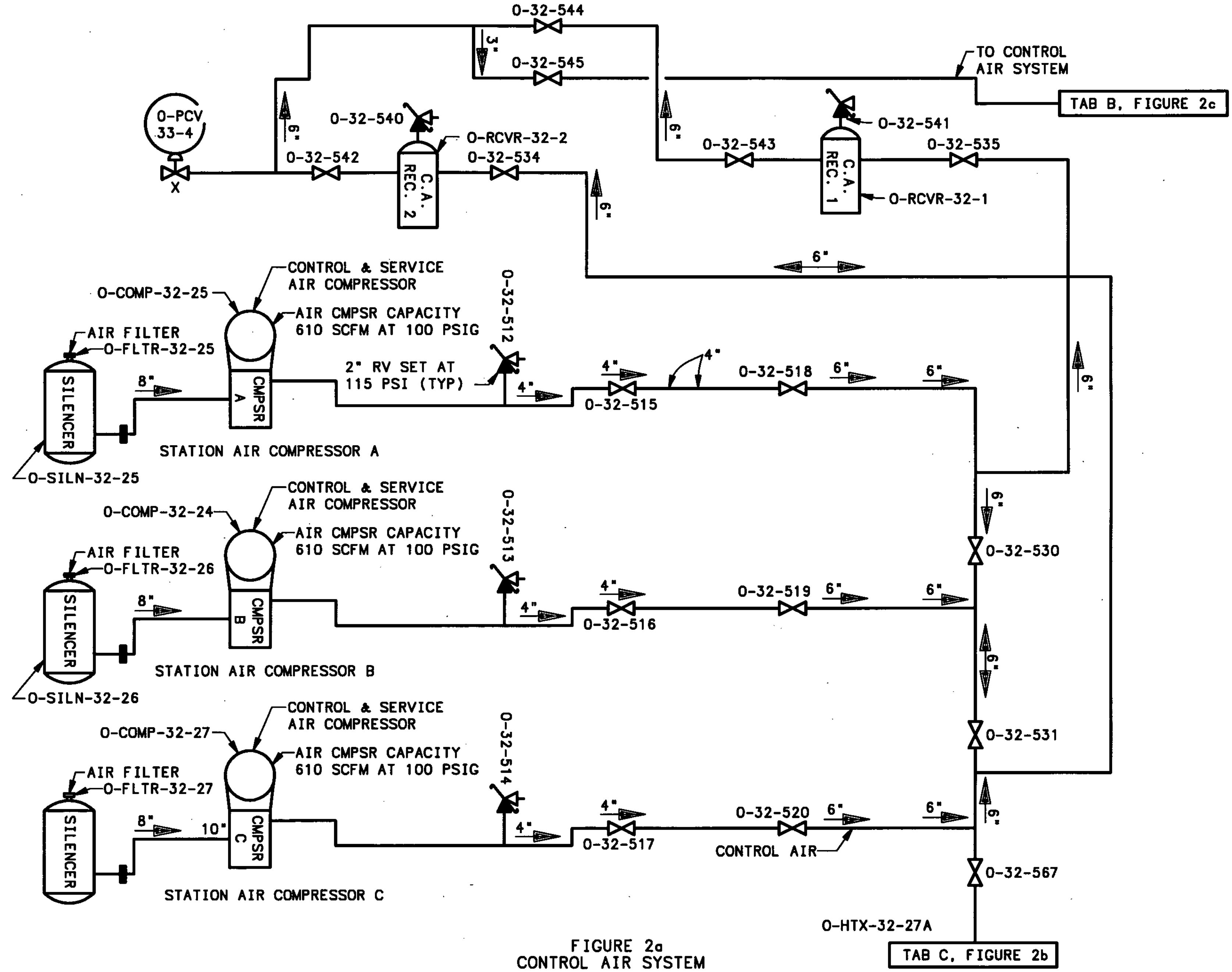


FIGURE 2a
CONTROL AIR SYSTEM

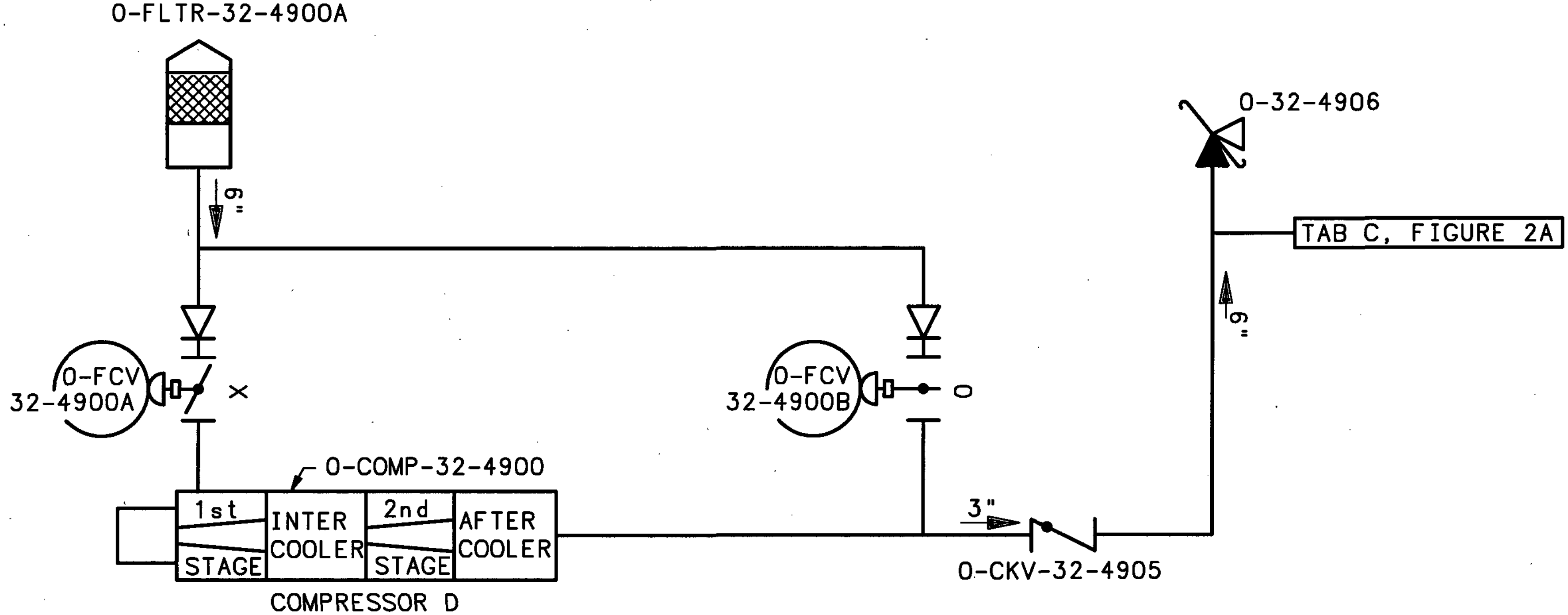


FIGURE 2b
CONTROL AIR SYSTEM

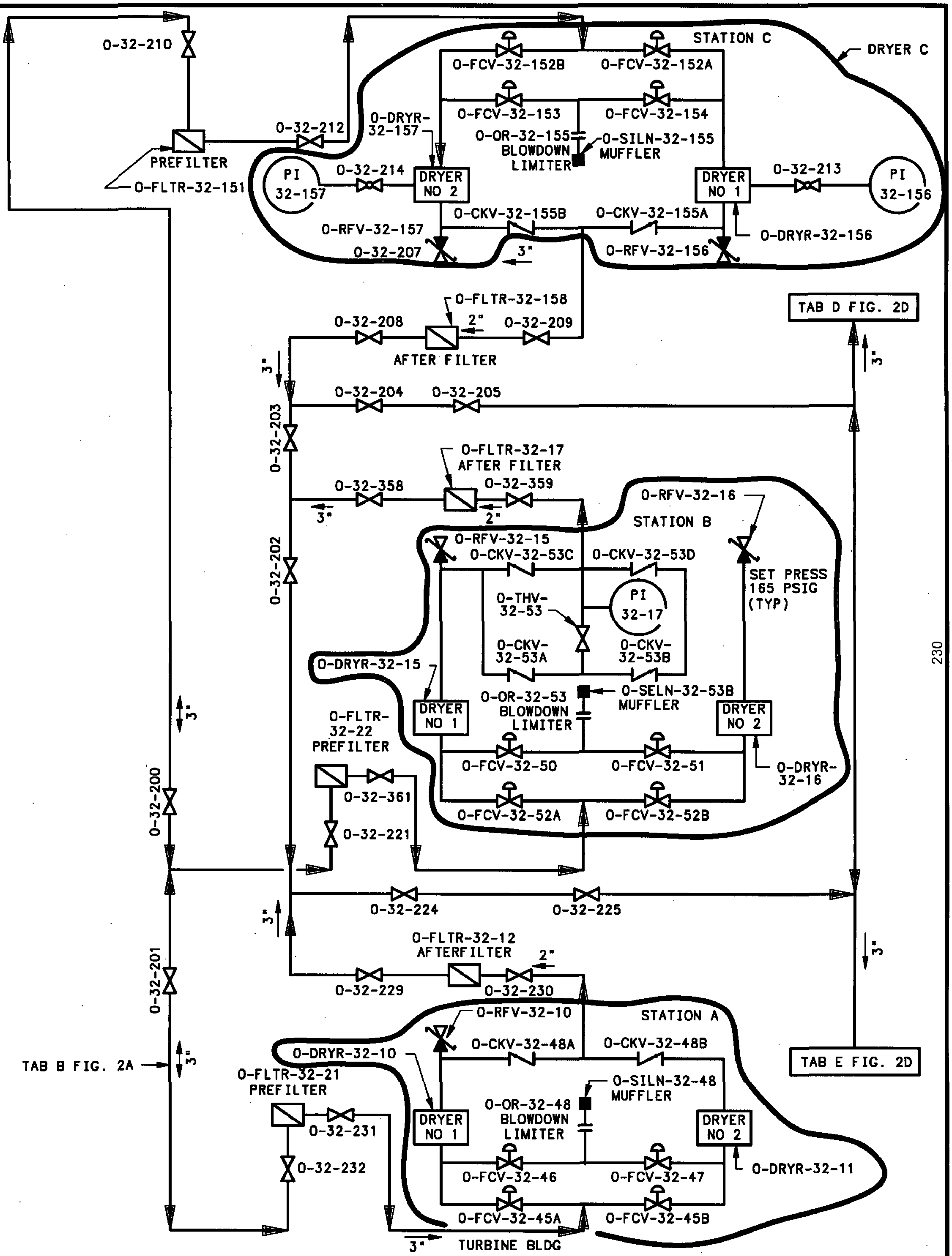


FIGURE 2c
CONTROL AIR SYSTEM

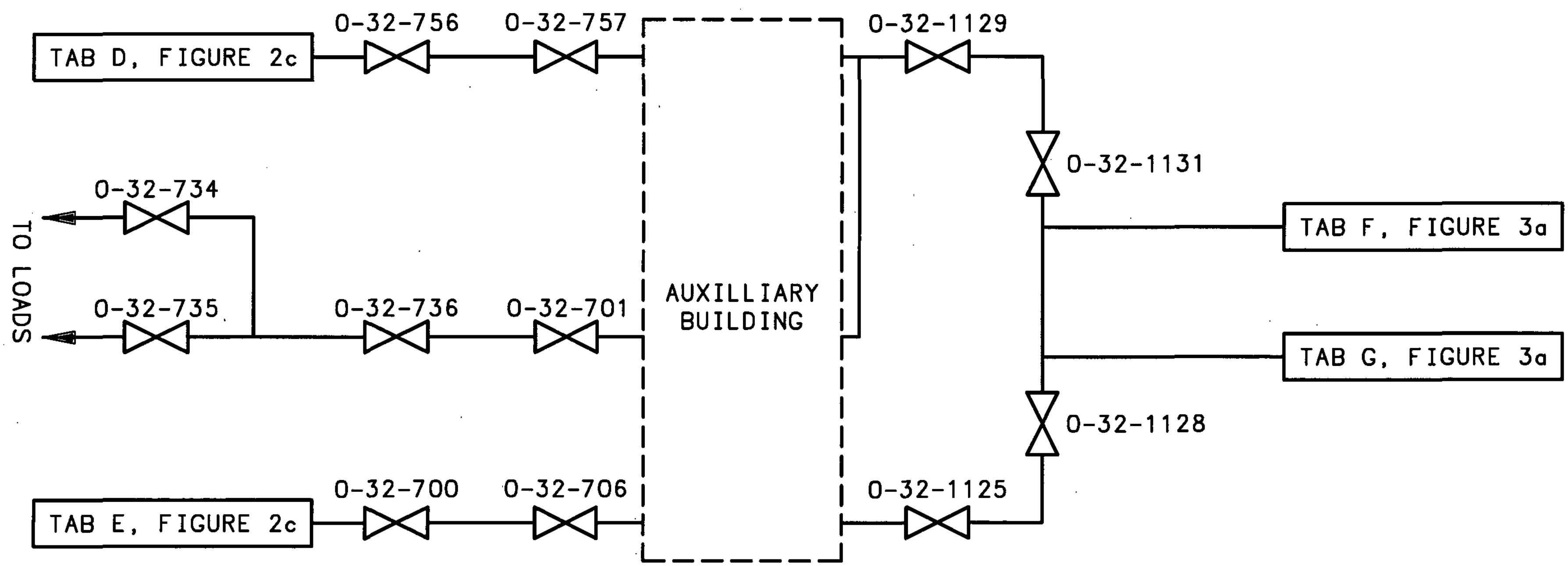


FIGURE 2d
CONTROL AIR SYSTEM

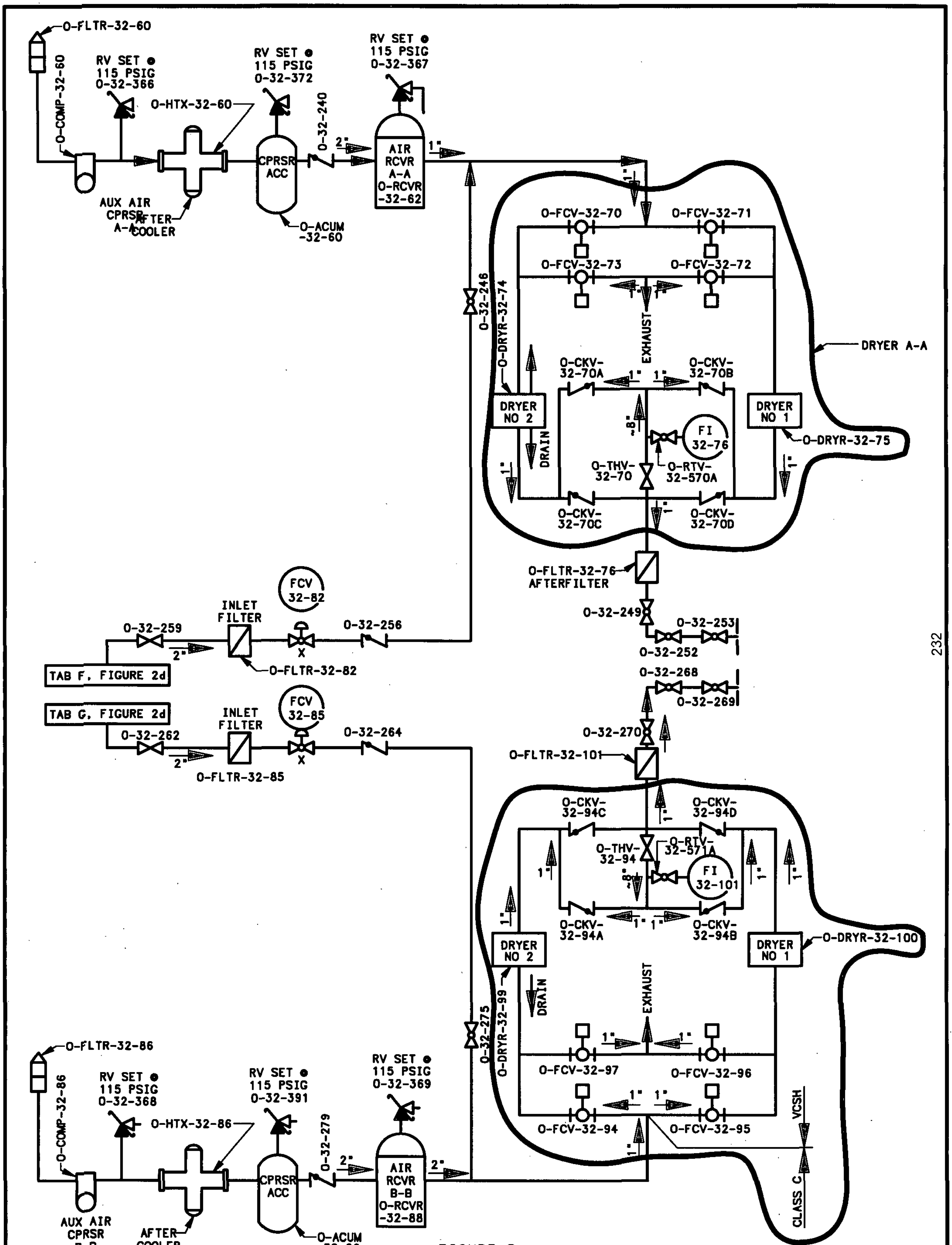


