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10 CFR 50.54(f)

December 17, 2014

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261 / RENEWED LICENSE NO. DPR-23

Subject: H. B. Robinson Steam Electric Plant, Unit No. 2 Expedited Seismic Evaluation Process Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident

References:

1. NRC Letter, *Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident*, dated March 12, 2012

2. NEI Letter, *Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations*, dated April 9, 2013, ADAMS Accession No. ML13101A379

3. NRC Letter, *Electric Power Research Institute Report 3002000704, "Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," as an Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluations*, dated May 7, 2013, ADAMS Accession No. ML13106A331

Ladies and Gentlemen:

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued a 50.54(f) letter to all power reactor licensees and holders of construction permits in active or deferred status. Enclosure 1 of Reference 1 requested each addressee located in the Central and Eastern United States (CEUS) to submit a Seismic Hazard Evaluation and Screening Report within 1.5 years from the date of Reference 1.

In Reference 2, the Nuclear Energy Institute (NEI) requested NRC agreement to delay submittal of the final CEUS Seismic Hazard Evaluation and Screening Reports so that an update to the Electric Power Research Institute (EPRI) ground motion attenuation model could be completed and used to develop that information. NEI proposed that descriptions of subsurface materials and properties and base case velocity profiles be submitted to the NRC by September 12, 2013, with the remaining seismic hazard and screening information submitted by March 31, 2014. NRC agreed with that proposed path forward in Reference 3.

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Reference 1 requested that licensees provide interim evaluations and actions taken or planned to address the higher seismic hazard relative to the design basis, as appropriate, prior to completion of the risk evaluation. In accordance with the NRC endorsed guidance in Reference 3, the attached Expedited Seismic Evaluation Process Report for H. B. Robinson Steam Electric Plant, Unit No. 2 provides the information described in Section 7 of Reference 3 in accordance with the schedule identified in Reference 2.

This letter contains no new regulatory commitments.

If you have any questions or require additional information, please contact Richard Hightower, Manager, Nuclear Regulatory Affairs at (843)-857-1329.

I declare under the penalty of perjury that the foregoing is true and correct.

Executed on December 17, 2014.

Sincerely,

R. Michael Glover

R. Michael Glover
Site Vice President

RMG/shc

Enclosure: Expedited Seismic Evaluation Process Report for H. B. Robinson Steam Electric Plant, Unit No. 2

cc: Ms. M. C. Barillas, NRC Project Manager, NRR
Mr. K. M. Ellis, NRC Senior Resident Inspector
Mr. V. M. McCree, NRC Region II Administrator

Expedited Seismic Evaluation Process Report

For

H. B. Robinson Steam Electric Plant, Unit No. 2

EXPEDITED SEISMIC EVALUATION PROCESS REPORT

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EXECUTIVE SUMMARY

An Expedited Seismic Evaluation Process has been completed for the H.B. Robinson Steam Electric Plant site based on endorsed guidance outlined in Electric Power Research Institute (EPRI) 3002000704 (Reference 2). The work includes screening, equipment selection, development of the RLGM and in-structure demands, evaluating seismic capacity of components and development of High Confidence of Low Probability of Failure (HCLPF) calculations, and implementation of necessary plant modifications. HCLPF calculations revealed that Motor Control Center (MCC-A) required modification for the beyond design basis ground motion. Modifications have been developed and implemented for MCC-A and a similar cabinet, MCC-B. Seismic margin above 2X SSE was also added to a group of instrument racks (Hagan Racks) by validating the bolting integrity of the top braces. All items in the ESEL have seismic capacity that exceeds the demand of the RLGM. The ESEL has been updated to consider new equipment in FLEX strategy as outlined in the updated Overall Integrated Plan. The FLEX strategy was subjected to critical path analysis and all the items required under the ESEP guidelines are included in the ESEL list.

1.0 Purpose and Objective

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 10 CFR 50.54(f) letter on March 12, 2012 (Reference 1), requesting information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance. Depending on the comparison between the reevaluated seismic hazard and the current design basis, further risk assessment may be required. Assessment approaches acceptable to the staff include a seismic probabilistic risk assessment (SPRA), or a seismic margin assessment (SMA). Based upon the assessment results, the NRC staff will determine whether additional regulatory actions are necessary.

This report describes the Expedited Seismic Evaluation Process (ESEP) undertaken for H.B. Robinson Steam Electric Plant (RNP). The intent of the ESEP is to perform an interim action in response to the NRC's 50.54(f) letter (Reference 1) to demonstrate seismic margin through a review of a subset of the plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events.

The ESEP is implemented using the methodologies in the NRC endorsed guidance in EPRI 3002000704, *Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic* (Reference 2).

The objective of this report is to provide summary information describing the ESEP evaluations and results. The level of detail provided in the report is intended to enable NRC to understand the inputs used, the evaluations performed, and the decisions made as a result of the interim evaluations.

2.0 Brief Summary of the FLEX Seismic Implementation Strategies

The H.B. Robinson Steam Electric Plant FLEX strategies for Reactor Core Cooling and Heat Removal, Reactor Inventory Control/Long-term Subcriticality, and Containment Function are summarized below. The FLEX flow path is shown in Attachment D. The summary is derived from the H.B. Robinson Steam Electric Plant Overall Integrated Plan (OIP) in Response to the March 12, 2012, Commission Order EA-12-049 (Reference 3), as supplemented by six-month updates (References 30, 31, and 32). Note that the H.B. Robinson Overall Integrated Plan (as amended in 6 month updates) is based on Engineering Change (EC) 88926 (Reference 33).

Reactor Core Cooling and Heat Removal

NEI 12-06, Diverse and Flexible Coping Strategies (FLEX) Implementation Guideline, Revision 0 (Reference 34), requires that Auxiliary Feedwater (AFW) cooling be available to provide secondary makeup sufficient to maintain or restore Steam Generator (SG) level with installed equipment to the greatest extent possible. Beyond the use of installed equipment, steam generators must be able to be depressurized in order to support makeup via portable pumps. Multiple and diverse connection points for the portable pumps must be provided and cooling water must be available indefinitely. Refer to Attachment B (Reactor Coolant System Cooling Strategies) for depiction of the following discussion.

The H.B Robinson Steam Electric Plant FLEX strategies require that the AFW be in operation within 61 minutes of event initiation. With the loss of AC power, a minimum of one steam supply valve (MS-V1-8A, MS-V1-8B, or MS-V1-8C) to Steam Driven Auxiliary Feedwater Pump (SDAFWP) and one AFW valve (AFW-V2-14A, AFW-V2-14B, AFW-V2-14C) to the steam generators must be manually operated. These required valves are all located in seismic Class 1 bay of the Turbine Building.

Additional portable backup for Steam Generator makeup is required per Section 3.2.2(13) of NEI 12-06. The H.B. Robinson Steam Electric Plant has two strategies for portable backup. The first strategy developed to satisfy this requirement is staging of two (2) intermediate pressure pumps (300 gpm at pressure of 1,000 psig) for all seismic events as described in detail below. The second strategy developed to satisfy the condition of Section 3.2.2(13) of NEI 12-06 is to store a Hale pumper in a seismically robust Permanent FLEX Storage Building (PFSB). This strategy will involve the use of the same primary and alternate connections described in the following paragraph, and will require SG depressurization.

The two (2) pre-staged portable pumps (300 gpm at 1,000 psig) eliminate the need to depressurize the Steam Generators in the event the backup AFW feed capability is needed due to an AFW interruption early in the ELAP transient as a result of seismic event. Either of the portable pumps can take suction from a variety of plant sources (described below) and can be tied directly into the auxiliary feedwater system. Engineering Change 95266, Isolation Valves And Connection For AFW - FUKUSHIMA-Admin (Reference 48) was developed to add a FLEX tee connection (AFW-166) to the SDAFWP discharge at AFW-121 (see Figure 2-1). Access to this primary connection is through the seismically qualified Turbine Building Class 1 bay. Engineering Change 90623, New Pipe Tee And Standard Connection For NTTF 4.2 (FLEX) (Reference 47) develops an alternate mechanical FLEX connection (AFW-165) inside the MDAFWP room on line 4-AFW-23 and upstream of AFW-54 (See Figure 2-2). EC90623 will be implemented during Refueling Outage, RO229. The MDAFW room is housed in the seismic Class 1 Reactor Auxiliary Building (RAB).

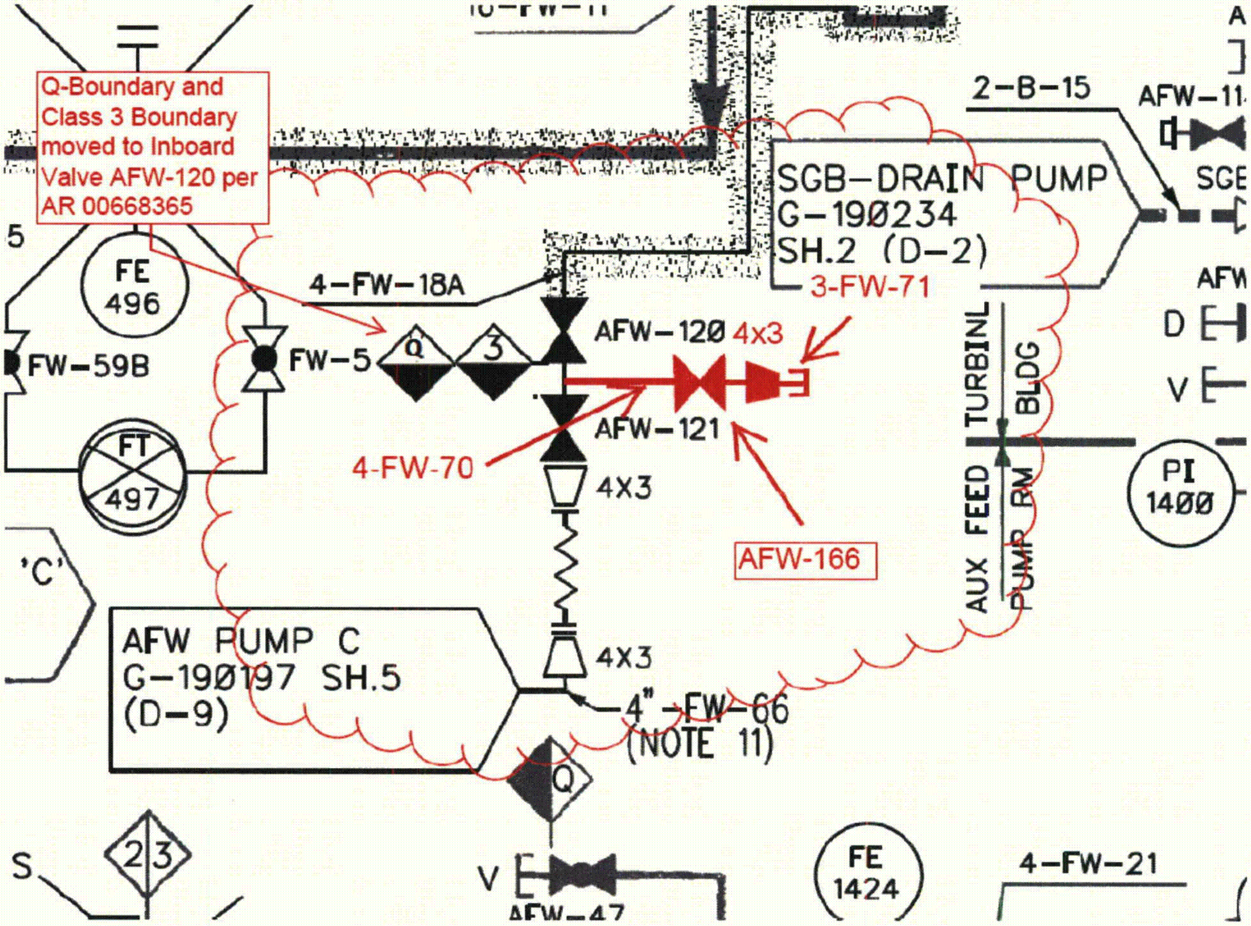


Figure 2-1: Mark-Up Showing Addition of FLEX Tee Connection (AFW-166) to SDAFW Discharge at AFW-121

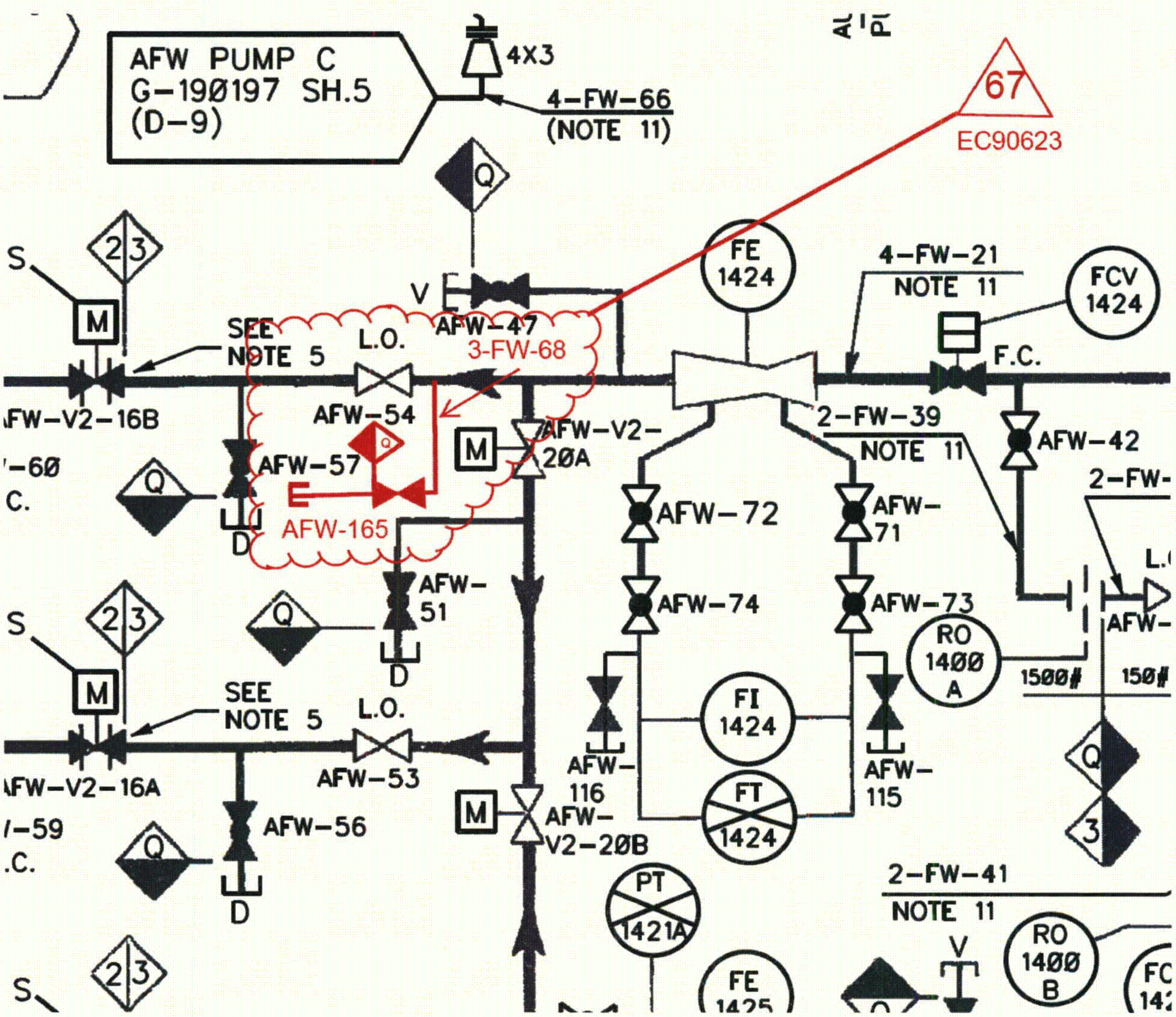


Figure 2-2: Mark-Up Showing Alternate Mechanical FLEX Connection (AFW-165) inside the MDAFWP Room on Line 4-AFW-23 and Upstream of AFW-54

There are several sources for sustained cooling water supply. The primary source of AFW inventory is the seismically qualified condensate storage tank (CST) and its level instrumentation. The CST is seismically robust and is the installed source of AFW to the SDAFWP. However, the CST inventory is not sufficient for indefinite coping (mission time is approximately 4 hours using the SDAFWP). A secondary source of AFW inventory is the "Tank Farm" (portable pump) inside the protected area that supplies the two pre-staged portable pumps (each with capacity of 300 gpm and 1,000 psig pressure as noted in the seismic strategy above). This source has a capacity of approximately 120,000 gallons and 10 hours of mission time using a pre-staged portable pump.

The only other assured source of water is the Ultimate Heat Sink (Lake Robinson) which per restrictions outlined in NEI 12-06 can only be accessed using portable equipment (assumes normal

access to the ultimate heat sink is lost). Given these limitations, one Phase 2/3 seismic strategy is to provide an indefinite supply of water to the CST and the SDAFWP by staging a portable diesel pumper at Lake Robinson with hoses routed to the CST. EC 90622, Standard Piping Connections For NTTF 4.2 (FLEX) (Reference 43) adds a FLEX connection at valve C-66 to provide an indefinite water supply to the CST. This can be accomplished during the initial CST/Tank Farm mission time of 14 hours.

The H.B. Robinson Steam Electric Plant has developed several options for the Steam Generator depressurization capability. The Steam Generator Power Operated Relief Valves (PORVs) are normally operated using the Instrument Air System or with backup Nitrogen System and aligned using Attachment 2 of EOP-ECA-0.0 (Reference 35). However, neither the primary Instrument Air nor the backup Nitrogen System are seismically qualified. Therefore, the primary Instrument Air and the backup Nitrogen System cannot be relied upon during or after seismic events. The Main Steam Safety Valves are an alternate option to depressurize the Steam Generators but this option is not recommended per the PA-PSC-0965, PWROG Core Cooling Position Paper (Reference 37) and WCAP-17601-P, Revision 1, Reactor Coolant System response to Extended Loss of AC Power Event for Westinghouse, Combustion Engineering, and Babcock & Wilcox NSSS Designs for Phase Boration, August 2012 (Reference 36), which state that remaining on the Main Steam Safety Valves for an extended period may lead to failure of the valve(s) which subsequently will cause excessive and uncontrolled RCS cooldown.

Current strategy is to align portable nitrogen tanks to the Steam Generator PORV header using Attachment 1 (Connecting Emergency Pressure Source to Operate SG PORVS) or Attachment 2 (S/G Manual Depressurization) of RNP procedure EDMG-004, Steam Generators (Reference 38). In addition to the SG PORV capabilities recommended in Reference 37, the H.B. Robinson Steam Electric Plant has also developed a strategy to cooldown the RCS using the main steam line isolation valve bypass lines. The strategy is detailed in Section 3.27 (Cooldown Using MSIV Bypass Lines) of calculation RNP-M/MECH-1712, Appendix R Mechanical Basis (Reference 39). This capability results in a cooldown rate of 83°/hr which bounds the recommended Westinghouse cooldown rate of 75°/hr described in Reference 37.

