PSNH PUBLIC SERVICE
Company of New Hampshire

SEABROOK STATION
Engineering Office:
1671 Worcester Road
Framingham, Massachusetts 01701
(617) - 872 - 8100

January 20, 1983

SBN- 427
T. F. B7.1.2

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

Attention: Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

References: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) PSNH Letter SBN-289, dated July 2, 1982, "FSAR Section 8, Open Item List, Power Systems Branch (Electrical)" J. DeVincentis to F. Miraglia

Subject: Open Item Responses

Dear Sir:

We have enclosed responses to the following open items w 'ch were discussed with representatives of the NRC Staff in a meeting conducted on January 18, 1983.

NRC BRANCH	SRP SECTIoN	COMMENTS
PSB (Electrical)	8.3.1.1.1 - Compliance with position B1 of BTP PSB-1	See attached revised RAI 430.15
PSB (Electrical)	8.3.3.3.1 - Interaction between circuits of different voltage levels	See attached revised RAI 430.149
	Testing of Non-Class 1E protective devices	See attached revised RAI 430.149
PSB (Electrical)	8.3.3.1.1 - Submerged electrical equipment as a result of a LOCA	See attached revised RAI 430.62
PSB (Electrical)	8.3.1.8 - Automatic transfer of loads and electrical interconnections	See attached revised RAI 430.44

B001

The enclosed responses will be included in a future Amendment to the OL Application.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

John DeVincentis

John DeVincentis
Project Manager

ALL/tn

cc: Atomic Safety and Licensing Board Service List

430.44 (8.3.1) (8.3.2) Redundant Buses E5 and E6 are interconnected through Bus E52, MCC 523, Charger ED-13C-2B, 125 V Bus 12B, UPS ED-I-2B and Bus E63. Either eliminate or justify this interconnection.

Provide the results of an analysis that identifies and justifies all electrical interconnections between redundant divisions.

RESPONSE: (11/82) The interconnection described above is imaginary and from a practical standpoint is nonexistent. For this interconnection to exist, multiple failures of diverse equipment would have to occur first, followed by a transformation from a 3-Wire AC System to a 2-Wire DC System and back again to a 3-Wire AC System. The safety divisions which are purported to be interconnected are in addition effectively isolated through the inherent blocking feature between input and output of the chargers, the current limiting feature of the UPS and the charger and the various protective devices. If this imaginary interconnection is to be given any credence, then so must the equally imaginary interconnection that could exist between redundant divisions through the common ground path and ground grid at the station.

We have performed an analysis to identify all other electrical interconnections between redundant divisions and it remains our position that there are no electrical interconnections between redundant divisions.

RESPONSE: (1/83) Although we still believe that the above interconnection does not violate RG 1.6, it has been decided to modify the power supplies to UPS ED-I-2B in order to eliminate the above concern.

A similar concern, as described above, exists with UPS ED-I-4; modification to the power supplies will also resolve this concern.

Our analysis indicates that we have no other electrical interconnections or automatic load transfers that violate RG 1.6.

We have instituted an investigation to verify if we have any violations of the physical independence between redundant divisions or associated circuits. Upon completion of our analysis, the results will be forwarded to you for your evaluation. If violations are identified they will either be justified or they will be corrected.

430.62 Identify all electrical equipment, both safety and non-safety, that may become submerged as a result of a LOCA. For all such equipment that is not qualified for service in such an environment provide an analysis to determine the following:

1. The safety significance of the failure of this electrical equipment (e.g. spurious actuation or loss of actuation function) as a result of flooding.

2. The effects on Class 1E electrical power sources serving this equipment as a result of such submergence; and

3. Any proposed design changes resulting from this analysis.

RESPONSE: All electrical equipment that may become submerged as a result of a LOCA is listed in Table 430.62-1.

TABLE 430.62-1

EQUIPMENT OR COMPONENT OF EQUIPMENT LOCATED IN THE CONTAINMENT BELOW THE FLOOD LEVEL OF (-) 20'-8"

VALVE LIST

TAG	V-VITAL NV - NON-VITAL	EQUIPMENT OR COMPONENT SUBMERGED	APPLICATION
CS-V-59	NV	Solenoid & Limit Switch	RCP LD Seal Water Return
CS-V-145	NV	Solenoid & Limit Switch	Letdown HX-E-2 to HX-E8
CS-V-170	NV	Solenoid & Limit Switch	Letdown HX-E-3 to RCDT
CS-V-175	NV	Solenoid & Limit Switch	Excess Letdown Line
CS-V-176	NV	Limit Switch	Excess Letdown Line
CS-V-177	NV	Limit Switch	HX-E2 to Cold Leg 4
CS-V-180	NV	Solenoid & Limit Switch	HX-E2 to Cold Leg 1
CS-V-185	NV	Solenoid & Limit Switch	HX-E2 to Pressurizer
CS-V-168	V	Motor Operator	RCP Seal Water Isolation
NG-V-17	NV	Solenoid & Limit Switch	Accumulator 9A Ni Line
NG-V-19	NV	Limit Switch	Accumulator 9B Ni Line
NG-V-21	NV	Solenoid & Limit Switch	Accumulator 9C Ni Line
NG-V-23	NV	Limit Switch	Accumulator 9D Ni Line
RC-LCV-459	NV	Solenoid & Limit Switch	Letdown Isolation Valve
RC-LCV-460	NV	Solenoid & Limit Switch	Letdown Isolation Valve
RC-V-81	NV	Motor Operator	RC Loop 3 Letdown to HX-E2
RMW-V-28	NV	Solenoid	RMW-TK 12 to RC TK 11
RMW-V-180	NV	Solenoid	RC-P-1B Seal Pressurizer Equalizing Valve
RMW-V-181	NV	Solenoid	RC-P-1A Seal Pressurizer Equalizing Valve
RH-V-27	V	Solenoid & Limit Switch	HX-E-9B Header Test
RH-V-28	V	Solenoid & Limit Switch	HX-E-9A Header Test
RH-V-49	V	Solenoid & Limit Switch	HX-E-9A Injection Test
RH-V-54	NV	Solenoid & Limit Switch	SI-P-6A Discharge Test
RH-V-55	NV	Limit Switch	SI-P-6B Discharge Test
SI-V-03	V	Stem Mounted Limit Switch	Accumulator Iso. Valve Stem Limit Switch
SI-V-04	NV	Limit Switch	Accum. Test Valve
SI-V-15	NV	Solenoid & Limit Switch	Accum. Fill Valve
SI-V-17	V	Stem Mounted Limit Switch	Accum. Iso. Valve Stem Limit Switch
SI-V-18	NV	Solenoid & Limit Switch	Accum. Test Valve
SI-V-23	NV	Solenoid & Limit Switch	Accum. Fill Valve
SI-V-32	V	Stem Mounted Limit Switch	Accum. Iso. Valve Stem Limit Switch

TABLE 430.62.1 (Continued)

VALVE LIST

TAG	V-VITAL NV - NON-VITAL	EQUIPMENT OR COMPONENT SUBMERGED	APPLICATION
SI-V-33	NV	Solenoid & Limit Switch	Accum. Test Valve
SI-V-38	NV	Limit Switch	Accum. Fill Valve
SI-V-47	V	Stem Mounted Limit Switch	Accum. Iso Valve - Stem Limit Switch
SI-V-48	NV	Solenoid & Limit Switch	Accum Test Valve
SI-V-53	NV	Solenoid & Limit Switch	Accum Fill Valve
SI-V-131	V	Limit Switch	SI - Cold Leg Test
SI-V-132	NV	Limit Switch	SI - Hot Leg 3 Test
SI-V-133	NV	Limit Switch	SI - Hot Leg 2 Test
SI-V-134	V	Solenoid & Limit Switch	SI - Hot Leg Test
SI-V-158	V	Solenoid	Charging Pump Test
SI-V-160	V	Limit Switch	SI Pump - Test Line Iso. Valve
WLD-FV-1403	NV	Solenoid & Limit Switch	RCDT Transfer Valve