FIGURE 3
AUXILIARY CONTROL AIR SYSTEM

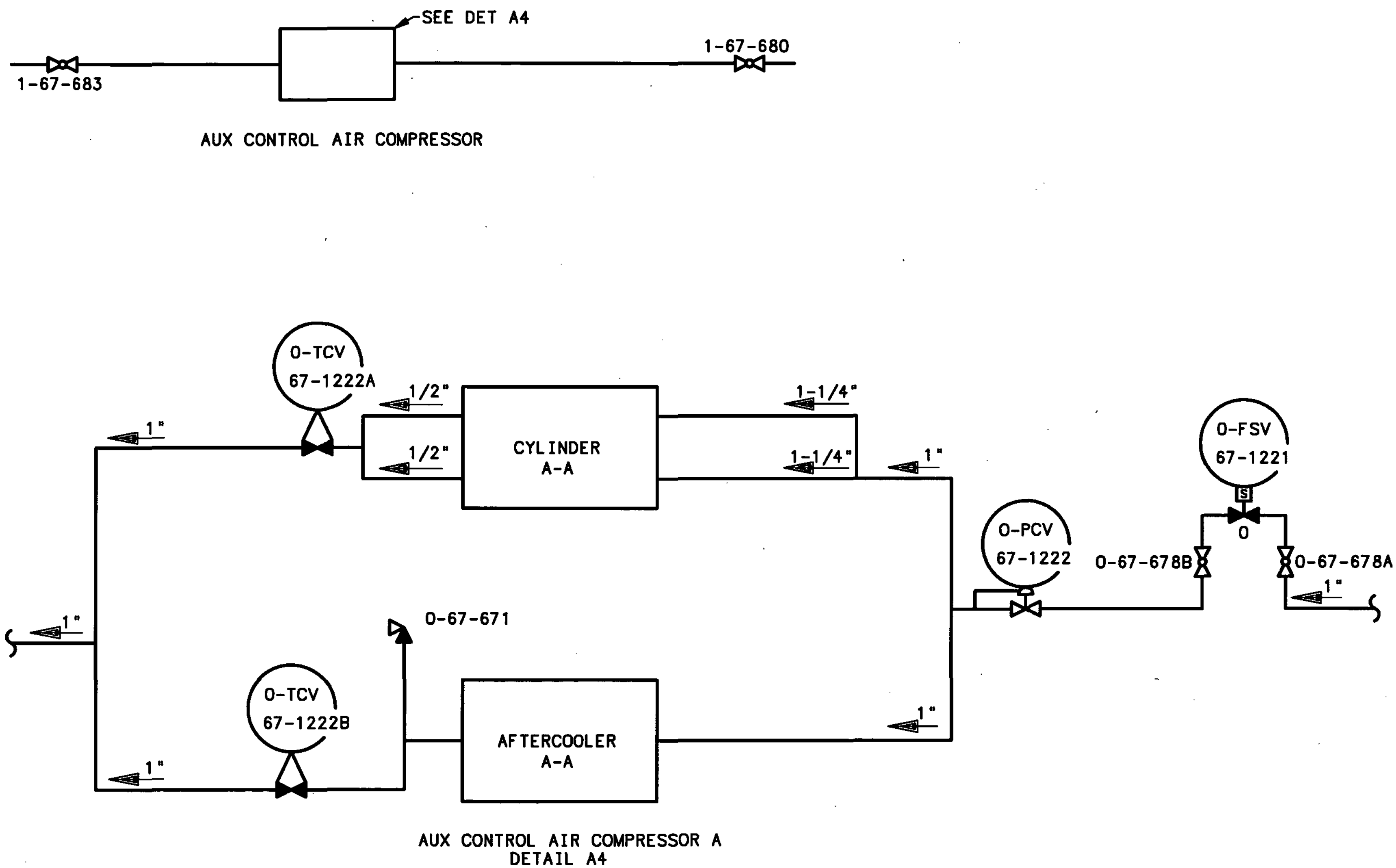


FIGURE 4a
ERCW COOLING FOR ACAS COMPRESSOR A-A

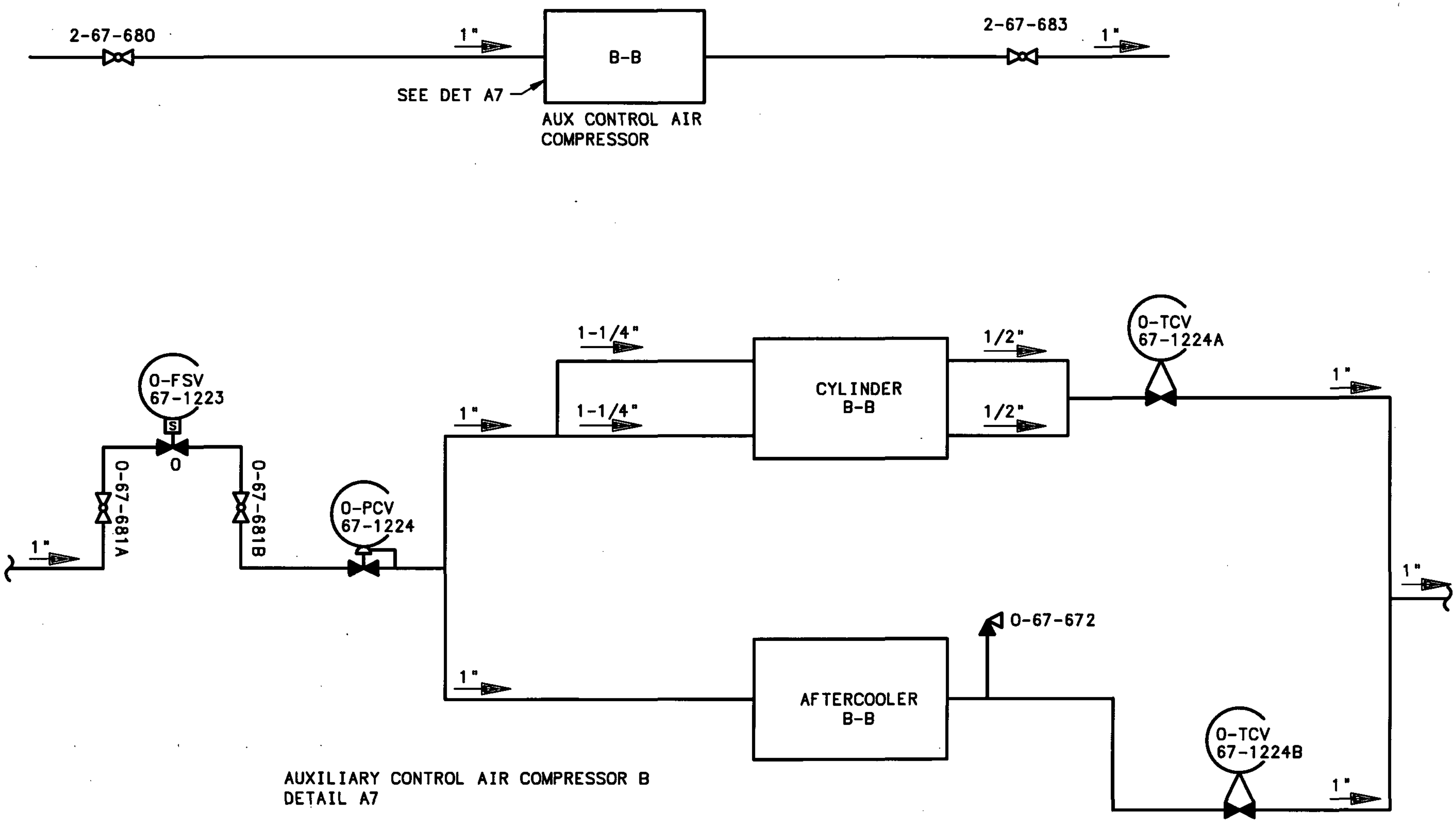


FIGURE 4b
ERCW COOLING FOR COMPRESSOR B-B

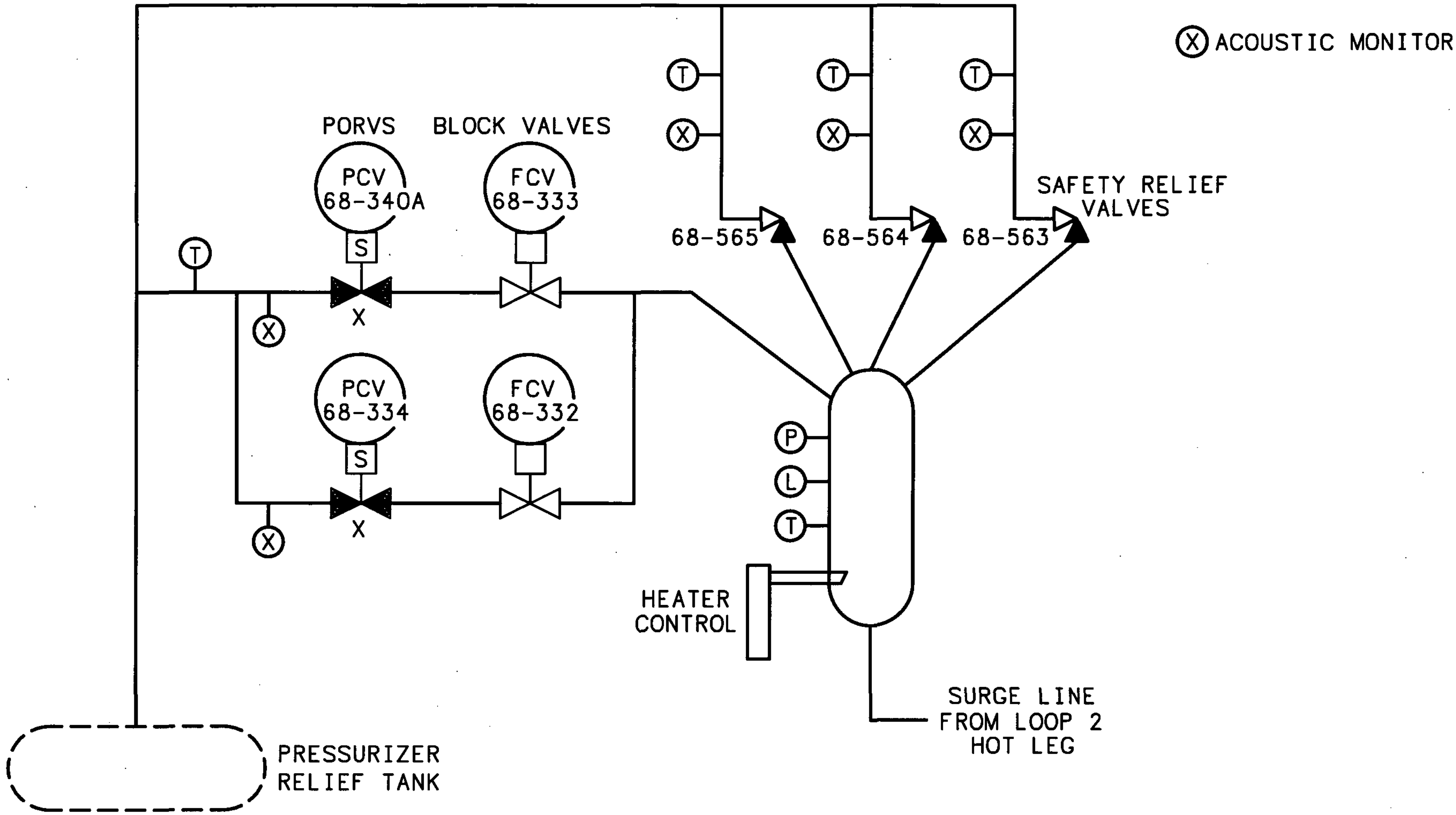
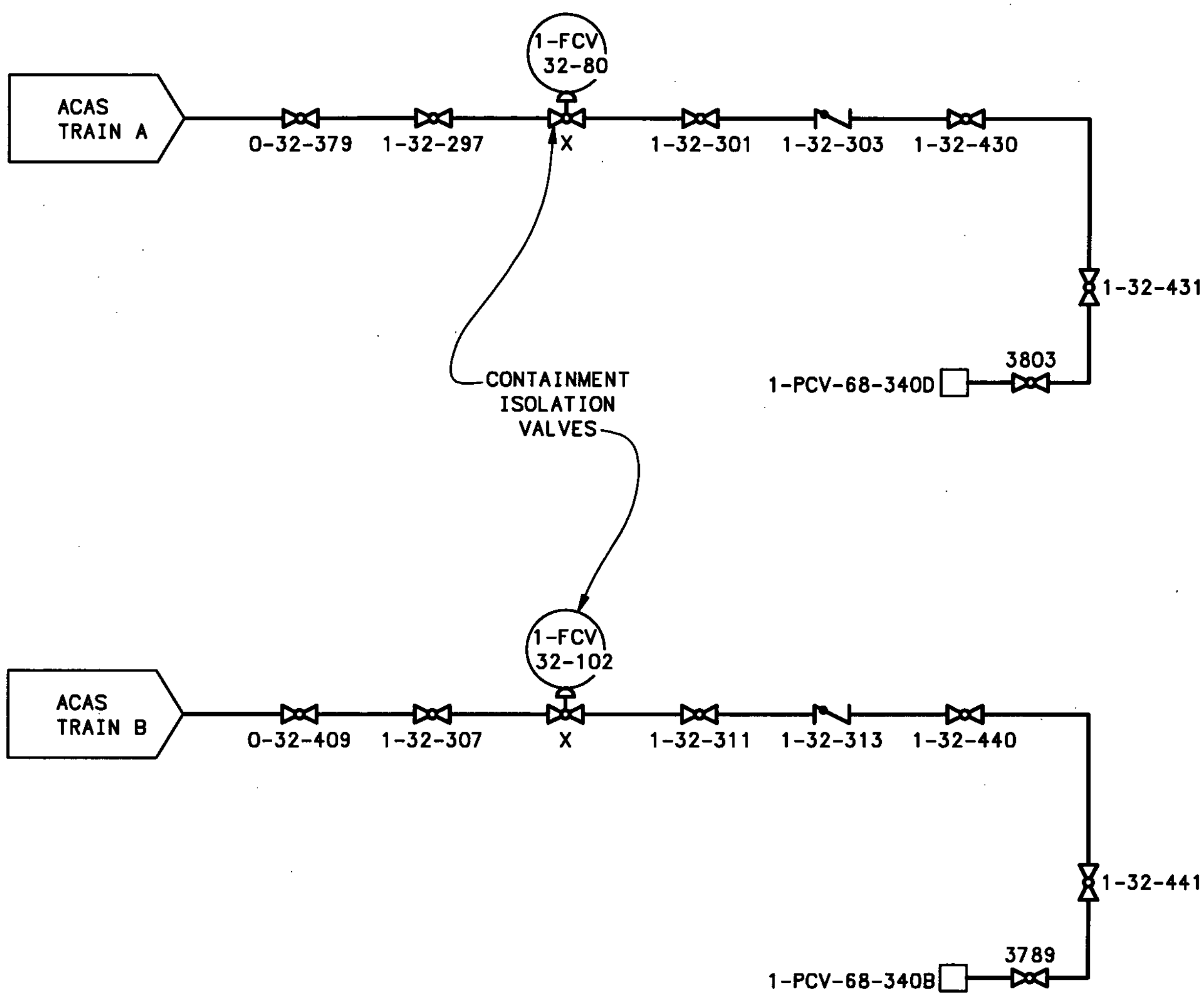


