February 3, 1994

Docket No. 52-001

Mr. Joseph Quirk
GE Nuclear Energy
175 Curtner Avenue Mail Code - 782
San Jose, California 95125

Dear Mr. Quirk:

SUBJECT: INDEPENDENT QUA GROUP COMMENTS ON THE GENERAL ELECTRIC (GE) ADVANCED P ER REACTOR CERTIFIED DESIGN MATERIAL AND STANDARD SAFET REPORT

As a result of the Indepen ity Review Group effort, the inspections, test, analyses, and acceptance iteria (ITAAC) review team members have determined that the group's comments listed in the Enclosure 1 require disposition by GE. Enclosure 2 contains the comments.

Please provide your disposition of the comments within two weeks to allow prompt review and resolution by the ITAAC review team. If you have any questions, please contact Kris Shembarger at (301) 504-1114.

Sincerely,

(Original signed by)

R. W. Borchardt, Director
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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Enclosure 1

INDEPENDENT REVIEW GROUP COMMENTS TO BE SENT TO GE

Section	Comment No.
2.1.1	4
2.1.2	1, 2, 4, 5, 6, 7
2.1.3	1
2.2.1	3, 7
2.2.2	1, 3, 4
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2.2.5	1, 2
2.2.7	1, 2, 6
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2.4.4	4, 5, 6
2.6.1	2, 3, 4, 5
2.6.2	6
2.6.3	1, 3, 5
2.8.1	New Comment
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2.10.1	2
2.10.2	3,4
2.10.7	1
2.10.9	1, 2d
2.10.21	1, 2, 3
2.10.22	2, 3
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2.11.1	1, 2, 4
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2.11.6	1, 2
2.11.9	1, 2, 3, 4
2.11.11	1
2.11.13	1
2.11.21	1
2.12.1	1, 2, 3, 4, 8, 9, 10, [SSAR Sec 8.2 (No. 1, 2)]
2.12.10	3
2.12.11	6, 8
2.12.12	1, 2
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2.14.4	1, 2, 3
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Section	Comment No.
2.14.8	2, 3
2.15.5	1, 3, 4, 12, 13, 14
2.15.6	1
3.1	3, 5, 6, 7, 9, 10, 12, 13, 15
3.2	4b, 4c, 4d, 4f, 4h
3.3	1
3.4	2(1), 2(4), 5, 6
MISC.	1, 2, 3, 4, 5, 6

INDEPENDENT REVIEW GROUP COMMENTS REQUIRING DISPOSITION BY GE

Section 2.1.1 Comment No. 4

Comment:

Correct typographical error on attached page 5.3-7 of SSAR.

Resolution:

Enclosure 2

- Second Capsule: After 20 effective full-power years.
- Third Capsule: With an exposure not to exceed the peak EOL fluence.
- Fourth Capsule: Schedule determined based on results of first two capsules per ASTM E-185, Paragraph 7.6.2 (see Section 5.3.4.2 for additional capsule requirements). Fracture toughness testing of irradiated capsule specimens will be in accordance with requirements of ASTM E-185 as called out for by 10CFR50 Appendix H.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Subsections 4.1.4.5 and 4.3.2.2.

5.3.1.6.3 Predicted Irradiation Effects on Beltline Materials

Transition temperature changes and changes in upper-shelf energy shall be calculated in accordance with the rules of Regulatory Guide 1.99. Reference temperatures shall be established in accordance with 10CFR50 Appendix G and NB-2330 of the ASME Code.

Since weld material chemistry and fracture toughness data are not available at this time, the limits in the purchase specification were used to estimate worst-case irradiation effects.

These estimates show that the adjusted reference temperature at end-of-life is less than 34°C, and the end-of-life USE exceeds 69 kg-m (see response to Question 251.5 for the calculation and analysis associated with this estimate). 6.9 ?

5.3.1.6.4 Positioning of Surveillance Capsules and Methods of Attachment Appendix H.II B (2)

The surveillance specimen holders, described in Subsections 5.3.1.6.1 and 3.9.5.1.2.10, are located at different azimuths at common elevation in the core beltline region. The locations are selected to produce lead factor of approximately 1.2 to 1.5 for the inserted specimen capsules. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel. The capsules can be removed form and reinserted into the surveillance specimen holders. See Subsection 5.3.4.2 for COL license information requirements pertaining to the surveillance material, lead factors, withdrawal schedule and neutron fluence levels.

In areas where brackets (such as the surveillance specimen holder brackets) are located, additional nondestructive examinations are performed on the vessel base metal and stainless steel weld-deposited cladding or weld-buildup pads during vessel manufacture. The base metal is ultrasonically examined by straight-beam techniques to a depth at least equal to the thickness of the bracket being joined. The area examined is the area

Section 2.1.2 Comment No. 1

Comment:

Fig. 2.1.2b: The piping symbol for the drain line piping (shown horizontally inside and outside the containment) is NNS though this piping is Class 1. This should be corrected.

Resolution:

Section 2.1.2 Comment No. 2

Comment:

The design description for the ADS states that the high drywell pressure bypass timer is less than or equal to 8 minutes which is in conflict with SSAR Table 6.3-1 which shows this value to be >= 8 minutes.

Resolution:

Section 2.1.2 Comment No. 4

Comment:

In Fig. 2.1.2b and 2.1.2e, the temperature element should be shown as T, not TE, or the symbol for temperature in App. A should be changed to TE.

Resolution:

Section 2.1.2 Comment No. 5

Comment:

All figure numbers in Table 2.1.2 and some figure numbers in the description are incorrect.

Resolution:

Section 2.1.2 Comment No. 6

Comment:

In the ITA column for entry #6, the word "conducted" should be added after "...MSIV will be ..."

Resolution:

Section 2.1.2 Comment No. 7

Comment:

The design description (page 2.1.2-4) should include the requirement that the maximum elapsed time between receiving the overpressure signal at the valve actuator and actual start of SRV motion will not exceed 0.1 sec (SSAR 5.2.2.4.1) and be verified by an ITAAC entry.

Resolution:

Section 2.1.3 Comment No. 1

Comment:

The minimum dry rotating inertia (17.5 Kg-m2) of the RIP stated in the design description conflicts with the value of 19.5 kg-m2 stated in SSAR Table 5.4-1. The description and ITAAC entry #4 should be revised.

Resolution:

Section 2.2.1 Comment No. 3

Comment:

ITAAC design commitment #4 references "automatic thermal power monitor". Design description (2) describes a "automatic thermal limit monitor" and SSAR 7.7.1.2.1 references an "automatic thermal limit monitor, page 7.7-15. The terms need to be made consistent. ITAAC #4 should read "automatic thermal power limit".

Resolution:

Section 2.2.1 Comment No. 7

Comment:

The wording for SAR section 7.4.2.1.2 (1) is not clear. Is the intent to state that any single failure will cause at most only one operable control rod insertion failure (page 7.4-18)? The sentence requires clarification.

Resolution:

Section 2.2.2 Comment No. 1

Comment:

The design description for the switches that detect separation of the FMCRD piston and ball nut need to be identified as Class 1E (see attached mark-up). Reference SSAR section 4.6.2.2.6 page 4.6-9.

Resolution:

Section 2.2.2 Comment No. 3

Comment:

The design description should be corrected as shown in the attached mark-up.

Resolution:

Section 2.2.2 Comment No. 4

Comment:

The electrical separation between Class 1E and non-1E is not addressed in ITAAC Item 9 for the power to the FMCRDs. Suggest that appropriate portions of ITAAC item 8 that deal with electrical separation be copied into DD, ITA, and AC for Item 9.

Resolution:

2.2.2 Control Rod Drive System

Design Description

The Control Rod Drive (CRD) System controls changes in core reactivity during power operation by movement and positioning of the neutron absorbing control rods within the core in fine increments in response to control signals from the Rod Control and Information System (RCIS). The CRD System provides rapid control rod insertion (scram) in response to manual or automatic signals from the Reactor Protection System (RPS). Figure 2.2.2h shows the basic system configuration and scope.

delete to read: 2.2.2

X

The CRD System consists of three major elements: (1) the electro-hydraulic fine motion control rod drive (FMCRD) mechanisms, (2) the hydraulic control unit (HCU) assemblies, and (3) the control rod drive hydraulic system (CRDHS). The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) for abnormal operating conditions. Simultaneous with scram, the FMCRDs also provide electric-motor driven run-in of control rods as a path to rod insertion that is diverse from the hydraulic-powered scram. The hydraulic power required for scram is provided by high pressure water stored in the individual HCUs. An HCU can scram two FMCRDs. It also provides the flow path for purge water to the associated drives during normal operation. The CRDHS supplies pressurized water for charging the HCU scram accumulators and purging to the FMCRDs.

There are 205 FMCRDs mounted in housings welded into the reactor vessel bottom head. The FMCRD has a movable hollow piston tube that is coupled at its upper end, inside the reactor vessel, to the bottom of a control rod. The FMCRD can move the control rod up or down over its entire range, by a ball nut and ball screw driven at a speed of 30 mm/sec ±10% by the electric stepper motor. In response to a scram signal, the piston inserts the control rod into the core hydraulically using stored energy in the HCU scram accumulator. The scram water is introduced into the drive through a scram inlet connection on the FMCRD housing, and is then discharged directly into the reactor vessel via clearances between FMCRD parts. The average scram times of all FMCRDs with the reactor pressure as measured at the vessel bottom below 76.3 kg/cm^{2}g are:

Percent Insertion	Time (sec)
10	≤ 0.42
40	≤ 1.00

Percent Insertion	Time (sec)
60	≤ 1.44
100	≤ 2.80

These times are measured starting from loss of signal to the scram solenoid pilot valves in the HCUs.

The FMCRD has an electro-mechanical brake with a minimum holding torque of 5 kg-m on the motor drive shaft and a ball check valve at the point of connection with the scram inlet line. ~~The FMCRD switches are protected from~~

Two, Class 1E, redundant and separate switches in the FMCRD detect separation of the hollow piston from the ball nut. Independence is provided between the Class 1E divisions for these switches.

There are 103 HCUs, each of which provides water stored in a pre-charged accumulator for scramming two FMCRDs. Figure 2.2.2 shows the major HCU components. The accumulator is connected to its associated FMCRDs by a hydraulic line that includes a scram valve held closed by pressurized control air. To cause a scram, the RPS provides a signal to de-energize the scram solenoid pilot valve (SSPV) that vents the control air from the scram valve, which then opens by spring action. Loss of either electrical power to the SSPV or loss of control air pressure causes scram. A pressure switch detects low accumulator gas pressure and actuates an alarm in the main control room.

The CRD System also provides alternate rod insertion (ARI) as a means of actuating hydraulic scram when an anticipated transient without scram (ATWS) condition exists. Following receipt of an ARI signal, solenoid valves on the scram air header open to reduce pressure in the header, allowing the HCU scram valves to open. The control rod drives then insert the control rods hydraulically.

The CRDHS has pumps, valves, filters, instrumentation, and piping to supply pressurized water for charging the HCUs and purging the FMCRDs.

The CRD System components classified as safety-related are: the HCU components required for scram; the FMCRD components required for scram; the scram inlet piping; the FMCRD reactor coolant primary pressure boundary components; the FMCRD brake and ball check valve; the internal drive housing support; the FMCRD separation switches; and the HCU charging water header pressure instrumentation.

The CRD System components classified as Seismic Category I are: the HCU components required for scram; the FMCRD components required for scram; the scram inlet piping; the FMCRD reactor coolant primary pressure boundary components; the FMCRD brake and ball check valve; the internal drive housing support; the FMCRD separation switches; and the HCU charging water header pressure instrumentation.

Figure 2.2.2 shows the ASME Code class for the CRD System piping and components.

The CRD System is located in the Reactor Building. The FMCRDs are mounted to the reactor vessel bottom head inside primary containment. The HCUs and CRDHS equipment are located in the Reactor Building at the basemat elevation.

Each of the four divisional HCU charging header pressure sensors are powered from their respective divisional Class 1E power supply. Independence is provided between the Class 1E divisions for these sensors and also between the Class 1E divisions and non-Class 1E equipment.

For their preferred source of power, the FMCRDs are collectively powered from one Class 1E division; for their alternate source of power, they are collectively powered from one non-Class 1E Plant Investment Protection (PIP) bus.

The hydraulic portion of the CRD System which performs the scram function is physically separated from and independent of the Standby Liquid Control System.

The CRD System has the following alarms, displays, and controls in the main control room:

(1) Alarms for separation of the hollow piston from the ball-nut and low HCU accumulator gas pressure.

(2) Parameter displays for the instruments shown in Figure 2.2.2.

(3) Controls and status indication for the CRD pumps and flow control valves shown on Figure 2.1.2. 2

(4) Status indication for the scram valve position.

The following CRD System safety-related electrical equipment are located in either the Reactor Building or primary containment and are qualified for a harsh environment: the HCU charging header pressure instrumentation, the scram solenoid pilot valves, and FMCRD separation switches.

The check valves (CVs) shown inside the HCU boundary on Figure 2.2.2 and the FMCRD ball check valves have active safety-related functions to close under system pressure, fluid flow, and temperature conditions.

The piping and components of the CRD pump suction supply, which extends from the CRD System interfaces with the Condensate Feedwater and Air Extraction (CFCAE)

Table 2.2.2 Control Rod Drive System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
6. Two redundant and separate switches in the FMCRD detect separation of the hollow piston from the ball nut.	6. Tests of each as-built FMCRD will be conducted.	6. Both switches in each FMCRD detect separation of the hollow piston from the ball nut.
7. Following receipt of an ARI signal, solenoid valves on the scram air header open to reduce pressure in the header, allowing the HCU scram valves to open.	7. Tests will be conducted on the as-built ARI valves using a simulated actuation signal.	7. Following receipt of a simulated ARI signal, solenoid valves on the scram air header open to reduce prescure in the header, allowing the HCU scram valves to open.
8. Each of the four divisional HCU charging header pressure sensors are powered from their respective divisional Class 1E power supply. For the four HCU charging ~~water header pressure sensors,~~ independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	8. a. Tests will be conducted on the as-built ~~charging water header sensors~~ switches by providing a test signal in only one Class 1E division at a time. b. Inspections of the as-installed ~~charging water header sensor~~ switch Class 1E divisions will be conducted.	8. a. The test signal exists only in the Class 1E Division under test. b. Physical separation or electrical isolation exists between Class 1E divisions. Physical separation or electrical isolation exists between these Class 1E divisions and non-Class 1E equipment.
9. For their preferred source of power, the FMCRDs are collectively powered from one Class 1E division; for their alternate source of power, they are collectively powered from one non-Class 1E PIP bus.	9. Inspections of the as-built CRD System will be conducted.	9. For their preferred source of power, the FMCRD motors are collectively powered from one Class 1E division; for their alternate source of power, they are collectively powered from one non-Class 1E PIP bus.
10. Main control room alarms, displays and controls provided for the CRD System are defined in Section 2.2.2.	10. Inspections will be performed on the main control room alarms, displays and controls for the CRD System.	10. Alarms, displays and controls exist or can be retrieved in the main control room as defined in Section 2.2.2.

Section 2.2.3 Comment No. 1

Comment:

ITAAC verification is needed for three element control mode discussed in the design description. Proposed insertion for Table 2.2.3 is the following:

Design Commitment: At high FW flow, the FDWC system controls FW flow in automatic/manual three-element modes, using RPV water level, steam flow, and FW flow.

Inspections, Test and Analysis: Tests will be performed by simulating an increase/decrease in RPV water level or steam flow.

Acceptance Criteria: A signal to increase/decrease FW flow (corresponding to the input signal) will occur to maintain RPV water level.

Resolution:

Section 2.2.5 Comment No. 1

Comment:

Design description 2.2.5 states that the SRNM monitors neutron flux up to 15% of rated power. The SSAR 7.2.1.1.4.2 states that the SRNM monitor flux to 10% of rated power. SSAR section 7.7 has 15% as the range. Revise 7.2.1.1.4.2.

Resolution:

Section 2.2.5 Comment No. 2

Comment:

The SSAR 7.2.1.1.4.2 does not list OPRM Oscillation Power range monitor reactor trip signal. Listed in design description 2.2.5, ITAAC and TS. Modify the SSAR to add the OPRM function in SSAR section 7.2.

Resolution:

Section 2.2.7 Comment No. 1

Comment:

Reactor Protection Section design description 2.2.7 does not list a reactor trip for high main steamline radiation. ITAAC states that a simulated process variable input will be used (as listed in 2.2.7 and figure 2.2.7a) to test the RPS channel inputs.

Resolution:

Section 2.2.7 Comment No. 2

Comment:

The SSAR 7.2.1.1.4.2 does not list OPRM as an initiating condition. (This comment was previously provided to GE for inclusion in Amendment 33.)

Resolution:

Section 2.2.7 Comment No. 6

Comment:

Figures 2.2.7a and 2.2.7b use solid and dotted lines for signal flow. No description or legend is provided.

Resolution:

Section 2.2.8 Comment No. 1

Comment:

The design description (2nd para) states 2 MG sets, each of which supplies 3 of 10 ASDs power 10 RIPs. Clarify how the other 4 ASDs (and 4 associated RIPs) are powered, the description should be consistent with SSAR figure 8.3-1.

Resolution:

Section 2.2.9 Comment No. 1

Comment:

Figure 2.2.9 lists TCS, RCIC, and RFC systems but design description does not define these abbreviations.

Resolution:

Section 2.2.9 Comment No. 2

Comment:

Chapter 7 Appendix 7B "Implementation Requirements for Hardware/Software development", first paragraph, last sentence appears incomplete. See attached. (This comment was previously provided to GE for inclusion in Amendment 33.)

Resolution:

Section 2.2.9 Comment No. 3

Comment:

Section 7.7.1.5.2, Power Generation Control Subsystem, states that the PGCS issues reactor command signals to the "APR". APR is not defined.

Resolution:

Section 2.2.9 Comment No. 4

Comment:

7.7.1.5.3 Safety Evaluation, references 7.7.1.5.1 as providing the explanation of PGCS signals to the APR system. The correct reference might be 7.7.1.5.2.

Resolution:

Section 2.2.9 Comment No. 5

Comment:

In 7.7, Control Systems not Required for Safety, the system list is inconsistent with the system included in Chapter 7.7. See attached.

Resolution:

7B Implementation Requirements for Hardware/Software Development

This section defines the requirements to be met by the hardware and software development implementation activities that are to be made available for review by the NRC. Software Development

Software Management Plan ~~3.4.3.6 Pm~~

(1) The Software Management Plan shall define:

(a) the organization and responsibilities for development of the software design; the procedures to be used in the software development; the interrelationships between software design activities; and the methods for conducting software safety analyses.

Within the defined scope and content of the Software Management Plan, accepted methods and procedures for the above activities are presented in the following documents:

(i) IEEE 730, Standard for Software Quality Assurance Plans, Section 3.4;

(ii) ASME NQA2a, Part 2.7, Quality Assurance Requirements of Computer Software for Nuclear Facility Application;

(iii) ANSI/IEEE-ANS-7-4.3.2, Application Criteria for Digital Computers in Safety Systems for Nuclear Facilities (to be replaced by the issued version of P 7-4.3.2, "Standard Criteria for Digital Computers Used in Safety Systems of Nuclear Power Generation Stations");

(iv) IEC 880, Software for computers in the safety systems of nuclear power stations, Section 3.1;

(v) IEEE *(draft H)*, Standard for Software Safety Plans;

(vi) IEEE 1012, Standard for Software Verification and Validation Plans, Section 3.5;

(vii) IEEE 830, Guide to Software Requirements Specifications, Section 5;

(viii) IEEE 1042, Guide to Software Configuration Management.

Note that within the set of documents listed above, differences may exist regarding specific methods and criteria applicable to the Software Management Plan. In situations where such differences exist, all of the methods and criteria presented within those documents are considered

7.7 Control Systems Not Required for Safety

7.7.1 Description

This subsection provides discussion (or provides references to other chapter discussions) for instrumentation and controls of systems which are not essential for the safety of the plant, and permits an understanding of the way the reactor and important subsystems are controlled, and why failure of these systems does not impair safety functions. The systems include the following:

- Nuclear Boiler System—Reactor Vessel Instrumentation
- Rod Control and Information System
- Recirculation Flow Control System
- Feedwater Control System
- Process Computer System
- Neutron Monitoring System—ATIP Subsystem and systems
- Fire Protection System (Chapter 9)
- Drywell Cooling System (Chapter 9)
- Instrument Air Systems (Chapter 9)
- Makeup Water System (Chapter 9)
- Atmospheric Control System (Chapter 9)
- Fuel Pool Cooling and Cleanup System (Chapter 9)

7.7.1.1 Nuclear Boiler System—Reactor Vessel Instrumentation

Figure 5.1-3 (Nuclear Boiler System P&ID) shows the instrument numbers, arrangements of the sensors, and sensing equipment used to monitor the reactor vessel conditions. The NBS interlock block diagram (IBD) is found in Figure 7.3-2. Because the NBS sensors used for safety-related systems, engineered safeguards, and control systems are described and evaluated in other portions of this document, only the non-safety-related sensors for those systems are described in this subsection.

(1) System Identification

The purpose of the NBS instrumentation is to monitor and provide control input for operation variables during plant operation.