TABLE 430.62.1 (Continued)

INSTRUMENTATION LIST

SAFETY-RELATED INSTRUMENTS THAT MAY BECOME SUBMERGED

TAG	DESCRIPTION	ACTION
NI-NE-41 A/B	Power Range Neutron Detectors	Not required post-LOCA
NI-NE-42 A/B	Power Range Neutron Detectors	Not required post-LOCA
NI-NE-43 A/B	Power Range Neutron Detectors	Not required post-LOCA
NI-NE-44 A/B	Power Range Neutron Detectors	Not required post-LOCA
RM-RM-6535A	Manipulator Crane Radiation Monitor	Not required post-LOCA
RM-RM-6535B	Manipulator Crane Radiation Monitor	Not required post-LOCA
RC-FT- 414, 415, 416	RC System Loop 1 - Flow	Will be raised above the flood level
424, 425, 426	RC System Loop 2 - Flow	Will be raised above the flood level
434, 435, 436	RC System Loop 3 - Flow	Will be raised above the flood level
444, 445, 446	RC System Loop 4 - Flow	Will be raised above the flood level

TABLE 430.62-1 (Continued)

NON-VITAL INSTRUMENTS THAT MAY BECOME SUBMERGED

TAG	DESCRIPTION
CAH-TE-5640 - 5647	NI Detector Wall Temp.
CAS-AE-8815	Hydrogen Analyzers in RC TK 55 Area
CAS-AE-8816	Hydrogen Analyzers in RC CS-E-3 Area
CAS-AE-3817	Hydrogen Analyzers in Cntmnt. Valve Rm.
CAS-AE-8818	Hydrogen Analyzers in CS-E-2 Area
CCP-PT-1787	
CS-PT-124	Excess Letdown HX, CS-E-3 Outlet Pres.
CS-TE-126	Regenerative HX, CS-E-2 Charging Line Temp.
CS-FT-154 - 157	RCP Low Range Leakage Flow
CS-FIS-191 - 194	RCP Seal Flow
LD-LT-8333	Sump B Level
RC-LT-9405	Press. Relief Tank Level
RM-RX-6578 - 6581	Radiation Monitor
SF-LT-2629	Refueling Canal Level
SM-XS, XT-6701	Seismic Monitor
SM-XS, ST-6709	Seismic Monitor
WLD-LT-1403	RCDT, TK 55 - Level
WLD-TE-1403	RCDT, TK 55 - Temp.
WLD-FT-1406	RCDT, TK 55 - Flow
WLD-FT-1411	RCDT, EX E-43 Inlet Temp.
WLD-TE-1413	RCDT, EX E-43 Outlet Temp.
WLD-PT-1412	RCDT, EX E-43 Pump P33 A/B Discharge Pres.
WLD-PT-1420	RCDT, EX E-43 Pump P33 A/B Section Pres.
WLD-LSH-6266	Containment Drains Sump A Level
WLD-LSH-6267	Containment Drains Sump B Level

TABLE 430.62-1 (Continued)

MISC. NON-VITAL EQUIPMENT THAT MAY BECOME SUBMERGED

PUMPS

TAG	DESCRIPTION
RC-P-271	Pressurizer Relief Tank, TK 11, Recirc. Pump
SF-P-272	Refueling Canal Drain Pump
WLD-P-5A, 5B	Containment Sump A Pumps
WLD-P-5C, 5D	Containment Sump B Pumps
WLD-P-33A, 33B	RCDT, TK 55, Pumps

INSTRUMENT RACKS

Individual instruments located on these racks have been identified above.

CONTROL PANELS

TAG	DESCRIPTION
WLD-CP-280	Containment Sump A - Control Panel
WLD-CP-281	Containment Sump B - Control Panel

LIGHTING

TAG	DESCRIPTION
ED-X-16F	Transformer Supply for Panel L17
ED-X-16A	Transformer Supply for Panels PP-8B and EL 13
ED-X-16J	Transformer Supply for Panel L41
Panel L41	Lighting Panel

TERMINAL BOXES

TAG	DESCRIPTION
X45	Pressurizer Heater Backup Group C
X46	Pressurizer Heater Backup Group D

1. The following analysis discusses the safety significance of the failure as a result of flooding of the electrical equipment listed in Table 430.62-1.

   A. Valves - Safety-Related

      (1) Stem-mounted Limit Switches for SI Accumulator Valves

      Stem-mounted limit switches only provide an alternate valve position indication. Failure of these switches could cause loss of the alternate valve position indication circuits but would not affect valve operation; valve position indication will still be provided by the normal valve limit switches.

      (2) RCP Seal Water Isolation Valve, CS-V-168

      This motor-operated valve is driven closed upon a containment isolation signal, therefore, this valve would fail in its safe position. Submergence will not change the fail-safe position. Motor-operated valves are powered from individual circuits of a motor control center so failure of this circuit would not affect the remaining loads on the motor control center. Failure of the limit switch internal to the operator could effect the valve position indication at the control switch on the MCB and also the post-accident monitor (PAM) light indication of this valve.

      This circuit will be modified by adding an interposing relay to the circuit of CS-V-168 . The relay will be located in the control building thus not affected by flood and will prevent loss of the remaining PAM monitor circuits upon flooding of the valve.

      (3) SI Pump Cold and Hot Leg Test Line Isolation Valves SI-V-131, and SI-V-160 and Charging Pump Test Line Isolation Valve, SI-V-158

      These Train B air-operated valves are closed on a containment isolation signal and submergence will not alter their position.

      Loss of power on this circuit will also cause loss of power to two other safety-related valves, SI-V-174 and SI-V-70, which are powered from the same circuit. These valves which are not subject to submergence are already closed on safety injection or containment isolation signals, therefore loss of power results only in loss of valve position indication at the MCB control

switches. The limit switches for valves SI-V-131, V-158 and V-160 are also used in the Train B PAM monitor light circuits, and this circuit may also be lost.

These circuits will be modified by adding interposing relays to the circuits of SI-V-131, SI-V-158 and SI-V-160. The relays will be located in the control building thus not affected by flood and will prevent loss of the remaining Train B PAM monitor circuits upon flooding of the above valves.

(4) SI Hot Leg Test Line Isolation Valves, SI-V-134

This valve is a normally closed test valve that also receives a containment isolation signal to ensure that it is closed. Valve SI-V-134 is a Train A valve, and loss of power on this circuit, because of submergence, will cause loss of power to four other safety-related valves which are not subject to submergence, SI-V-165 and V-173 (already closed on safety injection) and SI-V-62 and V-157 (already closed on containment isolation). Valve position indication at their respective MCB control switches will be lost.

Limit switch contacts for SI-V-134 are also used in the PAM monitor light circuits. An interposing relay will be added to SI-V-134 circuit to prevent loss of the remaining PAM monitor circuits upon flooding of SI-V-134.

