GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

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MFN 008-83
RWS 001-83

January 21, 1983

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Mr. D.G. Eisenhut, Director
Division of Licensing

SUBJECT: IN THE MATTER OF 238 NUCLEAR ISLAND
GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT
(GESSAR II) DOCKET NO. STN 50-447

Attached please find final draft responses to the Instrumentation and Control Systems Branch (ICSB) questions in the Commission's October 5, 1982 request for additional information. These responses reflect the NRC/GE information exchange meetings held in Bethesda October 14 & 15, 1982; San Jose December 7-9, 1982; and again in Bethesda January 11-13, 1983.

Most questions are addressed in this transmittal. Responses to questions 421.06, 10, 11, 15, 24, 25, 29, 33, 43, 45, 46, 49 and 54 remain unchanged since their initial submittal on November 19, 1982. Therefore, they are final drafts and are not duplicated here.

In accordance with the schedule submitted on December 22, 1982, the remainder of the ICSB final draft responses will be sent on January 28, 1983.

An amendment is scheduled for February 1983 to formalize the responses.

Sincerely,

Willa for

Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

Attachments

cc: M.J. Virgilio, NRC
D.C. Scaletti, NRC
L.S. Gifford, GE-Bethesda (Without Attachments)
F.J. Miraglia, (Without Attachments)
C.O. Thomas, (Without Attachments)

E003

421.01 QUESTION FINAL DRAFT

You indicate in Section 7.1.2.2, 7.1.2.3 and 7.1.2.4 of your FSAR that your statements regarding the applicability of the conformance of each of your proposed systems with the General Design Criteria (GDC), regulatory guides and the appropriate industry standards are included in Table 7.1-3 through 7.1-6. However, Tables 7.1-3 through 7.1-6 are inconsistent with Table 7-1 of Section 7.1 of the Standard Review Plan (SRP). Identify and deviations between Tables 7.1-3 through 7.1-6 of your FSAR an Table 7-1 of the SRP.

421.01 RESPONSE

The existing tables (7.1-3 through 7.1-6) were arranged consistent with the previous SRP = NUREG 75/087, Table 7-1. However, these tables are in process of being revised with headings consistent with NUREG 0800, Table 7-1. Since there is actually less information required (ie, applicability of regulatory criteria is more specifically refined to appropriate systems), no deviations are anticipated. BTP applicability will not be indicated directly in the GESSAR II Tables. However a note will be provided to reference the response to question 421.02. Assessments for all BTP's in Table 7-1 will be provided in that response.

Though incomplete at this time, a preliminary draft of the new tables is attached for reference.

Table 7.1-3

RPS AND ESF SYSTEMS REGULATORY REQUIREMENTS APPLICABILITY MATRIX[1]

10CFR50 Appendix A GDCS

Systems	55A[2]	2	4	13	19	20	21	22	23	24	25	29	34	35	38	41
REACTOR PROTECTION SYSTEM (RPS)	X	X	X	X	X	X	X	X	X	X	X	X				
ENGINEERED SAFETY FEATURED SYSTEMS																
ECCS	X	X	X	X	X	X	X	X	X	X		X	X	X		
CRVICS[3]	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X
MSPLCS	X	X	X	X	X	X	X	X	X	X		X				
Containment Spray Cooling RHR	X	X	X	X	X	X	X	X	X	X		X	X		X	
Suppression Pool Cooling RHR	X	X	X	X	X	X	X	X	X	X		X		X	X	
SPMU	X	X	X	X	X	X	X	X	X	X		X		X		
CCGCS	X	X	X	X	X	X	X	X	X	X		X				
SGTS	X	X	X	X	X	X	X	X	X	X		X				X
Shield Building Annulus Mixing	X	X	X	X	X	X	X	X	X	X		X				
Secondary Containment Isolation Control System	X	X	X	X	X	X	X	X	X	X						
Containment Isolation Valve Leakage Control	X															
Standby Power System	(See Table 8.1-1)															
Diesel Generator Auxiliaries	(															
Essential Service Water System	X															
ESF Area Cooling	X															
Pneumatic Supply System	X															
Main Control Room HVAC	X															
Control Building Chilled Water System	X															

Table 7.1-3 (CONT.)

RPS AND ESF SYSTEMS REGULATORY REQUIREMENTS APPLICABILITY MATRIX

NRC Reg. Guide[1] No: / Corresponding IEEE: / Systems	1.22	1.47	1.53 / 379 / 1972	1.62	1.75 / 384 / 1974	1.97	1.105	1.118 / 338 / 1975	— / 279 / 1971
REACTOR PROTECTION SYSTEM (RPS)	X	X	X	X	X	X	X	X	X
ENGINEERED SAFETY FEATURED SYSTEMS									
ECCS	X	X	X	X	X	X	X	X	X
CRVICS[3]	X	X	X	X	X	X	X	X	X
MSPLCS	X	X	X	X	X	X	X	X	X
Containment Spray Cooling RHR	X	X	X	X	X	X	X	X	X
Suppression Pool Cooling RHR	X	X	X	X	X	X	X	X	X
SPMU	X	X	X	X	X	X	X	X	X
CCGCS	X	X	X	X	X	X	X	X	X
SGTS	X	X	X	X	X	X	X	X	X
Shield Building Annulus Mixing	X	X	X	X	X	X	X	X	X
Secondary Containment Isolation Control System	X	X	X	X	X	X	X	X	X
Containment Isolation Valve Leakage Control	X	X	X	X	X	X	X	X	X
Standby Power System	(See Table 8.1-1)								
Diesel Generator Auxiliaries	(" " " " ?)								
Essential Service Water System	X	X	X	X	X	X	X	X	X
ESF Area Cooling	X	X	X	X	X	X	X	X	X
Pneumatic Supply System	X	X	X	X	X	X	X	X	X
Main Control Room HVAC	X	X	X	X	X	X	X	X	X
Control Building Chilled Water System	X	X	X	X	X	X	X	X	X

NOTES

1. Generic conformance statements appear in subsections 7.1.2.9, 7.1.2.1[illegible] and 7.1.2.11. Where required specific conformance statements are given in each system's analysis section. BTP: are assessed in response to ques[illegible] /42

TABLE 7.1-4
SAFE SHUTDOWN SYSTEMS REGULATORY REQUIREMENTS APPLICABILITY MATRIX[1]

Systems	55A[2]	10CFR 50 Appendix A GDCS 2	4	13	19
SAFE SHUTDOWN SYSTEMS					
Reactor Core Isolation Cooling System	X	X	X	X	X
Standby Liquid Control System	X	X	X	X	X
RHR/Reactor Shutdown Cooling	X	X	X		X
Remote Shutdown System		X	X	X	X
SAFETY-RELATED INSTRUMENTS	X	X	X	X	X

NRC Regulatory Guide No.	1.22	1.47	1.53	1.62	1.75	1.97	1.105	1.118	
Corresponding IEEE Standard			379 1972		384 1974			338 1975	279 1971
SAFE SHUTDOWN SYSTEMS									
Reactor Core Isolation Cooling System	X	X	X	X	X		X	X	X
Standby Liquid Control System	X	X	X	X	X		X	X	X
RHR/Reactor Shutdown Cooling	X	X	X	X	X		X	X	X
Remote Shutdown System					X				
SAFETY-RELATED INSTRUMENTS	X	X	X		X	X	X	X	X

NOTES

1. Generic conformance statements appear in subsections 7.1.2.9, 7.1.2.10 and 7.1.2.11. Where required, specific conformance statements are given in each systems analysis section. BTP's are assessed in response to question 421.
2. 10CFR 50.55a(h) requires use of IEEE-279 1971.

GESSAR II
238 NUCLEAR ISLAND

TABLE 7.1-5
OTHER SYSTEMS REQUIRED FOR SAFETY REGULATORY REQUIREMENTS APPLICABILITY MATRIX

Systems	55A[2]	2	4	13	19
OTHER SAFETY SYSTEMS					
Neutron Monitoring System	X	X	X	X	X
Process Radiation Monitoring System	X	X	X	X	X
~~Refueling Interlocks Function~~					
Leak Detection System (Safety related portion)	X	X	X	X	X
Rod Pattern Control System	X	X	X		X
~~HP/LP System Interlock Function~~					
Recirculation Pump Trip System	X	X	X	X	X
Fuel Pool Cooling and Cleanup System	X	X	X		X
Drywell/Containment Vacuum Relief System	X	X	X	X	X
Containment and Reactor/Auxiliary/Fuel Building Ventilation and Pressure Control System	X	X	X	X	X
Containment Atmosphere Monitoring System	X	X	X	X	X
Suppression Pool Temperature Monitoring	X	X	X	X	X

TABLE 7.1-5 (CONT.)

OTHER SYSTEMS REQUIRED FOR SAFETY REGULATORY REQUIREMENTS APPLICABILITY MATRIX[1]

NRC Regulatory Guide No.	1.22	1.47	1.53	1.62	1.75	1.105	1.118	
Corresponding IEEE Standard			379 1972		384 1974		338 1975	279 1971
SYSTEM								
OTHER SAFETY SYSTEMS								
Neutron Monitoring System →	X	X	X	X	X	X	X	X
Process Radiation Monitoring System →	X	X	X	X				
Leak Detection System (safety related portion) →								
Rod Pattern Control System →								
Recirculation Pump Trip System →								
Fuel Pool Cooling and Cleanup System →								
Drywell/Containment Vacuum Relief System →								
Containment and Reactor/Auxiliary Fuel Building →								
Ventilation and Pressure Control System →								
Containment Atmosphere Monitoring System →								
Suppression Pool Temperature Monitoring →								

421.03 QUESTION

FINAL DRAFT

In Section 7.1.2.10.18 of your FSAR, you provide information regarding the conformance of your proposed design with the guidance provided in Regulatory Guide 1.75. Discuss the details of your separation criteria for protection channel circuits, protection logic circuits and nonsafety-related circuits using one-line drawings, schematics or other drawings as appropriate, in light of the guidance provided in this regulatory guide.

421.03 RESPONSE

Details of the separation methods and techniques are provided in GESSAR II, Chapter 8, as required by Regulatory Guide 1.70, revision 3 (See Subsections 8.3.1.1.5.1, 8.3.1.3 and 8.3.1.4). Also, in conjunction with PGCC separation, see the NRC approved topical report = NEDO-10466-A, as referenced in GESSAR II, subsections 7.1.2.10.18.E, 7.7.1.9.A, 7.7.2.9.B, 8.3.1.4.1.2(6), and 8.3.1.4.2.3.2(8).

Physical separation of redundant Class 1E circuits and devices/components is provided within each Class 1E control panel so that no single credible event can prevent the proper functioning of any Class 1E system.

Separation is accomplished by mounting the redundant Class 1E equipment on physically separated control panels. When operational design dicates the redundant Class 1E equipment be in close proximity, separation is achieved by a fire-retardant barrier or an air space. Wiring is supported in a manner such that the designed air space would be maintained throughout the entire life of the panel. Such Class 1E control panels are located and protected in a manner such that a single credible event is limited to an internally generated fire.

Examples of acceptable separation barriers are:

1. Two sheets of fire-retardant material separated by an air space or thermal insulating material
2. A single barrier with a one inch maintained air space or thermal insulating material between the components or devices and the barrier
3. Metallic conduit

The minimum separation distance between redundant Class 1E equipment and circuits internal to the control panels is established by analysis of proposed installation. This analysis based on tests performed to determine the flame-retardant characteristics of the wiring materials, equipment, and other materials internal to the control board. Where the control board materials are flame retardant and analysis is not performed, the minimum separation distance is 6 inches. Wherever the separation distances are not maintained, barriers are installed between redundant Class 1E wiring.

Under certain circumstances it is possible that various divisions of Class 1E circuits must enter the same device. This may occur with switches, a typical example being the reactor mode switch where four divisions come together. When common devices are used, the divisional wiring is separated by means of separation barriers between divisional wiring. Non-Class 1E circuits, within the Class 1E enclosures, which are in close proximity of the Class 1E wiring or devices, are treated as an associated circuits. Associated circuits conform to the same requirements as applicable to the Class 1E circuits such as cable derating, environmental qualification, flame retardance, splicing restrictions and raceway fill. Interface between divisional circuits, and divisional and non-divisional circuits, where required, is accomplished by means of optical isolators.

Cable routing is fixed in accordance with the elementary diagrams in Appendix 7A of GESSAR II. All BOP interfaces are defined on the appropriate elementary diagram.

FINAL DRAFT

421.05 QUESTION

The information you provide in your FSAR discussing your conformance with Regulatory Guide 1.118 and IEEE Std 338 is insufficient. Accordingly, provide the following additional information:

a. Discuss your proposed testing of response time, including the use of sensors, in relation to the guidance provided in Regulatory Guide 1.118 and Section 6.3.4 of IEEE Std. 338. Include in your discussion the effects of thermo wells, restrictions, orifices or other instruments in relation to the overall response.

b. Provide examples and descriptions of typical response time tests for the reactor protection system (RPS) and the ESF systems.

421.05 RESPONSE (for both a and b)

Regulatory Guide 1.118 and Section 6.3.4 of IEEE Std. 338 were reviewed and it was confirmed that the GESSAR II design meets the intent of these guidelines. In addition, we practice the state of the art in response time testing in that we comply with the EPRI study NP-267 dated October, 1976 (Sensor Response Time Verification).

All ESF instruments (sensors and electronics) can be periodically response-time tested with the exception of gamma and neutron detectors which require a step or ramp change in gamma or neutron flux levels when compared to a suitable, agreed-upon flux-measuring standard. Such flux levels are usually high and there is no proven methodology acceptable in industry for response-time testing such devices.

Response-time testing methods were reviewed at the GE/NRC meeting held in San Jose December 7-9, 1982 in conjunction with the response to this question. An example of the response time testing specification was presented for NRC review. The NRC requested ~~agreed to close this issue provided GE supply the~~ the following additional information:

The techniques with ramps, steps, etc., used for response testing RPS instruments are as outlined in the GE preoperational test specification for the RPS. Acceptable response criteria which the test must satisfy are based upon the response characteristics of the instruments that were assumed in the analysis.

FINAL DRAFT

421.07 QUESTION

In footnote 2 to Appendix A of 10 CFR Part 50, we require the assumption that: "single failures of passive components in electric systems should be assumed in designing against a single failure." Accordingly, discuss how you consider passive failures in all safety-related instrumentation and control systems in your proposed facility. Provide assurance that passive failures were included in a failure mode and effects analysis (FMEA) performed in response to the concerns identified in Question 421.08.

421.07 RESPONSE

Single failures of passive components in electric systems are assumed in the design of safety systems.

Passive electrical failures have been included in the GESSAR II FMEAs, in Appendix 15c.

FINAL DRAFT

421.08 QUESTION

We state our position in Section 7.3.2 of Regulatory Guide 1.70 that a FMEA should be a detailed analysis demonstrating that the appropriate regulatory requirements have been met. However, it is not clear in your FSAR if a FMEA addressing all credible failures has been performed. Verify that the appropriate FMEA's have been performed and address the following:

a. The FMEA is applicable to all ESF equipment.

421.08 a RESPONSE

In accordance with Regulatory Guide 1.70, Appendix 15C provides for FMEAs on selected systems of Chapters 6,7 and 9. There are a total of 32 FMEAs; either completed in detail in Appendix 15C or identified as requiring the Applicant to provide. In addition, several interfacing systems are identified as requiring FMEAs to be provided by the applicant. The combination of the FMEAs detailed in Appendix 15C and those identified to be provided by the applicant is applicable to all ESF equipment.

421.08 b QUESTION

b. The FMEA is applicable to all design changes and modifications to date.