After initiation of depressurization, it is desirable to isolate the Safety Injection (SI) Accumulators in order to prevent injection of nitrogen into the RCS which will impede natural circulation cooldown. During an ELAP, power to the SI Accumulator isolation valves is lost. Although, the isolation valves can be operated manually, they are located inside the Containment Building and it is undesirable to perform this operation at this time due to personnel safety. The valves are powered by MCC 5 and MCC 6 and will be re-powered via Emergency Buses E1 and E2 with portable diesel generators staged in the seismic Class 1 Reactor Auxiliary Building (Drumming Room) for re-powering the A and B Battery Chargers (see EC 90617 [Reference 40]). DB-50 Bus Feed Adapters can be installed in each of the Emergency Buses E1 and E2 and will be connected to the output of the Diesel Generators. As part of the Phase 2 strategy, Steam Generator pressure will be maintained above the pressure corresponding to the SI Accumulator injection (240 psig) until the SI Accumulator isolation valves are closed using FLEX Support Guideline (FSG) 10, Passive RCS Injection Isolation (Reference 41).

Reactor Inventory Control/Long-Term Subcriticality

Refer to Attachment C (Reactor Coolant System Boration and Makeup Strategies) for a depiction of the following discussion. There is no installed means of providing borated makeup following an ELAP. The primary method of boration and inventory control is the use of portable high pressure and low volume pump directly connected to the Charging Lines or Safety Injection Headers from the Refueling Water Storage Tank (RWST) or a portable tanker containing borated water (see EC95216, NTTF 2.1 Interim Action RCS Injection [Reference 42]). The RWST is seismically

designed and will remain operational during and after a design basis seismic event. The makeup capacity of the portable pump is 60 gpm at a pressure of 2,000 psig which is adequate for the bounding analysis in WCAP-1760-P (Reference 36). Phase 3 inventory control will be accomplished using the same portable Phase 2 boration/makeup strategy. Portable high pressure pumping and portable tanker capability will be stored in the PFSB to support this strategy.

EC 90622 (Reference 43) adds a FLEX connection to the exposed end downstream of the normally locked closed drain valve (SI-837) located at the base of the RWST to access this borated water if it available. This portable strategy will deliver borated water to the RCS through valves CVC-121A/B (primary) or SI-888P/S (alternate).

Containment Function

Calculation RNP-M/MECH-1877, RNP Extended Loss of AC Power (ELAP) Containment Response (Reference 45) was developed to determine the containment temperature and pressure response assuming an ELAP and a trip from 100% reactor power at 100 days into the cycle. Results in Reference 45 indicate that the Containment Building design limits for temperature and pressure will not be challenged in the first 43 days following the event. This analysis assumes that: (1) no action is taken to cool, spray, or vent the containment; and (2) low leakage RCP seals are installed. Therefore, Phase 1 and 2 strategies are not required. There is sufficient time and resources in Phase 3 to assemble a strategy using the National Safer Response Center (NSRC) pumpers and generators, prefabricated electrical connections, and prefabricated SW connections that will be stored in the PFSB. FSG-12, Alternate Containment Cooling (Reference 46) provides instructions for several existing strategies including external containment cooling which does not require use of any plant system. These particular activities will be determined and directed by the Emergency Response Organization (Technical Support Center) based on the effects of the Beyond Design Basis External Event (BDBEE) and the state of existing equipment.

Instrumentation

Instrumentation channels that are powered by station batteries will be lost upon depletion of the batteries. FLEX strategies to improve battery coping occur by extending Phase 1. Phase 1 is extended by strategic load shedding followed by additional deep load shedding in the first hour of the event to extend battery coping times to 3.25 – 3.75 hours. Phases 2 and 3 battery coping require portable diesel generators to power the vital battery chargers. Two FLEX diesel generators will be mounted in their deployed positions near the battery chargers and within the Reactor Auxiliary Building. Each generator will be sized to power two vital battery chargers, room air supply and exhaust fans, and safety injection accumulator isolation valves. Electrical cables and pre-installed connectors will be routed from the FLEX diesel generators to the battery room for quick connection of the cables to each of the battery chargers. The primary strategy is to power the A and B vital battery chargers from one or both of the pre-staged FLEX generators. The alternate is to power the A-1 and B-1 vital battery chargers from one or both of the pre-staged FLEX generators. See Reference 40 for complete details of this strategy.

3.0 Equipment Selection Process and ESEL

The selection of equipment for the Expedited Seismic Equipment List (ESEL) followed the guidelines of EPRI 3002000704. The complete ESEL for H. B. Robinson Unit 2 is presented in Attachment A.

3.1 Equipment Selection Process and ESEL

The selection of equipment to be included on the ESEL was based on installed plant equipment credited in the FLEX strategies during Phase 1, 2 and 3 mitigation of a BDBEE as described in the H.B. Robinson Steam Electric Plant OIP (Reference 3) in response to the March 12, 2012 Commission Order EA-12-049 as revised in References 30 through 32.

The scope of "installed plant equipment" includes equipment relied upon for the FLEX strategies to sustain the critical functions of core cooling and containment integrity consistent with Reference 3 and References 30 through 32. FLEX recovery actions are excluded from the ESEP scope per EPRI 3002000704. The overall list of planned FLEX modifications and the scope for consideration herein is limited to those required to support core cooling, reactor coolant inventory and subcriticality, and containment integrity functions. Portable and pre-staged FLEX equipment (not permanently installed) are excluded from the ESEL per EPRI 3002000704.

The ESEL component selection followed the EPRI guidance outlined in Section 3.2 of EPRI 3002000704.

1. The scope of components is limited to that required to accomplish the core cooling and containment safety functions identified in Table 3-2 of EPRI 3002000704. The instrumentation monitoring requirements for core cooling/containment safety functions are limited to those outlined in the EPRI 3002000704 guidance, and are a subset of those outlined in the H.B. Robinson Steam Electric Plant OIP and as revised in the first , second and third six-month status reports.
2. The scope of components is limited to installed plant equipment, and FLEX connections necessary to implement the H.B. Robinson Steam Electric Plant OIP (Reference 3) in response to the March 12, 2012 Commission Order EA-12-049 and as revised in References 30 through 32. and as described in Section 2.
3. The scope of components assumes the credited FLEX connection modifications are implemented, and are limited to those required to support a single FLEX success path (i.e., either "Primary" or "Back-up/Alternate").
4. The "Primary" FLEX success path is to be specified. Selection of the "Back-up/Alternate" FLEX success path must be justified.
5. Phase 3 coping strategies are included in the ESEP scope, whereas recovery strategies are excluded.
6. Structures, systems, and components excluded per the EPRI 3002000704 (Reference 2) guidance are:
   - Structures (e.g. Reactor Containment Building, Reactor Auxiliary Building, etc.)
   - Piping, cabling, conduit, HVAC, and their supports.
   - Manual valves and rupture disks .
   - Power-operated valves not required to change state as part of the FLEX mitigation strategies.

- Nuclear steam supply system components (e.g. reactor pressure vessel and internals, reactor coolant pumps and seals, etc.)

7. For cases in which neither train was specified as a primary or back-up strategy, then only one train component (generally 'A' train) is included in the ESEL.

3.1.1 ESEL Development

The ESEL was developed by reviewing the H.B. Robinson Steam Electric Plant OIP (Reference 3) and revisions in three subsequent six-month status reports to determine the major equipment involved in the FLEX strategies. Further reviews of plant drawings (e.g., Process and Instrumentation Diagrams (P&IDs)(EC92103R0, Attachment Z03R0 Mechanical Documents [Reference 49]), and Electrical One Line Diagrams (EC92103R0, Attachment Z05R0 Electrical Documents [Reference 50]) were performed to identify the boundaries of the flowpaths to be used in the FLEX strategies and to identify specific components in the flowpaths needed to support implementation of the FLEX strategies. Boundaries were established at an electrical or mechanical isolation device (e.g., isolation amplifier, valve, etc.) in branch circuits / branch lines off the defined strategy electrical or fluid flowpath. P&IDs were the primary reference documents used to identify mechanical components and instrumentation. The flow paths used for FLEX strategies were selected and specific components were identified using detailed equipment and instrument drawings, piping isometrics, electrical schematics and one-line drawings, system descriptions, design basis documents, etc., as necessary.

3.1.2 Power Operated Valves

Page 3-3 of EPRI 3002000704 notes that power operated valves not required to change state are excluded from the ESEL. Page 3-2 also notes that "functional failure modes of electrical and mechanical portions of the installed Phase 1 equipment should be considered (e.g. AFW trips)." To address this concern, the following guidance is applied in the H.B. Robinson Steam Electric Plant ESEL for functional failure modes associated with power operated valves:

- Power operated valves not required to change state as part of the FLEX mitigation strategies were not included on the ESEL. The seismic event also causes the ELAP event; therefore, the valves are incapable of spurious operation as they would be de-energized.
- Power operated valves not required to change state as part of the FLEX mitigation strategies during Phase 1, and are re-energized and operated during subsequent Phase 2 and 3 strategies, were not evaluated for spurious valve operation as the seismic event that caused the ELAP has passed before the valves are re-powered.

3.1.3 Pull Boxes

Pull boxes were deemed unnecessary to add to the ESEL as these components provide completely passive locations for pulling or installing cables. No breaks or connections in the cabling are included in pull boxes. Pull boxes were considered part of conduit and cabling, which are excluded in accordance with EPRI 3002000704.

3.1.4 Termination Cabinets

Termination cabinets, including cabinets necessary for FLEX Phase 2 and Phase 3 connections, provide consolidated locations for permanently connecting multiple cables. The termination cabinets and the internal connections provide a completely passive function; however, the cabinets are included in the ESEL to ensure industry knowledge on panel/anchorage failure vulnerabilities is addressed.

3.1.5 Critical Instrumentation Indicators

Critical indicators and recorders are typically physically located on panels/cabinets and are included as separate components.

3.1.6 Phase 2 and Phase 3 Piping Connections

Item 2 in Section 3.1 above notes that the scope of equipment in the ESEL includes "... FLEX connections necessary to implement the H.B. Robinson Steam Electric Plant OIP as described in Section 2." Item 3 in Section 3.1 also notes that "The scope of components assumes the credited FLEX connection modifications are implemented, and are limited to those required to support a single FLEX success path (i.e., either "Primary" or "Back-up/Alternate")."

Item 6 in Section 3.1 above goes on to explain that "Piping, cabling, conduit, HVAC, and their supports" are excluded from the ESEL scope in accordance with EPRI 3002000704.

Therefore, piping and pipe supports associated with FLEX Phase 2 and Phase 3 connections are excluded from the scope of the ESEP evaluation. However, any active valves in FLEX Phase 2 and Phase 3 connection flow path are included in the ESEL.

3.2 Justification for Use of Equipment That Is Not The Primary Means for FLEX Implementation

In accordance with EPRI 3002000704, the H.B. Robinson Steam Electric Plant used equipment that is the primary means of implementing FLEX strategy. The complete ESEL for the H.B. Robinson Steam Electric Plant is presented in Attachment A.

4.0 Ground Motion Response Spectrum (GMRS)

4.1 Plot of GMRS Submitted by the H.B. Robinson Steam Electric Plant

Following completion of the seismic hazard re-evaluation as requested in Reference 1, the NRC 10 CFR 50.54(f) letter, a screening process is needed to determine if an interim seismic risk evaluation like the EPRI ESEP is required. The screening GMRS was determined with control point seismic hazard re-evaluation. In accordance with the 50.54(f) letter and following the guidance in EPRI Screening, Prioritization, and Implementation Details (SPID) (Reference 15), Probabilistic Seismic Hazard Analysis (PSHA) was performed using the 2012 CEUS Seismic Source Characterization for Nuclear Facilities (Reference 20), a Regional Seismic Catalog Correction (Reference 61), and updated EPRI Ground Motion Model (GMM) for the CEUS (Reference 21). Development of the H.B. Robinson Steam Electric Plant Ground Motion Response Spectra (GMRS) is documented in References 4 and 62. The GMRS and Uniform Hazard Response Spectra (UHRS) are tabulated in Table 4-1 and then compared in Figure 4-1 with the 5% damped horizontal SSE. Note that additional seismic hazard analysis and GMRS development is underway for H.B. Robinson Steam Electric Plant to support completion of the seismic probabilistic risk analysis. In the analysis, newly acquired geophysical testing results are being used to update the site response analysis. The results of the screening evaluation discussed will not change as a result of the newly acquired geophysical testing. These new geophysical testing data allow for a more accurate representation of seismic hazard and seismic probabilistic risk assessment by eliminating a significant source of uncertainty.

Table 4-1: GMRS and UHRS for the H.B. Robinson Steam Electric Plant

Freq. (Hz)	10^{-4} UHRS (g)	10^{-5} UHRS (g)	GMRS
100	4.20E-01	9.17E-01	4.71E-01
90	4.23E-01	9.31E-01	4.77E-01
80	4.27E-01	9.48E-01	4.85E-01
70	4.35E-01	9.73E-01	4.97E-01
60	4.54E-01	1.02E+00	5.19E-01
50	4.98E-01	1.11E+00	5.66E-01
40	5.74E-01	1.25E+00	6.43E-01
35	6.21E-01	1.35E+00	6.95E-01
30	6.63E-01	1.46E+00	7.50E-01
25	7.23E-01	1.61E+00	8.21E-01
20	7.92E-01	1.75E+00	8.97E-01
15	8.09E-01	1.82E+00	9.27E-01
12.5	8.35E-01	1.82E+00	9.36E-01
10	8.52E-01	1.86E+00	9.55E-01
9	8.40E-01	1.84E+00	9.42E-01
8	8.58E-01	1.84E+00	9.49E-01
7	8.98E-01	1.92E+00	9.88E-01
6	8.87E-01	1.95E+00	9.99E-01
5	8.57E-01	1.87E+00	9.61E-01
4	8.40E-01	1.83E+00	9.39E-01
3.5	7.71E-01	1.76E+00	8.94E-01
3	6.79E-01	1.59E+00	8.04E-01
2.5	6.08E-01	1.38E+00	7.04E-01
2	5.37E-01	1.30E+00	6.52E-01
1.5	3.97E-01	1.05E+00	5.20E-01
1.25	3.23E-01	8.58E-01	4.23E-01
1	2.26E-01	6.44E-01	3.13E-01
0.9	1.87E-01	5.52E-01	2.67E-01
0.8	1.56E-01	4.69E-01	2.26E-01
0.7	1.31E-01	3.95E-01	1.90E-01
0.6	1.10E-01	3.25E-01	1.57E-01
0.5	8.86E-02	2.51E-01	1.22E-01
0.4	7.09E-02	2.01E-01	9.79E-02
0.35	6.20E-02	1.76E-01	8.57E-02
0.3	5.32E-02	1.51E-01	7.34E-02
0.25	4.43E-02	1.26E-01	6.12E-02
0.2	3.55E-02	1.00E-01	4.90E-02
0.15	2.66E-02	7.54E-02	3.67E-02
0.125	2.22E-02	6.28E-02	3.06E-02
0.1	1.77E-02	5.02E-02	2.45E-02

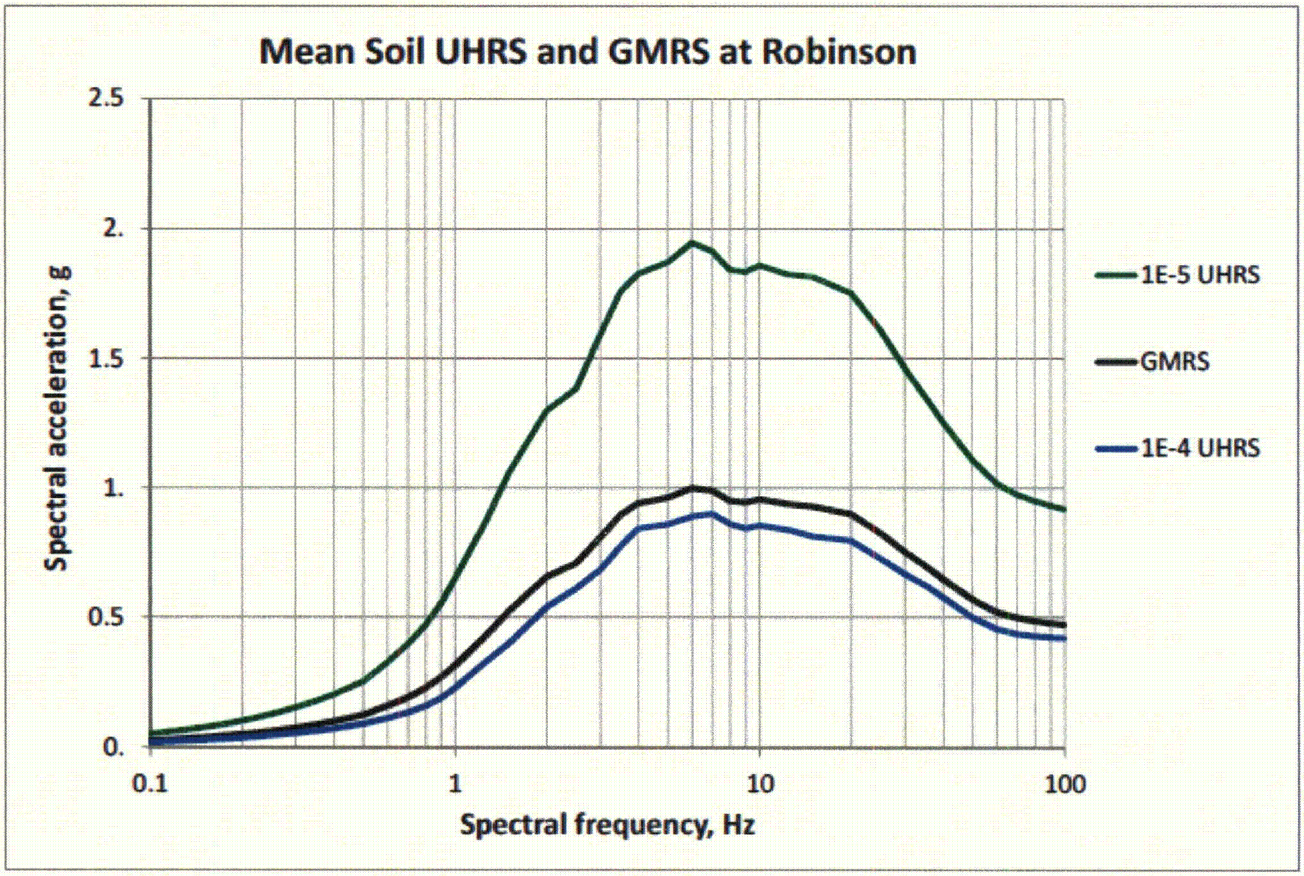


Figure 4-1: Plot of 1E-4 and 1E-5 UHRS and GMRS at Control Point for the H.B. Robinson Steam Electric Plant (5% Damped Response Spectra)

Control point hazard curves were used to develop the UHRS and the GMRS. The methodology described in SPID (Reference 15) was used to compute site-specific control point hazard curves. The selection of control point elevation is based on recommendations in Section 2.4.2 of the SPID (Reference 15). The control point elevation for the H.B. Robinson Steam Electric Plant is at El. 226 feet based on information in Sections 2.5 and 2.7 of the Updated Final Safety Analysis Report (Reference 51).