(5) RHR Test Valves RH-V-27 and RH-V-49

These valves are normally closed Train A test valves that receive a containment isolation signal. The open contact of the containment isolation signal isolates the solenoids. If the containment isolation signal is reset, the circuit for RH-V-27 and V-49 could fail resulting in loss of power. This circuit failure would also cause loss of power to safety-related valve RH-V-16, and solenoids for FY-618-1 and HCV-606. Valve RH-V-16 is already closed on containment isolation. Loss of power to FY 618-1 and HCV-606 solenoids will result in full flow of the RH System A loop through the RHR heat exchanger E-9A and the closing of the bypass line. This scenario is addressed in response to RAI 440.133. Valve position indication for these valves at the MCB control switches RH-V-27 and V-49 will be lost.

Limit switch contacts for RH-V-27 and RH-V-49 are also used in PAM monitor light circuits. Interposing relays will be added to these valve circuits to prevent loss of the remaining PAM monitor circuits upon flooding of the above valves.

(6) RHR Test Valve RH-V-28

Valve RH-V-28 is a normally closed Train B test valve that also receives a containment isolation signal. The open contact of the containment isolation signal isolates the solenoid. This valve is similar to RH-V-27 (above) and failure of this circuit would result in loss f power to safety-related valve RH-V-17 and sol noids for FY-619-1 and HCV-607. This wi l resu in full flow through RHR heat exchanger E-9B. Valve position indication for these valves at the MCB control switches will be lost.

Limit switch contacts for RH-V-28 are also used in PAM monitor light circuits. An interposing relay will be added to this valve circuit to prevent loss of the remaining PAM monitor circuits upon flooding of the above valve.

B. Valves - Non-Safety

Non-safety-related, air-operated valves that may be submerged following LOCA, are listed in Table 430.62-1. Failure of the stem-mounted limit switch or pilot solenoid may cause the entire valve circuit to lose power with the results that any other non-safety-related valves on that particular circuit may de-energize to their fail safe position. Valve position indication will also be lost.

A non-safety-related motor-operated valve that may be submerged is listed in Table 430.62-1. This MOV is supplied from a motor control center which does not feed any safety-related loads.

C. Instrumentation

All safety-related instrumentation, with the exception of the excore neutron detectors and maniulator crane radiation monitors is located or will be relocated above the flood level. The excore neutron detectors and manipulator crane radiation monitors are not required following a LOCA.

Non-safety-related instrumentation located below the flood level may fail. None of this instrumentation is required following a LOCA.

D. Miscellaneous Non-Safety-Related Equipment

(1) Pumps

Those pump motors that may become submerged and fail are either protected by redundant Class 1E breakers or are de-energized during normal plant operation. Failure of the particular circuit will not affect other circuits. These non-safety-related motors are not required following a LOCA.

(2) Control Panels

The sump pump control panels are not required following a LOCA.

(3) Lighting

The lighting system inside containment is normally off. Control of the system is at control stations located outside the personal air lock. Failure of lighting equipment due to flooding will not affect other circuits.

(4) Pressurizer Heater Terminal Boxes

Flooding of the terminal boxes for pressurizer heater backup groups C and D, will cause de-energization of these heaters. Pressurizer heater backup groups A and B which are part of the NUREG-0737 requirements, will not be affected because their terminal boxes are above the flood level. Backup groups A and B are fed from the diesel generators.

The above analysis demonstrates that there is no safety significance from the failure of the equipment described above as a result of flooding. In areas where safety concerns were identified, the modifications mentioned above will be implemented to alleviate these concerns.

2. A. There is no detrimental effect on the Class 1E electric power sources as a result of equipment submergence.

All submerged equipment can be divided into the following three categories:

Category 1: Class 1E load supplied by a Class 1E power supply.

Category 2: Non-Class 1E load supplied by a Class 1E power supply.

Category 3: Non-Class 1E load supplied by a non-Class 1E power supply.

For Category 1 and 2 equipment, Class 1E power supply is protected by the Class 1E protective device which feeds the circuit of the submerged equipment. Only the circuit which feeds the submerged equipment is de-energized. For the loads that are connected to Class 1E power supplies, analysis of the protective devices has shown that in the unlikely event of simultaneous failure of these loads during submergence, the Class 1E power supply will not be lost.

For Category 3 equipment there is no connection to the Class 1E power supply; therefore, there is no effect due to submergence.

Instrumentation

Safety-related and non-safety-related instruments are powered through separate instrumentation panels. These panels are powered from separate distribution circuits. Protective devices of the internal power supplies further isolate the individual circuits from the distribution system. Therefore, the Class 1E power sources will not be affected by submergence of any instrumentation.

Miscellaneous

(1) Pumps

The non-vital pumps that may be submerged are powered from the non-Class 1E power system. Failure of these motor circuits will not affect the Class 1E sources.

(2) Control Panels

The control panels for the containment sump pumps are powered from non-Class 1E sources and their failure will not affect the Class 1E system.

(3) Lighting System

The lighting system inside the containment is normally de-energized during plant operation; the part of the lighting system that might be submerged is powered from Motor Control Centers that do not carry safety-related loads.

(4) Pressurizer Heater Terminal Boxes

Backup heater groups C and D are not powered from the Class 1E power system and their failure will not affect the Class 1E system.

Proposed Design Changes

Safety-Related Valves

As discussed in Item 1, interposing relays will be added to certain valve circuits to electrically isolate the valve PAM monitor light circuit from the valve limit switch circuit. This modification prevents submergence of the valve limit switch from tripping the entire PAM monitor light circuit. No additional changes are proposed.

Non-Safety-Related Valves

No design changes are proposed.

Instrumentation

Safety-related reactor coolant flow transmitters will be raised above the flood level.

No additional design changes are proposed.

Miscellaneous Equipment

No design changes are proposed.

RAI 430.149

Provide the results of an analysis to prove that any challenges to Class 1E Circuits from associated circuits do not prevent the safe shutdown of the plant.

RESPONSE

A. General

The Seabrook Station complies with the requirements of FSAR Appendix 8A, IEEE 384-1974 and Regulatory Guide 1.75, Rev. 2. These documents describe acceptable methods of complying with IEEE 279-1971 and Criteria 3, 17, and 21 of Appendix A to 10 CFR Part 50 with respect to the physical independence of the circuits and electrical equipment comprising or associated with the Class 1E power system, the protection system, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection system and the systems it actuates to perform their safety-related functions.

In accordance with the provisions of Section 4.5a and 4.6.2 of FSAR Appendix 8A, Sections 4.5(1) and 4.6.1 of IEEE 384-1974, and Position C4 of Regulatory Guide 1.75, Revision 2, we have elected to associate all the Non-Class 1E circuits with Class 1E circuits. This application of associated circuits allows the plant to be designed with one less separation group; that is, instead of having five separation groups consisting of four safety-related separation groups and one non-safety-related separation group, Seabrook has only four separation groups. The major advantages of this approach are the ability to provide greater separation distances between the groups as well as to reduce the raceway system's exposure to fire.

As a result of this design, all plant circuits are specifically assigned to one of the following four separation groups as noted in Figure 1.

Group A - Train A, Channel I and Train A Associated Circuits
Group B - Train B, Channel II and Train B Associated Circuits
Group C - Channel III
Group D - Channel IV

The great majority of associated circuits are with Group A, a very limited number are with Group B, and none are with Groups C and D.