7.7.1.1 Nuclear Boiler System
7.7.1.2 Rod Control
7.7.1.3 Recirculation Flow Control
7.7.1.4 Feedwater Control
7.7.1.5 Process Computer
7.7.1.6 Neutron Monitoring
7.7.1.7 APR
7.7.1.8 Steam Bypass & Pressure Control
7.7.1.9 Non-Essential multiplexing System
7.7.1.9 Fuel Pool Cooling and Cleanup System
7.7.1.11 Other non-safety Related Control Systems

Section 2.2.10 Comment No. 1

Comment:

Figure 2.2.10 labelling of interfacing systems are listed as Turbine Control System, Turbine Bypass System and RFC system. The labelling is inconsistent.

Resolution:

Section 2.2.10 Comment No. 2

Comment:

Figure 15E-1 ATWS mitigation logic indicates that the SB&PC system provides the steam dome pressure input to the RFC system. Per the system descriptions, the APR system is the RFC interface for the SB&PC system. See CDM 2.2.9 and 2.2.10 attached.

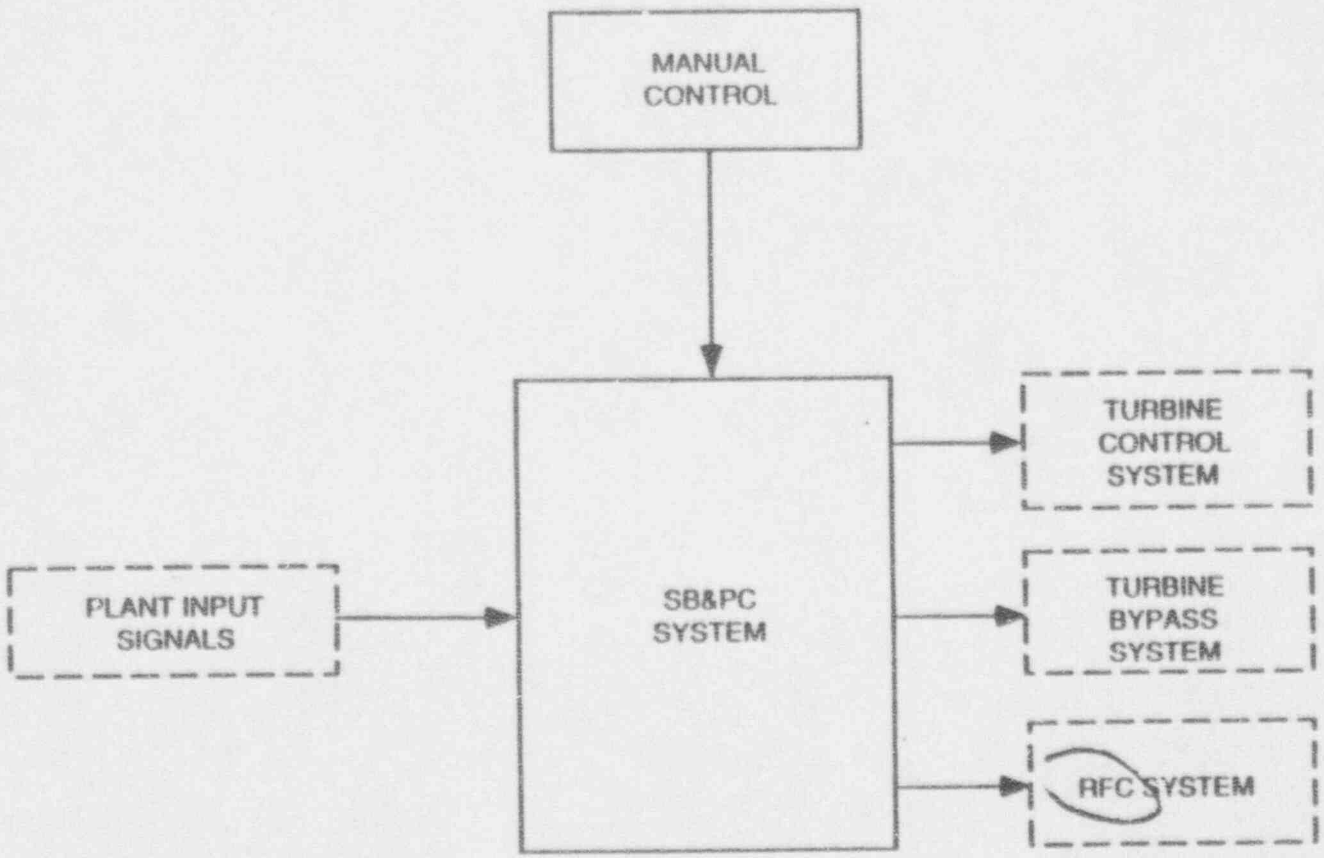
Resolution:

Section 2.2.10 Comment No. 3

Comment:

The title of figure 15E-2 is not consistent with the title of figure 15E-1. See SSAR, page 15E-18.

Resolution:



NOTE:
1. INTERCONNECTIONS MAY BE FIBER-OPTIC OR METALLIC.

Figure 2.2.10 Steam Bypass and Pressure Control System Control Interface Diagram

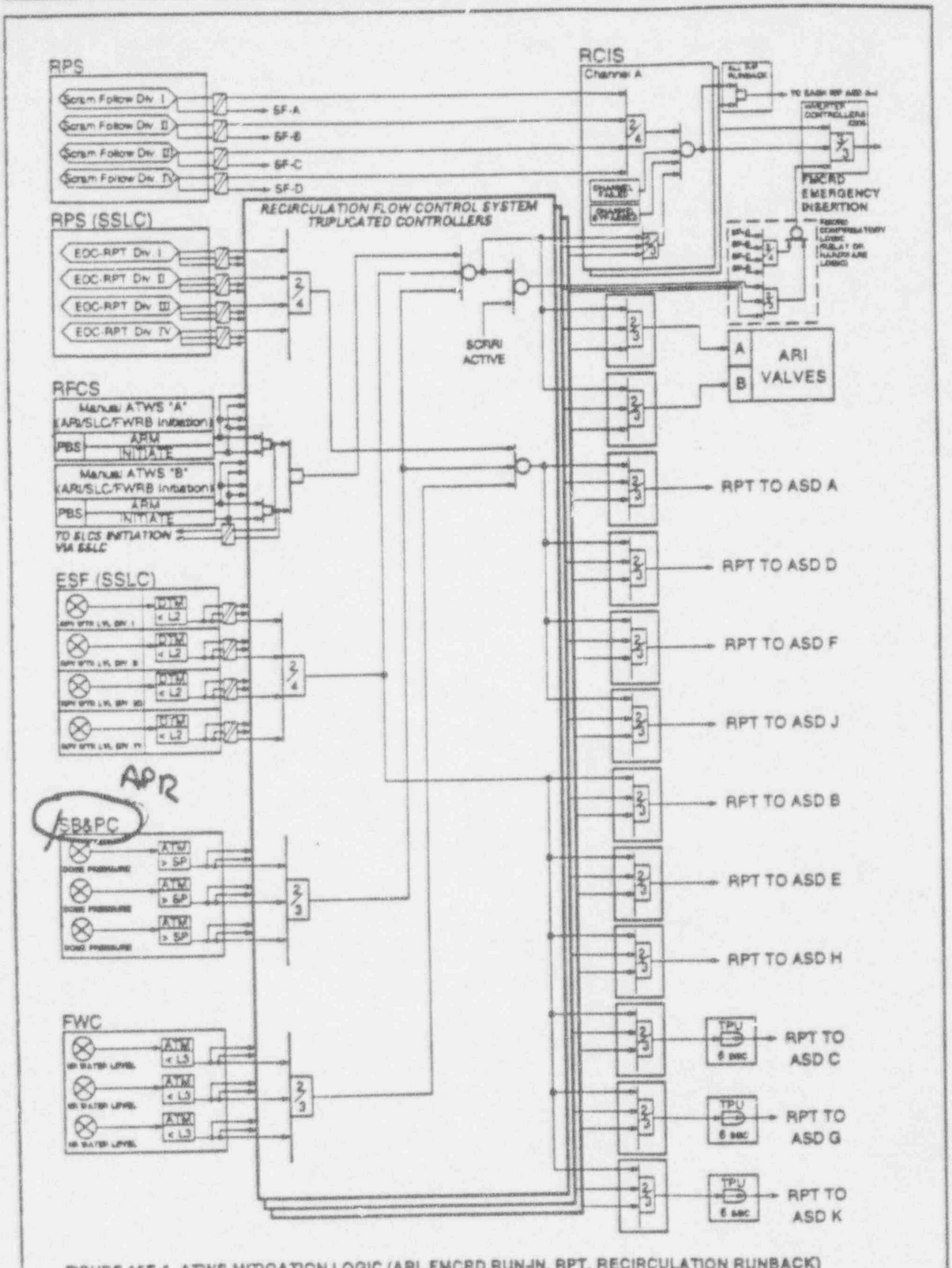


FIGURE 15E-1 ATWS MITIGATION LOGIC (ARI, FMCRD RUN-IN, RPT, RECIRCULATION RUNBACK)

Figure 15E-1 ATWS Mitigation Logic

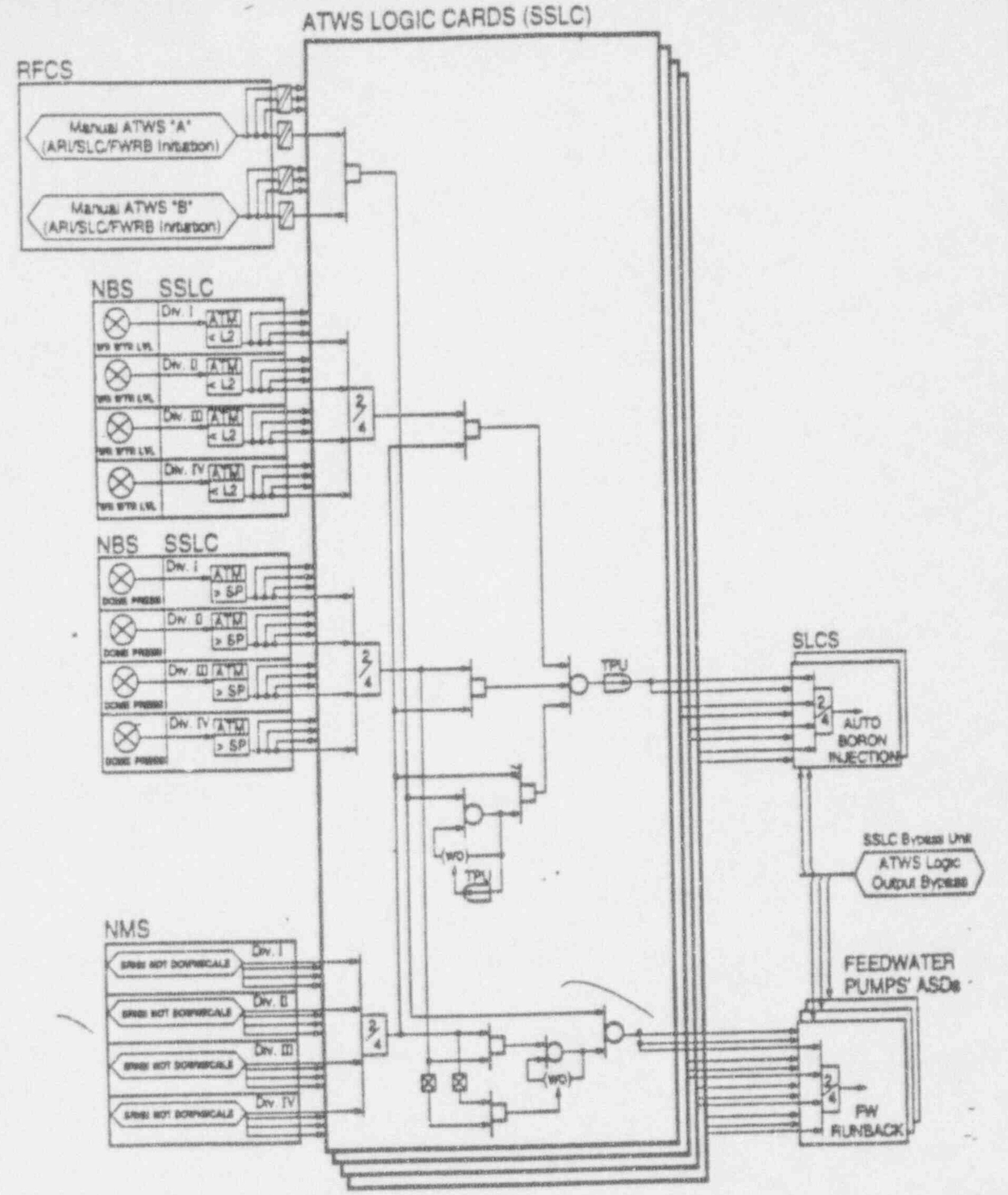


Figure FIGURE 15E-2 ATWS MITIGATION LOGIC (SLCS INITIATION, FEEDWATER RUNBACK)

15E-2 ATWS Mitigation Logic ~~(SLCS Initiation, Feedwater Runback)~~

Section 2.2.11 Comment No. 1

Comment:

CDM material 2.2.11 references the ATLM, "Automated thermal limit monitor". Acronym is not consistent with other references. See ITAAC review 2.2.1 Rod Control and Information System, Comment No. 3.

Resolution:

Section 2.3.1 Comment No. 5

Comment:

The SSAR states the RW/B exhaust vent monitor reads out in both cpm and mR/hr (eg. 11.5.5.2(6) - cpm, Table 11.5-1(B) - mR/hr, Table 11.5-2 - cpm). The SSAR should be corrected to state cpm, not mR/hr.

Resolution:

Section 2.4.1 Comment No. 1

Comment:

On page 2.4.1-2, the reactor pressure at minimum RHR rated flow is stated as 2.8 kg/cm2. In SSAR Table 6.3-1, this pressure is listed incorrectly as 28 kg/cm2.

Resolution:

Section 2.4.1 Comment No. 4

Comment:

The SSAR does not list the acronyms "SW" and "CS" used in the SSAR Table 18F-1.

Resolution:

Section 2.4.1 Comment No. 5

Comment:

Figures 2.4.1a & b should show that the RHR pump discharge pressures are displayed on the RSS panels, i.e., add symbol "R" to the pressure instruments.

Resolution:

Section 2.4.1 Comment No. 6

Comment:

The 2nd note at the bottom of SSAR Table 6.3-2 needs revision.

Resolution:

Section 2.4.1 Comment No. 7

Comment:

Why are the isolation valves between FPC and RHR in Figures 2.4.1b and 2.4.1c of different type?

Resolution:

Section 2.4.1 Comment No. 9

Comment:

Correct the attached CDM typos.

Resolution:

Outside the primary containment, each mechanical division of the RHR System (Divisions A, B, and C) is physically separated from the other divisions.

The RHR System has the following displays and controls in the main control room:

(1) Parameter displays for the instruments shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c.

(2) Controls and status indication for the active safety-related components shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c.

(3) Manual system level initiation capability for the following modes:

(a) LPFL initiation

(b) Standby

(c) Shutdown cooling

(d) Suppression pool cooling

(e) Drywell spray

RHR System components with displays and control interfaces with the Remote Shutdown System (RSS) are shown on Figures 2.4.1a and 2.4.1b.

The safety-related electrical equipment shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c located inside the primary containment and the Reactor Building is qualified for a harsh environment.

The motor-operated valves shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c have active safety-related functions and perform these functions to open, close, or both open and close, under differential pressure, fluid flow, and temperature conditions.

Valves

The check calves (CVs) shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c have safety-related functions to open, close, or both open and close under system pressure, fluid flow, and temperature conditions.

The RHR System main pumps are interlocked to prevent starting with a closed suction path.

Each RHR loop has a continuously running jockey pump to maintain the system piping continuously filled with water. The jocky pump is stopped by a RHR initiation signal or may be stopped or started manually.

The piping and components outside the shutdown cooling suction line containment isolation valves and outside the suppression pool containment isolation valves, and upstream of the suction side of the pump with all its branches have a design pressure of

Appendix A Legend for Figures

For a number of the systems presented in Section 2, figures depicting the Basic Configuration of the systems have been provided to help facilitate the Design Description. For I&C systems, the figures represent a diagram of significant aspects of the logic of the system. For other systems and buildings, these figures represent a functional diagram, representation, or illustration of design-related information. Unless otherwise specified explicitly, these figures are not necessarily indicative of the scale, location, dimensions, shape, or spatial relationships of as-built structures, systems, and components. In particular, the as-built attributes of structures, systems and components may vary from the attributes depicted on these figures, provided that those safety functions discussed in the Design Description are not adversely affected.

The figures contain information that uses the following conventions:

Mechanical Equipment

Line classification:

		Figure Designation
ASME Code Class 1	——— – – ——— – – ———	1
ASME Code Class 2	— — — — — — — — —	2
ASME Code Class 3	—————————————	3
Non-ASME Code/ Non-Nuclear Safety	——— – ——— – ——— – ———	NNS
Other Line Type:	—//——//——//——//	This legend can be used for pneumetric lines when needed for clarity. ASME Code class for such lines is defined on the system figure. ← Typo

Section 2.4.2 Comment No. 3

Comment:

Correct the attached typos.

Resolution:

Table 2.4.2 High Pressure Core Flooder System (Continued)

	Inspections, Tests, Analyses and Acceptance Criteria	
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
d. The HPCF System flow in each division is not less than a value corresponding to a straight line between a flow of 182 m^3/hr at a differential pressure of 82.8 kg/cm^3 and a flow of 727 m^3/hr at a differential pressure of 7 kg/cm^2.	d. Tests will be conducted on each division of the as-built HPCF System in the HPCF high pressure flooder mode. Analyses will be performed to convert the test results to the conditions of the Design Commitment.	d. The converted HPCF flow satisfies the following: The HPCF System flow in each division is not less than a value corresponding to a straight line between a flow of 182 m^3/hr at a differential pressure of 82.8 kg/cm^3 and a flow of 727 m^3/hr at a differential pressure of 7 kg/cm^2.
e. The HPCF System has the capability to deliver at least 50% of the flow rates in item 3d with 171°C water at the pimp suction. Typo	e. Analyses will be performed of the as-built HPCF System to assess the system flow capability with 171°C water at the pump suction.	e. The HPCF System has the capability to deliver at least 50% of the flow rates in item 3d with 171°C water at the pump suction.
f. System flow into the reactor vessel is achieved within 16 seconds of receipt of an initiation signal and power available at the emergency busses.	f. Tests will be conducted on each HPCF division using simulated initiation signals.	f. The HPCF System flow is achieved within 16 seconds of receipt of a simulated initiation signal.
g. The HPCF pumps have sufficient NPSH available at the pumps.	g. Inspections, tests and analyses will be performed upon the as-built system. NPSH tests of the pumps will be performed in a test facility. The analyses will consider the effects of: – Pressure losses for pump inlet piping and components. – Suction from the suppression pool with water level at the minimum value. – 50% minimum blockage of the pump suction strainers.	g. The available NPSH exceeds the NPSH required by the pumps.

Section 2.4.3 Comment No. 1

Comment:

The Design Description should include discussion of the main steam line radiation level trip of the MSIVs which is an input from PRMS. This is discussed in SSAR 7.3.1.1.2 as part of LDS.

Resolution:

Section 2.4.4 Comment No. 4

Comment:

See comments on attached copy of Figure ?.4.4a.

Resolution:

Section 2.4.4 Comment No. 5

Comment:

In Table 2.4.4, Item 1, 3rd column: change the end of the statement to read " --- Figures 2.4.2a and 2.4.2b."

