PSNH PUBLIC SERVICE
Company of New Hampshire

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SBN- 384
T.F. B 7.1.2

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. George W. Knighton, Chief
Licensing Branch #3
Division of Licensing

References:
(a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) USNRC Letter, dated February 16, 1982, "Request for Additional Information," F. J. Miraglia to W. C. Tallman
(c) USNRC Memo, dated March 23, 1982, "Additional Agenda Items for Meeting with Seabrook Applicant on Instrumentation and Controls," T. P. Speis to R. L. Tedesco
(d) PSNH Letter, dated April 1, 1982, "Meeting Notes; Instrumentation and Controls Systems Branch (ICSB)," J. DeVincentis to R. Stevens
(e) PSNH Letter, dated June 10, 1982, "Meeting Notes; Instrumentation and Control Systems Branch (ICSB)," J. DeVincentis to F. J. Miraglia
(f) PSNH Letter, dated August 10, 1982, "Meeting Notes, Instrumentation and Control Systems Branch (ICSB)," J. DeVincentis to F. J. Miraglia
(g) PSNH Letter, dated October 14, 1982, "Meeting Notes, Instrumentation and Control Systems Branch (ICSB)," J. DeVincentis to J. Kerrigan

Subject: Meeting Notes; Instrumentation and Controls Systems Branch (ICSB)

B001

Dear Sir:

We have attached notes from the previous ICSB review meetings with additional responses prepared subsequent to the September 14 and 15 meeting. These meetings were based on the ICSB Requests for Additional Information (RAIs) which were forwarded in References (b) and (c). The notes also include those items discussed at the March 23-25, 1982; May 12 and 13, 1982; July 15 and 16; and September 14 and 15, 1982 review meetings that have been revised.

We have indicated the date of the meeting at which the response or a revision to a response was made. The attachments to the previous meeting notes [References (d), (e), (f), and (g)] are not included with this letter.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

J. DeVincentis
Project Manager

ALL/ces

cc: Mr. Robert Stevens
Instrumentation and Control Systems Branch
Division of Systems Integration

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420.5 (7.1) As called for in Section 7.1 of the Standard Review Plan, provide information as to how your design conforms with the following TMI Action Plan Items as described in NUREG-0737:

(a) II.D.3 - Relief and safety valve position indication,

(b) II.E.1.2 - Auxiliary feedwater system automatic initiation flow indication,

(c) II.E.4.2 - Containment isolation dependability (positions 4, 5 and 7),

(d) II.F.1 - Accident monitoring instrumentation (positions 4, 5 and 6),

(e) II.F.3 - Instrumentation for monitoring accident conditions (Regulatory Guide 1.97, Revision 2),

(f) II.K.3 - Final recommendations
.9 - PID controller
.12 - Anticipatory reactor trip.

RESPONSE: 3/23

(a) II.D.3 The single acoustic device to monitor all safety valves is not redundant but is safety grade. Limit switches for each PORV are not redundant but position indication is safety grade. Position indication system is seismically and environmentally qualified. There will be control room alarm for acoustical device and for either PORV not closed. There is backup temperature indication downstream of each safety valve and one temperature indication for both PORVs, all are alarmed in the control room. The FSAR will be revised.

(b) II.E.1.2 Auxiliary feedwater system automatic initiation is safety grade. Flow indication meets Item 2a and b of II.E.1.2.5, NUREG-0737.

(c) & (d) II.E.4.2 and II.F.1 will be handled by containment systems branch.

(e) II.F.3 will be covered by Regulatory Guide 1.97, Response 420.51.

(f) II.K.3.9 and .12, provided response in letter SBN-212, dated 2/12/82. Reviewed by staff and found acceptable.

ADDITIONAL RESPONSE: 5/12

(a) NUREG-0737, Item II.D.3, Clarification was made that the final design of the safety and relief valve position indication is not complete. The project documents and the FSAR will be revised. The block valves, position indication and their manual controls will be Class 1E.

(b) NUREG-0737, Item II.E.1.2, will be addressed in the overall discussions of the emergency feedwater system.

FSAR Figure 7.2-1, Sheet 15 and Page 7.3-23, will be corrected to indicate that both A & B train actuate the turbine driven emergency feedwater pump.

ADDITIONAL RESPONSE: 9/14

(a) FSAR 5.2.2.8 will be revised to provide the information on relief and safety valve position required by NUREG 0737 II.D.3. A handout of the draft FSAR revision is included in the meeting minutes.

(b) The information required by NUREG 0737 II.E.1.2 is provided in the following FSAR sections that are keyed to the 0737 positions:

Part I

(1) 6.8.1 h, 6.8.5
(2) 6.8.1 a
(3) 6.8.4, 7.3.2.2
(4) 8.3
(5) 6.8.1 h
(6) 8.3
(7) 6.8.1 h

The automatic initiation signals and circuits are safety grade.

Part II

(1) 6.8.5
(2) 6.8.5

Note that 6.8 is being revised to include this and other information on EFW changes, a copy of the draft revision is attached as part of the response to RAI 420.36. FSAR Figure 7.2-1, Sheet 15, and p. 7.3-23 will be revised to show that both A & B trains actuate the turbine driven pump. A copy of the FSAR markups are attached.

HANDOUT: 9/14 11/82

Revised FSAR 5.2.2.8 for RAI 420.5 (a).

5.2.2.8 Process Instrumentation

Instrumentation is provided in the control room to give the open/closed status of the pressurizer safety and Power Operated Relief (PORV) Valves. Each PORV is monitored by limit switches that operate red and green indicating lights on the main control board. The safety valves are monitored by an acoustic monitor that senses the acoustic emissions associated with flow in the discharge line that is common to the three safety valves.

All instrumentation will be environmentally and seismically qualified, will be powered from a vital instrument bus, and will actuate VAS alarms. The indication will not be redundant, therefore, backup indication and alarms are provided by | 11/82

temperature indication on the discharge of each safety valve and the common discharge from the PORVs and by primary relief tank temperature, pressure, and level.

The primary and backup instrumentation will be integrated into the emergency procedures and operator training. The human factors analysis will be performed as part of the control room design review.

STATUS: 9/14 — Confirmatory pending ICSB review.

420.6 (7.1) — Provide an overview of the plant electrical distribution system, with emphasis on vital buses and separation divisions, as background for addressing various Chapter 7 concerns.

RESPONSE: 3/23 — Discussed at meeting, no further response required.

STATUS: 5/12 — Closed.

420.7 (7.1) — Describe features of the Seabrook environment control system which insure that instrumentation sensing and sampling lines for systems important to safety are protected from freezing during extremely cold weather. Discuss the use of environmental monitoring and alarm systems to prevent loss of, or damage to systems important to safety upon failure of the environmental control system. Discuss electrical independence of the environmental control system circuits.

RESPONSE: 3/23 — Written response reviewed by the NRC and attached to meeting notes. We reviewed the freeze protection for the refueling water storage tank (RWST) after the meeting. It was determined that the instruments and sensing lines are in the building that encloses the RWST and is maintained above 32°F by the heated RWST. Additional freeze protection is not required. RAI 440.104 is related. This item is under review by the staff.

ADDITIONAL RESPONSE: 5/12 7/15 — Fluid systems are protected from freezing by being 1) located in an area with a heating system; 2) located in an enclosure with a heated tank; or 3) provided by heat tracing.

The majority of the safety-related piping is located in areas that are provided with heating systems. Low ambient temperature is alarmed in the control room. The alarms are not safety grade. The alarm is electrically independent of the heating system. The areas are accessed periodically as part of the operators inspections. The operator will be instructed to notice abnormal ambient temperatures that could result from failure of the heating system.

The tank farm enclosure is maintained above the freezing temperature by the heat lost from the heated RWST. Low ambient,

RWST, and spray additive tank temperatures are alarmed in the control room to warn of abnormal conditions in the tank farm enclosure.

Safety-related piping that is not in heated areas or that require the maintenance of temperatures higher than the design ambient temperatures is provided with dual heat tracing circuits and low temperature alarms.

The alarm and heat tracing circuits are electrically independent, therefore, failure of the heating circuit will not result in loss of the low temperature alarm. Loss of power to the low temperature alarm and heat tracing circuits will be alarmed in the control room.

HANDOUT: 3/23

To ensure that instruments, including sensing and sampling lines, are protected from freezing during cold weather, electrical heat tracing is provided. Heat tracing on safety-related piping is protected by redundant, non-safety-related, heat tracing. On the boron injection line only, the primary heat tracing circuit is train A associated. The backup heat tracing circuit is train B associated. This backup circuit is normally de-energized. On the remaining lines, the redundant heat tracing circuit is energized from the same train as the primary circuit.

Integrity of each circuit is continuously monitored. Low and high temperature alarms are available at the heat tracing system control cabinet. Additionally, failures as detailed below are indicated at the heat tracing control cabinets that are located in the general vicinity of the systems being heat traced:

a) Loss of voltage,

b) Ground fault trip for each heating element circuit,

c) Overload trip of branch circuit breakers,

Trouble alarms are provided in the main control room.

STATUS: 9/14

Closed.

420.8 (7.1)

Provide and describe the following for NSSS and BOP safety-related setpoints:

(a) Provide a reference for the methodology used. Discuss any differences between the referenced methodology and the methodology used for Seabrook,

(b) Verify that environmental error allowances are based on the highest value determined in qualification testing,

(c) Document the environmental error allowance that is used for each reactor trip and engineered safeguards setpoint,

(d) Identify any time limits on environmental qualification of instruments used for trip, post-accident monitoring or engineered safety features actuation. Where instruments are qualified for only a limited time, specify the time and basis for the limited time.

RESPONSE: 3/23 Seabrook uses the same methodology as W used for DC Cook, North Anna and Sumner, there are no differences. DC Cook and North Anna were submitted and approved. This is applicable for both NSSS and BOP safety-related setpoints.

WCAP 8587 and 8687 describe the determination of environmental error allowances.

ADDITIONAL RESPONSE: 9/14 The use of the Westinghouse statistical methodology was accepted by the NRC for Virgil C. Sumner (NUREG 0717 Supplement No. 4). The determination of the Seabrook setpoints will be consistent with the method used for Sumner.

STATUS: 9/14 Confirmatory pending review of formal documentation.

420.9 (7.1.2.5) There is an inconsistency between the discussions in FSAR Section 1.8 and FSAR Section 7.1.2.5 pertaining to the compliance with Regulatory Guide 1.22. FSAR Section 1.3 states that the main reactor coolant pump breakers are not tested at full power. FSAR Section 7.1.2.5 does not include these breakers in the list of equipment which cannot be tested at full power. Please provide a discussion as to whether the operation of the reactor coolant pump breakers is required for plant safety. If not, then please justify. Also, please correct the inconsistency described above and, as a minimum, provide a discussion per the recommendations of Regulatory Position D.4 of Regulatory Guide 1.22.

RESPONSE: 3/23 Revised 1.8 provided to staff and attached to meeting notes, reactor does not trip on opening of reactor coolant pump breakers.

STATUS: 9/14 Closed.

420.10 (1.8) (7.1.2.6) (7.5) Using detailed plant design drawings (schematics), discuss the Seabrook design pertaining to bypassed and inoperable status indication. As a minimum, provide information to describe:

1. Compliance with the recommendations of Regulatory Guide 1.47,

2. The design philosophy used in the selection of equipment/systems to be monitored,

3. How the design of the bypass and inoperable status indication systems comply with Positions B1 through B6 of ICSB Branch Technical Position No. 21, and

4. The list of system automatic and manual bypasses within the BOP and NSSS scope of supply as it pertains to the recommendations of Regulatory Guide 1.47.

The design philosophy should describe, as a minimum, the criteria to be employed in the display of inter-relationships and dependencies on equipment/systems and should insure that bypassing or deliberately induced inoperability of any auxiliary or support system will automatically indicate all safety systems affected.

RESPONSE: 3/23 Handout given to staff. Overview of systems covered and description of operation given including automatic and manual modes, and interaction between systems. Handout as amended during meeting will be attached to the meeting minutes.

System description of computer and video alarm system (VAS) presented during meeting and will be followed up by written description to staff as response to RAI 420.49. A meeting will be held with the staff in Washington at a later date to review all aspects of plant computer operation.

Staff presented concern that some guarantee must be considered as to percent of time computer will be operating and that plant will not continue to operate for any length of time, without appropriate corrective action, when and if computer should be out of service. A possible solution would be to refer operating and repair times to safety review committee although it is agreed that the computer is not a safety-related system. Staff asked for additional information concerning level of validation and verification of software.

HANDOUT: 3/23

1. Systems are designed to meet the recommendations of Regulatory Guide 1.47.

2. Design philosophy is discussed in FSAR Section 7.1.2.6. The selection of equipment is given in Item 4.

3. System design meets the recommendation of ICSB-21 as follows:

   B1 - Refer to FSAR Section 7.1.2.6(a).

   B2 - System design meets the requirements. Refer to logic diagrams listed in FSAR Section 7.1.2.6(f).

   B3 - Erroneous bypassed/inoperable alarm indications could be provided by any of the following:

   - dirty relay contacts
   - dirty limit switch contacts.

   B4 - The bypass indication system does not perform functions essential to safety. (Refer to FSAR Section 7.1.2.6)

   - A system design is supplemented by administrative procedures. The operator will not rely solely on the indication system.

B5 - The indication system does not perform any safety-related functions and has no effect on plant safety systems. The indication system is located at the MCB separately for each train on system level basis.

B6 - All bypass indicators and plant video annunciator systems are capable of being tested during normal system operation.

4. The list of the equipments for which bypass/inoperable alarms and indication are provided.

A1 - Service Water System (SW)

Service	Equipment	Logic Diagram	Schematic
Service Water Pumps	SW-P-41A/41B	M-503968	M-301107 Sh. AG3,AR3
	-41C/41D	M-503969	M-301107 Sh. AG4,AR4
Cooling Tower Pumps	SW-P-110A	M-503966	M-301107 Sh. AU2
	-110B	M-503967	M-301107 Sh. AU6
Cooling Tower Fans	SW-FN-51A	M-503951	M-301107 Sh. AV4
	-51B	M-503452	M-301107 Sh. AW4
Cooling Tower/Service Water Bypass/Inop.		M-503973	M-310951 EH9/EHO

Note: There are separate lights for the service water pump and the cooling tower subsystems.

A2 - Primary Component Cooling Water System (CC)

Service	Equipment	Logic Diagram	Schematic
Primary Cooling Water Pumps	CC-P-11A	M-503270	M-310895 Sh. A58/A78
	11B/11C/11D		A59,A79
PCCW Bypass Inop.		M-503277	M-310951 EH9/EHO

A3 - Containment Building Spray (CSB)

Service	Equipment	Logic Diagram	Schematic
Containment Spray Pumps	CBS-P-9A/9B	M-503257	M-310900 Sh. A61,A81
Containment Sump Iso. Vlv.	CBS-V8/V14	M-503252	M-310900 Sh. B84,D40
Cont. Spray Add. Iso. Vlv.	CBS-V39/V44	M-503259	M-310900 Sh. 4b
Cont. Spray Nozzle Iso. Vlv.	CBS-V13/V19	M-503259	M-310900 Sh. 4b

Service	Equipment	Logic Diagram	Schematic
Primary Comp. Cooling Water to Containment HX	CC-V131/V260	M-503259	M-310895 Sh. 4a
Primary Comp. Cooling Water		M-503259	

A4 - Residual Heat Removal (RH)

Service	Equipment	Logic Diagram	Schematic
RH Cold Leg Inj. Iso. Vlv.	RH-V14/26	M-503768/503769	M-310887 Sh. B57,B65
RH Hot Leg Inj. Iso. Vlv.	RH-V32/70	M-503768/503769	M-310887 Sh. B58,D90
Chg. Pump Suc. Iso. Vlv.	RH-V35	M-503768/503763	M-310887 Sh. B59,B66
SI Pump Suc. Iso. Vlv.	RH-36	M-503768/503763	M-310887
Cont. Sump Iso. Vlv.	CBS-V8/V14	M-503252	M-310900 Sh. B84,D40
Prim. Comp. Cooling Water to HX	CC-V133/V258	M-503768	M-310895 Sh. 4A
Residual Ht. Removal Pumps	RH-P-8A/8B	M-503761	M-310877 Sh. A57,A77

A5 - Safety Injection System (SI)

Service	Equipment	Logic Diagram	Schematic
SI Pumps	SI-P-6A/6B	M-503900	M-310890 Sh. A56/A76
Cont. Sump Iso. Valve	CBS-V8/V14	M-503918	
SI Cold Leg Iso. Valve	SI-V114	M-503918	M-310890 Sh. B49
SI-P-CA-6B to Hot Legs Isolation Valve	SI-V102/V77		
SI-P-6A/6B to RWST Isolation Valve	SI-V89/V90	M-503918	M-310890 Sh. B41/B42
SI-Pump Cross Connect	SI-V111/V112	M-503918	M-310890 Sh. B47/B48
Prim. Comp. Cooling Wtr.		M-503918	M-310895 Sh. EH9/3 EA

A6 - Chemical and Volume Control System (CS)

Service	Equipment	Logic Diagram	Schematic
Charging Pump	CS-P-2A/2B	M-503372,M-503330	M-310891 Sh. A62,A82
Prim. Comp. Cooling Wtr.		M-503372	

A7 - Feedwater (FW)

Service	Equipment	Logic Diagram	Schematic
Emer. Feedwater Pump	FW-P-37B	M-503586	M-310844 Sh. A80
Emer. FW Pump 37A/37B	FW-V71/73	M-503599	M-310844 Sh. 4
Discharge and Bypass Vlvs.	FW-V65/67	M-503599	M-310844 Sh. 4

A8 - Diesel Generator

Service	Equipment	Logic Diagram	Schematic
DG Control Power Lost		M-503495	M-310102
DG Breaker Control Power Lost		M-503495	M-310102
EPS Control Power Lost		M-503495	M-310102
Protection Relays not Reset		M-503495	M-310102
DG - Barring Devices Engaged		M-503495	M-310102
Starting Air Pressure Lo-Lo		M-503495	M-310102
Control Switch Pull to Lock		M-503495	M-310102
Selector Switch Maintenance		M-503495	M-310102

B - Interrelationship Between Auxiliary Systems and Safety Systems

Auxiliary systems such as service water system (SW), primary component cooling water system (CC), and diesel generator system (DG) are dependent on the operation of other auxiliary systems or are required for the operation of other auxiliary or safety systems.

The VAS will automatically indicate the dependent auxiliary and safety systems that are made inoperable by an inoperable auxiliary system. Initiation of the Emergency Power Inoperable indication will automatically initiate all the indicators for the same train on the bypass and inoperable status panel. Initiation of an indicator on the bypass and inoperable status panel is performed manually and will automatically initiate indication of dependent auxiliary and safety systems on the bypass and inoperable status panel.

Reference logic drawings:

M-503277 - M-503973
M-503259 - M-503768
M-503918 - M-503372

ADDITIONAL RESPONSE: 5/12

The handout will be revised to indicate that alarms and indicators are provided. The indication on the bypass and inoperable status panel is on the system level for each train. All automatic initiation is through the VAS. Indication on the status panel is manually initiated in response to the VAS alarm or when the system is bypassed or made inoperable with devices not monitored by the VAS. The VAS and the status panel have logic that will automatically indicate all systems made inoperable when a support system is inoperable.

Typographical errors on A7 and A8 will be corrected.

This items remains open pending the review of the VAS.

After the meeting, a note to clarify the service water indicators was added to A1 of the 3/23 handout. A8 was deleted as the Diesel Generator status monitoring lights and alarms are not considered part of the bypass and inoperable status monitoring system, since the events monitored occur less than once per year. FSAR 7.1.2.6, copy attached, will be revised.

ADDITIONAL RESPONSE: 7/15

Item A8, diesel generator, will be returned to the list as data for other diesels indicate that they may require maintenance outages more than once per year.

The functions that are listed all initiate a VAS common alarm which indicates that a train is inoperable, TRN EMERG POWER INOPERABLE.

Diesel generator status is indicated on the diesel generator status light panel on Section HF of the MCB, not on the bypass and inoperable status light panel on Section CF of the MCB. These status monitoring lights along with specific and common VAS alarms provide continuous status of the diesel generators.

We will add the bypass/inoperable status monitoring system pushbuttons to the computer inputs that initiate the VAS bypass/inoperable alarms. This will ensure that the same information on system status is available at the monitoring system or through the VAS. A summary of the current status of the VAS bypass/inoperable alarms will be available on demand to ensure that operator is aware of the status of redundant systems when a system is bypassed/made inoperable. A system level VAS alarm will be initiated if the redundant trains are bypassed/made inoperable.