The circuits that are associated with Train A consist of:

1. Non-Class 1E power, control, and instrument circuits contained within the Nuclear Island.

2. Non-Class 1E power, control, and instrumentation circuits that traverse the Nuclear Island boundary.

3. Non-Class 1E power, control, and instrument circuits outside the Nuclear Island.

The circuits that are associated with Train B consist of:

1. Non-Class 1E power, control, and instrument circuits contained within the Nuclear Island.

2. Non-Class 1E power, control, and instrumentation circuits that traverse the Nuclear Island boundary.

The Nuclear Island boundary is shown in Figure 2. This figure denotes the buildings, structures, duct banks, etc., which are part of the Nuclear Island. All other buildings, structures, etc., are considered to be outside the Nuclear Island.

The following analysis examines the design features and modes of failure of associated circuits of each separation group to determine any interaction and challenges with other separation groups. The overall objective is to assure that the ability to achieve a safe plant shutdown under Design Basis Event (DBE) conditions is not compromised.

B. <u>Train A Associated Circuit Analysis</u>

1. <u>Associated Circuits Contained within the Nuclear Island</u>

Non-Class 1E circuits that remain within the Nuclear Island are permitted to share the same raceway as Train A Class 1E circuits. These circuits are classified as Train A Associated Circuits and are designed and installed to meet all the requirements placed on associated circuits as required by the compliance documents listed earlier.

Challenges to Class 1E circuits, because of failure in an associated circuit, have been examined and determined to have no detrimental effect because:

a. When Class 1E power supplies are utilized, failure of a Non-Class 1E motor, load, or device connected to this power supply will be promptly isolated by operation of Class 1E protective devices.

Non-Class 1E loads connected to Class 1E buses are in all cases protected by Class 1E devices. The breakers protecting Non-Class 1E loads are coordinated such that failure of all Non-Class 1E loads, with proper operation of their own breakers, will not result in tripping of the incoming breaker to the bus.

Further, in the few cases where credit is taken for the incoming bus feeder breaker to provide backup protection to meet Regulatory Guide 1.63, the associated bus is dedicated to Non-Class 1E loads only and therefore will not degrade a Class 1E bus.

b. In cases where Non-Class 1E power supplies, such as switchgear, motor control centers, and distribution panels, are utilized, these are of identical design to the Class 1E counterparts and

have been purchased to the same specification requirements inclusive of quality control. Mounting of the Non-Class 1E power supplies within the Nuclear Island is identical to the mounting of their Class 1E counterparts; therefore, credit can be taken for this equipment to function under DBE conditions.

c. All Non-Class 1E protective circuit breakers will be periodically inspected approximately once every five years according to a program developed for the inspection of Non-Class 1E equipment. This program will be in accordance with manufacturer's recommendations for maintenance and inspections.

   Since Class 1E and Non-Class 1E protective devices are identical, any generic degradation such as setpoint drift, manufacturing deficiencies, and material defects will be detected and corrected as a result of the rigorous program performed on the Class 1E protective devices to satisfy the requirements of ANSI N-18.7-1976 and Regulatory Guide 1.63; therefore, credit can be taken for this equipment to function under DBE conditions.

d. The probability of an ensuing fire is minimized because all cables utilized for these associated circuits are specified, designed, manufactured, and installed to the same criteria as Class 1E cables. Factors that have been taken into consideration include flame retardancy, non-propagating and self-extinguishing properties, splicing restrictions, appropriate limitations on raceway fill, appropriate cable derating, and environmental qualifications.

e. Degradation of an associated circuit because of a raceway failure during a DBE, has been eliminated because all electrical raceway systems within the Nuclear Island are seismically analyzed.

f. Other design considerations that contribute to the integrity of these associated circuits are:

   - Cables associated with one train are never routed in raceways containing Class 1E or associated cable of another train or channel.

   - All cables for instrumentation circuits utilize shielded construction which minimize any unacceptable interaction between Class 1E and associated circuits.

   - All circuits entering the reactor containment are provided with protective devices complying with Regulatory Guide 1.63.

Based on the above design features and analysis, we do not consider these associated circuits to pose any challenges to any Class 1E circuits. Therefore, the ability for safe plant shutdown under DBE conditions has not been jeopardized.

2. Train A Associated Circuits That Traverse the Nuclear Island Boundary

For analysis purposes, the associated circuits that traverse the Nuclear Island boundary can be further subdivided into two basic types: (a) those that have their protective device located in the Nuclear Island and (b) those that have their protective device outside the Nuclear Island. It should be noted that there are a limited number of power cables in these categories.

a. Associated Circuits That Have Protective Device Located in the Nuclear Island

These circuits are also designed and installed to meet all the requirements as outlined in Sections B.1a, b, c, d and f. Though the raceway system outside the Nuclear Island is not seismically analyzed, this is of no concern because the circuit protective devices inside the Nuclear Island are assumed to perform their protective function.

Concerns that design basis events such as a seismic event may cause high voltage cables that are not in seismically analyzed raceways and not located in Category I buildings to interact with lower voltage cables are analyzed below:

Recent seismic tests performed on raceways representing typical installations on SEP plants, proved that the raceways can withstand seismic events with no significant failures. Since the typical non-seismic installation at Seabrook is superior to the tested SEP installations, it can be assumed that they will survive a seismic event. Failures of raceways resulting from collapse of the non-seismically designed buildings can be dismissed because the conservative criteria and UBC seismic loading used in the construction of the building will ensure little likelihood of collapse.

Not withstanding the preceding, any event involving the raceway system that can cause a higher voltage cable to come in contact with another lower voltage cable will first cause the higher voltage cable to be grounded. Contributing factors to this are: 1) the cables are in grounded metallic trays or enclosures, 2) the 13.8 kV and 4.16 kV power cables are of armored construction, and 3) as indicated in FSAR Section 8.3.1.4a, separate raceways are designated for the different voltage levels.

A ground fault in the low resistance grounded 13.8 kV system will cause protective circuit breakers to open and isolate the fault. In the high resistance grounded 4 kV and 480 V system, although a single ground fault will not cause circuit breaker operation, it is highly probable that under such a failure, the faults will be such that will cause breaker operation.

Although the protective devices might not be in Category I buildings, they are of identical design to the Class 1E devices and based on operating experience of protective devices that

have been subject to actual and simulated seismic conditions, it is highly probable that the protective devices will maintain its structural integrity and perform its function.

In view of the above design considerations and analysis, any possible interaction between cables of different voltage levels is deemed nonexistent.

It is therefore concluded that the ability for safe plant shutdown under DBE conditions has not been jeopardized by these associated circuits.

b. Associated Circuits That Have Protective Device Outside the Nuclear Island

Non-Class 1E switchgear, motor control centers and distribution panels outside the nuclear island have been purchased to the same specification requirements as their Class 1E counterparts; therefore the probability of failure under DBE conditions is greatly minimized.

These protective devices which are not located in a LOCA environment are identical to Class 1E devices except for seismic requirements. Other design basis events such as pipe break, fire, flood, etc., will not cause failure of the protective device located outside the nuclear island simultaneously with the failure of load which is located in the nuclear island. Hence, credit can be taken for their proper operation.

However, if one postulates their misoperation under a seismic event, such an event is likely to disable the power source itself which is neither seismically qualified.

Analysis of concerns on interaction between cables of different voltage levels is shown in (a), above.

We conclude, therefore, that these circuits will not degrade Class 1E circuits, since the Non-Class 1E power supply is lost and all Non-Class 1E equipment becomes de-energized.

For the above reasons, the ability for the safe plant shutdown under DBE conditions has not been jeopardized by these circuits.

3. Train A Associated Circuits Outside the Nuclear Island

The design features, analyses, and conclusions listed under Section B.2.b are applicable to all these circuits.