421.08 b RESPONSE

As presented in Subsection 15C.0.6 the FMEA system-defining documents (electrical. instrumentation, and control drawings, and piping and instrumentation diagrams) utilized in conducting the FMEAs are annotated versions of the corresponding documents listed in Table 1.7-1. However, some of the Table 1.7-1 documents were revised (updated) after the FMEAs were completed. In each case the impact of the document update(s) was assessed and it was determined that the FMEA results were still valid.

421.08 c QUESTION

c. Provisions exist to assure that future design changes or modifications are included in the FMEA.

421.08 c RESPONSE

To assure that future design changes or modifications are included in the FMEAs, a statement will be added to Subsection 15C.0.6 that commits the Applicant to assess the impact of design changes or modifications, subsequent to the FDA. This will also be added to section 1.9 as an interface requirement. Drawings referenced in 1.7 and FMEA's shown in Chapter 15 will be consistant at the time of FDA.

FINAL DRAFT

421.09 QUESTION

Identify any nonsafety-related electrical equipment which is assumed in Chapter 15 of your FSAR to successfully operate to mitigate the consequences of anticipated operational occurrences and accidents. For each piece of equipment identified, provide the corresponding anticipated operational occurrence(s) and accidents for which that equipment is expected to function.

421.09 RESPONSE

No assumption is made in the FSAR Chapter 15 about the operation of nonsafety - related electrical equipment to mitigate the consequences of accidents.

In the transient analyses, no assumption is made about the failure of the nonsafety-related electrical equipment unless that is the effect that is specifically being analyzed.

Unless specifically addressed in the analysis, the operation of the reactor recirculation, the feedwater, the main steam bypass and the pressure control systems are not assumed to malfunction during a transient.

In the feedwater controller failure - maximum demand transient, a nonsafety, reactor vessel high water level trip is used to trip the feedwater system and main turbine stop valves. However, a separate safety-grade high water level trip is used to trip the reactor and subsequently ~~[illegible]~~ deprive the steam supply to the feedwater turbine. The logic for the feedwater & turbine trip is two-out-of-three with the power for the sensors from three independant sources.

Turbine building cable routing for class 1E interfaces is discussed in the response to question 421.38.

FINAL DRAFT

RECEIVED
NOV 23 1982
R.W. STRONG

NUCLEAR POWER SYSTEMS ENGINEERING DEPARTMENT MEMO

DATE: November 18, 1982

cc: A. Alam
BE Lawler
JS Post
MA Ross
R. Villa
PDK-82-65

TO: R. W. Strong

FROM: P. D. Knecht M/C 754 Ext. 56215

SUBJECT: GESSAR QUESTION 421.12

The following is a response to the subject question.

~~NRC Question~~ 421.12 Question

The BWR Owners Group submitted BWR Emergency Procedure Guidelines, Revision 2, by a letter dated June 1, 1982 (BWROG8219). Confirm the applicability of these guidelines to your proposed design. Review these guidelines to: (1) insure tht the instruments identified in the guidelines are identified in Section 7.5 of your FSAR under post-accident monitoring; and (2) determine the consistency of the requirements for instrumentation accuracy contained in these guidelines. For example, Item SP/L-1 contains the statement that: "Maintain the suppression pool water level between 12 ft. 6 in. and 12 ft. 2 in." Confirm that the accuracy of the suppression pool level instrument is consistent with this requirement. Caution #6 contains the statement that: "Whenever temperature near the instrument reference leg...". Identify the temperature instrumentation you have provided to implement this caution.

421.12 Response

The BWR Emergency Procedure Guidelines (EPG), Revision 2, have been reviewed to verify the adequacy of installed instrumentation in the 238 Nuclear Island design both in terms of availability of instrumentation and in terms of design criteria (including range, accuracy and precision).

The availability of the existing instrumentation and the adequacy of its range was reviewed as part of the Human Factors Engineering Systems analysis and is described in Chapter 18. Definition of EPG parameters which may require operator action was made as part of the Type A variable determination for assessment of 238 Nuclear Island instrumentation against the requirements of Regulatory Guide 1.97. This discussion may be found in Appendix 1D.

Table 7.5-1 lists safety related display instrumentation. The parameters associated with following BWR/6 Emergency Procedures based on EPG's are shown in Table 18.2-3a. A comparison of these two tables shows that, with two exceptions, all EPG related parameters are included in Table 7.5-1. The exceptions are Standby Liquid Control Tank Level and Condensate Storage Tank Level. Both of these instrument channels are specified as RG 1.97

421.12 RESPONSE (CONT.)

Category 3 (refer to Table 1DA-1) which does not require a safety related classification. Thus including these instruments on Table 7.5-1 would be inappropriate.

As Stated previously, the adequacy of instrumentation range to follow EPG's was reviewed as part of the Chapter 18 Human Factors review. No instances were identified where the Emergency Procedures based on EPG's specified action at points outside the range of the installed instrumentation. Requirements for instrumentation accuracy are specified by the system designers based on intended use during routine operation and following design basis events. ~~Verifiation of these accuracies or justification of deviations is part of the environmental qualification program and is the responsibility of the applicant to address.~~

Details on each of the post-accident monitoring instrumentation channels are to be found in Appendix 1D.

QUESTION

421.14 We convey information based on operating experience to licensees and applicants by issuing Office of Inspection and Enforcement (IE) Bulletins, Circulars and Information Notices (IEB, IEC and IEN). Although only the IE Bulletins require written responses, we expect licensees and applicants to take appropriate action based on the information provided in the Circulars and Information Notices which is applicable to their designs. Attachment 1 is a list of IE Bulletins, Circulars and Information Notices which are applicable to BWR's. Provide additional information on this matter which includes the following:

a. Your procedures for determining the applicability of the various IEB, IEC, and IEIN to your facility.

b. Your procedures or methods for factoring the applicable information or criteria into your proposed design.

c. Details of specific design modifications including their implementation, resulting from your response to Items a and b above.

421.14 RESPONSE (a, b, c)

(See GESSAR II, Section 1.9). ~~noted~~

APPLICANT WILL PROVIDE ~~EXCEPT AS BELOW~~
~~IN RESPONSE TO ITEM d~~ ~~e~~

421.14 d QUESTION

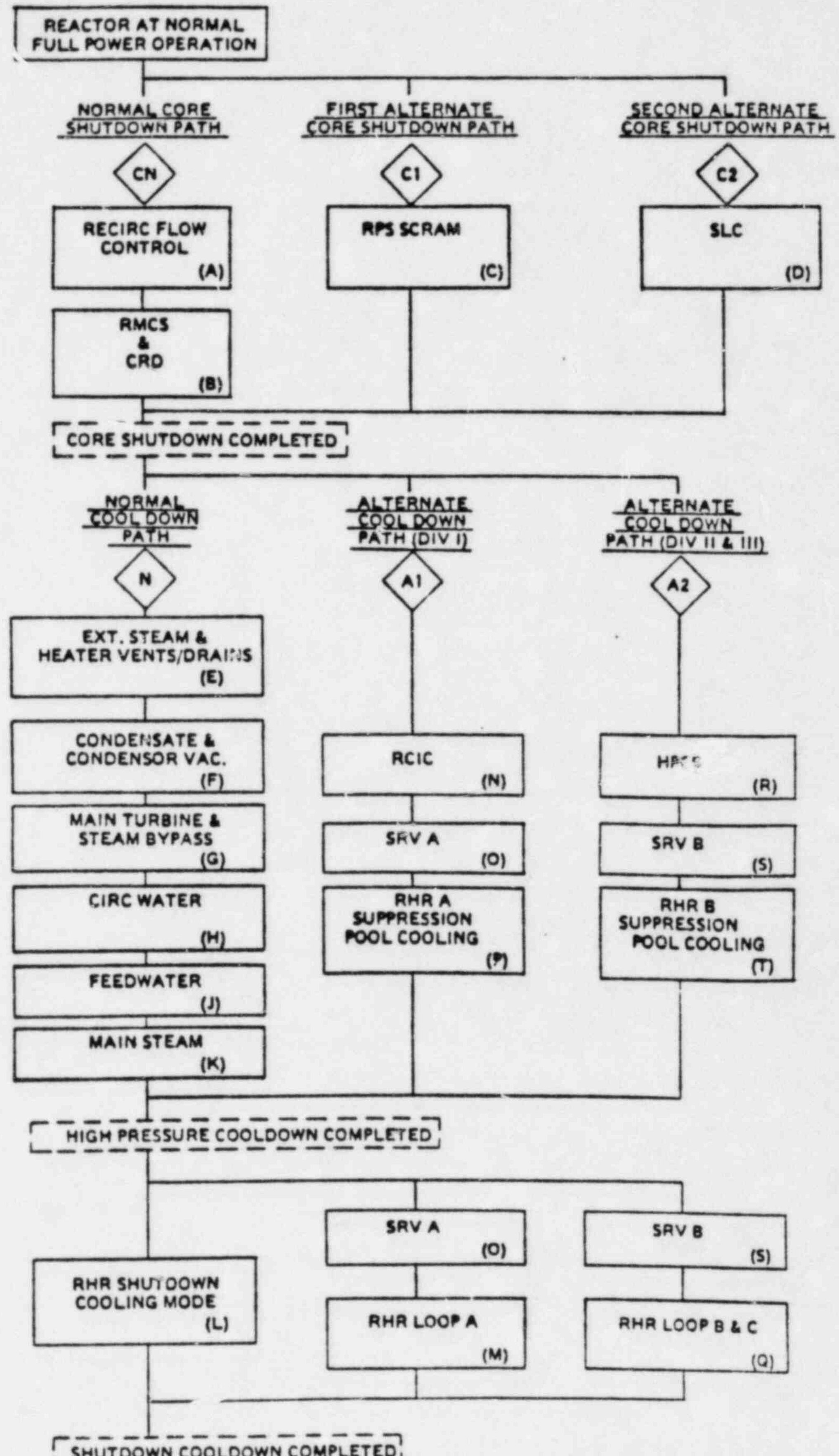
Provide a ~~A~~ detailed analysis of your response to IEB 79-27 and IEB 80-06, including the results.

In responding to this question, provide a cross-reference to Question 421.48. (80-06 only)

421.14 d RESPONSE

Safe shutdown is assured through alternate paths for any electrical single failure condition as shown on the attached figure. A detailed list of the busses providing power to the components used for the normal shutdown path and the alarms associated with the loss of these busses will be provided by the applicant as a response to IEB 79-27. The systems to be considered in this response are those shown on the attached figure.

Response to IEB 80-06 is contained in GESSAR Section ~~Appendix 1E~~. 1C.2-14 ~~1C.2.14~~



NOTE: Other variations of systems are possible using RHRC (LPCI) ADS and LPCS.

FIGURE 1. - COLD SHUTDOWN PATHS - -

MP 11/82

421.14 e QUESTION

Provide a

A detailed analysis, including the results, of your response to IEIN 79-22 which you performed to assure that consequential control system failures in the event of a high-energy line break do not result in consequences more severe than those shown in your accident analyses in Chapter 15 of your FSAR.

421.14 e RESPONSE

Applicant will provide. See GESSAR Section 1.9

Item e. [submit] a detailed analysis, including the results, of your response to IEB 79-22 which you performed to assure that consequential control system failures in the event of a high-energy line break do not result in consequences more severe than those shown in your accident analysis in Chapter 15 of your FSAR.

Answer: Analysis of other BWR plants has shown that, with the exception of the condenser vacuum and feedwater heater control components, system interactions caused by pipe breaks do not result in consequences more severe than the Chapter 15 event.

Appropriate guidance will be provided in the interface requirements of GESSER Section 1.9 so that feedwater heater controls and condenser vacuum components are not physically located such that single HELB conditions would cause their interaction.

FINAL DRAFT

421.17 QUESTION In Table 3.2-1 of your FSAR, your provided a "Q-List" of structures,
(7.1) systems and components whose safety functions require conformance to
(7.6) the applicable quality assurance requirements of Appendix B to 10 CFR Part 50. Verify that all safety-related instrumentation and controls (I&C) described in Section 7.1 thru 7.6 and other safety-related I&C equipment used in safety-related systems are subject to your QA program implementing the requirements of Appendix B. Indicate how we may determine which specific components shown in the electrical drawings referenced in Chapter 1.7 are classified as safety-related.

RESPONSE

All safety-related instrumentation and control (I&C) equipment described in Chapter 7 Sections 1 through 6 and other safety-related I&C equipment used in safety-related systems are subject to quality assurance programs which implement the requirements of 10CFR50 Appendix B, per Table ~~3-2-1~~ 3.2-1 and Chapter 17.

Electrical drawings identified in Chapter 1, Section 7 that contain safety-related components are so indicated on the drawing. Specific safety-related components are identified by safety division classifications or special symbols as shown in Chapter 1, Figures ~~1-7-1a~~ 1.7-1a thru 3b and 1.7-4.

FINAL DRAFT

421.19 QUESTION

Discuss your methodology and rationale for determining the setpoint values associated with the various leak detection systems (LDS) discussed in Section 7.6 of your FSAR. Discuss details of the manual bypass switch which will be used during testing of the leak detection system for the RCIC, including its conformance with the guidance provided in Regulatory Guide 1.47. Discuss the applicability of your response on this specific leak detection system to other such systems described in Section 7.6 of your FSAR.

421.19 RESPONSE

Setpoint methodology and rationale for the Leak Detection System is the same as for the RPS and ESF systems addressed in question 421.18. At the GE/NRC meeting December 6-9, 1982, it was agreed between the NRC and GE to close this portion of the question on the basis of the response to 421.18.

The manual bypass switch (keylocked) is placed in the bypass position when testing the leak detection logic used for RCIC system isolation. Control room annunciation of the bypass condition is provided as "LOGIC A IN BYPASS" and "LOGIC B IN BYPASS". This bypass and annunciation applies to a portion (leak detection) of the RCIC isolation logic and does not affect RCIC system initiation or operation. This arrangement meets the intent of Regulatory Guide 1.47.

QUESTION FINAL DRAFT,

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421.20 Discuss in Chapter 7 of your FSAR, the design criteria you have established to prevent trapping of air or noncondensable gases in the reactor pressure vessel instrument sensing lines. Discuss the applicability of these criteria to safety-related instrument sensing lines.

RESPONSE

The design criteria to prevent trapping of non-condensible gases in the RPV instrument sensing lines are as follows:

1) the instrument lines are required to be sloped downward from the vessel to the sensors. Whereever the lines or the local instrument racks can not be located such as to accommodate the 1/4" slope, high point vents are provided. A slope of approximately 1" per foot is specified to assure that gas bubbles will migrate back into the RPV. Lesser slopes requires engineering approval. The instrument line slope is required to be maintained through the drywell penetrations area except where structural interferences make it impractical to maintain the 1" per foot slope. In such cases a ~~1"~~ 1/4" per foot slope is the minimum allowed in this area.

2) Instrument lines for liquid service are required to be installed for self venting back into the process or be provided with high point vents to release trapped non condensibles after initial filling and thereafter if necessary. High point vents in the drywell have their vent valves located outside the drywell, inside the containment, for access during operatio
Globe valves are required to be mounted with their stems horizontal to reduce the amount of gas that can be trapped in the valve body to a practical minimum.
Orifices in impulse lines are concentric to obtain optimum accommodation of venting and avoidance of plugging by foreign particles. The slight heel of non condensibles resulting from this practice does not introduce any instrument error.