ADDITIONAL RESPONSE: 9/14 — The 3/23 handout, Part B, is revised to include the Diesel Generator in the discussion of the interrelationship of the auxiliary systems. Logic diagrams will be changed.

STATUS: 9/14 — Confirmatory pending review of formal documentation.

420.11 (7.1) — Summarize the status of those instrumentation and control items discussed in the Safety Evaluation Report (and supplements) issued for the construction permit which required resolution during the operating license review.

RESPONSE: 3/23 — There are no unresolved items relating to Chapter 7 of the SAR identified in the construction permit SER (Supplements 1 to 4).

STATUS: 5/12 — Closed.

420.12 (7.1.2.2) — Various instrumentation and control system circuits in the plant (including the reactor protection system, engineered safety features actuation system, instrument power supply distribution System) rely on certain devices to provide electrical isolation capability in order to maintain the independence between redundant safety circuits and between safety circuits and non-safety circuits.

1. Identify the type of isolation devices which are used as boundaries to isolate non-safety grade circuits from the safety grade circuits or to isolate redundant safety grade circuits.

2. Describe the acceptance criteria and tests performed for each isolation device which is identified in response to Part 1 above. This information should address results of analyses or tests performed to demonstrate proper isolation and should assure that the design does not compromise the required protective system function.

RESPONSE: 3/23

1. BOP uses the same type W 7300 system, with the same qualifications, as is used by NSSS (NSSS equipment for Seabrook is identical to that for SNUPPS).

2. Radiation data management system will require submittal of further documentation of isolation devices used.

3. Power supply distribution isolation is covered under RAI 430.40A.

ADDITIONAL RESPONSE: 9/14

The current status of the RDMS isolators was discussed. Further discussion is deferred pending overall resolution of train separation criteria.

STATUS: 9/14

Open pending documentation of testing to be performed to show that the isolator will perform the required isolation function. The maximum credible fault voltage and current should be justified.

ADDITIONAL RESPONSE: 11/82

The design of the RDMS supplied by the General Atomic Company is consistent with the criteria for physical independence of electrical systems established in "Attachment C" of AEC letter dated December 14, 1973 (see FSAR Appendix 8A) and in Regulatory Guide 1.75, Revision 2. In addition, the independence of Class 1E equipment and circuits follows IEEE Standard 384-1981, Section 7, regarding specific electrical isolation criteria.

All Class 1E equipment is supplied with power from the appropriate Class 1E power source train.

Communications within the RDMS System between the various microcomputer based monitors takes place via redundant semiduplex lines, transmitting and receiving low level digitally coded signals. All of these monitors are provided with semiconductor-based optical isolators that isolate all communication lines from the internal circuitry of the monitors.

Further, all Class 1E monitors are provided with state-of-the-art fault isolation devices. Each communication line is provided with overcurrent and overvoltage protection. Overcurrent protection is provided by incorporating a low current fuse in each line just before it enters the optical isolator circuitry which is part of each monitor. The overvoltage protection is provided by the use of a Transzorb device between the two communication lines and from each communication line to ground.

The Transzorbs are semiconductor-based devices incorporating a zener diode and Silicon Controlled Rectifier (SCR) units. Upon exceeding the predetermined safe voltage, the zener diode senses the excess voltage and activates the appropriate SCR which shorts the fault voltage to ground or between the lines, whichever is the case. If the power in the fault voltage is of a significant nature, it will cause the fuse to blow out, which will result in complete circuit isolation.

The qualification plan which is proposed for the fuse/Transzorb combination used as an isolation device consists of the following two steps:

1) A Maximum Credible Fault Voltage test to prove that the components, when exposed to the maximum credible voltage, will protect the RM-80 such that it will continue proper operation. This test has been performed, and the results are available (UE&C Foreign Print #72046), and

2) A study to prove that the Transzorb and fuse have no age-related failures over the 40-year life of the plant.

The study will be as follows:

a. The Transzorb is a solid-state device with an activation energy of 1 ev. The manufacturer on a periodic basis samples test units to 150-200°C for 50 hours. By extrapolation on an Arrhenius curve using the activation energy and the test temperature and test time, the life of the device is several orders of magnitude greater than 40 years at normal operating conditions (40°C). Therefore, the Transzorb has no significant age-related failures.

b. A fuse is nothing more than a piece of wire which has no age-related failures which would cause it not to blow upon high current through it. There are no insulation materials in the device which would degrade with age.

420.13 (7.1.2.2) (7.5.3.3) (7.7.2.1) The discussion in Section 7.1.2.2 states that Westinghouse tests on the Series 7300 PCS system covered in WCAP-8892 are considered applicable to Seabrook. As a result of these tests, Westinghouse has stated that the isolator output cables will be allowed to be routed with cables carrying voltages not exceeding 580 volts ac or 250 volts dc. The discussion of isolation devices in Section 7.5.3.3 of the FSAR, however, considered the maximum credible fault accidents of 118 volts ac or 140 volts dc only. Also, the statement in Section 7.7.2.1 implies that the isolation devices were tested with 118 volts ac and 140 volts dc only. In order to clarify the apparent inconsistency, provide the following:

(a) Specify the type of isolation devices used for Seabrook process instrumentation system. If they are not the same as the Series 7300 PCS tested by Westinghouse, specify the fault voltages for which they are rated and provide the supporting test results.

(b) Provide information requested in (a) above for the isolation devices of the nuclear instrumentation system. As implied in WCAP-8892, the tests on Series 7300 PCS did not include the nuclear instrumentation system.

(c) Describe what steps are taken to insure that the maximum credible fault voltages which could be postulated in Seabrook, as a result of BOP cable routing design, will not exceed those for which the isolation devices are qualified.

RESPONSE: 3/23 The isolation devices used are as described in 420.12.

Isolation device design is identical and has been qualified the same as for SNUPPS. The routing of cables leaving the cabinets is consistent with the interface criteria in WCAP 8892A.

STATUS: 5/12 Closed.

420.14 (7.1.2.2) The FSAR information provided describing the separation criteria for instrument cabinets and the main control board is insufficient. Please discuss the separation criteria as it pertains to the design criteria of IEEE Standard 384-1977, Sections 5.6 and 5.7. Detailed drawings should be used to aid in verifying compliance with the separation criteria.

RESPONSE: 3/23 Handout submitted to staff. Overview of main control board was presented using drawings and pictures. FSAR Sections 7.1.2.2 and 1.8 will be revised to be applicable to both balance of plant and NSSS control panels. The design criteria of IEEE Standard 384-1977, Sections 5.6 and 5.7 for the main control board and instrument cabinets has been met.

STATUS: 9/14 Closed.

HANDOUT: 3/23

1. Instrument Cabinets

Section 5.7 of IEEE-384-1977 is met by having independent cabinets for redundant Class 1E instruments, examples of this separation may be found on instrument cabinets MM-CP-152A and MM-CP-152B, both located in the main control room, control building Elevation 75'-0".

2. Main Control Board (MCB)

Sections 5.6.1 through 5.6.6 of IEEE-384-1977 are met as follows, and as described in UE&C Specification 9763-006-170-1, Revision 5:

(a) Section 5.6.1 - The main control board, seismically qualified by analysis and testing per UE&C Specifications 9763-006-170-1 Revision 5, and 9763-SD-170-1, Revision 0, is located in the main control room of the Seabrook station control building (Elevation 75'-0") which is a Seismic Category I structure.

(b) Sections 5.6.2 through 5.6.6 - MCB Zone "B" (front contains the low pressure safety injection; rear contains miscellaneous systems like steam generator blowdown, heat removal, spent fuel) will be used to describe compliance with above referenced sections of IEEE-384-1977. UE&C drawings 9763-F-510102 Revision 6,

9763-F-510115 Revision 4 and 9763-F-510116 Revision 4 could be used to ascertain the compliance with the standard.

b.1 Internal Separation (5.6.2) - the front section of Zone B is divided into Class 1E train "A" (and it's associated non-Class 1E circuits train "AA") on the left-hand side, separated from the Class 1E train "B" (and it's associated non-Class 1E circuits train "BA") by a full size top-to-bottom steel barrier. However, due to process requirements there are instruments of the opposite train, "B", on the train "A" side; they are separated by a steel enclosure fully surrounding the instrument or open at the rear after a depth 6" deeper than the instrument itself.

The rear section of Zone B is all Class 1E train "A" or it's associated non-Class 1E circuit train "AA". Again, as in the front section due to process requirements, there are instruments of the opposite train which are separated by a steel enclosure in the same fashion as in the front section.

Refer to next Item, b.2, for wiring separation.

b.2 Internal Wiring Identification (5.6.3) - All wiring within each section is identified by different jacket colors, as follows:

Class 1E train "A" - red
Class 1E train "B" - white
Non-Class 1E train "AA" - black with red stripe
Non Class 1E train "BA" - black with white stripe

Each wire/cable insulation is qualified to be flame retardant per either IPCEA-S-19-81 (NEMA WC3) paragraph 6.13.2 or UL-44 Section 85 or IEEE Standard-383 Section 2.5. In addition, all wiring within each section is run in covered wireways formed from solid or punched sheet steel. Minimum wire bundles were allowed where it was physically impossible to install wireways or where it would have been hazardous to the operator/maintenance personnel.

Class 1E and Non-Class 1E wiring of the same train are run in the same wireway. The wireways were further identified with red "A" or white "B" to depict the train assignment of the wire being run within the particular wireway.

b.3 Common Terminations (5.6.4) - No common terminations were allowed in the MCB.

b.4 Non-Class 1E Wiring (5.6.5) - Class 1E and Non-Class 1E associated circuits wiring of the same train are run together in the same metallic wireway but are separated by specific identifying jacket colors as described above (b.2).

b.5 Cable Entrance (5.6.6) - Field cables to be terminated on the MCB terminal blocks are routed in train assigned raceways through the cable spreading room which is located directly under the main control room (refer to UE&C Drawing 9763-F-500091, Revision 6). The raceways run all the way up to the floor slots of the same assigned train located in the floor right underneath the MCB. (The floor slots location and train assignment are shown on UE&C Drawings

9763-F-500100 Revision 6, 9763-F-101347 Revision 5 and 9763-F-310432 Revision 8).

420.15 (7.1) Identify all plant safety-related systems, or portions thereof, for which the design is incomplete at this time.

RESPONSE: 3/23 The design of all safety-related systems has been completed. The design details associated with procurement and installation are on-going in accordance with the project schedule.

STATUS: 5/12 Closed (design modifications are being covered under the other RAIs).

420.16 (7.1) Identify where microprocessors, multiplexers, or computer systems are used in or interface with safety-related systems.

RESPONSE: 3/23 NSSS does not use microprocessors, multiplexers or computers in or to interface with safety-related systems (multiplexors are used for information transmission).

The radiation data management uses microprocessors and computers. Detailed descriptions on how the system works will be submitted later.

ADDITIONAL RESPONSE: 5/12 The RDMS is functionally identical to the systems installed at Byron-Braidwood, St. Lucie 2, Waterford 3, SNUPPS and Comanche Peak.

NRC will review handout presented, copy attached. More information is needed on the 1E microprocessor software and design features.

The Class 1E monitors are identified in FSAR Tables 12.3-13, 12.3-14 and 12.3-15. They are described in Section 12.3.4.

ADDITIONAL RESPONSE: 9/14 Software design control and testing was discussed. The controls will be documented. Information on the testing will be provided.

STATUS: 9/14 Open.

ADDITIONAL RESPONSE: 11/82 A description of the Radiation Data Management System (RDMS) and its major functional components has been previously submitted to the NRC.

Verification of monitor software performance is accomplished via functional testing of the performance as demonstrated in the vendors' acceptance test procedure (a typical test procedure is UE&C Foreign Print #72797). In addition, verification of the monitor response to radiation sources is accomplished via an acceptance test and transfer calibration procedure (UE&C Foreign Print #72761).

Documentation and tracking of software versions for the RM-80 microcomputer is accomplished via a multistep method which is detailed below:

Software Documentation Procedure

1. System Requirements and Design Basis

   The System Data Base Document and Block Diagram reflect the customer's specification requirements. These drawings define the functional software task. Changes to these drawings are controlled via engineering change orders.

2. RM-80 Software Design Basis

   The Software Design Task is defined by the Software Design Basis Document. It is developed by an iterative process that includes coding, checkout, reviews, and debugging. It is the design specification for the software. Changes to this document are controlled by Engineering Change Orders (ECO).

3. Testing of RM-80 Software

   After the design related debugging, reviewing and testing steps are finished, a generic software test is performed according to an approved test procedure. Changes to the test procedure are ECO controlled.

4. Final Design Review

   A Final Design Review is held. Minutes of this design review and all other reviews are maintained in the corresponding software design file.

5. Software Release

A RM-80 Software Checklist is completed to insure all the proper steps have been followed.

6. Software Documentation

The software is controlled by the GA software librarian who assures conformance to the documentation control specified in the GA Quality Assurance Manual.

420.17 (7.1) (7.2) (7.3) (1.8) The FSAR information which discusses conformance to Regulatory Guide 1.118 and IEEE-338 is insufficient. Further discussion is required. As a minimum, provide the following information:

1. Confirm that the Technical Specifications will provide detailed requirements for the operator which insure that blocking of a selected protection function actuator circuit is returned to normal operation after testing.

2. Discuss response time testing of BOP and NSSS protection systems using the design criteria described in Position C.12 or Regulatory Guide 1.118 and Section 6.3.4 of IEEE 338. Confirm that the response time testing will be provided in the Technical Specifications.

3. The FSAR states that, "Temporary jumper wires, temporary test instrumentation, the removal of fuses and other equipment not hard-wired into the protection system will be used where applicable". Identify where procedures require such operation. Provide further discussion to describe how the Seabrook test procedures for the protection systems conform to Regulatory Guide 1.118 (Revision 1) Position C.14 guidelines. Identify and justify any exceptions.

4. Confirm that the Technical Specifications will include the RPS and ESFAS response times for reactor trip functions.

5. Confirm that the Technical Specifications will include response time testing of all protection system components, from the sensor to operation of the final actuation device.

6. Provide an example and description of a typical response time test.

RESPONSE: 3/23 Handout was distributed and found acceptable with changes discussed during meeting. The revised handout is included in the meeting minutes.

STATUS: 9/14 Confirmatory pending correction of an editorial error to show that the correct revision is Revision 2, dated June 1978.

ADDITIONAL RESPONSE: 11/82 The comparison to Regulatory Guide 1.118 has been changed back to Revision 1, see the 3/23 Handout, to be consistent with the commitment to IEEE 338-1975 made in the PSAR.

HANDOUT: 3/23

1. Technical Specification Tables 3.3-1 reactor trip system, 3.3-3 engineered safety features actuation, and 3.3-5 reactor trip/ESF actuation system interlocks, provide the operator with the minimum operable channel criteria and the appropriate action statement.

2. BOP and NSSS protection system time response tests will be conducted in accordance with Regulatory Guide 1.118 Revision 1, IEEE-338-1975, ISA dS67-06, and draft Regulatory Guide Task IC 121-5, January, 1982, with the following exceptions and positions:

   (a) Task IC 121-5 Regulatory Position C1 states that the term "nuclear safety-related instrument channels in nuclear power plants" should be understood to mean instrument channels in protection systems.

   (b) Response time testing will be performed only on those channels having a limiting response time established and credited in the safety analysis.

   (c) The revised discussion of Regulatory Guide 1.118 in FSAR Section 1.8 (copy attached).

   Response time testing is specified in Tables 3.3-2 and 3.3-4.

3. It is not anticipated that any Seabrook test procedures performed on protection systems will require the use of temporary jumpers, lifted wires or pulled fuses. All procedures will, in fact, utilize the hard-wired test points within the system and therefore, comply with Regulatory Guide 1.118, Revision 1, Position C14.

   If during plant operation, conditions or test requirements show that deviation from this guide is the only practical method of obtaining the desired test results, then all affected testing will be performed and documented under the control of a special test procedure. We will inform ICSB, prior to licensing, of any temporary modifications identified during preparation of the surveillance procedures.

4. Response times are specified in Tables 3.3-2 and 3.3-4.

5. Compliance with Regulatory Guide 1.118, Revision 1, IEEE-338-1975, and ISA dS67-06 ensures that the complete channel is tested with the exception noted on Table 3.3-2 of Seabrook Technical Specifications.

6. Response time tests have not yet been prepared. Test methods to be employed are outlined below:

   Pressure Sensors

   The process variable will be substituted by a hydraulic ramp, the ramp rate to be selected based on the transient for which the sensor is required to respond.

In the event that the sensor is required to respond to more than one transient, the ramp rates will be selected to represent the fastest and slowest transients.

Temperature Sensors

Will be tested in place using the loop current step response (LCSR) method. See NUREG-0809.

Impulse Lines

Tests will be conducted during the startup testing phase to establish the relationship between response time and impulse line flow, subsequent tests will be limited to flow testing.

Electronic Channel

The signal conditioning and logic section of the instrument channel will be tested by inputting a step change at the input of the process racks, and measuring the time required until the final device in the channel actuates.

420.18 (7.1.2.11) It is stated in FSAR Section 7.1.2.11 that, "A periodic verification test program for sensors within the Westinghouse scope for determining any deterioration of installed sensor's response time, is being sought". NUREG-0809, "Review of Resistance Temperature Detector Time Response Characteristics", and draft Standard ISA-dS67.06, "Response Time Testing of Nuclear Safety-Related Instrument Channels in Nuclear Power Plants", are documents which propose acceptable methods for response time testing nuclear safety-related instrument channels. Please provide further discussion on this matter to unequivocally indicate the test methods to be used for Seabrook.

RESPONSE: 3/23 See our Response to 420.17 for a discussion of the proposed response time testing program. The referenced portion of 7.1.2.11 will be deleted (see attached copy).

STATUS: 9/14 Closed.

420.19 (7.1.1.1) FSAR Section 7.1.1 does not provide sufficient information to distinguish between those systems designed and built by the nuclear steam system supplier and those designed or built by others. Please provide more detailed information.

RESPONSE: 3/23 Draft revision of FSAR 7.1.1 provided to staff and found acceptable and is attached to the meeting notes.

STATUS: 9/14 Closed.

420.20 (7.1.2.7) Section 7.1.2.7 of the FSAR discusses conformance to Regulatory Guide 1.53 and IEEE Standard 379-1972. The information provided addresses only Westinghouse provided equipment and associated topical reports. Provide a conformance discussion that addresses the BOP portions of the plant safety systems and auxiliary systems required for support of safety systems.

RESPONSE: 3/23 FSAR has been revised to cover single failure criteria for BOP and NSSS and is attached to the meeting minutes.

ADDITIONAL RESPONSE: 5/12 The change to FSAR 7.1.2.7 was reworded. Copy is attached.

STATUS: 9/14 Closed.

420.21 (7.2.1.1) The information in Section 7.2.1.1.b.6, "Reactor Trip on Turbine Trip", is insufficient. Please provide further design bases discussion on this subject per BTP ICSB 26 requirements. As a minimum you should:

1. 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).

2. Discuss how the routing within the non-seismic Category I turbine building is such that the effects of credible faults or failures in this area on these circuits will not challenge the reactor trip system and thus degrade the RTS performance. This should include a discussion of isolation devices.

3. Describe the power supply arrangement for the reactor trip on turbine trip circuitry.

4. Provide discussion on your proposal to use permissive P-9 (50% power).

5. Discuss the testing planned for the reactor trip on turbine trip circuitry.

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 reactor trip, emergency safeguards equipment such as auxiliary feedwater system and safety-grade interlocks. Verification should be provided to show that such sensors and circuits meet IEEE-279 and are seismically and environmentally qualified. Identify the testing or analyses performed which insures that failures of non-seismic structures, mountings, etc. will not cause failures which could interfere with the operation of any other portion of the protection system.

RESPONSE: 3/23

Add to the SNUPPS response to "Reactor Trip on Turbine Trip" that circuits and sensors used in a non-seismic structure are Class 1E and are run in separate conduits meeting Regulatory Guide 1.75 with the exception of seismic qualification. Hydraulic pressure and limit switches on the turbine stop valves are two examples. the response will be attached to the meeting minutes.

Permissive P-9 has an adjustable setpoint between 10 - 50%.

Reactor trip on turbine trip circuitry is testable at power.

The turbine impulse chamber pressure transmitters are Class 1E and routed as Class 1E, with the seismic exception.

There are no other safety-grade sensors routed through non-seismic areas. The only safety-related outputs in non-seismic areas are signal to close the feedwater control valves, close the condenser dump valves and trip the turbine generator. These circuits are designed as described above.