FIGURE 1
PRESSURIZER PORV's, SV's

REFERENCE DRAWING:
1-47W813-1



REFERENCE DRAWINGS:
1-47W848-1
1-47W848-10

FIGURE 2
AUX CONTROL AIR SUPPORT FOR SPRAY VALVES

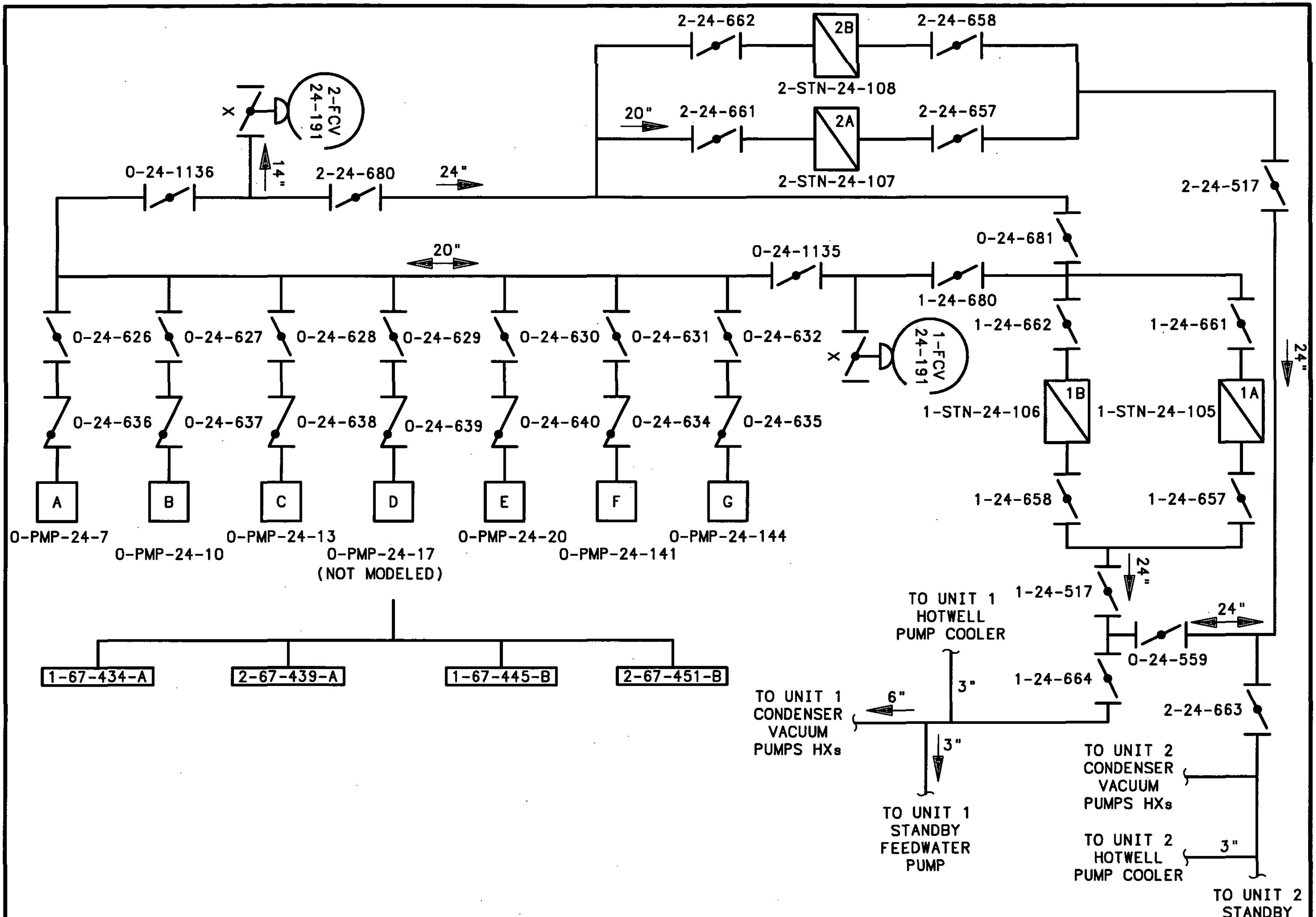


FIGURE 1
RAW COOLING WATER SYSTEM SUPPLY

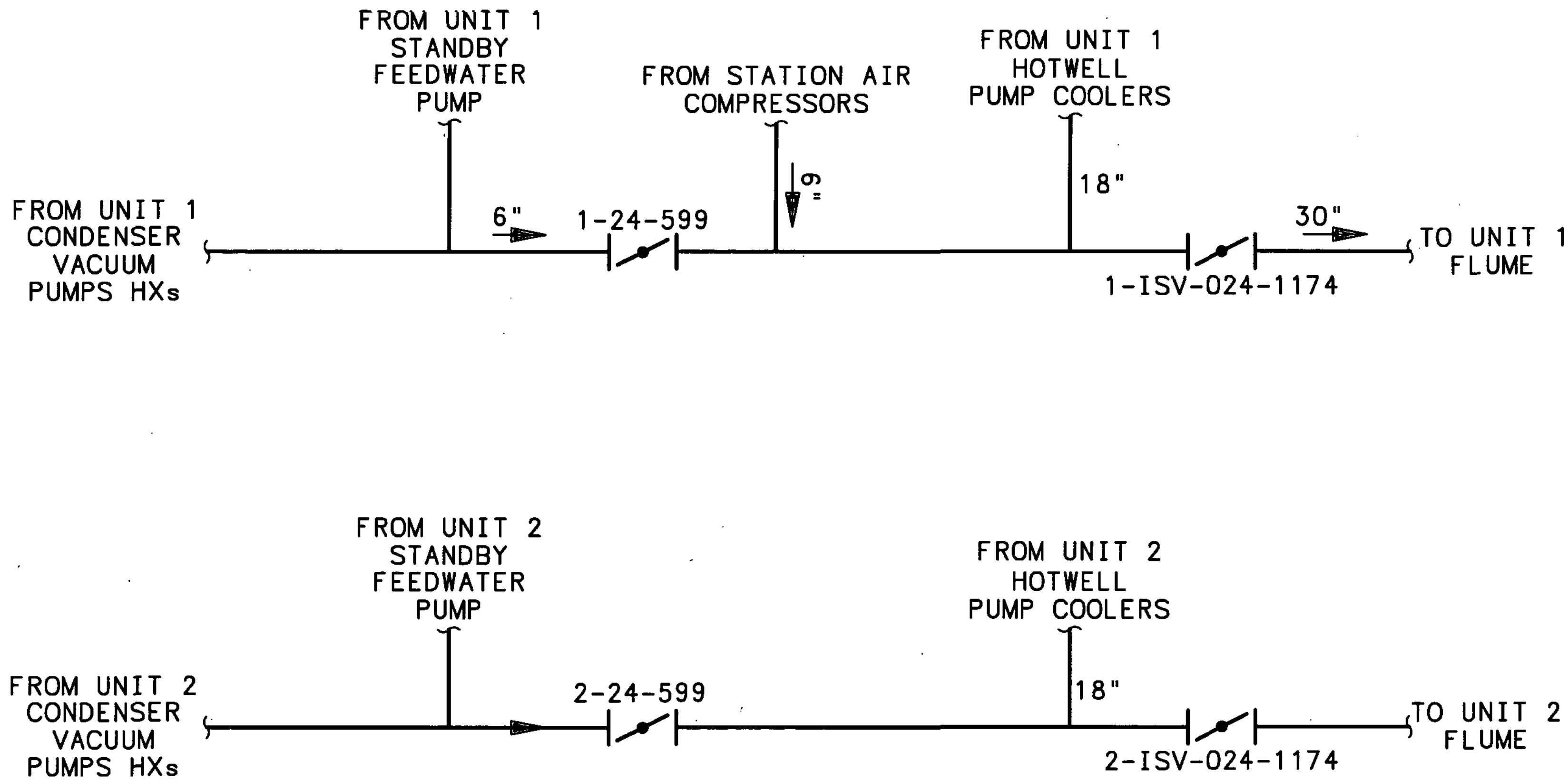


FIGURE 2
RAW COOLING WATER SYSTEM DISCHARGE