21.20
(cont'd)

3) Condensing chambers are connected to the RPV by 1" IPS minimum nozzles which are insulated to within 18 inches of the condensing chamber, thus making the condensing area and the amount of condensate draining back to the RPV more uniform and predictable. The ~~evaluation~~ elevation of the condensing chamber above the vessel instrument tap is limited to provide favorable conditions for the non condensibles to diffuse back into the steam volume so that the accumulation of non condensibles will not become high enough to impair the condensing chambers' function of maintaining the reference column level. Conservative analysis has determined that with a condensing chamber located 4 feet above the vessel nozzle the maximum non condensibles partial pressure is less than 300 psi which would not reduce the condensing rate unacceptably since it would not prevent sufficient steam from condensing to maintain the reference level.

The foregoing measures are consistent with the safety functions performed by the vessel sensing lines.

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421.21 QUESTION Provide impulse line routing criteria for safety-related pressure and flow instrumentation.

RESPONSE:

Safety-related pressure and flow instruments sensing lines are required to be routed such that the single failure criterion is complied with and also such as to avoid unacceptable errors from trapped non-condensibles. Instrument lines are assigned to mechanical separation divisions and the instruments served by them are assigned to electrical divisions of the same number. In some cases it is necessary to serve more than one division of instruments from a single flow element. These special cases are analyzed on a case by case basis for compliance with safety criteria and shown to be acceptable.

Redundant sets of instrument lines for flow sensing (e.g., Leak Detection sensors) are required to be separated so that an event for which these lines provide sensory information necessary to initiate the mitigating action cannot cause disabling of the sensing lines unless there is provided additional backup by means of diverse sensing or additional redundancy not affected by the same event. (An example of diverse backup is ambient temperature backup for excess flow sensors).

Redundant sensing lines are required to be physically separated except where convergence is unavoidable such as at the flow element itself as in the case of the main and shutdown steam flow sensors and the recirculation flow sensors. Each of these cases has been analysed to show that localized failure of redundant sensing lines does not impair the safety function as explained ~~on the next page:~~ in the following:

421.21
(cont'd)

1. Main Steam Flow Sensing for Main Steam Isolation Valve Closure:

   A main steam line break within the drywell/containment does not have to be sensed by the steam line flow sensors because the MSIV closure cannot isolate such a break. The high flow sensing is to protect against a break outside the drywell/containment. The sensing lines are widely separated outside the drywell inside the containment, they are also outside the containment's steam tunnel and on opposite sides of it so they are not vulnerable to damage from the event they protect against.

2. Shutdown steam flow sensing for RCIC/RHR steam isolation valve closure is similar to 1 above.

3. ~~2.~~ Recirculation Flow Sensors for Flow Reference Scram:

   The instrument lines for Recirculation flow converge at a single sensing ~~event~~ element ~~causing~~ allowing potential for pipe whip or jets that could break or crimp an instrument line. It has been determined that a break of sufficient magnitude to be considered damaging to these lines would be sufficient to increase the drywell pressure to the scram point very quickly and thus obviate any need for the flow reference scram to be operative.

Pressure sensing lines for the reactor vessel are also designed and routed to serve as reference pressure lines for the reactor pressure vessel level measurements. Therefore, they follow the same venting, draining, and azimuthal dispersion and limited vertical drop in the drywell as specified for the level reference lines. The routing criteria for these lines are as follows:

1. Redundant sets of instrument nozzles for reactor vessel level (pressure) are required to be widely dispersed around the periphery of the vessel. (Azimuths are 15°, 165°, 195° and 345°).
2. Instrument lines are required to maintain divisional separation as they run radially from the vessel nozzles through the drywell and thence to local instrument racks located in the corresponding four quadrants of the containment.

421.21 (cont'd)

3. Vertical elevation changes for the pairs of level sensing lines are required to be equal within + or - 1 foot inside the drywell where ambient temperatures can vary over a wide range. (from normal to LOCA environment) This practice results in automatic correction for drywell temperature effects and thus keeps errors resulting from varying water density predictably low.
4. Slopes of the instrument lines are required to be adequate for effective venting of non-condensibles and to permit effective flushing. This is obtained by a slope of approximately 1" per foot, with 1/4" per foot as an absolute minimum in limited areas.
5. The vertical elevation difference from the condensing chambers to the drywell penetration is required to be not more than four feet to limit the potential error which might result from instrument line boiloff under condition in which the RPV is cooler than the drywell (in the event that the drywell is maintained above 212°F for a time sufficient to permit gross boiloff of the reference column.)

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421.23 Question

Discuss the susceptability of safety-related equipment to electromagnetic interference (EMI).

Response

Refer to attachment on following pages.

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421.23 Response

The basic elements of the decision making logic of the NSPS are standard MIL grade CMOS logic elements, in dual in-line epoxy packages, mounted on multilayer printed circuit cards.

CMOS logic was chosen for the NSPS application because of its high noise immunity compared to other types of solid state devices. With the CMOS devices powered by 12 vdc, it takes an input greater than approximately 4 volts to switch the output on a low to high transition, and less than approximately 8 volts to switch on a high to low transition. Thus, noise spikes of considerable magnitude can be tolerated on the input lines without causing erroneous logic states. As a comparison, TTL logic which must be operated at +5V has a low to high minimum threshold of approximately .7 volt.

Numerous design techniques have been utilized to reduce the possibility of any significant electrical noise being coupled into the logic circuitry. All inputs and outputs that leave the NSPS cabinets are buffered and isolated, and internal wiring is routed to prevent "crosstalk" or radiated electromagnetic interference.

Specifically, prevention of electromagnetic conducted interference is accomplished in the following ways.

<u>Power Lines</u>: Conduction of EMI via power lines to the logic elements is prevented by the use of switching power supplies which are specified by the manufacturer to have a maximum noise spike of 62 mv. In addition, each logic card has single pole filters on the power input to remove any remaining high frequency noise.

<u>Input signal lines</u>: Inputs from other separation divisions, and from nondivisional sources are processed through optical isolators which are also filtered on the input side. Inputs from same-division sources such as the control room panels or field sources are processed through Digital Signal Conditioners (DSC's) which are filtered and optically coupled. Inputs to trip units are current loops and therefore not vulnerable to EMI.

Output signal lines: Outputs to actuated devices pass through load drivers which have pulse transformer coupling between input and output stages. Outputs to other logic elements in other divisions pass through optical isolators. Analog outputs to actuated devices are current loops which are not vulnerable to EMI.

Internal wiring: Interconnections between logic cards is on a backplane of wire wrapped terminals. The connections are made point to point so that groups of wires do not run in parallel for long distances. Power wiring is routed as far from signal wiring as possible. The high current wiring of the drives to the pilot valve solenoids is run in conduit, as is the wiring for utility services (lighting).

Card layout: All signal inputs at the card level are buffered by a 100 K ohm resistor. The use of ground planes over large areas of the boards also insures electrically quiet circuitry.

All standards of good practice were applied during the design and construction of the solid state safety system to prevent any problem with EMI.

EMI testing will be performed. according to the EMI susceptibility test guide (provided under separate cover letter dated _____, as Attachment:) or an equivalent method at the time of NSPS panel qualification testing.

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421.26 In Section 7.2 of your FSAR, you indicate that interconnections
(7.1) between redundant safety divisions are allowable through isolation
(7.7) devices. These isolating devices are used to maintain independence between safety-related circuits and between safety-related and nonsafety circuits. Provide the following additional information:

a. Identify the types of isolation devices used.

b. Provide the details of the testing which has been performed, including the results, to ensure that the isolation devices provide adequate protection against EMI, microphonics, short-circuit failures, voltage faults and voltage surges.

c. Discuss the applicability of the tests performed in Item (b) above.

RESPONSE:

a. Optical isolators are the principal devices used to provide physical and electrical isolation between safety-related circuits and between safety-related and non-safety circuits.

An optical isolator is an optical coupler with a high degree of electrical and physical separation. The working parts consist of a LED (light Emitting Diode) photo receiver (phototransistor or photo diode) pair separated by an optical barrier that will permit light to travel from the LED to the photo receiver, but will provide the necessary physical separation to satisfy USNRC Regulatory Guide 1.75. The LED's are mounted near the edge of an input circuit card that also contains the appropriate excitation and logic circuitry; the card is slid into one side of a specially designed double-sided printed circuit card file. The output circuit card containing photo receivers and appropriate output circuitry is located on the opposite side. A refractory material between the two sides contains holes filled with clear quartz rods which permit the light to pass while providing the necessary impervious physical barrier. The printed circuit card file is designed to be mounted in a control panel wall or other bulkhead between redundant divisions or between divisional and non-divisional bays or ducts to provide signal continuity while maintaining electrical and physical separation.

Several different input and output circuit boards will handle a variety of input and output signal levels and characteristics. Some can be intermixed for maximum flexibility ~~iwht~~ with a minimum number of different card types.

Other isolation devices, if any, are the responsibility of the ~~utility~~ applicant

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b. Optical isolators are tested to conform to the following requirements:

1. Provide electrical isolation between the input and output sides so that any abnormal circuit condition which occurs on one side will not affect the functional capability of circuitry on the other side. Electrical isolation between the input and output is sufficient so that a voltage of 5 kV applied to the input or output will not impair the function of devices on the other side of the barrier. The applicable requirements of USNRC Regulatory Guide 1.75 are also satisfied.

2. Provide physical isolation between the input side and the output side so that any environmental abnormality (such as fire) that occurs on one side and affects circuit operation will be inhibited in affecting the functional capability of circuitry on the other side. The center barrier between the input and output sides of the isolator card file is designed using special materials as are required to allow light to pass with negligible loss while providing a physical barrier capable of preventing fire or other severe environments from having easier access between control panel divisions than if there were no isolators.

3. Provide the means of coupling between the input and output sides to allow electrical stimuli on the input side to produce the desired electrical response at the output.

4. The isolators are capable of operation within specifications when exposed to the following environments:

   a. Temperature, humidity, pressure, and radiation according to the requirements specified in ~~FSAR 3.11.~~ GESSAR II, Section 3.11.

   b. Seismic vibration according to IEEE 344 and the requirements specified in GESSAR II, Section ~~FSAR~~ 3.10 using standard plant response spectrum multi-frequency excitation while mounted in control panels in which used.

c. The optical isolators have wide application and therefore meet codes and standards as separate equipment as well as part of system control panels in which they are to be located.

1. Institute of Electrical and Electronics Engineers (IEEE)

   a. IEEE 323-1974: IEEE Standard For Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

   b. IEEE 344-1975: IEEE Recommended Practices For Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations

   c. IEEE 384-1974: IEEE ~~Trial Use~~ Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits.

2. United States Nuclear Regulatory Commission (US NRC)

   a. USNRC Regulatory Guide 1.75 (Physical Independence of Electrical Systems) Rev. 1.

   b. USNRC Regulatory Guide 1.89 (Qualification of Class 1E Equipment for Nuclear Power Plants)

   c. USNRC Regulatory Guide 1.100 (Seismic Qualification of Electrical Equipment for Nuclear Power Plants)

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421.27 QUESTION

In our review of the Clinton application for an OL, we were concerned about the seismic and environmental qualification of the analog trip modules (ATM) and the optical isolators (OI). In response to that concern, the applicant stated that a qualification test of these devices is underway. State whether the ATM's and the OI's proposed in your design are identical to those used in the Clinton facility. If not, discuss how they will be qualified.

421.27 RESPONSE

The seismic and environmental qualifications of the optical isolators are fully discussed in the response to Question 421.26. The ATM's and optical isolators are identical to those used in the Clinton facility. They are qualified as part of the NSPS panel qualification program as was done for Clinton.

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421.28 QUESTION

Provide a discussion in Section 7.1.2 of your FSAR of your proposed separation criteria for instrumentation and control equipment and components for the safety-related systems identified in Section 7.1.1. It is our position that these separation criteria should assure that safety-related equipment is not located in a steam leakage zone insofar as is practicable. Alternatively, they should be designed for short term exposure to the high temperature and humidity associated with a steam leak. In this regard, provide the following additional information:

a. Identify the specific systems and the electrical equipment or components which are located in a steam zone and/or subjected to an abnormal temperature pressure, humidity or other environmental stress.

b. Discuss the safety-related function of the equipment and components.

c. Confirm that the equipment and components are included in your environmental qualification program.

421.28 RESPONSE

Safety related equipment in all cases will be environmentally qualified for its safety function.

Physical separation and independence criteria and conformance for safety-related systems are discussed in ~~FSAR~~ GESSAR II sections 7.1.2.10.18 and 7.1.2.11.~~10~~ 9.

Redundant divisions of electrical equipment and cabling are located in separate areas and/or are provided with spatial separation or barriers such that no single event can disable more than one of the redundant divisions or prevent safe shutdown.

Qualifications of electrical equipment and components, including equipment locations, environmental requirements, and method of qualification is contained in Section 3.11. (See Table 3.11-9).

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421.30 QUESTION

We discuss our requirements for anticipated transients without scram (ATWS) in Volume 4 of NUREG-0460. However, we note that no description of the instrumentation and controls to implement these requirements for your proposed design has been provided in Chapter 7 of your FSAR. Accordingly, discuss your design and its conformance with our requirements in NUREG-0460 for ATWS. Identify all non-safety related equipment relied upon in your design to satisfy our ATWS requirements.

421.30 RESPONSE

The GESSAR II design incorporates the safety-related Recirculation Pump Trip (RPT) as required by the NRC for the BWR. Its safety design basis is stated in Subsection 7.1.2.6.6 and the system technical descriptions and analysis are found in Subsections 7.6.1.6 and 7.6.2.6 respectively.

(additional information is provide on the following pages)

RECIRCULATION PUMP TRIPS

GESSAR II has two recirculation pump trip (RPT) design features for tripping the recirculation pumps in response to plant transients. One is known as the reactor protection system (RPS) end-of-cycle (EOC) RPT; the other is the anticipated-transient-without-scram (ATWS) RPT. These two RPT designs are described below.

EOC-RPT

The RPS EOC-RPT trips the power supply from the motors for the recirculation pumps when the reactor protection system logic detects a turbine stop valve closure or a turbine cont.ol valve fast closure event. The EOC-RPT function reduces the severity of the thermal transient that the fuel experiences for turbine/generator load rejection events. The EOC-RPT is a Seismic Category 1 and Class 1E system powered by the divisions of the 125V dc power supplied to the RPS Each of the four divisions of the EOC-RPT will take the RPS trip signal and supply it to the trip coil for one of the two in-series Class 1E circuit breakers on one of the two recirculation pump motor power supplies. That is, each recirculation pump has two in-series EOC-RPT breakers, each of which is fed by an RPS divisional trip signal. Although the initiation logics and inputs to the RPS trip are de-energized to trip, the final actuation signals from the RPS are energized to trip. RPS signals used to trip the recirculation high speed circuit breakers are also used to start the low frequency motor generator (LFMG) of the associated recirculation drive flow loop.

ATWS-RPT

The ATWS-RPT trips the power supply from the motors for the recirculation pumps when a high reactor dome pressure or low water level condition is detected. The ATWS-RPT function reduces the maximum transient reactor pressure for ATWS events. Each recirculation pump power supply circuit breaker used by the ATWS-RPT is tripped by a redundant one-out-of-two relay logic, i.e., one-out-of-four logic fed by two level and two pressure switches is used on each pump. The wiring from these eight switches to the control room is separated from any RPS cabling, and the ATWS-RPT logic is located in different panels from the RPS. 125V dc nonessential power is used to energize the trip coil on the power supply circuit breakers. The ATWS-RPT logic also provides a trip signal to the LFMG power supply breaker associated with each recirculation pump motor.

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15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

15.8.1 Requirements

The issue of a postulated failure to scram the reactor following an anticipated transient, i.e., an anticipated transient without scram (ATWS), has been under consideration by the NRC for some length of time. As a result of its assessment, the NRC has required the recirculation pump trip (RPT) feature for the BWR.