ADDITIONAL RESPONSE: 5/12

The handout was discussed and revised.

Each turbine stop valve is monitored by two independent switches.

STATUS: 7/15

Closed. ICSB will follow PSB review of separation per Regulatory Guide 1.75.

HANDOUT: 3/23 9/14

Revised SNUPPS Submittal

Evaluations indicate that the functional performance of the protection system would not be degraded by credible electrical faults such as opens and shorts in the circuits associated with reactor trip or the generation of the P-7 interlock. The contacts of redundant sensors on the steam stop valves and the trip fluid pressure system are connected through the grounded side of the ac supply circuits in the solid state protection system. A ground fault would therefore produce no fault current. Loss of signal caused by open circuits would produce either a partial or a full reactor trip. Faults on the first stage turbine pressure circuits would result in upscale, conservative, output for open circuits and a sustained current, limited by circuit resistance, for short circuits. Multiple failures imposed on these redundant circuits could potentially disable the P-13 interlock. In this event, the nuclear instrumentation power range signals would provide the P-7 safety interlock. Refer to Functional Diagram, Sheet 4 of Figure 7.2-1.

SSPS input circuits and sensors in non-seismic structures are Class 1E. The electrical and physical independence of the connecting cabling conforms to Regulatory Guide 1.75.

STATUS: 9/14

Closed.

420.22 (7.2.1.1) FSAR Section 7.2.1.1.b.8 states that, "The manual trip consists of two switches with two outputs on each switch. One output is used to actuate the train A reactor trip breaker, the other output actuates the train B reactor trip breaker." Please describe how this design satisfies the single failure criterion and separation requirements for redundant trains.

RESPONSE: 3/23 Manual trip design is identical to SNUPPS, Watts Bar, Byron-Braidwood. Drawing was reviewed and found acceptable.

STATUS: 5/12 Closed.

420.23 (7.2) Describe how the effects of high temperatures in reference legs of steam generator and pressurizer water level measuring instruments subsequent to high energy breaks are evaluated and compensated for in determining setpoints. Identify and describe any modifications planned or taken in response to IEB 79-21. Also, describe the level measurement errors due to environmental temperature effects on other level instruments using reference legs.

RESPONSE: 3/23 The steam generator level transmitter reference legs will be insulated to prevent excessive heating under accident conditions. Setpoints will include errors for high energy line breaks with the insulation.

For the pressurizer level, we will review SNUPPS report and determine applicability to Seabrook.

REVISED RESPONSE: 5/12 SNUPPS did not insulate reference legs in containment. We are evaluating their approach for application to Seabrook and will advise the NRC on our final corrective action.

STATUS: 9/14 Open. Evaluation of transient heating of steam generator reference leg continues. A complete response will be submitted to the NRC.

420.24 (7.2) (7.3) (7.4) (7.6) State whether all of the systems discussed in Sections 7.2, 7.3, 7.4 and 7.6 of the FSAR conform to the recommendations of Regulatory Guide 1.62 concerning manual initiation. Identify any exceptions and discuss how they do not conform to the recommendations. Provide justification for nonconformance areas.

RESPONSE: 3/23 Systems discussed in Sections 7.2, 7.3, 7.4 and 7.6 of the FSAR conform to the recommendations of Regulatory Guide 1.62 concerning manual initiation. There are no exceptions taken.

STATUS: 5/12 Closed.

420.25 (7.2.2.2) The information provided in Section 7.2.2.2.c.10.(b) on testing of the power range channels of the nuclear instrumentation system, covers only the testing of the high neutron flux trips. Testing

of the high neutron flux rate trips is not included. Provide a description of how the flux rate circuitry is tested periodically to verify its performance capability.

RESPONSE: 3/23 The power range nuclear instrumentation system and all associated bistables including the rate trips are testable at power.

STATUS: 5/12 Closed.

420.26 (7.2) (7.3) Identify 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. The intent of this item is to verify that a single failure in a common instrument line or tap (such as break or blockage) cannot defeat required protection system redundancy.

RESPONSE: 3/23 Identical to SNUPPS except we do not share taps for pressurizer pressure. There are no shared taps for redundant BOP safety instruments.

STATUS: 5/12 Closed.

420.27 (7.3) If safety equipment does not remain in its emergency mode upon reset of an engineered safeguards actuation signal, system modification, design change or other corrective action should be planned to assure that protective action of the affected equipment is not compromised once the associated actuation signal is reset. This issue is addressed by I&E Bulletin 80-06. Please provide a discussion addressing the concerns of the above bulletin. This discussion should assure that you have reviewed the Seabrook design per each of the I&E Bulletin 80-06 concerns. Results of your review should be given.

RESPONSE: 3/23 We have reviewed the electrical schematics for engineered safety feature (ESF) reset controls. In the Seabrook design, all systems serving safety-related functions remain in the emergency mode upon removal of the actuating signal and/or manual resetting of ESF actuation signals. The required testing (per 80-06) will be performed as part of the start-up test program described in Chapter 14.

STATUS: 5/12 Closed.

420.28 (7.3.1.1) The description of the emergency safety feature systems which is provided in the FSAR Section 7.3.1.1 is incomplete in that it does not provide all of the information which is requested in Section 7.3.1 of the standard format for those safety-related systems, interfaces and components which are supplied by the applicant and mate with the systems which are within the Westinghouse scope of supply. Provide all of the descriptive and design basis information which is requested in the standard format for these systems. In addition, provide the results of an analysis, as is

requested in Section 7.3.2 of the standard format, which demonstrates how the requirements of the general design criteria and IEEE Standard 279-1971 are satisfied and the extent to which the recommendations of the applicable Regulatory Guide are satisfied. Identify and justify any exceptions.

RESPONSE: 3/23 Tables supplied in response to 420.32 and the additional information to be supplied when answering 420.29 will satisfy the requirements of this question.

ADDITIONAL RESPONSE: 5/12 See 420.29.

STATUS: 7/15 Closed.

420.29 (7.3.2.1) Confirm that the FMEA referenced in FSAR Section 7.3.2.1: (1) is applicable to all engineered safety features equipment within the BOP and NSSS scope of supply, and (2) is applicable to design changes subsequent to the design analyzed in the referenced WCAP.

RESPONSE: 3/23 Discussion of this item was deferred to the next meeting.

ADDITIONAL RESPONSE: (28&29) 5/12 The Seabrook design complies with the interface criteria in Appendix B of WCAP 8584, Revision 1. The FMEA in WCAP 8584 is applicable to all BOP and NSSS safety features equipment at Seabrook including design changes made to the systems analyzed in WCAP 8584.

STATUS: 7/15 Closed.

420.30 (7.3) Section 7.3.2.2 of the FSAR indicates that conformance to Regulatory Guide 1.22 is discussed in Section 7.1.2.8. However, Section 7.1.2.8 addresses Regulatory Guide 1.63. Correct this discrepancy.

RESPONSE: 3/23 The reference to Section 7.1.2.8 will be changed in Amendment 45 to Section 7.1.2.5 where Regulatory Guide 1.22 is addressed.

STATUS: 9/14 Closed.

420.31 (7.3.2.2) Using detailed drawings, discuss the automatic and manual operation of the containment spray system including control of the chemical additive system. Discuss how testing of the containment spray system conforms to the recommendations of Regulatory Guide 1.22 and the requirements of BTB ICSB 22. Include in your discussion the tests to be performed for the final actuation devices.

RESPONSE: 3/23 Draft of response submitted to staff. Overview of containment spray system was presented using drawings. System description and operation were reviewed. Staff questioned redundancy of temperature system. Tank temperature is monitored by a temperature indicating switch that actuates a VAS alarm and by an independent temperature indicating controller that controls auxiliary steam to the tank. Fluid systems are totally separable into trains "A" and "B". The electrical systems are also completely separable into trains "A" and "B" as per the piping systems. Provisions are available for on-line testing of CBS system as described in FSAR 7.3.2.2.

The assignment of components to slave relays for on-line testing is indicated in the ESF table in the response to 420.32.

ADDITIONAL RESPONSE: 5/12 The response was clarified to specify that the spray additive tank is the tank being discussed.

This item is considered closed.

STATUS: 5/12 Closed.

420.32 (7.3) Please provide a table(s) listing the components actuated by the engineered safety features actuation system. As a minimum, the table should include:

1. Action required,

2. Component description,

3. Identification number,

4. Actuation signal and channel.

RESPONSE: 3/23 Tables supplied at the meeting are attached.

STATUS: 5/12 Closed.

420.33 (7.3.2.2) Section 7.3.2.2.e.12 discusses testing during shutdown. Describe provisions for insuring that the "isolation valves" discussed here are returned to their normal operating positions after test.

RESPONSE: 3/23 Administrative controls to ensure that equipment and systems are restored to normal after testing will be addressed in equipment control procedures that follow the guidance of ANS 18.7, 1976. The system inoperative status monitoring panel will be manually actuated when a system is made inoperative.

STATUS: 5/12 Closed.

420.34 (7.3) Portions of paragraph 7.3.1.2.f, appear not to apply to ESFAS response times. In particular, the discussion on reactor trip breakers, latching mechanisms, etc., should be replaced by a discussion of ESF equipment time responses. The applicant should provide a revised discussion for ESFAS (a) defining specific beginning and end points for which the quoted times apply, and (b) relating these times to the total delay for all equipment and to the accident analysis requirements.

RESPONSE: 3/23 FSAR 7.3.1.2.f will be revised as indicated on the attached markup.

STATUS: 9/14 Closed.

420.35 (7.2 & 7.4) Using detailed drawings, describe the ventilation systems used to support engineered safety features areas including areas containing systems required for safety shutdown. Discuss the design bases for these systems including redundancy, testability, etc.

RESPONSE: 3/23 Overview given at meeting on HVAC system for control room. Equipment for system is redundant and safety grade. The HVAC instrumentation and control required for safety-related equipment is Class 1E and trains "A" and "B" oriented. Radiation detectors for intake air are redundant and safety related. Other systems in the control building are redundant and safety related.

Control of safety-related HVAC systems are operated from the control room and those systems required for remote safe shutdown also have local control. The control room outside air intake lines are shared between Units 1 and 2. Each unit has its own controls and isolation valves.

STATUS: 5/12 Closed.

420.36 (7.3.2.3) Using detailed system schematics, describe how the Seabrook auxiliary feedwater system meets the requirements of NUREG-0737, TMI Action Plan Item II.E.1.2 (See question 420.01). Be sure to include the following information in the discussion:

a) the effects of all switch positions on system operation.

b) the effects of single power supply failures including the effect of a power supply failure on auxiliary feedwater control after automatic initiation circuits have been reset in a post-accident sequence.

c) any bypasses within the system including the means by which it is insured that the bypasses are removed.

d) initiation and annunciation of any interlocks or automatic isolations that could degrade system capability.

e) the safety classification and design criteria for any air systems required by the auxiliary feedwater system. This should include the design bases for the capacity of air reservoirs required for system operation.

f) design features provided to terminate auxiliary feedwater flow to a steam generator affected by either a steam line or feed line break.

g) system features associated with shutdown from outside the control room.

RESPONSE: 3/23 Overview of emergency feedwater system was presented to staff using drawings for description of system operation.

Emergency feedwater system was discussed with staff and it is considered an open item. Significant concerns identified:

a) Lack of safety-grade air system.

b) Single failure in pneumatic control valve.

c) Loss of one train of power while operating from remote safe shutdown panel.

d) On-off control of the EFW control valves.

ADDITIONAL RESPONSE: 9/14 The concerns expressed in this RAI and in the letter, dated April 22, 1982 (Items A - K), were discussed in meetings with ICSB, ASB, RSB, YAEC, PSNH, and UE&C on June 23 and 24 and July 14 and 15, 1982. Our letter SBN-300, dated July 27, 1982, provided response to your April 22 letter. Our letter SBN-321, dated September 7, 1982, described the changes that are being made to the emergency feedwater system. A draft copy of the revision to FSAR Section 6.8 reflecting these changes is attached.

STATUS: 9/14 Confirmatory pending ASB review of recirc line modification and ICSB review of the formal documentation.

420.37 (7.3) Using detailed system schematics, describe the sequence for periodic testing of the:

a) main steam line isolation valves

b) main feedwater control valves

c) main feedwater isolation valves

d) auxiliary feedwater system

e) steam generator relief valves

f) pressurizer PORV

The discussion should include features used to insure the availability of the safety function during test and measures taken to insure that equipment cannot be left in a bypassed condition after test completion.

RESPONSE: 3/23 Periodic testing was discussed using detailed drawings. Significant discussion items are:

a) To be presented at next meeting.

b) Standard Westinghouse testing system used.

c) When testing main feedwater control and main feedwater isolation valves using train "A", the system for train "B" remains completely operable.

d) During testing of emergency feedwater pumps the discharge valve is closed and recirculation valve opened. The system inoperable indication is in accordance with Regulatory Guide 1.47.

During testing, the capability exists to test the entire ESFAS as including actuation of the EFW pump.

e) Discussed with no comments.

f) Discussed with no comments.

ADDITIONAL RESPONSE: 9/14 The MSIV logic has been redesigned so that periodic testing can be performed during normal power operation as a series of overlapping tests. Since the MSIVs cannot be fully closed at power, the actuation logic is blocked by a signal from the solid state protection system (SSPS) test cabinet when the test relay is energized. Operation of the slave relay and the test switch actuates the isolation logic. Proper operation of the logic is indicated at the logic gate that has been blocked.

After the SSPS is returned to normal, the MSIV is exercised by partial stroke closure at a reduced speed. The exercise signal overlaps the actuation test to verify the operability of the complete logic.

The restoration of the flow restrictor after the exercise test is monitored.

STATUS: 9/14 Confirmatory pending review of formal documentation.

420.38 (7.4.1) The information supplied in FSAR Section 7.4.1 does not adequately describe the systems required for safe shutdown as required by Section 7.4.1 of the standard format. Therefore, provide all the descriptive and design basis information which is requested by Section 7.4.1 of the standard format. Also, provide the results of an analysis, as requested by Section 7.4.2 of the standard

format, which demonstrates how the requirements of the general design criteria and IEEE Std. 279-1971 are satisfied and the extent to which the recommendations of the applicable regulatory guides are satisfied. Identify and justify any exceptions.

RESPONSE: 3/23 Staff to review handouts presented at this meeting and come back with any further questions. Update list for 420.39 and submit with minutes. YAEC given written position on safe shutdown, to be forwarded formally. Rewritten FSAR 7.4 is attached.

ADDITIONAL RESPONSE: 5/12 The analog instruments associated with the remote shutdown panel are Non-1E and are independent of the control room instruments.

The controls at the remote shutdown locations have the same qualification as the controls at the main control board.

REVISED RESPONSE: 9/14 The design of the controls at the remote shutdown locations have undergone considerable revision to comply with the requirements of Appendix R and to be consistent with the changes required for safety grade cold shutdown from the control room.

Since the same safety grade equipment will be used for remote shutdown without a fire, all the associated controls at the remote shutdown locations are safety grade and meet the applicable requirements of IEEE 279-1971, 324-1974, and 344-1975.

The instrumentation at the remote locations (with the exception of the wide range nuclear instrumentation) are separate loops that are completely independent of the instrument loops that provide indication in the control room. Since the remote shutdown locations are not required to have the controls and indication necessary to control the plant during accidents, the instrumentation at the remote shutdown locations do not meet all the requirements for safety grade equipment. We have determined that the electronics and indicators at the remote shutdown panels (CP-108 A & B) and the field wiring do meet the requirements of IEEE 344-1975. The transmitters and indicators are mechanically similar to transmitters and indicators that are qualified to 344-1975. We are obtaining the necessary documentation to certify that the transmitters and indicators will be operable following a seismic event. We will be able to certify that the instruments at the remote shutdown panels will be available following all postulated natural phenomena and, therefore, will meet the design basis of the remote shutdown equipment. This documentation will be available for audit prior to fuel loading.

The design for the safety grade wide-range nuclear instrumentation has the electronics mounted such that they would not be affected by a fire in the control room cable spreading room. The indication that will be provided at the remote shutdown location will be safety grade. We are reviewing a conflict between our Appendix R response (de-energization of the SSPS) and the ICSB guidance to meet Appendix K (do not disable ESF actuation prior to cooldown). We will provide our position on this item.

The draft revision to FSAR 7.4 submitted with the March 23, 1982, meeting minutes is being revised to reflect the latest design of the remote shutdown equipment and will address the positions in your April 21, 1982 letter, item-by-item.

STATUS: 9/14 Compliance with the Appendix K guidelines remains open, the remainder is confirmatory.

420.39 The information supplied for remote shutdown from outside the control room is insufficient. Therefore, provide further discussion to describe the capability of achieving hot or cold shutdown from outside the control room. As a minimum, provide the following information:

a. Provide a table listing the controls and display instrumentation required for hot and cold shutdown from outside the control room. Identify the safety classification and train assignments for the safety-related equipment.

b. Design basis for selection of instrumentation and control equipment on the hot shutdown panel.

c. Location of transfer switches and remote control station (include layout drawings, etc.).

d. Design criteria for the remote control station equipment including transfer switches.

e. Description of distinct control features to both restrict and to assure access, when necessary, to the displays and controls located outside the control room.

f. Discuss the testing to be performed during plant operation to verify the capability of maintaining the plant in a safe shutdown condition from outside the control room.

g. Description of isolation, separation and transfer/override provisions. This should include the design basis for preventing electrical interaction between the control room and remote shutdown equipment.

h. Description of any communication systems required to coordinate operator actions, including redundancy and separation.

i. Description of control room annunciation of remote control or overridden status of devices under local control.

j. Means for ensuring that cold shutdown can be accomplished.

k. Explain the footnote in FSAR Section 7.4.1.4 which states that, "Instrumentation and controls for these systems may require some modification in order that their functions may be performed from outside the control room". Discuss the modifications required on the instrumentation and controls of

the pressurizer pressure control including opening control for pressurizer relief valves, heaters and spray and the nuclear instrumentation that are necessary to shutdown the plant from outside the control room. Also discuss the means of defeating the safety injection signal trip circuit and closing the accumulator isolation valves when achieving cold shutdown.

RESPONSE: 3/23 See 420.38.

ADDITIONAL RESPONSE: 5/12 We will investigate the absence of pressurizer level indication in the table that was provided in response to Item a.

Response to Item g should refer to 7.4.1.1 and 7.4.1.3.a.5 vice 7.4.11.

See 420.36.

HANDOUT: 3/23 5/12

a) Table is attached.

b) See response to Item 440.13 (attached).

c) Transfer switches are at the same location as the controls.

d) Controls are the same safety classification as the controls in the control room. Instrumentation is not safety-related.

e) The controls are located in areas that are controlled by the security system. The transfer switches are key-locked.

f) Verification of the capability of maintaining the plant in a safe shutdown condition from outside control room will be in accordance with commitment in Chapter 14, Table 14.2-5, Item 33. Reactor coolant pumps will not be tripped for this test. Verification of natural circulation will be in accordance with commitment in Chapter 14, Table 14.2-5, Item 22.

g) Isolation is discussed in FSAR 7.4.1.1 and 7.4.1.3.a.5.

h) See response to 430.67 (attached).

i) Any switch that is in the local position is alarmed by the VAS.

j) See Items a and b.

k) The footnote has been deleted. See rewritten 7.4 submitted in 420.38.

ADDITIONAL RESPONSE: 9/14

a) A revised table will be attached to the meeting minutes.

b) Item-by-item compliance with RSB BTP 5-1 will be documented in our response to RAI 440.133.

d) See 420.38 for the design of instrumentation.

e) The remote shutdown locations are the vital switchgear rooms on elevation 21' 6", tow level directly below the control room on elevation 75. Access is through the stairwell on the south side of the control building or through stairwells in the turbine building.

Access to all levels of the control building is controlled by the station security system. The operators key cards will allow access to all levels of the control building. Administered controlled keys are also available to assure access should the security system be inoperable.

i) VAS will be reviewed under 420.49.

STATUS: 9/15 Confirmatory, closely related to 420.38.

420.40 Concerning safe shutdown from outside the control room, discuss the likelihood that the auxiliary feedwater system will be automatically initiated on low-low steam generator level following a manual reactor trip and describe the capability of resetting the initiating logic from outside the control room. Describe the method of controlling auxiliary feedwater from outside the control room.

RESPONSE: 3/23 Even though the emergency feedwater system may be automatically initiated as the main control room is evacuated, the emergency feedwater system can be controlled from the remote safe shutdown panel without resetting the actuation logic. Additional information required by staff is furnished in the response to 420.38 and 420.39.

STATUS: 9/14 Closed.