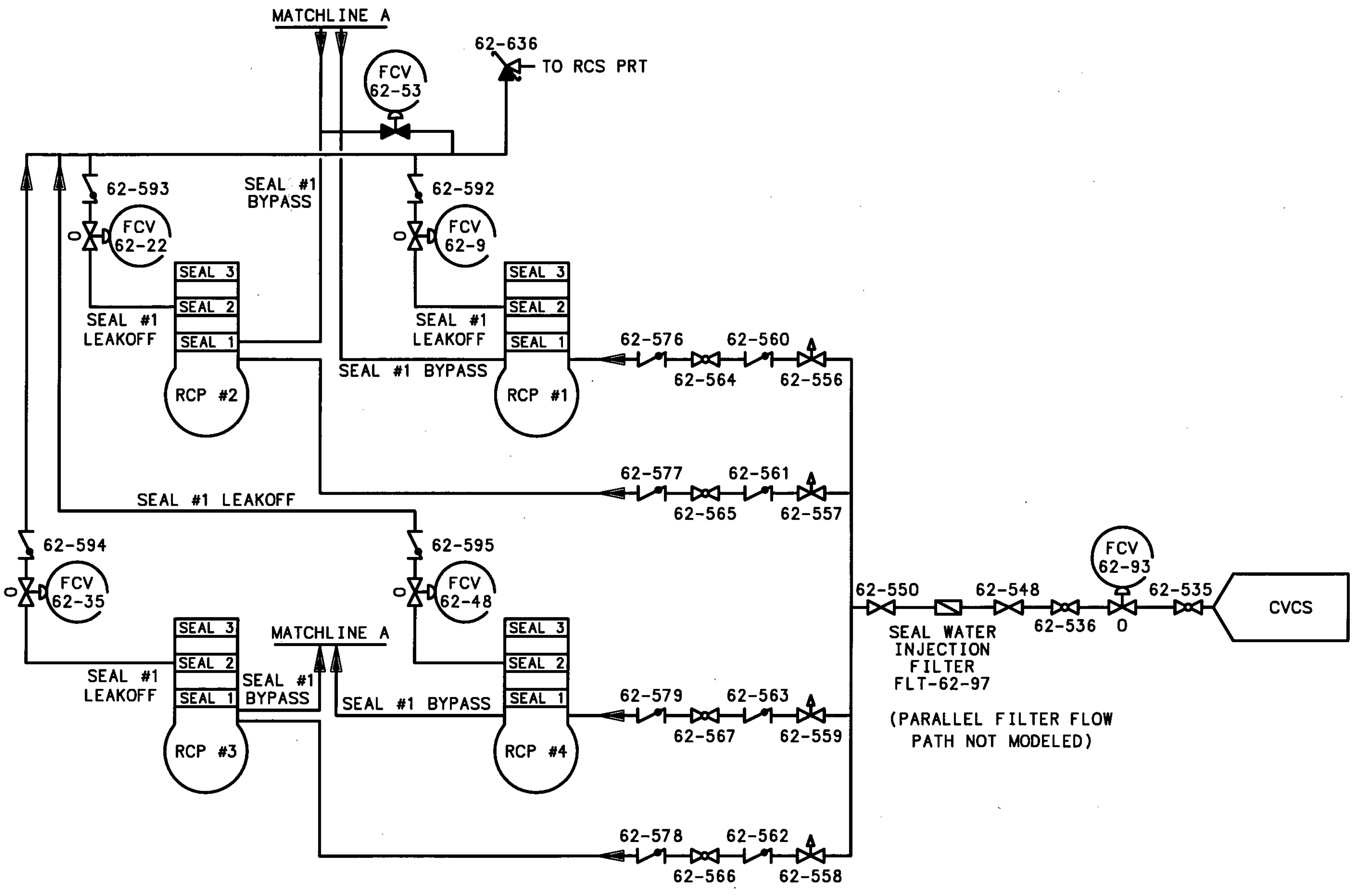


FIGURE 1
RCP SEAL INJECTION

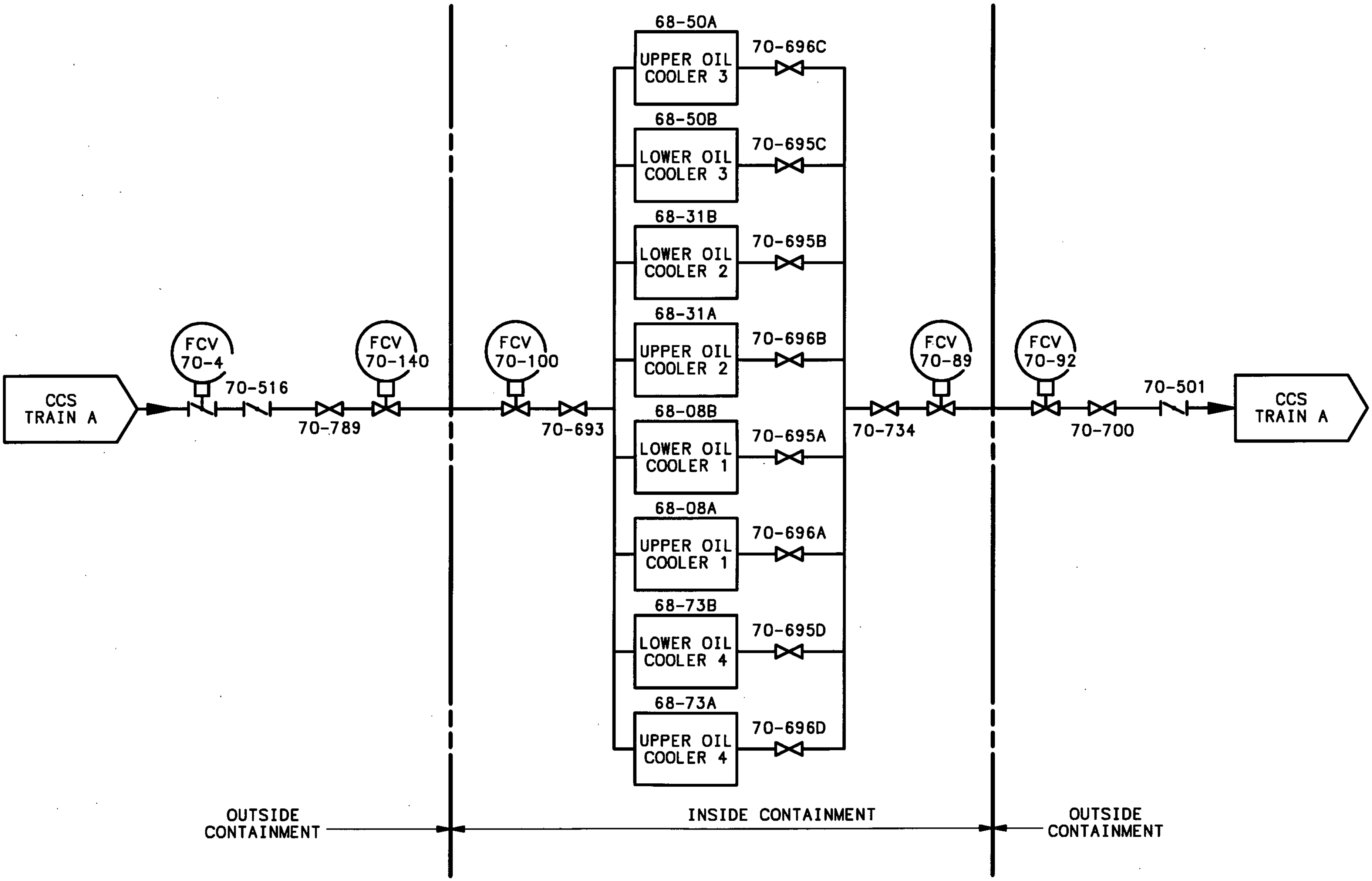


FIGURE 2
RCP BEARING OIL COOLING

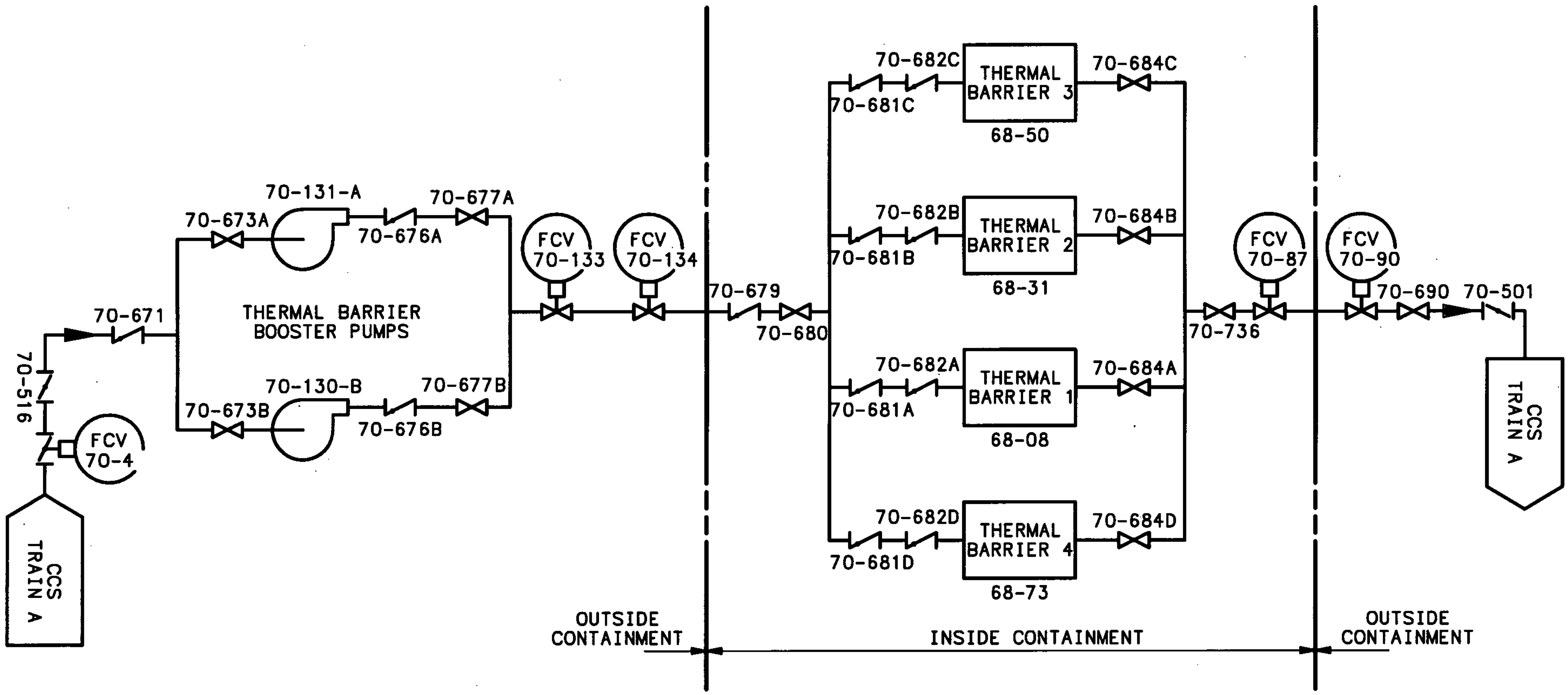


FIGURE 3
THERMAL BARRIER HEAT EXCHANGER

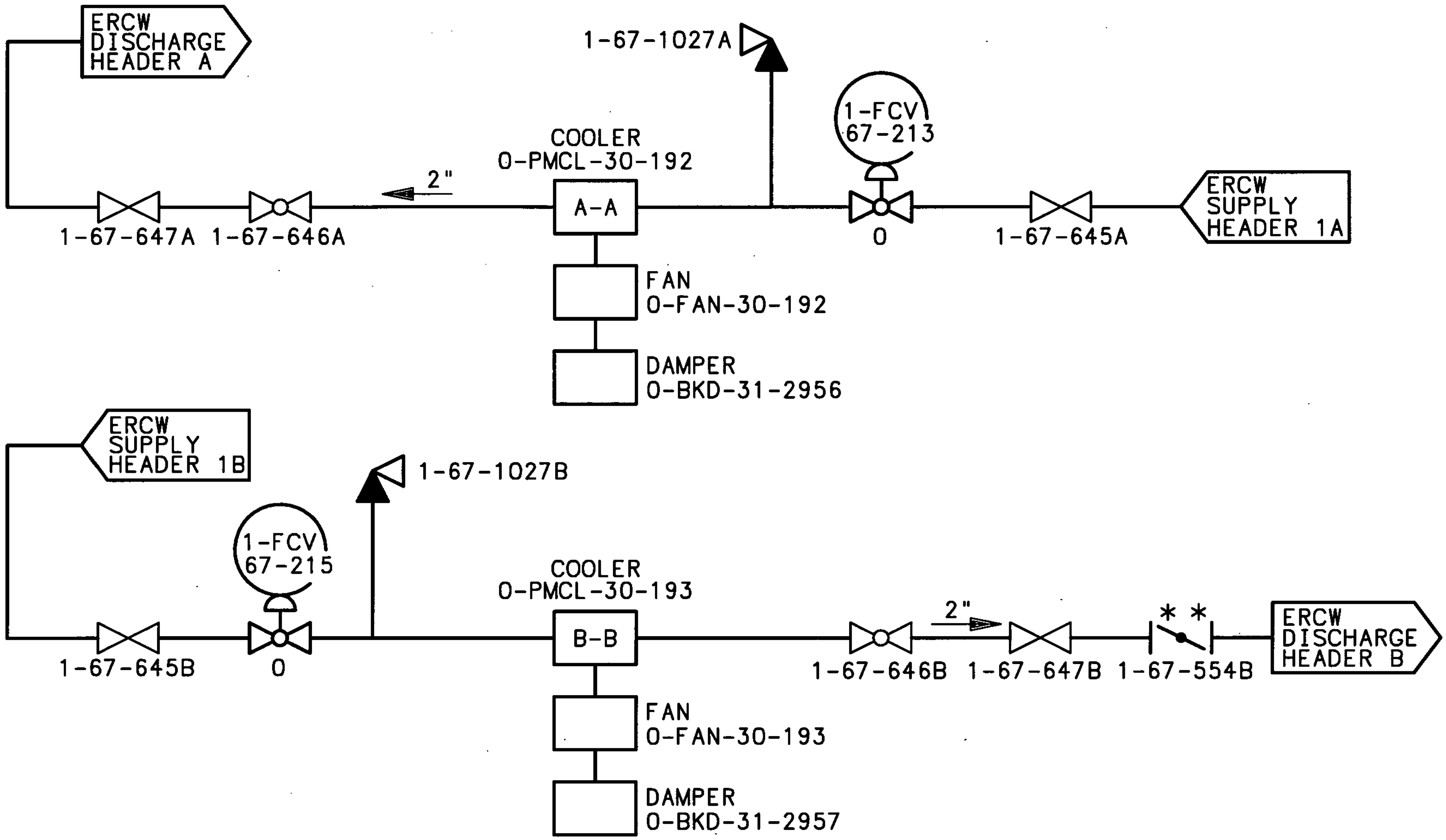


FIGURE 4
SPENT FUEL PIT PUMP AND TB BOOSTER