Plant requirements for ATWS in addition to the RPT have been proposed and are currently being reviewed by the NRC. It is not clear what, if any, additional ATWS requirements will result from this review. It should be noted that the NRC has determined that the current risk from an ATWS event is acceptably small, and therefore any additional plant modifications would only be required for long-term resolution of the ATWS issue and such modifications need not satisfy the requirements for a design basis event.

15.8.2 Plant Capabilities

The GESSAR II design utilizes diverse, highly redundant, and very reliable scram systems. [redacted] these systems are frequently tested and would insert the control rods even if multiple component failures should occur, thus making the possibility of an ATWS event extremely remote.

The plant has the ATWS-RPT feature which prevents reactor vessel overpressure and possible short-term fuel damage for the most limiting postulated ATWS event. Subsequent to an ATWS event [redacted], the long-term shutdown of the reactor can be accomplished by either manual insertion of the control rods or [redacted] boron injection into the vessel.

15.8.3 Additional Modifications

Should the NRC mandate additional ATWS modifications, the GESSAR II design will be appropriately modified.

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421.31 QUESTION

As a result of an event at the Brown's Ferry facility where a complete insertion of the control rods was not successful until after several attempts were made, we required design modifications related to hydraulic coupling and level monitoring to resolve this problem. You indicate in Paragraph 7.2.1.1.D.2(g) of your FSAR that four nonindicating level sensors provide scram discharge volume high water level inputs to the RPS. We conclude from this that your proposed system for monitoring the level of the scram discharge volume lacks diversity. Discuss what modifications are planned to meet the recommendations of the Office for Analysis and Evaluation of Operational Data (AEOD) presented in NUREG-0785.

421.31 RESPONSE

Scram discharge volume (SDV) level instrumentation is being changed to provide diversity in SDV high water level sensing. Necessary changes to GESSAR subsection 7.2.1.1.D.2(g) are shown on the attached sheets.

7.2.1.1.D.2.f System Description (Continued)

The arrangement of signals within the trip logic requires closing of at least one valve in two or more steamlines to cause a scram. In no case does closure of two valves in one steamline cause a scram due to valve closure. The wiring for position-sensing channels feeding the different trip channels is separated.

Main steamline isolation valve closure trip channel operating bypasses are described in Subsection 7.2.1.1.D.4.(c).

(g) Scram Discharge Volume

INSERT

~~Four nonindicating level sensors provide scram discharge volume high water level inputs to the reactor protection system. Each sensor provides an input to one instrument channel. The sensors are arranged so that no single event will prevent a reactor scram caused by scram discharge volume high water level.~~

With the predetermined scram setting, a scram is initiated when sufficient capacity still remains in the tank to accommodate a scram.

Scram discharge volume water level trip channel operating bypasses are described in Subsection 7.2.1.1.D.4(d).

The environmental conditions for the RPS are described in Section 3.11. The piping arrangement of the scram discharge volume level sensors is shown in Section 4.6.

(h) Drywell Pressure

Drywell pressure is monitored by four nonindicating pressure transmitters mounted on instrument racks

Insert - Page 7.2-12

Four non-indicating float-type level switches (one for each channel) provide scram discharge volume (SDV) high water level inputs to the four RPS channels. In addition, a level transmitter and trip unit for each channel provide redundant SDV high water level inputs to the 4 RPS channels. This arrangement provides diversity, as well as redundancy, to assure that no single failure could prevent a scram caused by SDV high water level.

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QUESTION:

421.34 In Section 7.2.1.1.D.6 of your FSAR, you indicate that pilot solenoids for the scram valves "are not part of the RPS" and that the RPS interfaces with the pilot solenoids. Discuss this interface using detailed schematics and drawings as appropriate, including a discussion of the backup scram valves, their classification and their interaction or interface with the RPS.

RESPONSE:

421.34 SECTION 7.2.1.1.D.6 DOES INDICATE THAT: "THE INDIVIDUAL CONTROL RODS AND THE SCRAM VALVES, PILOT SOLENOIDS AND THEIR CONTROLS ARE NOT PART OF THE RPS."

THE ACTUATING DEVICES OR LOAD DRIVERS WHICH ARE USED TO PROVIDE OR INTERUPT THE POWER TO THE SOLENOIDS OF THE SCRAM PILOT VALVES ARE PART OF THE RPS. HOWEVER, THE ACTUATED DEVICES OR THE SCRAM PILOT VALVES THEMSELVES, AND THEIR SOLENOIDS, ARE NOT PART OF THE RPS, BUT RATHER ARE COMPONENTS OF THE CONTROL ROD DRIVE (CRD) SYSTEM. EACH HYDRAULIC CONTROL UNIT (HCU) OF THE CRD SYSTEM HAS A SINGLE SCRAM PILOT VALVE WHICH CONTROLS THE AIR SUPPLY TO THE TWO SCRAM VALVES OF THAT HCU. THE SCRAM PILOT VALVE WILL BLOCK SCRAM AIR HEADER AIR AND EXHAUST THE AIR TO THE TWO SCRAM VALVES WHEN BOTH OF THE TWO SOLENOIDS OF THE SCRAM PILOT VALVE ARE DEENERGIZED.

FIGURES 7A-2-1y, 7A-2-1z AND 7A-2-1w SHOW THE INTERFACES BETWEEN THE LOAD DRIVERS OF THE RPS AND THE SOLENOIDS OF THE SCRAM PILOT VALVES OF THE CRD SYSTEM.

THE HCU'S AND CONTROL RODS ARE ARRANGED INTO FOUR SCRAM GROUPS. IN EACH SCRAM GROUP, THE POWER TO ALL "A" SOLENOIDS OF THE SCRAM PILOT VALVES IN THAT SCRAM GROUP ARE CONTROLLED BY THE ACTIONS OF TWO RPS LOAD DRIVERS. SIMILARLY, THE POWER TO ALL "B" SOLENOIDS OF THE SCRAM PILOT VALVES IN THAT SCRAM GROUP ARE CONTROLLED BY TWO DIFFERENT LOAD DRIVERS. A TOTAL OF SIXTEEN LOAD DRIVERS ARE USED TO CONTROL THE POWER TO THE "A" AND

(421.34 RESPONSE CONT.)

"B" SOLENOIDS OF THE SCRAM PILOT VALVES OF THE FOUR SCRAM GROUPS.

EACH OF THE FOUR LOAD DRIVERS ASSOCIATED WITH A SINGLE SCRAM GROUP RECEIVES ITS TRIP SIGNAL FROM A DIFFERENT RPS DIVISION. THE INTERFACE BETWEEN THE RPS LOAD DRIVERS AND THE SOLENOIDS OF THE SCRAM PILOT VALVES ARE SUCH THAT TRIP CONDITIONS IN THE TRIP LOGICS OF EITHER OF TWO RPS DIVISIONS WILL CAUSE THE INTERUPTION OF POWER TO ALL OF THE "A" SOLENOIDS OF A SCRAM GROUP, AND TRIP CONDITIONS IN THE TRIP LOGICS OF EITHER OF THE TWO REMAINING RPS DIVISIONS WILL CAUSE THE INTERUPTION OF POWER TO ALL OF THE "B" SOLENOIDS OF THE SAME SCRAM GROUP. THE INTERFACE BETWEEN THE FOUR RPS DIVISIONS AND THE SCRAM PILOT VALVE SOLENOIDS OF EACH SCRAM GROUP IS THUS OF A ONE-OUT-OF-TWO-TAKEN-TWICE DESIGN CONFIGURATION. All parts of the HCU's are testable at power so long as procedures are properly followed (ie, only one division is tested at a time).

THE ATTACHED SKETCH A SHOWS THE INTERFACES OF THE RPS LOAD DRIVERS WITH THE VALVES OF THE (RD) SYSTEM PRIOR TO IMPLEMENTATION OF ECA's 810108-1, REV 0 AND 800911-1, REV. 0. ATTACHED SKETCH B SHOWS THE INTERFACES THAT WILL EXIST AFTER THE ECA's ARE IMPLEMENTED FOR GESSAR II. ALL FURTHER DISCUSSION OF THE INTERFACES WILL ASSUME CONFIGURATION AS PER SKETCH B.

FOUR LOGIC LEVEL SIGNALS USED TO TRIP THE RPS LOAD DRIVERS ORIGINATE FROM EACH OF THE FOUR RPS DIVISIONS. FROM DIVISION 1, THESE FOUR SIGNALS CONTROL FOUR OF THE SIXTEEN LOAD DRIVERS ASSOCIATED WITH THE SCRAM PILOT VALVE SOLENOIDS. THE FOUR SIGNALS ARE IDENTICAL IN SIGNAL CONTENT, AND WHENEVER THE SIGNALS ARE AT LOGIC LEVEL ~~ZERO~~ (0), ONE OF THE SIGNALS WILL INTERUPT POWER TO ALL "A" SOLENOIDS IN SCRAM GROUP 1; THE SECOND WILL INTERUPT POWER TO ALL "B" SOLENOIDS IN SCRAM GROUP 2; THE THIRD WILL INTERUPT POWER TO

(421.34 RESPONSE CONT.)

ALL "B" SOLENOIDS IN SCRAM GROUP 3; AND, THE FOURTH WILL INTERUPT POWER TO ALL "A" SOLENOIDS IN SCRAM GROUP 4. SIMILARLY, THE FOUR LOGIC LEVEL SIGNALS ORIGINATING FROM RPS DIVISION 2, WHEN IN THE LOGIC LEVEL ZERO (0) STATE, WILL INTERUPT THE POWER TO ALL "A" SOLENOIDS IN SCRAM GROUPS 2 AND 1 AND TO ALL "B" SOLENOIDS IN SCRAM GROUPS 3 AND 4. THE FOUR SIGNALS ORIGINATING FROM RPS DIVISION 3, WHEN IN THE LOGIC LEVEL ZERO (0) STATE, WILL INTERUPT THE POWER TO ALL "A" SOLENOIDS IN SCRAM GROUPS 3 AND 2 AND TO ALL "B" SOLENOIDS IN SCRAM GROUPS 4 AND 1. THE REMAINING FOUR SIGNALS ORIGINATING FROM RPS DIVISION 4, WHEN IN THE LOGIC LEVEL ZERO (0) STATE, WILL INTERUPT POWER TO ALL "A" SOLENOIDS IN SCRAM GROUPS 4 AND 3 AND TO ALL "B" SOLENOIDS IN SCRAM GROUPS 1 AND 2.

NOTE THAT THE TRIP SIGNALS TO THE LOAD DRIVERS CONTROLLING THE POWER TO THE SCRAM PILOT VALVE SOLENOIDS DO NOT JUST REPRESENT TRIPS OF INDIVIDUAL RPS DIVISION TRIP LOGICS. THE FOUR SIGNALS ORIGINATING FROM RPS DIVISION 1 WILL CHANGE FROM A LOGIC LEVEL ONE (1), UNTRIPPED STATE TO A LOGIC LEVEL ZERO (0), TRIPPED STATE IF THE DIVISION 1 TRIP LOGIC HAS BEEN TRIPPED, OR IF BOTH THE DIVISION 2 AND DIVISION 3 TRIP LOGICS ARE SIMULTANEOUSLY IN A TRIPPED CONDITION, OR IF BOTH THE DIVISION 2 AND DIVISION 4 TRIP LOGICS ARE SIMULTANEOUSLY IN A TRIPPED CONDITION. THE FINAL TRIP SIGNALS FROM THE THREE OTHER RPS DIVISIONS ARE SIMILARLY CONFIGURED BUT ARE DIFFERENT FOR EACH RPS DIVISION. THE FOUR SIGNALS FROM RPS DIVISION 2 WILL CHANGE TO THE LOGIC LEVEL ZERO (0) STATE IF DIVISION 2 ALONE HAS BEEN TRIPPED, OR IF DIVISIONS 3 AND 4 HAVE BOTH BEEN TRIPPED, OR IF DIVISIONS 3 AND 1 HAVE BOTH BEEN TRIPPED. THE FOUR SIGNALS FROM RPS DIVISION 3 WILL CHANGE TO THE LOGIC LEVEL ZERO (0) STATE IF DIVISION 3 ALONE HAS BEEN TRIPPED, OR IF DIVISIONS 4 AND 1 HAVE BOTH BEEN

(421.34 RESPONSE CONT.)

TRIPPED, OR IF DIVISIONS 4 AND 2 HAVE BOTH BEEN TRIPPED. THE FOUR SIGNALS FROM RPS DIVISION 4 WILL CHANGE TO THE LOGIC LEVEL ZERO (0) STATE IF DIVISION 4 ALONE HAS BEEN TRIPPED, OR IF DIVISIONS 1 AND 2 HAVE BOTH BEEN TRIPPED, OR IF DIVISIONS 1 AND 3 HAVE BOTH BEEN TRIPPED.

THE UNIQUE SIGNAL CONTENT OF THE FINAL TRIP LOGIC SIGNALS FROM EACH RPS DIVISION, COMBINED WITH THE FINAL ONE-OUT-OF-TWO-TAKEN-TWICE CONFIGURATION INTERFACE OF THE FOUR DIVISIONS WITH EACH OF THE FOUR SCRAM GROUPS RESULTS IN A DESIGN WHEREBY ALL SCRAM PILOT VALVES OF ALL SCRAM GROUPS WILL BE ACTUATED IF ANY TWO OF THE FOUR RPS DIVISION LOGICS ARE TRIPPED. NO SCRAM RESULTS IF ONLY A SINGLE RPS LOGIC IS TRIPPED. THE RESULTING COMBINATION TRIP LOGIC BETWEEN THE INDIVIDUAL RPS DIVISION TRIPS AND THE ACTUATION OF THE SCRAM PILOT VALVES COULD BEST BE DESCRIBED AS ONE-OUT-OF-TWO TWICE COMBINED WITH PARTIAL TWO-OUT-OF FOUR TWICE COMBINATION LOGIC. THE DESIGN ALLOWS THE FUNCTIONAL TESTING OF THE FINAL ACTUATED ELEMENTS DURING PLANT OPERATION WITHOUT SUBJECTING THE PLANT TO AN UNWANTED SCRAM AND YET HAS A PROBABILITY OF SUCCESSFUL SCRAM THAT IS GREATER THAN EITHER A ONE-OUT-OF-TWO-TWICE DESIGN OR A TWO-OUT-OF-FOUR DESIGN.