420.41 (7.4.2) Subsection 7.4.2 states that, "The results of the analysis which determined the applicability to the Nuclear Steam Supply System safe shutdown systems of the NRC General Design Criteria, IEEE Standard 279-1971, applicable NRC Regulatory Guides and other industry standards are presented in Table 7.1-1". This statement does not address the balance of plant (BOP) safe shutdown systems. Also, sufficient information giving results of the analysis performed for safe shutdown systems cannot be found from Table 7.1-1. Therefore, provide the results and a detailed discussion of how the BOP and NSSS systems required for safe shutdown meet GDCs 13, 19, 34, 35, and 38; IEEE Standard 279 requirements; Regulatory Guides 1.22, 1.47, 1.53, 1.68, and 1.75. Be sure that you include a discussion of how the remote shutdown station complies with the above design criteria.

RESPONSE: 3/23 Closely related to Items 38 and 39. Staff will review to see if more response is required.

ADDITIONAL RESPONSE: 9/14 Table 7.1-1 will be revised to include the GDCs, Standards, and Regulatory Guides listed as being applicable to Section 7.4. A draft revision of Table 7.1-1 is attached.

STATUS: 9/14 Confirmatory.

420.42 (7.4.2) FSAR Section 7.4.2 states that, "It is shown by these analyses, that safety is not adversely affected by these incidents, with the associated assumptions being that the instrumentation and controls indicated in Subsections 7.4.1.1 and 7.4.1.2 are available to control and/or monitor shutdown". Please provide a discussion pertaining to the phrase "associated assumptions". Your discussion should address loss of off-site power associated with plant load rejection or turbine trip.

RESPONSE: 3/23 Covered in the response to 420.38.

ADDITIONAL RESPONSE: 9/14 The phrase "associated assumptions" will be deleted. Loss of off-site power will be addressed in the revised 7.4 (see 420.38).

STATUS: 9/14 Confirmatory.

420.43 (7.4.2) Please discuss how a single failure within the station service water system and/or the primary component cooling water system affects safe shutdown.

RESPONSE: 3/23 Each of the independent and redundant flow trains of the station service water system and the primary component cooling water system is capable of performing their safety functions necessary to effect a safe shutdown assuming a single failure. See Sections 9.2.1, 9.2.2 and 9.2.5 for further details.

STATUS: 5/12 Closed.

420.44 (9.2.5.5) Using detailed electrical schematics and logic diagrams, discuss the tower actuation (TA) signal which is generated to isolate the normal service water system and initiate the cooling tower system. Be sure to include in your discussion the possibilities of inadvertent switchover (loss of off-site power, etc.) and the affects this would have.

RESPONSE: 3/23 The tower actuation circuit is being revised. The revised drawings will be submitted for review.

ADDITIONAL RESPONSE: 9/14 The TA actuation logic is being revised to correct deficiencies in the logic and to provide the design features described in 420.73. Latch relays are now used that require a signal to actuate and another signal to reset. Loss of off-site power or loss of power to the TA circuit will not cause inadvertent actuation. The redundant cooling tower train will provide the service water function if one cooling tower train does not actuate. FSAR 9.2.5.5 will be revised, marked-up copy is attached.

STATUS: 9/14 Confirmatory pending review of formal documentation.

420.45 (7.4.2) FSAR Section 7.4.2 states that, "Loss of plant air systems will not inhibit ability to reach safe shutdown from outside the control room". Using detailed drawings, please provide further discussion on this matter. Clearly indicate any function required to reach safe shutdown from outside the control room which is dependent on air and the means by which the air is provided.

RESPONSE: 3/23 Instrument air system is redundant, piping is safety grade and seismically supported but appropriate safety-grade compressor has not been located. Critical to define how long system can operate from accumulator tanks. Staff questioned atmospheric relief valve

as to safety classification - valve itself is safety grade but control system is not. This item is still open.

REVISED RESPONSE: 9/14 Instrument air is no longer required for safe shutdown as the emergency feedwater control valves and the atmospheric dump valves no longer have pneumatic operators and the RHR system can be operated without the use of instrument air.

STATUS: 9/14 Confirmatory.

420.46 (7.4) Describe the procedures to borate the primary coolant from outside the control room when the main control room is inaccessible. How much time is there to do this?

RESPONSE: 3/23 Handout given to NRC. Staff questioned if MOV's and controls mentioned are safety grade. Items are safety grade. If problem exists during review, it will be covered under overall discussion of shutdown. "Adequate time" mentioned in response is minimum of four hours.

STATUS: 9/14 This issue was discussed at the June 23 and 24, 1982, meeting, and is closed.

HANDOUT: 3/23 Boration of the primary coolant will require an alignment of the suction of charging pumps from the refueling water storage tank (RWST) to the boric acid storage tank (BAST). This will be required once the plant starts its cooldown. The gravity feed from the BAST to the suction of the charging pumps contains manual

isolation valves located in the primary auxiliary building. The RWST suction valves contain motor-operated valves (MOV) that can be controlled from the motor control center in the switchgear. If need be, the MOV's can be operated locally. There is adequate time for an operator to follow the procedure since the plant is in a safe hot shutdown condition.

420.47 (7.4) Using detailed drawings (schematics, P&IDs'), describe the automatic and manual operation and control of the atmospheric relief valves. Describe how the design complies with the requirements of IEEE-279 (i.e., testability, single failure, redundancy, indication of operability, direct valve position, indication in control room, etc.).

RESPONSE: 3/23 Operation of these valves from a remote location is not considered a safety-related function; therefore, they are not designed to meet IEEE-279. Overview of operation given at meeting. Item still under review by staff and considered open.

REVISED RESPONSE: 9/14 The operators for the atmospheric dump valves are being changed to safety grade operators that will comply with the requirements of IEEE 323-1974 and 344-1975. Safety grade manual control will be provided and will override the non-1E automatic controls. The preliminary design was discussed.

STATUS: 9/14 Confirmatory pending review of formal documentation.

420.48 (7.4.2) (7.3) Using detailed electrical schematics and piping diagrams, please discuss the automatic and manual operation and control of the station service water system and the component cooling water system. Be sure to discuss interlocks, automatic switchover, testability, single failure, channel independence, indication of operability, isolation functions, etc.

RESPONSE: 3/23 Reviewed system design and operation from drawings and schematics. Staff will review isolation of non-seismic portion of service water system during earthquake without another accident.

ADDITIONAL RESPONSE: 5/12 Low service water pump discharge pressure (could be the result of tunnel blockage due to an earthquake) will result in tower actuation (TA). The TA signal will isolate the non-seismic portion of the SW system.

ADDITIONAL RESPONSE: 9/14 An analysis was performed that shows that a complete failure of the non-seismic SW piping will reduce SW pump discharge pressure below the tower actuation setpoint. The non-seismic SW piping is isolated on tower actuation, safety injection and loss of off-site power (see revised 9.2.5.5 in response to 420.44).

STATUS: 9/14 Open pending ICSB review with ASB.

ADDITIONAL RESPONSE: 11/82 As was discussed in the 9/14 meeting, we have performed an analysis that shows that a complete failure of the safety service water piping will result in a tower actuation that will isolate the non-safety piping and restore flow to the safety users. It was also pointed out that the non-safety piping is isolated by a safety injection signal or a loss of off-site power. Since the isolation is performed for worst case break conditions and for the critical condition II, III and IV events the remaining concern relates to the effect of reduced flow to the safety users for failures of the non-safety piping smaller than the complete failure.

Reduced service water flow to the safety users, component cooling heat exchangers and the diesel generators, could affect the temperature of the component cooling water or the diesel generator cooling water. The degree of the effect is determined by the amount the flow is reduced, the service water temperature and the heat load on the systems being cooled.

A low service water pump discharge pressure alarm is provided at 35 psig (tower actuation is at 30 psig). High temperature alarms are provided at the discharge of the component cooling heat exchanger and on the components being cooled by component cooling water to alert the operator to the problem with heat removal caused by a reduction in SW flow. A similar but more rapid occurrence would be a mechanical failure of a component cooling water pump that results in the loss of all cooling water flow.

The operator action in all cases of reduced heat removal would be to verify component and service water flows/pressures and taken action necessary to restore flow. One of the primary means to restore or increase service water flow would be to start the standby service water pump if available. If this action proved inadequate, the next step would be to isolate the non-safety piping.

The diesel generator scenario is similar.

420.49 (7.5) The information supplied in FSAR Section 7.5 concentrates on the post accident monitoring instrumentation and does not provide sufficient information to describe safety related display instrumentation needed for all operating conditions. Therefore, please expand the FSAR to provide as a minimum additional information on the following:

1. ESF Systems Monitoring

2. ESF Support Systems Monitoring

3. Reactor Protective System Monitoring

4. Rod Position Indication System

5. Plant Process Display Instrumentation

6. Control Boards and Annunciators

7. Bypass and Inoperable Status Indication

8. Control Room Habitability Instrumentation

9. Residual Heat Removal Instrumentation

Please use drawings as necessary during your discussion.

RESPONSE: 3/23 All except Item 6 will be covered in response to Regulatory Guide 1.97. Summary of VAS and annunciator system will be provided.

ADDITIONAL RESPONSE: 5/12 Letter SBN-268, dated 5/4/82, forwarded additional information on the main plant computer system and the VAS.

The annunciators are standard lightboxes that respond to digital inputs. Power is supplied from inverters and the dc system. Audible alarms and controls are shared with the VAS.

The alarm sequence is:

	Condition	Operator Action	Visual	Alarm Audible	Ringback Audible
1.	Normal	-	Off	Off	Off
2.	Off Normal	-	Fast Flash	On	Off
3.	Off Normal	Silence	Fast Flash	Off	Off
4.	Off Normal	Acknowledge	Steady	Off	Off
5.	Normal	-	Slow Flash	Off	On (momentary)
6.	Normal	Reset	Off	Off	Off

The annunciator alarms are a subset of the VAS alarms and were selected to provide essential alarms if the VAS is inoperable. The alarm points are shown on Drawings 9763-C-509109 through 509114. Some VAS inputs are obtained from relays in the annunciator that duplicate the input to the annunciator. Failure of the VAS will not affect the annunciator.

FSAR 7.5 will be revised in our response to Regulatory Guide 1.97, Revision 2.

STATUS: 7/15 SBN-268 was discussed on 6/21/82 by NRC/PSNH/YAEC. Information was requested on software QA and security; control of alarm priority (criteria and method for assigning priorities); management functions; and the use as a Regulatory Guide 1.47 monitor (see RAI 420.10).

ADDITIONAL RESPONSE: 9/14

VAS Software QA and Security

1. The testing of the video alarm system (VAS) is being conducted as part of the startup test program in two phases. Phase 1 will be run after installation of the computer equipment at the plant site and will validate the functional operation of the VAS system. Tests will be run using projected worst case conditions derived from simulator data. Phase 2 will verify operation of individual computer inputs as plant systems are checked out.

2. Changes to the software after the Phase 1 testing has been completed will be controlled by procedure. This procedure, under control of the Station Plant Manager, will ensure that changes to the tested software are authorized and adequately tested before they are implemented. The change control procedure will require operator authorization to make the change, documentation of the change, retest of the affected system, and integration into the procedures and operator training as applicable.

3. The following operator change functions are under keylock and administrative procedure control:

   delete/restore a point from alarming
   delete/restore a group of points from alarming
   delete/restore a point from scan
   modify a point's alarm limits
   modify a point's engineering value

4. Procedures will be available for review three months prior to fuel loading:

VAS Alarm Priority

The Operations Group is in the process of reviewing the VAS alarms for priority, alarm message, point identification and destination. Their comments will be incorporated in the project documents. The following priority guidelines are being used:

Priority One - Immediate operator response required to:

A. Prevent plant shutdown.

B. Minimize the consequences of a shutdown.

Priority Two - Occurrence of alarm indicates a degradation of a major plant system that could result in plant shutdown, power reduction, or reduced availability of a safety system.

Priority Three - Occurrence of alarm indicates degradation of a system component or are informational items describing a change of state.

STATUS: 9/14 The VAS software response will be reviewed by the NRC and discussed during a conference call to be scheduled later. FSAR 7.5, 7.2.2.2 (13) and (20) are being revised to provide the additional information requested.

VAS

ADDITIONAL RESPONSE: 11/82 A telephone conversation was held on 9/27 with representatives of the NRC (R. Stevens, J. Joyce, J. Rosenthal), PSNH (G. Gelineau, D. Johnson), and YAEC (W. Fadden, R. Marie). The additional response, dated 9/14, was discussed in detail. Significant items of discussion were:

1. The VAS software was produced prior to implementation of formalized quality control procedures for production of software. THe VAS software requirements (functional description) were reviewed extensively by PSNH operations and YAEC (a summary of the development of the VAS software up to the installation of the computer at Seabrook is attached). Computer startup and preparations for Phase I and II testing is in progress.

2. The software change control procedure will be implemented prior to the start of the Phase II testing.

3. All procedures associated with software change control or testing will have an independent review performed.

4. Limited alarm suppression is employed, mainly associated with the status of specific equipment or supression of redundant alarms (see CBS logic diagrams M-503257 and M-503260).

5. The NRC expressed concern that system unreliability be identified and appropriate corrective action taken.

RESPONSE: The Seabrook computer will be maintained by the Computer Engineering Department that has expended considerable money and effort to establish the in-house resources required to provide prompt repair of the computer. As the com[illegible] provides many aids to the operating staff, its performance i[illegible]ly visible to station management. Any evidence of unacceptable performance will be promptly identified and corrected.

6. The main computer and the CPU at the remote locations, are provided with full capability backups that will automatically assume all functions on failure of the operating computer or CPU. Only data that changes state and returns to its original state during the less than 5 second transfer time will be lost.

7. Redundant I/O equipment is not available. Critical paraments are monitored by different IRTUs so that critical data will not be lost.

8. CRT functions can be manually transferred without loss of data to other CRTs on the MCB.

FSAR: Attached are draft copies of revised FSAR Section 7.2.2.2.c(13) and 7.5 that provide the additional information requested.

420.50 (7.5) If reactor controls and vital instruments derive power from common electrical distribution systems, the failure of such electrical distribution systems may result in an event requiring operator action concurrent with failure of important instrumentation upon which these operator actions should be based. IE Bulletin 79-27 addresses several concerns related to the above subject. You are requested to provide information and a discussion based on each IE Bulletin 79-27 concern. Also, you are to:

1. Confirm that all a.c. and d.c. instrument buses that could affect the ability to achieve a cold shutdown condition were reviewed. Identify these buses.

2. Confirm that all instrumentation and controls required by emergency shutdown procedures were considered in the review. Identify these instruments and controls at the system level of detail.

3. Confirm that clear, simple, unambiguous annunciation of loss of power is provided in the control room for each bus addressed in item 1 above. Identify any exceptions.

4. Confirm that the effect of loss of power to each load on each bus identified in item 1 above, including ability to reach cold shutdown, was considered in the review.

5. Confirm that the re-review of IE Circular No. 79-02 which is required by Action Item 3 of Bulletin 79-27 was extended to include both Class 1E and Non-Class 1E inverter supplied instrument or control buses. Identify these buses or confirm that they are included in the listing required by Item 1 above.

RESPONSE: 3/23 9/14 Refer to the attached response to IE Bulletin 79-27 and two attached responses to IE Circular 79-02.

1. All 1E and non-1E ac and dc instrument buses were reviewed. Refer to the listing of buses reviewed in the attached response to Bulletin 79-27.

2. Redundant instrumentation and controls required for safe shutdown are available at the control room and the remote shutdown location. Loss of an entire power train will not prevent the ability to accomplish cold shutdown with the control and indication powered by the other train.

3. Annunciation of loss of power is provided in the main control room through Seabrook video alarm system. The wording of all alarms is subject to review by the station operating staff to insure clarity.

4. See Item 2.

5. Refer to the two attached responses to Circular 79-02. The buses are listed in the response to Bulletin 79-27.

ADDITIONAL RESPONSE: 5/12

Item 1 was revised. We will clarify the reviews performed for Items 2 and 4. All required instrumentation and controls will be identified.

Our emergency procedures will contain the items requested by I&E Bulletin 79-27, Items 2.a, 2.b and 2.c.

We will provide additional information on our inverters as requested by I&C Circular 79-02 (time-delay, modifications).

ADDITIONAL RESPONSE: 7/15

Item 1 was revised. The NRC clarified the additional information requested in Items 2 and 4. A handout on inverters was reviewed and is included in the meeting minutes.

HANDOUT: 7/15

Time Delay Circuits on Inverters

1. Class 1E 7.5 kVA inverters (I-1A, -1B, -1C, -1D, -1E and -1F).

   There are no time delays on the voltage sensing circuits on the Class 1E inverters. High dc voltage at the output of the rectifier section will result in tripping the ac input only. Power will continue to be supplied from the 125 V dc battery.

2. Non-Class 1E 60 kVA inverters (I-2A and I-2B).

   There are no time delays on the voltage sensing circuit, on these inverters. High or low dc voltage at the rectifier section output and high or low ac voltage at the inverter section output will trip the inverter off and force an automatic transfer to the backup ac supply through the solid state transfer switch.

3. Non-Class 1E 25 kVA inverter (I-4).

   There are no time delays on the voltage sensing circuits on this inverter. High or low dc voltage at the inverter section input will trip the inverter input breaker and force an automatic transfer to the backup ac supply through the solid state transfer switch.

No modifications to the 1E and non-1E inverter were found necessary as a result of the re-review of IE Circular 79-02.

STATUS: 9/14

Closed.

420.51 (7.5) Table 7.1-1 indicates that conformance to R.G. 1.97 is discussed in Section 7.5.3.2. However, Section 7.5.3.2 is a section of definitions only. We find partial discussion on conformance in Section 7.5.3.1. Correct Table 7.1-1. Also, FSAR Section 1.8 states that Regulatory Guide 1.97, Revision 2, is presently being reviewed and the extent of compliance will be addressed at a later date. Discuss the plans and schedule for complying with R.G. 1.97, Revision 2.

RESPONSE: 3/23 Applicant is working on response to Regulatory Guide 1.97, Revision 2. Schedule will be supplied at a later date.

STATUS: 9/14 We have continued to review Seabrook for compliance with Regulatory Guide 1.97, Rev. 2. We are following the applicable discussions within the NRC, particularly those of the CRGR in relation to SECY 82-111. Will not be an open item on the SER.

420.52 (7.6.2) Provide a discussion (using detailed drawings) on the residual heat removal (RHR) system as it pertains to Branch Technical Position ICSB 3 and RSB 5-1 requirements. Specifically address the following as a minimum:

1. Testing of the RHR isolation valves as required by branch position E of BTP RSB 5-1.

2. Capability of operating the RHR from the control room with either on-site or only off-site power available as required by Position A.3 of BTP RSB 5-1. This should include a discussion of how the RHR system can perform its function assuming a single failure.

3. Describe any operator action required outside the control room after a single failure has occurred and justify.

In addition, identify all other points of interface between the Reactor Coolant System (RCS) and other systems whose design pressure is less than that of the RCS. For each such interface, discuss the degree of conformance to the requirements of Branch Technical Position ICSB No. 3. Also, discuss how the associated interlock circuitry conforms to the requirements of IEEE Standard 279. The discussion should include illustrations from applicable drawings.

RESPONSE: 3/23 The RHR isolation valves can be tested while on RHR by operating only one RHR pump, removing power from one valve associated with the operating pump, simulating high pressure in the isolation channel for the valve that has power removed and verifying that the associated valve in the non-operating loop closes. The system is restored, the sequence repeated for the other isolation channel, cooling shifted to the other loop and the test sequence repeated.

NRC will review reply to RAI 440.23 and 440.24 that address power sources.

There is no other system interfacing with the reactor coolant system (RCS) whose design pressure is less than that of the RCS.

STATUS: 9/14 — The RSB has concerns with the response to RAI 440.23. They are continuing their review. Additional information will be provided on the design of the RHR suction valve controls and indication and time available to restore RHR flow following inadvertent closure RHR suction valves. Information will include alarms for switch position, need for temporary modification, alarm to indicate valve closure, analysis to consider worst case conditions for all modes, and operator action required. FSAR 5.4.7.2 will be revised.

ADDITIONAL RESPONSE: 11/82 — We will add alarms that will actuate if either suction valve for an operating RHR pump is not fully open or if the flow through the RHR pump is below the minimum required for pump protection.

If the suction valves close due to a power failure in the logic circuit (circuit is designed to fail to the isolation condition to ensure protection of the low pressure piping), the valves can be reopened at the remote shutdown location. This operation can be performed expeditiously, less than 10 ten minutes, since the controls for remote shutdown bypass all interlocks. Temporary circuit modifications are not required. FSAR 5.4.7.2.f will be revised; a draft copy of the change is attached.