PUMP SPACE COOLING

* * COMMON TO CVCS, SI, RHR, AND CS

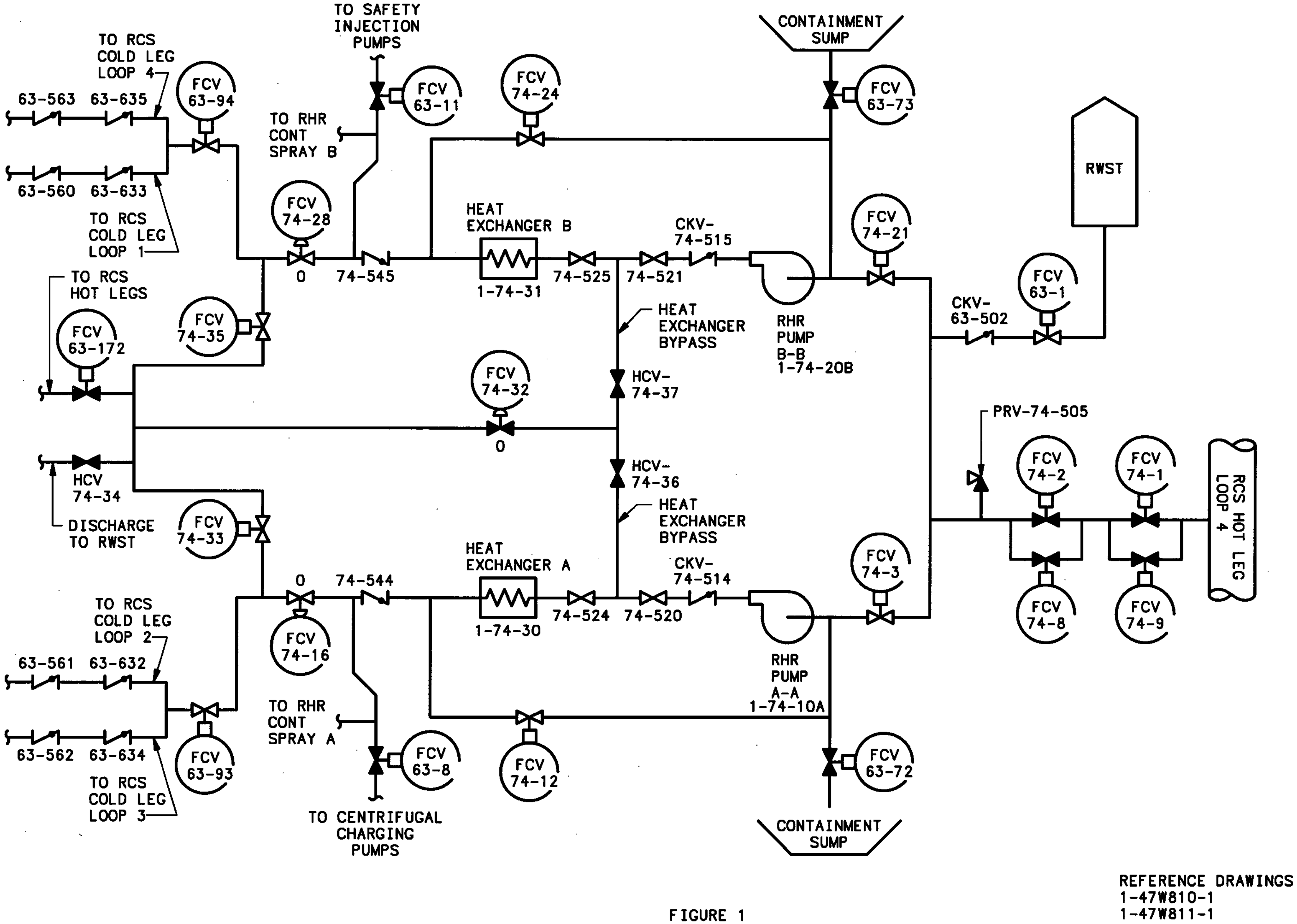


FIGURE 1
RHR STANDBY ALIGNMENT

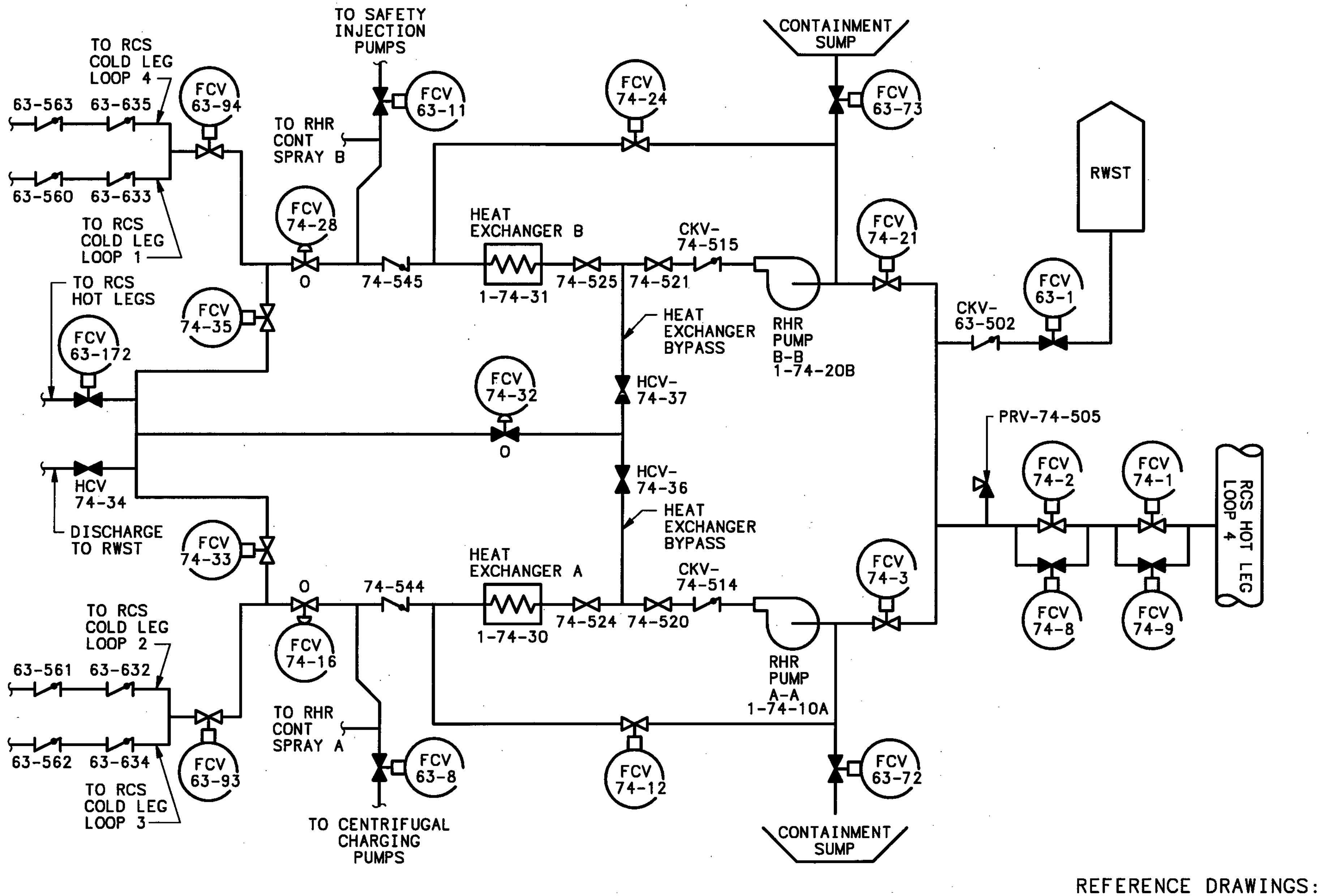
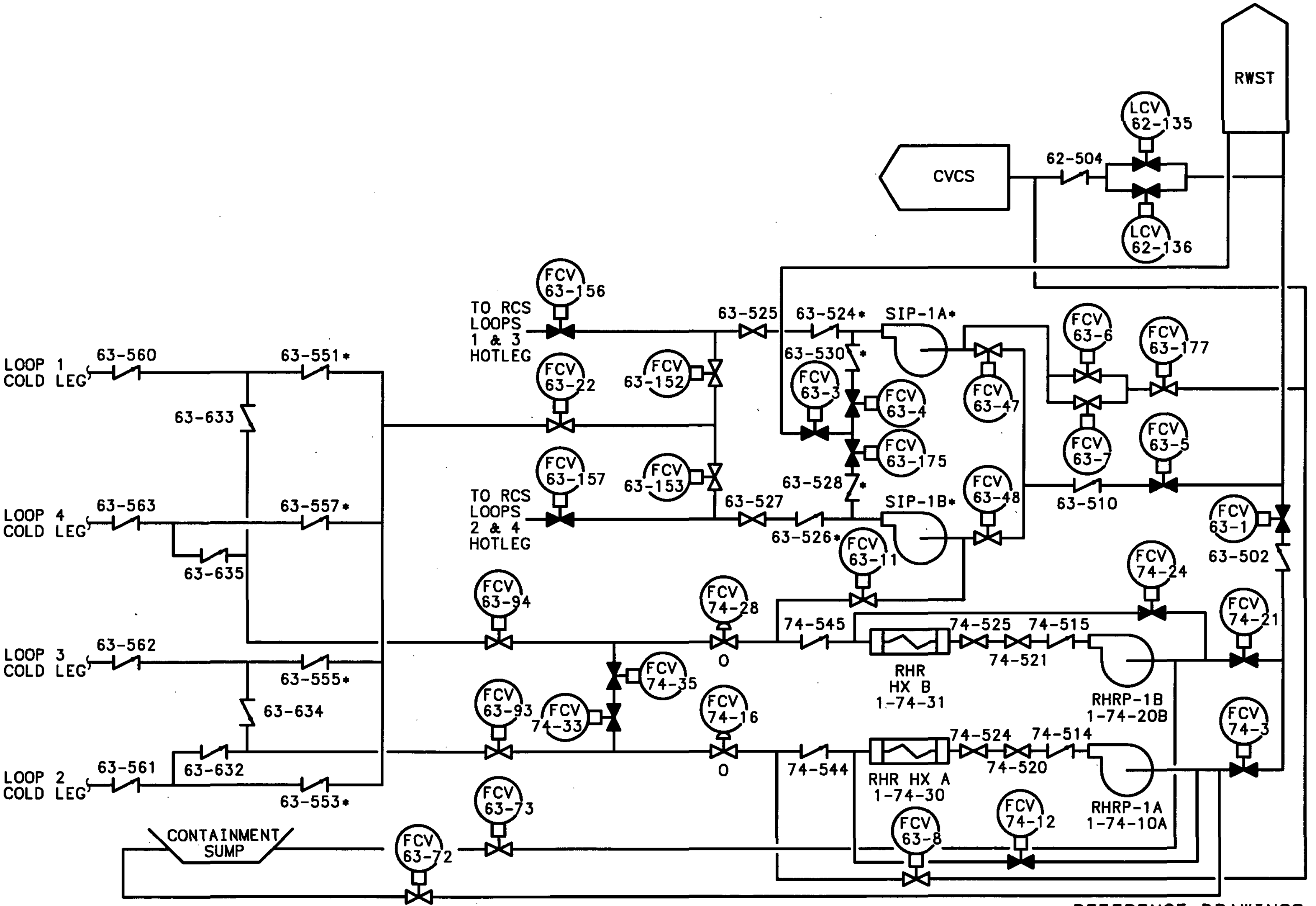


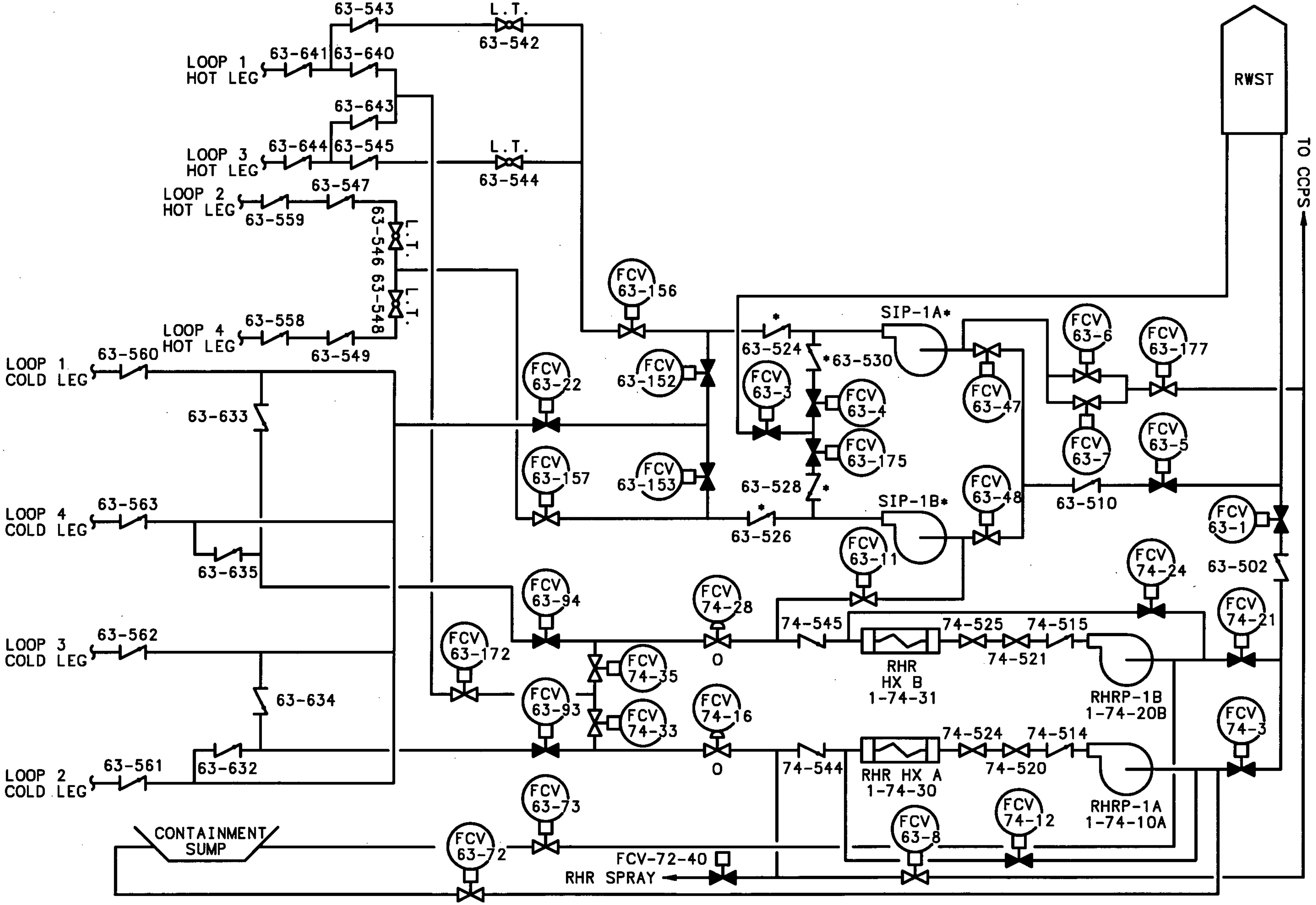
FIGURE 2
RHR ALIGNMENT FOR NORMAL COOLDOWN



* NOT MODELED IN THIS SYSTEM.