THE INTERFACE OF THE RPS WITH THE BACKUP SCRAM VALVES IS ALSO SHOWN ON SKETCH B. TWO DC LOAD DRIVERS OF THE RPS ARE USED TO PROVIDE POWER TO THE SOLENOIDS OF THE TWO BACKUP SCRAM VALVES IN THE EVENT OF A FULL REACTOR SCRAM. THE SOLENOID OF BACKUP SCRAM VALVE C11-F110A WILL BE ENERGIZED IF A DIVISION TRIP EXISTS IN RPS

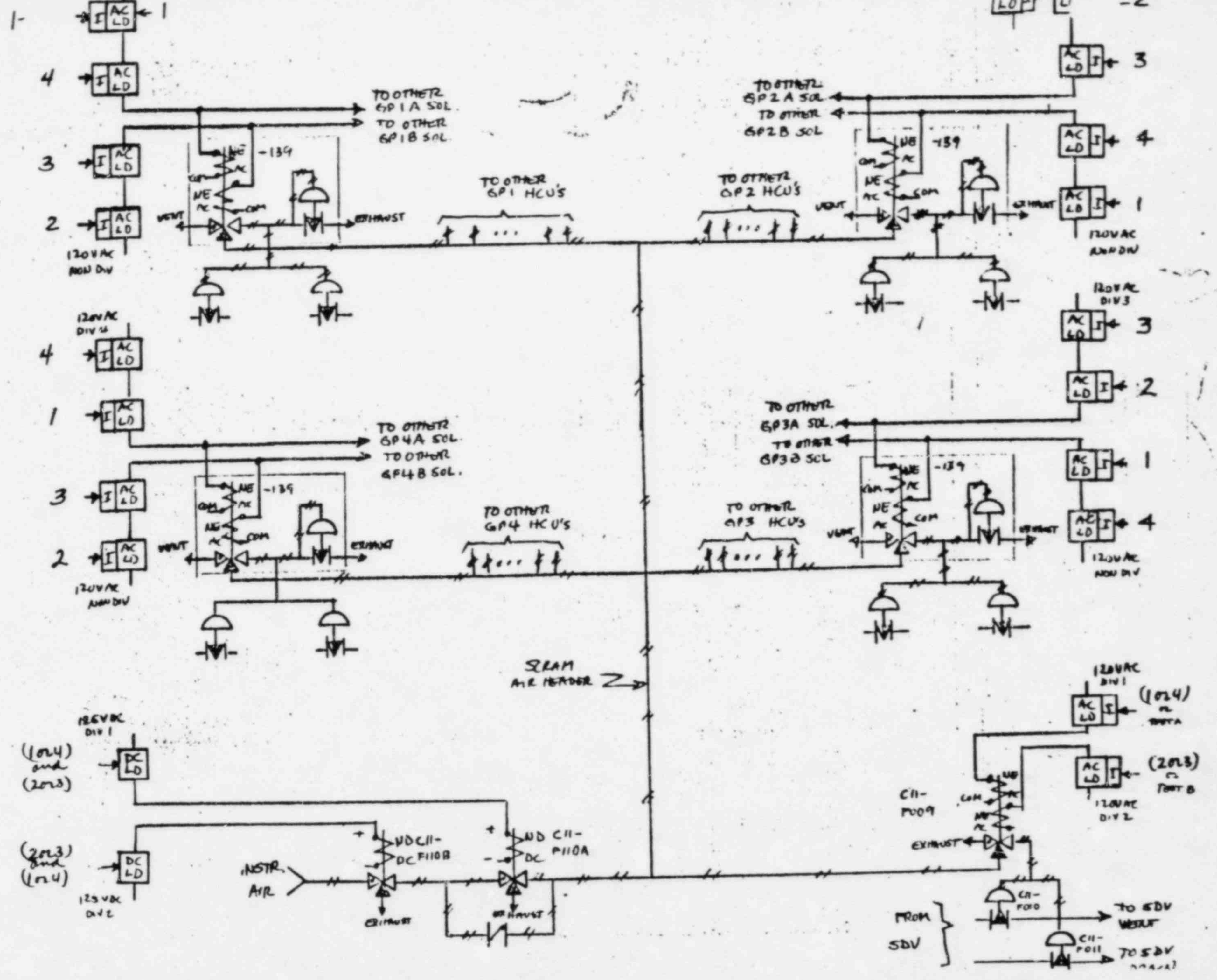
(421.34 RESPONSE CONT.)

DIVISION 1 OR DIVISION 2 AND AT THE SAME TIME A DIVISION TRIP EXISTS IN RPS DIVISION 3 OR DIVISION 4. THE SOLENOID OF BACKUP SCRAM VALVE C11-F110B WILL BE ENERGIZED IF A DIVISION TRIP EXISTS IN RPS DIVISION 2 OR DIVISION 3 AND AT THE SAME TIME A DIVISION TRIP EXISTS IN RPS DIVISION 4 OR DIVISION 1. ENERGIZATION OF THE SOLENOIDS OF EITHER OF THE TWO BACKUP SCRAM VALVES WILL BLOCK THE INSTRUMENT AIR SUPPLY TO THE SCRAM AIR HEADER AND AT THE SAME TIME EXHAUST THE AIR IN THE SCRAM AIR HEADER. THE LOAD DRIVERS ARE PART OF THE RPS WHEREAS THE BACKUP SCRAM VALVES AND THEIR SOLENOIDS ARE COMPONENTS OF THE CRD SYSTEM.

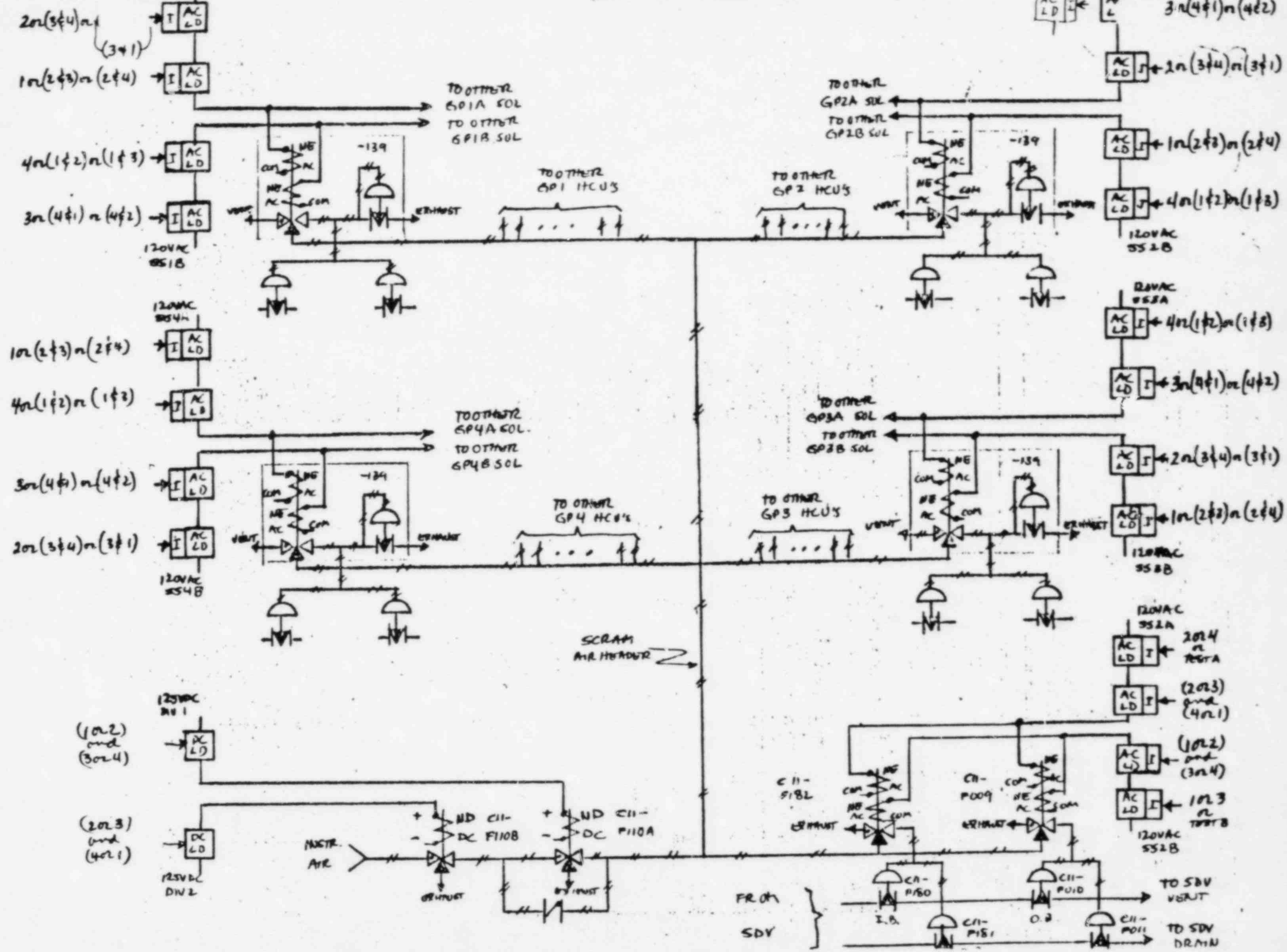
THE BACKUP SCRAM VALVES AND THEIR SOLENOIDS ARE CLASSIFIED AS NOT IMPORTANT TO SAFETY AS NO CREDIT IS TAKEN IN ANY SAFETY ANALYSIS FOR THE OPERATION OF THE BACKUP SCRAM VALVES. THEIR ONLY FUNCTION IS TO EXHAUST THE AIR IN THE SCRAM AIR HEADER AFTER A REACTOR SCRAM SUCH THAT, IF ANY SINGLE SCRAM PILOT VALVE HAS FAILED TO OPERATE AS IT SHOULD UPON THE DEENERGIZATION OF ITS TWO SOLENOIDS, THE AIR SUPPLIED TO THE TWO SCRAM VALVES ASSOCIATED WITH THAT PILOT VALVE WILL STILL EVENTUALLY BE EXHAUSTED AND THE ASSOCIATED CONTROL ROD WILL BE INSERTED INTO THE REACTOR CORE. THERE IS NO SAFETY CONSEQUENCE ASSOCIATED WITH THE FAILURE OF A SINGLE WITHDRAWN CONTROL ROD TO INSERT VIA SCRAM LOGIC. SHOULD THE SCRAM PILOT VALVE FAIL TO OPERATE AND BOTH BACKUP SCRAM VALVES ALSO FAIL TO OPERATE, THE CONTROL ROD CAN STILL BE MANUALLY INSERTED BY THE OPERATOR VIA THE NORMAL CRD SYSTEM CONTROLS.

ATTACHED: 421.34 SKETCH A
421.34 SKETCH B

421.2II - SKETCH A



42134 - SKETCH B



FINAL DRAFT

421.35 QUESTION
(7.7) In Figure 5.1-3C of your FSAR, you indicate that the RPV pressure and water level instruments use the same instrument lines. Identify all other instances where instrument sensors or transmitters supplying information to more than one protection channel, are located in a common instrument line or connected to a common instrument tap. Verify that a single failure in a common instrument line or tap (such as a break or a blockage) cannot defeat the required protection system redundancy. Identify where instrument sensors or transmitters supplying information to both a protection channel and one or more control channels are located in a common instrument line or connected to a common instrument tap. Verify that a single failure in a common instrument line or tap cannot cause an initiating event and also defeat protection channels or functions. Provide a list of the shared equipment identified in response to these questions. Include the turbine stop valve/control valves as well as the RPV instrumentation in your analysis.

RESPONSE:

The figure cited (5.1-3C) shows a part of the nuclear boiler system and its safety related level and pressure instrumentation and illustrates schematically the grouping of instruments on the four sets of instrument taps at the four different azumuths. The vessel level and pressure instruments of each single division are connected to the same tap. This is consistent with the single failure criterion which assumes failure of an entire division of equipment as a Single Failure.

There are instances where a single instrument line or tap serves instruments in more than one division of the protection system. These cases are:

1. Main Steam Line high flow sensors for isolation of large main steam line breaks outside containment.

2. Reactor Recirculation flow sensors for flow reference scram inputs.

3. RPV level sensors for RPS division 4 and HPCS division 3 level inputs.

4. The Main Turbine First Stage pressure taps providing power level information to the RPS to permit scram on turbine trip above a specific power level.

Each of these four cases has been analyzed for single line break or blockage consequences and found to be acceptable as follows:

1. ~~The~~ Each main steam flow sensing element has two sets of ΔP taps/lines which run in divergent directions to two local instrument racks located outside the drywell on opposite sides inside the containment outside the containment's steam tunnel. Each local rack has two steam flow transmitters, which are assigned to different electrical divisions located on ~~different~~ separated sections of the same rack. A postulated instrument line/tap failure could cause two high flow signals to be disabled. Such a failure could not be the result of a main steam line break outside the containment because of the location of the instrument taps, lines and racks. Therefore, the failure can be considered a random failure. The remaining two channels of flow information emanating from the second instrument rack provide~~s~~ the signals to the 2/4 logic which will initiate isolation as required by a large main steam line break. These main steam flow taps serve no control functions.

2. ~~The~~ Each Reactor Recirculation Line flow element is an elbow tap which has two sets of instrument lines which run in divergent directions to two local instrument racks outside the drywell and in two different quadrants of the containment. Each set of instrument lines serves two ΔP transmitters which are assigned to different electrical divisions ~~and so are~~ located on separate sections of the sub divided local instrument rack. A postulated instrument line/tap failure could affect two of the four channels of flow information to the flow reference scram circuit but the remaining two operative channels would

provide the necessary 2/4 inputs to obtain a scram on the 2/4 logic IF low recirc flow were to occur. Instrument line damage in the vicinity of the elbow taps as a result of a LOCA induced pipe whip or jet could not result from a leak so small that it ~~could~~ would not quickly raise the drywell pressure to the scram set point. Therefore, failure of these lines as consequences of a LOCA is not a safety concern.

These recirculation flow taps serve a rod block function but do not cause any active control action that would initiate a transient.

3. The division 4 RPV level sensors includes level transmitters for the HPCS system which is a Division 3 system. Therefore these transmitters have 24 VDC circuits from division ~~3~~ 4 ~~coming~~ going into a Division 4 CABINET and on into ~~THE~~ DIVISION 3. The Division 3 ~~rack. These~~ circuits are ~~required to be~~ ~~separated~~ ISOLATED from the Division 4 circuits by ~~separate enclosures~~ CLASS IE ISOLATORS IN THE DIVISION 4 CABINET and routed in a Division 3 raceway or conduit, ~~even though the 4-20 ma signal poses no threat to the Division 4 circuits in the vicinity. Conversely, the Division 4 circuits are in their own raceway so they cannot pose a threat~~ to the Division 3 ~~circuits~~ CABINET. It is also noted that the HPCS has a separate set of sensors located on the other side of the vessel/ containment. The division 4 RPV level taps do not serve any control function.

4. The main turbine first stage pressure connections are not always separable into four separate taps because of physical constraints. Where only two taps are available each tap serves two sensors, one in each of two divisions. The blocking or breaking of an instrument line can thus disable two sensors. However in the 2/4 logic the two remaining operable ~~taps~~ sensors would give the required two inputs to permit the turbine stop-valve-closure scram on pressure above their set point.

The first stage pressure taps provide input to transmitters used in the rod block circuits. Each tap serves one of the rod block circuits so failure of a tap could disable one of two rod block circuits leaving the other active. This failure would not initiate any transient that could cause a need for the first stage pressure safety signals.

5. The Turbine Control Valve fast closure signals are not shared since they are taken from four separate taps. The only other instrument taps that serve both safety and control functions are the RPV level taps on divisions 1 and 2. The transient analysis covering a failure of one of these taps as an initiating event is covered in detail elsewhere but in summary a single failure that could initiate a RPV level transient that exceeded normal operating limits would cause either a high or low level scram which would not be disabled by an additional single failure, since the two remaining operable sensors would give the required 2/4 inputs to initiate scram.

[illegible] 11-15-82

QUESTION

FINAL DRAFT

421.36 (7.1) (7.7) Provide an evaluation of the effects of high temperatures on the reference legs of the water level measuring instruments resulting from exposure to high-energy line breaks.