FSAR 5.4.7.2 discusses the effects of temporary loss of RHR flow.

420.53 (7.6.4) — FSAR Section 7.6.4, Accumulator Motor-Operated Valves, states that, "During plant operation, these valves are normally open, and the motor control center supplying power to the operators is de-energized". Describe how power is removed and how the system complies to Positions B.2, B.3 and B.4 of BTP ICSB 18 (PSB). Also, identify any other such areas of design and state your conformance to the positions of BTP ICSB 18.

RESPONSE: 3/23 — Covered in response to 420.59.

STATUS: 5/12 — Closed.

420.54 (7.3.1.1) (7.6.5) — FSAR Section 7.3.1.1 states that, "The transfer from the injection to the recirculation phase is initiated automatically and completed manually by operator action from the main control board". Describe automatic and manual design features permitting switchover from injection to recirculation mode for emergency core cooling including protection logic, component bypasses and overrides, parameters monitored and controlled and test capabilities. Discuss design features which insure that a single failure will neither cause premature switchover nor prevent switchover when required. Discuss the reset of Safety Injection actuation prior to automatic switchover from injection to recirculation and the potential for defeat of the automatic switchover function. Confirm whether the low-low level refueling water storage tank alarms which determine the time at which the containment spray is switched to recirculation mode are safety grade.

RESPONSE: 3/23 Will be discussed later.

RESPONSE: 5/12 The step-by-step automatic and manual switchover operations are described in detail in FSAR Section 6.3.2.8 and Table 6.3-7. The ECCS/Containment Spray Recirculation Signal is generated for each train by a combination of the safety injection signal and low-low level in the RWST. The level signal uses 2 out of 4 logic to prevent premature switchover and to ensure switchover is accomplished. Each ESF train uses completely redundant equipment for recirculation to ensure that the safety functions are accomplished. The operator is provided with safety grade indicators for RWST and containment sump level, and manual controls for all the valves required for recirculation so that recirculation can be accomplished without any automatic action. Non-safety grade but independent low-low level alarms are available from the VAS and the annunciator to alert the operator of the need for recirculation.

The safety injection signal sets latching relay K740 that requires separate action to reset after the safety injection signal has been reset. This ensures automatic recirculation on low-low level in the RWST even if the safety injection signal is reset before the low-low level is reached. Lights will be provided on MCB AF and BF to indicate when K740 is latched to ensure that it is reset after periodic testing. The light has a lamp test feature. Its operation is also verified as part of the periodic testing.

ADDITIONAL RESPONSE: 7/15 The independence of the non-safety grade RWST low-low level alarms was discussed. Details will be provided later. Level setpoints are provided in Figure 6.3-6 (Amendment 45).

ADDITIONAL RESPONSE: 9/14 The four transmitters that provide the low-low level recirculation signal will provide an annunciator alarm when any of the low-low level bistables have tripped. A wide range level transmitter will provide an analog input to the station computer. The station computer will generate a VAS low-low level alarm at the same setpoint as the annunciator alarm.

STATUS: 9/14 Confirmatory pending review of formal documentation.

420.55 (5.2.5.8) (7.6) FSAR Section 5.2.5.8 states that calibration and functional testing of the leakage detection systems will be performed prior to initial plant startup. Please provide justification since Position C.8 of Regulatory Guide 1.45 states that, "leakage detection systems should be equipped with provisions to readily permit testing for operability and calibration during plant operation".

RESPONSE: 3/23 The electronics can be tested with plant at power. There are readouts that can be checked during plant operation. Radiation sensors can be tested at power because they have check source in them. Level sensors will be channel calibrated in accordance with Technical Specifications.

STATUS: Closed.
5/12

420.56 (7.6) As shown on Drawing 9763-M-310882 SH-B54a, two circuit breakers in series are employed in the power and control circuits for the residual heat removal inlet isolation valves. Tripping of either breaker will remove power from the position indicating lights and valve position indication will be lost. Discuss how this arrangement complies with Branch Technical Position ICSB No. 3 which calls for suitable valve position indication to the control room.

RESPONSE: Handout submitted to staff. Valve position indicator lights will
3/23 be powered from different source so that true valve position will always be indicated when power is removed from valve motor by racking out breaker. This applies to RHR interface valves.

STATUS: Confirmatory pending review of formal documentation.
9/14

HANDOUT: Two circuit breakers in series are employed in the circuits of
3/23 motor-operated valves inside containment. This is part of the containment penetration protection provided in response to Regulatory Guide 1.63. Refer to FSAR Section 8.3.1.1.c.7a.

Valve position indication is provided on both RCS-RHR interface valves which are in series. As with any circuit, when power is removed because of a fault, indication will also be lost.

We believe that our revised design meets the intent of ICSB 3 position B4.

In addition to the normal valve position indication lights, the valve full closed position is also monitored by the station computer to alarm whenever the valve is not fully closed and the reactor coolant system is above the pressure rating of the RHR system.

420.57 (7.6) Section 7.6.2.1 indicates that the interlock circuits of the residual heat removal isolation valves, RC-V22 and RC-V87, have a transmitter that is diverse from the transmitter associated with valves RC-V23 and RC-V88. Discuss the method(s) used to achieve this diversity.

RESPONSE: Different manufacturers for pressure transmitters are used to
3/23 achieve the diversity.

STATUS: Closed.
5/12

420.58 (7.6) Discuss conformance of the accumulator motor-operated valves to the recommendations of Branch Technical Positions ICSB No. 4.

RESPONSE: Handout submitted to staff. Change response to indicate valve
3/23 position is monitored through video alarm system (VAS). Details of VAS will be in the response to 420.49.

Staff will review adequacy of alarm.

STATUS: 9/14 Closed.

HANDOUT: 3/23 The design of the accumulator motor-operated valves conforms to the recommendations of ICSB No. 4. Refer to FSAR Section 7.6.4 for a response to Branch Technical Positions B1 and B2.

Branch Technical Position B3:

Valve position is monitored and alarmed by the video alarm system.

Branch Technical Position B4:

The automatic safety injection signal bypasses all main control board switch functions which may have closed the SI accumulator valve.

The safety injection signal will not automatically return power to the de-energized motor control center.

420.59 (7.6) Section 7.6.9 of the FSAR lists the motor-operated valves which will be protected from spurious actuation by removal of motor and control power by de-energizing their motor control centers (MCC 522 and MCC 622). The FSAR also states that control of the breakers supplying power to these MCCs is provided in the main control room. Provide the following information:

(a) The control the the MCC breaker from the Main Control Board for a typical Safety Injection System accumulator isolation valve is not shown on schematic diagram 9763-M-310890 Sh. B35a. Identify the drawing where this is shown.

(b) The residual heat removal inlet isolation valves are not included in the list of valves protected against spurious operation. State whether protection against spurious action of these isolation valves is planned and if so, provide information on how it is accomplished. If not, then justify.

RESPONSE: 3/23 (a) Refer to FSAR Section 8.3.3. Alarm is provided in the control room when the breaker is closed.

(b) Reply given in response to RAI 440.23 and will be reviewed by the staff.

ADDITIONAL RESPONSE: 5/12 We will explain the operation of valves 35, 36, 89, 90 and 93 and the effects of failure of valve 93 or its position switches.

STATUS: 9/14 The valve interlocks were discussed during the meeting held June 23, 1982. Additional information on interlock testing is required.

ADDITIONAL RESPONSE: 11/82

A telephone conversation was held on 11/10/82 with representatives of the NRC (R. Stevens), PSNH (R. LaRhette), and YAEC (W. Fadden) to discuss the operation and testing of the interlocks associated with ECCS recirculation. Significant items of discussion were:

1. The pump recirculation line isolation valves (SI-V89, 90, 93) are provided to prevent RWST contamination and subsequent release to the environment. The valves are arranged so that the line will be isolated assuming any single failure. Failure to isolate will not affect the performance of the ECCS for cooling the core.

2. The RHR recirculation valves (RH-V35 and 36) are provided so that either RHR pump can supply both safety injection and both centrifugal charging pumps. No single failure of the SI pump recirculation valves (SI-V89, 90, 93) or the associated interlocks, will prevent the operation of either RHR recirculation valve.

3. These valves and the associated interlocks will be periodically tested. The test schedule and procedures will be available for review three months before fuel loading.

420.60 (7.6)

The following apparent errors have been noted in the schematic diagrams.

(a) Drawing M-310980, Sh. B35d, Rev. 0

Contacts 5-5C on LOCAL REMOTE SWITCH SS-2403 appear incorrectly developed. An X indicating contacts closed should appear under the REMOTE column for contact 5 to allow remote closing of the accumulator valves.

(b) Drawing 9763-M-310900, Sh. B52a, Rev. 1

Motor starter 42 open coil is mislabeled 42/C instead of 42/O.

RESPONSE: 3/23

We agree with your observation of drawing errors on the two schematic sheets mentioned and this will be corrected in the next revision of these drawings.

STATUS: 5/12

Closed.

420.61 (7.6.6)

FSAR Section 7.6.6 discusses interlocks for RCS pressure control during low temperature operation. Using detailed schematics, discuss how this interlock system complies with Positions B.2, B.3, B.4 and B.7 of BTP RSB 5-2. Be sure to discuss the degree of redundancy in the logic for the low temperature interlock for the RCS pressure control. Also, include a discussion on block valve control.

RESPONSE: 3/23 Reply for the low temperature operation of the RCS pressure control will be under RAI 440.11.

The block valves and manual controls are Class 1E, train oriented, with controls being on the main control board.

REVISED RESPONSE: 5/12 Design of the cold overpressure interlocks will be changed to make them single failure proof.

ADDITIONAL RESPONSE: 9/14 The single failure problem with the cold overpressure interlocks was related to the use of one auctioneer card in each circuit to arm the other circuit and actuate the same circuit. Redundant auctioneer cards will be added to each circuit so that the arming and actuating signals will be independent, therefore, no single failure will prevent operation of both relief valves. FSAR Figures 7.6-4 will be revised.

STATUS: 9/14 Confirmatory pending review of formal documentation.

420.62 (7.7) If control systems are exposed to the environment resulting from the rupture of reactor coolant lines, steam lines or feedwater lines, the control systems may malfunction in a manner which would cause consequences to be more severe than assumed in safety analyses. I&E Information Notice 79-22 discusses certain non-safety grade or control equipment, which if subjected to the adverse environment of a high energy line break, could impact the safety analyses and the adequacy of the protection functions performed by the safety grade systems.

The staff is concerned that a similar potential may exist at light water facilities now under construction. You are, therefore, requested to perform a review per the I&E Information Notice 79-22 concern to determine what, if any, design changes or operator actions would be necessary to assure that high energy line breaks will not cause control system failures to complicate the event beyond the FSAR analysis. Provide the results of your review including all identified problems and the manner in which you have resolved them.

The specific "scenarios" discussed in the above referenced Information Notice are to be considered as examples of the kinds of interactions which might occur. Your review should include those scenarios, where applicable, but should not necessarily be limited to them.

RESPONSE: 3/23 We will identify key control systems that effect plant safety and analyze for effects of high energy line break. Review will be completed and formal response to I&E Information Notice 79-22 submitted.

STATUS: (420.62 & .63) 9/14 We have received the memo from Check to Tedesco that provides additional guidance. Our review is in progress and the required reports will be submitted later.

420.63
(7.7)

If two or more control systems receive power or sensor information from common power sources or common sensors (including common headers or impulse lines), failures of these power sources or sensors or rupture/plugging of a common header or impulse line could result in transients or accidents more severe than considered in plant safety analyses. A number of concerns have been expressed regarding the adequacy of safety systems in mitigation of the kinds of control system failures that could actually occur at nuclear plants, as opposed to those analyzed in FSAR Chapter 15 safety analyses. Although the Chapter 15 analyses are based on conservative assumptions regarding failures of single control systems, systematic reviews have not been reported to demonstrate that multiple control system failures beyond the Chapter 15 analyses could not occur because of single events. Among the types of events that could initiate such multiple failures, the most significant are, in our judgment, those resulting from failure or malfunction of power supplies or sensors common to two or more control systems.

To provide assurance that the design basis event analyses adequately bound multiple control system failures, you are requested to provide the following information:

(1) Identify those control systems whose failure or malfunction could seriously impact plant safety.

(2) Indicate which, if any, of the control systems identified in (1) 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.

(3) Indicate which, if any, of the control systems identified in Item 1 receive input signals from common sensors. The sensors considered should include, but should not necessarily be limited to, common hydraulic headers or impulse lines feeding pressure, temperature, level or other signals to two or more control systems.

(4) Provide justification that any simultaneous malfunctions of the control systems identified in (2) and (3) resulting from failures or malfunctions of the applicable common power source or sensor are bounded by the analyses in Chapter 15 and would not require action or response beyond the capability of operators or safety systems.

RESPONSE: 3/23 We will submit formal response similar to that submitted on other Westinghouse plants.

STATUS: 9/14 See 420.62.

420.64 (7.7.1) FSAR Section 7.7.1 discusses steam generator water level control. Discuss, using detailed drawings, the operation of this control system. Include information on what consequences (i.e., overfilling the steam generator and causing water flow into the steam piping, etc.) might result from a steam generator level control channel failure. Be sure to discuss the high-high steam generator level logic used for main feedwater isolation.

RESPONSE: 3/23 High-high steam generator level trip will be changed to two out of four logic.

ADDITIONAL RESPONSE: 5/12 S/G level is not programmed as a function of power level. 420.67 from the draft memo dated 3/22/82 is now 420.70.

STATUS: 9/14 Confirmatory pending review of formal documentation.

420.65 (7.2) (7.3) Recent review of a plant (Waterford) revealed a situation where heaters are to be used to control temperature and humidity within insulated cabinets housing electrical transmitters that provide input signals to the reactor protection system. These cabinet heaters were found to be unqualified and a concern was raised since possible failure of the heaters could potentially degrade the transmitters, etc.

Please address the above design as it pertains to Seabrook. If cabinet heaters are used, then describe as a minimum the design criteria used for the heaters.

RESPONSE: 3/23 Class 1E electronic transmitters are not mounted in an insulated cabinet with heaters for temperature and humidity control. The subject design, therefore, does not pertain to Seabrook.

STATUS: 5/12 Closed.

Note: The NRC memo dated March 22, 1982, on the SSPS slave relay contacts is now 420.81.

420.66 (7.2) It is not clear from the drawings provided and the description of the turbine trip circuits and mechanisms that the equipment used to trip the turbine following a reactor trip meets the criteria applicable to equipment performing a safety function.

It is the staff position that the circuits and equipment used to trip the turbine following a reactor trip should meet the criteria applicable to a safety function with the exception of the fact that the circuits may be routed through non-seismic qualified structures and the turbine itself is not seismically qualified. Please provide further discussion on how the Seabrook design meets the staff position.

RESPONSE: 5/12 We will comply with the attached Westinghouse Interface Criteria for Implementation of Turbine Trip on Reactor Trip. We are discussing the design changes required with General Electric Co., the turbine supplier.

ADDITIONAL RESPONSE: 9/14 We will provide redundant, safety grade (except for seismic qualification) solenoids powered from the vital busses, that are energized to trip the turbine.

STATUS: 9/14 Confirmatory pending review of formal documentation.

420.67 (7.2) The reactor coolant system hot and cold leg resistance temperature detectors (RTD) used for reactor protection are located in reactor coolant bypass loops. A bypass loop from upstream of the steam generator to downstream of the steam generator is used for the hot leg resistance temperature detector and a bypass loop from downstream of the reactor coolant pump to upstream of the pumps is used for the cold leg resistance temperature detector. The magnitude of the flow affects the overall time response of the temperature signals provided for reactor protection.

It is the staff's position that the magnitude of the RTD bypass loop flow be verified to be within required limits at each refueling period and that this requirement be included into the plant technical specifications. Please provide discussion on how the Seabrook design complies with the staff's position. If there are any exceptions please describe and provide justification.

RESPONSE: 5/12 Westinghouse letter SNP-4340, attached, evaluates the potential for reduced flow in the RTD Bypass System due to corrosion product deposition. Based on their analysis, we do not consider flow reduction due to crud to be a problem.

We will verify the bypass flow rates during the preoperational testing program. The low flow alarm in the combined return line will be set at a value to indicate unacceptable flow degradation in either the cold or hot leg bypass manifolds.

This response is the same as was made to Catawba.

This item is open pending NRC review.

STATUS: 7/15 The NRC reiterated the position that the bypass flow be reverified each refueling. Technical Specification revision is required.

ADDITIONAL RESPONSE: 9/14 Preoperational verification of bypass flow will be by test procedure that follows the guidance of NAH/NCH-SU-2.1.9, Resistance Temperature Detector Bypass loop Flow Verification. Surveillance procedures that verify the bypass loop flow will be available 90 days before fuel loading. The surveillance procedure will be performed every refueling. Any required Technical Specification will be generated as part of procedures outlined in NUREG 0452, Revision 4.

STATUS: 9/14 Closed.

420.68 (7.2) Operation of either of two manual reactor trip switches de-energizes the reactor trip breaker undervoltage coils and, at the same time, energizes the breaker shunt coils for the breakers associated with both protection logic trains.

It is the staff's position that the plant technical specifications include a requirement to periodically, independently verify the

operability of the undervoltage and shunt trip functions. Please describe how the Seabrook design complies with our position. If there are any exceptions please identify with sufficient justification.

RESPONSE: 5/12 We defer response pending generic resolution of this item by Westinghouse and the NRC (Ref. NS-EPR-2588, dated 4/29/82).

STATUS: 9/14 The NRC has responded to the Westinghouse letter. Issue is still open. Westinghouse is preparing for further discussions.

420.69 (7.2) Several safety system channels make use of lead, lag or rate signal compensation to provide signal time responses consistent with assumptions in the Chapter 15 analyses. The time constants for these signal compensations are adjustable setpoints within the analog portion of the safety system. The staff position is that the time constant setpoint be incorporated into the plant technical specifications. Please provide a discussion on this matter.

RESPONSE: 5/12 The time constants are in Tables 2.2-1 and 2.2-2 of the Technical Specification. Attached is a revised Table 2.2-2 with editorial corrections and inclusion of the time constants that clarify Item 4.E.

STATUS: 9/14 Closed.

420.70 (7.2) (7.3) The present Seabrook design shows that three steam generator level channels are to be used in a two-out-of-three logic for isolation of feedwater on high steam generator level and that one of the three level channels is used for control. This design for actuation of feedwater isolation does not meet Paragraph 4.7 of IEEE-279 on "Control and Protection System Interaction". For example, the failure of the level channel used for control in the low direction could defeat the redundancy requirements (i.e., a single failure of one of the remaining channels defeats the two-out-of-three requirements). Therefore it is the staff's position that the system be modified (i.e., addition of a fourth protection channel) to meet the redundancy requirements or provide an analysis justifying that isolation of feedwater on high-high steam generator level is not required for safety. Please provide a discussion based on the above staff requirements.

RESPONSE: 5/12 This was addressed in the March 23-25 meetings as Item 420.67. Commitment was made to change the S/G high level trip to 2 out of 4 (see 420.64).

STATUS: 9/14 — Confirmatory pending review of formal documentation.

420.71 (7.2) — FSAR Figure 7.2-1, Sheet 2 shows a reactor trip initiated by a General Warning Alarm from the Solid State Protection System. The information presented in the FSAR does not sufficiently describe this trip signal. Therefore, please provide additional information to describe and justify this reactor trip.

RESPONSE: 5/12 — The Seabrook SSPS is functionally similar to that discussed at Catawba. FSAR Section 7.2.2.2 will be revised per attached markup as was done at Catawba.

STATUS: 9/14 — Closed.

420.72 (7.3) — Using detailed drawings (schematics, P&ID's), describe the automatic and manual operation and control of the main steam and feedwater isolation valves. Describe as a minimum how the design complies with the requirements of IEEE-279 (i.e., single failure, redundancy indication of operability, direct valve position indication in the control room, automatic actuation, etc.).

RESPONSE: 5/12

(a) Discussions on circuit modifications to the MISV controls continue. Response is deferred pending resolution (see 420.37a).

(b) The MFWIV's were discussed with 420.37.

STATUS: 9/14 — Closed (items called out above were discussed with those of 420.37).