C. Train B Associated Circuit Analysis

1. Associated Circuits Contained Within the Nuclear Island

Non-Class 1E circuits that remain within the Nuclear Island are permitted to share the same raceways as Train B Class 1E circuits.

These circuits are classified as Train B Associated Circuits and are designed and installed to meet all the requirements placed on associated circuits as required by the compliance documents listed earlier. Challenges to Class 1E circuits because of failure in an associated circuit will have no detrimental effect because all Train B power supplies utilized by these circuits such as motor control centers, distribution panels, etc., and their protective devices are Class 1E equipment. Failure of a Non-Class 1E motor, load, or device connected to the Class 1E power supply will be promptly isolated by operation of a Class 1E protective device. Therefore utilizing the analysis performed for Train A associated circuits we conclude that the ability for the safe plant shutdown under DBE conditions has not been jeopardized by these circuits.

2. <u>Associated Circuits That Traverse the Nuclear Island Boundary</u>

For analysis purposes, the associated circuits that traverse the Nuclear Island boundary can be further subdivided into two basic types: (a) those that have the protective devices located in the Nuclear Island and (b) those that have their protective devices outside the Nuclear Island.

a. <u>Associated Circuits That Have Protective Devices Located in the Nuclear Island</u>

There are very few Train B associated circuits that traverse the Nuclear Island boundary. These circuits are unavoidable either because of plant design constraints, such as the need for interlocks and permissives for the preferred power supply circuits to Train B emergency buses, or because of features provided to improve plant reliability, such as power supply and control for the station service air compressors fed from Train B buses. The portion of these circuits which are outside the Nuclear Island are routed in dedicated embedded or exposed conduits; therefore, the potential of harmful interactions with other associated circuits or other voltage level cables is minimized. For applicable details on the interaction between cables of different voltage levels, see analysis under B2a.

The design features described in Section C.1 for associated circuits contained within the Nuclear Island are also applicable to these circuits. Though the conduit system outside the Nuclear Island is not seismically analyzed, this is of no concern because the circuit protective devices located in the Nuclear Island are assumed to perform their protective function.

Based on the above, we conclude that the ability for the safe plant shutdown under DBE conditions has not been jeopardized by these few circuits.

b. Associated Circuits That Have Protective Devices Outside the Nuclear Island

The only circuits under this category are the 15 kV cables to the reactor coolant pumps for motor feeders and potential transformers. These interlocked armor cables are routed in embedded conduit outside the Nuclear Island and are in dedicated seismically analyzed raceway systems in the Nuclear Island. Furthermore, the portion of the circuit entering the containment is protected by qualified fuses located in the electrical penetration area which would open the circuit in the event of a failure of a reactor coolant pump.

The 15 kV cables used on these circuits meet all the construction and material requirements placed on the 5 kV Class 1E cables, i.e., flame retardancy, etc., but do not have documented LOCA/MSLB qualifications.

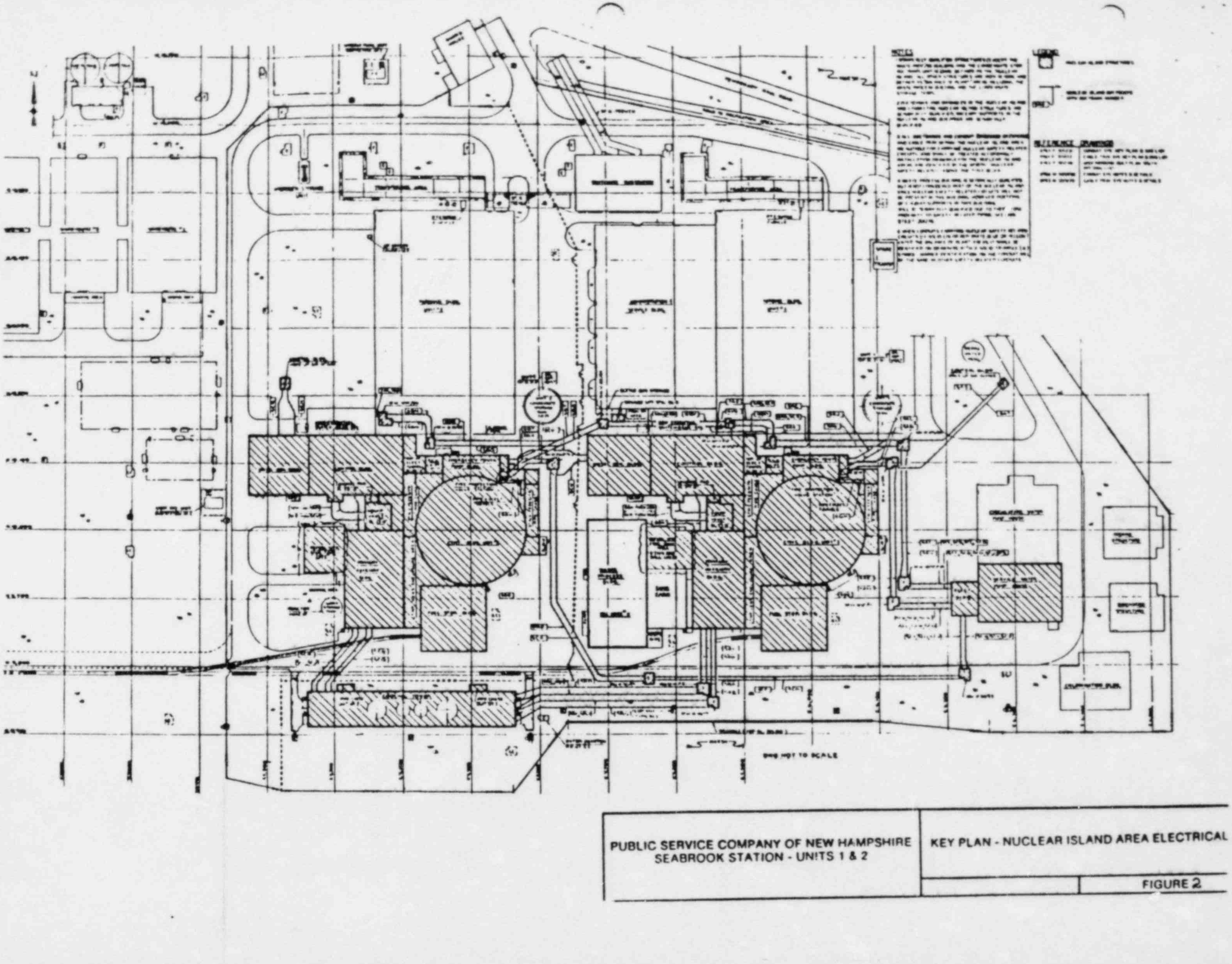
Based on the above design features, we conclude that these circuits do not pose any challenges to Class 1E circuits.

D. Group C and D Circuits

Separation Groups C and D, which are comprised of circuits for Channels III and IV, do not have any associated circuits. Since these channels meet all requirements as defined in the compliance documents listed earlier, these channels are not susceptible to any challenges from any associated circuits; therefore, the ability for the safe plant shutdown under a DBE cannot be jeopardized.

E. Results of Analysis

Based on the above analysis and the fact that the associated circuits for both trains are designed and installed according to the requirements of IEEE 384-1974, FSAR Appendix 8A and Regulatory Guide 1.75, Rev. 2, it is concluded that the reliability of both trains has not been compromised and that under any Design Basis Event the safe shutdown capability of the plant has not been impaired.



PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE SEABROOK STATION - UNITS 1 & 2	KEY PLAN - NUCLEAR ISLAND AREA ELECTRICAL
	FIGURE 2

CABLE SEPARATION GROUPS

CHANNEL and/or TRAIN	PURPOSE	INSULATION LEVEL	A	B	C	D
	POWER	15000 V	○ ⊔	○ ⊔		
	POWER	5000 V	○ ⊔	○ ⊔		
	POWER	600 V ①	○ ⊔	○ ⊔		
	POWER	600 V ②	○ ⊔	○ ⊔		
	POWER	600 V ③	○ ⊔			
	CONTROL	600 V	○ ⊔	○ ⊔	○ ⊔	○ ⊔
	LOW LEVEL INSTRU.	300 V	○ ⊔	○ ⊔	○ ⊔	○ ⊔
	LOW LEVEL INSTRU.	SPECIAL	○	○	○	○
		CIRCUIT TYPES	TR A CH I A Associated	TR B CH II B Associated	CH III	CH IV
CABLE JACKET COLOR		Class IE	Red	White	Blue	Yellow
CABLE JACKET COLOR		Associated	Black/Red	Black/White		

NOTES:

① For 480 VAC, 120 VAC & 125 VDC feeders requiring cables 4/0 AWG and larger.

② For 480 VAC, 120 VAC & 125 VDC feeders requiring cables 2/0 AWG and smaller.

③ Reserved for Control Rod Drive power feeders only.

FIGURE 1

430.15 Section 8.3.1.1.b.4.b of the FSAR indicates that activation of the second level of undervoltage protection generates only an alarm when there is no accident signal and generates an off-site power source trip signal when there is an accident signal. Based on this information, it appears that the design does not meet Position 1 of Branch Technical Position PSB-1. Justify non-compliance.

RESPONSE: We are providing an acceptable alternative to Position 1 of Branch Technical Position PSB-1. This alternative design is described in our letter WYR-80-83 to the USNRC which is included as Attachment 430.15-1.

Letter WYR-80-83 describes Yankee Atomic's generic approach to degraded voltage protection and is applicable to Yankee Rowe, Vermont Yankee, Maine Yankee and Seabrook Station.

The Yankee Atomic approach has been accepted by the NRC staff. Attachment 430.15-2, USNRC letter to YAEC, dated September 27, 1982, documents the NRC's acceptance of our approach.

The Safety Evaluation attached to the letter concludes that our proposal to use operator action to disconnect the Class 1E buses from a degraded off-site power source under non-accident conditions is justified because YAEC has shown that adequate safety systems, which are not exposed to or rendered inoperable by degraded grid voltage, are available to place and maintain the plant in a safe shutdown condition.

As part of the response to RAI 420.38, Seabrook has submitted a list of systems and equipment used for safe shutdown. The equipment in the list will be available in the event of degraded grid voltage because of one or more of the following reasons:

1. Not powered by the degraded power source;

2. Does not rely on electric power;

3. In standby and, therefore, not connected to the degraded source;

4. Equipment will continue to run unimpeded under degraded voltage conditions because of margin between equipment rating and duty;

5. Not sensitive to degraded grid voltage (resistive load); or

6. Time is available for corrective action.

The equipment in this list will not be exposed to or rendered inoperative by degraded voltage and, therefore, would be available to place the plant in a safe shutdown status under non-accident conditions.

In addition to approving the YAEC degraded voltage protection design, the conclusion section of the letter, dated September 27, 1982, required YAEC to submit three additional items in order to finalize acceptance of the design. For Seabrook, these three items are addressed as follows:

1. Design details and a description of the operation of the proposed load shedding bypass circuitry and how this feature will be reinstated on a diesel generator breaker trip are found in FSAR Section 8.3 and in the response to RAI 430.14.

2. The setpoints and tolerances, limiting conditions operation and surveillance testing for the undervoltage protective relaying system are included in the Technical Specifications.

3. Plant operating procedures to cover operator actions for degraded grid under non-accident conditions will be available.

We conclude that the Seabrook design meets the requirements of Position 1 of Branch Technical Position PSB-1 by providing a degraded voltage protection system which protects equipment required to mitigate an accident and assures that adequate equipment will be available for safe shutdown of the reactor.

ATTACHMENT 430.15-1

Telephone 617 366-9011

TWX 710-380-0739

YANKEE ATOMIC ELECTRIC COMPANY

B.3.2.1
WYR 80-83

20 Turnpike Road Westborough, Massachusetts 01581

July 24, 1980

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

Attention: Office of Nuclear Reactor Regulation
Mr. T. A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

References:
1. License No. DPR-3 (Docket No. 50-29)
2. License No. DPR-28 (Docket No. 50-271)
3. License No. DPR-36 (Docket No. 50-309)
4. Docket No. 50-443 and 50-444
5. USNRC Letter, dated 8/12/76 (typical)
6. USNRC Letter, dated 6/3/77 (typical)
7. USNRC Letter, dated 10/16/79
8. VYNPC Letter No. WVY-76-114, dated 9/16/76 (typical)
9. VYNPC Letter No. WVY-77-65, dated 7/18/77 (typical)
10. VYNPC Letter No. WVY-79-139, dated 12/6/79

Subject: Mitigating the Effects of Grid Degradation on Safety Related Electrical Equipment

Dear Sir:

This letter is being written by Yankee Atomic Electric Company on behalf of the Yankee Rowe, Vermont Yankee, Maine Yankee, and Seabrook nuclear stations. These facilities have been identified as References (1, 2, 3 and 4).

BACKGROUND INFORMATION

The NRC position on degraded grid voltage (Reference 5, 6 and 7) requires automatic disconnection of the supply from the grid (offsite power supply) to the plant emergency buses any time the voltage drops below a pre-determined limit. The NRC is concerned that a sustained variation outside the safety related equipment's design rated limit could result in a loss of capability if the equipment were simultaneously required to perform its safety function.

Yankee Atomic has steadfastly opposed the NRC's position on degraded grid voltage because we believe that any changes made in equipment or circuitry

~~8007280439~~ PDR

should increase rather than decrease the level of overall nuclear plant safety; it is our opinion that in attempting to resolve one safety concern additional safety concerns should not be introduced in the process.

Yankee Atomic has proposed an alternative position (References 8, 9 & 10) which relies on operator action in lieu of an automatic trip to resolve the NRC's basic concerns on degraded grid voltage. On receipt of our letter dated 12/6/79 (Reference 10), the NRC requested a meeting to discuss Yankee Atomic's generic position on grid undervoltage. The meeting was held at the NRC offices on May 5, 1980.

At the above meeting, Yankee Atomic engineers acknowledged that the NRC's concerns for continued operation of safety related equipment under degraded voltage conditions were valid, but, stated that they could not ignore the fact that additional safety concerns were being introduced by the NRC position. These safety concerns were categorized into three areas:

a. Violations of GDC-17,
b. Disintegration of the entire grid,
c. Being left with a less reliable source of power or no source of power.

It was pointed out that the Yankee Atomic position (References 8, 9 & 10) relied on the station operator to assess the situation relating to grid degradation and to take appropriate action to ensure that degradation was being corrected. Failing this he would take ad 'tional steps to protect safety-related equipment from the influence of degraded grid voltage. The operator action would ensure that a further deterioration of safety would not result from any action directed at correcting the degradation.

At the meeting, the NRC stated that they had an ongoing concern with operator action; they did not have confidence in operator action and for that reason were opposed to placing any reliance on it.