RESPONSE:

Exposure of the RPV level reference columns to the high ambient temperature associated with a High Energy line break will heat the column. ~~at a rate corresponding to a thermal time constant of approximately 10 minutes~~. Therefore, the water in the reference column will gradually approach the temperature of the drywell environment. ~~in about 30 minutes.~~ This temperature will depend upon the nature of the break. A large break will give a relatively low temperature (approx. 280°F) whereas a Small break will superheat the drywell to approximately ~~340~~ 330°F. This is the condition which is of greatest concern because in the small break case, it is expected that the vessel will be depressurized after a short time and vessel water temperature will then be lower than the reference column water so that the reference column will boil. The boiling will be rapid if the vessel depressurization is rapid and it has been determined that approximately 20% of the reference column exposed to the high drywell temperature could flash quickly. This is based on a vertical column and will be less for a sloping reference line such as exists in the drywell because the volume of water per unit of vertical drop will depend on the slope which will typically be ~~¼~~ ¼ to 1 inch per foot compared with 12 inches per foot for a vertical pipe. After the initial rapid flashing there would be a gradual boiling over a period of hours if the drywell ambient was maintained above the vessel temperature. Thus the reference column could be gradually depleted. The effect of this depletion will be limited by the vertical height differential between the reference level and the instrument line penetration through the drywell wall. This distance is limited by design specification ~~recommendation~~ to 4 feet so as to limit the boil off potential error to not more than that equivalent to four feet of reference column. The error would be in the direction to make the indicated level higher than the actual level. This error will not exist prior to the depressurization of the vessel because there will be no boiloff and the reference and variable legs will heat up at very nearly the same rate and thus compensate each other.

It has been determined that an error of the magnitude cited will not result in incorrect operator action or unsafe reactor conditions during recovery from LOCA.

The configuration of the instrument piping is the responsibility of the ~~utility~~ applicant, and the location of the containment penetrations must be such as to accomodate the limited drop specified.

FINAL DRAFT

421.37 QUESTION

Verify that there is sufficient redundancy in the water level instrumentation to prevent a sensing line failure (i.e., break, blockage or leak), concurrent with a random single electrical failure, defeating an automatic reactor protection or an ESF actuation.

421.37 RESPONSE

In BWR/6 solid-state plants, the RPS logic is any 2-out-of-4 channels to scram. Therefore, if one RPS channel reads erroneously high due to the instrument line failure and any additional RPS channel is assumed to fail-short, there are still 2 remaining channels left to accomplish normal scram. Assuming an instrument line break in Division 1 (worst-case in the Grand Gulf analysis), it is possible to fail either RCIC or HPCS by postulating the additional failure in ECCS busses 2 or 3 respectively. However, both systems cannot fail due to a single electrical failure and there will always be a normal Level 3 scram prior to automatic initiation of either (or both) high-pressure system.

The worst-case scenerio is postulated to be the reference line break coupled with HPCS failure. Normally, the operator would switch feedwater control from the bad instrument line to the good one as soon as the level mis-match is detected by the annunciator alarm. This would immediately restore normal water level. Should he neglect to do this, the water level would continue to drop until it reaches Level 2. This level would normally initiate both HPCS and RCIC and trip the recirc pumps. Assuming the additional electric failure of HPCS, only RCIC will start. Since a successful scram occurred at Level 3, RCIC is sufficient to cause water level to turn around between Level 2 and Level 1 and rise; slowly filling the vessel as power decays. If still unattended, the vessel level will gradually increase until it reaches Level 8 which trips the RCIC turbine and assures closure of the main turbine stop valves. Thus, level will drop back toward Level 2 and the cycle will continue to repeat itself even slower due to residual heat decay occuring in the vessel. This will limit vessel level between Level 2 and Level 8 indefinitely until the operator takes the remaining shutdown action. The postulated scenerio therefore has no adverse safety consequences for BWR/6 solid-state plants.

See also question 421.35 and its associated response.

FINAL DRAFT

QUESTION

421.38 (7.2) In Section 7.2.2.2.C.1.e of your FSAR, you state that the turbine stop valve closure trip and the turbine control valve closure trip are not guaranteed to function during a safe shutdown earthquake (SSE). We recognize that full conformance to the requirements of IEEE Std. 279 and the other standards referenced in IEEE Std. 279 is not possible in those plants where the turbine building is not a seismic Category I structure. These limitations are acceptable if you install a system which has adequate reliability. Accordingly, verify that the design of these trips, up to the trip solenoids, conforms to those sections of IEEE Std. 279 addressing: (1) single failure (Section 4.2); (2) Quality (Section 4.3); (3) Channel Integrity (Section 4.5 excluding seismic); (4) Channel Independence (Section 4.6); and (5) Testability (Section 4.10).

421.38 RESPONSE

(1) GESSAR II Section 7.2.2.2.C.1.b verifies that the design of these trips conforms to Section 4.2 (single failure criterion) of IEEE Standard 279. Four separate divisions of turbine stop valve and control valve closure sensors and circuits provide trip signals to the Reactor Protection System. Trips from any two of the four divisions will cause an RPS trip (scram). Any single failure within the RPS will not prevent proper protective action at the system level when required.

(2) & (3) GESSAR II Sections 7.2.2.2.C.1.c and e verify that the design of these trips conforms to Sections 4.3 (Quality of Components and Modules) and 4.5 (Channel Integrity), excluding seismic only within the turbine building, of IEEE Std 279. The ~~utility/~~ applicant will provide the information on the quality and integrity of the turbine stop valve and control valve closure sensors and circuits within the turbine building. Generally, GE requires all hardware which contributes to scram to be qualified per IEEE 279 (excluding seismic in the turbine building if desired by the ~~utility/~~ applicant).

(4) GESSAR II Section 7.2.2.2.C.1.f verifies that the design of these trips conforms to Section 4.6 (Channel Independence) of IEEE 279. The logic and control circuits for the four RPS divisions are independent and physically separated by barriers and/or distance. The ~~utility/~~ applicant will provide the information on channel independence of the turbine stop valve and control valve closure sensors and circuits within the turbine building.

(5) The RPS logic and control circuits are testable up to the trip solenoid even during plant operation. GESSAR II Section 7.2.2.2.C.1.j verifies that the design of these trips conforms to Section 4.10 (Capability for Test and Calibration) of IEEE Std 279. The ~~utility/~~ applicant will provide information on the capability for test and calibration of the turbine stop valve and control valve closure trip sensors.

421.38 a QUESTION

Verify that your proposed design includes a highly reliable power source which assures availability of the system.

421.38 a RESPONSE

The RPS is supplied by highly reliable class 1E power sources which assure availability of the system. See Figure 8.3-1 of GESSAR II for the class 1E power sources; also Figure 7A.2-1 for their connections to the system.

421.38 b QUESTION

Using detailed drawings, describe the routing and separation for this trip circuitry from the sensor in the turbine building to the final actuation in the reactor trip system (RTS).

421.38 b RESPONSE

Using detailed drawings, the ~~utility/~~applicant will describe the routing and separation for this trip circuitry from the sensors in the turbine building, which is not a seismic category 1 structure, to their interfaces with the class 1E routed and separated circuits in the Nuclear Island buildings, which are seismic category 1 structures. The routing and separation in the Nuclear Island buildings is discussed in GESSAR II Sections 7.2.1.1.D.7 (Separation) and 7.2.2.2.C.9 (IEEE Std 384-1974).

421.38 c QUESTION

Discuss how the routing within the non-seismically qualified turbine building provides assurance that the effects of credible faults or failures in these circuits will not challenge the reactor trip system and/or degrade the RPS performance. Your response should include a discussion of any isolation devices you have or may propose to install.

421.38 c RESPONSE

See item (1) preceding. In addition, the failure of any one division will not degrade RPS performance. The main turbine trip signals from the turbine building are optically decoupled from the RPS logics and circuits in the control room. Any failure in the main turbine trip sensors in the turbine building will not propagate to the RPS.

The turbine stop valve closure trip and turbine control valve closure trip are backed up by the reactor high pressure trip and the high neutron flux trip.

From above, the routing of the turbine trip signals does not degrade the RPS integrity and finction.

421.38 d QUESTION

The position indicator lights for the turbine stop valves are not part of the RPS. Provide details of the design interface areas using appropriate drawings. Provide your basis for assuring conformance to the requirements of Section 4.20 of IEEE Std. 279-1971.

421.38 d RESPONSE

The ~~utility/~~applicant will provide the details of the design interface areas using appropriate drawings, and the basis for assuring conformance with Section 4.20 of IEEE 279-1971.

421.38 e QUESTION

Identify any other sensors or circuits used to provide input signals to the protection system or perform a function required for safety which are located or routed through non-seismically qualified structures. This should include sensors or circuits providing input for a reactor trip and emergency safeguards equipment including safety-grade interlocks. Discuss the degree of conformance of your design to IEEE Std. 279 and its referenced standards.

421.38 e RESPONSE

The turbine stop valve closure trip, the turbine control valve closure trip, and their bypasses based on turbine first stage pressure are the only sensors or circuits used to provide signals to the RPS or perform a function required for safety; which are located in and/or routed through a non-seismically qualified structure.

FINAL DRAFT

421.39 (7.3.1) QUESTION In Section 7.3.1.1.2.K of your FSAR, you indicated that the containment and reactor vessel isolation control system (CRVICS) is capable of operation during any unfavorable ambient conditions anticipated during normal operation. Discuss the capability of the CRVICS to function during abnormal and accident conditions such high-energy line breaks.

RESPONSE:

The CRVICS is made up of two separate divisions of equipment controlling two sets of valves; one set outside the containment and the other set on the inside of the containment with certain lines having their inboard valves within the drywell. All of the CRVICS valves close on low reactor vessel level and all except the MSIVs and those valves associated therewith (MS drain valves and Reactor Water sample) close on Drywell High Pressure. Isolation valves within the drywell are ~~required~~ Qualified to withstand the temperature and pressure and radiation conditions of all normal, abnormal and LOCA ~~with a time limit on the duration of the LOCA environment because of their short function time for closure on a LOCA signal. Since all the valves that close on drywell high perssure start to close when the drywell pressure exceeds two psig they do not have time to reach LOCA ambient steady state conditions before they are closed and their isolation mission is completed.~~ check qualif. time on MOV.

Consideration of localized damage to equipment as a result of a LOCA focuses attention on the inboard isolation valves and their ability to withstand jet forces and missiles associated with a LOCA. While it is true that an inboard valve may be affected by such forces, it is beyond the design basis to impose a LOCA pipe break and more than one single failure beyond those which can be postulated as consequential. With this groundrule it is evident that the inboard

421.39
(7.3.1)
(cont'd)

isolation valve failure as a result of consequential damages would not open a release path for radioactivity if the line involved were part of a closed system and had another isolation valve on the outside of the containment.

The following considers postulated damage to various inboard isolation valves and cites mechanisms of potential failures together with the isolation condition resulting.

<u>MSIVS Normally Open- Fail Closed</u>

Electrical or air service interruption may be impaired by LOCA. The valves are capable of closure on loss of air or electric power or both. Additionally a third manually operated Motor Operated valve is provided.

<u>MAIN STEAM DRAIN VALVES - Motor Operated Valves.</u>

These valves are normally closed during power operation but open during low power operation. Therefore, failures (electrical cable damage or mechanical damage to the operator) could open a release path to the main condenser if the outboard drain valve failure was the SAF. Because of this possibility, the MS drain valves inside the drywell are located in a protected area within the guard piped area of the main steam lines and considered to be out of the LOCA consequential damage zone. The ~~utility~~ applicant is responsible for the installation adequacy with regard to explicit protection methods. ~~The AE and constructor are responsible for the design and installation adequacy with regard to explicit protection methods. (CF Braun is to verify the foregoing statement).~~

<u>SHUTDOWN COOLING Suction Valves, Normally Closed MOV's.</u>

Since the inboard valve is normally closed and no electrical failure as a consequence of a LOCA can command the valve to open there is no release path established through this valve.

421.39 (7.3.1) (cont'd) Reactor Water Cleanup Inboard Isolation Valve - Motor Operated. Fail-as-Is.

Damage mechanisms include cable damage or mechanical damage to the operator rendering the valve incapable of closure.

In view of the fact that the portion of the RWCU system outside containment is closed system and also protected by a second isolation valve; no radiation release path will result from inboard valve failure and a single active failure. (or single passive failure provided the outboard valve operates.)

OTHER Inboard Containment Isolation Valves. Motor Operated, Normally Open.

Closed cooling water, chilled water and air systems are examples of systems that could communicate with the drywell atmsophere in the event of a LOCA and consequential breakage of one of these pipes. The damage could also be postulated to damage the inboard valve but in each case the outboard valve and the closed system piping outside the containment would accommodate a single active failure or single passive failure without opening a release path to the environs.

FINAL DRAFT

421.41 QUESTION

In your discussion of the high pressure core spray (HPCS) system in Section 7.3.1.1.1.1.C of your FSAR, you state that the HPCS system provides water to the reactor as long as a high drywell pressure signal is present, regardless of the water level in the vessel. The control logic has been modified in the HPCS designs of other BWR's (e.g., Grand Gulf and Clinton) to stop the HPCS when the water level reaches the high level trip. This modification was implemented to prevent possible flooding of the steam lines and subsequent damage to safety/relief valves and the primary system piping. Discuss your proposed HPCS control and its "termination" logic.

421.41 RESPONSE

The GESSAR II design is to be modified just like the other BWR's mentioned. Engineering Change Authorization (ECA) number 801203-1 is already in place to facilitate the change in the GESSAR II documentation. Attached is a mark-up showing how the text will be modified to delete the high drywell signal which inhibits the level 8 trip of HPCS. The HPCS Elementary and FCD will also be revised in accordance with this change.

7.3.1.1.1.1.C High Pressure Core Spray System Instrumentation and Controls (Continued)

transmitter provides an input to an analog trip module (ATM). The output trip signals from the analog trip modules feed into one-out-of-two twice logic. The initiation logic for HPCS sensors is shown in Figure 7.3-1.

Drywell pressure is monitored by four pressure transmitters (two in Division 3 and two in Division 4). 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 the trip logic shown in Figure 7.3-1. The trip logic inputs are electrically connected to a one-out-of-two twice circuit.

The HPCS system is initiated on receipt of a reactor vessel low water level signal or drywell high-pressure signal from the trip logic. The HPCS system reaches its design flow rate within 27 seconds of receipt of initiation signal. Makeup water is discharged to the reactor vessel until the reactor high water level is reached. The HPCS then automatically stops flow by closing the injection valve if the high water level signal is available, ~~and drywell pressure is below the trip setting.~~ The system is arranged to allow automatic or manual operation. The HPCS initiation signal also initiates the HPCS Division 3 diesel generator.

Two ac motor operated valves are provided in the HPCS pump suction. One valve lines up pump suction from the condensate storage tank, the other from the suppression pool. The control arrangement is shown in Figure 7.3-1. Reactor grade water in the condensate storage tank is the preferred source. On receipt of an HPCS initiation signal, the condensate storage tank suction valve is automatically signaled to open (it is normally in the open position) unless the pump suction from the suppression pool

7.3.1.1.1.1.C High Pressure Core Spray System Instrumentation and Controls (Continued)

The valves in the test line to the condensate storage tank are interlocked closed, if the suppression pool suction valve is not fully closed, to maintain the quantity of water in the suppression pool.

4. Redundancy and Diversity

The HPCS is actuated by reactor vessel low water level or drywell high pressure. Both of these conditions may result from a design basis loss-of-coolant accident.

The HPCS system logic requires two independent reactor vessel water level measurements to concurrently indicate the high water level condition. When the high water level condition is reached following HPCS operation, ~~and drywell pressure is below the trip setting,~~ these two signals are used to stop HPCS flow to the reactor vessel by closing the injection valve until such time as the low water level initiation setpoint again is reached. Should this latter condition recur, HPCS will be initiated to restore water level within the reactor.