Resolution:

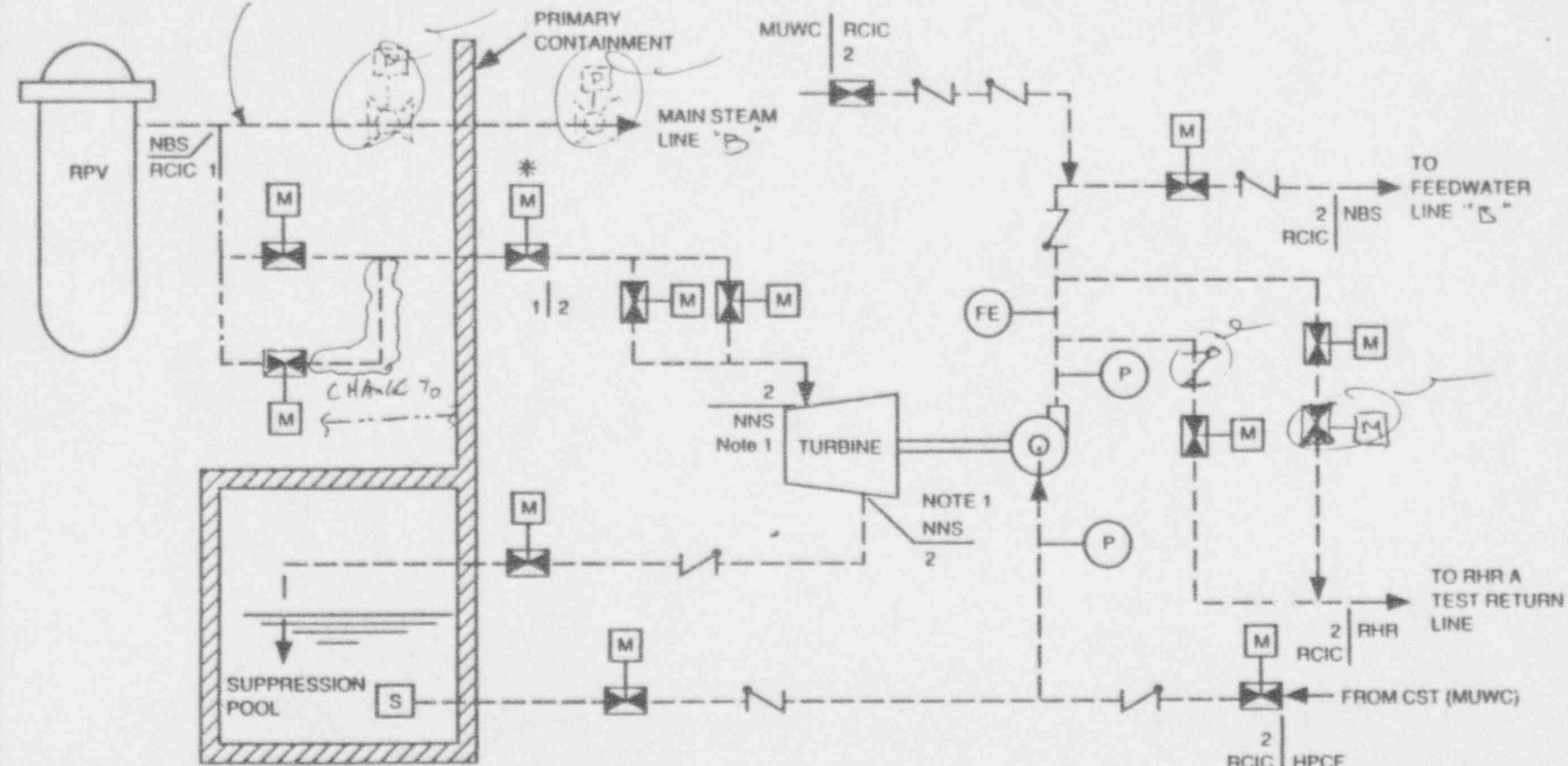
Section 2.4.4 Comment No. 6

Comment:

Table 2.4.4, Item 3i, add the following condition to the 1st and 3rd column "within 29 seconds after the signal to start." In the 3rd column, change the numeral 2 to an exponent.

Resolution:

LINE NBS FROM RPV TO OUTBOARD MSIV
IS ASME CODE CLASS 1



NOTES:

1. RCIC TURBINE IS NOT COVERED BY ASME CODE SECTION III BUT IS DESIGNED, FABRICATED, AND INSTALLED TO SAFETY-RELATED STANDARDS AND IS CONSISTENT WITH THE ASME CODE SECTION III REQUIREMENTS. THE TURBINE INCLUDES THE TURBINE TRIP AND THROTTLE VALVE.

2. ALL RCIC SYSTEM COMPONENTS SHOWN ON THIS FIGURE ARE POWERED FROM CLASS 1E DIVISION I EXCEPT FOR THE OUTBOARD CONTAINMENT ISOLATION VALVE (*) WHICH IS CLASS 1E DIVISION II.

3. ALL RCIC SYSTEM COMPONENTS SHOWN ON THIS FIGURE EXCEPT THE INBOARD CONTAINMENT ISOLATION VALVES, ARE POWERED FROM DC SOURCES.

COMMENTS:

1. CHANGE PIPING CODE CLASS IN 2 PLACES.

~~2. ADD PHANTOM MSIVS.~~

3. STATE SPECIFICALLY LINE "B" FOR MAIN STEAM + FEEDWATER

Figure 2.4.4a Reactor Core Isolation Cooling System

~~4. SHOW CHECK VALVE AND SECOND MOV IN ACCORDANCE SSAR TABLE 3.9.8.~~

ITAAC COMMENTS PAGE 3 OF 3

GEORGE + GARY
[illegible]

Section 2.6.1 Comment No. 2

Comment:

Need to develop acronyms for: main condenser, check valve, and containment isolation valve. (Example of problem: "CV" is used for "check valve", but the App. B shows "CV" as "control valve)

Resolution:

Section 2.6.1 Comment No. 3

Comment:

In the Certified Design Material, SSAR and P&ID, use one consistent parameter for the centerline of the tee connection; select either "at least 460 mm above the centerline of the variable leg nozzle" or "at least 389 mm above the top of active fuel".

Resolution:

Section 2.6.1 Comment No. 4

Comment:

Reference attached markup of Figures 2.6.1, 5.1-3, and 5.4-12 for comments.

Resolution:

Section 2.6.1 Comment No. 5

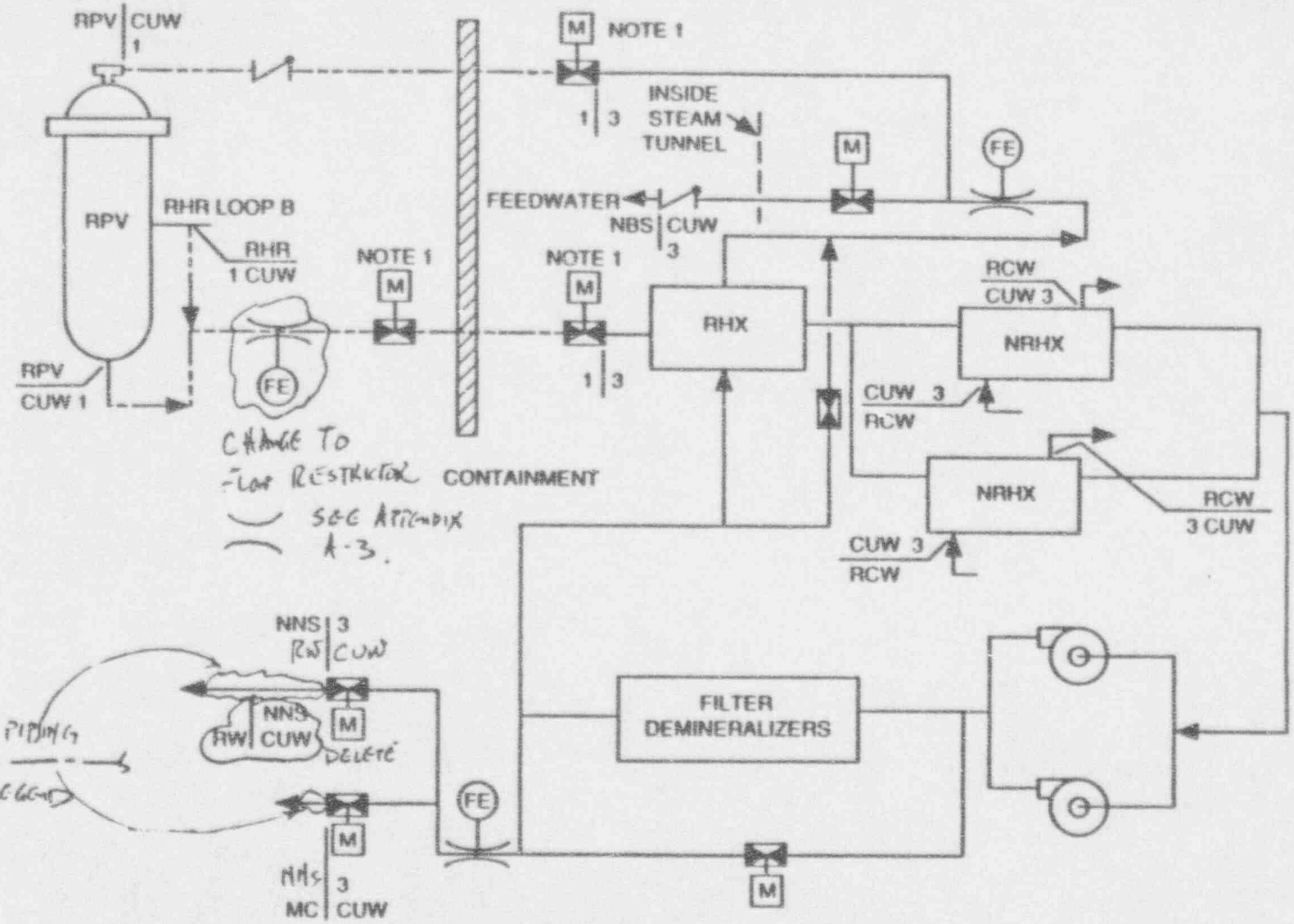
Comment:

Certified Design Material Table 2.6.1 in ITAAC:

Item 3. - change "non-1E" to "non- Class 1E"

Item 5a provided the closure time of <= 30 sec for two of the three containment isolation MOVs, whereas SSAR Table 5.4-6 requires all three MOVs to close within a time constraint of <30 sec. Should be revised to reflect a closure time of <30 seconds.

Resolution:



NOTES:
1. THE INBOARD CONTAINMENT ISOLATIONVALVE IS POWERED FROM CLASS 1E DIVISION II AND THE OUTBOARD ISOLATION VALVES ARE POWERED FROM CLASS 1E DIVISION I.

Figure 2.6.1 Reactor Water Cleanup System

ADD ACRONYM MC
TO APPENDIX B.

ITAAC COMMENTS P. 2a OF 2

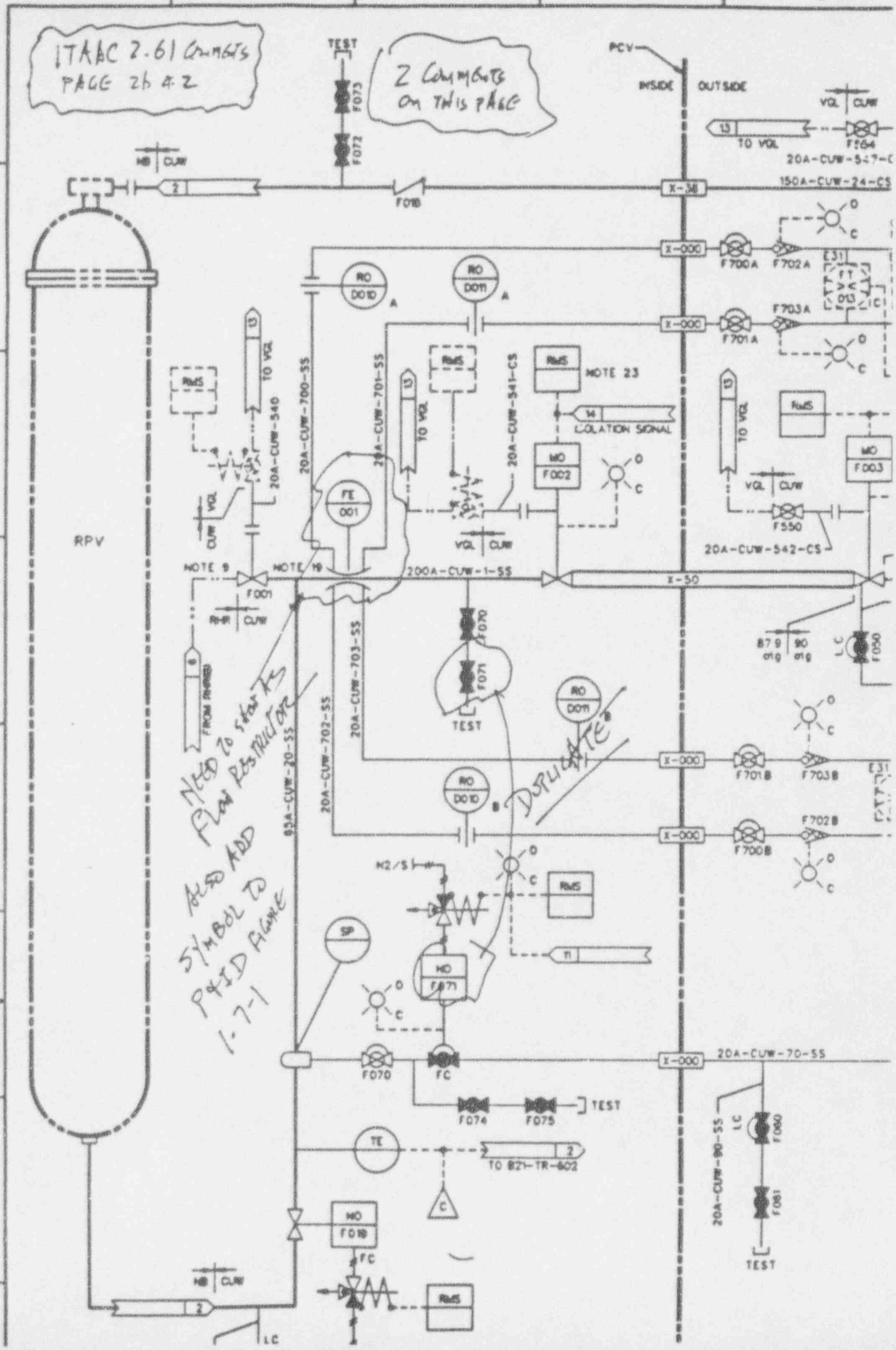
ITAAC 2.61 Comments
PAGE 26 42

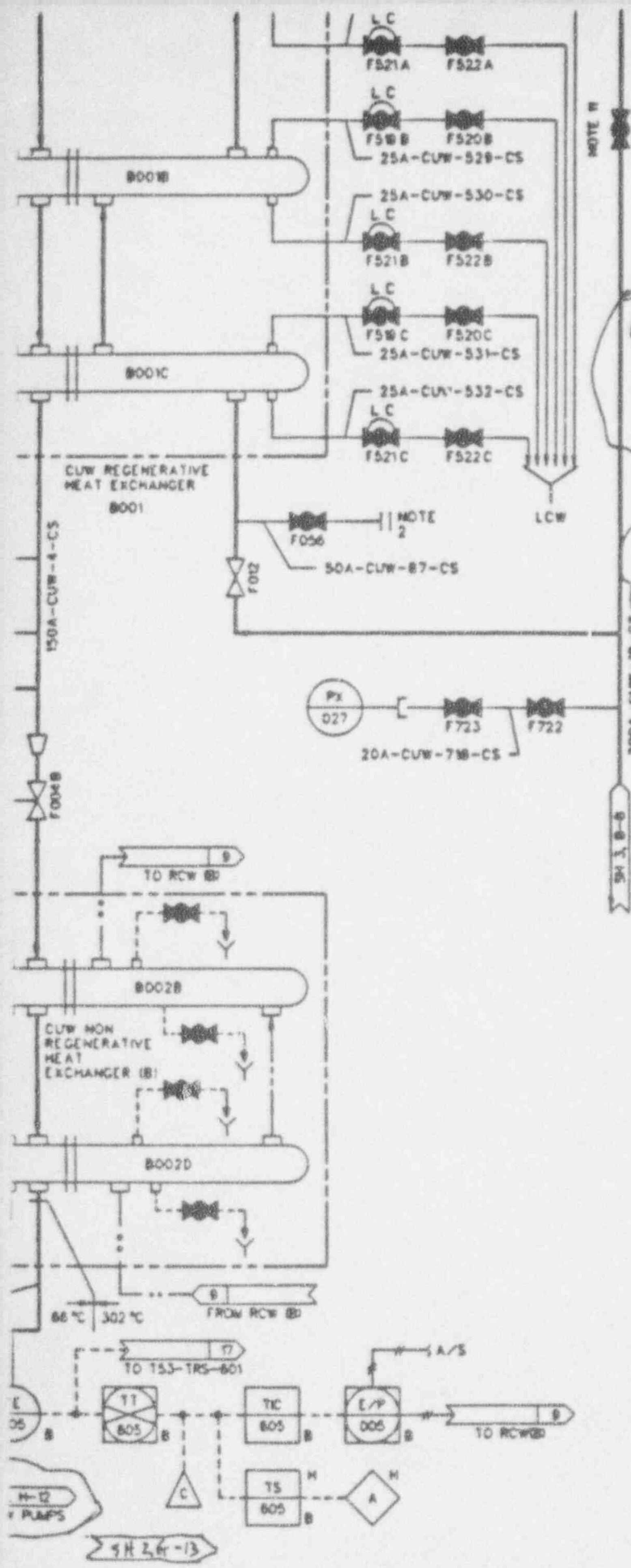
2 Comments on this page

Need to show as Flow Restrictor

Also add symbol to P&ID figure 1.7-1

DUPLICATE





VALVE POSITION INDICATION LIGHTS LOCATED IN VICINITY OF VALVE CONTROL SWITCH.

13. TE006 INCLUDES TT FUNCTION.

14. NO VALVES AND INSTRUMENTS TO BE LOCATED IN THE SHIELDED COMPARTMENT CONTAINING THE FILTER DEMINERALIZER.

15. DESIGN CONDITIONS ARE FOLLOWING

(A) FLUID	WATER
(B) RADIOACTIVE CONCENTRATION	≥ 1 µCi/CC
(C) SCHEDULE	INTERFACE

16. FILTER DEMINERALIZER VALVE CONTROL SWITCHES, VALVE POSITION INDICATION LIGHTS AND ALARMS SHOULD BE INSTALLED IN LOCAL CONTROL PANEL.

17. A COMMON TROUBLE ALARM FROM LOCAL ANNUNCIATOR SHALL ALARM IN THE MAIN CONTROL ROOM.

18. WO INDICATES AIR OPERATED VALVES FAIL AS IS ON LOSS OF AIR AND FAIL CLOSED ON LOSS OF ELECTRIC POWER.

Consistency 389 mm or 460 mm

19. VESSEL HEAD DRAINLINE TEE CONNECTION TO THE CUW SUCTION LINE SHALL BE INSTALLED AT AN ELEVATION OF AT LEAST 389 mm ABOVE THE TOP OF ACTIVE FUEL.

20. MAXIMUM THROAT DIAMETER OF FLOW RESTRICTOR FE-F001 SHALL BE 70% OF THE INTERNAL PIPE DIAMETER.

CHANGE TO 135 mm

21. PIPE WITH A DESIGN PRESSURE OF 28.8 KG/SQ CM OR GREATER SHALL HAVE ITS MINIMUM WALL THICKNESS NO LESS THAN THAT OF A STANDARD WEIGHT PIPE. THICKER THAN THE STANDARD WEIGHT PIPE SHALL BE USED IF REQUIRED BY THE DESIGN PRESSURE OR OTHER REQUIREMENTS.

22. VALVES WITH A DESIGN PRESSURE OF 28.8 KG/SQ CM OR GREATER SHALL HAVE A MINIMUM OF CLASS 300, OR OF A HIGHER CLASS IF REQUIRED BY THE DESIGN PRESSURE.

23. THE INBOARD CONTAINMENT ISOLATION VALVE F002 MANUAL CONTROL AND VALVE POSITION STATUS INDICATION SHALL BE HARDWIRED (NOT MULTIPLEXED) TO THE MAIN CONTROL ROOM.