4.2 Comparison to SSE

Original design of the H.B. Robinson Steam Electric Plant was based on the 0.2g Housner Spectrum. Table 4-2a shows the spectral acceleration values as a function of frequency for the 5% damped horizontal SSE. As will be discussed in more detail in Section 5.2, original design in-structure response spectra was developed based on conservative time history. The Ground Level Response Spectrum that results from this time history is reported in Table 4-2b.

A comparison of the Ground Level Response Spectrum, SSE, and GMRS is shown in Figure 4-2. As shown in Figure 4-2, in the 1 to 10 Hz frequency range of the response spectrum, the GMRS exceeds the SSE and the Ground Level Response Spectrum. The GMRS also exceeds the SSE and the Ground Level Response Spectrum at frequency values higher than 10 Hz.

Table 4-2a: Original SSE Based on 0.2g Housner Spectrum for the H.B. Robinson Steam Electric Plant (5% Damping)

Frequency (Hz)	**SSE (g)**
1.0	0.17
1.5	0.230
2.0	0.260
2.5	0.290
3.0	0.3
3.5	0.310
4.0	0.32
5.0	0.305
6.0	0.290
7.0	0.265
8.0	0.255
9.0	0.240
10.0	0.23
12.50	0.210
15.0	0.2
20.0	0.2
25.0	0.2
30.0	0.2
33.0	0.2
35.0	0.2

Table 4-2b: Ground Level Response Spectrum Based on Time History for H.B. Robinson Steam Electric Plant (5% Damping)

Frequency (Hz)	**Ground Level Response Spectrum (g) from Time History**
1.0	0.300
1.5	0.455
2.0	0.441
2.5	0.417
3.0	0.445
3.5	0.468
4.0	0.489
5.0	0.455
6.0	0.415
7.0	0.380
8.0	0.351
9.0	0.316
10.0	0.281
12.50	0.221
15.0	0.232
20.0	0.246
25.0	0.258
30.0	0.267
33.0	0.273
35.0	0.275

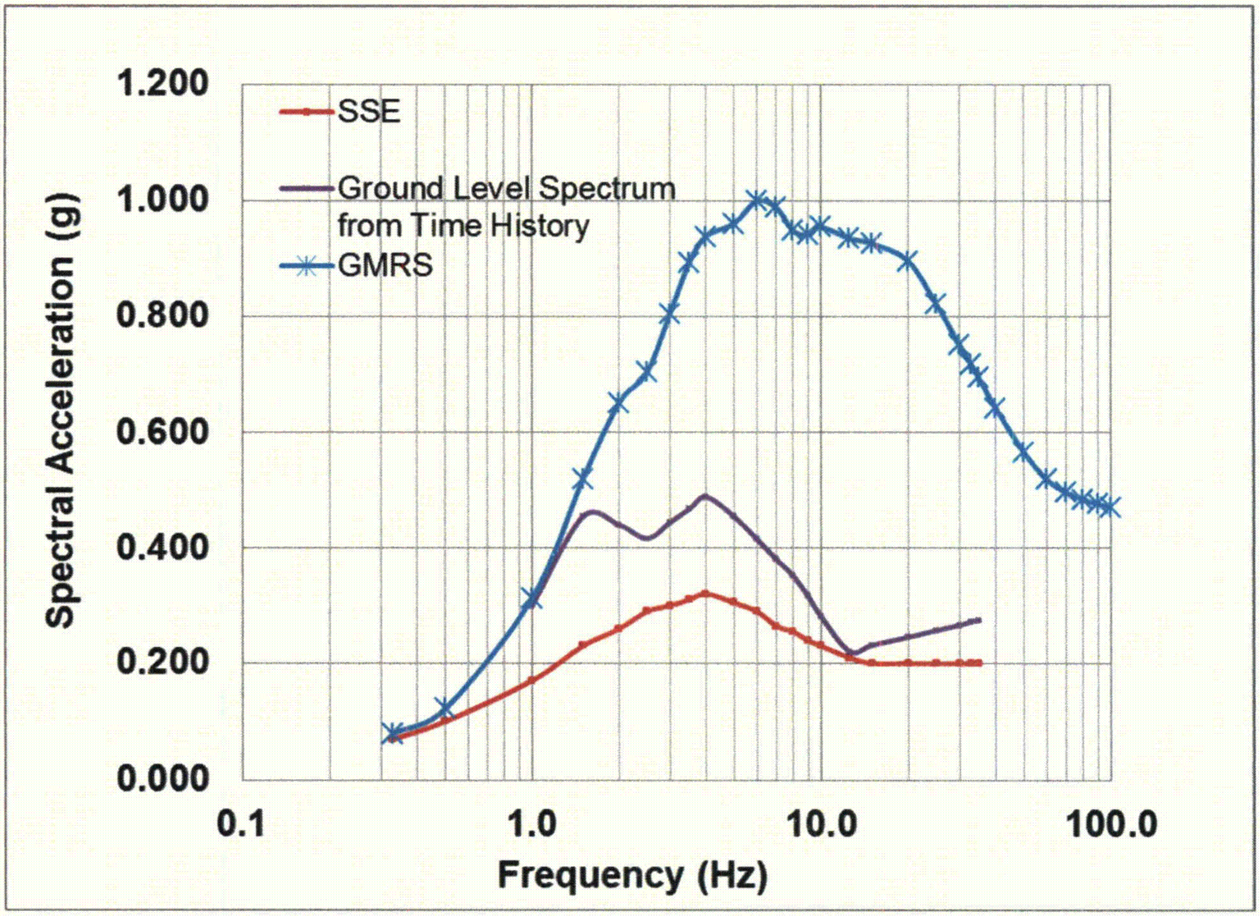


Figure 4-2: Comparison of GMRS, SSE and Ground Level Response Spectrum from Time History

5.0 Review Level Ground Motion (RLGM)

5.1 Description of RLGM Selected

Plants for which the GMRS exceeds the SSE in the 1.0 to 10.0 Hz frequency range do not screen out of the ESEP and require further seismic evaluation. The further seismic evaluation is performed to a Review Level Ground Motion which consists of a response spectrum above the SSE level. The RLGM is defined as a response spectrum reflecting an earthquake level that is above the plant's design basis SSE. The RLGM can be computed using one of the following criteria as described in Reference 2:

1. The RLGM can be derived by linearly scaling the SSE by the maximum ratio of the horizontal GMRS to the 5% damped SSE, between the 1 and 10 Hz frequency range, but not to exceed a ratio greater than 2 times the SSE. The in-structure seismic motions corresponding to the RLGM would be derived using existing SSE-based In-Structure Response Spectra (ISRS) scaled with the same factor.

2. Alternatively, licensees who have developed appropriate structural/soil-structure interaction (SSI) models capable of calculating ISRS based on site GMRS/Uniform Hazard Response Spectrum (UHRS) input may opt to use these ISRS in lieu of scaled SSE ISRS. In this case, the GMRS would represent the RLGM. EPRI 1025287 and the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard give guidance on acceptable methods to compute both the GMRS and the associated ISRS. Section 4 of Reference 2 contains full description of this task.

The RLGM for the H.B. Robinson Steam Electric Plant was developed in Reference 52 and in accordance with the methodology and objectives in EPRI ESEP guidance Reference 2. The RLGM is the SSE multiplied by a factor of 2.0. Table 5-1 is the RLGM as a function of frequency and acceleration at 5% damping. As discussed under Sections 4.2 and 5.2, original design in-structure response spectra were developed based on a conservative time history. The Ground Level Response Spectrum that resulted from this time history is reported in Table 4-2b and Figure 5-2. For consistency between component screening and component evaluations, the Ground Level Response Spectrum was scaled by 2 to represent an effective RLGM for component screening. Therefore, both screening and evaluation of ESEL items were conservatively based on 2 x Ground Level Response Spectrum (see Figure 6-1 for plot of 2 x Ground Level Response Spectrum) instead of 2 x SSE.

Table 5-1: RLGM for H.B. Robinson Steam Electric Plant

Frequency (Hz)	SSE (g)	RLGM (g)
1.0	0.17	0.34
1.5	0.230	0.460
2.0	0.260	0.520
2.5	0.290	0.58
3.0	0.3	0.60
3.5	0.310	0.62
4.0	0.32	0.64
5.0	0.305	0.61
6.0	0.290	0.58
7.0	0.265	0.53
8.0	0.255	0.51
9.0	0.240	0.48
10.0	0.23	0.46
12.50	0.210	0.42
15.0	0.2	0.4
20.0	0.2	0.4
25.0	0.2	0.4
30.0	0.2	0.4
33.0	0.2	0.4
35.0	0.2	0.4

The ratio of the GMRS to the SSE is summarized in Table 5-2. The maximum ratio of the GMRS to SSE is 4.635 and this occurs at frequency of approximately 15Hz. In the frequency range of 1 to 10Hz, the maximum ratio of the GMRS to SSE is 4.152. As limited in EPRI 3002000704, the RLGM is determined multiplying the SSE by a factor of 2.0.

Table 5-2: Ratio of GMRS to SSE

Frequency (Hz)	GMRS (g)	SSE (g)	GMRS/SSE
1.0	0.313	0.17	1.841
1.5	0.520	0.230	2.261
2.0	0.652	0.260	2.508
2.5	0.704	0.290	2.428
3.0	0.804	0.3	2.680
3.5	0.894	0.310	2.884
4.0	0.939	0.32	2.934
5.0	0.961	0.305	3.151
6.0	0.999	0.290	3.445
7.0	0.988	0.265	3.728
8.0	0.949	0.255	3.722
9.0	0.942	0.240	3.925
10.0	0.955	0.23	4.152
12.50	0.936	0.210	4.457
15.0	0.927	0.2	4.635
20.0	0.897	0.2	4.485
25.0	0.821	0.2	4.105
30.0	0.750	0.2	3.750
33.0	0.717	0.2	3.585
35.0	0.695	0.2	3.475

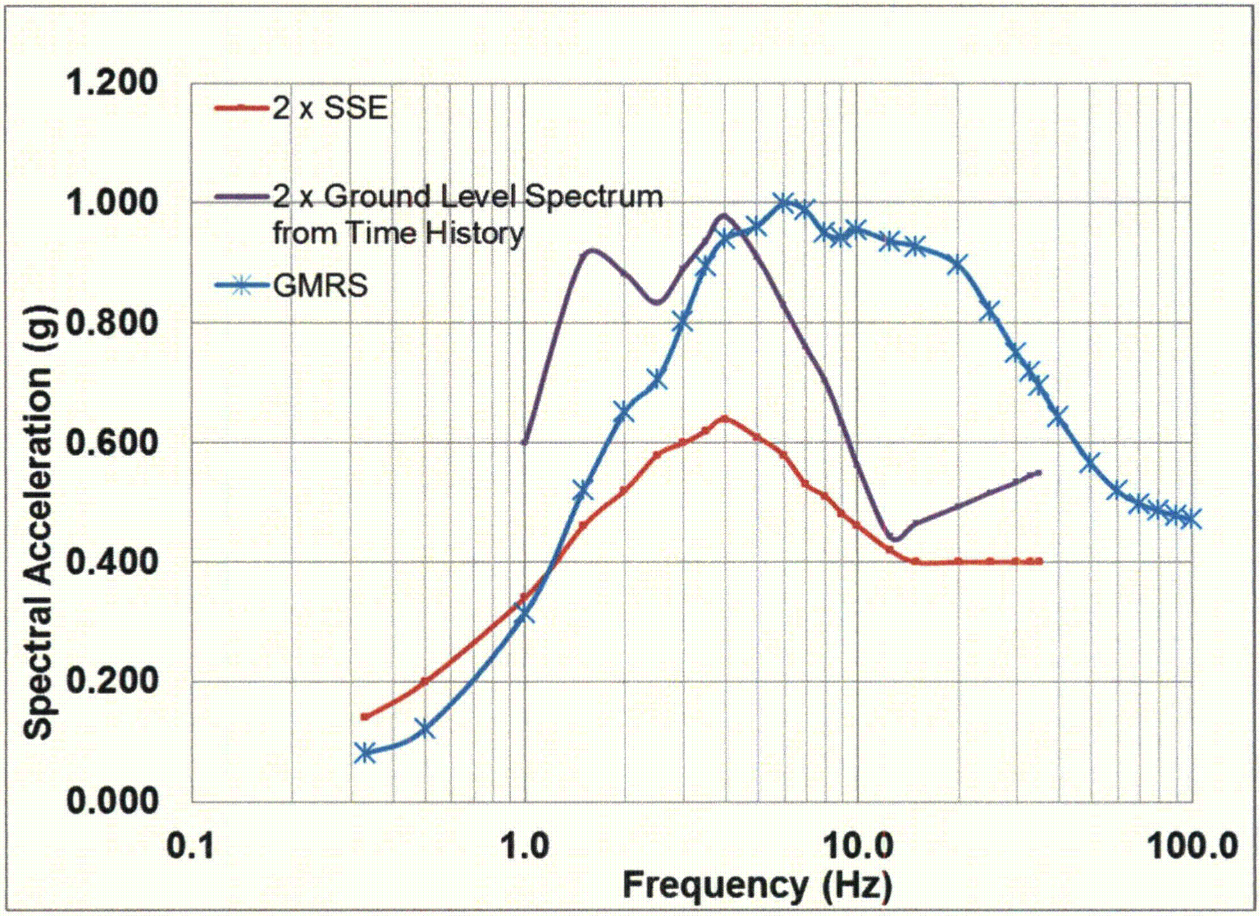


Figure 5-1: Plot of 5% Damping 2xSSE, 2 x Ground Level Response Spectrum, and GMRS

5.2 Method to Estimate ISRS

The seismic demand of the ESEL items/element mounted rigidly to the structure can be specified in terms of the In-Structure Response Spectra (ISRS). For use in the ESEP, the in-structure seismic demand for an element listed in the ESEL is defined by the ISRS scaled by the same factor used to obtain the RLGM from the SSE. The guidance under Section 4 of Reference 7 recommends broadening the peaks of the ISRS to account for the uncertainty in the civil structure frequency calculation. The extent of broadening is suggested to be at least 15 percent of the frequency approaching and proceeding spectral peaks but can be increased beyond the minimum recommendation based on the level of uncertainty associated with the structural model.

The original design basis ISRS for the H.B. Robinson Steam Electric Plant were generated in 1970 by Westinghouse Electric Corporation using mathematical building models developed by Ebasco Services, Inc. The original ISRS or floor spectra generated by Westinghouse was limited in scope and only considered the 0.20g design basis earthquake at damping ratio of 0.005 (0.5 percent). These ISRS include conservatisms that result from conservative selection of the time history and excessive bounding of design spectra. Figure 4-2 shows plot of: (1) Ground Level Response Spectrum; (2) GMRS; and (3) SSE.

Additional ISRS for other damping values were generated. The task of generating the additional floor response spectra was complicated by lack of availability of time history data from the original Westinghouse analysis. Consequently, synthetic ground motion time history that generates ISRS

comparable to the original Westinghouse floor spectra was used. The ISRS were generated by inputting the synthetic ground motion through the original Ebasco structural models. Scale factors as a function of frequency were developed by comparing the spectra at the desired damping ratio against the 0.50 percent damping spectra. The factors were then used to scale the original Westinghouse 0.50 percent damped spectra to the desired damping ratio. The reconstituted ISRS at the various damping ratios have been incorporated into the H.B. Robinson Steam Electric Plant's design basis ISRS documentation in Reference 18.

The ISRS from Reference 18 were peak broadened in accordance with guidance in Regulatory Guide 1.122 (Reference 19). Since the ISRS in Reference 18 are already broadened, these spectra are scaled by a factor of 2.0 for ESEP.

In summary, in-structure response spectra developed with the conservative Ground Level Response Spectrum were scaled by a factor of 2 for use in ESEP. Figure 5-1 shows plot of the 2 x SSE (RLGM), 2 x Ground Level Response Spectrum, and GMRS.

6.0 Seismic Margin Evaluation Approach

It is necessary to demonstrate that ESEL items have sufficient seismic capacity to meet or exceed the demand characterized by the RLGM and the corresponding scaled in-structure response spectra. The seismic capacity is characterized as the peak ground acceleration (PGA) for which there is a high confidence of a low probability of failure (HCLPF). The PGA is associated with a specific spectral shape, in this case the 5%-damped 2 x Ground Level Response Spectrum shape. The HCLPF capacity must be equal to or greater than the RLGM PGA. The criteria for seismic capacity determination are given in Section 5 of EPRI 3002000704.

There are two basic approaches for developing HCLPF capacities:

1. Deterministic approach using the conservative deterministic failure margin (CDFM) methodology of EPRI NP-6041, A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1) (Reference 7).
2. Probabilistic approach using the fragility analysis methodology of EPRI TR-103959, Methodology for Developing Seismic Fragilities (Reference 8).

6.1 Summary of Methodologies Used

The H. B. Robinson Steam Electric Plant completed a seismic margin assessment (SMA) in 1993. The SMA is documented in Reference 9 and consisted of screening, walkdowns by SRT, and HCLPF anchorage calculations. The screening and walkdowns used the screening tables from Chapter 2 of EPRI NP-6041 (Reference 7) for peak spectral acceleration less than 0.8g. The walkdowns were conducted by engineers trained in EPRI NP 6041 (the engineers attended the EPRI SMA Add-On course in addition to the SQUG Walkdown Screening and Seismic Evaluation Training Course), and were documented on Screening Evaluation Work Sheets from EPRI NP-6041. Anchorage capacity calculations used the CDFM criteria from EPRI NP-6041. Seismic demand was the IPEEE Review Level Earthquake (RLE) for SMA (mean NUREG/CR-0098 [Reference 11] ground response spectrum anchored to 0.3g PGA).

Figure 6-1 shows the mean NUREG/CR-0098 ground response spectrum used as the IPEEE RLE compared to the 2 x Ground Level Response Spectrum. The figure shows that the ESEP input motion enveloped the IPEEE RLE at all frequencies except between 10 Hz and 15 Hz where the IPEEE RLE slightly exceed the ESEP input motion. The frequency of interest for ESEL items is between 1 Hz and 10Hz.

The ESEP methodology included screening and extensive walkdown by the Seismic Review Team (SRT), and HCLPF calculations to evaluate structural capacity of the ESEL items against the RLGM. Function evaluation of relays was also performed. The walkdowns were documented on Screening Evaluation Worksheets (SEWS) from EPRI NP-6041. Based on outcome of the seismic walkdown and documentation in SEWS, six (6) HCLPF calculations were performed to envelope the thirteen (13) ESEL items identified during the walkdowns.