5. Actuated Devices

All motor-operated valves in the HPCS system are equipped with remote-manual functional test feature. The entire system can be manually operated from the main control room. Motor-operator valves are provided with limit switches to turn off the motor when the full open or closed positions are reached. Torque switches also control valve motor forces while the valves are seating.

The HPCS valves must be opened sufficiently to provide design flow rate within 27 seconds from receipt of the initiation signal.

7.3.1.1.1.1.C High Pressure Core Spray System Instrumentation and Controls (Continued)

The HPCS pump discharge line is provided with an ac motor operated injection valve. The control scheme for this valve is shown in Figure 7.3-1. The valve opens on receipt of the HPCS initiation signal. The pump injection valve closes automatically on receipt of a reactor high water level signal. ~~and when drywell pressure is below the trip setting.~~

6. Separation

Separation within the Emergency Core Cooling System is in accordance with criteria given in Subsection 8.3.1.4.2. It is such that no single design basis event can prevent core cooling when required. Control and electrically driven equipment wiring is segregated into three separate electrical divisions, designated 1, 2, and 3 (Figure 8.3-1).

HPCS is a Division 3 system augmented by redundant Division 4 instrument channels (Figure 8.3-1). In order to maintain the required separation, HPCS control logic, cabling, manual controls and instrumentation are mounted so that divisional separation is maintained. System separation is as shown in Table 8.3-1.

7. Testability

The high pressure core spray instrumentation and control system is capable of being tested during normal unit operation to verify the operability of each system component. Testing of the initiation transmitters which are located outside the drywell is accomplished by valving out each transmitter, one at a time, and applying a test pressure source. This verifies the

6.3.2.2.1 High Pressure Core Spray (HPCS) System (Continued)

opening with the maximum differential pressure across the valve expected for any system operating mode, including HPCS pump shutoff head. The valve opens within 12 sec following receipt of a signal to open. This valve is normally closed to back up the inside testable check valve for containment integrity purposes. A drain line is provided between the two valves. The test connection line is normally closed with two valves to assure containment integrity.

Remote controls for operating the motor-operated components and diesel generator are provided in the main control room. The controls and instrumentation of the HPCS System are described, illustrated and evaluated in detail in Section 7.3.

The HPCS System is designed to pump water into the reactor vessel over a wide range of pressures. For small breaks that do not result in rapid reactor depressurization, the system maintains reactor water level and depressurizes the vessel. For large breaks, the HPCS System cools the core by a spray.

If a LOCA should occur, a low water level signal or a high drywell pressure signal initiates the HPCS and its support equipment. The system can also be placed in operation manually.

The HPCS System is capable of delivering rated flow into the reactor vessel within 27 sec following receipt of an automatic initiation signal.

When a high water level in the reactor vessel is signaled, the HPCS is automatically stopped by a signal to the injection valve to close. ~~unless a high drywell pressure signal exists. If a high drywell pressure signal exists in conjunction with a high reactor water level signal, HPCS injection will continue until~~

6.3.2.2.1 High Pressure Core Spray (HPCS) System (Continued)

~~manually stopped.~~ The HPCS System also serves as a backup to the RCIC System in the event the reactor becomes isolated from the main condenser during operation and feedwater flow is lost.

If normal auxiliary power is not available, the HPCS pump motor is driven by its own onsite power source. The HPCS standby power source is discussed in Section 8.3.

The HPCS pump head flow characteristic used in LOCA analyses is shown in Figure 6.3-3. When the system is started, initial flow rate is established by primary system pressure. As vessel pressure decreases, flow will increase. When vessel pressure reaches 200 psid*, the system reaches rated core spray flow. The HPCS motor size is based on peak horsepower requirements.

The elevation of the HPCS pump is sufficiently below the water level of both the condensate storage tank and the suppression pool to provide a flooded pump suction and to meet pump NPSH requirements with the containment at atmospheric pressure and the suction strainer 50% plugged. The available NPSH has been calculated in accordance with Regulatory Guide 1.1.

A motor-operated valve is provided in the suction line from the suppression pool. The valve is located as close to the suppression pool penetration as practical. This valve is used (1) to isolate the suppression pool water source when HPCS System suction is from the condensate storage system, and (2) to isolate the system from the suppression pool in the event a leak develops in the HPCS System.

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*psid = differential pressure between the reactor vessel and the suction source.

FINAL DRAFT

QUESTION

421.42 (7.3) In Section 7.3.1.1.1.1 of your FSAR, you indicate that the HPCS system will automatically initiate, if required, during testing with specific exceptions. Parts of the system which are bypassed or rendered inoperable are indicated in the control room at the system level. In your response to Question 421.04, provide details relating to the HPCS system. Specifically, discuss the interlock which prevents HPCS injection into the reactor when test plugs are inserted during logic testing. Resolve the discrepancy between your statements in Sections 7.3.1.1.1.1.C.7 and 7.3.2.1.C.1.j.

RESPONSE

421.42 The High Pressure Core Spray (HPCS) system is capable of being completely tested during normal plant operation. Motor-operated valves can be exercised by the appropriate control relays and starters. Should HPCS be initiated during testing, valves will re-align, allowing high pressure core spray into reactor vessel. A motor-operated valve (MOV) test switch in control room removes the over-torque interlock bypass associated with MOV's for testing. This is considered less reliable mode of operation, but does not prevent HPCS initiation (HPCS OUT OF SERVICE light illuminates in control room). During plant normal operation HPCS system can be flow tested by discharging into condensate storage tank.

421.42 (Continued)

HPCS logic is tested by applying a test signal to each analog trip module (ATM) in turn and observing that channel trip device changes state. ~~To verify that both elements in one out of two twice logic are functional a plug in test box is used to operate the logic as one out of one for verification of single element function.~~

If desired, the variable associated with the ATM can be varied and, in conjunction with the ATM output indicator light and appropriate instruments, both the transmitter and ATM outputs can be verified. In those cases where the sensor is disconnected from the process variable to allow testing, an out-of-service alarm will be indicated in control room by administrative action or automatically when analog comparator trip unit is in calibration. Test specification allows this system (division 3 power augmented by division 4 channels) to be down for testing during plant normal operation. The <u>HPCS OUT OF SERVICE</u> light in control room will indicate HPCS is at degraded performance or inoperable during these conditions.

Though not implemented to meet the requirements of testability, the Automatic Pulse Test (APT) continuously and automatically performs end to end testing of all active circuitry. The APT improves availability of HPCS system by minimizing time to detect and locate failures.

7.3.2.1.2.C.1.i) Specific Regulatory Requirements Conformance (Continued)

The sensors can be calibrated by application of pressure from a low pressure source (instrument air or inert gas bottle) after closing the instrument valve and opening the calibration valve.

However, transmitter output is continually monitorable from the control room by observing meters on master trip units. Accuracy checks can be made by cross comparison of each of the four channels (A, E, B and F). For this reason, transmitters need not be valved out of service more than once per operating fuel cycle.

The trip units mounted in the control room are calibrated separately by introducing a calibration source and verifying the setpoint through the use of a digital readout on the trip calibration module.

j) Capability for Test and Calibration (IEEE-279-1971, Paragraph 4.10)

1) HPCS

HPCS control system is capable of being completely tested during normal plant operation to verify that each element of the system, active or passive, is capable of performing its intended function. Sensors can be exercised by applying test pressures. ~~Logic can be exercised by means of plug-in test switches used alone or in conjunction with single sensor tests.~~ Pumps can be started by the appropriate breakers, to pump against system injection valves and/or return to the suppression pool through test valves while the reactor is at pressure. Motor-operated valves can be exercised by the appropriate control relays and starters, and all indications and annunciations can be observed as the system is tested. Check valves are testable by a remotely operable pneumatic piston. HPCS water will not actually be introduced into the vessel except initially before fuel loading.

condensate storage tank

FINAL DRAFT

421.47 In Section 7.3.1.1.1.2.C of your FSAR, your briefly mention testing of the automatic depressurization system (ADS) solenoid valves. These valves cannot be fully tested with the plant at power. Provide a discussion of your proposed method for integrated testing of these valves and circuits, including the frequency of testing. Identify other ESF systems where either a portion of the actuation circuitry or the actuated device is not routinely tested with the actuation circuits. Discuss your proposed method for integrated testing of the circuits and components, including the test frequency.

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RESPONSE:

Integrated testing of the ADS solenoid valves and circuitry is not performed with the plant operating at power which is consistent for safety systems where the final actuating device(s) would cause temporary modification of plant processes such as fluid injection or discharges. The design provides for a functional partially integrated test without valve actuation. This is supplemented by a manual one-at-a-time valve test using associated actuation circuitry from the transmitter trip units with the reactor shutdown but with steam dome pressure equal or greater than 100 psig. This test interval is 18 months. Additionally, the transmitter/trip units that provide sensory inputs to the ADS are checked by control room personnel and the logic chain up to the solenoid is tested by the automatic pulse test performed by the self test sub-system and described as the sixth test in the discussion in 7.1.2.1.6

Other safety system such as RPS, portions of CRVICS, MS-PLCS, HPCS, LPCS, RHR/LPCI, RHR/containment spray mode, RHR/suppression pool cooling mode, safety relief valves, and water positive seal system likewise have components which are not activated or tested with a complete integrated testing procedure. Each of these systems has a modified test procedure that utilizes a manual test which allows for independently checking of individual components. This includes "verification of flow" tests by using the installed return piping such that the motors, pumps and valves are operated, with the associated installed sensors and circuits monitored to verify proper operation. The injection valves are checked independently and separately by manual initiation.

FINAL DRAFT

421.48 QUESTION

In Section 7.3.1.1.1.2.C of your FSAR, you indicate that the ADS can be manually reset after initiation and its delay timers recycled. The operator can delay or prevent subsequent automatic opening of the ADS valves if such delay or prevention is deemed prudent by the operator. Discuss the details of the manual reset capability, using appropriate drawings. Provide the following additional information:

a. The conditions and information which the operator used in making a judgement to exercise the manual over-ride of a subsequent automatic signal.

b. Address the concerns identified in Question 421.14.

421.48 RESPONSE

After receipt of an ADS initiation signal, the 105-second delay timers are started; the ADS valves will not open until the timers time out. Before time out, the operator may reset the timers for additional delay by activating the timer reset push buttons.

The delay in starting the ADS functions allows the high pressure systems sufficient time to arrest the decline of reactor water lever and refill the vessel, while allowing enough time for the low pressure systems to come up to rated conditions.

Resetting the ADS timers does not change the state of the initiating circuits, it merely extends the time delay before the ADS function takes place or until the initiating condition ceases.

The operator should base his decision to reset the timers on information provided by safety-related displays; i.e, reactor pressure, reactor water level, and water inventory make-up system performance.

QUESTION

FINAL DRAFT

421.51 (7.7.1) You describe the performance monitoring system in Section 7.7.1.5 of your FSAR. Provide the following additional information in this section:

a) Identify all safety-related parameters which will be monitored with the performance monitoring system during initial operation.

b) For each safety parameter identified above, provide a concise description of how its associated circuitry connects (either directly or indirectly by means of isolation devices) with the performance monitoring system circuitry. Where appropriate, supplement this description with detailed electrical schematics.

c) Describe your proposed design provisions to prevent failures of the performance monitoring system degrading safety-related systems.

d) Provide the above information for the startup "transient monitoring system," if provided and distinct from the performance monitoring system.

RESPONSE

421.51 a) The following parameters in safety-related systems will be monitored during initial operation:

SYSTEM	PARAMETERS
Nuclear Boiler/Nuclear Steam Supply Shutoff System (NBS)	Vessel Wide Range Level
	ADS/SRV Position
	ADS/SRV Initiation Signal
	MSIV's Position
	MSIV's Isolation Trip Signal
	Vessel High/Low Level Alarm
	RHR/ADS/LPCS/HPCS Low Water Level Initiation Signals
	RHR/ADS/LPCS/HPCS High Drywell Pressure Initiation Signals
Neutron Monitoring System (NMS)	APRM Output
	APRM Heat Flux
	LPRM Output
	Recirc. System Flow

SYSTEM	PARAMETERS
Reactor Protection System (RPS)	Reactor Manual Scram Reactor Scram Trip System
Residual Heat Removal System (RHR)	RHR System Flow (A,B,C) RHR Heat Exchanger Iniet Temp (A,B) RHR Heat Exchanger Outlet Temp (A,B) RHR System Pressure
Low Pressure Core Spray System (LPCS)	LPCS System Pressure LPCS System Flow
High Pressure Core Spray System (HPCS)	HPCS System Pressure HPCS System Flow

b) Isolation will be accomplished by means of optical isolators. The isolation will be accomplished downstream of signal conditioning and analog-to-digital conversion. Figure 1 demonstrates a typical signal flow from a safety-system parameter to the non-safety PMS. The optical isolators shall be qualified in accordance with Regulatory Guides 1.75 and 1.89. The isolators provide a means for preventing a fault in the non-divisional wiring from affecting the safety-system circuitry. Figure 2 exhibits the power and signal connnections to the isolators.

c) To maintain the PMS as a highly reliable system, its normal power will be supplied from an uninterruptible power source (UPS). In addition, interfaces to safety system will be by means of isolation devices. Failures in the PMS will not affect safety-system operation other than possible erroneous operator information.

~~d) Based upon current transient monitoring requirements, the following safety systems will have interface with the safety transient monitoring system:~~

~~Neutron Monitoring System~~
~~Reactor Protection System~~
~~Nuclear Boiler System~~
~~Nuclear Steam Supply Shutoff System~~
~~Diesel Generator System~~
~~4.16 kV Power Distribution System~~
~~High-Pressure Core Spray System~~
~~Residual Heat Removal System~~

d) The start-up Transient Monitoring System is not to be reviewed for GESSAR II docket because the system design and its interfaces cannot be specified at this time.