420.73 (7.3) (7.4) — Instrumentation for process measurements used for safety functions such as reactor trip or emergency core cooling typically are provided with the following:

a) An indicator in the control room to provide the operator information on the process variable being monitored which can also be used for periodic surveillance checks of the instrument transmitter.

b) An alarm to indicate to the operator that a specific safety function has been actuated.

c) Indicator lights or other means to inform the operator which specific instrument channel has actuated the safety function.

d) Rod positions, pump flows, or valve positions to verify that the actuated safety equipment has taken the action required for the safety function.

e) Design features to allow test of the instrument channel and actuated equipment without interfering with normal plant operations.

During recent reviews, it has been found that one or more of the features above was not provided for certain instrumentation used to initiate safety functions. Examples include instrumentation used to isolate essential service water to the air compressors, instrumentation used to isolate the non-safety-related portion of the component cooling water system, and instrumentation used to isolate the spray additive tank on low-low level.

The staff position is that instrumentation provided to perform safety functions such as isolating non-seismic portions of systems, closing valves when tank levels reach low level setpoints, and similar functions should be provided with alarms and indicators commensurate with the importance of the safety function and should be testable without interfering with normal plant operations. The applicants should provide the staff with a list of all instrument channels which perform a safety function where one or more of the features listed in a through e of the concern above are not currently provided. For each of these instrument channels, the applicants should indicate which of the features a through e are not currently provided. The staff position on these instrument channels is further that the applicants should:

a) Provide an alarm to indicate that the safety function has been actuated if such an alarm is not in the current design.

b) If not in the current design, provide means to inform the operator which specific channel has actuated the safety function.

c) If not in the current design, provide indication that the actuated safety equipment has taken the action required for the safety function.

d) If not in the current design, provide the capability for testing each safety function without interfering with normal plant operations and without lifting instrument leads or using jury rigs. The capability for testing should include the transmitter where indicators are not provided to perform operability checks of the transmitters.

The staff will provide requirements in the plant technical specifications for testing these safety functions. Please provide discussion on how the Seabrook design meets the above stated staff position. If there are any exceptions please describe and provide justification.

RESPONSE: 5/12 — A preliminary list was provided. We are evaluating the missing features and will respond at the next meeting.

STATUS: 9/14 — Our review continues. A complete report will be submitted at a later date.

ADDITIONAL RESPONSE 11/82 — Safety function instrumentation at Seabrook can be divided into two general classifications; actuation instrumentation and control instrumentation.

Actuation instrumentation performs functions that are considered protective functions (i.e. reactor trip and engineered safety features actuation) or are necessary to provide essential auxiliary functions (cooling tower actuation, isolation of the non-safety component cooling water piping). This instrumentation is designed to meet the requirements of IEEE 279 and typically has the following features:

a) Dedicated indicator in the Control Room.

b) Alarm on actuation of a specific safety function.

c) Indicator lights on the MCB, VAS alarm, channel indication at the instrument cabinets to alert the operator to a channel in the trip condition and to identify the specific channel, this indication is not applicable to functions that only have one sensing instrument.

d) Indication to monitor the performance of the actuated equipment.

e) Capability to perform the surveillance tests specified in the Technical Specifications (see Section 7.2.2.2(c)).

Control instrumentation performs functions associated with the control of auxiliary supporting features in response to changes in a measured variable (start of cooling fans to maintain environmental conditions, operation of valves to meet minimum flow conditions for a pump). These control functions only affect the operation of one of the redundant safety trains, the other train is available to perform the safety function if one train fails. This instrumentation is not designed to meet the requirements of IEEE 279 and typically has the following features:

f) Control Room or local indication to monitor the controlled variable.

g) Independent alarm if the controlled variable exceeds the expected control band.

h) Capability to perform periodic calibration and functional tests.

We have reviewed the safety function instrumentation at Seabrook to verify the availability of the typical features discussed previously. Table 420.73-1 lists all the instrumentation that does not have all of the applicable features listed, the missing feature is specified with corrective action planned to provide the feature or justification why the feature is not required.

TABLE 420.73-1

Safety Function Instrumentation Design Features

1) Safety Function - Cooling tower actuation signal (TA).

Missing Features - c) No alarm if one pressure channel has tripped.

d) Not all of the actuated equipment can be tested at power.

Remarks - c) An alarm will be provided if any pressure channel is tripped, the specific channel will be indicated at the instrument cabinet.

d) The actuated equipment will be divided into test groups so that the equipment that cannot be tested during the routine testing can be blocked, the following functions are blocked and will be tested each refueling:

- Trip of the service water pump, this is provided for pump or diesel generator protection only and does not provide a safety function.

- Realignment of the service water discharge valves to return to the cooling tower (SW-V19, 20, 23, 34), the function of providing service water from the cooling tower can be performed without these valves re-aligning, the proper re-alignment is required to conserve water inventory, sufficient time is available for this function to be performed manually.

2) Safety Function - Isolation of non-safety component cooling water piping on low level in the head tank.

Missing Features - c) No alarm if only one level channel has tripped.

d) The containment isolation valves (CC-V57, 121, 122, 168, 175, 176, 256, 257) are not tested during power operation.

Remarks - c) An alarm will be provided if any level channel is tripped, the specific channel will be indicated at the instrument cabinet.

d) The actuation of these valves would cause a loss of cooling to the reactor coolant pumps, the actuation signal is blocked and continuity testing will be performed as discussed in FSAR 7.3.2.2.e.

TABLE 420.73-1
(continued)

Safety Function Instrumentation Design Features

3) Safety Function - RWST lo-lo level recirculation actuation.

Missing Features - a) Indication from level transmitters (CBS-LT-930, 931, 932, 933) is not available.

c) Two out of four channels tripped is alarmed.

Remarks - a) Level indicators will be provided to permit channel comparison.

c) Alarm is considered adequate since there are channel tripped indicators at the instrument cabinet and the function will not actuate unless there is a coincident safety injection signal.

4) Safety Function - Emergency feedwater high flow isolation.

Missing Features - a) Indication is not provided for the backup instrumentation (i.e., B Train instruments for S/G A&C, A Train instruments for S/G B&D).

Remarks - a) Provisions are available for periodic calibration, the indicator for the primary instrumentation is available to check the transmitter zero, there is flow in the system only when supplying EFW to the S/G, the zero for.

5) Safety Function - RHR pump low flow recirculation valve control.

Missing Features - g) An independent low flow alarm is not available.

Remarks - g) An independent low flow alarm will be provided.

6) Safety Function - High temperature start of cooling fans for the emergency feedwater pump house, service water pump house and cooling tower switchgear area.

Missing Features - f) Local indication is not provided.

Remarks - f) Local temperature indication will be provided.

420.74 (7.3) On November 7, 1979, Westinghouse notified the Commission of a potential undetectable failure which could exist in the engineered safeguards P-4 interlocks. Test procedures were developed to detect failures which might occur. The procedures require the use of voltage measurements at the terminal blocks of the reactor trip breaker cabinets.

In order to minimize the possibility of accidental shorting or grounding of safety system circuits during testing, suitable test jacks should be provided to facilitate testing of the P-4 interlocks. Provide a discussion on how the above issue will be resolved for Seabrook.

RESPONSE: 5/12 In SBN-120, dated May 15, 1980, we committed to the tests described in NS-TMA-2204.

ADDITIONAL RESPONSE: 7/15 We will provide suitable circuits for testing the P-4 interlock. Details will be provided later.

ADDITIONAL RESPONSE: 9/14 Test switches and meters will be permanently installed to perform the tests outlined in SBN-120.

STATUS: 9/14 Closed.

420.75 (7.3) (9.3.4) (6.3) On May 21, 1981, Westinghouse notified the Commission of a potentially adverse control and protection system interaction whereby a single random failure in the Volume Control Tank level control system could lead to a loss of redundancy in the high head safety injection system for certain Westinghouse plants. Please determine whether this generic problem exists on Seabrook and, if so, how the problem is to be resolved.

RESPONSE: 5/12 The generic problem is applicable to Seabrook. We are evaluating Westinghouse recommendations for procedural changes.

ADDITIONAL RESPONSE: 9/14 In SBN-164, dated June 18, 1981, we committed to reviewing the plant procedures to ensure that the operators would be properly alerted and would take appropriate action. The procedures will be available for review 3 months prior to fuel loading. An analysis performed by Westinghouse (see NAH-1935, dated April 23, 1982, copy attached) indicates that there is in excess of ten minutes from the VCT low level alarm until the VCT is empty.

STATUS: 9/14 Open pending NRC review.

420.76 (7.4) Discuss the likelihood that emergency core cooling will be automatically initiated following a manual reactor trip initiated during a temporary evacuation of the control room. For example, is it possible for the reactor coolant system to be cooled to the point that the pressurizer empties during the time interval

between manual reactor trip and the time an operator can take control of auxiliary feedwater outside the control room? Analyses and operating experience from plants similar to Seabrook should be presented during the discussion. Based upon the likelihood of emergency core cooling actuation following a manual reactor trip, should the capability for resetting the equipment be provided outside the control room?

RESPONSE: 9/14 11/82

Westinghouse has analyzed the transient resulting from evacuation of the control room using the following assumptions:

1. The reactor, turbine, MSIVs, and RCPs were tripped, in this order, prior to leaving the control room, no other operator action was taken.

2. The trip was from various power levels from 0 to 100% power with no decay heat (50% power was the most limiting).

3. EFW temperature was 40°F.

4. Both EFW pumps operate at the time of reactor trip and provide 1440 GPM.

11/82

The analysis shows that low main steam pressure safety injection will not occur until more than 568 seconds after the reactor trip. This will provide sufficient time for the operator to throttle EFW flow to stop the cooldown. It should be noted that Assumptions 2 and 4 are extremely conservative. A more detailed analysis using realistic decay heat loads expected during a power ascension and delay in actuation of EFW (actuation on the initial shrink after a trip is not expected) will show considerably more time is available to throttle EFW.

If safety injection is actuated, the operator has the capability of terminating flow by stopping the charging and RHR pumps from CP 108 A & B and by tripping the SI pumps at the switchgear. These pumps can be restarted from outside the control room without temporary modifications if necessary. Automatic start of the SI pumps is not defeated by local trip of the breaker.

STATUS: 9/14

Open pending NRC review with RAI 420.38.

420.77 (7.4) (5.4.10.3)

The FSAR states that the pressurizer auxiliary spray valve is used during cooldown when the reactor coolant pumps are not operating and FSAR Section 7.4 lists the auxiliary spray as a system required for safe shutdown. FSAR Figure 9.3-13 shows this system as a single path with a single diaphragm operated valve. A single failure could conceivably:

1) Prevent the use of auxiliary spray for cooldown,

2) Cause inadvertent actuation, or

3) Prevent isolation of the system.

Using detailed fluid and schematic drawings, please provide further discussion describing the operation of the auxiliary spray system.

RESPONSE: 9/14 The safety grade power operated relief valves will be used to depressurize the RCS during safe shutdown; therefore, the auxiliary spray valves have been deleted from FSAR 7.4. See the draft revision provided for RAI 420.38.

STATUS: 9/14 Confirmatory pending review of formal documentation.

420.78 (7.4) Provide a discussion on the termination of possible inadvertent boron dilution. Will automatic equipment be used for termination?

RESPONSE: 5/12 The revised criteria for the boron dilution accident promulgated by NUREG-0800 are under review.

ADDITIONAL RESPONSE: 9/14 We will meet the operator response times specified in NUREG 0800 following receipt of a flux increase alarm from the safety grade wide range neutron monitor.

STATUS: 9/15 Closed pending ICSB discussions with RSB.

420.79 (7.7.1.2) Describe the design features used in the rod control system which

1) Limit reactivity insertion rates resulting from single failures within the system.

2) Limit incorrect sequencing or positioning of control rods. The discussion should cover the assumptions for determining the maximum control rod withdrawal speed used in the analyses of reactivity insertion transients.

RESPONSE: 5/12 Section 7.7.1.2.2 of the FSAR will be revised per attached markup to describe features that limit reactivity insertions, maximum rod speeds and incorrect sequencing resulting from single failures within the system. This evaluation is identical to that made for the SNUPPS review. The SNUPPS and Seabrook rod control systems are functionally identical.

STATUS: 9/14 Closed.

420.80 The FSAR (Section 5.2.2.8) information describing direct position indication of relief and safety valves is insufficient to allow the staff to complete its review. Therefore, please provide additional information on how the Seabrook design complies with each specific requirement of NUREG-0737, TMI Item II.D.3.

RESPONSE: 5/12 The FSAR will be revised when the details of the valve position indication system are known (see 420.05 response).

STATUS: 9/14 Confirmatory (see 420.05(a)).

480.81 During the Seabrook drawing review it was discovered that safeguards actuation circuits have parallel relay contacts to handle specific load requirements. The slave relays used for the output of the solid state protection system (SSPS) have apparently been qualified by Westinghouse for use in circuits drawing a maximum current of 4.4 amps. It is our understanding that the Seabrook 5 Kv and 15 Kv systems expose the SSPS slave relay contacts to a magnitude of 5.2 amps upon safeguards actuation. The applicant has decided to use parallel contacts to carry the current, relying on simultaneous closure (and opening) of the safeguards contacts upon protection signal actuation.

This design concept is unacceptable to the staff. We have concluded that paralleling contacts may not solve the concern with the current ratings of the Westinghouse slave relay contacts since closure (or opening) of the SSPS slave relay contacts at the exact same time cannot be assured. One set of contacts will, in most instances, function before its redundant counterpart thus allowing the full 5.2 amps to that set of contacts. Also, it appears that the present test methods do not allow for checking operation of each individual set of contacts when paralleled. It is the staff's position that the relays used in the protection system should be qualified for the maximum expected current.

The applicant is requested to modify the Seabrook design to comply with the above staff position.

RESPONSE: 5/12 We will perform an independent test to verify the contact current carrying capabilities of the SSPS slave relays. The test will be performed on single contacts controlling actual switchgear components.

Upon completion of the tests, the NRC will be notified on the disposition of the issue regarding the use of these relays.

The NRC expressed concern that the testing meet similar requirements as were utilized during the W testing. Departures should be justified.

STATUS: 9/14 We are discussing this test with Worcester Polytechnic Institute. We expect to have results available by January 1983. A test plan that includes acceptance criteria and justification will be provided.

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SEABROOK STATION MAIN PLANT COMPUTER SYSTEM

A Summary of the
Video Alarm System Software Development Methodology
and Verification and Validation Techniques

The Video Alarm System (VAS) is a computer-based method of presenting alarms to the control board operator. Traditionally this has been done by several assemblies of light-box annunciators. However by incorporating recent advances in computer technology and human factor engineering the VAS has been designed to further improve the operator's awareness of plant operating conditions. This document summarizes the design and development techniques used to construct the VAS.

A draft functional description was first written by Yankee Atomic Electric Company (YAEC). This description was reviewed by groups within both PSNH and YAEC. Many suggestions, enhancements and changes were incorporated into the second revision. The second revision was again reviewed by YAEC and PSNH with the addition of the PSNH operations group as reviewers. Comments resulting from this last review were incorporated resulting in a final document approved by all groups.

The functional description was then analyzed. Each functional area was identified and defined. These areas consist of the following:

A. Remote multiplexor software
   This includes digital input scanning, failed or unstable input processing, and data transfer of exception reports.

B. Administration and control of inputting exception report buffers from remote multiplexors.

C. Logic-by-computer processing including alarm logging, archival storage, and audio annunciation via horns.

D. Alarm presentation to the control board operator. Included here is
   1. unacknowledged alarm presentation
   2. operator acknowledgement of alarms
   3. display of alarms including paging
   4. return-to-normal presentation
   5. operator acknowledgement of return-to-normals

E. Operator functional management of VAS alarm points such as delete from scan, restore to scan, deleting and retrieving groups of points, and construction of these groups.

F. Operator management of display devices to handle failures.

G. Initialization and start-up processor.

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Careful examination was made of these areas both singly and as they relate to the overall system. Changes were carefully evaluated for impact on other functional areas. Global data bases and file structures were then defined to support the functional area breakdown.

Each area was further divided into modules with further definition given to supporting global and local data bases. These modules were coded, debugged, tested, and verified. For any detected design errors potential solutions were evaluated for impact on remaining code in this module, the data base, and on other interacting modules. Revised modules were re-tested and verified.

Completed modules were then assembled into tasks and again tested and verified for correctness. Of particular concern was the interaction of tasks with themselves and with the operating system. This was extensively checked and resulted in combining several smaller tasks into one large task. This allowed better control over execution sequences and data file usage.

Extensive validation was then performed on three computer systems. First was a minimum (or "nucleus") configuration containing 4 CRT devices and 2 remotes. This allowed considerable testing of the remote multiplexor software, data transfer scheme, logic by computer processing and alarm presentation. A second installation on the Seabrook Simulator allowed in-depth analysis of the alarm presentation software. Certain buffer size changes resulted from the simulator testing in order to adequately handle major disturbances resulting in many alarms. Plant operators were involved in the checkout and review of the simulator system.

Finally the system was installed on the main plant computer system at its staging location in Manchester. Complete testing was accomplished under the fully configured system with all remote multiplexors and CRT devices available. At this point, only minor changes were installed to fine tune the system for optimum performance.

System documentation includes the functional specification and module descriptions, some of which are currently out of revision and need updating. There are data base definition documents for global regions and mass storage data files. The code itself is nearly all FORTRAN and is well commented.

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7.5 SAFETY-RELATED DISPLAY INSTRUMENTATION

7.5.1 Description

Monitoring of safety functions, ESF Systems, and ESF Support Systems is accomplished using analog indicators, status lamp arrays, VAS alarms, hardwired annunciator windows, CRT displays and breaker and valve position indicating lights. These indications are all used in conjunction with one another to enable the operator to monitor the status of the plant under all operating conditions.

Table 7.5-1 lists the safety-related analog indications provided to the operator to enable him to monitor safety functions, take any manual actions required to support the accomplishment of safety functions, and to determine the effect of manual actions taken following a reactor trip due to a Condition III or IV event, as defined in Chapter 15. Table 7.5-1 also lists the analog indications required to maintain the plant in a hot shutdown condition, or to proceed to cold shutdown within the limits of the Technical Specifications.

Table 7.5-2 lists the indications available to the operator to monitor the performance of individual safety systems after a reactor trip and to monitor radioactive releases. These indications supplement those of Table 7.5-1 and can be used to corroborate the indications of Table 7.5-1.

Table 7.5-3 lists the information available to the operator for monitoring conditions in the reactor, the Reactor Coolant System, and in the Containment and key process systems throughout all normal operating conditions of the plant, including anticipated operational occurrences.

Status lamp arrays are used to indicate both a demand for a protective function/ESF actuation and the appropriate valve line-ups for ESF actuations. These status lamp arrays provide the operator with sufficient information to enable him to take any manual actions required to support the accomplishment of safety functions. Status lamp arrays are provided for the following safety functions:

A. Reactor Trip
B. Safety Injection
C. Containment Isolation
D. Steam Line Isolation
E. Feedwater Line Isolation

To monitor valve line-ups and emergency power availability, status lamp arrays are provided for the following:

A. Cold Leg Injection
B. Cold Leg Recirculation
C. Hot Leg Recirculation
D. Containment Isolation, Phase A
E. Containment Isolation, Phase B
F. Main Steam and Feedwater Isolation
G. Tower Actuation

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H. Diesel Generator Status
I. Emergency Power Sequencer

A computer-based Video Alarm System (VAS) is provided to alert the operator when various process limits are exceeded. The incoming alarms are prioritized to allow the operators to focus on high priority alarms during major plant upsets. Three levels of priority have been established. Incoming alarms are also broken down into primary and secondary sides; primary side alarms are displayed on the alarm CRTs in Sections A and D while secondary side alarms are displayed on the alarm CRTs in Sections F and I.

The power supply for the VAS is two battery-backed UPSs. The system is completely redundant with the exception of IRTU #7; manual transfer capability is available to align this IRTU and other peripherals onto the available UPS.

A Hardwired Annunciator System has been installed to back up the VAS should a complete Computer System failure occur. This system also has a limited "First Out" capability to assist the operator in determining the cause of a reactor trip. The Hardwired Annunciator System monitors a limited set of essential parameters and (except First Out) will be used to monitor plant status until the VAS is back in service.

The Hardwired Annunciator System is powered from a vital instrument panel; this power source is independent of the power supply for the VAS.

Various CRT based dynamic displays will also be developed to serve the needs of the operating crew. These displays will supplement those described above.

Bypassed and inoperable condition of safety systems is displayed ON THE VAS AND on status lamp arrays on the MCB - one per train. Refer to Subsection 7.1.2.6 for a complete discussion.