FIGURE 3
COLD LEG RECIRCULATION MODE

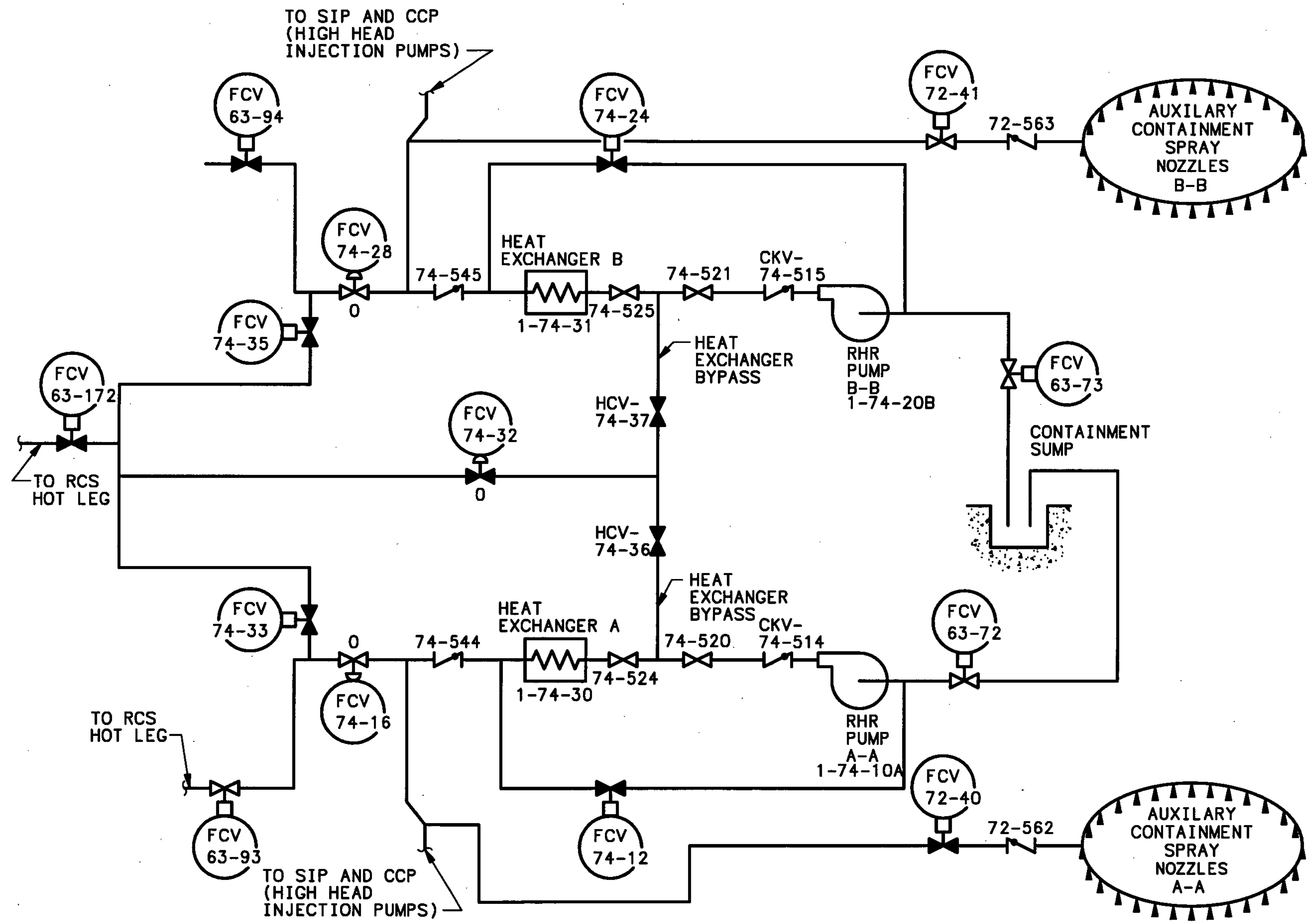
REFERENCE DRAWINGS:
1-47W810-1
1-47W811-1



* NOT MODELED IN THIS SYSTEM

FIGURE 4
HOT LEG RECIRCULATION

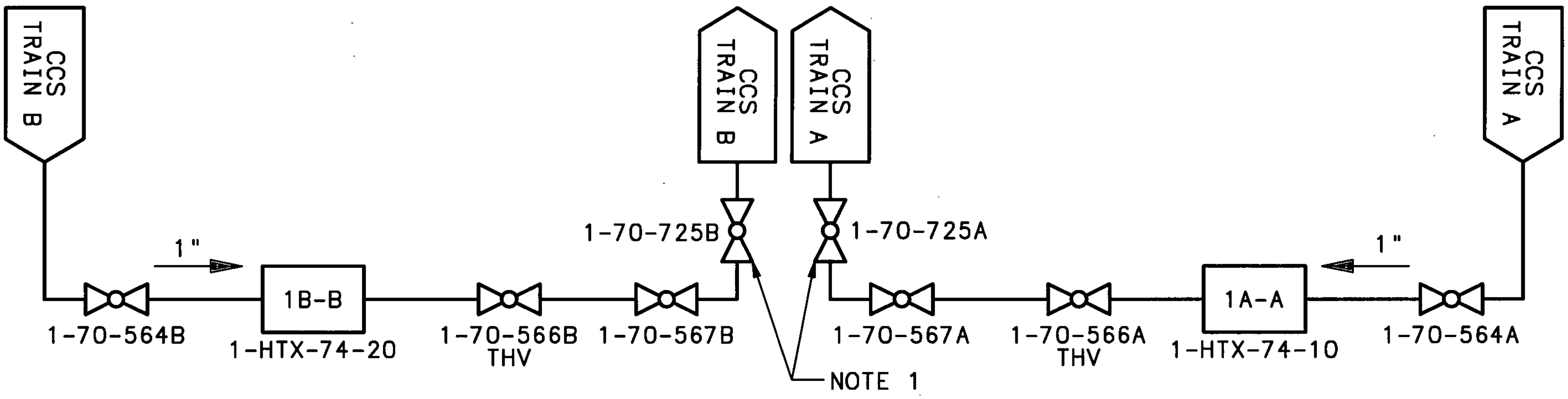
REFERENCE DRAWINGS:
1-47W810-1
1-47W811-1



NOTE: ONLY TRAIN B IS MODELED

FIGURE 5
RHR CONTAINMENT SPRAY

REFERENCE DRAWINGS:
1-47W810-1
1-47W811-1



REFERENCE DRAWING:
1-47W859-4

NOTE 1: COMMON TO CS AND RHR

FIGURE 6
RHR PUMP SEAL COOLING WATER

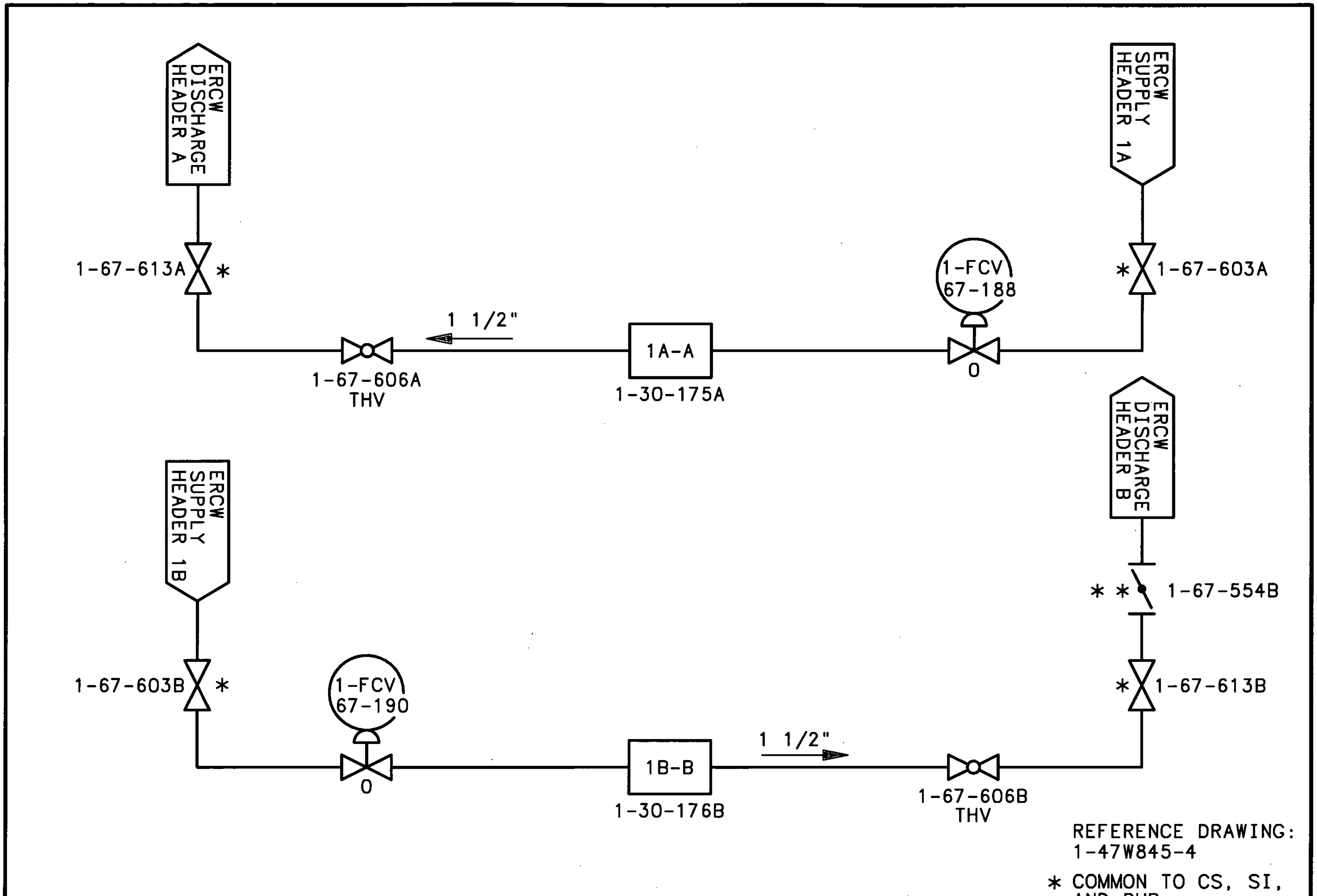
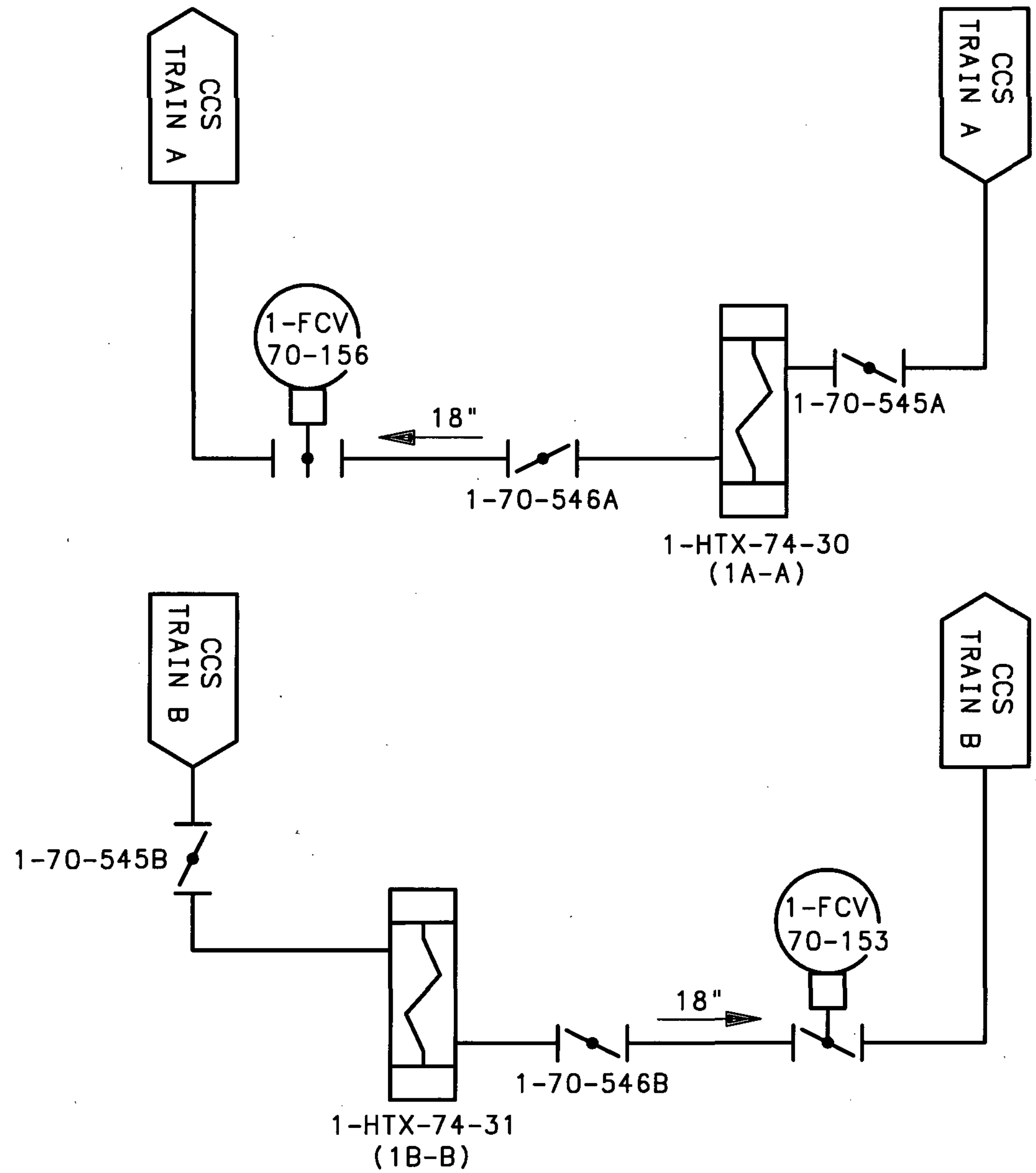
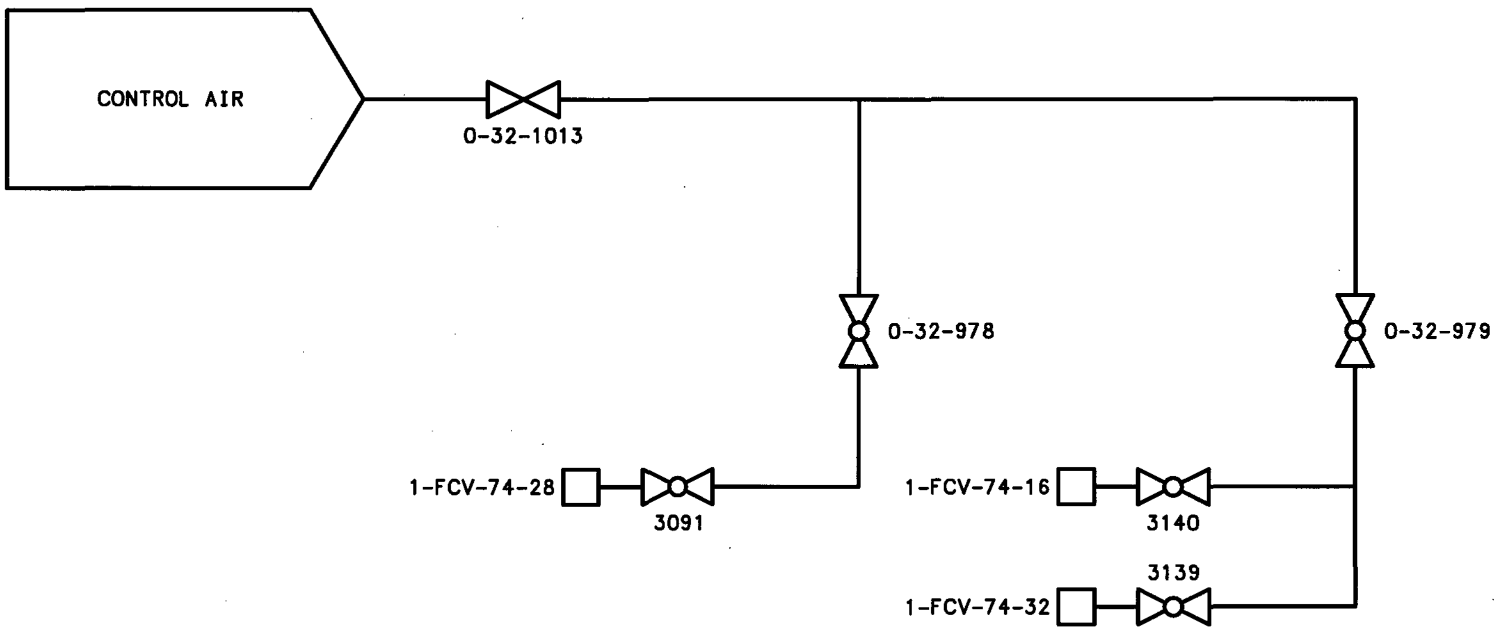