REFERENCE DOCUMENTS	MPL NO.
1. REACTOR WATER CLEANUP SYS PFD	G31-1020
2. NUCLEAR BOILER SYSTEM P&ID	B21-1010
3. RADIOACTIVE WASTE (LIQUID, SOLID), RADWASTE SYS P&ID	K17-1010
4. LCW, RADWASTE SYSTEM P&ID	K17-1010
5. REACTOR WATER CLEANUP SYS IBD	G31-1030
6. RESIDUAL HEAT REMOVAL SYS P&ID	E11-1010
7. SERVICE AIR SYS P&ID (REAC BLDG)	P51-1010
8. PIPING & INSTRUMENT SYMBOLS DIAGRAM	A10-3030
9. REAC BLDG CLNG WATER SYS P&ID	P21-1010
10. MUWC SYS P&ID (REAC BLDG)	P13-1010
11. SAMPLING SYSTEM P&ID	P91-1010
12. CONTROL ROD DRIVE SYS P&ID	C12-1010
13. VALVE GLAND LEAKAGE TREATMENT, RADWASTE SYS P&ID	K17-1010
14. LEAK DETECTION AND ISOLATION SYS IED	E31-1010
15. FUEL POOL CLNG & CLEANUP SYS P&ID	G41-1010
16. FEEDWATER CONTROL SYS IBD	C31-1030
17. SUPPRESSION POOL TEMPERATURE MONITORING SYSTEM P&ID	T53-1010
18. MAIN CONDENSER SYSTEM P&ID	N61-1010

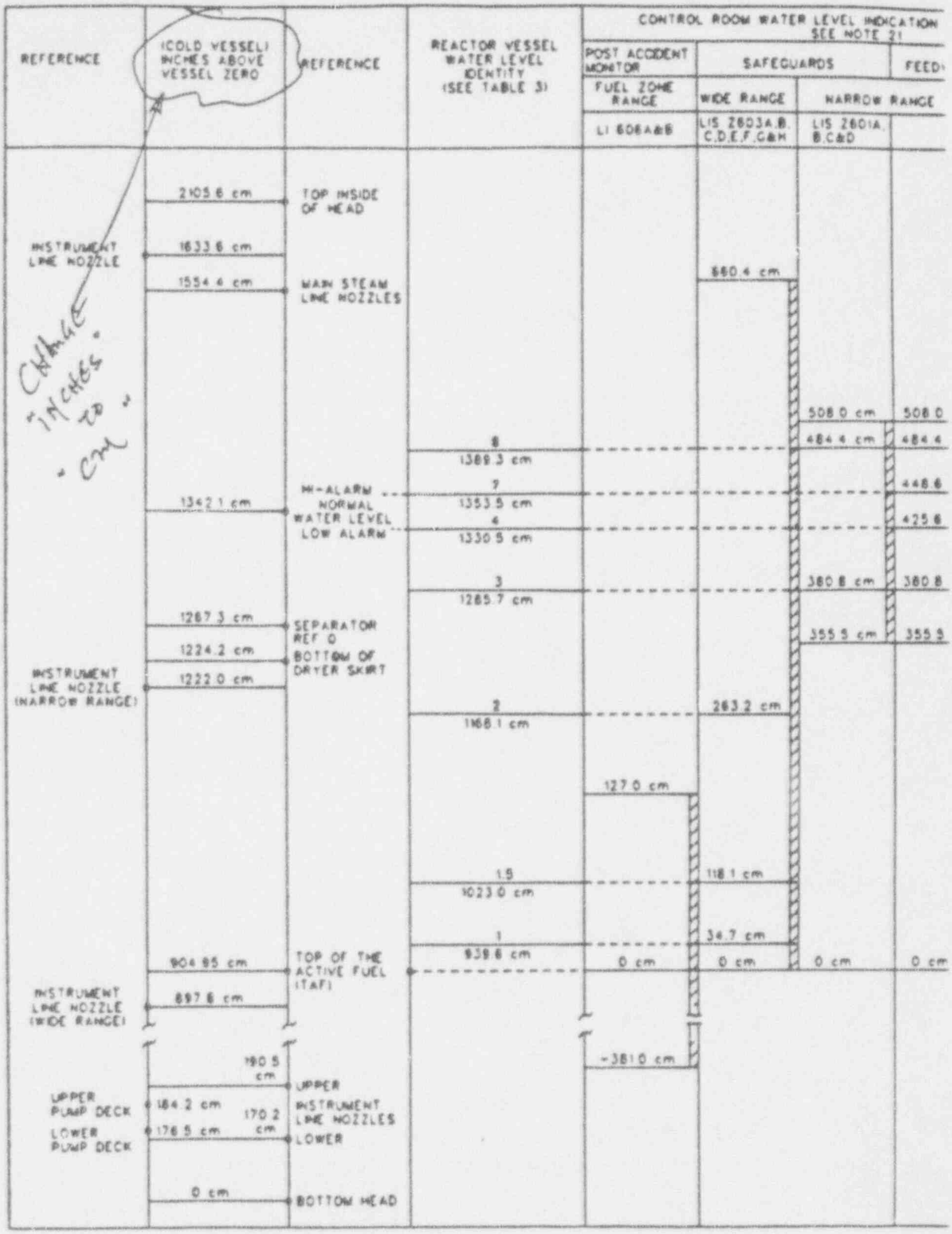
MPL NO. G31-1010

3 Comments on THIS PAGE

ITAAC 2.6.1 Comments PAGE 2C OF 2

FIGURE 5.4-12 REACTOR WATER CLEANUP SYSTEM P&ID (Sheet 1 of 4)

TABLE 2 ELEVATION CORRELATION CHART



ITAAC 2.6.1 COMMENTS
PAGE 2d of 2

PrFD FIGURE 5.1-3
SHEET 9 OF 11

Section 2.6.2 Comment No. 6

Comment:

Add the following statement:

Piping penetrations and arrangements of piping connected to the pool are configured to ensure that the pool is not drained below a minimum level of water above the stored spent fuel in the event of a pipe break, or inadvertent operator action, or siphonic action.

Resolution:

Section 2.6.3 Comment No. 1

Comment:

Page 2.6.3-1 first paragraph: add acronym (CST) after condensate storage tank.

Resolution:

Section 2.6.3 Comment No. 3

Comment:

Figure 2.6.3: Change "SURGE TANK" to "SURGE TANKS".

Resolution:

Section 2.6.3 Comment No. 5

Comment:

SSAR Section 9.5.9, page 9.5-57, add acronym (D/S) after dryer/separator.

Resolution:

Section 2.8.1 Comment No. N/A

Comment:

In the Design Description, delete the first sentence starting: "The fuel assembly --- 100", since this refers to 10 CFR 20, 50 and 100.

Resolution:

Section 2.9.1 Comment No. 1

Comment:

The valves listed in SSAR Table 6.2-7, Containment Isolation Valve Information Radwaste System, are shown on Figures 11.2-2, sheets 29 and 31 of 36. GE deleted these figures in Amendment 33. However, the staff understands the figures will be readded. GE should correct SSAR Table 6.2-7, page 6.2-165 to refer to these figures. F-103 and F-104 are HCW H2O, not LCW H2O. Also, page 6.2-122 entry page should be 6.2-165.

Resolution:

Section 2.9.1 Comment No. 4

Comment:

Features of the radioactive drain transfer system should be discussed in 2.9.1. See attached.

Resolution:

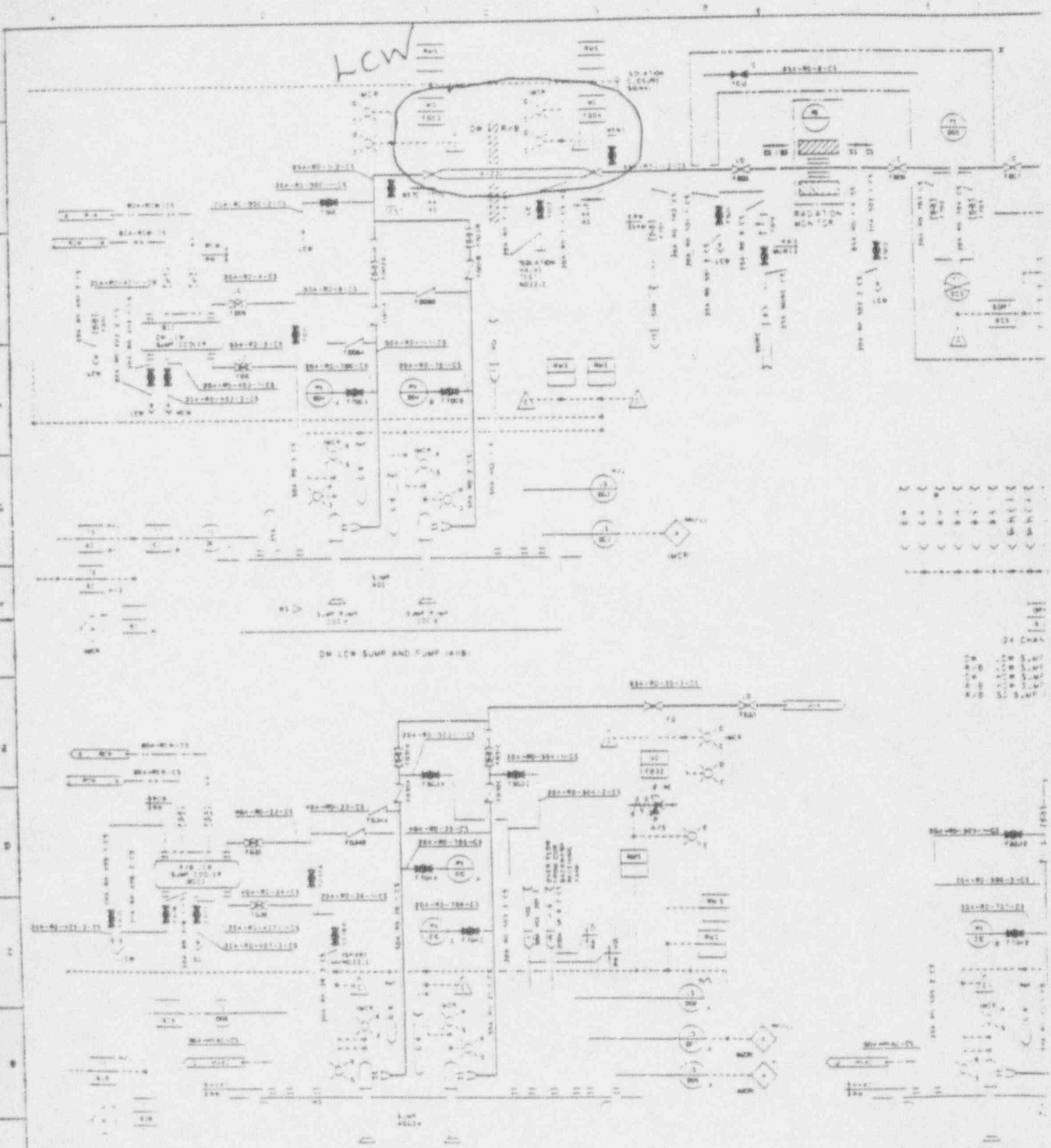


Figure 11.2-2 RADW

proprietary

HCW

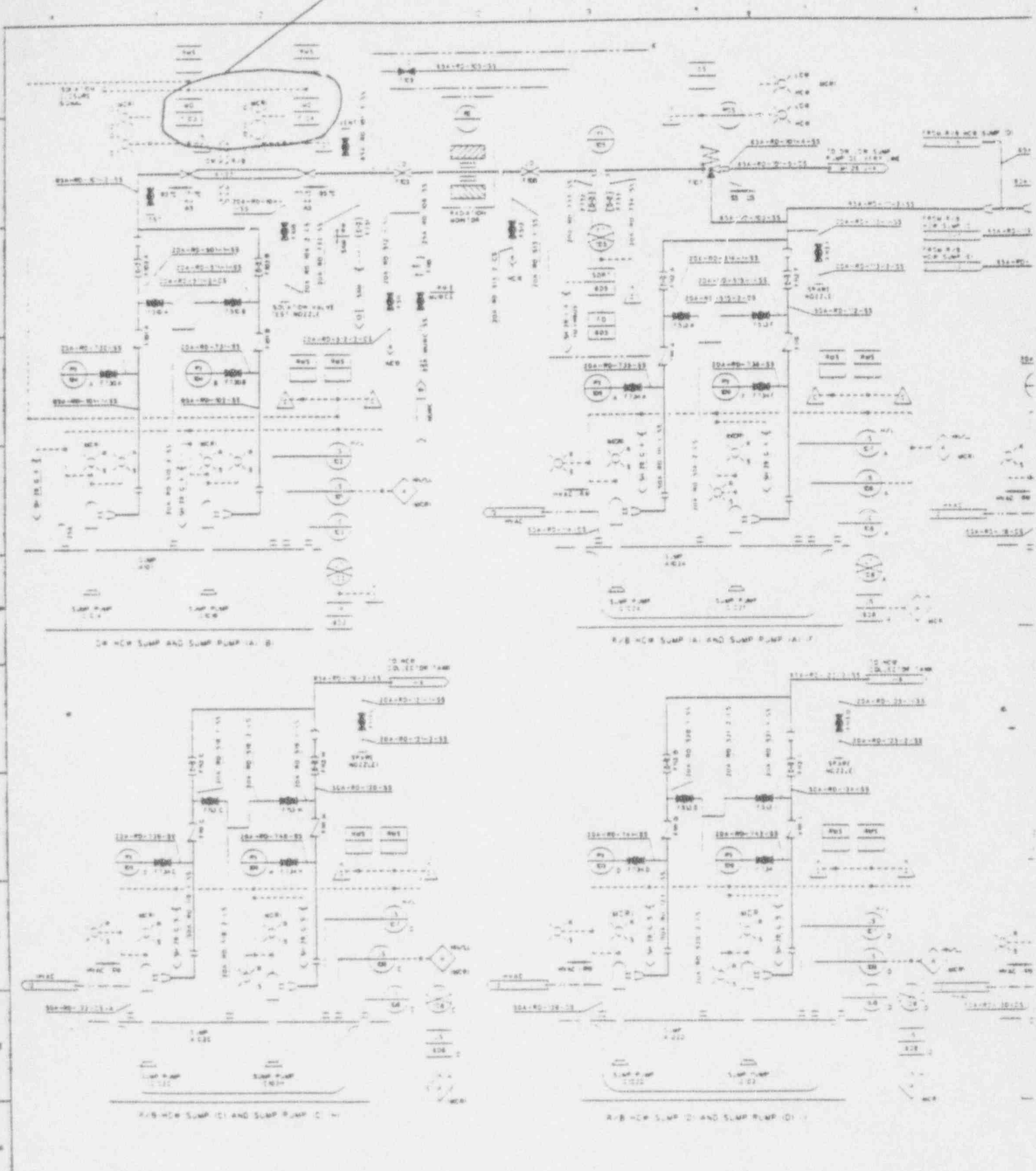


Figure 11.2-2 RAD\

Proprietary

Section 2.10.1 Comment No. 2

Comment:

Reference attached SSAR tables for comments:

a) Table 3.2-1 pages 3.2-19 & 3.2-55.
b) Table 10.3-1

Resolution:

P

Table 3.2-1 Classification Summary (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
1	6. Piping including supports—MSL (including branch lines to first valve) from the seismic interface restraint up to but not including the turbine stop valve and turbine bypass valve	N	SC,T	B	F Should be a 'B' or 'E' No 'F' in the note e.	—	(r)
	7. Piping from FW shutoff valve to seismic interface restraint	N	SC	D	E	I	(ee)
	8. Deleted						
	9. Deleted						
	10. Pipe whip restraint—MSL/FW	3	SC,C	—	B	—	
	11. Piping including supports—other within outermost isolation valves						
	a. RPV head vent	1	C	A	B	I	(g)
	b. Main steam drains	1	C,SC	A	B	I	(g)
	12. Piping including supports—other beyond outermost isolation or shutoff valves						
	a. RPV head vent beyond shutoff valves	N	C	C	E	—	
1	b. Main steam drains to first valve	2/N	SC,T	B	B	I/—	(r)
	c. Main steam drains beyond first valve	N	SC, T	D	E	—	(r)

Notes and footnotes are listed on pages 3.2-53 through 3.2-60

W	=	Radwaste Building
X	=	Control Building
F	=	Firewater Pump House*
U	=	Ultimate Heat Sink Pump House*
P	=	Power Cycle Heat Sink Pump House*

d. A,B,C,D= Quality groups defined in Regulatory Guide 1.26 and Subsection 3.2.2. The structures, systems and components are designed and constructed in accordance with the requirements identified in Tables 3.2-2 and 3.2-3.

— = Quality Group Classification not applicable to this equipment.

e. B = The quality assurance requirements of 10CFR50, Appendix B are applied in accordance with the quality assurance program described in Chapter 17.

E = Elements of 10CFR50, Appendix B are generally applied, commensurate with the importance of the equipment's function.

f. I = The design requirements of Seismic Category I structures and equipment are applied as described in Section 3.7, Seismic Design.

— = The seismic design requirements for the safe shutdown earthquake (SSE) are not applicable to the equipment. However, the equipment that is not safety-related but which could damage Seismic Category I equipment if its structural integrity failed is checked analytically and designed to assure its integrity under seismic loading resulting from the SSE.

g. 1. Lines one inch and smaller which are part of the reactor coolant pressure boundary and are ASME Code Section III, Class 2 and Seismic Category I.

2. All instrument lines which are connected to the reactor coolant pressure boundary and are utilized to actuate and monitor safety systems shall be Safety Class 2 from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation.

3. All instrument lines which are connected to the reactor coolant pressure boundary and are not utilized to actuate and monitor safety systems shall

* Pump House structures are out of the ABWR Standard Plant scope.

Table 10.3-1 Main Steam Supply System Design Data

Main Steam Piping	
Design flow rate at 69.25 kg/cm^2a and 0.40% moisture, lb/hr	~17,000,000
Number of lines	4
Nominal diameter	700A
Minimum wall thickness, mm	38.1
Design pressure, kg/cm^2a	87.89
Design temperature, °C	315.56
Design code	ASME III, Class 2
Seismic design	Analyzed for SSE design loads

Comments:

1. DESIGN PRESSURE IS gauge "g" on PAGE 10.3-2.

2. DESIGN TEMPERATURE IS 315.55 °C on PAGE 10.3-

Section 2.10.2 Comment No. 3

Comment:

Page 2.10.2-4, Figure 2.10.2b:

Valve operators shown are pneumatic, whereas on SSAR figure 10.4-1, they are motor operators. Reconcile the type of valve operators used for the valves on piping from SJAEs to inlet of vacuum pump as shown in ITAAC figure 2.10.2b.

Resolution:

Section 2.10.2 Comment No. 4

Comment:

Page 2.10.2-6, table 2.10.2b, item 2: The acceptance criteria requires the SJAE discharge valves to close. These valves are not shown on figure 2.10.2b. Show SJAEs discharge valves on ITAAC figure 2.10.2b as shown in SSAR figure 10.4-1. Also, explain why one of the SJAE discharge goes to "offgas-A" but not diverted to "turbine compartment exhaust system" in certain condition in SSAR figure 10.4-1 (see ITAAC figure 2.10.2b).

Resolution:

Section 2.10.7 Comment No. 1

Comment:

Add acronyms for "high pressure" and "low pressure" as "HP" and "LP" respectively in CDM 2.10.7 Design Description. Also, add acronyms "HP", "LP", "ISVs" and "IVs" in CDM Appendix B.

CDM Table 2.10.7 should show acronym "MTSVs", not "MSVs" in items 2.b and 2.c as listed in CDM Appendix B.

Delete the word "other" in 2nd sentence of CDM 2.10.7 Design Description.

See attached markup.

Resolution:

2.10.7 Main Turbine

Design Description

The Main Turbine (MT) uses the energy in steam from the reactor to drive the plant generator. DELETE

The other major turbine components are: DELETE ACRONYMS FOR APPENDIX B

(1) A high pressure (HP) section

(2) An intermediate section (between HP and LP sections)

(3) Low pressure (LP) sections

The major fluid system boundaries are:

(1) Turbine Main Steam 2.10.1

(2) Main Condenser 2.10.21

(3) Turbine Gland Seal 2.10.9

(4) Extraction System 2.10.12

The MT is classified as non-safety-related.

The MT has the following features that prevent overspeed:

(1) Main turbine stop valves (MTSV)/Control valves (CV) [MTSVs trip, CVs trip and modulate] DELETE ACRONYMS FOR APPENDIX B

(2) Combined intermediate valves (CIVs) consist of intercept valves (IVs) and intercept stop valves (ISVs) [IVs trip and modulate, ISVs trip]

(3) Extraction line non-return valves (trip)

(4) Redundant valve closure mechanisms (i.e., fast acting solenoid valves and emergency trip fluid system)

(5) Redundant normal speed control

Three levels of signals to MT valves (i.e., normal speed control, overspeed trip, backup overspeed trip)

Table 2.10.7 Main Turbine System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the MT System is as described in Section 2.10.7.	1. Inspection of the as built MT will be conducted.	1. The as built MT conforms with the basic configuration described in Section 2.10.7.
2. MT System overspeed protective actions are as defined in Section 2.10.7.	2. Tests will be conducted on the as-built MT System using simulated overspeed signals.	2. The following protective actions occur: **Overspeed Condition** — **Protective Action** a. Exceeds normal speed control setpoint. — Normal speed control signals the CVs and IVs to close. b. Exceeds overspeed trip setpoint. — Overspeed trip signals MSVs, CVs, ISVs, IVs, and extraction line non-return valves to close. c. Exceeds backup overspeed trip setpoint. — Backup overspeed trip signals MSVs, CVs, ISVs, IVs, and extraction line non return valves to close. MTSVs
3. The turbine MTSV closes in 0.10 seconds or greater.	3. Tests will be conducted on the as-built turbine MTSV.	3. The turbine MTSV closes in 0.10 seconds or greater.
4. The turbine CV trip closure is 0.08 seconds or greater.	4. Tests will be conducted on the as-built turbine CV.	4. The turbine CV trip closure is 0.08 seconds or greater.

Section 2.10.9 Comment No. 1

Comment:

Correct typos on page 2.10.9-1 as follows:

- "two exhaust blowers" need not be listed as "two full capacity exhaust blowers" in 2nd paragraph of CDM 2.10.9.
- "main turbine system" should be "main turbine", and "turbine main steam supply systems" should be "turbine main steam supply system" in 3rd paragraph.

Resolution:

Section 2.10.9 Comment No. 2d

Comment:

On Figure 2.10.9, steam leaving the main turbine stop and control valves and the bypass valves are shown being directed to the feedwater heater. On SSAR Figure 10.4-2, they are directed to the cross around piping. Resolve this discrepancy. Delete note 2 is "FW HEATER" is deleted.

Resolution:

Section 2.10.21 Comment No. 1

Comment:

Revise CDM Appendix B to add acronym "MC" for "Main Condenser".

Resolution:

Section 2.10.21 Comment No. 2

Comment:

Revise CDM 2.10.21 Design Description, 1st paragraph, to state "TB", not "TBP".

Resolution:

Section 2.10.21 Comment No. 3

Comment:

Revise SSAR section 10.4.1.2.1, 2nd paragraph, to state "Figure 10.4-5b", not "Figure 10.4-6b".

Resolution:

Section 2.10.22 Comment No. 2

Comment:

SSAR Chapter 11.3.6 should be supplemented with information pertaining to the automatic isolation of the OGS, as verified in ITAAC #3.

Resolution:

Section 2.10.22 Comment No. 3

Comment:

SSAR Chapter 11.3 should be supplemented with information pertaining to the OGS capability to withstand a hydrogen explosion, as verified in ITAAC #6.

Resolution:

Section 2.10.23 Comment No. 1

Comment:

Revise ITAAC figure 2.10.23 to conform with CDM "Appendix A" for "NNS" piping class.

Resolution:

Section 2.11.1 Comment No. 1

Comment:

Section 2.11.1 requires a figure/diagram to supplement the text. This is necessary because of the primary containment isolation function which is safety-related.