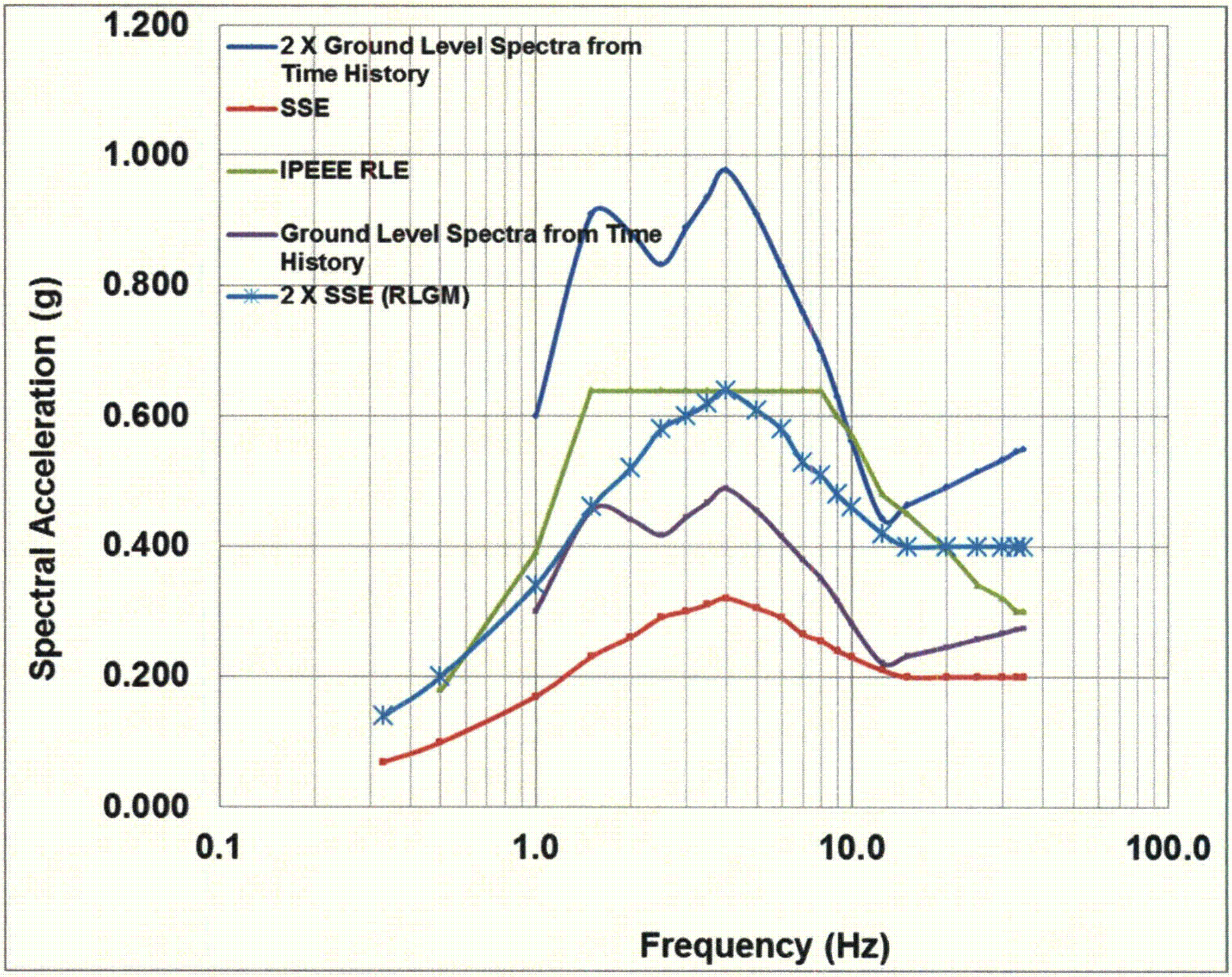


Figure 6-1. Comparison of the H.B. Robinson Steam Electric Plant IPEEE RLE, ESEP RLGM, SSE, Ground Level (El. 226ft) Spectrum from Time History, and 2 x Ground Level (El. 226ft) Spectrum from Time History

6.2 HCLPF Screening Process

The HCLPF screening and calculations were based on 2 x Ground Level Response Spectrum peak ground acceleration. Screening tables in EPRI NP-6041 (Reference 7) are based on peak spectral acceleration of ≤ 0.8g, 0.8g to 1.2g, and ≥ 1.2g. Since 2 x Ground Level Response Spectrum peak ground acceleration is 0.978g, screening of ESEL items was based on the 0.8g to 1.2g range criteria. The screening guidelines were supplemented by Appendix A of EPRI NP-6041 SL which provides the basis for the seismic capacity screening guidelines.

Anchorage capacity calculations were based on 2 x Ground Level Response Spectrum. Equipment for which the screening caveats were met and for which the anchorage capacity exceeded 2 x Ground Level Response Spectrum seismic demand were screened out from ESEP seismic capacity determination.

6.3 Seismic Walkdown Approach

6.3.1 Walkdown Approach

Walkdowns were performed in accordance with the criteria provided in Section 5 of EPRI 3002000704 (Reference 2), which refers to EPRI NP-6041 (Reference 7) for the Seismic Margin Assessment process. Pages 2-26 through 2-30 of EPRI NP-6041 describe the seismic walkdown criteria, including the following key criteria.

"The SRT [Seismic Review Team] should "walk by" 100% of all components which are reasonably accessible and in non-radioactive or low radioactive environments. Seismic capability assessment of components which are inaccessible, in high-radioactive environments, or possibly within contaminated containment, will have to rely more on alternate means such as photographic inspection, more reliance on seismic reanalysis, and possibly, smaller inspection teams and more hurried inspections. A 100% "walk by" does not mean complete inspection of each component, nor does it mean requiring an electrician or other technician to de-energize and open cabinets or panels for detailed inspection of all components. This walkdown is not intended to be a QA or QC review or a review of the adequacy of the component at the SSE level.

If the SRT has a reasonable basis for assuming that the group of components are similar and are similarly anchored, then it is only necessary to inspect one component out of this group. The "similarity-basis" should be developed before the walkdown during the seismic capability preparatory work (Step 3) by reference to drawings, calculations or specifications. The one component or each type which is selected should be thoroughly inspected which probably does mean de-energizing and opening cabinets or panels for this very limited sample. Generally, a spare representative component can be found so as to enable the inspection to be performed while the plant is in operation. At least for the one component of each type which is selected, anchorage should be thoroughly inspected.

The walkdown procedure should be performed in an ad hoc manner. For each class of components the SRT should look closely at the first items and compare the field configurations with the construction drawings and/or specifications. If a one-to-one correspondence is found, then subsequent items do not have to be inspected in as great a detail. Ultimately the walkdown becomes a "walk by" of the component class as the SRT becomes confident that the construction pattern is typical. This procedure for inspection should be repeated for each component class; although, during the actual walkdown the SRT may be inspecting several classes of components in parallel. If serious exceptions to the drawings or questionable construction practices are found then the system or component class must be inspected in closer detail until the systematic deficiency is defined.

The 100% "walk by" is to look for outliers, lack of similarity, anchorage which is different from that shown on drawings or prescribed in criteria for that component, potential SI [Seismic Interaction[1]] problems, situations that are at odds with the team members' past experience, and any other areas of serious seismic concern. If any such concerns surface, then the limited sample size of one component of each type for thorough inspection will have to be increased. The increase in sample size which should be inspected will depend upon the number of outliers and different anchorages, etc., which are observed. It is up to the SRT to ultimately select the sample size since they are the

[1]EPRI 3002000704 page 5-4 limits the ESEP seismic interaction reviews to "nearby block walls" and "piping attached to tanks" which are reviewed "to address the possibility of failures due to differential displacements." Other potential seismic interaction evaluations are "deferred to the full seismic risk evaluations performed in accordance with EPRI 1025287'

ones who are responsible for the seismic adequacy of all elements which they screen from the margin review. Appendix D gives guidance for sampling selection."

As part of the ESEP, demonstration that the components listed in the ESEL have a HCLPF capacity that exceeds the effective RLGM (2 x Ground Level Response Spectrum) verifies adequate seismic ruggedness. Section 5 of EPRI ESEP guidance specifies that the methodology in EPRI NP-6041 SL may be used for the development of the HCLPF capacity. The major steps in Reference 7 include pre-screening, walkdowns, and the CDFM HCLPF calculations.

In order to ensure efficiency while performing the walkdowns and during seismic capacity evaluations, each of the items listed in the ESEL were subjected to pre-screening. The initial pre-screening effort consisted of data collection in the form of drawings, calculations, specifications, and vendor documents for each item in the ESEL. After identification of documentation for a specific item, the pre-screening process followed the general seismic capacity screening guidelines presented in Reference 7 for civil structures, equipment, and subsystems to be considered screened out from further review. The caveats and footnoted exceptions and restrictions listed are followed.

For the purpose of completing the ESEP for the H. B. Robinson Steam Electric Plant, only Table 2-4 of Reference 7 is relevant for applying seismic screening criteria for plant equipment listed in the ESEL. In addition to using the screening criteria in Reference 7 during plant walkdown, the SRT also exercised their collective experience and judgment while using the criteria for specific component. The screening criteria can be used for equipment that is approximately 40ft above grade or lower. EPRI Report No 1019200 (Reference 23) provides guidance on screening criteria for equipment that is greater than 40ft above grade. Screening criteria in Reference 7 do not include considerations for anchorage. Therefore, structural integrity of anchorage was evaluated separately. Some simple cases were documented on the SEWS form.

Plant walkdowns were performed for items in the ESEL using guidance in Reference 7. Information extracted from existing documentation such as equipment location, seismic input elevation, relevant drawing details, and previous seismic capacity calculations were recorded on the ESEP SEWS and used during the walkdowns. In accordance with the ESEP guidance, the SEWS that were used in the ESEP walkdowns were consistent with content and format of the SEWS presented in Appendix F of EPRI NP-6041 SL.

A major part of the ESEP walkdowns was the investigation of equipment anchorages. Therefore, cabinets with anchorages located internally were opened. Furthermore, the ESEP guidance states that components that are anchored to sub-structural elements that may not have the same capacity as the main structural system (e.g. block walls, frames, stanchions etc.) should also be reviewed. Nearby block walls were identified and evaluated as necessary. Piping attached to tanks were also reviewed. Other potential seismic interaction evaluations were deferred to a full Seismic Risk Evaluation (SRE) as discussed in the SPID References 14 and 15, and were not addressed in the ESEP walkdowns.

Walkdown assessment for the H.B. Robinson Steam Electric Plant ESEL items were completed by the SRT between August 2013 and February 2014. Some of the components were previously walked down during the IPEEE, USI A-46, or NTTF 2.3: Seismic and relevant information such as the equipment location, seismic input elevation, drawing details and previous seismic calculations were recorded on the ESEP SEWS. Previous walkdowns were credited since they were performed by qualified Seismic Review Team. A walk-by of these components was performed and documented. The objective of the walk-by is to confirm and verify that the components and their anchorage have not degraded since the previous walkdown.

Items included in the ESEL that have not been previously walked down and evaluated, were automatically included for a detailed walkdown.

The SRT was comprised of at least two SQUG trained engineers and often included two additional structural engineers (Reference 57). The results of the walkdowns were documented on the SEWS for each item. The completed SEWS and pictures taken during the walkdowns for the ESEL are documented in Reference 55. Follow-up inspections and walkdowns were completed where additional information was necessary.

6.3.2 Application of Previous Walkdown Information

Previous seismic walkdowns from IPEEE and USI A-46 were used to support the ESEP seismic evaluations. Some of the components and items on the ESEL were included in the NTTF 2.3 seismic walkdowns (Reference 17). Those walkdowns were well documented and recent enough that they did not need to be repeated for the ESEP.

Several ESEL items were previously walked down during the H.B. Robinson Steam Electric Plant Seismic IPEEE program. Those walkdown results were reviewed and the following steps were taken to confirm that the previous walkdown conclusions remained valid.

- A walk by was performed to confirm that the equipment material condition and configuration is consistent with the walkdown conclusions and that no new significant interactions related to block walls or piping attached to tanks exist.
- If the ESEL item was screened out based on the previous walkdown, that screening evaluation was reviewed and reconfirmed for the ESEP.

6.3.3 Significant Walkdown Findings

Consistent with guidance from NP-6041, no significant outliers or anchorage concerns (except MCC-A) were identified during the H.B. Robinson Steam Electric Plant seismic walkdowns. The following findings were noted during the walkdowns.

- Nearby block walls were identified in the proximity of ESEL item. These block walls were assessed for their structural adequacy to withstand the seismic loads resulting from the RLGM. There is no case where the block wall represented the HCLPF failure mode for an ESEL item.

- Piping attached to tanks were reviewed and evaluated for their structural integrity to withstand seismic-induced loads from RLGM.

- Cabinets with anchorage located internally were opened and evaluated against RLGM.

- Thirteen (13) components were identified by the SRT during the plant walkdowns and six (6) HCLPF calculations were performed to envelope the thirteen components identified.

6.4 HCLPF Calculation Process

ESEL items not included in the previous IPEEE evaluations at H.B. Robinson Steam Electric Plant were evaluated using the criteria in EPRI NP-6041. Those evaluations included the following steps:

- Performing seismic capability walkdowns for equipment not included in previous seismic walkdowns (SQUG, IPEEE, or NTTF 2.3) to evaluate the equipment installed plant conditions. Results of the walkdowns which are documented in the ESEP SEWS identified thirteen (13) components that require HCLPF calculation.
- Performing screening evaluations using the screening tables in EPRI NP-6041 as described in Section 6.2 and
- Performing HCLPF calculations considering various failure modes that include structural failure modes (e.g. anchorage, load path etc.) and functional failure modes.

Items based on similarity of model, function and anchorage were grouped together. Based on EPRI NP-6041-SL rule of similarity, a bounding anchorage evaluation was performed for equipment grouped together. The calculations evaluate the demand and capacity of the equipment anchorage and derived a HCLPF capacity from the results of the anchorage evaluation. The functional failure mode(s) are also evaluated.

Equipment that were identified as requiring a HCLPF capacity calculation in Reference 55 were evaluated using the CDFM methodology as outlined in EPRI NP-6041-SL. The HCLPF calculations are documented in Reference 10 and References 25 through 29. Thirteen components were identified by the SRT during walkdown and six HCLPF calculations were completed to envelope all the components which include I&C and Hagan rack; Pressure Vessel; MCC; Battery Charger; and Auxiliary DC Panel.

6.5 Functional Evaluations of Relays

Based on review of ESEL and associated single line diagrams, two relays (Under-Voltage Alarm Relay 27/MCC-A and Under-Voltage Alarm Relay 27/MCC-B) were identified. However, these

relays do not have lockout or seal-in mechanism (Reference 59) and are not required during FLEX implementation. 27/MCC-A and 27/MCC-B are not designed to operate during and following DBE and BDBEE. Therefore, these relays were not included on the ESEL list. Extensive review of the single line diagrams did not identify any other relay or contactor that will be of concern.

6.6 Tabulated ESEL HCLPF Values (Including Key Failure Modes)

Tabulated ESEL HCLPF values are provided in Table 6-1. The following notes apply to the information in the table:

- For items screened out using NP 6041 screening tables, the screening level can be provided as >RLGM and the failure mode can be listed as "Screened", (unless the controlling HCLPF value is governed by anchorage).
- For items where anchorage controls the HCLPF value, the HCLPF value is listed in the table and the failure mode is noted as "anchorage."

Six HCLPF calculations were performed for items listed in the ESEL. Items that are based on similarity of equipment model, function, and anchorage are grouped together. Based on EPRI NP-6041 SL rule of similarity, some items were grouped together and a bounding anchorage evaluation was performed. The six HCLPF capacity evaluations are documented in Reference 10 and References 25 through 29. Each capacity calculation evaluates the demand and capacity of the equipment anchorage and derives a HCLPF capacity from the results of the anchorage evaluation. The functional failure modes for each ESEL item were identified and documented in the calculation. The functional and anchorage HCLPF capacity of items identified by the SRT for a seismic capacity evaluation is presented in Table 6-1.

Table 6-1: Functional and Anchorage HCLPF Capacity Results

Equipment Group	Equipment	Functional HCLPF Capacity	Anchorage/Structural HCLPF Capacity
Instrumentation and Control Panels and Rack	Main Control Board	$\geq$ 0.40g	0.414g
Hagan Racks	Rack -4 Rack -11 Rack -12 Rack -13	$\geq$ 0.40g	0.445g
Pressure Vessels	Boron Injection Tank	$\geq$ 0.40g	0.541g
Battery Chargers	Battery Charger - A Battery Charger - A1 Battery Charger - B Battery Charger - B1	$\geq$ 0.40g	0.755g
Motor Control Centers	MCC-A	$\geq$ 0.40g	0.250g
	MCC-B	$\geq$ 0.40g	0.406g
Auxiliary DC Panel GD	AUX-PNL-GD	$\geq$ 0.40g	0.596g

7.0 Inaccessible Items

7.1 Identification of ESEL items inaccessible for walkdowns

All ESEL items were accessible with the exception of TE-423. This temperature element is rugged and due to installation internal to the pipe, it is also protected from seismic interaction. An evaluation was performed based on available information and this item was determined to be acceptable by the SRT with no visual examination.

7.2 Planned Walkdown / Evaluation Schedule / Close Out

No ESEL item requires future walkdown.

8.0 ESEP Conclusions and Results

8.1 Supporting Information

The H.B. Robinson Steam Electric Plant has performed the ESEP as an interim action in response to Reference 1, the NRC's 10 CFR 50.54(f) letter. It was performed using the methodologies in Reference 2, the NRC endorsed guidance in EPRI 3002000704.

The ESEP provides an important demonstration of seismic margin and expedites plant safety enhancements through evaluations and potential near-term modifications of plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events.

The ESEP is part of the overall H.B. Robinson Steam Electric Plant response to the NRC's 50.54(f) letter. On March 12, 2014, NEI submitted to the NRC results of Reference 12, a study of seismic core damage risk estimates based on updated seismic hazard information as it applies to operating nuclear reactors in the Central and Eastern United States (CEUS). The study concluded that "site-specific seismic hazards show that there has not been an overall increase in seismic risk for the fleet of U.S. plants" based on the re-evaluated seismic hazards. As such, the "current seismic design of operating reactors continues to provide a safety margin to withstand potential earthquakes exceeding the seismic design basis."

The NRC's May 9, 2014 NTTF 2.1 Screening and Prioritization letter (Reference 14) concluded that the "fleetwide seismic risk estimates are consistent with the approach and results used in the GI-199 safety/risk assessment." The letter also stated that "As a result, the staff has confirmed that the conclusions reached in GI-199 safety/risk assessment remain valid and that the plants can continue to operate while additional evaluations are conducted."

An assessment of the change in seismic risk for H.B. Robinson Steam Electric Plant was included in the fleet risk evaluation submitted in the March 12, 2014 NEI letter therefore, the conclusions in the NRC's May 9 letter also apply to H.B. Robinson Steam Electric Plant.

In addition, Reference 12, the March 12, 2014 NEI letter, provided an attached "Perspectives on the Seismic Capacity of Operating Plants," which (1) assessed a number of qualitative reasons why the design of SSCs inherently contain margin beyond their design level, (2) discussed industrial seismic experience databases of performance of industry facility components similar to nuclear SSCs, and (3) discussed earthquake experience at operating plants.