7.5.2 Analyses

The indicator channels (see Table 7.5-1) required to enable the operator to take the correct action during the course of a Condition III or IV accident or during the recovery phase are designed to the criteria listed in Section 7.5.3. These indicators are designated as Post-Accident Monitoring (PAM) and are identified on the MCB by uniquely colored nameplates.

The indicators in Table 7.5-1 are also used for the operational monitoring of the plant and thus, are under surveillance by the operator during normal plant operation. The indicators are functionally arranged on the control board to provide the operator with ready understanding and interpretation of plant conditions. Comparisons between duplicate information channels or between functionally related channels will enable the operator to readily identify a malfunction in a particular channel. The range of the readouts extends over the maximum expected range of the variable being measured, as listed in Table 7.5-1. The combined indicated accuracies, as shown in Table 7.5-1, are within the errors used in the safety analyses.

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The indicator channels listed in Table 7.5-1 are fully qualified for the environmental conditions under which they must operate. This ensures that reliable information will be presented to the operator at all times.

The indications in Table 7.5-2 supplement the indicators of Table 7.5-1. They are designed to provide information on the status/performance of individual safety systems. They are functionally grouped according to system to permit ready assessment of system performance. The range of the readouts extends over the maximum range of the variable being measured, as listed in Table 7.5-2.

The status lamp arrays are functionally arranged on the control board to enable the operator to quickly and accurately monitor safety system status. Information from duplicate channels is presented to enable the operator to readily identify a malfunction in a particular channel. Status lamp arrays for valve line-ups and emergency power availability are provided for both Train A and Train B to allow the operator to readily monitor system status in both trains.

The Control Room Design Review will also identify the instruments required during the course of an accident or in the recovery phase. Various plant emergency procedures will be reviewed and analyzed to determine the instrumentation required. The results of this review will be used to update both Table 7.5-1 and Table 7.5-2 as required. This review will ensure that the instruments required are arranged to support both normal operations and accident conditions and fully support the emergency procedures to be used by the operator.

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7.5.3 Design Criteria

7.5.3.1 Scope

The scope of IEEE Standard 279-1971 covers protection systems that initiate automatic protective actions. Therefore, in the absence of applicable industry standards for the safety-related post-accident monitoring (PAM) instrumentation, the following criteria were developed, using applicable sections of IEEE Standard 279-1971 as a model for this purpose.

The environmental and seismic qualification of these sensors is covered in Sections 3.11 and 3.10, respectively.

The general design criteria and applicable documents are discussed in Section 7.1. The following criteria establish requirements for the functional performance and reliability of the safety-related PAM indications listed in Table 7.5-1. The safety-related PAM indicators encompass those display devices which provide information needed to:

- Enable the operator to take the correct manual action during the course of a Condition III or IV fault or during recovery from a Condition III or IV fault.
- Maintain hot shutdown condition.
- Proceed to cold shutdown conditions.

7.5.3.2 Definitions

The definitions in this section establish the meanings of words in the context of their use in these criteria.

Channel - An arrangement of components and modules as required to generate a single information signal to monitor a generating station condition.

Components - Items from which the system is assembled (for example, resistors, capacitors, wires, connectors, transistors, tubes, switches, springs, etc.).

Module - Any assembly of interconnected components which constitute an identifiable device, instrument, or piece of equipment. A module can be disconnected, removed as a unit, and replaced with a spare. It has definable performance characteristics which permit it to be tested as a unit. A module could be a card or other subassembly of a larger device, provided it meets the requirements of this definition.

Post-Accident Monitoring Function - A post-accident monitoring function consists of the sensing of one or more variables associated with a particular safety function, signal processing, and the presentation of visual information (including recorded information, if required) to the operator.

Type Test - Tests made on one or more units to verify adequacy of design.

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7.5.3.3 Requirements

a. General Functional Requirements

The nuclear power generating station post-accident monitoring instrumentation shall function with precision and reliability to continuously display the appropriate monitored variables. This requirement shall apply for the full range of conditions and performance enumerated.

b. Single Failure Criterion

Any single failure within the post-accident monitoring instrumentation shall not result in the loss of the monitoring function. ("Single failure" includes such events as the shorting or open-circuiting or interconnecting signal or power cables. It also includes single credible malfunctions or events that cause a number of consequential component, module, or channel failures. For example, the overheating of an amplifier module is a "single failure" even though several transistor failures result. Mechanical damage to a mode switch would be a "single failure" although several channels might become involved.)

c. Quality of Components and Modules

Components and modules shall be of a quality that is consistent with minimum maintenance requirements and low failure rates. Quality levels shall be achieved through the specification of requirements known to promote high quality, such as requirements for design, for the derating of components, for manufacturing, quality control, inspection, calibration, and test.

d. Equipment Qualification

Type test data or reasonable engineering extrapolation based on test data shall be available to verify that Monitoring System equipment shall meet, on a continuing basis, the performance requirements determined to be necessary for achieving the system requirements. Seismic and environmental qualification programs are addressed in Sections 3.10 and 3.11, respectively.

e. Channel Integrity

All Monitoring System channels shall be designed to maintain necessary functional capability, including accuracy and range, under extremes of conditions (as applicable) relating to environment, energy supply, and malfunctions during the circumstances throughout which the Monitoring System must function.

f. Channel Independence

Channels that provide signals for the same monitoring function shall be independent and physically separated to accomplish decoupling of the

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effects of unsafe environmental factors, electric transients, and physical accident consequences documented in the design basis, and to reduce the likelihood of interactions between channels during maintenance operations or in the event of channel malfunction. Malfunctions, accidents, and other unusual events shall include, for example, fire, explosions, missiles, lightning, earthquakes, etc.

g. Power Source

The post-accident monitoring display instrumentation shall be capable of operating independent of off-site power availability.

h. Post-Accident Monitoring Instrumentation and Control System Interaction

1. Classification of Equipment

Any equipment that is used for both post-accident monitoring and control functions shall be classified as part of the post-accident monitoring instrumentation and shall meet all the necessary requirements.

2. Isolation Devices

The transmission of signals from the post-accident monitoring equipment for control or monitoring shall be through isolation devices which shall be classified as part of the post-accident monitoring instrumentation and shall meet all the requirements of this document. No credible failure at the output of an isolation device shall prevent the associated PAM channel from meeting the minimum performance requirements considered in the design bases. Examples of credible failures include short circuits, open circuits, grounds, and the application of the maximum credible ac or dc potential (140 V dc or 118 V ac). A failure in an isolation device is evaluated in the same manner as a failure of other PAM equipment.

i. Derivation of System Inputs

Inputs to the Monitoring System are derived from signals that are direct measures of the desired variables. The channels listed also, in many cases, bear a known relationship to each other during normal plant operation.

j. Capability for Sensor Checks

Means shall be provided for checking, with a high degree of confidence, the operational availability of each system input sensor during reactor operation.

This may be accomplished in various ways, for example:

1. By perturbing the monitored variable; or

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2. By introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable; or

3. By cross-checking between channels that bear a known relationship to each other and that have readouts available.

k. Capability for Verifying Operability

Means shall be available for verifying the operability of the Monitoring System channels. Where channels exhibit a dynamic response during normal plant operation or are required frequently for normal plant operation, verification of operability is inherent in the normal functioning of the channels. For channels which monitor a normally static parameter, provisions must be included to allow periodic testing thereby verifying channel operability. Identification of malfunctions will be adequately identified by cross-checking between duplicate redundant channels or cross-checking between channels that bear a known relationship to each other during normal plant operation.

l. Channel Bypass or Removal from Operation

The system shall be designed to permit any one channel to be maintained when required during power operation. During such operation, the active parts of the system need not themselves continue to meet the single failure criterion. As such, Monitoring Systems comprised of two redundant channels are permitted to violate the single failure criterion during channel bypass provided that acceptable reliability of operation can be otherwise demonstrated. The bypass time interval allowed for a maintenance operation will be specified in the plant Technical Specifications.

m. Access to Means of Bypassing

The design shall permit the administrative control of the means for manually bypassing channels.

n. Access to Setpoint Adjustments, Calibration, and Test Points

The design shall permit the administrative control and access to all setpoint adjustments, module calibration adjustments, and test points.

o. System Repair

The system shall be designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.

p. Identification

In order to provide assurance that the requirements given in this document can be applied during the design, construction, maintenance, and operation of the plant, the post-accident monitoring instrumentation

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channels (for example, interconnecting wiring, components, modules, etc.) shall be identified with the appropriate channel colors to distinguish between redundant channels. The channel and equipment colors used are shown in Subsections 7.1.2.3 and 8.3.1.3.

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TABLE 7.5-1
(Sheet 1 of 8)

MAIN CONTROL BOARD INDICATORS AVAILABLE TO THE OPERATOR
(CONDITION III AND IV EVENTS)

1. Wide Range T_{hot} and T_{cold}

a. Channels provided: Loop 1 T_{hot}, T_{cold}
Loop 2 T_{hot}, T_{cold}

b. Range: 0 to 700°F.

c. Purpose:

	Indicated Accuracy (1)
1. Maintain the plant in a safe shutdown condition.	± 1.5% of full range.
2. Ensure proper cooldown rate.	
3. Ensure proper relationship between system pressure and temperature.	

2. Core Exit Thermocouples — Indicated Accuracy: (later)

a. Channels provided:

56 monitoring four quadrants.

b. Range: 0-2300°F

c. Purpose:

1. Provide indication of adequate core cooling.

2. Ensure proper cooldown rate.

---

(1) Indicated Accuracy: Channel accuracy at the readout device; includes sensor and readout device inaccuracies under worst case conditions. (For level measurements, environmental effects on sensing lines will be addressed later.)

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TABLE 7.5-1
(Sheet 2 of 8)

MAIN CONTROL BOARD INDICATORS AVAILABLE TO THE OPERATOR
(CONDITION III AND IV EVENTS)

3. Pressurizer Water Level

a. Channels provided:

2 channels on separate power supplies.

b. Range: Entire distance between taps (0 to 100%).

c. Purpose:	Indicated Accuracy
1. Maintain proper reactor reactor coolant inventory.	± 15% of full range.
2. Determine return of water level to pressurizer following steam break and steam generator tube ruptures.	

4. System Wide Range Pressure

a. Channels provided:

2 channels on separate power supplies.

b. Range: 0 to 3000 psig.

c. Purpose:	Indicated Accuracy
1. Ensure proper relationship between system pressure and temperature.	(later)

5. Containment Pressure (narrow and wide range)

a. Channels provided:

2 narrow range channels are provided on separate power supplies.
2 wide range channels are also provided on separate power supplies.

b. Range: 0 to 60 psig (0 to 115% of containment design pressure) for narrow range.
-5 to 160 psig (-5 psig to 300% of containment design pressure) for wide range.

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TABLE 7.5-1
(Sheet 3 of 8)

MAIN CONTROL BOARD INDICATORS AVAILABLE TO THE OPERATOR
(CONDITION III AND IV EVENTS)

c. Purpose:	Indicated Accuracy
1. Monitor containment conditions following primary or secondary system break inside containment.	(later) (narrow range) (later) (wide range).

6. Steam Generator Pressure

a. Channels provided:

2 channels per steam generator on separate power supplies.

b. Range: 0 to 1300 psig.

c. Purpose:	Indicated Accuracy
1. Needed to determine type of accident that has occurred and the proper recovery procedure to use.	(later)
2. Determine that plant is in a safe shutdown condition.	

7. Steam Generator Water Level (narrow and wide range)

a. Channels provided:

Four narrow range channels, one per steam generator, are provided on separate power supplies. Four wide range channels, one per steam generator, are also provided on different power supplies. Thus, there is one wide and one narrow range channel per steam generator, and these two channels for the same steam generator are on different power supplies.

b. Range: 0 to 100% of span for both wide or narrow range.

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TABLE 7.5-1
(Sheet 4 of 8)

MAIN CONTROL BOARD INDICATORS AVAILABLE TO THE OPERATOR
(CONDITION III AND IV EVENTS)

	Indicated Accuracy
c. Purpose:	
1. Maintain adequate heat sink following an accident.	±15% of full range.
2. Needed in recovery procedure following steam generator tube rupture.	
3. Ensure that steam generator tubes are covered following a LOCA.	

8. Refueling Water Storage Tank Level

a. Channels provided:

2 channels on separate power supplies.

b. Range: 0 to 100% of span.

	Indicated Accuracy
c. Purpose:	
1. Determine when to perform the necessary manual actions following switchover from the injection phase to the recirculation phase of safety injection after a LOCA.	± 1.8% of level span.

9. Boric Acid Tank Level (2 tanks)

a. Channels provided:

1 level channel per tank, each on a separate power supply.

b. Range: 0 to 100% level.

	Indicated Accuracy
c. Purpose:	
1. To ensure that borated water is leaving the boric acid tanks.	± 1.8% of level span.

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TABLE 7.5-1
(Sheet 5 of 8)

MAIN CONTROL BOARD INDICATORS AVAILABLE TO THE OPERATOR
(CONDITION III AND IV EVENTS)

10. Emergency Feedwater Flow (4 steam generators)

a. Channels provided:

1 channel per steam generator. One power supply serves two steam generators.

b. Range: 0 to 300 gpm.

c. Purpose:	Indicated Accuracy
1. To assure maintenance of residual heat removal and to determine uniformity of flow to each steam generator.	± 1.8% of full range.

11. Emergency Feed Pump Suction Pressure

a. Channels provided:

1 channel per feed pump.

b. Range:

0-30 psig.

c. Purpose:	Indicated Accuracy
1. To ensure adequate NPSH and provide gross indication of Condensate Storage Tank Level.	(later)

12. Containment Wide Range Level

a. Channels provided:

2 channels on separate power supplies.

b. Range: 0 to 6 feet.

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TABLE 7.5-1
(Sheet 6 of 3)

MAIN CONTROL BOARD INDICATORS AVAILABLE TO THE OPERATOR
(CONDITION III AND IV EVENTS)

	Indicated Accuracy
c. Purpose:	
1. To provide extended range level monitoring in the event of sump overflow and containment flooding.	(later)

13. Cooling Tower Level

a. Channels provided:

2 channels on separate power supplies.

b. Range: Top 20 feet of sump.

	Indicated Accuracy
c. Purpose:	
1. Assure that service water is available to diesel generator jacket cooling heat exchangers and PCCW heat exchangers.	(later)
2. Monitor rate of level decrease to detect pipe breaks.	
3. Monitor adequacy of makeup.	

14. Diesel Generator Parameters

a. Channels provided:

1 channel is provided for voltage, current, frequency and power from each diesel generator.

b. Range: - Voltage - 0 to 5000 Volts
Current - 0 to 2000 Amps
Frequency - 55 to 65 Hz
Power - 0 to 9000 kW

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TABLE 7.5-1
(Sheet 7 of 8)

MAIN CONTROL BOARD INDICATORS AVAILABLE TO THE OPERATOR
(CONDITION III AND IV EVENTS)

	Indicated Accuracy
c. Purpose:	
1. These measurements confirm satisfactory diesel generator operation relative to power available for engineered safety features.	± 1.6% of full range.

15. Control Room Air Intake Radiation Monitor

a. Channels provided:

2 channels on separate power supplies.

b. Range: 10^1 to 10^6 CPM.

	Indicated Accuracy
c. Purpose:	
1. Assure the operator of continued habitability in the control room and indicate site radiation background levels.	(later)

16. Containment Post-LOCA Area Radiation Monitor

a. Channels provided:

2 channels on separate power supplies. These indicators are in the control room on the radiation monitor Class 1E cabinets and not on the main control board.

b. Range: 10^0 to 10^7 R/hr.

	Indicated Accuracy
c. Purpose:	
1. Measure radiation level inside containment during post-accident conditions.	(later)

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TABLE 7.5-1
(Sheet 8 of 8)

MAIN CONTROL BOARD INDICATORS AVAILABLE TO THE OPERATOR
(CONDITION III AND IV EVENTS)

17. Containment Hydrogen Analyzer

a. Channels provided:

2 channels on separate power supplies.

b. Range: 0 to 10% H_2.

c. Purpose:	Indicated Accuracy
1. Provide operator with hydrogen concentration levels inside the containment and aid in assessing the effectiveness of H_2 recombiner operation.	± 2.5% of full range.

18. Nuclear Instrumentation

a. Channels provided:

2 channels on separate power supplies.

b. Range: (later)

c. Purpose:	Indicated Accuracy
1. To verify reactor trip.	(later)
2. To maintain the reactor subcritical.	

19. Containment Isolation (Indicating light array)

a. Channels provided:

1 channel per valve; 2 valves per penetration are provided.

b. Range: Closed/not closed, open/not open.

c. Purpose:	Indicated Accuracy
1. To provide operator with a concise display of the status of containment isolation.	N/A

TABLE 7.5-2
(Sheet 1 of 9)

MAIN CONTROL ROOM INDICATION AVAILABLE TO THE OPERATOR
FOR MONITORING OF INDIVIDUAL SYSTEMS AND RADIOACTIVE RELEASES

Parameter	No. of Channels Available	Range	Indicated Accuracy	Indication	Location	Notes
NUCLEAR INSTRUMENTATION SYSTEM						
1. Source Range Flux NI-NI-31B,32B	2	1 to 10^6 counts/sec	±7%	Both channels indicated. Either may be selected for recording.	Main Control Board	
2. Intermediate Range NI-NI-35B, 36B	2	10^{-11} to 10^{-3} amps	±7% ±3% between 10^{-4} and 10^{-3} amps	Both channels indicated. Either may be selected for recording using the recorder in Item 1 above.	Main Control Board	
3. Power Range NI-NI-41B, 42B, 43B, 44B	4	0 to 120%	±1%	Indicators/Recorders	Main Control Board	
REACTIVITY CONTROL						
Rod Position Indication System		Full out to full in	N/A	Indicating light array	Main Control Board	
Soluble Boron Concentration CS-X-7329-1	1	0-5000 ppm	Ltr.	Digital Indicator	Main Control Board	

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TABLE 7.5-2
(Sheet 2 of 9)

MAIN CONTROL ROOM INDICATION AVAILABLE TO THE OPERATOR
FOR MONITORING OF INDIVIDUAL SYSTEMS AND RADIOACTIVE RELEASES

Parameter	No. of Channels Available	Range	Indicated Accuracy	Indication	Location	Notes
REACTOR COOLANT SYSTEM						
Primary System PORV and Safety Valve Position	1/Valve for PORVs 1 Channel for Safety Valves	Open-Closed	N/A	Indicating lights and VAS alarm pt.	Main Control Board	Valve limit switches for PORVs. Acoustic monitoring of common discharge line for the safety valves. Refer to Section 5.2.2.8 for a complete discussion.
Pressurizer Relief Tank Temperature RC-T-468	1	50-250°	Ltr.	Indicator	Main Control Board	
Pressurizer Relief Tank Pressure RC-P-469	1	0-100 psig	Ltr.	Indicator	Main Control Board	
Pressurizer Relief Tank Level RC-L-470	1	0-100%	Ltr.	Indicator	Main Control Board	
Margin to Saturation	Ltr.	Ltr.	Ltr.	CRT display on demand	Main Control Board	System is in design.