FIGURE 7
RHR PUMP ROOM COOLER



REFERENCE DRAWING:
1-47W859-4

FIGURE 8
RHR HEAT EXCHANGERS



REFERENCE DRAWING:
1-47W848-5

FIGURE 9
CONTROL AIR SYSTEM TO RHR

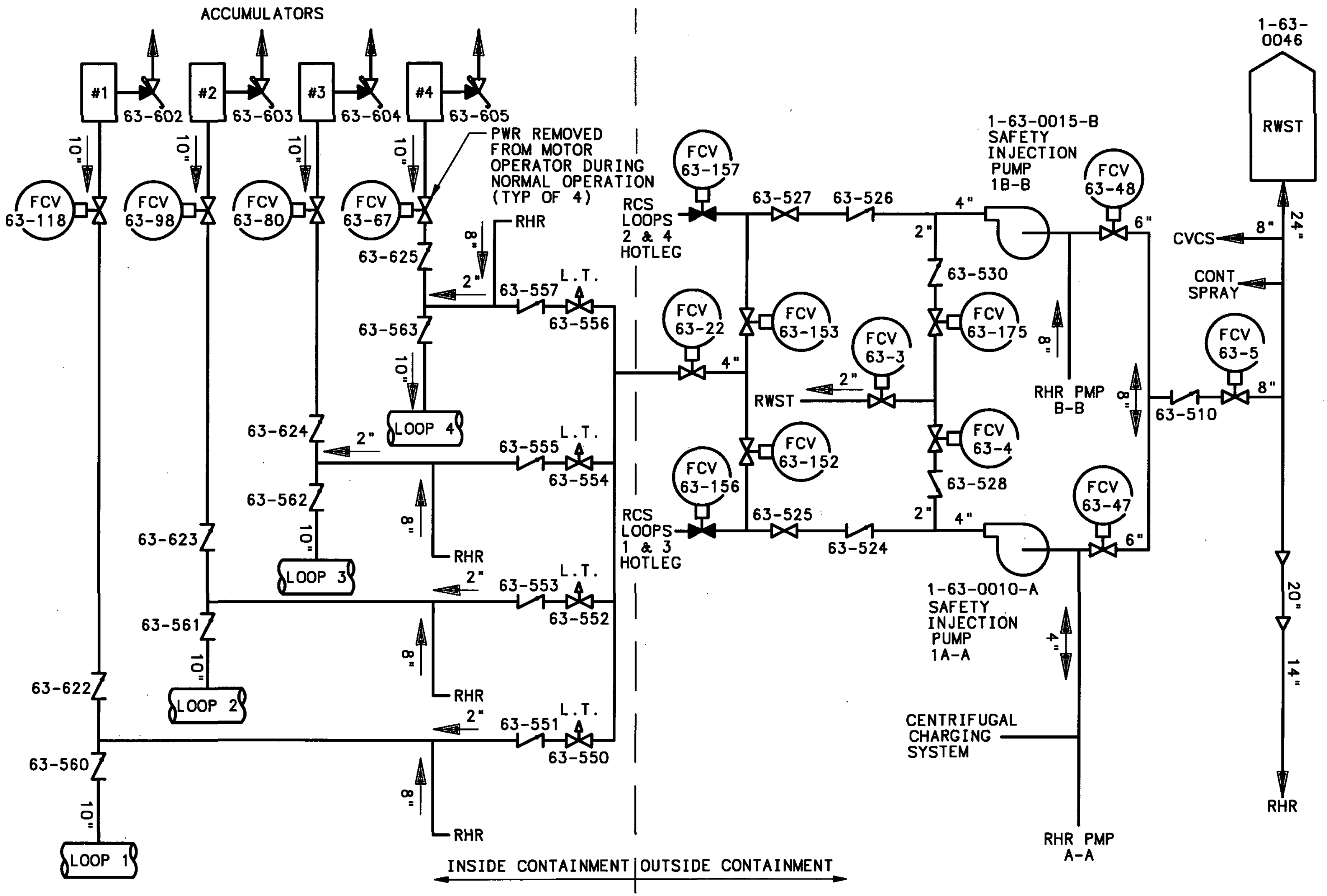
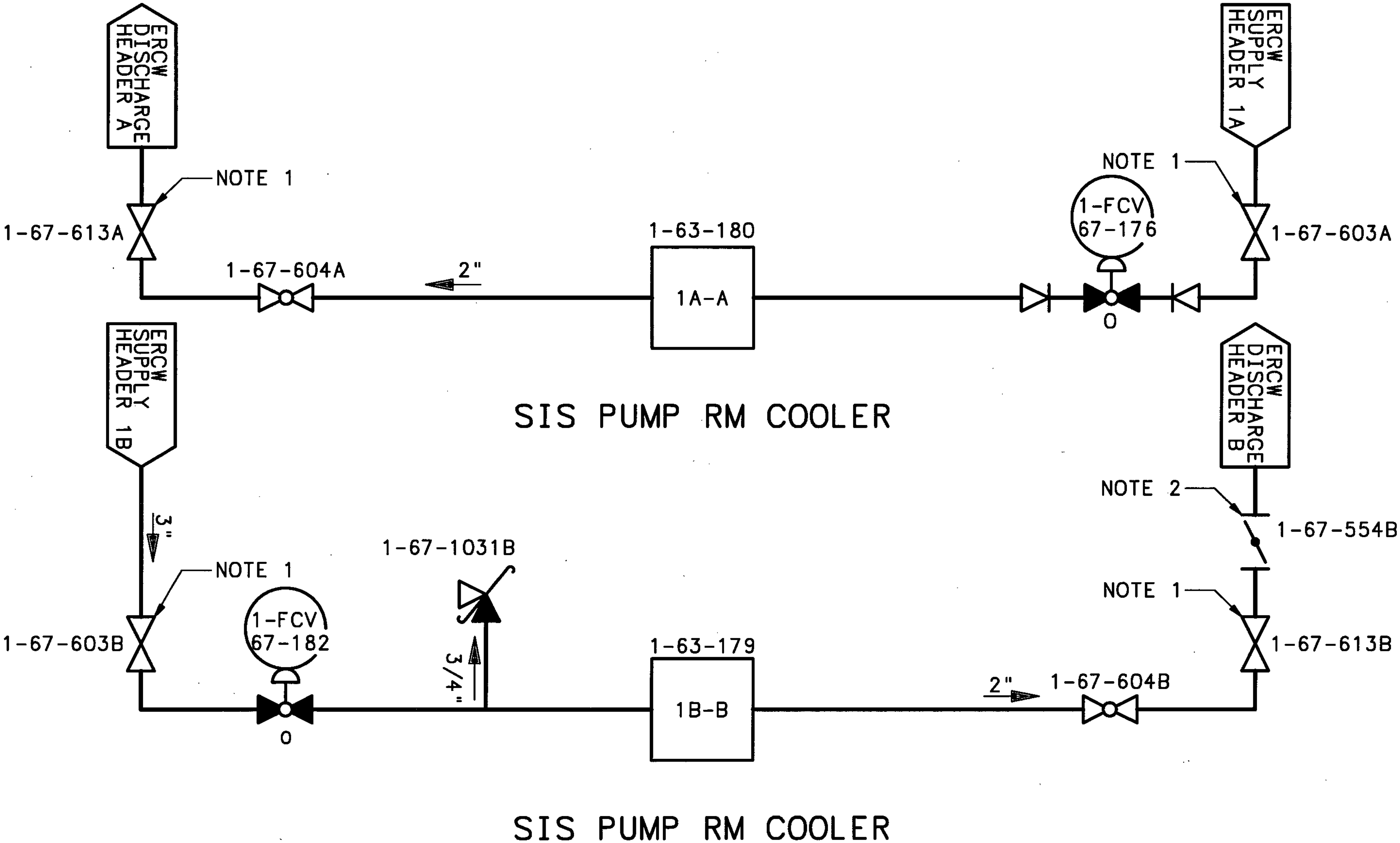


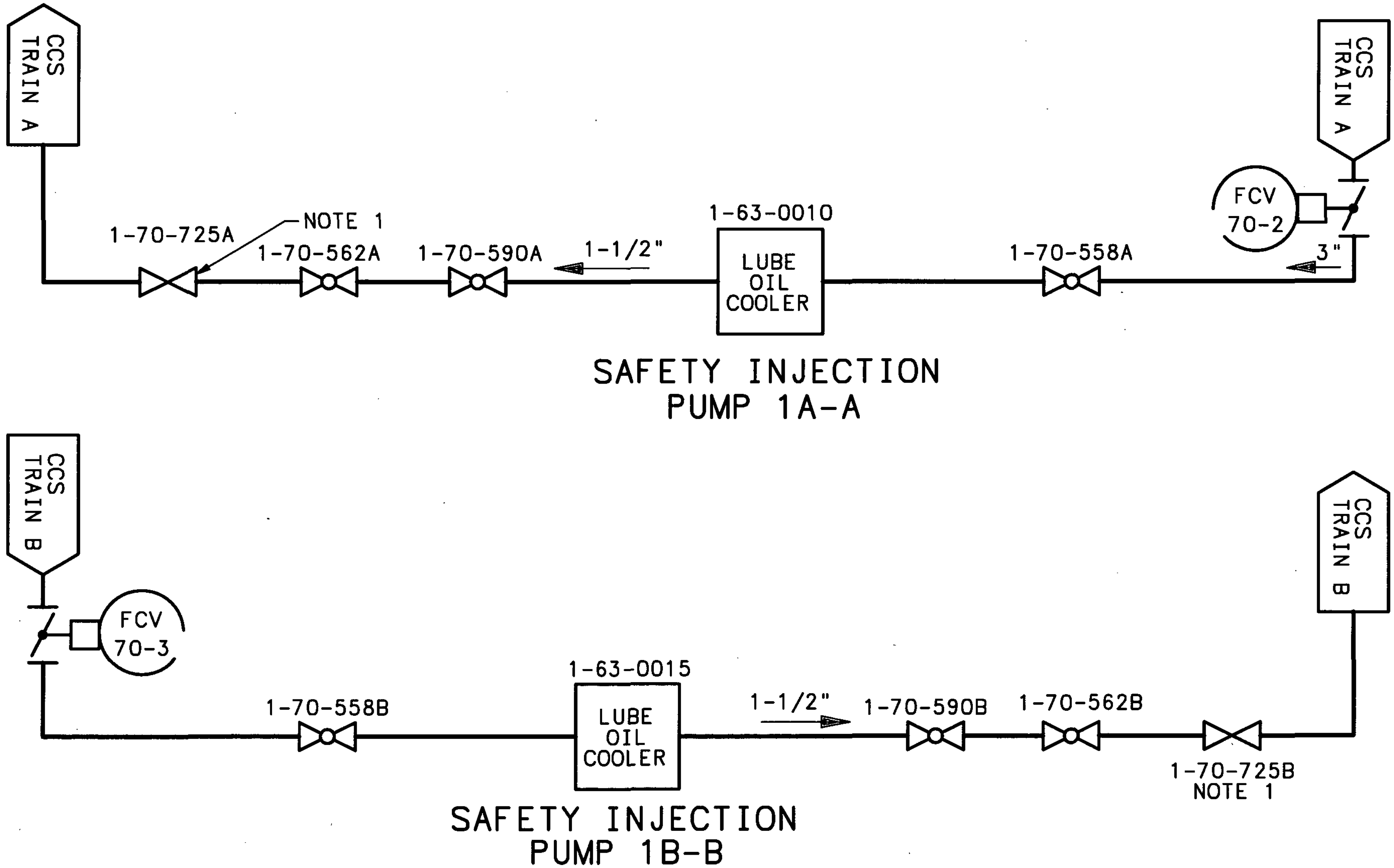
FIGURE 1
SAFETY INJECTION SYSTEM
"COLD LEG" INJECTION MODE