As one alternative (henceforth known as Alternative 1), the NRC suggested that we consider interlocking the automatic trip with an accident signal. An automatic trip of the offsite power supply would then result only if a simultaneous grid degradation and an accident occurred.

Another alternative (henceforth known as Alternative 2), suggested by the NRC for our consideration subsequent to the meeting, was that we interlock the automatic trip with a signal indicating that the main generator was off-line. An automatic trip of the offsite power supply would then result only if grid degradation occurred when the generator was not synchronized to the grid.

Both the above alternatives assumed that manual operator action would be utilized in modes when the automatic trip was not applicable.

DISCUSSION:

We have considered the two NRC suggestions and have analyzed the merits of each scheme. These alternatives have then been compared with both the original NRC position (Reference 5, 6 & 7) and the Yankee Atomic position (References 8, 9 & 10) on degraded grid voltage.

Alternative 1

This alternative requires that the offsite power circuit breaker be automatically tripped if a simultaneous grid degradation and accident occurred.

The circuit breaker connecting offsite power to the emergency bus is shown in Figure 1. The first level undervoltage relay is shown as device 27A. The second level undervoltage relay is shown as device 27B. Both relays sense voltage on the emergency bus.

When voltage is degraded below that required to ensure the continued operation of safety-related equipment, the second level voltage relay 27B will be activated. Contacts of relay 27B will close in the breaker trip circuit as well as in the alarm circuit. The breaker will trip automatically if an accident signal is also received.

If bus voltage is severely degraded or lost altogether, the first level voltage relay 27A will also be activated. Contacts of relay 27A in the breaker trip circuit will cause an instantaneous trip of the circuit breaker.

With Alternative 1, a grid degradation experienced without an accident signal will only cause an alarm. Established plant procedures require the operator to take specific steps to assess the magnitude and expected duration of the grid degradation. If he is not assured that the disturbance is transitory, and that recovery is imminent, he may choose to manually trip the offsite power circuit breakers after ensuring that a further deterioration of safety will not result from his proposed action.

Advantages

1. Violations of GDC 17 are precluded.

2. An accident signal would itself cause a trip of the main generator; therefore, any resulting collapse or disintegration of the grid could not be attributed to this circuit modification.

3. The reliance on operator action in the event of a simultaneous accident and degraded grid condition is avoided.

4. Operator action would be maintained for all non-accident conditions, thus precluding our concerns expressed in References (8, 9 and 10).

Disadvantages

1. If the reactor is at power and all onsite ac power is determined to be unavailable, (i.e. all diesel generators lost) the reactor will be brought to a cold shutdown conditfon in accordance with technical specification requirements. If, during this mode, an accident signal and a grid degradation were to occur, the offsite power supply breaker would trip leaving the plant with a total loss of all onsite and offsite ac power.

This scenario is being identified in spite of its extremely low probability because firstly, the basis for the NRC position on grid degradation is to design for a simultaneous accident and grid degradation, and secondly, loss of all onsite ac power has occurred at a number of facilities. The combined probability, however, remaining extremely low.

Alternative 2

This alternative requires that the offsite power circuit breaker be tripped automatically if a grid degradation occurs when the generator is not connected to the grid.

The circuit breaker connecting offsite power to the emergency bus is shown in Figure 2. The first level undervoltage relay is shown as device 27A. The second level undervoltage relay is shown as device 27B. Both relays sense voltage on the emergency bus.

When voltage is degraded below that required to ensure the continued operation of safety-related equipment, the second level voltage relay 27B will be activated. Contacts of relay 27B will close in the breaker trip circuit as well as in the alarm circuit. The circuit breaker will trip automatically if it is determined by a logic circuit that the generator is not connected to the grid.

If bus voltage is severely degraded or lost altogether, relay 27A will also be activated. Contacts of relay 27A in the breaker trip circuit will cause an instantaneous trip of the circuit breaker.

With Alternative 2, a grid degradation experienced with the generator connected to the grid will only cause an alarm. Established plant procedures require the operator to take specific steps to assess the magnitude and expected duration of the grid degradation. If he is not assured that the disturbance is transitory, and that recovery is imminent, he may choose to manually trip the offsite power circuit breakers after ensuring that a further deterioration of safety will not result from his proposed action.

Advantages

1. Violations of GDC-17 are precluded.

2. Disintegration of the entire grid is precluded.

3. This alternative prevents the second level undervoltage relay from automatically tripping the offsite power circuit breaker during plant operation. Our concerns expressed in References (8, 9 and 10) relating to a plant trip are thereby removed.

Disadvantages

If the reactor is at power and all onsite ac power is determined to be unavailable (i.e. all diesel generators lost), the reactor will be brought to a cold shutdown condition in accordance with technical specification requirements. Once the generator is disconnected from the grid, this circuit

will cause the circuit breaker to trip if a grid degradation also occurs; the plant will now face a total loss of all onsite and offsite ac power. This situation is extremely undesirable because of the unpredictable consequences of this transient.

CONCLUSION

We have carefully analyzed four possible methods of mitigating the effects of grid degradation on safety related equipment. These methods were:

(a) NRC Position
(b) Yankee Atomic Position
(c) Alternative 1
(d) Alternative 2

Of these four methods, we believe Alternative 1 is the most desirable scheme for our facilities. Additionally, the low probability disadvantages of Alternative 1 are outweighed by the advantages. We, therefore, propose to adopt Alternative 1 for mitigating the effects of a grid degradation. The sensors, and circuit to be utilized will be as detailed in Figure 1, and as described in the text above.

PROPOSED ACTION AND SCHEDULE

We are assuming your continued endorsement of Alternative 1, and will therefore immediately commence engineering changes to incorporate this modification for our Yankee Rowe, Vermont Yankee, and Maine Yankee facilities. It is anticipated that engineering for these changes will be completed by November 1980. Installation will follow at the first opportune shutdown following completion of engineering and receipt of materials. Similar changes will be made on our Seabrook facility and will be documented in the FSAR and installed prior to commencement of fuel loading.

Should you have any comments on this proposed course of action and schedule, please notify us by August 15, 1980.

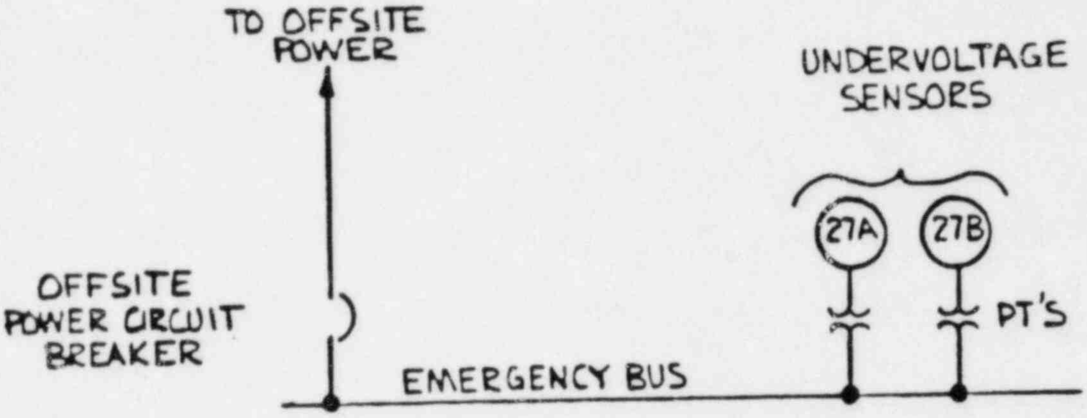
Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