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TABLE 7.5-2
(Sheet 3 of 9)

MAIN CONTROL ROOM INDICATION AVAILABLE TO THE OPERATOR

FOR INFORMATION OF INDIVIDUAL SYSTEMS AND INDICATION PANELS

Parameter	No. of Channels Available	Range	Indicated Accuracy	Indication	Location	Notes
CONTAINMENT COOLING SYSTEM						
1. Containment Spray Pump Suction Pressure CBS-P-2312, 2314	1	0-30 psig	Ltr.	Indicator	Main Control Board	
2. Containment Spray Pump Discharge Pressure CBS-P-2313, 2315	1	0-500 psig	Ltr.	Indicator	Main Control Board	
EMERGENCY FEEDWATER SYSTEM						
1. Condensate Storage Tank Level CO-L-4079, 4096	2	0-40 ft	Ltr.	Indicators	Main Control Board	Backup Indication to EFP Suction Pressure
SAFETY INJECTION SYSTEM						
1. Accumulator Tank Level SI-L-950, 951, 952 953, 954, 955, 956, 957	2/ Accumulator	0-100%	Ltr.	Both channels indicated	Main Control Board	
2. Accumulator Tank Pressure SI-P-960, 961, 962, 963, 964, 965, 967,	2/	0-700 psig Accumulator	Ltr.	Both channels indicated	Main Control Board	

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TABLE 7.5-2
(Sheet 4 of 9)

MAIN CONTROL ROOM INDICATION AVAILABLE TO THE OPERATOR
FOR MONITORING OF INDIVIDUAL SYSTEMS AND RADIOACTIVE RELEASES

Parameter	No. of Channels Available	Range	Indicated Accuracy	Indication	Location	Notes
SAFETY INJECTION SYSTEM (Cont.)						
3. Accumulator Isolation Valve Position SI-Z-2403-1, 2413-1, 2423-1, 2433-1	1/Valve	Open-Closed	N/A	Red and green indicator lights for each valve	Main Control Board	
4. Boron Injection Tank Charging Header Flow SI-F-917	1	0-1000 gpm	Ltr.	Indicator	Main Control Board	
5. SI Flow SI-F-918, 922	2	0-800 gpm	Ltr.	Indicators	Main Control Board	
RESIDUAL HEAT REMOVAL SYSTEM						
1. RHR System Flow RH-F-618, 619	2	0-5000 gpm	Ltr.	Indicators	Main Control Board	
2. RHR Heat Exchanger Inlet Temp. RH-T-604, 605	2	50-400°F	Ltr.	Recorder	Main Control Board	
3. RHR Heat Exchanger Outlet Temp. RH-T-612, 613	2	50-400°F	Ltr.	Recorder	Maine Control Board	

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TABLE 7.5-2
(Sheet 5 of 9)

MAIN CONTROL ROOM INDICATION AVAILABLE TO THE OPERATOR
FOR MONITORING OF INDIVIDUAL SYSTEMS AND RADIOACTIVE RELEASES

Parameter	No. of Channels Available	Range	Indicated Accuracy	Indication	Location	Notes
CHEMICAL AND VOLUME CONTROL SYSTEM						
1. Makeup Flow In CS-F-121	1	0-200 gpm	Ltr.	Indicator	Main Control Board	
2. Letdown Flow Out CS-F-132	1	0-200 gpm	Ltr.	Indicator	Main Control Board	
3. Volume Control Tank Level CS-L-185	1	0-100%	Ltr.	Indicator	Main Control Board	
COMPONENT COOLING WATER SYSTEM						
1. Component Cooling Water Temp. to ESF System CC-T-2171, 2271	2	0-175°	Ltr.	Indicators	Main Control Board	
2. Component Cooling Water Flow to ESF System CC-F-2103, 2203	2	0-14 x 10^3 gpm	Ltr.	Indicators	Main Control Board	

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TABLE 7.5-2
(Sheet 6 of 9)

MAIN CONTROL ROOM INDICATION AVAILABLE TO THE OPERATOR
FOR MONITORING OF INDIVIDUAL SYSTEMS AND RADIOACTIVE RELEASES

Parameter	No. of Channels Available	Range	Indicated Accuracy	Indication	Location	Notes
VENTILATION SYSTEM						
Emergency entilation Damper Position	1/Damper	Open-Closed	N/A	Red and green indicating lights	Main Control Board (Rear)	
DP-12A,B	"	"	"	"	"	
DP-36A,B	"	"	"	"	"	
DP-35A,B	"	"	"	"	"	
DP-43A,B	"	"	"	"	"	
DP-44A,B	"	"	"	"	"	
DP-27A,B	"	"	"	"	"	
Dr-53A,B	"	"	"	"	"	
POWER SOURCES						
1. Bus 5 Voltage EDE-VM-9891-2	1	0-5000 V	1.6%	Indicator	Main Control Board	
2. Bus 6 Voltage EDE-VM-9746-2	1	0-5000 V	1.6%	Indicator	Main Control Board	

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TABLE 7.5-2
(Sheet 7 of 9)

MAIN CONTROL ROOM INDICATION AVAILABLE TO THE OPERATOR
FOR MONITORING OF INDIVIDUAL SYSTEMS AND RADIOACTIVE RELEASES

Parameter	No. of Channels Available	Range	Indicated Accuracy	Indication	Location	Notes
POWER SOURCES (Cont.)						
3. 125 V dc Buses						
Battery Bus 11A - Voltage EDE-VM-9750	1	0-150 V	1.6%	Indicator	Main Control Board	
- Current EDE-AM-9751	1	0-1000 A	1.6%	Indicator	Main Control Board	
Battery Bus 11B - Voltage EDE-VM-9752	1	0-150 V	1.6%	Indicator	Main Control Board	
- Current EDE-AM-9753	1	0-1000 A	1.6%	Indicator	Main Control Board	
Battery Bus 11C - Voltage EDE-VM-9754	1	0-150 V	1.6%	Indicator	Main Control Board	
- Current EDE-AM-9755	1	0-1000 A	1.6%	Indicator	Main Control Board	
Battery Bus 11D - Voltage EDE-VM-9756	1	0-150 V	1.6%	Indicator	Main Control Board	
- Current EDE-AM-9757	1	0-1000 A	1.6%	Indicator	Main Control Board	

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TABLE 7.5-2
(Sheet 8 of 9)

MAIN CONTROL ROOM INDICATION AVAILABLE TO THE OPERATOR
FOR MONITORING OF INDIVIDUAL SYSTEMS AND RADIOACTIVE RELEASES

Parameter	No. of Channels Available	Range	Indicated Accuracy	Indication	Location	Notes
Process Radiation Monitoring System						
Main Steam Line Radiation Monitor RM-RM-6481 6482	1/Steam Line	Ltr.	Ltr.	Indicated on radiation monitoring CRT	Facing Main Control Board	
Condenser Air Evacuation Monitor RM-RM-6505	1	10^1 to 10^6 cpm	Ltr.	Indicated on radiation monitoring CRT	Facing Main Control Board	
Plant Vent Stack Radiation Monitor RM-RM-6533	1	10^{-6} to 10^5 uCi/cc	Ltr.	Indicated on radiation monitoring CRT	Facing Main Control Board	
S.G. Blowdown Sample Monitors RM-RM-6510, 6511, 6512, 6513	4	10^{-6} to 10^{-2} uCi/cc	Ltr.	Indicated on radiation monitoring CRT	Facing Main Control Board	
S.G. Blowdown Sample Monitor RM-RM-6519	1	10^{-7} to 10^{-3} uCi/cc	Ltr.	Indicated on radiation monitoring CRT	Facing Main Control Board	
Reactor Coolant Letdown Gross Activity Monitor RM-RM-6520-1, 2	2	10^{-4} to 10^3 uCi/cc	Ltr.	Indicated on radiation monitoring CRT	Facing Main Control Board	

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TABLE 7.5-2
(Sheet 9 of 9)

MAIN CONTROL ROOM INDICATION AVAILABLE TO THE OPERATOR
FOR MONITORING OF INDIVIDUAL SYSTEMS AND RADIOACTIVE RELEASES

Parameter	No. of Channels Available	Range	Indicated Accuracy	Indication	Location	Notes
Meteorology						
Wind Direction	1	0 to 360 degrees	± 3%	Recorder	Main Control Board	El. 209
Wind Speed	2	0 to 100 mph	± 1%	Recorder	Main Control Board	El. 209 & 43
Temperature	1	-30° to $+110^{o}F$	$\pm 0.95\%^{o}F$-Sensor	Recorder	Main Control Board	El. 43
Delta Temperature	2	10° to $\pm 18^{o}F$	±.1% of span-bridge	Recorder	Main Control Board	El. 150 & 43 El. 290 & 43
Dew Point	1	$-30^{o}$to $+100^{o}F$	$\pm 0.36^{o}F$	Recorder	Main Control Board	El. 43
Precipitation	1	N/A	±1% to ± 6%	Recorder	Main Control Board	
Solar Radiation	1	$0-2Cal/cm^{2}-min$	± 5%	Recorder	Main Control Board	

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TABLE 7.5-3
(Sheet 1 of 8)

CONTROL ROOM INDICATORS AND/OR RECORDERS AVAILABLE TO THE OPERATOR TO MONITOR SIGNIFICANT PLANT PARAMETERS DURING NORMAL OPERATION INCLUDING OPERATIONAL OCCURRENCES

Parameter	No. of Channels Available	Range	Indicated Accuracy (1)	Indicator/ Recorder	Location	Notes
NUCLEAR INSTRUMENTATION						
1. Source Range						
a. Count rate	2	1 to 10^6 counts/sec.	±7% of the linear full scale analog voltage	Both channels indicated. Either may be selected for recording.	Control Board	One two-pen recorder is used to record any of the 8 nuclear channels (2 source range, 2 intermediate range and 4 power range)
b. Startup rate	2	0.5 to 5.0 decades/ min.	±7% of the linear full scale analog voltage	Both channels indicated.	Control Board	
2. Intermediate Range						
a. Current	2	10^{-11} to 10^{-3}	±7% of the linear full scale analog voltage and ±3% of the linear full scale voltage in the range of 10^{-4} to 10^{-3} amps	Both channels indicated. Either may be selected for recording using the recorder in Item 1 above.	Control Board	

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TABLE 7.5-3
(Sheet 2 of 8)

Parameter	No. of Channels Available	Range	Indicated Accuracy (1)	Indicator/ Recorder	Location	Notes
NUCLEAR INSTRUMENTATION (Cont.)						
b. Startup rate	2	0.5 to 5.0 decades/ min.	±7% of the linear full scale analog voltage	Both channels indicated	Control Board	
3. Power Range						
a. Uncalibrated ion chamber current (top and bottom uncompensated ion chambers)	4	0 to 120% of full power current	±1% of full power current	All 8 current signals indicated.	NIS racks in Control Room	
b. Calibrated ion chamber current (top and bottom uncompensated ion chamber)	4	0 to 125% of full power current (0 to 5mA)	±2% full power current	All 8 current signals recorded on four 2-pen recorders. Recorder 1 - upper currents for two diagonally opposed detectors. Recorder 2 - upper currents for remaining detectors. Recorder 3 - lower currents for two diagonally opposed detectors. Recorder 4 - lower currents for remaining detectors.	Control Board	

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TABLE 7.5-3
(Sheet 3 of 8)

Parameter	No. of Channels Available	Range	Indicated Accuracy (1)	Indicator/ Recorder	Location	Notes
NUCLEAR INSTRUMENTATION (Cont.)						
c. Upper and lower ion chamber current difference	4	-60 to 60%	±3% of full power	Diagonally opposed channels may be selected for recording at the same time using recorder in Item 1.	Control Board	
d. Average flux of the top and bottom ion chamber (% full power)	4	0 to 120% of full power	±3% of full power for indication ±2% for recording	All 4 channels indicated. Any 2 of the four channels may be recorded using recorder in Item 1 above.	Control Board	
e. Average flux of the top and bottom ion chambers (power range overpower)	2	0 to 200% of full power	±2% of full power to 120% ±6% of full power to 200%	Both channels recorded.	Control Board	
f. Flux difference of the top and bottom ion chambers	4	-30 to 30%	±4%	All 4 channels indicated.	Control Board	
REACTOR COOLANT SYSTEM						
1. $T_{average}$ (measured)	1/loop	530° - 630°F	± 4° F	All channels indicated.	Control Board	

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TABLE 7.5-3
(Sheet 4 of 8)

Parameter	No. of Channels Available	Range	Indicated Accuracy (1)	Indicator/ Recorder	Location	Notes
REACTOR COOLANT SYSTEM (Cont.)						
2. ΔT (measured)	1/loop	0 to 150% of full power ΔT	$\pm$4% of full power ΔT	All channels indicated. One channel is selected for recording.	Control Board	
a. T_{cold} or T_{hot} (measured, wide range)	1-T_{hot}, 1-T_{cold} per loop	0 to 700°F	$\pm$4%	4T_{hot} channels are recorded on 2 - two-pen recorders. 4 T_{cold} channels are recorded on 2 - two-pen recorders.	Control Board	
3. Overpower ΔT Setpoint	1/loop	0 to 150% of full power ΔT	$\pm$ of full power ΔT	All channels indicated. One channel is selected for recording.	Control Board	
4. Overtemperature ΔT Setpoint	1/loop	0 to 150% of full power ΔT	$\pm$4% of full power ΔT	All channels indicated. One channel is selected for recording.	Control Board	
5. Pressurizer Pressure	4	1700 to 2500 psig	$\pm$28 psi	All channels indicated.	Control Board	
6. Pressurizer Level	3	Entire distance between taps	$\pm$3.5% ΔP level at 2250 psia	All channels indicated. One channel is selected for recording.	Control Board	Two-pen recorder used, second pen records reference level signal.
7. Primary Coolant Flow	3/loop	0 to 120% of rated flow	Repeatability of $\pm$4.5% of full flow	All channels indicated.	Control Board	

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TABLE 7.5-3
(Sheet 5 of 8)

Parameter	No. of Channels Available	Range	Indicated Accuracy (1)	Indicator/ Recorder	Location	Notes
REACTOR COOLANT SYSTEM (Cont.)						
8. Reactor Coolant Pump Motor Current	1/loop	0 to 400 ac amps	±1.0%	All channels indicated.	Control Board	One channel for each pump.
9. System Pressure Wide Range	2	0 to 3000 psig	±1.8%	All channels indicated and recorded.	Control Board	
REACTOR CONTROL SYSTEM						
1. Rod Speed	1	5 to 75 steps/min.	±2%	The one channel is indicated.	Control Board	
2. Auctioneered T_{avg}	1	530° to 630°F	±4°F	The one channel is recorded.	Control Board	Any one of the T_{avg} channels into the auctioneer may be bypassed.
3. $T_{reference}$	1	530° to 630°F	±4°F	The one channel is recorded.	Control Board	
4. Control Rod Position						If system not available, borate and sample accordingly.
a. Number of steps of demanded rod withdrawal	1/group	0 to 230 steps	±1 step	Each group is indicated during rod motion.	Control Board	These signals are used in conjunction with the measured position signals (4b) to detect deviation of any

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TABLE 7.5-3
(Sheet 6 of 8)

Parameter	No. of Channels Available	Range	Indicated Accuracy (1)	Indicator/ Recorder	Location	Notes
REACTOR CONTROL SYSTEM (Cont.)						
						individual rod from the demanded position. A deviation will actuate an alarm and annunciator.
b. Full length rod measured position	1 for each rod	0 to 228 steps	±4 steps	Each rod position indicated.	Control Board	
5. Control Rod Bank Demanded Position	4	0 to 230 steps	±2.5% of total bank travel	All 4 control rod bank positions are recorded along with the low-low limit alarm for each bank.	Control Board	1. One channel for each control bank. 2. An alarm and annunciator is actuated when the last rod control bank to be withdrawn reaches the withdrawal limit, when any rod control bank reaches the low insertion limit, and when any rod control bank reaches the low-low insertion limit.

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TABLE 7.5-3
(Sheet 7 of 8)

Parameter	No. of Channels Available	Range	Indicated Accuracy (1)	Indicator/ Recorder	Location	Notes
CONTAINMENT SYSTEM						
1. Containment Pressure	2 2	0 to 60 psig -5 to 160 psig	±1.8%	All 4 channels indicated and 1 is recorded.	Control Board	
FEEDWATER AND STEAM SYSTEMS						
1. Emergency Feedwater Flow	1/feed line	0 to 300 gpm	later	All channels indicated and recorded.	Control Board	One channel to measure the flow to each steam generator.
2. Steam Generator Level (narrow range)	3/steam generator	0 to 100%	±4% of ΔP level (hot)	All channels indicated. The channels used for control are recorded.	Control Board	
3. Steam Generator Level (wide range)	1/steam generator	0 to 100%	±5% of level (cold)	All channels recorded.	Control Board	
4. Steam Generator Level Signal		+7 to -5 feet	±4%	The one channel is indicated.		
5. Main Feedwater Flow	2/steam generator	0 to 5 x 10^6 lbs/hr	±5%	All channels indicated. The channels used for control are recorded.	Control Board	
6. Magnitude of Signal Controlling Main and Bypass	1/main 1/bypass	0 to 100% of valve opening	±1.5%	All channels indicated.	Control Board	1. One channel for each main and bypass feedwater

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TABLE 7.5-3
(Sheet 8 of 8)

Parameter	No. of Channels Available	Range	Indicated Accuracy (1)	Indicator/ Recorder	Location	Notes
FEEDWATER AND STEAM SYSTEMS (Cont.)						
Feedwater Control Valves						control valve. 2. OPEN/CLOSED indication is provided in the Control Room for each main and bypass feedwater control valve.
7. Steam Flow	2/steam generator	0 to 5 x 10^6 lbs/hr	±5.5%	All channels indicated. The channels used for control are recorded.	Control Board	Accuracy is equipment capability; however, absolute accuracy depends on applicant calibration against feedwater flow.
8. Steam Line Pressure	3/loop	0 to 1300 psig	±4%	All channels indicated and 1 is recorded.	Control Board	
9. Steam Dump Demand	1	0-100% of steam dump valves open	±1.5%	The one channel is indicated.	Control Board	OPEN/CLOSED indication is provided in the Control Room for each steam dump valve.
10 Turbine Impulse Chamber Pressure	2	0 to 860 psig	±3.5%	Both channels indicated.	Control Board	OPEN/CLOSED indication is provided in the Control Room for each turbine stop valve.

(1) Includes channel accuracy and environmental effects

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INDICATION OF BYPASS FOR THE REACTOR PROTECTION SYSTEM IS PROVIDED BY VAS ALARMS. BYPASS OF A REACTOR TRIP BREAKER ~~SB 1 & 2~~ AND BYPASS OF THE PROTECTION SYSTEM OUTPUT ARE ALARMED FOR EACH TRAIN ON THE VAS.

coincidence logic required for reactor trip. Additional information is given in Subsection 7.2.2.2c. 420.49

12. Operating Bypasses

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Where operating requirements necessitate automatic or manual bypass of a protective function, the design is such that the bypass is removed automatically whenever permissive conditions are not met. Devices used to achieve automatic removal of the bypass of a protective function are considered part of the protective system and are designed in accordance with the criteria of this section. Indication is provided in the control room if some part of the system has been administratively bypassed or taken out of service.

13. Indication of Bypasses

~~Bypass indication is discussed in Subsection 7.1.2.6 and Section 1.8.~~

14. Access to Means for Bypassing

The design provides for administrative control of access to the means for manually bypassing channels or protective functions.

15. Multiple Setpoints

For monitoring neutron flux, multiple setpoints are used. When a more restrictive trip setting becomes necessary to provide adequate protection for a particular mode of operation or set of operating conditions, the protective system circuits are designed to provide positive means or administrative control to assure that the more restrictive trip setpoint is used. The devices used to prevent improper use of less restrictive trip settings are considered part of the protective system and are designed in accordance with the criteria of this section.

16. Completion of Protective Action

The protection system is so designed that, once initiated, a protective action goes to completion. Return to normal operation requires action by the operator.

17. Manual Initiation

Switches are provided on the control board for manual initiation of protective action. Failure in the automatic system does not prevent the manual actuation of the protective functions. Manual actuation relies on the operation of a minimum of equipment.

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emergency feedwater system is capable of performing this function for an extended period of time following plant shutdown.

The RHR system is provided with two residual heat removal pumps, and two heat exchangers arranged in two separate, independent flow paths. To assure reliability, each residual heat removal pump is connected to a different vital bus. Each residual heat removal train is isolated from the RCS on the suction side by two motor-operated valves in series with each valve receiving power via a separate motor control center and from a different vital bus. Each suction isolation valve is also interlocked to prevent exposure of the RHR system to the normal operating pressure of the RCS. (See Subsection 5.4.7.2d.)

RHR system operation for normal conditions and for major failures is accomplished completely from the control room. The redundancy in the RHR system design provides the system with the capability to maintain its cooling function even with major single failures, such as failure of an RHR pump, valve, or heat exchanger, since the redundant train can be used for contained heat removal.

Although such major system failures are within the system design basis, there are other less significant failures which can prevent opening of the RHR suction isolation valves from the control room. Since these failures are of a minor nature, improbable to occur, and easily corrected outside the control room, with ample time to do so, they have been realistically excluded from the engineering design basis. Such failures are not likely to occur during the limited time period in which they can have any effect (i.e. when opening the suction isolation valves to initiate RHR operation); however, even if they should occur, they have no adverse safety impact and can be readily corrected. In such a situation, the emergency feedwater system and steam generator power operated relief valves can be used to perform the safety function of removing residual heat and in fact can be used to continue the plant cooldown below 350°F, until the RHR system is made available.

One failure of this type is a failure in the interlock circuitry which is designed to prevent exposure of the RHR system to the normal operating pressure of the RCS (See Subsection 5.4.7.2d). In the event of such a failure, RHR system operation can be initiated by defeating the failed interlock through corrective action at the Solid State Protection System cabinet or by TAKING LOCAL CONTROL at the individual affected motor control centers.

The other type of failure which can prevent opening the RHR suction isolation valves from the control room is a failure of an electical power train. Such a failure is extremely unlikely to occur during the few minutes out of a year's operating time during which it can have any consequence. If such an unlikely event should occur,