Resolution:

Section 2.11.1 Comment No. 2

Comment:

Section 2.11.1, 4th paragraph, states that the outboard containment isolation valve is locked closed during normal operation. SSAR Section 9.2.10.2 item 7 states " --- locked closed during standby, not standby and power operation." This requires clarification.

Resolution:

Section 2.11.1 Comment No. 4

Comment:

SSAR Table 9.2-3: combine the two sections on pages 9.2-50 and 9.2-51 into a single unit.

Resolution:

Section 2.11.2 Comment No. 1

Comment:

Add the following statement to the CDM: "The Condensate Storage Tank (CST) capacity includes sufficient water for operation of the RCIC System during station blackout." Also, it should be verified in an ITAAC.

Resolution:

Section 2.11.2 Comment No. 2

Comment:

Revise figure 2.11.2 to incorporate the following:

- 3 pumps should be shown, or indicate that this is 1 of 3 pumps.
- A note should be added to figure 2.11.2 to indicate that RCIC, HPCF, and SPCU take suction from the CST.
- The extraneous piping shown on figure 2.11.2 should be deleted.

Resolution:

Section 2.11.2 Comment No. 3

Comment:

Revise Table 2.11.2 to add the boilerplate ITAAC on hydrostatic testing.

Resolution:

Section 2.11.2 Comment No. 4

Comment:

SSAR Section 9.2.9.2, Item (9):

- Add to the first sentence, "radwaste building control room, and Remote Shutdown System."
- Add to the second sentence, "and low water level shall be alarmed in the main control room."

Resolution:

Section 2.11.3 Comment No. 1

Comment:

Figure 2.11.3b - relocate piping class break immediately after the MOV before the Fuel Pool Cooling HX.

Resolution:

Section 2.11.3 Comment No. 2d

Comment:

F175, 3 valves, one for each system

In accordance with the P&IDs and the CDM figures, these are MOVs supplying cooling water to the fuel pool cooling HX room coolers. Table 3.9-8 of the SSAR describes these valves as "cooling water supply to RHR system HX pressure relief valve". Resolve this discrepancy.

Also, both P&IDs and CDM figures showed a total of 2 valves, one for RCW-A and the second for RCW-B. Resolve this discrepancy.

Resolution:

Section 2.11.4 Comment No. 1

Comment:

Figure 2.11.4 shows the surge tank is shared with the HVAC Normal Cooling Water System (HNCW). Whereas on SSAR figure 9.2-6a, the surge tank is shared with the HNCW and the Hot Water Heating (HWH) Systems. Resolve this discrepancy.

Also, the HWH discussion in section 9.2 was deleted and should be reinserted.

Resolution:

Section 2.11.6 Comment No. 1

Comment:

On figures 2.11.6a and 2.11.6b, DP should be dP.

Resolution:

Section 2.11.6 Comment No. 2

Comment:

ITAAC 9, the Design Commitment figures 2.11.3a and 2.11.3b should be 2.11.6a and 2.11.6b.

Resolution:

Section 2.11.9 Comment No. 1

Comment:

Page 2.11.9-1, second paragraph, and page 2.11.9-2, item (3): change " --- loss-of-coolant accident (LOCA) signal, ---." to " --- loss-of-coolant accident and/or loss of preferred power (LOCA and/or LOPP) signal, ---."

Resolution:

Section 2.11.9 Comment No. 2

Comment:

The CDM describes valves F003 and F005 to have active safety-related functions. In Table 3.9-8 of SSAR, they are classified as "PASSIVE". Resolve this discrepancy.

Resolution:

Section 2.11.9 Comment No. 3

Comment:

Table 2.11.9, change "LOCA" to "LOCA and/or LOPP" for all three parts across the page.

Resolution:

Section 2.11.9 Comment No. 4

Comment:

SSAR Section 9.2.15.1.2, change paragraph to read as follows: " --- shutdown; (d) testing; and (e) loss of preferred power."

Resolution:

Section 2.11.11 Comment No. 1

Comment:

The SSAR does not reference the Fig. 9.3.7, sheets 1&2, Service Air System, in SSAR chapter 9.3.7.

Resolution:

Section 2.11.13 **Comment No. 1**

Comment:

ITAAC item #6 requires each of two HPIN divisions to be powered from the respective Class 1E divisions. This requirement should be included in the HPIN description in SSAR Chapter 6.7.2 or appropriate section in SSAR chapter 3.

Resolution:

Section 2.11.21 Comment No. 1

Comment:

See attached markup for SSAR editorial comment.

Resolution:

(7) Any purified water storage tank shall be provided located outdoors with adequate freeze protection and adequate diking and other means to control spill and leakage.

9.2.8.3 System Description (Conceptual Design)

The MWP System consists of both mobile and permanently installed water treatment systems.

The permanently installed system consists of a well, filters, reverse osmosis modules and demineralizers which prepare demineralized water from well water. The demineralized water is sent to storage tanks until it is needed. Pumps are provided to keep the Makeup Water Preparation (MWP) System pressurized at all times. The components of the MWP System are listed in Table 9.2-15 and the system block flow diagram is in Figure 9.2-10.

While it is planned to install both permanent divisions, only one division may be installed if plant water requirements and economic conditions indicate that the second division will not be needed.

Mobile water treatment systems will be used before the permanent system is installed and later if water requirements exceed the capacity of the permanent system or if economic condition make use of mobile equipment attractive compared to operating and maintaining the permanent system.

9.2.8.3.1 Well System

A well, well water storage tank and two well water forwarding pumps are provided which can produce sufficient water to meet the concurrent needs of the MWP System and the PSW System.

9.2.8.3.2 Pretreatment System

Two dual media filters are provided in parallel which are backwashed when needed using one of two backwash pumps and water from a filtered water storage tank. This tank is provided with a heater to maintain a water temperature of at least 10°C at all times. Water may be sent from the filtered water storage tank to the PSW System or to the next components of the MWP System.

9.2.8.3.3 Reverse Osmosis Modules

Chemical addition tanks, pumps and controls are provided to add sodium hexametaphosphate and sodium hydroxide to the filtered water.

Four high pressure, horizontal multistage reverse osmosis (RO) feed pumps provide a feed pressure of approximately 32 kg/cm^2g. Reverse osmosis membranes are arranged in two parallel divisions of two passes each with the permeate of the first passes going to

Section 2.12.1 Comment No. 1

Comment:

Figure 2.12.1 shows "DG II" feeders for all divisions. It should be changed to "DG I, DG II, and DG III" as shown in attached markup. This may conflict with SSAR. See SSAR section 8.3.1.1.8.3 and Item 4 of SSAR section 8.3.3.6.2.3.2.

Resolution:

Section 2.12.1 Comment No. 2

Comment:

Incorrect page numbers are referenced on SSAR pages 8.0.iii/iv and v/vi.

Resolution:

Section 2.12.1 Comment No. 3

Comment:

SSAR descriptions use "mVA and mW" as abbreviations for showing the units of power for equipment such as transformers, DG, CTG and buses. This is inconsistent with drawing 8.3.1. Drawing used "MVA and MW" as abbreviations. SSAR descriptions need to be updated to be consistent with drawing.

Resolution:

Section 2.12.1 Comment No. 4

Comment:

Values referenced in acceptance criteria for ITAAC #s 3, 5, 6, 7, 8.b, and 21 are not described in the design description. These should be described. With respect to separation distances in #s 3, 5, 6, and 7, GE should consider removing the actual distances from the acceptance criteria column. Distances are in the SSAR.

Resolution:

Section 2.12.1 Comment No. 8

Comment:

SSAR TS Section should be clarified as noted in the attached markup.

Resolution:

Section 2.12.1 Comment No. 9

Comment:

ITAAC #23 and CDM design description should be revised as shown in the attached markup.

Resolution:

Section 2.12.1 Comment No. 10

Comment:

In response to comment no. 17 made during the pilot review, GE stated the proposed changes would be made post-amendment 33. This comment is being made to encourage follow-up on that specific item which is the legend in the SSAR for electrical symbols.

Resolution:

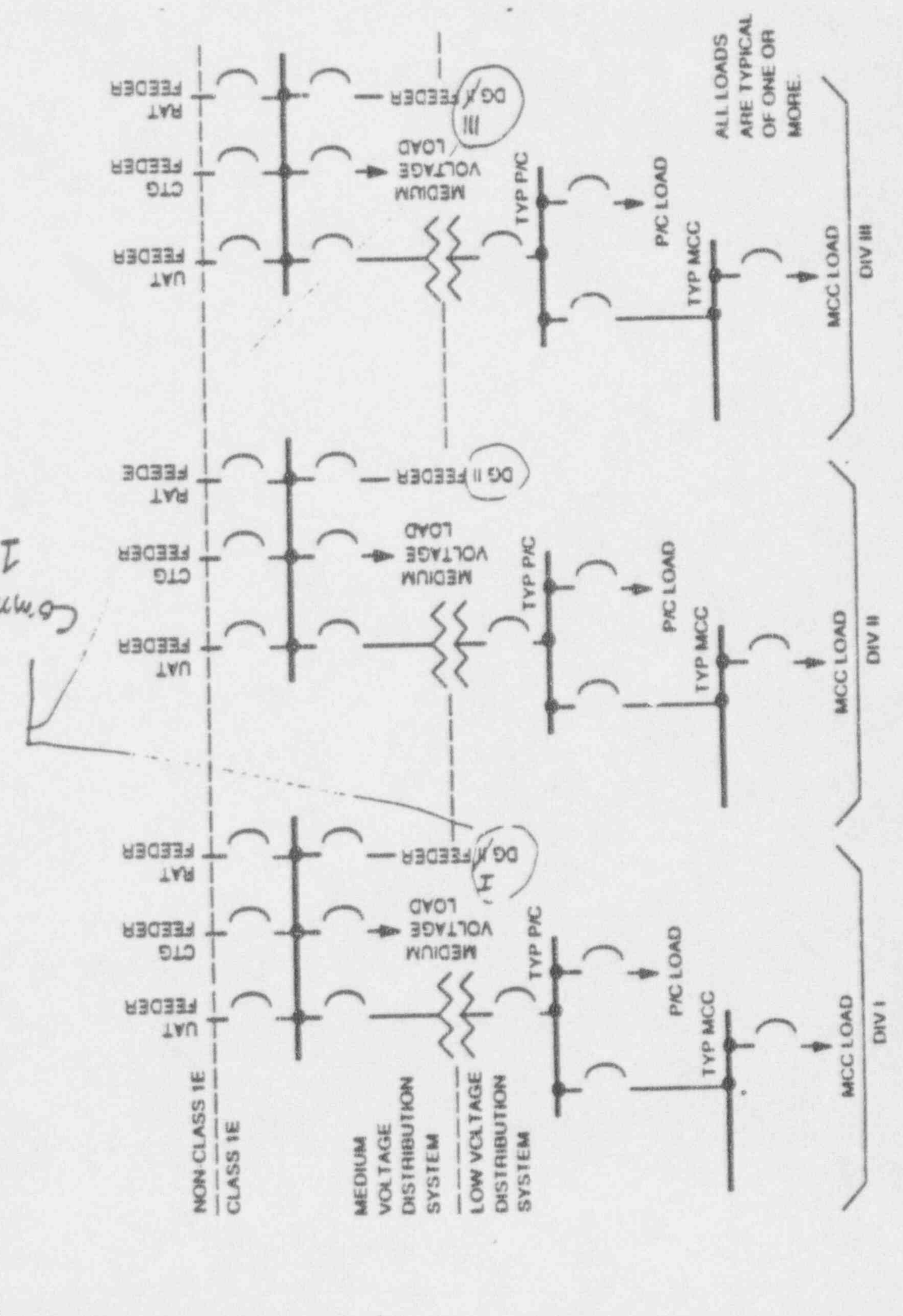


Figure 2.12.1 Class 1E Electrical Power Distribution System

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Comment 2

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Comme 2

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources—Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (normal preferred and alternate preferred) and the onsite standby power sources (Division I diesel generator (DG), Division II DG, and Division III DG). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The Class 1E AC distribution system supplies electrical power to three divisional load groups, with each division powered by an independent Class 1E 6.9 kV ESF bus (refer to LCO 3.8.9, "Distribution Systems—Operating"). Each ESF bus has two separate and independent preferred (offsite) sources of power and a dedicated onsite DG. Each ESF bus is also connectable to a combustion turbine generator (CTG). The ESF systems of any two of the three divisions provide for the minimum safety functions necessary to shut down the unit and maintain it in a safe shutdown condition.

Comment #8

CTG may be substituted for the second (delay access) offsite source

Offsite power is supplied to each of the 6.9 kV ESF buses from the transmission network via two electrically and physically separated circuits. In addition, ~~offsite power may be supplied~~ to any one ESF bus ~~from the CTG~~ (for a limited duration) when the ESF bus is being fed from the reserve auxiliary transformer while the unit auxiliary transformer associated with the ESF bus is out of service, or when the ESF bus is being fed from the unit auxiliary transformer while the reserve auxiliary transformer associated with the ESF bus is out of service. These offsite AC electrical power circuits are designed and located so as to minimize to the extent practicable the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A detailed description of the offsite power system and circuits to the onsite Class 1E ESF buses is found in SSAR, Chapter 8 (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, controls, and

(continued)

17 Class 1E medium voltage M/C switchgear and low voltage P/C switchgear and MCCs are identified according to their Class 1E division. Class 1E M/C and P/C switchgear and MCCs are located in Seismic Category I structures, and in their respective divisional areas.

19 Class 1E EPD System cables and raceways are identified according to their Class 1E division. Class 1E divisional cables are routed in Seismic Category I structure and in their respective divisional raceways.

Harmonic Distortion waveforms do not prevent Class 1E equipment from performing their safety functions.

The EPD System supplies an operating voltage at the terminals of the Class 1E utilization equipment that is within the utilization equipment's voltage tolerance limits.

An electrical grounding system is provided for (1) instrumentation, control, and computer systems, (2) electrical equipment (switchgear, distribution panels, and motors) and (3) mechanical equipment (fuel and chemical tanks). Lightning protection systems are provided for buildings and for structures and transformers located outside of the buildings. Each grounding system and lightning protection system is separately grounded to the plant ground grid.

The EPD System has the following alarms, displays and controls in the MCR:

(1) Alarms for degraded voltage on Class 1E medium voltage M/C switchgear.

(2) Parameter displays for PMG output voltage, amperes, watts, vars, and frequency.

(3) Parameter displays for EPD System medium voltage M/C switchgear bus voltages and feeder and load amperes.

(4) Controls for the PMG output circuit breaker, medium voltage M/C switchgear feeder circuit breakers, load circuit breakers from the medium voltage M/C switchgear to their respective low voltage P/C switchgear, and low voltage feeder circuit breakers to the low voltage P/C switchgear.

(5) Status indication for the PMG output circuit breaker and the medium voltage M/C switchgear circuit breakers.

The EDP System has the following displays and controls at the Remote Shutdown System (RSS):

(1) Parameter displays for the bus voltages on the Class 1E Divisions I and II medium voltage M/C switchgear.

Table 2.12.1 Electric Power Distribution System (Continued)

Design Commitment	Inspections, Tests, Analyses and Acceptance Criteria: Inspections, Tests, Analyses	Acceptance Criteria
22. The EPD System supplies an operating voltage at the terminals of the Class 1E utilization equipment that is within the utilization equipment's voltage tolerance limits. Comment # 19	22. Analyses for the as-built EPD System to determine voltage drops will be performed.	22. Analyses for the as-built EPD System exist and conclude that the analyzed operating voltage supplied at the terminals of the Class 1E utilization equipment is within the utilization equipment's voltage tolerance limits, as determined by their nameplate ratings.
23. An electrical grounding system is provided for (1) instrumentation, control, and computer systems, (2) electrical equipment (switchgear, distribution panels, and motors) and (3) mechanical equipment (fuel and chemical tanks). Lightning protection systems are provided for buildings and for structures and transformers located outside of the buildings. Each grounding system and lightning protection system is separately grounded to the plant ground grid.	23. Inspections of the as-built EPD System plant Grounding and Lightning Protection Systems will be conducted.	23. The as-built EDP System instrumentation, control, and computer grounding system, electrical equipment and mechanical equipment grounding system, and lightning protection systems provided for buildings and for structures and transformers located outside of the buildings are separately grounded to the plant ground grid.
24. MCR alarms, displays and controls provided for the EPD System are as defined in Section 2.12.1.	24. Inspections will be conducted on the MCR alarms, displays and controls for the EPD System.	24. Displays and controls exist or can be retrieved in the MCR as defined in Section 2.12.1.
25. RSS displays and controls provided for the EPD System are as defined in Section 2.12.1.	25. Inspections will be conducted on the as-built RSS displays and controls for the EPD System.	25. Displays and controls exist or can be retrieved on the RSS as defined in Section 2.12.1.

SSAR Section 8.2 Comment No. 1

Comment:

In responding to comment no. 24 on ITAAC 2.12.1, GE incorrectly interjected the letter "B" between "isolated" and "phase" in the next to the last paragraph on page 8.2-2.

Resolution:

SSAR Section 8.2 Comment No. 2

Comment:

In responding to comment no. 20 on ITAAC 2.12.1, GE incorrectly changes SSAR Fig. 8.2.1 (sh 2). "Gas Combustion Turbine Generator" should just be "Combustion Turbine Generator".

Resolution:

Section 2.12.10 Comment No. 3

Comment:

IEEE 317 "IEEE Standard for Electrical Penetration Assembly in Containment Structure for Nuclear Power Plant" should be referenced in SSAR for meeting design, construction, qualification, test and installation of electrical penetration assemblies.

Resolution:

Section 2.12.11 Comment No. 6

Comment:

Design Description states that "CTG is located in a non-safety related area of the plant." SSAR Section 9.5.11.3 states that "Adequate protection of the CTG against sabotage is provided by locating the unit inside the security protected area." Design Description should be revised to show that the CTG is located in the protected area of the plant and an ITAAC should be provided to verify this location.

Resolution:

Section 2.12.11 Comment No. 8

Comment:

Table 2.12.11, item 1, replace 2.12.1 with 2.12.11.

Resolution:

Section 2.12.12 Comment No. 1

Comment:

GE response to comment #9 of pilot review is not fully implemented. Figure 8.3.4 sheet 3 as well as other affected drawings (AC and DC systems) should be revised to show non-drawout type MCCBs.

Resolution:

Section 2.12.12 Comment No. 2

Comment:

One line diagram symbol legend needs to be added to SSAR section 1.7 as committed by GE in response to pilot review comments 10 & 11.

Resolution:

Section 2.12.13 Comment No. 1b

Comment:

Acceptance values (+/- 10%) voltage and +/- 2% frequency) should be deleted from ITAACs 4, 5, and 7 since the design requirement is to establish rated voltage and frequency. The above tolerance requirements are specified only during loading.

Resolution:

Section 2.12.13 Comment No. 2

Comment:

SSAR Section 8.3.1.1.8.2, item no. 14 states that "the maximum loads expected to occur for each division do not exceed 90% of the continuous power output rating of the diesel generator." This information should be incorporated in CDM and ITAAC #2 should be revised accordingly. See attached.

Resolution:

Section 2.12.13 Comment No. 6

Comment:

Load shedding requirements specified in SSAR 8.3.1.1.7 should be included in CDM. Design commitments 4 & 6 and ITAACs should be revised to take into consideration the shedding of large motors at a bus voltage equal to 30% of nominal.

Resolution:

Section 2.12.13 Comment No. 10

Comment:

Last paragraph of TS bases section B 3.8.1 should be revised to include power factor of DG.

Resolution:

Section 2.12.13 Comment No. 13

Comment:

Revise design commitment #7 as shown in the attached markup.

Resolution:

Section 2.12.13 Comment No. 14

Comment:

Page 2.12.13-1, 4th paragraph, states that the EDG is automatically connected to its respective divisional bus upon an undervoltage condition., However, this is not really true because in accordance with page 8.3-16, item 1 of the SSAR, it is necessary for large motors to trip first at a bus voltage equal to 30% before the EDG output breaker is closed. See comment 6 above on the same aspect.

Resolution:

Table 2.12.13 Emergency Diesel Generator System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
6. When LOCA and LOPP signals exist, the DG automatically connects to its respective divisional bus. After a DG connects to its respective bus, the LOCA loads are automatically sequenced onto the bus.	6. Tests on the as-built DG Systems will be conducted by providing simulated LOCA and LOPP signals.	6. In the as-built DG Systems, when LOCA and LOPP signals exist, the DG automatically connects to its respective divisional bus. The automatic load sequence begins at ≤ 20 seconds. Following application of each load, the bus voltage does not drop more than 25% measured at the bus. Frequency is restored to within 2% of nominal, and voltage is restored to within 10% of nominal within 60% of each load sequence time interval. The HPCF and RHR loads are sequenced on to the bus in ≤ 36 seconds for design basis events.
comment 13 7. A manual start signal from the MCR or from the local control station in the DG area starts a DG. After starting, the DG remains in a standby mode, unless a LOPP signal exists. (ie. running at ~~rated~~ required voltage and frequency)	7. Tests on the as-built DG Systems will be conducted by providing a manual start signal from the MCR and from the local control station, without a LOPP signal.	7. As-built DGs automatically start on receiving a manual start signal from the MCR or from the local control station, attain rated voltage (±10%), and rated frequency (±2%) in ≤ 20 seconds and remain in the standby mode.
8. When a DG is operating in parallel (test mode) with offsite power, a loss of the offsite power source used for testing or a LOCA signal overrides the test mode by disconnecting th[illegible] from its respective divisional bus.	8. Tests on the as-built DG Systems will be conducted by providing simulated loss of offsite power and LOCA signals while operat[illegible]g the DGs in the test mode.	8. When the as-built DG Systems are operating in the test mode with offsite power and a loss of offsite power or a LOCA signal is received, DGs automatically disconnect from their respective divisional buses.