NOTE 1: COMMON TO CS, SI, AND RHR

NOTE 2: COMMON TO CVCS, SI, RHR, AND CS

FIGURE 2
SAFETY INJECTION SYSTEM
PUMP ROOM COOLING



NOTE 1:
COMMON TO CONTAINMENT SPRAY AND RHR

FIGURE 3
PUMP LUBE OIL COOLING

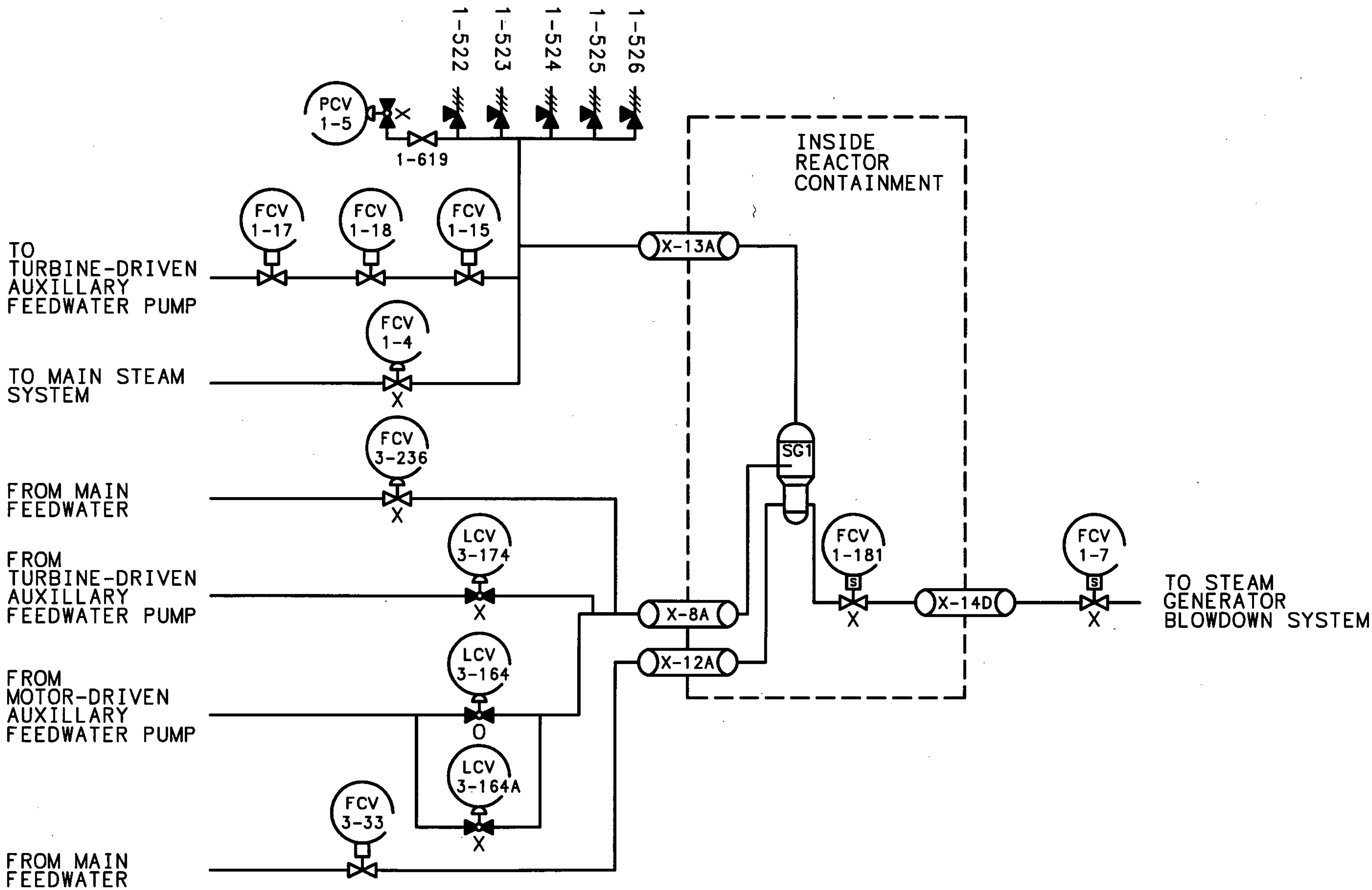


FIGURE 1
STEAM GENERATOR ISOLATION: SG 1

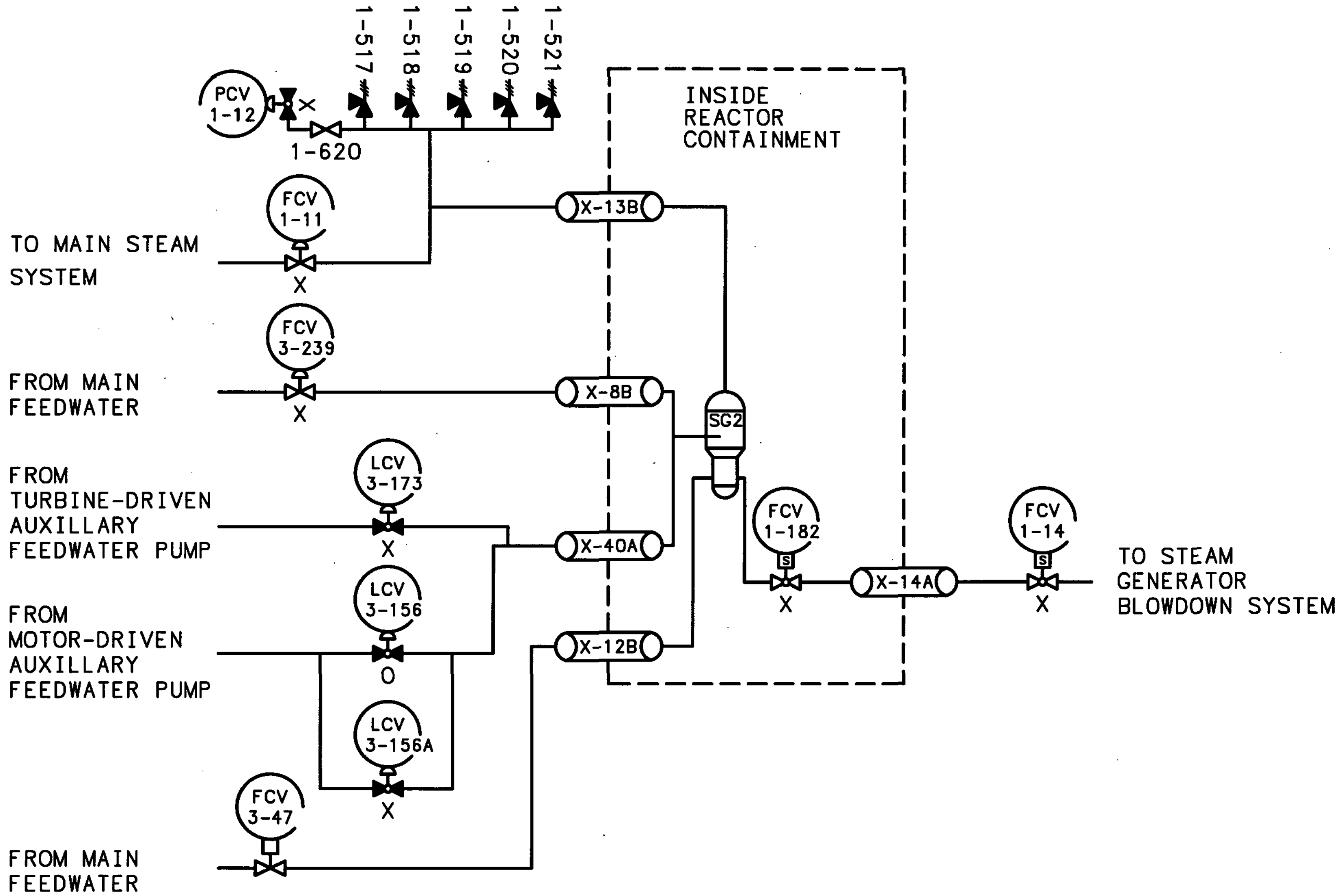


FIGURE 2
STEAM GENERATOR ISOLATION: SG 2

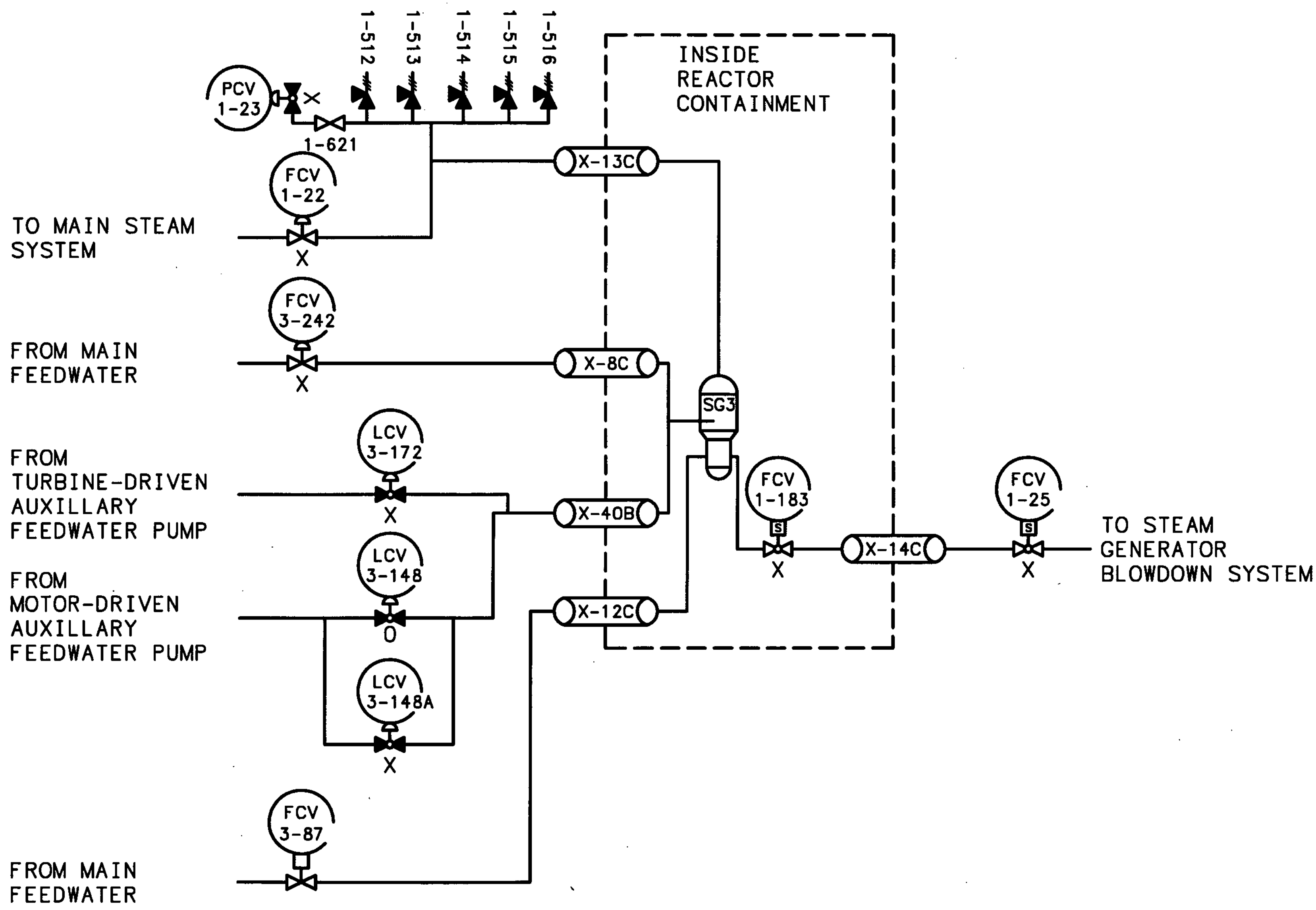


FIGURE 3
STEAM GENERATOR ISOLATION: SG 3

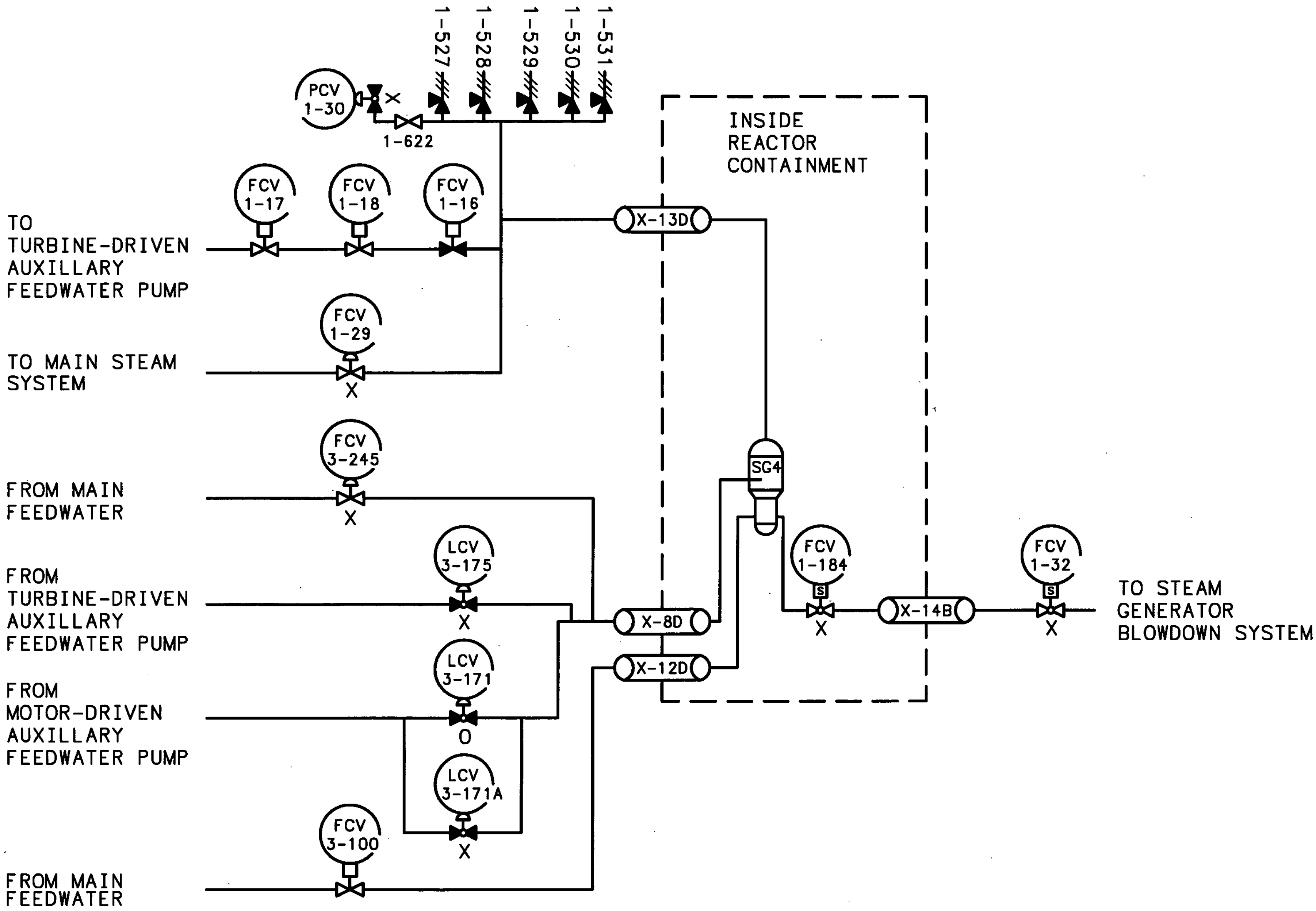


FIGURE 4
STEAM GENERATOR ISOLATION: SG 4

Enclosure 2

List of Commitments

1. Prior to fuel load, it will be confirmed that the Unit 2 Probabilistic Risk Assessment model matches the as-built, as-operated plant.