The fleet of currently operating nuclear power plants was designed using conservative practices, such that the plants have significant margin to withstand large ground motions safely. This has been borne out for those plants that have actually experienced significant earthquakes. The seismic design process has inherent (and intentional) conservatisms which result in significant seismic margins within structures, systems and components (SSCs). These conservatisms are reflected in several key aspects of the seismic design process, including:

- Safety factors applied in design calculations
- Damping values used in dynamic analysis of SSCs
- Bounding synthetic time histories for in-structure response spectra calculations
- Broadening criteria for in-structure response spectra
- Response spectra enveloping criteria typically used in SSC analysis and testing applications
- Response spectra based frequency domain analysis rather than explicit time history based time domain analysis
- Bounding requirements in codes and standards
- Use of minimum strength requirements of structural components (concrete and steel)
- Bounding testing requirements, and

- Ductile behavior of the primary materials (that is, not crediting the additional capacity of materials such as steel and reinforced concrete beyond the essentially elastic range, etc.).

These design practices combine to result in margins such that the SSCs will continue to fulfill their functions at ground motions well above the SSE.

The intent of the ESEP is to perform an interim action in response to the NRC's 50.54(f) letter to demonstrate seismic margin through a review of a subset of the plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events. In order to complete the ESEP in an expedited amount of time, the RLGM used for the ESEP evaluation is a scaled version of the plant's SSE rather than the actual GMRS. To more fully characterize the risk impacts of the seismic ground motion represented by the GMRS on a plant specific basis, a more detailed seismic risk assessment (SPRA or risk-based SMA) is to be performed in accordance with EPRI 1025287 (Reference 15). As identified in Reference 4, the H. B. Robinson Steam Electric Plant Seismic Hazard and GMRS submittal , the H.B. Robinson Steam Electric Plant screens in for a seismic risk evaluation. The complete seismic risk evaluation will more completely characterize the probabilistic seismic ground motion input into the plant, the plant response to that probabilistic seismic ground motion input, and the resulting plant risk characterization. H.B. Robinson Steam Electric Plant will complete that evaluation in accordance with the schedule identified in Reference 13, NEI's letter dated April 9, 2013 and endorsed by the NRC in Reference 16, their May 7, 2013 letter.

8.2 Identification of Planned Modifications

There are no planned future modifications for ESEP. The ESEP identified MCC-A as having a HCLPF capacity below the RLGM and not meeting the requirements of EPRI ESEP and NTTF Recommendation 2.1: Seismic. MCC-A has since been modified in accordance with EPRI 3002000704 to increase its seismic capacity to the RLGM. This was achieved by bracing the cabinet at the top. This modification eliminated flexible modes and resulted in reduced tensile load applied to the concrete expansion anchors. The HCLPF capacity of MCC-A is now greater than 0.4g.

The ESEP determined that the HCLPF capacity of MCC-B was slightly above the RLGM and meets the requirements of the EPRI ESEP such that no modification was required. However, a modification similar to that discussed above for MCC-A was implemented in order to increase the capacity of MCC-B anchorage and eliminate potential inertial forces at the top entry cable tray and conduit.

Seismic margin above 2 x SSE was also added to a group of instrument racks (Hagan Racks) by validating the bolting integrity of the top braces (a relatively minor scope of work). The HCLPF capacity of the Main Control Board is higher than the RLGM and meets the requirements of the EPRI ESEP. However, greater seismic capacity can be demonstrated by additional inspection of plug welds that form part of the anchorage. The additional inspection should confirm plug weld thickness and quality. Table 6-1 shows the capacities of the thirteen ESEL items that required HCLPF calculation. No additional modifications are planned for the H.B. Robinson Steam Electric Plant related to ESEP.

8.3 Modification Implementation Schedule

The only ESEL item that required modification based on the seismic walkdown and HCLPF capacity calculation was MCC-A. The modification has been developed and implemented as discussed in Section 8.2. The anchorage system for MCC-B is slightly different from that of MCC-A and has higher structural capacity. The HCLPF capacity of MCC-B slightly exceeds RLGM demand. However, similar modification developed for MCC-A was also implemented on MCC-B. Although, not considered a modification, the Hagan Rack cabinets bolts were tightened to improve structural capacity.

8.4 Summary of Planned Actions

The H.B. Robinson Steam Electric Plant has no follow-up action or planned modification to support the ESEP. All of the items identified in the ESEL currently have a HCLPF capacity at or above the RLGM and do not require further evaluation. The ESEL has been updated to consider new equipment that account for the changes in the FLEX strategy. The new FLEX strategy was subjected to critical path analysis and those items that fall under the ESEP guidelines have been added to the ESEL.

9.0 References

1) NRC (E Leeds and M Johnson) Letter to All Power Reactor Licensees et al., "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," March 12, 2012.
2) Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1 – Seismic. EPRI, Palo Alto, CA: May 2013. 3002000704.
3) Updated H.B. Robinson Steam Electric Plant Overall Integrated Plan (OIP) in Response to the March 12, 2012, Commission Order EA-12-049, August 2014.
4) H.B. Robinson Steam Electric Plant Seismic Hazard and GMRS submittal, dated March 31, 2014.
5) Nuclear Regulatory Commission, NUREG-1407, Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, June 1991
6) Nuclear Regulatory Commission, Generic Letter No. 88-20 Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR 50.54(f), June 1991
7) A Methodology for Assessment of Nuclear Power Plant Seismic Margin, Rev. 1, August 1991, Electric Power Research Institute, Palo Alto, CA. EPRI NP 6041
8) Methodology for Developing Seismic Fragilities, August 1991, EPRI, Palo Alto, CA. 1994, TR-103959
9) Appendix A to The H.B. Robinson Steam Electric Plant Unit No. 2 Individual Plant Examination for External Event Submittal: Seismic IPEEE
10) Calculation RNP-13-05-600-005,"High Confidence of Low Probability of Failure (HCLPF) for the H.B. Robinson Steam Electric Plant Motor Control Center A and B (MCC-A and MCC-B)}
11) Nuclear Regulatory Commission, NUREG/CR-0098, Development of Criteria for Seismic Review of Selected Nuclear Power Plants, published May 1978
12) Nuclear Energy Institute (NEI), A. Pietrangelo, Letter to D. Skeen of the USNRC, "Seismic Core Damage Risk Estimates Using the Updated Seismic Hazards for the Operating Nuclear Plants in the Central and Eastern United States", March 12, 2014
13) Nuclear Energy Institute (NEI), A. Pietrangelo, Letter to D. Skeen of the USNRC, "Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations", April 9, 2013
14) NRC (E Leeds) Letter to All Power Reactor Licensees et al., "Screening and Prioritization Results Regarding Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(F) Regarding Seismic Hazard Re-Evaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights From the Fukushima Dai-Ichi Accident," May 9, 2014.
15) Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic. EPRI, Palo Alto, CA: February 2013. 1025287.
16) NRC (E Leeds) Letter to NEI (J Pollock), "Electric Power Research Institute Final Draft Report xxxxx, "Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," as an Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluations," May 7, 2013
17) H.B. Robinson Steam Electric Plant NTTF 2.3 Seismic Walkdown Submittal dated February 27, 2014.

18) Carolina Power and Light Company (CP&L), Specification No CPL-HBR2-C-008,"Specification for Floor Response Spectra", Revision 1, 1991.
19) United States Nuclear Regulatory Commission, Regulatory Guide 1.122,"Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components", Revision 1, February 1978.
20) United States Nuclear Regulatory Commission NUREG-2115, Department of Energy Office of Nuclear Energy (DOE/NE)-0140, EPRI 1021097,"Central and Eastern United States Seismic Source Characterization for Nuclear Facilities", 6 Volumes, 2012.
21) Electric Power Research Institute (EPRI), Final Report No 3002000717,"EPRI (2004, 2006) Ground Motion Model (GMM) Review Project", June 2013.
22) URS Energy and Construction Calculation RNP-13-05-600-001,"Review Level Ground Motion (RLGM) and In-Structure Response Spectra (ISRS) for H.B. Robinson Steam Electric Plant Unit 2, Revision 0, July 2013.
23) Electric Power Research Institute (EPRI) Final Report No 1019200,"Seismic Fragility Applications Guide Update", 2009.
24) EC 92103,"Fukushima NTTF Recommendation 2.1: Seismic Reevaluation"
25) Calculation RNP-13-05-600-006,"High Confidence of Low Probability of Failure (HCLPF) for the H.B. Robinson Steam Electric Plant Battery Chargers"
26) Calculation RNP-13-05-600-004,"High Confidence of Low Probability of Failure (HCLPF) for the H.B. Robinson Steam Electric Plant Boron Injection Tank"
27) Calculation RNP-13-05-600-003,"High Confidence of Low Probability of Failure (HCLPF) for the H.B. Robinson Steam Electric Plant East Hagan Rack"
28) Calculation RNP-13-05-600-007,"High Confidence of Low Probability of Failure (HCLPF) for the H.B. Robinson Steam Electric Plant Auxiliary DC Panel GD".
29) Calculation RNP-13-05-600-002,"High Confidence of Low Probability of Failure (HCLPF) for the H.B. Robinson Steam Electric Plant Main Control Board".
30) RNP/RA-13-0087, First Six Month Status Report (Order EA-12-049) H. B. Robinson Steam Electric Plant (RNP), Unit 2.
31) RNP/RA-14-0008, Second Six Month Status Report (Order EA-12-049) H. B. Robinson Steam Electric Plant (RNP), Unit 2.
32) RNP/RA-14-0083, Third Six Month Status Report (Order EA-12-049) H. B. Robinson Steam Electric Plant (RNP), Unit 2.
33) Engineering Change (EC) 88926, FLEX Strategies and Implementation Plan, Rev. 3.
34) NEI 12-06, Diverse and Flexible Coping Strategies (FLEX) Implementation Guideline, Revision 0.
35) EOP-ECA-0.0, Loss Of All AC Power, Revision 0.
36) WCAP-1760-P, Revision 1, Reactor Coolant System Response to Extended Loss of AC Power Event for Westinghouse, Combustion Engineering, and Babcock & Wilcox NSSS Designs for Phase Boration, August 2012.
37) PA-PSC-0965, PWROG Core Cooling Position Paper.
38) EDMG-004, Steam Generators.
39) Calculation RNP-M/MECH-1712, Appendix R Mechanical Basis, Section 3.27, Cooldown Using MSIV Bypass Lines.
40) EC 90617, Pre-Staged Diesel Generator Design To Power 125VDC - A Train and B Train Battery Chargers For Fukushima Support (NTFF 4.2 - FLEX).
41) FSG-10, Passive RCS Injection Isolation.
42) EC 95216, NTTF 2.1 Interim Action RCS Injection.
43) EC 90622, Standard Piping Connections For NTTF 4.2 (FLEX).
44) EC 94745, Boric Acid and RCS Make Up Connections To The Safety Injection System NTTF 4.2 - Flexible Coping Strategies

45) Calculation RNP-M/MECH-1877, RNP Extended Loss of AC (ELAP) Power Containment Response

46) FSG-12, Alternate Containment Cooling

47) EC 90623, New Pipe Tee And Standard Connection For NTTF 4.2 (FLEX)

48) EC 95266, Isolation Valves And Connection For AFW - FUKUSHIMA-Admin Rev

49) EC 92103R0, Attachment Z03R0 Mechanical Documents

50) EC 92103R0, Attachment Z05R0 Electrical Documents

51) UFSAR, Section 02, Site Characteristics"

52) EC 92103R0, Attachment Z06R0

53) EC 92103, Attachment Z18R0

54) EC 92103, Attachment Z09R0

55) EC 92103, Attachment Z10R0

56) EC 92103, Attachment Z01R0

57) EC 92103, Attachment Z16R0

58) NRC Letter from NRC to Duke Energy and South Carolina Electric and Gas Company, Request for Additional Information Associated with Near-Term Force Recommendation 2.1, Seismic Re-Evaluations Related to Southeastern Catalog Changes (TAC NOS MF3724, MF3736, MF 3738, and MF 3831), dated October 23, 2014, ADAM Accession No ML14268A516.

59) H.B. Robinson Steam Electric Plant Control Wiring Diagram (CWD) B-190628, SH 00955 and 00956.

60) EC 92501, Attachment Z09R1,"Additional Seismic Interim Actions Studies for the H.B. Robinson Steam Electric Plant".

61) Letter Dated August 28, 2014 from EPRI to NRC Review of EPRI 1021097 Earthquake Catalog for RIS Earthquakes in the Southeastern U.S. and Earthquakes in South Carolina Near Time of the 1886 Charleston Earthquake Sequence.

62) Response to Request for Additional Information Associated with Near-Term Task Force Recommendation 2.1, Seismic Re-Evaluations Related to Southeastern Catalog Changes (TAC NOS. MF3724, MF3736, MF3737, MF3738, and MF3831), November 12, 2014