Table 2.12.13 Emergency Diesel Generator System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the DG System is described in Section 2.12.13.	1. Inspection of the as-built system will be conducted.	1. The as-built DG System conforms with the basic configuration described in Section 2.12.13.
2. The DGs are sized to supply their load demand following a LOCA.	2. Analyses to determine DG load demand, based on the as-built DG load profile, will be performed.	2. Analyses for the as-built DG systems exist and conclude that the DG System capacities exceed, as determined by their nameplate ratings, their load demand following a LOCA. by 10%
3. DG air start receiver tanks have capacity for five DG starts without recharging their tanks.	3. Tests on the as-built DG Systems will be conducted by starting the DGs five times.	3. As-built DGs start five times without recharging their air start receiver tanks.
4. A LOPP signal (bus under-voltage) from an EPD System medium voltage divisional bus automatically starts its respective DG, and initiates automatic connection of the DG to its divisional bus. A DG automatically connects to its respective bus when DG rated voltage and frequency conditions are established. After a DG connects to its respective bus, the non-accident loads are automatically sequenced onto the bus.	4. Tests on the as-built DG Systems will be conducted by providing a simulated LOPP signal.	4. As-built DGs automatically start on receiving a LOPP signal, attain rated voltage (±10%), and rated frequency (±2%) in ≤ 20 seconds, automatically connect to their respective divisional bus, and sequence their non-accident loads onto the bus.
5. LOCA signals from the RHR (Division I) and HPCF (Divisions II and III) System automatically start their respective divisional DG. After starting, the DGs remain in a standby mode (i.e. running at rated voltage and frequency, but not connected to their busses), unless a LOPP signal exists.	5. Tests on the as-built DG Systems will be conducted by providing a simulated LOCA signal, without a LOPP signal.	5. As-built DGs automatically start on receiving a LOCA signal, attain rated voltage (±10%), and rated frequency (±2%) in ≤ 20 seconds, and remain in the standby mode.

check

Section 2.12.14 Comment No. 1

Comment:

Why are drawout type molded case circuit breakers shown on SSAR figure 8.3.3?

Resolution:

Section 2.12.15 Comment No. 1

Comment:

Figure 2.12.15 and SSAR figure 8.3-2 do not agree in regard to the type of breakers utilized. Revise the SSAR figure to show non-drawout breakers.

Resolution:

Section 2.12.15 Comment No. 2

Comment:

Page 2.12.15-1, last paragraph-Selectively between interrupting devices is required; however this is difficult or almost impossible to achieve between molded-case circuit breakers, since their instantaneous trips are not adjustable. Figure 2.12.15 needs to be revised to employ other types of interrupting devices or this requirement needs to be eliminated or a disclaimer added.

Resolution:

Section 2.12.15 Comment No. 3

Comment:

Comment No. 2 also applies to Design Commitment No. 9 and the related tests and acceptance criteria. Unless the appropriate types of interrupting devices (typically fuses) are selected, this design requirement can not be met.

Resolution:

Section 2.12.16 Comment No. 6

Comment:

Paging Facilities SSAR 9.5.2.2.1, figure 9.5-2 lists paging equipment as T/B, R/B, Hx/B, S/B, Switching Station and outdoors. Acronyms are not listed for T/B, Hx/B, S/B and locations are not consistent with the ITAAC or SSAR descriptions. Locations are not shown on SSAR Figure 9.5.2 for the sound powered phone system and system is labelled as the communication facilities board for maintenance.

Resolution:

Section 2.12.17 Comment No. 6

Comment:

SSAR and CDM use different terminologies. SSAR refers to "Class 1E Associated lighting" whereas CDM refers to "Associated Class 1E lighting". Clarification should be provided - specifically, use of associated is acceptable when discussing circuits, however, GE should be consistent.

Resolution:

Section 2.14.4 Comment No. 1

Comment:

First paragraph: The sentence "SGTS consists of two redundant divisions." has been repeated; delete one.

Resolution:

Section 2.14.4 Comment No. 2

Comment:

SSAR Section 6.5: see attached pages for comments.

Resolution:

Section 2.14.4 Comment No. 3

Comment:

Figure 6.5-1 (Sh 2 of 3) and (3 of 3), coordinates 4/F: change the title of "EXHAUST" to "PROCESS".

Resolution:

6.5.1.2 System Design

6.5.1.2.1 General

The SGTS P&ID is provided as Figure 6.5-1.

6.5.1.2.2 Component Description

Table 6.5-1 provides a summary of the major SGTS components. The SGTS consists of two parallel and redundant filter trains. The two SGTS trains are located in two adjacent rooms. Each train is protected for fire, flood, pipe break and missiles. The electrical separation is provided by connecting the two trains to Divisions 2 and 3 electric power. The two trains are mechanically separated also. Suction is taken from the secondary containment, including above the refueling area, or from the primary containment via the Atmospheric Control System (ACS). The treated discharge goes to the main plant stack.

The SGTS consists of the following principal components:

(1) Two filter trains, each consisting of a of a moisture separator, an electric process heater, a prefilter, a high efficiency particulate air (HEPA) filter, a charcoal adsorber, a second HEPA filter, space heaters, and a cooling fan for the removal of decay heat from the charcoal.

(2) ~~Two independent process fans~~ located downstream of each filter train.

"A PROCESS FAN IS"

6.5.1.2.3 SGTS Operation

6.5.1.2.3.1 Automatic

Upon receipt of a high drywell pressure signal or a low reactor water level signal, or when high radioactivity is detected in the secondary containment or refueling floor ventilation exhaust, both SGTS trains are automatically actuated and one train is manually placed in the Standby mode. When the operation of both the trains is assured, one train is placed in the Standby mode. In the event that a malfunction disables an operating train, the standby train is automatically initiated.

6.5.1.2.3.2 Manual

T/PO

The SGTS is on standby during normal plant operation.It may be manually initiated for primary containment de-inerting in accordance with the Technical Specifications when required to limit the discharge of contaminants to the environment within 10CFR20 limits. Normal operation of the SGTS while the plant is in the startup, power, hot standby, and hot shutdown modes of operation is much less than 90 hours per year for both trains combined. However, if 90 hours of operation per year for either train (excluding tests) is to be exceeded, the COL applicant is required to demonstrate that maintenance or operating personnel activity or an incredible malfunction of the space heaters. In this case, a fire in the SGTS charcoal, like in the offgas system, would be a matter of plant availability and not of plant safety. The space heaters, located inside the SGTS filter housing, are powered only during SGTS standby and not during system operation. Therefore, the space heaters are not a potential cause of fire (and SGTS unavailability) when the SGTS is required to meet the licensing-basis release limits (and presumably inaccessible for repair).

Note that the space heaters each have a small fan which better distributes the heat and minimizes local warming by providing a more uniform temperature throughout the filter housing. This uniform heating further reduces the risk of fire by lowering local temperatures around the space heater and by improving the accuracy of the temperature measurements (used to detect high temperature) taken at necessarily discrete points within the filter housing.

(4) Degradation of the charcoal effectiveness between charcoal efficiency surveillance tests is not likely to occur. During normal operation, the filter is isolated, and valves upstream and downstream of the filter train are closed. DAMPERS Therefore, during SGTS standby, the potential for impurities entering the filter train and unacceptably reducing charcoal efficiency is small.

The ABWR SGTS charcoal bed thickness has been increased 5 cm to 15 cm as compared to the GESSAR II design. The additional 5 cm of charcoal provide an effective measure of protection against weathering or aging effects when the SGTS is placed into operation.

In addition to the increased charcoal bed depth, significantly more charcoal is provided than is required to meet the 2.5 mg iodine per gram carbon requirement. This added charcoal is used to meet the requirement specifying a residence time of 0.25 sec per 5 cm of bed depth. Approximately 332 kg of charcoal is required based on iodine loading calculated per Regulatory Guide 1.3 requirements, a 100% efficient charcoal adsorber, and no MSIV leakage. The SGTS charcoal adsorber is required to meet a 732 m/hr face velocity, which results in a normal 794 kg of charcoal assembly using a conservatively high 561kg/m^3 charcoal density with 6800 m^3/hr fan size, meeting the 0.25 sec per 5 cm of bed depth (732 m/hr) requirement of Regulatory Guide 1.52 (Position C.3.i), and using a conservatively high 561kg/m^3 charcoal density. The weight of charcoal will be adjusted to be consistent with the purchased charcoal density (usually less than 481 kg/m^3) and any dead space in the adsorber section itself.

THIS SHOULD BE "NOMINAL 794 kg"

Building ventilation exhaust radiation monitors during de-inerting, SGTS may be placed into service.

If purging (i.e., de-inerting) through the HVAC will [or does] result in a trip from the ventilation exhaust radiation monitors, then de-inerting will be [re-]initiated at a reduced rate through the SGTS. Use of SGTS during de-inerting is expected to be infrequent.

The design basis condition for the relevant dose analyses assumes that the large ventilation valves are closed, because the probability of a LOCA occurring at the same time the ventilation valves are open is very small. The large ventilation valves are, in fact, closed throughout normal plant operation except during inerting and de-inerting. The LOCA dose analyses do not assume any release from open containment isolation valves, either through the SGTS or through the normal ventilation system.

A realistic assessment of plant capability in support of the exclusion indicates that the ventilation valves, if open, would be isolated before significant fission products are transported to the containment atmosphere. "Significant" means fission products above that normally present in the primary system. A period much longer than the closing time of the ventilation valves would be required to generate conditions leading to the release of TID 14844-like source terms. Therefore, should a LOCA occur when the ventilation valves are open (valves expected to be open only during inerting or de-inerting), little fission product release to the environment would actually occur. Therefore, the plant design and analysis in this regard is conservative and bounds releases actually expected in the event of a LOCA.

CHANGE "VALVES" TO DAMPERS

6.5.1.4 Tests and Inspection

The SGTS and its components are periodically tested during construction and operation. These tests fall in three categories:

(1) Environmental qualification tests

(2) Acceptance tests as defined in ASME N509 and N510

(3) Periodic surveillance tests

The above tests are performed in accordance with the objectives of Regulatory Guide 1.52 and its references. Acceptance tests (including pre-operational tests) and periodic surveillance tests are defined and extensively described in ASME N509 and ASME N510. Testing requirements in ASME N509 are generally located in Section 5, "Components." ASME N510 provides details of each component functional test. These tests are summarized in Table 9-1 of ASME N509 and Table 1 of ASME N510. Specific surveillance testing requirements for SGTS are provided in Technical Specification 3.6.4.3 (Chapter 16). Environmental qualification testing is discussed in

Section 3.11 and is applicable to SGTS components. Dynamic qualification is addressed in Sections 3.9 and 3.10 for Seismic Category I equipment.

6.5.1.5 Instrumentation

Appendix 6B provides a discussion of the instrumentation for the SGTS. Control and instrumentation for the SGTS is also discussed in Subsections 7.3.1.1.5 and 7.3.2.5.

6.5.1.6 Materials

The construction materials used for the SGTS are compatible with normal and accident environments postulated for the area in which the equipment is located. The construction materials used in the dryer and filter trains are consistent with the recommendations of Regulatory Guide 1.52 and its references.

6.5.1.7 Operability and Effectiveness

Efficiency in the usual sense, can not be measured for adsorption systems. Adsorption is time dependent and therefore instantaneous containment-removal efficiency is meaningless. True efficiency tests are run on small, representative samples (test canisters) of the adsorbent using a radioactivity tagged tracer gas having similar properties and composition of those of the containment of interest (e.g., radioactive elemental iodine or methyl iodine). Because of the difficulty in handling radioactive materials, this type of test is generally not made in the field. The in-place field tests of installed systems are leak tests only. The iodine removal efficiency tests are carried out in a laboratory duplicating the field conditions as closely as possible.

The double filter train design for the SGTS depends on stationary components for normal (Routine) and accident operation. The pre-filter assembly is filled with glass fibers as are the pre and after HEPA filters. The charcoal iodine adsorber bed is located between the HEPA filters. All are located in a welded housing making up the filter train. The redundant active space heaters and fans operate only in the standby mode of the SGTS to dry the charcoal and maintain low relative humidity in the sealed train. Readiness for design operation is assured by effective surveillance tests.

The filter train availability depends on the stationary components replacement. The filter fiber glass sections are modularized for ease in handling. The charcoal is replaced by dumping old charcoal from below the bed and refilling with new charcoal from above. The integrity of the charcoal bed structure is maintained by limiting the moisture content of the charcoal in standby. The charcoal bed is oversized to reduce heating and weathering or aging effects. The bed has $\boxed{\text{795 kg}}$ of charcoal and is 150% thick over the calculated 335 kg required for adequate adsorber saturation and combustion protection.

OK IF P. 6.5-6 IS CHANGED TO NOMINAL 794 kg.

Section 2.14.6 Comment No. 4

Comment:

SSAR Table 6.2-7 pages 6.2-149 & 150:

a) valves T31-F32A/B and T31-F734A/D are listed as gate valves and are equipped with solenoid/electric operators. P&ID 6.2-39, Sheet 3 of 3 showed these as manually operated globe valves.

b) valves T31-F737A-D implies 4 valves whereas on P&ID, only valves A & B are shown.

Resolve discrepancies.

Resolution:

Section 2.14.8 Comment No. 2

Comment:

The CDD describes how the FCS component interfaces with the Remote Shutdown System. Figure 2.14.8 needs to be revised to reflect the RSS interfaces.

Resolution:

Section 2.14.8 Comment No. 3

Comment:

The CDD describes the cooling water requirements for operation of the FCS after a LOCA. Figure 2.14.8 needs to be revised to reflect the RHR interface with the FCS.

Resolution:

Section 2.15.5 Comment No. 1

Comment:

Revise CDM 2.15.5, page 2.15.5-12 to include ITAAC "Table 2.15-5b" description.

Resolution:

Section 2.15.5 Comment No. 3

Comment:

1. Revise ITAAC figure 2.15-5b to state "dP" not "DP" for differential instrumentation.

Resolution:

Section 2.15.5 Comment No. 4

Comment:

Revise ITAAC Table 2.15.5d, Item 10, to state "Section 2.15.5" not "Section t14".

Resolution:

Section 2.15.5 Comment No. 12

Comment:

Revise CDM design description on page 2.15.5-7 to state "On receipt of a DG start signal, both DG supply fans start. A space thermostat controls allow at least one fan in operation."

Resolution:

Section 2.15.5 Comment No. 13

Comment:

Revise ITAAC figure 2.15.5j to state "dP", not "DP" for the differential pressure instrument.

Resolution:

Section 2.15.5 Comment No. 14

Comment:

Revise SSAR section 9.4.5.1.1.2 to state "A negative pressure of 6.4 mm water gauge is normally maintained in the secondary containment relative to the outdoor atmosphere" as stated in SSAR section 6.5.1.3.1, 1st paragraph.

Resolution:

Section 2.15.6 Comment No. 1

Comment:

ITAAC item #3 includes minimum flows for the Reactor and Control Buildings. The SSAR, section 9.5.1.3.2, Fire Suppression System Requirements, specifies the Reactor Building. Resolve discrepancy.

Resolution:

Section 3.1 Comment No. 3

Comment:

Design acceptance criteria 1.b.(1) on page 3.1-4 should have the following words (from the SSAR) added "... in accordance with accepted human factors practices and principles."

Resolution:

Section 3.1 Comment No. 5

Comment:

Design acceptance criteria 2.a(1) on page 3.1-7 should add the following words (from the SSAR) "... in accordance with accepted human factors practices and principles."

Resolution:

Section 3.1 Comment No. 6

Comment:

Apply Comment No. 5 to 3.a(1)

Resolution:

Section 3.1 Comment No. 7

Comment:

Apply Comment No. 5 to 4.a(1)

Resolution:

Section 3.1 Comment No. 9

Comment:

Apply Comment No. 5 to 5.a(1)

Resolution:

Section 3.1 Comment No. 10

Comment:

Design acceptance criteria 5.a(1) should delete the word "equipment" so that HSI is not limited to equipment.

Resolution:

Section 3.1 Comment No. 12

Comment:

Apply Comment No. 5 to 6.a(1)

Resolution:

Section 3.1 Comment No. 13

Comment:

Design acceptance criteria 6.a.(4), page 3.1-15, correct typo to change "ask" to "task".

Resolution:

Section 3.1 Comment No. 15

Comment:

SSAR page 18C-1, paragraph 1 references section 18.5 (Operator Interface Design Implementation Requirements). This appears to be an incorrect reference. Section 18.5 is Remote Shutdown System.

Resolution:

Section 3.2 Comment No. 4b

Comment:

SSAR section 12.3.2.3 alludes to an Area Monitor in the spent fuel pool cleanup room. Could not locate one on the P&IDs or on the ARM equipment list.

Resolution:

Section 3.2 Comment No. 4c

Comment:

SSAR section 12.3.2.2.1(10), 1st sentence, "gr/cm3" should read "gm/cm3".

Resolution:

Section 3.2 Comment No. 4d

Comment:

See markup to correct typos on attached section 12A.

Resolution:

Section 3.2 Comment No. 4f

Comment:

SSAR Section 12.3.4.3, 3rd paragraph is confusing - "point" should be changes to "any point".

Resolution:

Section 3.2 Comment No. 4h

Comment:

SSAR Table 15.7.1 should be revised to indicate 400,000 uCi/sec Offgas Release Rate, 100,000 uCi/sec Design Basis Rate and 400,000 uCi/sec Maximum TS instead of the current 400,00; 100,00; and 400,00 uCi/sec respectively.

Resolution:

12A Appendix 12A Calculation of Airborne Radionuclides

12A.1 Calculation of Airborne Radionuclides

This appendix presents a simplified methodology to calculate the airborne concentrations of radionuclides in a compartment. This methodology is conservative in nature and assumes that diffusion and mixing in a compartment is basically instantaneous with respect to those mitigating mechanisms such as radioactive decay and other removal mechanisms. The following calculations need to be performed on an isotope-by-isotope basis to verify that airborne concentrations are within the limits of 10CFR20:

(1) For the compartment, all sources of airborne radionuclides need to be identified such as:

(a) Flow of contaminated air from other areas

(b) Gaseous releases from equipment in the compartment

(c) Evolution of airborne sources from sumps or water leaking from equipment

(2) Second, the primary sinks of airborne radionuclides need to be identified. This will primarily be outflow from the compartment but may also take the form of condensation onto room coolers.

(3) Given the above information the following equation will calculate a conservative concentration.

$$C_i = \frac{1}{V}\sum_j \frac{S_{ij}}{\left(\lambda_i + \sum_k R_{ijk}\right)}$$

Where:

C_i = Concentration of the i^{th} radionuclides in the room

V = Volume of room

S_{ij} = The j^{th} source (rate) of the i^{th} radionuclide to the room. These sources are discussed below.

R_{ijk} = The k^{th} removal constant for the j^{th} source and the i^{th} radionuclide as discussed below.

λ_i = Radionuclide decay constant

Evaluation Parameters

The following parameters require evaluation on a case-by-case basis dictated by the physical parameters and processes germane to the modeling process:

(1) S_{ij} is defined as the source rate for radionuclide i into the compartment. Typically, these sources take the form of:

(a) Inflow of contaminated air from an upstream compartment. Given the concentration of radionuclide i, c_i, in this air and a flow rate of "r", the source rate then becomes $S_{ij} = rc_i$.

(b) Production of airborne radionuclides from equipment. This typically takes two forms, gaseous leakage and liquid leakage.

(i) For gaseous leakage sources, the source rate is equal to the concentration of radionuclide i, c_i, and the leakage rate, "r", or $S_{ij} = rc_i$.

(ii) For liquid sources, the source rate is similar but more complex. Given a liquid concentration c_i and a leakage rate, "r", the total release from the leak is rc_i. The fraction of this release which then becomes airborne is typically evaluated by a partition factor, P_f which may be conservatively estimated from:

Noble Gases

$$P_f = 1$$

All others

$$P_f = \frac{h_t - h_f}{h_s - h_f}$$

where:

h_t = Saturated liquid enthalpy

h_f = Saturated liquid enthalpy at one atmosphere = 100.10 kcal/kg

h_s = Saturated vapor enthalpy at one atmosphere = 639.18 kcal/kg

Therefore, the liquid release rate becomes, rc_iP_f.

(2) R_{ijk} is defined as the removal rate constant and typically consists of:

(a) Exhaust rate from the compartment. This term considers not only the exhaust of any initially contaminated air, but also any clean air which may be used to dilute the compartment air.

(b) Compartment filter systems are treated by the equation:

$$R_{ijk} = (1-F_i)^* r_i$$

where

r_i = Filter system flow rate

F_i = Filter efficiency for radionuclide i

(c) Other removal factors on a case-by-case basis which may be deemed reasonable and conservative.

Example Calculation

(Values used below are examples only and should not be used in any actual evaluation.) This example will look at I-131 in a compartment 6.1 × 6.1 × 7.6 = 282.80 m^3 = V. First, all primary sources of radionuclides need to be identified and categorized.