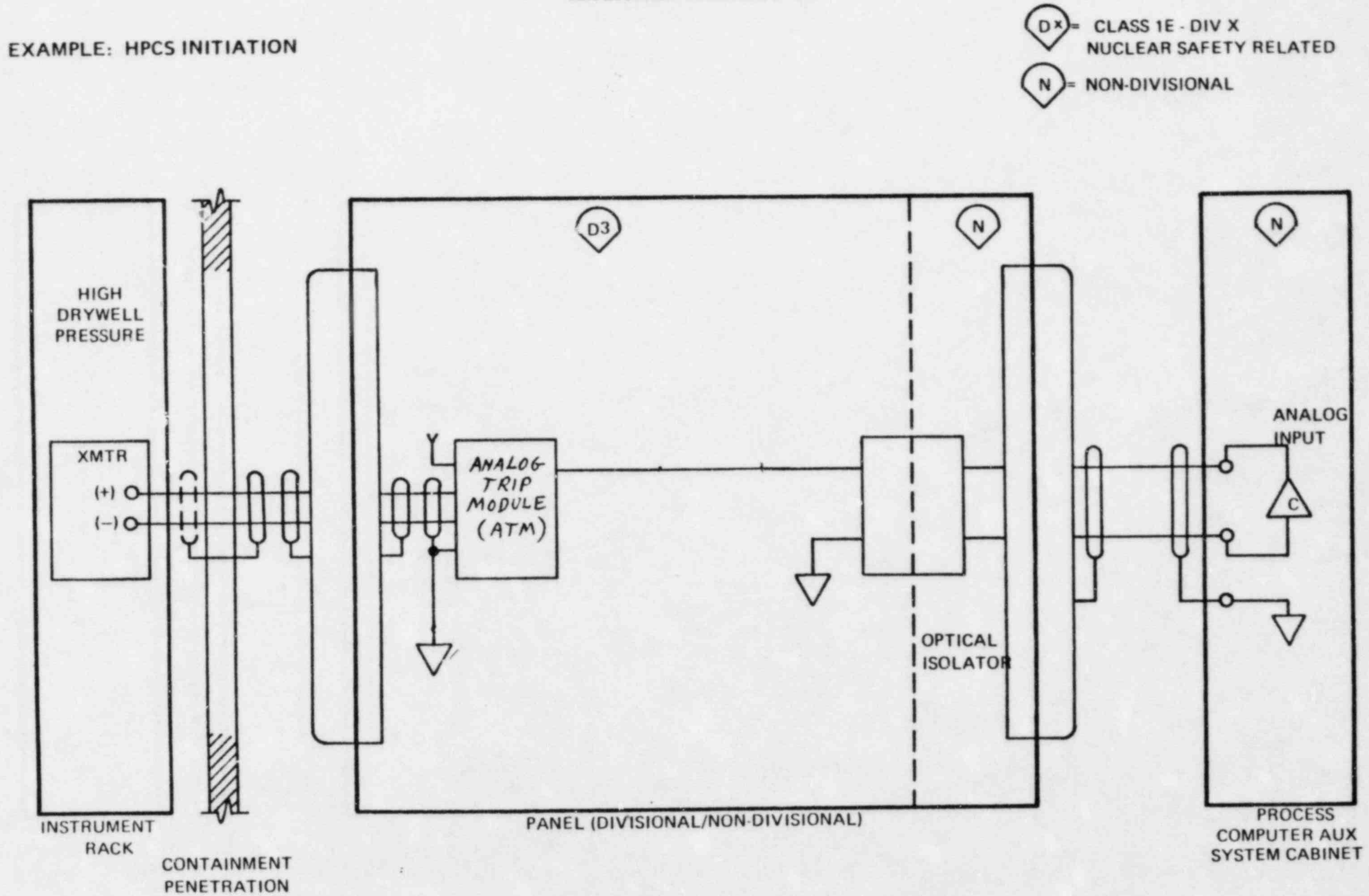
421.51 (Continued)

A concise description of how the associated circuitry merges or connects with the start-up transient monitoring system is inappropriate at this time because system design is not yet specified. Response at a later date, after system design, will be necessary to properly respond to this question. ~~C. F. Braun and Company will provide the system design.~~ ~~Utility/Applicant to provide~~

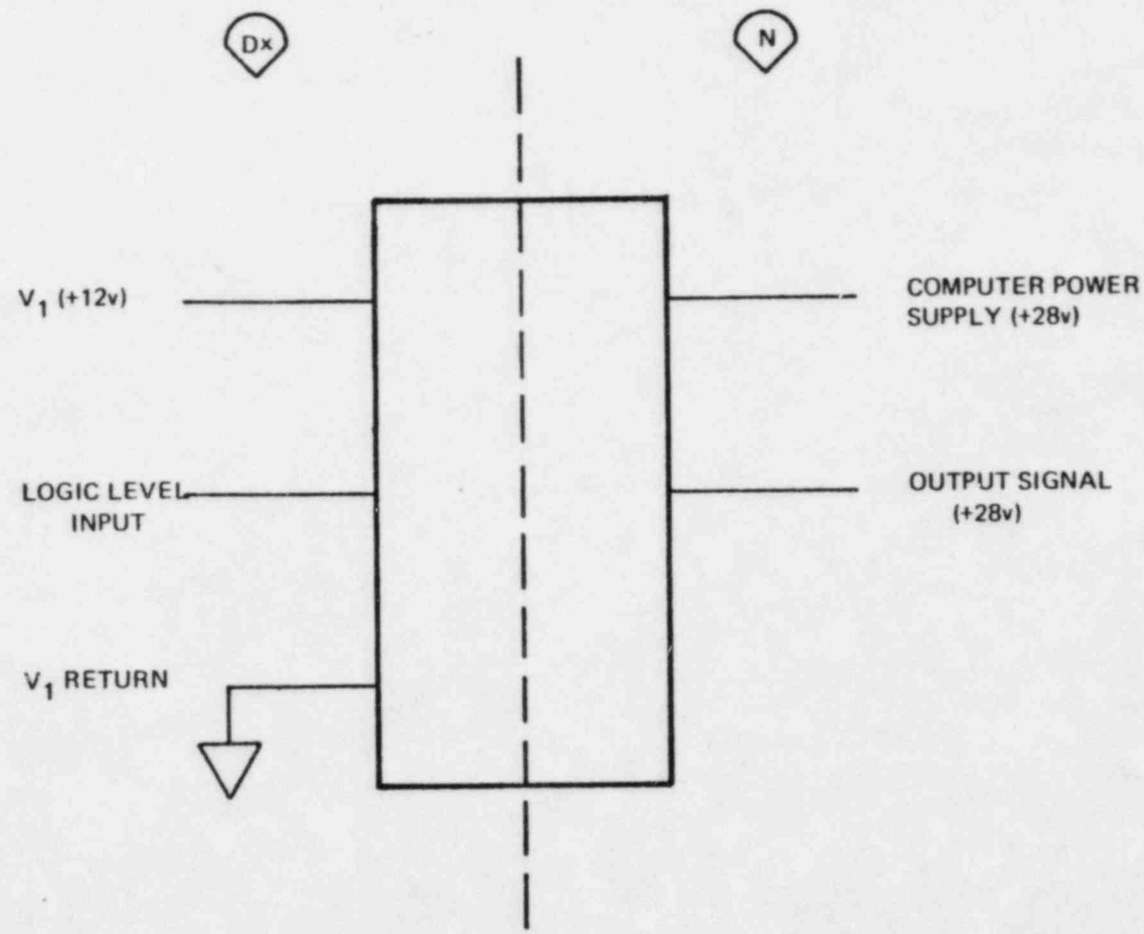
FIGURE 1

TYPICAL SAFETY PARAMETER ISOLATION PROTECTION

EXAMPLE: HPCS INITIATION



COMPUTER OPTICAL ISOLATOR



421.52 QUESTION
(7.7) The analyses discussed in Chapter 15 of your FSAR are intended to demonstrate the adequacy of safety systems in mitigating anticipated operational occurrences and accidents. Based on the conservative assumptions made in defining these "design bases" events and our detailed review of these analyses, it is likely that they adequately bound the consequences of single control system failures. To provide assurance that the design basis event analyses for your proposed design adequately bounds other more fundamental credible failures, provide the following additional information:

a. Identify those control systems whose failure or malfunction could seriously impact plant safety.

421.52 a RESPONSE

Control systems whose failure or malfunction could seriously impact plant safety are those that affect reactor pressure, water level, or power level. A list of those systems is attached.

(applicant to verify)

List 1

SYSTEMS WHICH COULD AFFECT REACTOR PARAMETERS

Condenser Air Removal

Reactor Plant Component Cooling Water

Turbine Plant Component Cooling Water

Condensate

Bearing Cooling Water

Main Steam Isolation Valve Seal

Circulating Water

Turbine Building Equipment Drains

Moisture Separator Vents and Drains

Moisture Separator RHTR Vents and Drains

Turbine Building Miscellaneous Drains

Extraction Steam

FDW Pump 4 Drive Lube Oil

FDW Pump Recirculation

Feedwater

Generator Leads Cooling

Generator Stator Cooling Water

Generator Hydrogen and CO_2

High-Pressure FDW Heater Drain

Low-Pressure FDW Heater Drain

Service Air

Instrument Air

List 1 (cont)

Main Steam

Offgas

Reactor Recirculation

Control Rod Drive

FDW Heater Relief Drains and Vents

Reactor Plant Sampling

Turbine Plant Sampling

Radwaste Building Sampling

Service Water

Turbine Trips

Turbine Generator E.H. Fluid System

Turbine Generator Gland Seal and Exhaust

Turbine Generator Lube Oil

Unit Runback

Turbine Generator Exhaust Hood Spray

Reactor Water Cleanup

Nuclear Boiler

Feedwater Control

Neutron Monitoring

Steam Bypass and Regulation

PROCESS RADIATION MONTORING

421.52 b QUESTION

Indicate which, if any, of the control systems identified in Item (a) receive power from common power sources. The power sources considered should include all power sources whose failure or malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses due to the failure of higher level distribution panels and load centers.

421.52 b RESPONSE

Applicant will provide (See Section 1.9 of GESSAR II).

~~Control systems whose failure or malfunction could seriously impact plant safety are those that affect reactor pressure, water level, or power level. A list of those systems is attached. Previous analysis has shown that of these systems, the ones that can produce reactivity increases in conjunction with delayed turbine trips due to single electrical failures are the most critical systems. Guidance will be provided in Section 1.9 to avoid common power or sensors in these systems.~~

421.52 c QUESTION

Indicate which, if any, of the control systems identified in Item (a) receive input signals from common sensors. The sensors considered should include common taps, hydraulic headers and impulse lines feeding pressure, temperature, level or other signals to two or more control systems.

421.52 c RESPONSE

Applicant will provide (See GESSAR II, Section 1.9).

~~ATTACHMENT AND ROUTING OF SENSOR LINES WILL BE APPLICANT UNIQUE. SUFFICIENT GUIDANCE WILL BE PROVIDED AS STATED IN PART b TO AVOID PLACING CRITICAL SYSTEM'S INSTRUMENTS ON COMMON LINES~~

421.52 d. QUESTION

Provide your analysis to show that any malfunctions of the control system identified in Items (b) and (c) resulting from failures or malfunctions of the applicable common power source or sensors including hydraulic components, are bounded by the analyses in Chapter 15 and would not require action or response beyond the capability of operators or safety systems. Where credit is taken for operator action, identify the time available for such action.

421.52 d RESPONSE

Applicant will provide (See Section 1.9 of GESSAR II)

~~TO BE PROVIDED BY APPLICANT~~

FINAL DRAFT

421.53 In Section 11.5.2.1.2 of your FSAR, you indicate that if one channel in both the A and B trip logic is downscale in the reactor containment heating, ventilation and air-conditioning (HVAC) radiation monitoring system, system isolation is not possible. Your design is such that any one downscale trip sounds an alarm in the control room. Discuss the details of your design which are provided to preclude downscale trips in one channel in each logic from occurring simultaneously. Discuss the required actions, either automatic or by the operator, including the procedures to be followed by the operator if a channel in one or both logics is downscale. Indicate whether the details provided in this discussion are applicable to the other radiation monitoring systems identified in Section 11.5.2.1 of your FSAR.

Response

The logic embodied in the reactor containment heating, ventilation and air conditioning (HVAC) radiation monitoring system (see figure 7A.6-4K) is such that either a downscale trip or an upscale per channel will be sufficient to provide one half of the required signal for the interlock. Figure 7.6-10C Note 7 also indicates that two-out-of-two high high/inop or downscale trips (in either A and D or B and C) will provide an interlock signal.

The procedure to be followed in case of a downscale trip will be provided in the technical specifications chapter of the Safety Analysis Report. Although the exact procedure will need to be reviewed by the applicant, in general, the following is typical: With the requirements for the minimum number of Operable channels not satisfied for one trip system, place the inoperable channel in the tripped condition within one hour or establish Secondary Containment Integrity with the standby gas treatment system operating within one hour. With the requirements for the minimum number of Operable channels not satisfied for both trip systems, establish Secondary Containment Integrity with the standby gas treatment system operating within one hour.

The details provided above are not directly applicable to the other radiation monitirong systems (i.e., containment space - refueling mode, fuel building ventilation exhaust, auxiliary building exhaust, standby gas treatment, shield annulus HVAC, and control building HVAC) in Section 11.5.2.1 because these systems are configured with a one-out-of-~~two~~ one trip logic, in each of two divisions.

Textual corrections for chapter 11 are attached for reference.

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11.5.2.1.2 Containment HVAC Radiation Monitoring System

This system monitors the radiation level exterior to the containment ventilation system exhaust duct. A high activity level in the ductwork could be due to fission gases from a leak or an accident.

The system consists of four redundant instrument channels. Each channel consists of a local detection assembly (a sensor and converter unit containing a GM tube and electronics) and a control room radiation monitor. Power is supplied to each channel, A, B, C, and D from RPS buses E, F, G, and H, respectively. Channels A and D are physically and electrically independent of channels B and C. One two-pen recorder powered from the 120 VAC instrument bus J2 allows the output of any two channels to be recorded by the use of selection switches. The detection assemblies are physically located outside and adjacent to the exhaust ducting upstream of the containment discharge isolation valves.

REPLACE WITH ATTACHED SHEET

Each radiation monitor has two trip circuits: one upscale (high-high) inoperative and one downscale. Two out of two upscale/ inoperative trips in channels A and C initiate closure of the containment ventilation outboard isolation valves and the drywell inboard isolation valves. The same condition for channels B and D initiates closure of the containment inboard valves and drywell outboard valves.

An upscale/inoperative trip is visually displayed on the affected radiation monitor and actuates a containment and drywell ventilation exhaust high-high radiation control room annuniciator for the affected channel.

A downscale trip is also visually displayed on the radiation monitor and actuates a downscale control room annunciator common to all channels. An additional trip signal for high radiation

11.5.2.1.2 Containment HVAC Radiation Monitoring System (Continued)

alarm is provided by the recorder and actuates a reactor building vent high radiation control room annunciator. Each radiation monitor visually displays the measured radiation level.

11.5.2.1.2.1 Containment Space - Refuel Mode Radiation Monitoring Subsystem

This system monitors the radiation level inside the containment above the top of the fuel pool. The radiation monitor elements are located approximately 50 feet above the top of the pool in four positions to facilitate detection of radioactivity instantly in the event of a fuel-handling accident.

The system consists of four instrument channels: A, B, E, and F. Each channel consists of a detector, converter, and a main control room radiation monitor. All four channels are physically independent of each other, but channels A and E share the same power supply. Channels B and F also share a common power supply. Channels A and B E are powered by the 120-vac RPS Bus E, Division 1. Channels B and F are powered by the 120-vac RPS Bus F, Division 2. Channels A and E are electrically independent of channels B and F. As a result there are two independent and redundant instrument systems. The failure of one system does not affect the other.

Each radiation monitor has two trip circuits: one upscale (high)/inoperative and one downscale. Both upscale and downscale trips are displayed on the appropriate radiation monitor and each one actuates either the high or low main control room annunciator.

A high radiation trip on either channel A or E initiates closure of the containment exhaust air isolation valves and containment supply air isolation valves for Division 1. A high radiation trip

GENERAL ELECTRIC CO.
Nuclear Energy Business Operations
ENGINEERING CALCULATION SHEET

NUMBER ______ DATE ______
SUBJECT ______ BY ______ SHEET ______ OF ______

UPDATE FOR GESSAR 11.5.2.1.2

"Each radiation monitor has two trip circuits: one upscale and one downscale. Two out of two trips in channels A and D, which on a per channel basis, may be caused by either a High High radiation signal, or a downscale signal or the Operate switch not in the Operate position, initiates closure of the containment ventilation outboard valves and the drywell inboard isolation valves. The same condition for channels B and C initiates closure of the containment inboard valves and drywell outboard valves.

Either an upscale trip or the Operate switch not in the Operate position for channels A or D actuates a "Containment Vent Division 1 or 4 High High Radiation or Inoperative" alarm. The same condition for channels B or C initiates an alarm corresponding to Divisions 2 or 3. The upscale (high - high) alarm is visually indicated on the radiation monitor.

A downscale trip is also visually displayed on the radiation monitor and actuates a control room annunciator common to all four channels.

An additional alarm signal for high radiation is provided and actuates a control room annunciator common to all channels. A reduction in high voltage associated with channels A, B, C or D causes a "High Voltage Inop" alarm.

Each radiation monitor assembly displays the measured radiation level."