Attachment A - H.B. Robinson Steam Electric Plant ESEL

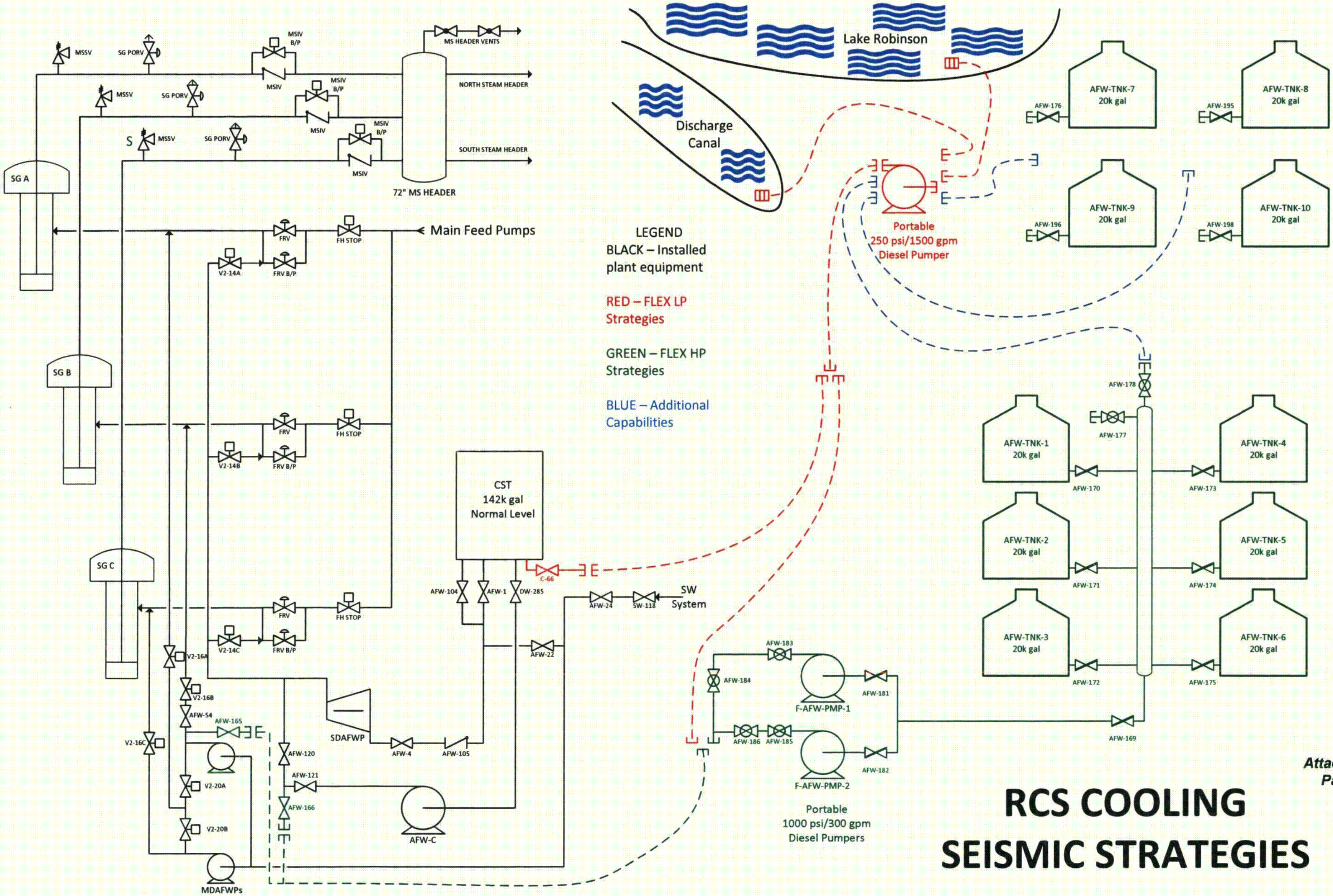
ESEL

ESEL #	Equipment ID	System	Description	Equipment Type	Building/Location	Room or Row/ Column	Equipment Normal State	Equipment Failed State	Equipment Desired State	Drawing	Notes
001	COND-STRG-TNK	3070	CONDENSATE STORAGE TANK	TNK	TUR1-AA6-0		N/A	N/A	N/A	G-190197 SH00001	
002	FLEX Piping connection at C-66	3070	CST PIPE CONNECTION		TUR1		AVAILABLE	N/A	N/A		TO BE INSTALLED PER EC 90622
003	SDAFW-PMP	3065	STEAM DRIVEN AUXILARY FEEDWATER PUMP	PMP	TUR1-H13-0		OFF	N/A	ON	G-190197 SH00004	
004	SDAFWP-LO-HTX	3065	SDAFW PUMP LUBE OIL HEAT EXCHANGER	HTX	TUR1-H13-3		N/A	N/A	N/A	G-190197 SH00004	SUPPORT FOR SDAFWP
005	SDAFWP-OIL-PMP	3065	STEAM DRIVEN AUX FEEDWATER PMP AUX OIL PMP	PMP	TUR1-H13-3		N/A	N/A	N/A	NONE	SUPPORT FOR SDAFWP
006	SDAFWP-PMP-TRT	3065	SDAFW PUMP THROTTLE VALVE	FCV	TUR1-J13-3		N/A	N/A	N/A	NONE	SUPPORT FOR SDAFWP
007	SDAFW-OIL-PMP-FLT	3065	STEAM DRIVEN AUX FEEDWATER PMP AUX OIL PMP FILTER	FLT	TUR1-H13-3		N/A	N/A	N/A	NONE	SUPPORT FOR SDAFWP
008	SDAFW-PMP-GOV	3065	STM DRVN AFW PUMP GOVERNOR	GOV	TUR1-J13-3		N/A	N/A	N/A	NONE	SUPPORT FOR SDAFWP
009	AFW-V2-14A	3065	SDAFW PUMP FW DISCH TO SG "A"	FCV	TUR2-G6-4		CLOSED	FAILS AS IS	OPEN	G-190197 SH00004	POWER OPERATED VALVE TO BE OPERATED MANUALLY
010	MS-V1-8A	3020	SG "A" STM SUPPLY TO STM DRIVEN AFW PUMP	ISV	TUR2-F14-25 PIPE JUNGLE		CLOSED	FAILS AS IS	OPEN	G-190196 SH00001	POWER OPERATED VALVE TO BE OPERATED MANUALLY
011	MS-V1-3A	3020	MSIV "A"	CHV	TUR3-G14-3		VARIABLE	CLOSED	CLOSED	G-190196 SH00001	POWER OPERATED VALVE TO BE OPERATED MANUALLY
012	RV1-1	3020	SG "A" STEAM LINE PORV	SRV	TUR3-F14-7		CLOSED	CLOSED	OPEN	G-190196 SH00001	
013	VALVES FOR ISOLATION/CONTROL OF PORTABLE BORATION PUMP							N/A			TO BE INSTALLED
014	BOR-INJ-TNK	2080	BORON INJECTION TANK	TNK	AUX1-Y17-0	NA	N/A	N/A	N/A	5379-1082 SH00001	
015	SI-870B	2080	BIT OUTLET	ISV	BIT ROOM	CLOSED	CLOSED	FAILS AS IS	OPEN	5379-1082 SH00001	POWER OPERATED VALVE TO BE OPERATED MANUALLY
016	STATION-A	5235	STATION BATTERIES A	BAT	AUX2	Batt Room D27	AVAILABLE	N/A	AVAILABLE	B-190628 SH00955, G-190190A, HBR2-08236	INCLUDING BATTERY RACKS
017	STATION-B	5235	STATION BATTERIES B	BAT	AUX2	Batt Room E29	AVAILABLE	N/A	AVAILABLE	B-190628 SH00956, G-190190A, HBR2-11891 SH 1	INCLUDING BATTERY RACKS
018	BAT-CHRGR-A	5235	STATION BATTERY CHARGER "A"	BYC	AUX2	Batt Room C28	AVAILABLE	N/A	AVAILABLE	B-190628 SH00955, G-190190A, HBR2-12645 SH 1+2	
019	BAT-CHRGR-B	5235	STATION BATTERY CHARGER "B"	BYC	AUX2	Batt Room C29	AVAILABLE	N/A	AVAILABLE	B-190628 SH00956, G-190190A, HBR2-12645 SH 1+2	
020	BAT-CHRGR-A-1	5235	STATION BATTERY CHARGER "A-1"	BYC	AUX2	Batt Room F28-0	AVAILABLE	N/A	AVAILABLE	B-190628 SH00955, G-190190A, HBR2-12645 SH 1+2	
021	BAT-CHRGR-B-1	5235	STATION BATTERY CHARGER "B-1"	BYC	AUX2	Batt Room E28-0	AVAILABLE	N/A	AVAILABLE	B-190628 SH00956, G-190190A, HBR2-12645 SH 1+2	
022	INVERTER-A	5235	INVERTER-A (7.5KVA)	IVT	AUX2	G26	AVAILABLE	N/A	AVAILABLE	HBR2-11472, G-190190A	
023	INVERTER-B	5235	INVERTER-B (7.5KVA)	IVT	AUX2	I29	AVAILABLE	N/A	AVAILABLE	HBR2-11472, G-190190A	
024	MCC-A	5235	125V DC LOAD CENTER "A"	MCC	AUX2	F27	AVAILABLE	N/A	AVAILABLE	B-190628 SH00955, G-190190A, 5379-02390	
025	MCC-B	5235	125V DC LOAD CENTER "B"	MCC	AUX2	F29	AVAILABLE	N/A	AVAILABLE	B-190628 SH00956, G-190190A, 5379-02390	
026	DP-A	5235	DISTRIBUTION PANEL A	PNL	AUX2	F27	AVAILABLE	N/A	AVAILABLE	B-190628 SH00955, G-190190A	BREAKERS: Ckt No. 11 (Type TM, 2 Poles, Trip 125A)
027	INST-2	5185	INSTRUMENT BUS 2	PNL	AUX2	J34-4	AVAILABLE	N/A	AVAILABLE	B-190627 SH00046, G-190190A	Breakers: Ckt No. 3, 6, 7, 8, 9, 11 and 18
028	INST-3	5185	INSTRUMENT BUS 3	PNL	AUX2	J30-4	AVAILABLE	N/A	AVAILABLE	B-190627 SH00047, G-190190A	BREAKERS: Ckt No. 8 and 25
029	INST-7A	5185	INSTRUMENT BUS 7A	PNL	AUX2	J33-4	AVAILABLE	N/A	AVAILABLE	HBR2-11466, G-190190A	BREAKERS: Ckt No. 4, 7, 8 and 9
030	INST-7B	5185	INSTRUMENT BUS 7B	PNL	AUX2	J33-4	AVAILABLE	N/A	AVAILABLE	HBR2-11466, G-190190A, B-190627 SH00050B	BREAKERS: Ckt No. 20, 25 and 35
031	INST-8	5185	INSTRUMENT BUS 8	PNL	AUX2	J30-4	AVAILABLE	N/A	AVAILABLE	B-190627 SH00051	BREAKERS: Ckt No. 9, 22 and 27
032	TE-423	1055	RCS LOOP B WIDE RANGE HOT LEG-TEMP RTD	TIN	CV1	K15	AVAILABLE	N/A	AVAILABLE	5379-03502, B-190628 SH00468, 5379-01971 SH00001, A-190301 SH00511A	POWER SOURCE: INST-7B Ckt No. 25 (via TM-577). INSTRUMENT LOOP: TI-423 (MN CNTRL BOARD), TM-423A (RACK-29), TE-423, TM-577
033	PT-511AA	1055	RVLIS TRAIN "A" WIDE RANGE RCS PRESS TRANSMITTER	PIN	AUX2	L24-2	AVAILABLE	N/A	AVAILABLE	B-190628 SH01281, A-190299 SH00511B, A-190301 SH00511A	POWER SOURCE: INST-7B Ckt No. 25 (via TM-577). INSTRUMENT LOOP: PT-511AA, TM-577
034	LI-475	3005	CH II SG 1 NARROW RANGE LEVEL INDICATOR	LIN	CRM	RTGB R37	AVAILABLE	N/A	AVAILABLE	B-190628 SH00418, 5379-03513, 5379-03440	HOST EQUIPMENT: MN CNTRL BOARD. POWER SOURCE: RACK-13. INSTRUMENT LOOP: LI-475, LT-475, CT-475, LQ-475, LM-475, LM-475A
035	LT-475	3005	MS SG A NARROW RANGE LEVEL TRANSMITTER	LIN	CV1	INST RACK 1B K8-6	AVAILABLE	N/A	AVAILABLE	G-190197 SH00004, 5379-03440, B-190628 SH00418	POWER SOURCE: RACK-13. INSTRUMENT LOOP: LI-475, LT-475, CT-475, LQ-475, LM-475, LM-475A
036	TI-579	1055	CH I CORE EXIT T/C & CORE COOLING MON	TIN	CRM	(PAM PNL A) 036	AVAILABLE	N/A	AVAILABLE	B-190628 SH01701, HBR2-11131 SH00004	HOST EQUIPMENT: PAMS-PANEL (Plasma display). POWER SOURCE: INST-7B Ckt No. 35
037	LI-1454B	3070	CST LEVEL INDICATOR	LIN	CRM	RTGB R36	Available	N/A	Available	A-190301 SH01453, B-190628 SH00601A, HBR2-11136 SH00005	Host Equipment: MN CNTRL BOARD. Power Source: INST-2 Ckt No. 18. Instrument Loop: LI-1454B, AUX-PNL-GD, LT-1454B, LM-1454B, PNL-B-65V/Q-4
038	LT-1454B	3070	CST LEVEL TRANSMITTER	LIN	TUR1	Y4-1	Available	N/A	Available	A-190301 SH01453, B-190628 SH00601A, B-190628 SH00966, G-190197 SH00001	Power Source: INST-2 Ckt No. 18. Instrument Loop: LI-1454B, AUX-PNL-GD, LT-1454B, LM-1454B, PNL-B-65V/Q-4
039	NI-52B	1045	WIDE RANGE NIS POWER INDICATOR (CHANNEL IV)	IDS	CRM	(PAM PNL B) 036	Available	Available	N/A	B-190628 SH00450D, HBR2-10010	HOST EQUIPMENT: PAMS-PANEL (NI-52A & NI-52B Indication). POWER SOURCE: INST-8 Ckt No. 22. INSTRUMENT LOOP: NI-52A, NI-52B, NM-52A, NM-52B, NM-52C
040	NM-52A	1045	EXCORE NEUTRON MONITOR AMPLIFIER(CHANNEL IV)	CNV	AUX1	X25-6	Available	Available	N/A	B-190628 SH00450C	POWER SOURCE: INST-8 Ckt No. 22. INSTRUMENT LOOP: NI-52A, NI-52B, NM-52A, NM-52B, NM-52C
041	NM-52B	1045	EXCORE NEUTRON MONITOR SIG PROCESSOR(CHANNEL IV)	CNV	AUX1	25-3	Available	Available	N/A	B-190628 SH00450C	POWER SOURCE: INST-8 Ckt No. 22. INSTRUMENT LOOP: NI-52A, NI-52B, NM-52A, NM-52B, NM-52C
042	NM-52C	1045	EXCORE NEUTRON MON ISOL EXP MODULE(CHANNEL IV)	CNV	AUX1	25-3	Available	Available	N/A	B-190628 SH00450C	POWER SOURCE: INST-8 Ckt No. 22. INSTRUMENT LOOP: NI-52A, NI-52B, NM-52A, NM-52B, NM-52C
043	EI-210A	5235	125 DC MCC-A BUS VOLTAGE INDICATOR	IDS	AUX2	MCC-A F27	AVAILABLE	N/A	AVAILABLE	5379-02392, B-190628 SH00955	HOST EQUIPMENT: MCC-A. THIS INSTRUMENT IS A LOCAL INDICATOR.

ESEL

ESEL #	Equipment ID	System	Description	Equipment Type	Building/Location	Room or Row/ Column	Equipment Normal State	Equipment Failed State	Equipment Desired State	Drawing	Notes
044	EI-210B	5235	125 DC MCC-B BUS VOLTAGE INDICATOR	IDS	AUX2	MCC-B F29	AVAILABLE	N/A	AVAILABLE	5379-02392, B-190628 SH00956	HOST EQUIPMENT: MCC-B. THIS INSTRUMENT IS A LOCAL INDICATOR.
045	LI-460	2005	PZR LEVEL	LIN	CRM	RTGB Q37	AVAILABLE	N/A	AVAILABLE	5379-03530, B-190628 SH00460	HOST EQUIPMENT: MN CNTRL BOARD. POWER SOURCE: RACK-11. INSTRUMENT LOOP: LI-460, LT-460, CT-460, LQ-460, LM-460
046	LT-460	2005	PZR LEVEL TRANSMITTER	LIN	CV1	P18-3	AVAILABLE	N/A	AVAILABLE	5379-03530, B-190628 SH00460, A-190301 SH00460, G-190293 SH00001	POWER SOURCE: RACK-11. INSTRUMENT LOOP: LI-460, LT-460, CT-460, LQ-460, LM-460
047	FI-1426A	3065	AFW SDAFW FLOW TO SG A	FIN	CRM	RTGB R36	AVAILABLE	N/A	AVAILABLE	G-190198 SH00623A, A-190301 SH01426A	HOST EQUIPMENT: MN CNTRL BOARD. POWER SOURCE: INST-7B Ckt No. 20. INSTRUMENT LOOP: FI-1426A, FT-1426A, FY-1426A, FE-1426A
048	FT-1426A	3065	SDAFW PMP DISCH TO SG A FLOW TRANSMITTER	FIN	TUR1	G7-4	AVAILABLE	N/A	AVAILABLE	G-190198 SH00623A, A-190301 SH01426A, A-190299 SH01426A, G-190181A, G-190197 SH00004	POWER SOURCE: INST-7B Ckt No.20. INSTRUMENT LOOP: FI-1426A, FT-1426A, FY-1426A, FE-1426A
049	PI-952	2118	CH II CONTAINMENT PRESSURE INDICATOR	PIN	CRM	RTGB Q37	AVAILABLE	N/A	AVAILABLE	A-190301 SH00952, B-190628 SH 00496, 5379-03503	HHOST EQUIPMENT: MN CNTRL BOARD. POWER SOURCE: RACK-12. INSTRUMENT LOOP: PI-952, PT-952, CT-952, PQ-952, PM-952
050	PT-952	2118	SIS CV PRESSURE TRANSMITTER	PIN	AUX2	K24-5	AVAILABLE	N/A	AVAILABLE	A-190299 SH00145, 5379-03503, B-190628 SH 00496	POWER SOURCE: RACK-12. INSTRUMENT LOOP: PI-952, PT-952, CT-952, PQ-952, PM-952
051	LI-511AB	1055	RV FULL RANGE LEVEL INDICATOR	LIN	CRM	PAMS PNL O36	AVAILABLE	N/A	AVAILABLE	A-190301 SH00511A	HOST EQUIPMENT: PAMS-PANEL. POWER SOURCE: INST-7B Ckt No. 25 (via TM-577). INSTRUMENT LOOP: LI-511AB, LT-511AB, TM-577
052	LT-511AB	1055	RVLIS TRAIN "A" NARROW RANGE LEVEL TRANSMITTER	LIN	AUX2	L24-3	AVAILABLE	N/A	AVAILABLE	A-190301 SH00511A, HBR2-09067	POWER SOURCE: INST-7B Ckt No. 25 (via TM-577). INSTRUMENT LOOP: LI-511AB, LT-511AB, TM-577
053	LI-920	2080	ACCUMULATOR "A" LVL INDICATOR	LIN	CRM	RTGB Q37	AVAILABLE	N/A	AVAILABLE	A-190301 SH00920, B-190628 SH00004	HOST EQUIPMENT: MN CNTRL BOARD. POWER SOURCE: RACK-4. INSTRUMENT LOOP: LI-920, LT-920, LC-920, LQ-920
054	LT-920	2080	SIS ACCUMULATOR A LVL TRANSMITTER	LIN	CV2	H25-3	AVAILABLE	N/A	AVAILABLE	A-190299 SH00920, 5379-00251, A-19301 SH00051A, 5379-1082 SH00005	POWER SOURCE: RACK-4. INSTRUMENT LOOP: LI-920, LT-920, LC-920, LQ-920
055	PI-474	3020	CH 1 SG "A" STEAM PRESS INDICATOR	PIN	CRM	RTGB R37	AVAILABLE	N/A	AVAILABLE	5379-03488, B-190628 SH00429	HOST EQUIPMENT: MN CNTRL BOARD. POWER SOURCE: RACK-13. INSTRUMENT LOOP: PI-474, PT-474, CT-474, PQ-474, PM-474A
056	PT-474	3020	MS SG A STEAM PRESS TRANSMITTER	PIN	TUR2	G14-3	AVAILABLE	N/A	AVAILABLE	G-190196 SH00001, G-190182A, G-190292	POWER SOURCE: RACK-13. INSTRUMENT LOOP: PI-474, PT-474, CT-474, PQ-474, PM-474A
057	LM-1454B	3070	CST LEVEL SIGNAL ISOLATOR	LIN	AUX2	H34	Available	N/A	Available	A-190301 SH01453, B-190628 SH00601A, B-190628 SH00966,	Power Source: INST-2 Ckt No. 18. Instrument Loop: LI-1454B, AUX-PNL-GD, LT-1454B, LM-1454B, PNL-B-65V/Q-4
058	NOT USED										
059	TI-950B	2118	CV AVERAGE TEMPERATURE INDICATOR	TIN	CRM	RTGB Q37	AVAILABLE	N/A	AVAILABLE	B-190628 SH00044, HBR2-11133 SH00008	HOST EQUIPMENT: MN CNTRL BOARD. POWER SOURCE: INST-8 Ckt No. 27. INSTRUMENT LOOP: TI-950B, TT-950B, TQ-950B, TC-1A, TC-2A, TC-3A, TC-4A, TC-5A
060	TT-950B	2118	CV TEMPERATURE TRANSMITTER	TIN	CV2	SEAL TABLE ROOM N35-5	AVAILABLE	N/A	AVAILABLE	B-190628 SH00044	POWER SOURCE: INST-8 Ckt No. 27. INSTRUMENT LOOP: TI-950B, TT-950B, TQ-950B, TC-1A, TC-2A, TC-3A, TC-4A, TC-5A
061	RACK-4	1080	HAGAN RACK NO.4	RAI	CRM	T35-0	AVAILABLE	N/A	AVAILABLE	5379-01994, 5379-02047	POWER SOURCE: INST-2 Ckt No. 3 & INST-7A Ckt No. 4. SUBCOMPONENTS: LC-920, LQ-920
062	RACK-11	1080	HAGAN RACK NO.11	RAI	CRM	T35-0	AVAILABLE	N/A	AVAILABLE	5379-02001, 5379-02059	POWER SOURCE: INST-2 Ckt No. 6 & INST-7A Ckt No. 7. SUBCOMPONENTS: CT-460, LQ-460, LM-460
063	RACK-12	1080	HAGAN RACK NO.12	RAI	CRM	T35-0	AVAILABLE	N/A	AVAILABLE	B-190628 SH00497, 5379-02002, 5379-02056	POWER SOURCE: INST-2 Ckt No. 7 & INST-7A Ckt No. 8. SUBCOMPONENTS: CT-952, PQ-952, PM-952
064	RACK-13	1080	HAGAN RACK NO.13	RAI	CRM	T35-0	AVAILABLE	N/A	AVAILABLE	5379-02003, 5379-02046	POWER SOURCE: INST-2 Ckt No. 8 & INST-7A Ckt No. 9. SUBCOMPONENTS: CT-474, CT-475, LM-475, LQ-475, PM-474A, PQ-474
065	RACK-29	1080	HAGAN RACK NO.29	RAI	CRM	T35-0	AVAILABLE	N/A	AVAILABLE	HBR2-10980, HBR2-11267	POWER SOURCE: INST-2 Ckt No. 3. SUBCOMPONENTS: LM-475A, LM-477, TM-423A
066	MN CNTRL BOARD	6010	MAIN CONTROL BOARD (RTGB)	CAB	RAB	MAIN CONTROL ROOM	AVAILABLE	N/A	AVAILABLE	HBR2-11133 THRU HBR2-11138 SH00001-SH00009	POWER SOURCE: INST-2 Ckt No. 9, INST-3 Ckt No. 8 & INST-8 Ckt No. 9. SUBCOMPONENTS: FI-1426A, LI-460, LI-475, LI-920, PI-474, PI-952, TI-423, TI-950B
067	PAMS-PANEL	2116	POST ACCIDENT MONITORING PANEL	PNL	CRM	O36-0	AVAILABLE	N/A	AVAILABLE	B-190628 SH01701, HBR2-10010, HBR2-10143	POWER SOURCE: INST-7B Ckt No. 35. SUBCOMPONENTS: LI-511AB, NI-52A, NI-52B, TI-579
068	TM-577	1055	TEMP MONITOR FOR TE-413-2	MON		CET/CCM CAB	AVAILABLE	N/A	AVAILABLE	B-190628 SH00468, 5379-03502, A-190301 SH00511A	POWER SOURCE: INST-7B Ckt No. 25.
069	FLEX DIESEL GENERATOR										ADDED PER DESIGN INPUT LETTER REV 2
070	AUX-PNL-GD	1080	AUXILIARY DC PANEL GD (45V DC)	PNL	AUX2	I32	Available	N/A	Available	B-190628 SH00601A, B-190628 SH00966, B-190629 SH00041	Power Source: INST-2 Ckt No. 18. Instrument Loop: LI-1454B, AUX-PNL-GD, LT-1454B, LM-1454B, PNL-B-65V/Q-4
071	Q-4	3070	65VDC DISTRUBUTION PNL B POWER SUPPLY	PWR	AUX2	H34-3 DIST PNL B	Available	N/A	Available	A-190301 SH01453, B-190628 SH00601A, B-190628 SH00966	Host Equipment: 65V DISTRIUTION PNL B. Power Source: INST-2 Ckt No. 18. Instrument Loop: LI-1454B, AUX-PNL-GD, LT-1454B, LM-1454B, PNL-B-65V/Q-4