(1) Flow into the compartment equals 424.8 m^3/hr with the input I-131 concentration equal to $2 \times 10^{-10} \mu Ci/ml$ (from upstream compartments) or $2.4 \times 10^{-11} Ci/sec$. No other sources of air either contaminated or clean air are assumed.

(2) The compartment contains a pump carrying reactor coolant with a maximum specified leakage rate of 0.000034 m^3/hr at 273.6°C.

(a) Conservatively it can be estimated based upon properties from steam tables (Note 1) that under these conditions 44% of the liquid will flash to steam and become airborne. Along with the flashing liquid, it is assumed that a proportional amount of I-131 will become airborne; therefore, P_f = 0.44.

(b) Using the design basis iodine concentrations for reactor water from Table 11.1-2 of 0.016μ Ci/gm of I-131, it is calculated that the pump is providing a source of I-131 of $5.0 \times 10^{-11} Ci/sec$ to the air (Note 2).

Second, the sinks for airborne material need to be identified. This example includes only exhaust which is categorized as flow out of the compartment at 150% per hour or 4.2×10^{-4} per second.

Therefore, for an equilibrium situation, the I-131 airborne concentration from this liquid source would be calculated from the following equation:

$$C = \frac{1}{V} S_1/(\lambda + R_1) + S_2/(\lambda + R_2)$$

where

V	=	Room volume = 282 80 m^3
S_1	=	Source rate in Curies per second = 5.0×10^{-11} Ci/sec from liquid
S_2	=	Source rate from inflow = 2.4×10^{-11} Ci/sec
λ	=	Isotope decay constant in units per second = 9.977×10^{-7}/sec
R_1	=	R_2 = removal rate constant per second (exfiltration) = 4.2×10^{-4} per second
C	=	6.2×10^{-10} µCi/ml of I-131.

NOTE:

(1) The assumption of 44% flashing at 273.6°C is extremely conservative; see Reference 12A-1 for a discussion of fission product transport.

(2) Water density assumed at 0.743 gm/cm^3 based upon standard tables for water at 273.6°C.

12A.2 References

12A-1 Paquette, et al, *Volatility of Fission Products During Reactor Accidents*, Journal of Nuclear Materials, Vol 130 Pg 129–138, 1985.

Section 3.[illegible] Comment No. 1

Comment:

Correct attached CDM typo.

Resolution:

For those piping systems using austenitic stainless steel materials as permitted by the design specification, the stainless steel piping material and fabrication process shall be selected to reduce the possibility of cracking during service. Chemical, fabrication, handling, welding, and examination requirements that reduce cracking shall be met.

Piping system supports shall be designed to meet the requirements of ASME Code Subsection NF.

For piping systems, the pipe applied loads on attached equipment shall be calculated and shown to be less than the equipment allowable loads.

Analytical methods and load combinations used for analysis of piping systems shall be referenced or specified in the ASME Code Certified Stress Report. Piping systems and their supports shall be mathematically modeled to provide results for piping system frequencies up to the analysis cutoff frequency. Computer programs used for piping system dynamic analysis shall be benchmarked.

typo

Systems, structures and components that shall be required to be functional during and following an SSE shall be protected against the dynamic effects associated with postulated high energy pipe breaks in Seismic Cateogry I and NNS piping systems. The Pipe Break Analyses Report shall specify the criteria used to postulate breaks and the analytical methods used to perform the pipe break analysis. For postulated pipe breaks, the Pipe Break Analysis Report shall confirm: (1) piping stresses in the containment penetration area shall be within their allowable stress limits, (2) pipe whip restraints and jet shield designs shall be capable of mitigating pipe break loads, and (3) loads on safety-related systems, structures and components shall be within their design loads limits. Piping systems that shall be qualified for leak-before-break design may exclude design features to mitigate the dynamic effects from postulated high energy pipe breaks.

typo

Structures, systems, and components that shall be required to be functional during and following an SSE shall be protected against the effects of spraying, flooding, pressure and temperature due to pstulated pipe breaks and cracks in Seismic Category I and NNS piping systems.

Piping systems shall be designed to provide clearance from structures, systems, and components where necessary for the accomplishment of the structure, system, or component's safety function as specified in the respective structure or system Design Description.

The as-built piping shall be reconciled with the piping design required by this section.

Section 3.4 Comment No. 2.1 & 2.4

Comment:

CDM 3.4B Instrumentation Setpoint Methodology, page 3.4-9. SSAR 7.1.2.10.9 Regulatory Guide 1.105, Instrumentation setpoints. CDM description is inconsistent with the standard and regulatory guide:

1. The CDM references a "nominal trip setpoint". This term is not defined in RG 1.105 or ISA 67.04-1982. See attached.

4. A definition of allowable value is not given in ISA 76.04-1982. RG 1.105 endorses the figure description as depicted in ISA 67.04-1982. The allowable value description listed in the CDM material is inconsistent with the standard (both 1982 and 1987) and RG 1.105. See attached.

Resolution:

Section 3.4 Comment No. 5

Comment:

Figure 4.3c, page 3.4-19 "NMS" should be blocked in with input designated to be consistent. See attached.

Resolution:

Section 3.4 Comment No. 6

Comment:

ITAAC Table 3.4, Item 5, ATWS, Design commitment. Reference is made to both APRM and SRNM not downscale. This is not consistent with other design material or Figure 15E-2.

Resolution:

The determination of nominal trip setpoints includes consideration of the following factors:

Design Basis Analytical Limit

In the case of setpoints that are directly associated with an abnormal plant transient or accident analyzed in the safety analysis, a design basis analytical limit is established as part of the safety analysis. The design basis analytical limit is the value of the sensed process variable prior to or at the point which a desired action is to be initiated. This limit is set so that associated licensing safety limits are not exceeded, as confirmed by plant design basis performance analysis.

Allowable Value

SEE ENCLOSED STANDARD

An allowable value is determined from the analytical limit by providing allowances for the specified or expected calibration capability, the accuracy of the instrumentation, and the measurement errors. The allowable value is the limiting value of the sensed process variable at which the trip setpoint may be found during instrument surveillance.

Nominal Trip Setpoint

SEE ENCLOSED STANDARD

The nominal trip setpoint value is calculated from the analytical limit by taking into account instrument drift in addition to the instrument accuracy, calibration capability, and the measurement errors. The nominal trip setpoint value is the limiting value of the sensed process variable at which a trip action will be set to operate at the time of calibration.

Signal Processing Devices in the Instrument Channel

Within an instrument channel, there may exist other components or devices that are used to further process the electrical signal provided by the sensor (e.g., analog-to-digital converters, signal conditioners, temperature compensation circuits, and multiplexing and demultiplexing components). The worst-case instrument accuracy, calibration accuracy, and instrument drift contributions of each of these additional signal conversion components are separately or jointly accounted for when determining the characteristics of the entire instrument loop.

5) Not all parameters have an associated design basis analytical limit (e.g., main steamline radiation monitoring). An allowable value may be defined directly based on plant licensing requirements, previous operating experience or other appropriate criteria. The nominal trip setpoint is then calculated from this allowabie value, allowing for instrument drift. [Where appropriate, a nominal trip setpoint may be determined directly based on operating experience.]

ISA 67.04 1994 - ONCE INSTALLED HISTORICAL RESULTS MAY BE EVALUATED AND STATS REVIEWED.

Procedures will be used that provide a method for establishing instrument nominal trip setpoint and allowable value. Because of the general characteristics of the instrumentation and processes involved, two different methods are applied:

(1) Computational

(2) Historical data

INCONSISTENT WITH PREVIOUS PAGE

The computational method is used when sufficient information is available regarding a dynamic process and the associated instrumentation. The procedure takes into account channel instrument accuracy, calibration accuracy, process measurement accuracy, primary element accuracy, and instrument drift. If the resulting nominal trip setpoint and allowable value are not acceptable when checked to ensure that they will not result in an unacceptable level of trips caused by normal operational transients, then more rigorous statistical evaluation or the use of actual operational data may be considered.

Some setpoint values have been historically established as acceptable, both for regulatory and operational requirements. These setpoints have non-critical functions or are intended to provide trip actions related to gross changes in the process variable. The continued recommendation of these historically accepted setpoint values is another method for establishing nominal trip setpoint and allowable values. This approach is only valid where the governing conditions remain essentially unaltered from those imposed previously and where the historical values have been adequate for their intended functions.

The setpoint methodology plan requires that activities related to instrument setpoints be documented and stored in retrievable, auditable files.

Equipment Qualification (EQ)

Qualification of safety-related instrumentation and control equipment is implemented by a program that assures this equipment is able to complete its safety-related function under the environmental conditions that exist up to and including the time the equipment has finished performing that function. Qualification specifications consider conditions that exist during normal, abnormal, and design basis accident events in terms of their cumulative effect on equipment performance for the time period up to the end of equipment life.

The material discussed herein identifies an EQ program that addresses the spectrum of design basis environmental conditions that may occur in plant areas where I&C equipment is installed. Not all safety-related I&C equipment will experience all of these conditions; the intent is that qualification be performed by selecting the conditions applicable to each particular piece of equipment and performing the necessary qualification.

As-built I&C components are environmentally qualified if they can withstand the environmental conditions associated with design basis events without loss of their safety functions for the time needed to be functional. Safety-related I&C components are designed to continue normal operation after loss of HVAC. The environmental conditions are as follows, as applicable to the bounding design basis events: Expected time-dependent temperature and pressure profiles, humidity, chemical effects, radiation, aging, seismic events, submergence, and synergistic effects which have a significant effect on equipment performance.

I&C equipment environmental qualification is demonstrated through analysis of the environmental conditions that would exist in the location of the equipment during and following a design basis accident and through a determination that the equipment is qualified to withstand those conditions for the time needed is functional. This determination may be demonstrated by:

(1) Type testing of an identical item of equipment under identical or similar conditions with a supporting analysis to show that the equipment to be qualified.

(2) Type testing of a similar item of equipment with a supporting analysis to show that the equipment is qualified.

(3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment is qualified.

(4) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions to show that the equipment is qualified.

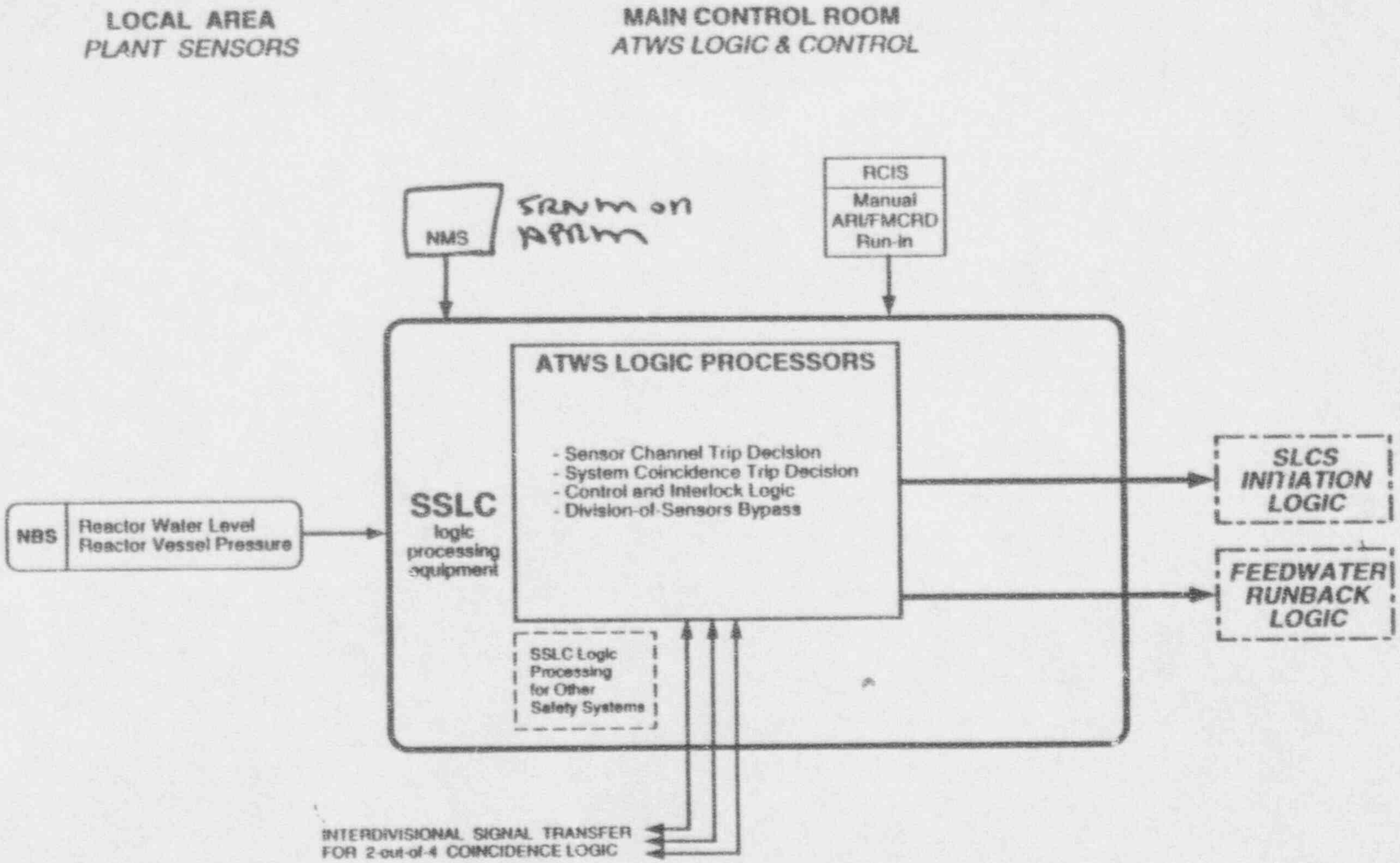
The installed condition of safety-related I&C equipment is assured by a program whose objective is to verify that the installed configuration is bounded by the test configuration and test conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 3.4, Items 7 through 15, provides a definition of the inspections, tests and analyses, together with associated acceptance criteria, which will be used to demonstrate compliance with the above commitments for hardware and software development, electromagnetic compatibility, instrument setpoint methodology, and equipment qualification.

C. Diversity and Defense-in-Depth Considerations

Subsection B discusses processes for developing hardware and software qualification programs that will assure a low probability of occurrence of both random and common-mode system failures for the installed ABWR I&C equipment. However, to address the



Notes:
1. Diagram represents one of four ATWS divisions.
2. Remaining ATWS functions are processed as part of Recirculation Flow Control System logic and Nuclear Boiler System logic.

Figure 3.4c Anticipated Transient Without Scram (ATWS) Control Interface Diagram

1 PURPOSE

The purpose of the standard is to develop a basis for establishing setpoints for actions determined by the design basis for protection systems and to account for instrument errors and drift in the channel from the sensor through and including the bistable trip device.

2 SCOPE

This standard defines minimum requirements for assuring that setpoints are established and held within specified limits in nuclear safety-related instruments in nuclear power plants.

3 DEFINITIONS

Accuracy - Degree of conformity of an indicated value to a recognized accepted standard value, or ideal value. [1]

Design Basis - The Design Basis for protection systems for nuclear power generating stations is delineated in IEEE Standard 279-1971, "IEEE Standard for Protection Systems for Nuclear Power Generating Stations," Part 3, Design Basis.

Drift - An undesired change in the output-input relationship over a period of time. [1,11]

Dynamic response - The behavior of the output of a device as a function of the input, both with respect to time. [1]

Foldover - A characteristic of the steady-state or dynamic conditions of a device for which, at a point, a further change in the input signal produces an output signal which reverses its direction from the specified input-output relationship.

Hysteresis - That property of an element evidenced by the dependence of the value of the output, for a given excursion of the input, upon the history of prior excursions and the direction of the current traverse. [1]

Instrument channel - An arrangement of components and modules as required to generate a single protective action signal when required by a generating station condition. A channel loses its identity where single protective action signals are combined. [2]

Instrument range - The region between the limits within which a quantity is measured, received, or transmitted, expressed by stating the lower and upper range values. [1]

Limiting Safety System Setting (LSSS) - Limiting Safety System Settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. [3]

Note: For the purposes of this standard, the phrase "nuclear reactors" used in this definition should be understood to mean "nuclear power plants."

Protective action - The initiation of a signal or operation of equipment within the protection system, or protective action system, for the purpose of accomplishing a protective function in response to a generating station condition having reached a limit specified in the design basis. [4]

Protective function - The sensing of one or more variables associated with a particular generating station condition, the signal processing, and the initiation and completion of the protective action within the values of the variables established in the design basis. [2]

Protection system - The electrical and mechanical devices (measured process variables to protective action system input terminals) involved in generating those signals associated with the protective functions. These signals include those that initiate reactor trip, engineered safety features, and auxiliary supporting features. [4]

Repeatability - The closeness of agreement among a number of consecutive measurements of the output for the same value of the input under the same operating conditions, approaching from the same direction, for full range traverses. [1]

Nuclear safety-related instrumentation - That which is essential to:

(1) emergency reactor shutdown;
(2) containment isolation;
(3) reactor core cooling;
(4) containment or reactor heat removal;
(5) prevent or mitigate a significant release of radioactive material to the environment; or is otherwise essential to provide reasonable assurance that a nuclear power plant can be operated without undue risk to the health and safety of the public.

Saturation - A characteristic of the steady state or dynamic conditions of a device under which, at a point, a further change in the input signal, produces no additional change in the output signal.

Sensor - That portion of a channel which responds to changes in a plant variable or condition, and converts the measured process variable into an instrument signal.

Setpoint - A predetermined level at which a bistable device changes state to indicate that the quantity under surveillance has reached the selected value. [5]

Test interval - The elapsed time between the initiation of identical tests on the same sensor, channel, train, load group, or other specified system or device. [5]

4 ESTABLISHMENT OF SETPOINTS

Setpoints in nuclear safety-related instruments shall be selected to provide sufficient margin between the trip setpoint and the safety limits to account for accuracies, drift, uncertainties and dynamic responses. Detailed requirements for safety-related instrument setpoint relationships are given in the sections which follow as illustrated in Figure 1.

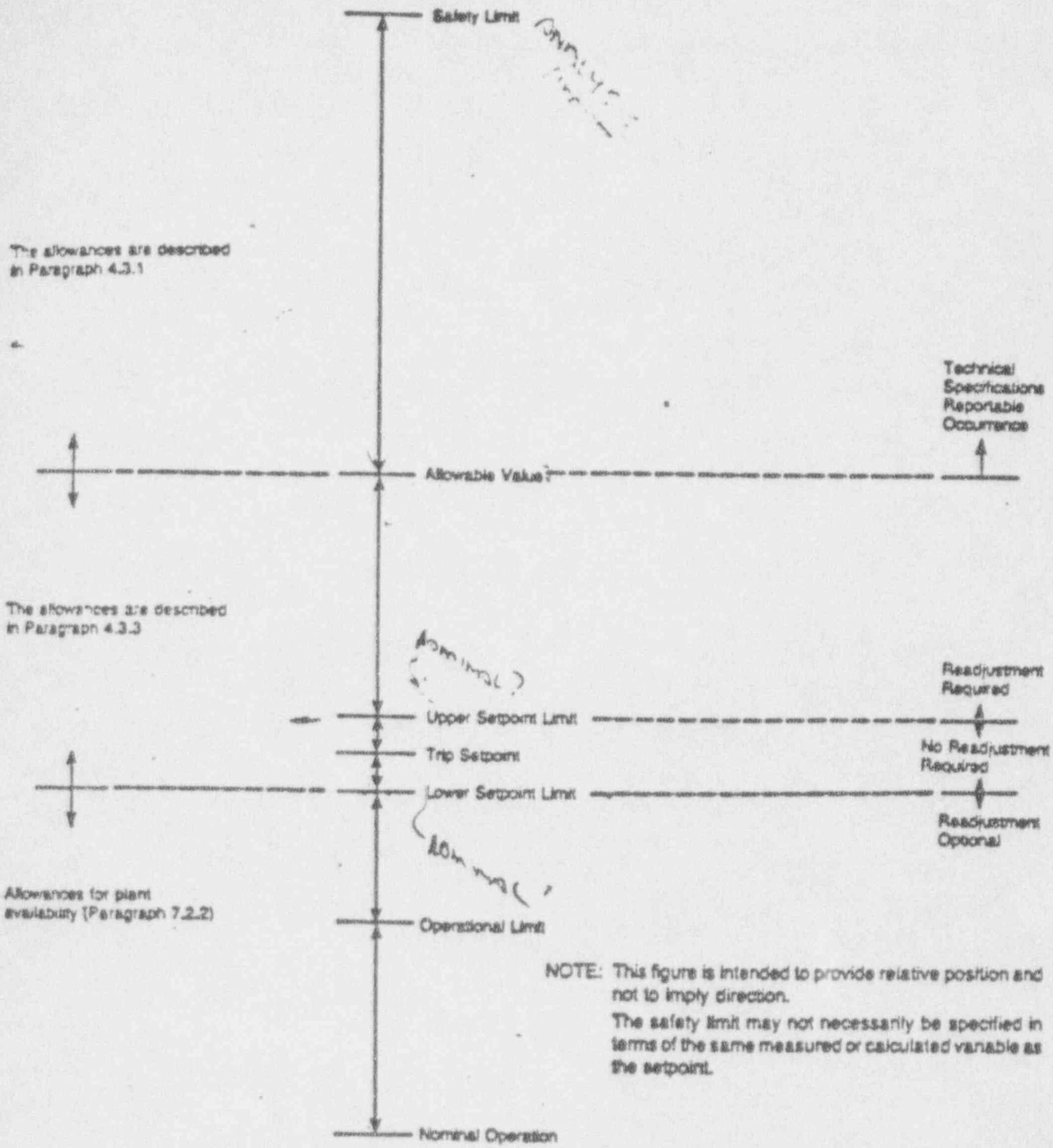


Figure 1. Nuclear Safety-Related Instrument Setpoints Relationships

4.1 Safety Limits

Safety limits for nuclear reactors are limits upon important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity. [3] The safety limit may not necessarily be specified in terms of the same measured or calculated variable as the setpoint. For example, a setpoint using temperature as a measured variable may be related to a safety limit specified in terms of Departure from Nucleate Boiling Ratio (DNBR)

4.2 Safety Analysis

The conclusions of the safety analysis are assured in part by establishing appropriate safety system setpoints to be stated in the technical specifications and maintained through operating procedures. The selection of setpoints for safety-related instruments shall be documented or referenced in the basis for the technical specifications including the parameters and assumptions upon which the setpoint selection was based.