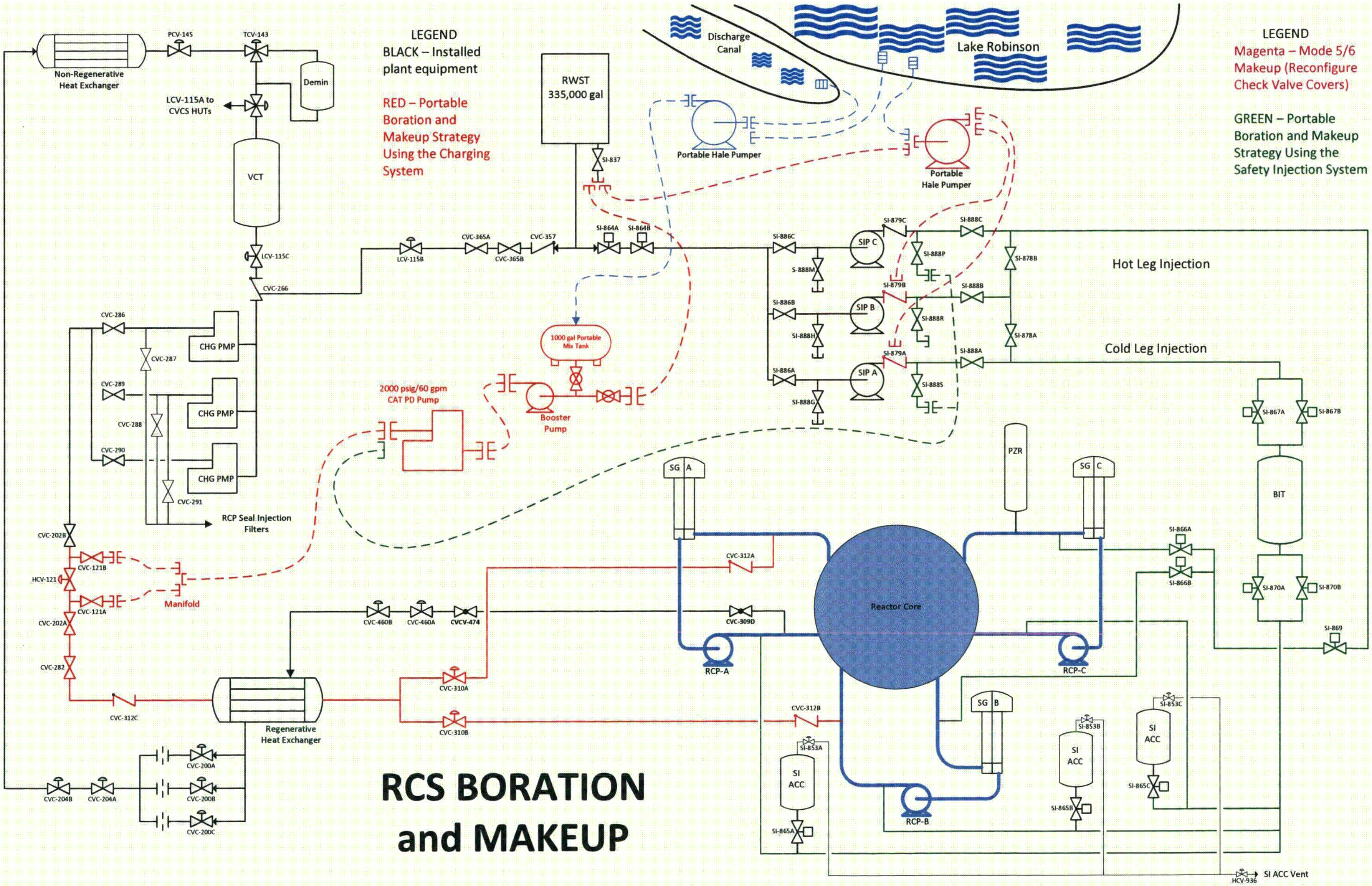
Attachment B – H.B. Robinson Steam Electric Plant Unit 2 RCS Cooling Strategies



Attachment B
Page 1 of 1

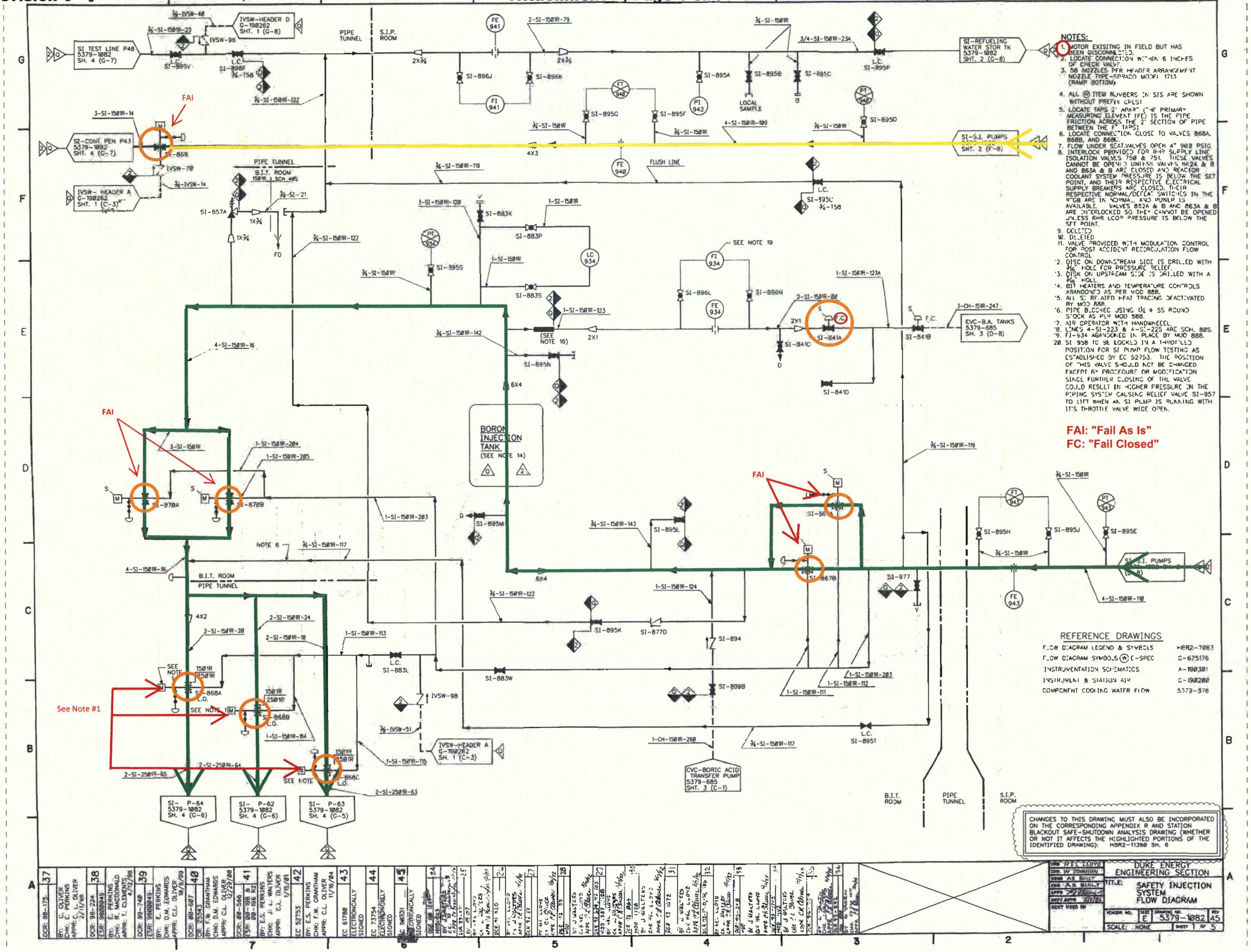
RCS COOLING SEISMIC STRATEGIES

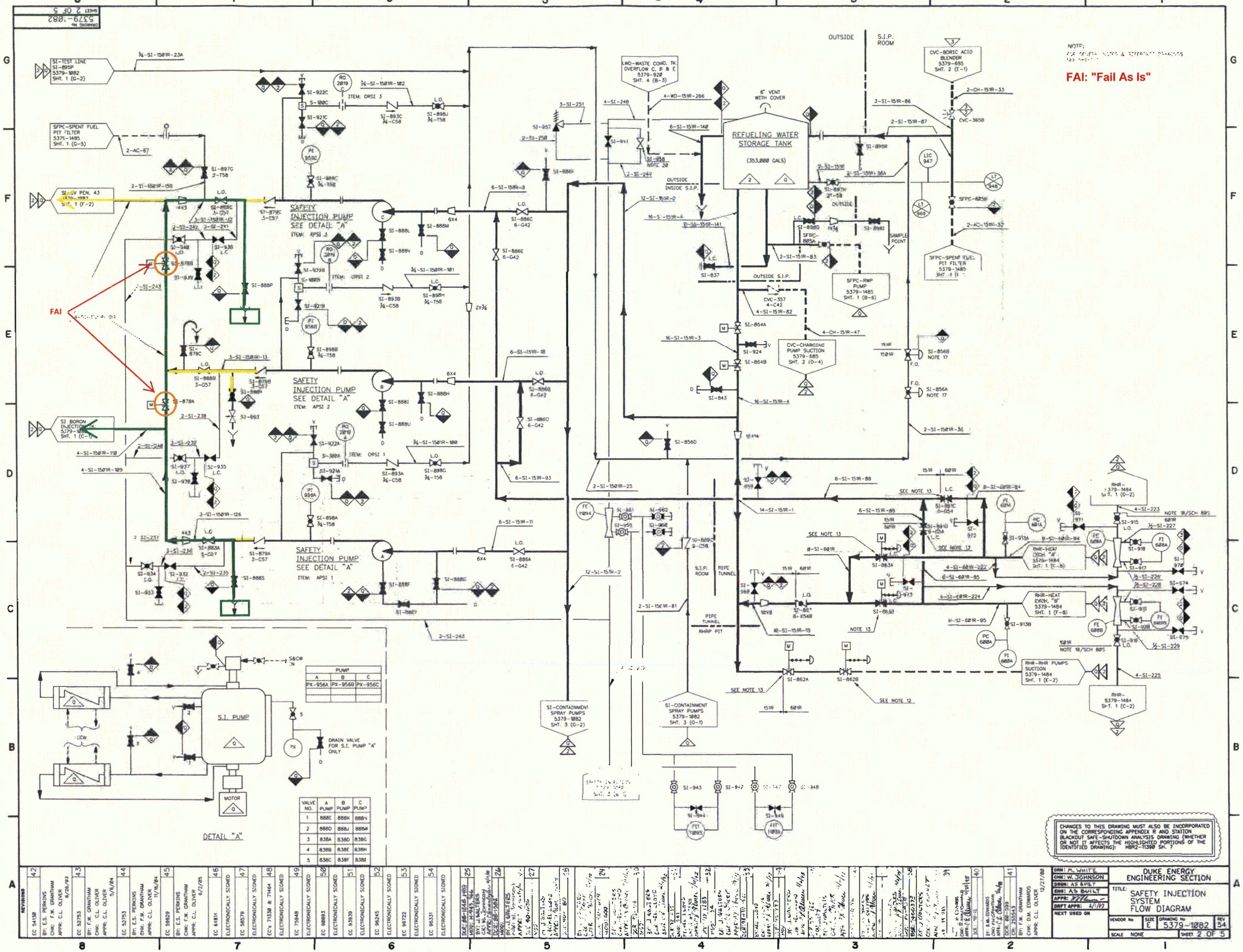
Attachment C – H.B. Robinson Steam Electric Plant Unit 2 RCS Boration and Makeup Strategies

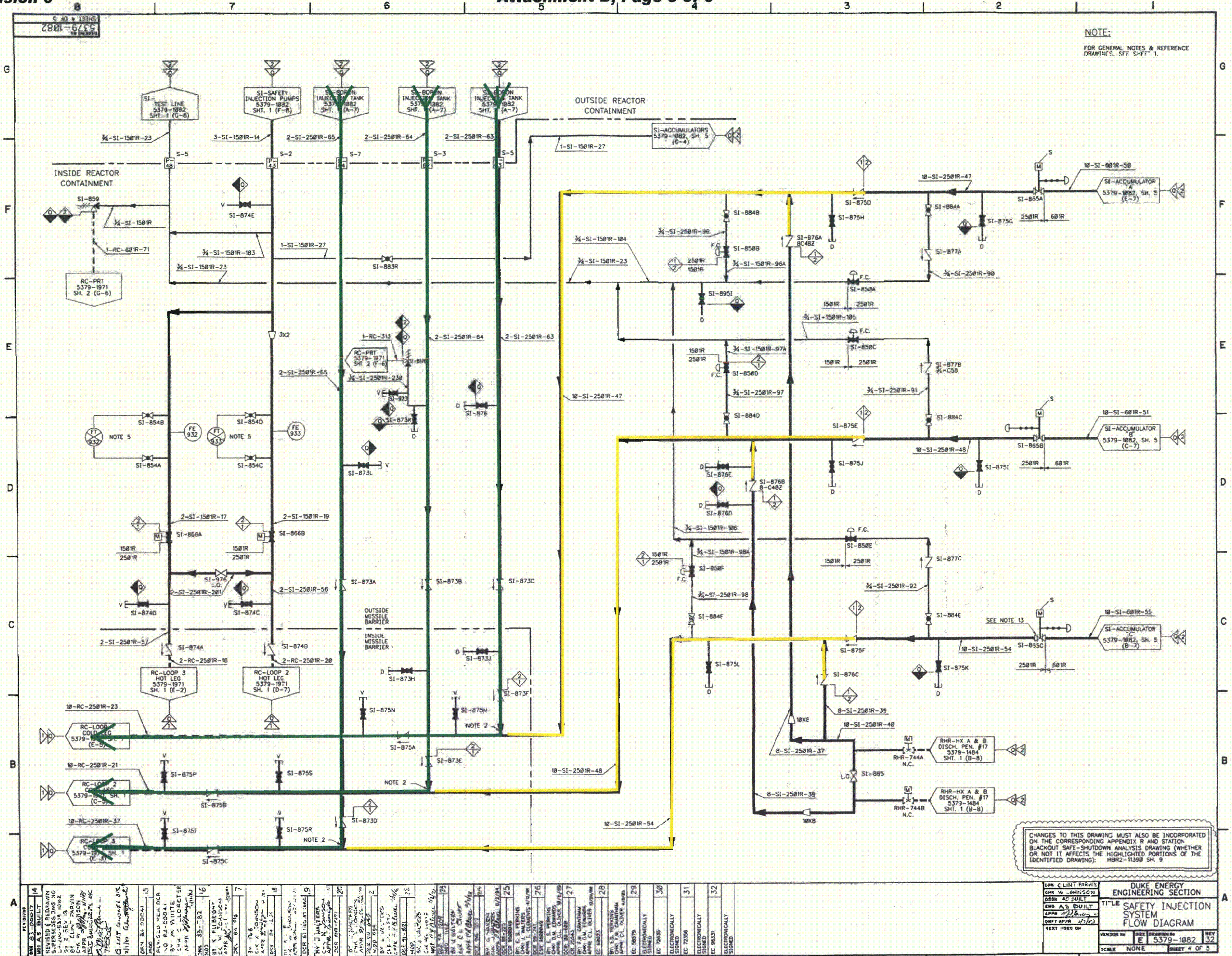


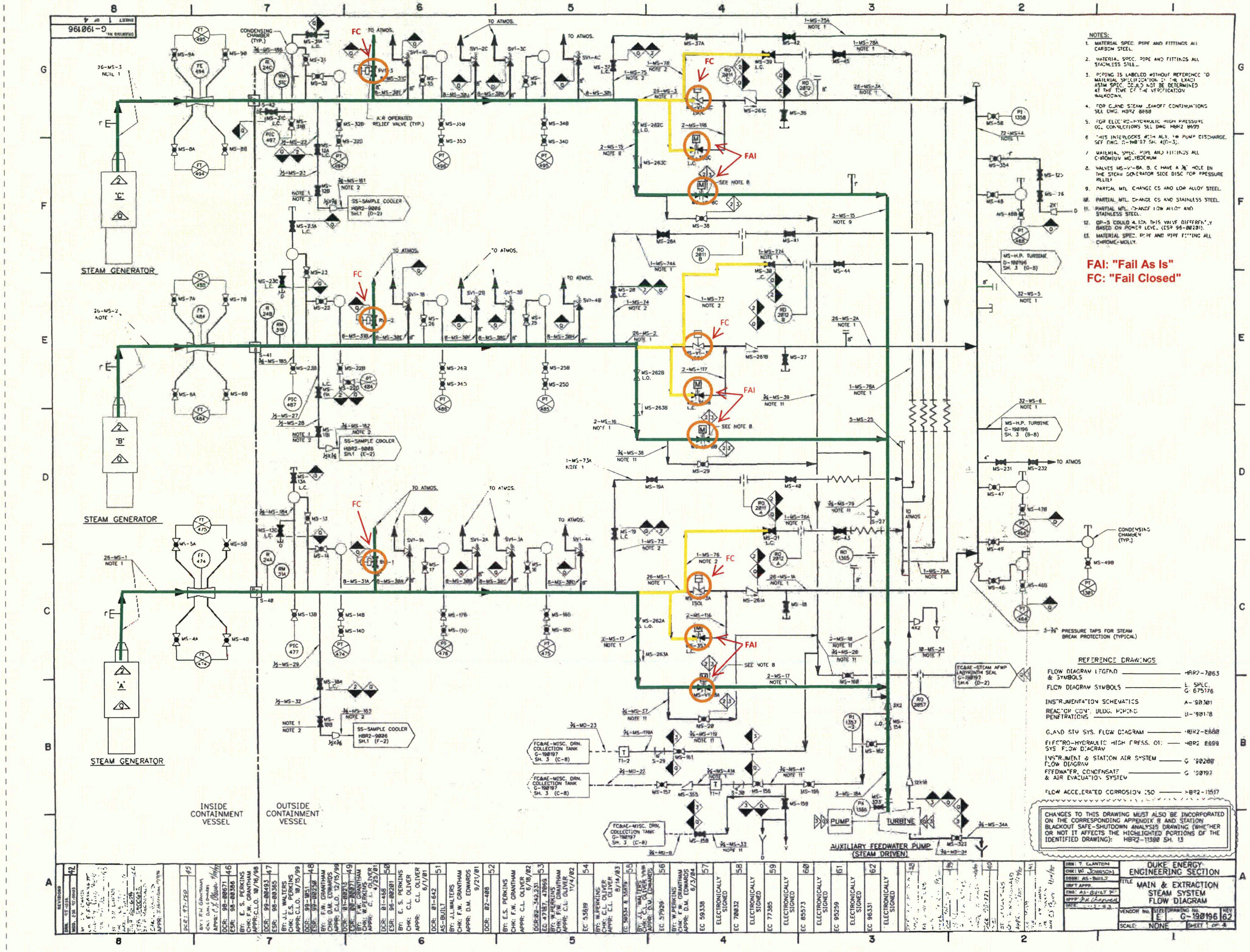
RCS BORATION and MAKEUP

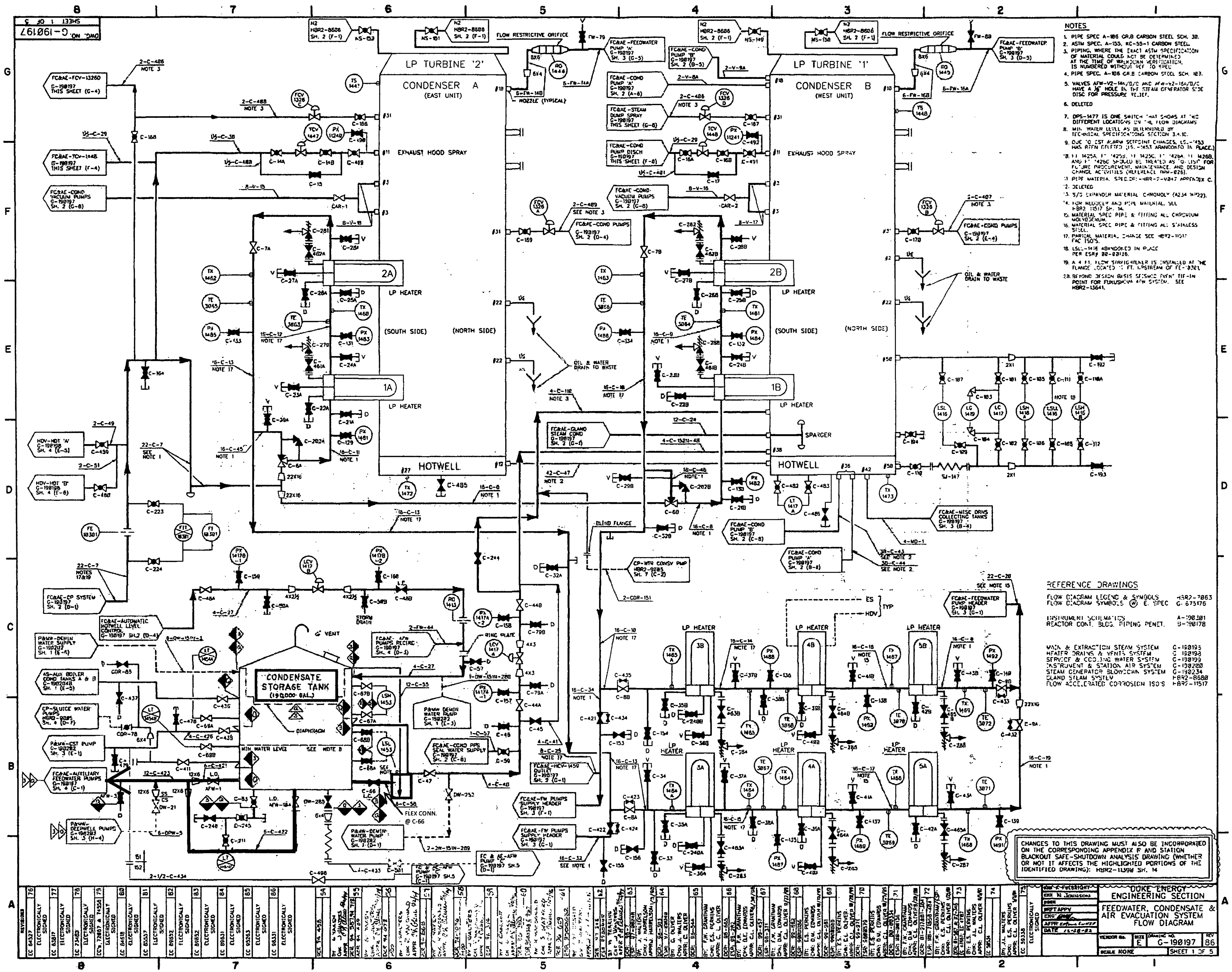
Attachment D – H.B. Robinson Steam Electric Plant FLEX Flow Path

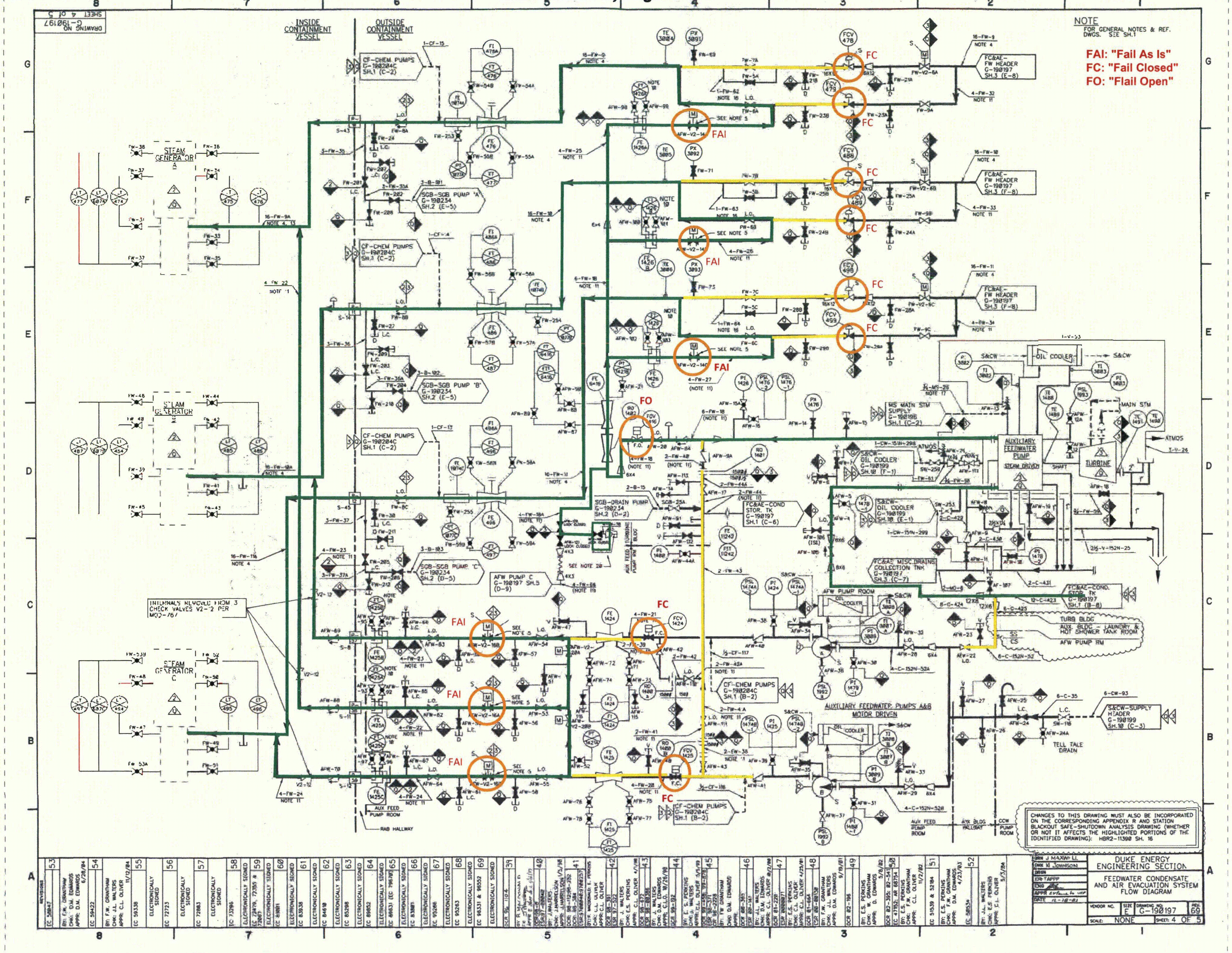

NOTES:
FAI: "Fail As Is"
FC: "Fail Closed"
FAI
See Note #1
BORON INJECTION TANK (SEE NOTE 14)
SI-S.I. PUMPS SHT. 2 (F-8)
SI-REFUELING WATER STOR TK 5379-1082 SHT. 2 (G-8)
SI-CONT PEN P43 5379-1082 SHT. 4 (C-7)
SI TEST LINE P48 5379-1082 SH. 4 (G-7)
IVSW-HEADER D G-190262 SHT. 1 (G-8)
IVSW-HEADER A G-190262 SHT. 1 (C-3)
CVC-B.A. TANKS 5379-685 SH. 3 (D-8)
CVC-BORIC ACID TRANSFER PUMP 5379-685 SHT. 3 (C-1)
SI- P-64 5379-1082 SH. 4 (G-6)
SI- P-62 5379-1082 SH. 4 (G-6)
SI- P-63 5379-1082 SH. 4 (G-5)
PIPE TUNNEL
S.I.P. ROOM
B.I.T. ROOM
REFERENCE DRAWINGS
FLOW DIAGRAM LEGEND & SYMBOLS HBR2-7063
INSTRUMENTATION SCHEMATICS A-190301
COMPONENT COOLING WATER FLOW 5379-376
CHANGES TO THIS DRAWING MUST ALSO BE INCORPORATED ON THE CORRESPONDING APPENDIX R AND STATION BLACKOUT SAFE-SHUTDOWN ANALYSIS DRAWING (WHETHER OR NOT IT AFFECTS THE HIGHLIGHTED PORTIONS OF THE IDENTIFIED DRAWING): HBR2-11390 SH. 6
DUKE ENERGY ENGINEERING SECTION
SAFETY INJECTION SYSTEM FLOW DIAGRAM
5379-1082
SCALE: NONE


FAI: "Fail As Is"
FAI
REFUELING WATER STORAGE TANK
(353,000 GALS)
SAFETY INJECTION PUMP
SEE DETAIL "A"
S.I. PUMP
MOTOR
DETAIL "A"
DRAIN VALVE FOR S.I. PUMP "A" ONLY
OUTSIDE
S.I.P. ROOM
DUKE ENERGY ENGINEERING SECTION
SAFETY INJECTION SYSTEM FLOW DIAGRAM
5379-1082
SHEET 2 OF 5


NOTE:
FOR GENERAL NOTES & REFERENCE DRAWINGS, SEE SHEET 1.
OUTSIDE REACTOR CONTAINMENT
INSIDE REACTOR CONTAINMENT
OUTSIDE MISSILE BARRIER
INSIDE MISSILE BARRIER
CHANGES TO THIS DRAWING MUST ALSO BE INCORPORATED ON THE CORRESPONDING APPENDIX R AND STATION BLACKOUT SAFE-SHUTDOWN ANALYSIS DRAWING (WHETHER OR NOT IT AFFECTS THE HIGHLIGHTED PORTIONS OF THE IDENTIFIED DRAWING): HBR2-11390 SH. 9
DUKE ENERGY
ENGINEERING SECTION
SAFETY INJECTION SYSTEM
FLOW DIAGRAM
5379-1082
SHEET 4 OF 5


FAI: "Fail As Is"
FC: "Fail Closed"
FC
FAI
STEAM GENERATOR
'A'
'B'
'C'
INSIDE CONTAINMENT VESSEL
OUTSIDE CONTAINMENT VESSEL
TO ATMOS.
AIR OPERATED RELIEF VALVE (TYP.)
SS-SAMPLE COOLER
SEE NOTE 8
AUXILIARY FEEDWATER PUMP (STEAM DRIVEN)
PUMP
TURBINE
CONDENSING CHAMBER (TYP.)
MS-H.P. TURBINE
PRESSURE TAPS FOR STEAM BREAK PROTECTION (TYPICAL)
NOTES:
REFERENCE DRAWINGS
FLOW DIAGRAM LEGEND & SYMBOLS
FLOW DIAGRAM SYMBOLS
INSTRUMENTATION SCHEMATICS
REACTOR CONT. BLDG. PIPING PENETRATIONS
GLAND STM SYS. FLOW DIAGRAM
ELECTRO-HYDRAULIC HIGH PRESS. OIL SYS FLOW DIAGRAM
INSTRUMENT & STATION AIR SYSTEM FLOW DIAGRAM
FEEDWATER, CONDENSATE & AIR EVACUATION SYSTEM
FLOW ACCELERATED CORROSION ISO
CHANGES TO THIS DRAWING MUST ALSO BE INCORPORATED ON THE CORRESPONDING APPENDIX R AND STATION BLACKOUT SAFE-SHUTDOWN ANALYSIS DRAWING (WHETHER OR NOT IT AFFECTS THE HIGHLIGHTED PORTIONS OF THE IDENTIFIED DRAWING): HBR2-11390 SH. 13
DUKE ENERGY ENGINEERING SECTION
MAIN & EXTRACTION STEAM SYSTEM FLOW DIAGRAM
G-190196
SCALE: NONE
ELECTRONICALLY SIGNED


LP TURBINE '2'
CONDENSER A
(EAST UNIT)
LP TURBINE '1'
CONDENSER B
(WEST UNIT)
EXHAUST HOOD SPRAY
FLOW RESTRICTIVE ORIFICE
NOZZLE (TYPICAL)
LP HEATER
(SOUTH SIDE)
(NORTH SIDE)
OIL & WATER DRAIN TO WASTE
HOTWELL
SPARGER
BLIND FLANGE
CONDENSATE STORAGE TANK
(190,000 GAL.)
DIAPHRAGM
MIN WATER LEVEL SEE NOTE 8
FLEX CONN. @ C-66
STORM DRAIN
RING PLATE
NOTES
1. PIPE SPEC A-106 GR.B CARBON STEEL SCH. 30.
2. ASTM SPEC. A-155, KC-55-1 CARBON STEEL.
3. PIPING, WHERE THE EXACT ASTM SPECIFICATION OF MATERIAL COULD NOT BE DETERMINED AT THE TIME OF WALKDOWN VERIFICATION, IS NUMBERED WITHOUT REF. TO SPEC.
4. PIPE SPEC. A-106 GR.B CARBON STEEL SCH. 160.
5. VALVES AFW-V2-14A/B/C AND AFW-V2-16A/B/C HAVE A 1/16" HOLE IN THE STEAM GENERATOR SIDE DISC FOR PRESSURE RELIEF.
6. DELETED
7. DPS-1477 IS ONE SWITCH THAT SHOWS AT TWO DIFFERENT LOCATIONS ON THE FLOW DIAGRAMS
8. MIN. WATER LEVEL AS DETERMINED BY TECHNICAL SPECIFICATIONS SECTION 3.4.10.
9. DUE TO CST ALARM SETPOINT CHANGES, LSL-1453 HAS BEEN DELETED (LSL-1453 ABANDONED IN PLACE.)
10. FT 1425A, FT 1425B, FT 1425C, FT 1426A, FT 1426B, AND FT 1426C SHOULD BE TREATED AS "Q-LIST" FOR FUTURE PROCUREMENT, MAINTENANCE, AND DESIGN CHANGE ACTIVITIES (REFERENCE DMM-026).
11. PIPE MATERIAL SPEC. DF: HBR-2-4047 APPENDIX C.
12. DELETED
13. S/G EXPANDER MATERIAL CHROMOLY (A234 WP22).
14. FOR REDUCER AND PIPE MATERIAL SEE HBR2 11517 SH. 14.
15. MATERIAL SPEC PIPE & FITTING ALL CHROMIUM MOLYBDENUM.
16. MATERIAL SPEC PIPE & FITTING ALL STAINLESS STEEL.
17. PARTIAL MATERIAL CHANGE SEE HBR2-11517 FAC ISO'S.
18. LSLL-1416 ABANDONED IN PLACE PER ESR# 00-00126.
19. A 4 FT. FLOW STRAIGHTENER IS INSTALLED AT THE FLANGE LOCATED 11 FT. UPSTREAM OF FE-10301
20. BEYOND DESIGN BASIS SEISMIC EVENT TIE-IN POINT FOR FUKUSHIMA AFW SYSTEM. SEE HBR2-13641.
REFERENCE DRAWINGS
CHANGES TO THIS DRAWING MUST ALSO BE INCORPORATED ON THE CORRESPONDING APPENDIX F AND STATION BLACKOUT SAFE-SHUTDOWN ANALYSIS DRAWING (WHETHER OR NOT IT AFFECTS THE HIGHLIGHTED PORTIONS OF THE IDENTIFIED DRAWING): HBR2-11598 SH. 14
DUKE ENERGY
ENGINEERING SECTION
FEEDWATER, CONDENSATE & AIR EVACUATION SYSTEM FLOW DIAGRAM
SIZE E
DRAWING NO. G-190197
REV 86
SCALE NONE
SHEET 1 OF 5
ELECTRONICALLY SIGNED


FAI: "Fail As Is"
FC: "Fail Closed"
FO: "Flail Open"
FC
FAI
FO
INSIDE CONTAINMENT VESSEL
OUTSIDE CONTAINMENT VESSEL
STEAM GENERATOR A
STEAM GENERATOR C
SGB-SGB PUMP "A"
SGB-SGB PUMP "B"
SGB-SGB PUMP "C"
CF-CHEM PUMPS
AFW PUMP C
AUXILIARY FEEDWATER PUMP STEAM DRIVEN
AUXILIARY FEEDWATER PUMPS A&B MOTOR DRIVEN
TURBINE
OIL COOLER
MAIN STM
ATMOS
AFW PUMP ROOM
TURB BLDG
AUX BLDG - LAUNDRY & HOT SHOWER TANK ROOM
TELL TALE DRAIN
INTERNALS REMOVED FROM 3 CHECK VALVES V2-2 PER MOD-767
CHANGES TO THIS DRAWING MUST ALSO BE INCORPORATED ON THE CORRESPONDING APPENDIX R AND STATION BLACKOUT SAFE-SHUTDOWN ANALYSIS DRAWING (WHETHER OR NOT IT AFFECTS THE HIGHLIGHTED PORTIONS OF THE IDENTIFIED DRAWING): HBR2-11380 SH. 16
DUKE ENERGY ENGINEERING SECTION
FEEDWATER CONDENSATE AND AIR EVACUATION SYSTEM FLOW DIAGRAM
G-190197
SHEET 4 OF 5
SCALE: NONE