4.3 Limiting Safety System Settings

Limiting Safety System Settings (LSSS) shall be selected such that operation within LSSS provides assurance that the physical barriers will not be damaged beyond acceptable limits during anticipated operational occurrences and accidents. For each LSSS a trip setpoint and its associated allowable value shall be established. (See Figure 1.)

4.3.1 The allowances between the allowable value and the safety limit shall include the following items unless they are included in the determination of the safety limit:

(1) Accuracy (including drift) of components not tested when setpoint is measured. Setpoint measurements shall be made by:
   (a) Perturbing the monitored variable (the same or a substitute process variable), and noting the point at which a channel trip occurs, or:
   (b) Substituting a known signal in the instrument channel as close to the monitored variable as practical and noting the point at which a channel trip occurs. Justification for selecting Item (b) over (a) shall be documented.

(2) Accuracy of test equipment for:
   (a) Measuring setpoints
   (b) Calibrating sensors for the case where sensors are not included in setpoint measurements.

(3) Process measurement accuracy. Examples are the effect of fluid stratification on temperature measurement and the effect of changing fluid density on level measurements.

(1) acc. of comp. not calibrated → — safety limit
— allowable value
(2) test equipment
(3) process measurement accuracy
(4) overshoot
→(6) environmental effects

(4) The effects of potential transient overshoot determined in the design basis events analyses.

(5) The effects of the time reponse characteristics of the total instrument channel, including the sensor.

(6) Environmental effects on equipment accuracy or time response characteristics caused by anticipated operational occurrences or accidents for those systems required to mitigate the consequences of such events.

The above items shall be combined in one of the following five ways:

(1) Algebraically,
(2) Square root of the sum of the squares,
(3) Statistically,
(4) Probabilistically, or
(5) Combinations of 1 thru 4.

Justification shall be provided for the adequacy of the method used.

4.3.2 Where items listed in Paragraph 4.3.1 are accounted for by compensating the signal(s) representing the monitored variable(s) prior to comparison with the trip setpoint, these items need not be considered in the allowance between the safety limit and the allowable value.

4.3.3 The trip setpoint shall be a value which allows margin for drift and adjustment. The trip setpoint shall be chosen so that the corresponding allowable value is not exceeded due to the following:

(1) Drift of that portion of the instrument channel which is tested when the setpoint is determined.

(2) Actual setting of the setpoint within an allowable tolerance of upper and lower setpoint limits. (See Figure 1.)

The band between upper and lower setpoint limits shall account for the ability to adjust the setpoint and minimize the need for frequent adjustments.

5 INSTRUMENT PERFORMANCE AND SETPOINT SETTING

Setpoints shall be specified in units of the monitored value.

Instrument performance requirements shall be specified such that during the interval between setpoint tests the actual setpoint does not exceed the allowable value due to expected drift.

Instrument performance requirements shall be specified for that portion of the instrument channel not tested (Paragraph 4.3.1) such that the parameters remain within the values assumed in the determination of the allowable value.

Setpoints shall be located in that portion of the instrument's range which has the required accuracy.

Instrument performance requirements shall be specified such that as long as the process variable exceeds the setpoint, the protective action of that instrument channel is not negated by saturation, foldover, or any other cause for expected values of the process variable.

Instrumentation calibration correction factors shall be identified and documented. Correction factors which have been incorporated in the determination of the setpoint (for example, to compensate for differences in physical location, temperature or pressure between the required point of measurement and actual sensor location) shall be separately identified.

6 QUALIFICATION

The nuclear safety-related instrumentation hardware and software qualification shall be documented and available to verify all parameters used in determining the setpoints, including:

(1) The value of setpoint drift during proposed test intervals due to expected exposure to normal operating temperature, pressure, humidity, power variation, electromagnetic interference, vibration, seismic acceleration and radiation exposure.

(2) The time response characteristics or other response characteristics of the instrument channel.

(3) The instrument channel performance such as accuracy, repeatability and hysteresis at the trip setpoint and at the allowable value under design basis conditions.

These requirements are supplemental to those of IEEE Standard 323-1974. [10]

7 MAINTENANCE OF SETPOINTS

Maintenance of setpoints shall include all actions taken to assure that the instrumentation is installed and continues to operate within the design requirements used to establish the setpoints. The following sections address those aspects of nuclear safety-related instrument setpoint maintenance that are necessary to support the establishment of the allowable values and trip setpoints as described in Section 4. Specific guidance for implementing each of the following maintenance activities can be found in other industry standards (See references 6 through 9, for examples.)

7.1 Installation

Installation requirements shall include:

(1) Receipt, storage and handling provisions to prevent instrumentation degradation.

(2) Provisions for necessary access and other design features to assure setpoint maintenance.

7.2 Operation

7.2.1 Initial Calibration and Operation

Nuclear safety-related instrument channels shall be calibrated, functionally tested and set at their trip setpoint as soon as practicable after installation and again prior to initial criticality, where practical, to determine if the drift rate of the channel meets design requirements. Inability to perform these tests shall be justified and documented.

If within this period the drift rate of the channel fails to meet the design requirements, an evaluation shall be conducted to determine the cause. The evaluation shall include consideration of the installation (including all possible environmental effects), adequacy of the supplied instrumentation, accuracy of calibration, and calibration techniques. This evaluation shall provide the basis for proper and timely resolution and shall be documented.

7.2.2 Periodic Testing

Testing of safety-related instrumentation shall be in accordance with the technical specifications. Written procedures shall be used to verify the proper operation of the instrumentation, including each instrument channel's compliance with design requirements related to setpoints. These procedures shall include, as a minimum, requirements to record sufficient data on each channel to determine the true setpoint in terms of measured or derived process variables, before any adjustments are made.

If the "as found" setpoint indicates the setpoint is within the "no readjustment" band (See Figure 1.) or that calculations based on the analog value would result in setpoints within the "no readjustment" band, documentation of the results is the only required action. If the "as found" setpoint exceeds the upper setpoint limit, readjustment shall be performed to bring this channel back within the "no readjustment" band. The "as found" and "as left" setpoint shall be recorded. If the "as found" setpoint was also beyond the allowable value, a review shall be conducted immediately to determine the availability of the other redundant channels of the same protective function and their setpoints. Based on this review and subsequent evaluation, it may be necessary to decrease the time between tests in order to ensure proper operation. A review of the parameters verified in Paragraph 7.2.1, above shall be required to determine the cause. The action taken when the allowable value has been exceeded shall be based on the measured drift rates determined by previous "as left" and current "as found" data.

This evaluation shall be documented.

If subsequent tests show the allowable value continues to be exceeded the following shall be considered:

(1) Upgrading the instrument system
(2) Revising the required tolerances for the trip setpoint
(3) Revising the upper setpoint limit and lower setpoint limit ("no readjustment" band)
(4) Revising the test interval.

This evaluation shall be documented.

unnessary

If the "as found" setpoint is below the lower setpoint limit, readjustment may be made to avoid necessary trips, but is not mandatory. The "as found" and "as left" setpoint shall be recorded.

Should these data indicate drift rates considerably less than originally expected, testing intervals or tolerances may be revised accordingly, with suitable justification and documented.

7.3 Test Equipment

A system shall be established to ensure the accuracy and adequacy of the test equipment used to verify setpoints and tolerances of safety-related instrumentation. Calibration records shall identify all test equipment by serial number. The test equipment shall be calibrated at specified intervals and shall be traceable to the U.S. National Bureau of Standards or have a known valid relationship to physical constants. If test equipment is found out of tolerance, an evaluation shall be conducted to determine the effect on safety-related instrumentation calibrated with that equipment since its last calibration. The evaluation including corrective action taken shall be documented. The accuracy of the test equipment used shall equal or exceed that required of the instrumentation under test.

7.4 Repair and Replacement

Replacement of material, parts and components shall be "in kind." Substitutions shall be evaluated and documented to assure equal or better performance than that provided in the design basis.

REFERENCES

1. Definition per ISA - S51.1 (1976) "Process Instrumentation Terminology."
2. Definition per IEEE Standard 279-1971 "Criteria for Protection System for Nuclear Power Generating Stations."
3. Definition per "Code of Federal Regulations" Title 10, Part 50, dated January 1, 1978, Paragraph 50.36.
4. Definition per IEEE Trial Use Standard 603-1977 "Criteria for Safety Systems for Nuclear Power Generating Stations."
5. Definition per IEEE Standard 380-1975 "Definition of Terms Used in IEEE Standards on Nuclear Power Generating Stations."
6. IEEE Standard 338-1975, "IEEE Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems."
7. ANSI N45.2-1971, "Quality Assurance Program Requirements for Nuclear Power Plants."
8. IEEE Standard 352-1975, "IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems."
9. IEEE Standard 498-1975, "IEEE Standard Supplementary Requirements for the Calibration and Control of Measuring and Test Equipment Used in the Construction and Maintenance of Nuclear Power Generating Stations."
10. IEEE Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."
11. The committee chose this specific definition for drift to explain the output-input relationship of a safety-related instrument channel.

INFORMATIVE REFERENCES

The Instrument Society of America (ISA) has developed standards for the nuclear industry through the SP67 Nuclear Power Plant Standards Committee (NPPSC)

ANSI/ISA-67.01-1981, "Transducer and Transmitter Installation for Nuclear Safety Applications."

ISA-S67.02, "Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants."

Miscellaneous Comments:

1. SSAR Acronym use: revise SSAR lis* as marked-up for VAC and VDC. SSAR list needs to reflect PRA as Probabilistic Risk Assessment. SSAR acronym list is incomplete, such as: TN, MPT, PIP, D/G, IED, NBS, UAT, PMG, M/C, RAT, SBO, MVA. Recommend total SSAR search to identify all missing acronyms.

   Resolution:

2. SSAR Section 14.2.12.1.45 used loss of offsite power (LOP) for loss of preferred power (LOPP). Revise to be consistent with CDM and other SSAR sections.

   Resolution:

3. SSAR page 9.3-9, revise as marked-up - see attached.

   Resolution:

4. SSAR pages 9.3-26 and 9.3-27 are not continuous, information is missing in section 9.3.8.2.3 that should be added to SSAR.

   Resolution:

5. SSAR Table 14.3-10, clarify the statement at top of page with respect to RCIC and HPCF suction piping as outlined in mark-up.

   Resolution:

6. SSAR page 7.3-3, revise as shown on markup.

   Resolution:

List of Acronyms (Continued)

TCS	Turbine Control System
TCV	Turbine Control Valve
TCW	Turbine Building Cooling Water (System)
TGSS	Turbine Gland Sealing System
THA	Time-History Accelerographs
TIP	Traversing Incore Probe or Traversing Ion Chamber
TIU	Technician Interface Unit
TLU	Trip Logic Unit
TRS	Test Response Spectra
TSC	Technical Support Center
TSV	Turbine Stop Valve
TSW	Turbine Service Water
U/D	Upper Drywell
UHS	Ulumate Heat Sink
UPS	Uninterruptible Power System
USE	Upper Shelf Energy
USMA	Uniform Support Motion Response Spectrum Analysis
USNRC	United States Nuclear Regulatory Commission
VAC	Volts Direct Current switch
VDC	Volts Alternating Current
VDU	Video Display Unit
VLC	Vent Line Clearing
VWO	Valves-Wide-Open
WDSC	Wetwell and Drywell Spray Cooling (Mode of RHR)
WDVB	Wetwell-to-Drywell Vacuum Breaker
WDVBS	Wetwell-to-Drywell Vacuum Breaker System
ZIS	Zinc Injection System
ZSI	Zone Selective Interlocks

for lead unit and standby unit of air compressors and dryers shall be switched periodically. The pressure setpoints for these operational changes are adjustable, depending on air requirements that might exist.

During normal operation, the nonsafety-related nitrogen users within containment are downstream of P52-F277 and P54-F208. (The safety-related nitrogen users are downstream of P54-F008A and B.) Should the AC/HPIN Systems become unable to supply nitrogen to the non-safety-related users downstream of P52-F277, the operator may remote manually open P52-F257 to supply instrument air to these users (Figure 20.3-55).

During refueling, the IAS provides compressed air instead of nitrogen gas to the users located inside containment in Figure 9.3-6.

Acceptance Criterion II.1 of SRP Section 9.3.1 requires that the maximum particle size of 3 microns in the air stream at the instrument. The corresponding maximum particle size for the ABWR design is 5 microns. Experience to date for plants with a maximum filtered particle size of 5 microns in the compressed gases has been very satisfactory.

All equipment using instrument air shall be capable of operating with air of the quality listed above.

9.3.6.3 Safety Evaluation

The operation of the IAS is not required to assure any of the following:

(1) Integrity of the reactor coolant pressure boundary.

(2) Capability to shut down the reactor and maintain it in a safe shutdown condition.

(3) Ability to prevent or mitigate the consequences of accidents which can result in potential offsite exposures comparable to the guideline exposures of 10CFR100.

However, the IAS incorporates features that assure this operation over the full range of normal plant operations. If IAS pressure falls below a desired limit, air from the Service Air System (SAS) is automatically added from a tie-line. An air receiver is provided to maintain air supply pressure if all of the IAS and SAS compressors fail. Pneumatic-operated devices are designed for a failsafe mode and do not require continuous air supply under emergency or abnormal conditions.

The instrument air system does p orivide air service to a number of safety-related systems and components. The loss of air to these systems will result in current or new valva positions. These positions have been evaluated. The subject system safety functions have

(3) **Provision of Spare Pumps**—All sumps which process radioactive wastes are supplied with two pumps each. Each pump is sized to handle the maximum anticipated flow into the sump. Thus, each sump has one operating pump and one pump on standby.

(4) **Leak Detection**—The Reactor Building and drywell sumps have instrumentation which permits detection of excessive leakage and provides for an alarm upon high leakage rates.

(5) **Sump Coolers**—The Reactor Building drywell equipment drain sumps each have provisions for measuring their sump liquid temperature and automatically recirculating the sump contents through a drain cooler to cool the sump contents if the temperature exceeds 60°C. In the event of a LOCA signal, all drywell sump pumps are automatically isolated, to preclude the possible uncontrolled release of primary coolant.

(6) **Detergent Drains**—The detergent drain sump collects laundry and shower drains. The detergent drains are transferred to the detergent drain tanks in the Radwaste System. These detergent wastes are kept separate from other wastes, since detergent wastes are processed in a separate process train in the Radwaste System.

9.3.8.2.3 Component Description

Drain System components are as follows:

(1) **Collection Piping**— In all area of potential radioactive contamination, the collection system piping for the liquid system is of stainless steel for embedded and chemical drainage, and carbon steel for suspended drainage. Offsets in the piping are provided, where necessary, for radiation shielding. In general, the fabrication and installation of the piping provides for a uniform slope that causes gravity flow to the appropriate sump. During construction, equipment drain piping is terminated not less than 5 cm above the finished floor or drain receiver at each location where the discharge from equipment is to be collected. The connections to the individual equipment are made after the equipment is installed in its proper location.

(2) **Collection Sumps** (potentially radioactive drains)—These sumps are provided with a well-fitting, but not gastight, steel plate access cover for convenient maintenance access, as well as to minimize airborne contamination.

(3) **Equipment Drains**—Equipment that may be pressurized during drainage, and that drains via direct or indirect drain connection to the floor drain system, is designed so that the equipment discharge flow does not exceed the gravity flow capacity of the drainage header at atmospheric pressure.

(4)

during system startup. It is also installed, as required, to preserve the integrity of the drainage systems. Floor drains in areas not restricted because of potential radioactivity are provided with caulked or threaded connections.

(5) Cleanouts—In collection system piping from areas of potential radioactivity, cleanouts are provided, when practicable, at the base of each vertical riser where the change of direction in horizontal runs is 90°, at offsets where the aggregate change is 135° or greater, and at maximum intervals of 50 feet. Equipment hubs and floor drains are also used as cleanout points. Cleanouts are welded directly to the piping and located with their access covers flush with the finished floor or wall.

9.3.8.2.4 Safety Evaluation

The Drain Transfer System is not safety-related. Sumps designated as containing radioactive wastes are equipped with charcoal filters in the vents. In the event of a LOCA signal, all drywell sumps are automatically isolated to preclude the uncontrolled release of primary coolant outside the PCV.

9.3.8.2.5 Tests and Inspections

Drywell and Reactor Building floor and equipment drain sumps are provided with the following instruments and controls:

(1) High and low level switches are provided on each sump pump to start and stop the sump pump automatically. A separate high-high level switch set at a higher level starts the second pump and simultaneously actuates an alarm in the main control room.

(2) Leak detection is effected by monitoring the frequency and duration of pump runs.

9.3.9 Hydrogen Water Chemistry System

9.3.9.1 Design Bases

9.3.9.1.1 Safety Design Basis

The Hydrogen Water Chemistry (HWC) System is non-nuclear, non-safety-related and is required to be safe and reliable, consistent with the requirement of using hydrogen gas. The hydrogen piping in the Turbine Building shall be designed in accordance with the guidance Regulatory Guide 1.29 "Seismic Design Classifications", Section C.2 to comply with modified BTP CMEB 9.5-1, Part C.5.d(5).

Table 14.3-10 TMI Issues (Continued)

SSAR Entry	Parameter	SSAR Value
	RCIC and HPCF Do not Share Any Common Suction Piping with RHR	
	RCIC	—
	HPCF	—
	LPFL	—
	ECCS Have Minimum Flow Protection for All Operating Modes	
	RCIC	—
	HPCF	—
	RHR	—
	Number of RCW Divisions	3
	Individual ECCS Pumps Can be Isolated Without Affecting Other ECCS Pumps	
	RCIC	—
	HPCF	—
	RHR	—
	ABWR has Water Level Measurement Directly on the Vessel	—
	Containment Sprays are Manually Initiated	—
	Essential Equipment Inside the Containment is Qualified for Harsh Environment	—
	ADS Automatically Depressurizes the Vessel on Low Water Level	—
	ABWR has Manual Vessel Depressurization Capability	—
1A.2.34	III.D.1.1(1) Review Information Submitted by Licensee Pertaining to Reducing Leakage from Operating Systems	
	Inboard and Outboard Isolation Valves on All Lines Which Penetrate Primary Containment	—
	ABWR has a Leak Detection and Isolation System	—
	MSIV Closure on:	
	High Temperature in Steam Tunnel	—
	High Temperature in Turbine Building	—
	High Radiation in HVAC Air Exhaust Results in:	
	Closure of HVAC Air Ducts to Reactor Building	—
	Closure of Containment Purge and Vent Lines	—

?

optical fiber data link to the logic processing units in the main control room. All four transmitter signals are fed into the two-out-of-four logic for each of the two divisions (II & III). The initiation logic for HPCF sensors is shown in Figure 7.3-1.

Drywell pressure is monitored by four pressure transmitters in the same four-division configuration described above. Instrument sensing lines that terminate outside the drywell allow the transmitter to communicate with the drywell interior. Each drywell high-pressure trip channel provides an input into two-out-of-four trip logic shown in Figure 7.3-1.

The HPCF System is initiated on receipt of a reactor vessel low water level signal (Level 1.5) or drywell high-pressure signal from the trip logic. The HPCF System reaches its design flow rate within 36 seconds of receipt of initiation signal. Makeup water is discharged to the reactor vessel until the reactor high water level is reached. The HPCF System then automatically stops flow by closing the injection valve if the high water level signal is available.

This valve will reopen if reactor water level subsequently decreases to the low initiation level. The system is arranged to allow automatic or manual operation. The HPCF initiation signal from the NBS also initiates the standby diesels in the respective divisions.

An AC motor-operated valve and a check valve are provided in both branches of the pump suction. The pump suction can be aligned through one branch to the condensate storage tank or aligned through the other branch to the suppression pool. The control arrangement is shown in Figure 7.3-1. Reactor grade water in the condensate storage tank is the preferred sour[illegible] On receipt of an HPCF initiation signal, the condensate storage tank suction valves are automatically signaled to open (they are normally in the open position unless the suppression pool suction valves are open). If the water level in the condensate storage tank fails below a preselected level, first the suppression pool suction valves automatically open and then the condensate storage tank suction valves automatically close. Four level transducers (one in each electrical division) are used to detect low water level in the condensate storage tank. Any two-out-of-four transducers can cause the suppression pool suction valves to open and the condensate storage tank valves to close. The suppression pool suction valves also automatically open if high water level is detected in the suppression pool. Four level transducers (one in each electrical division) monitor this water level and

fails