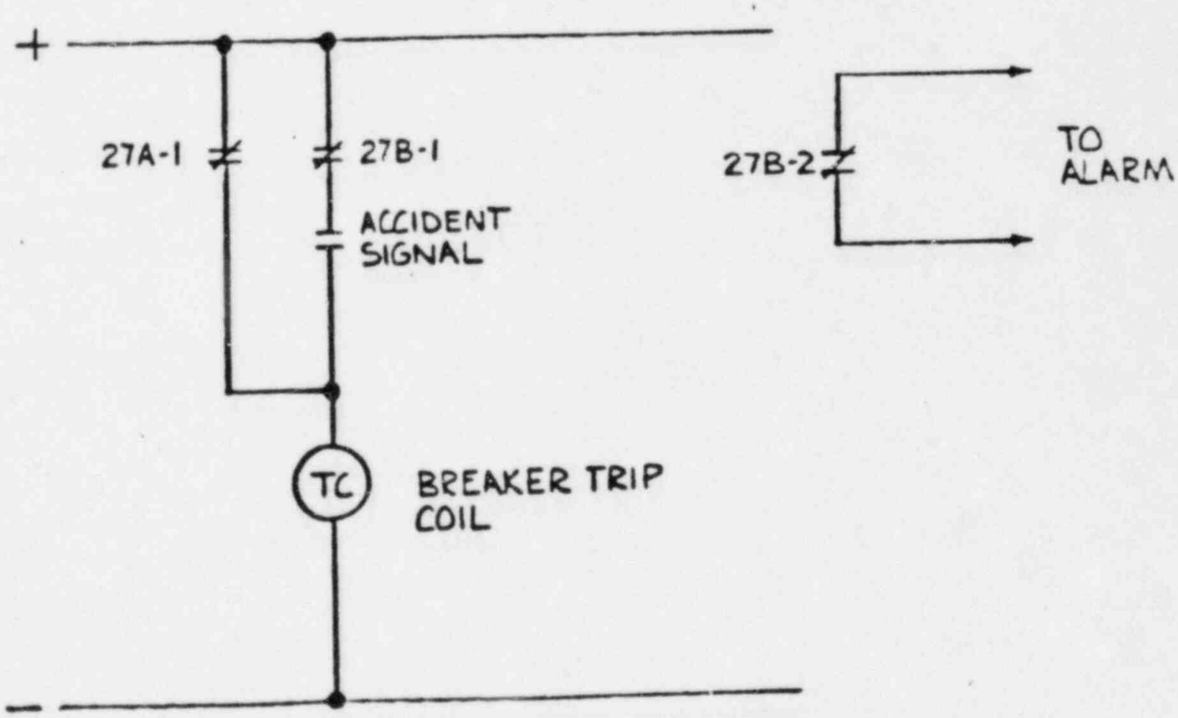
D. E. Vandenburgh

D. E. Vandenburgh
Senior Vice President

FIGURE 1

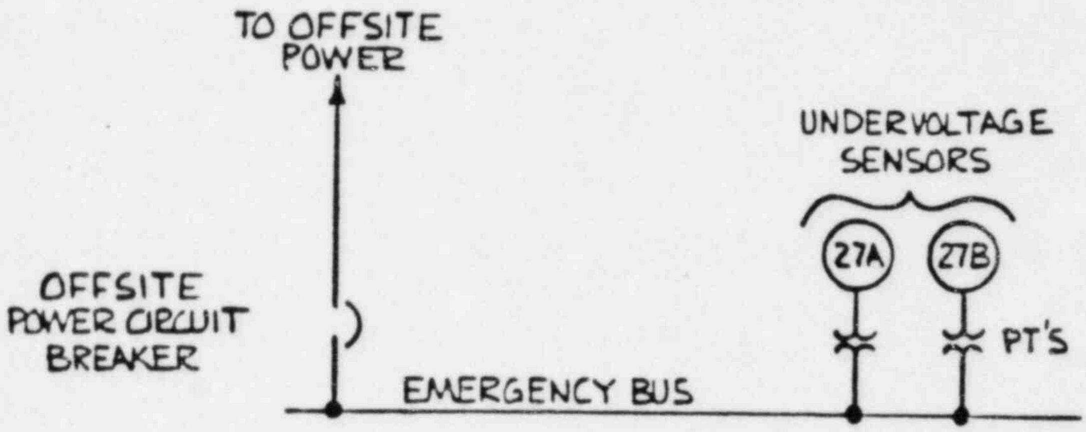


ONE LINE REPRESENTATION

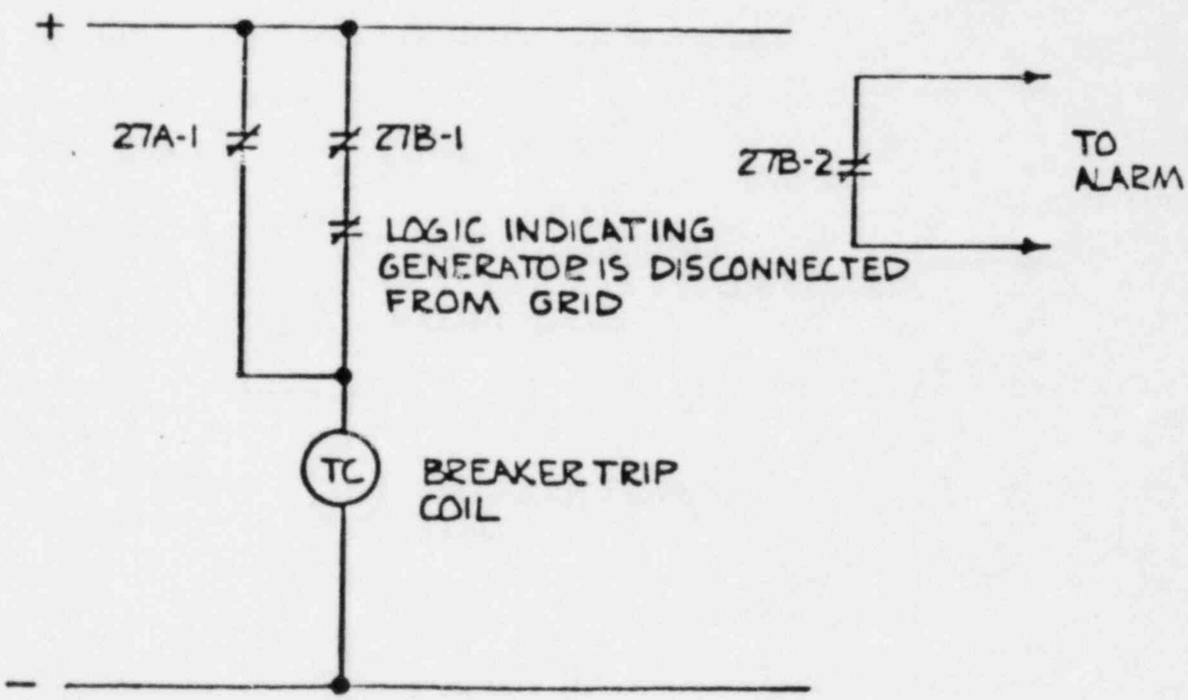


SCHEMATIC REPRESENTATION OF OFFSITE POWER CIRCUIT BREAKER TRIP CIRCUIT

FIGURE 2



ONE LINE REPRESENTATION



SCHEMATIC REPRESENTATION OF OFFSITE POWER CIRCUIT BREAKER TRIP CIRCUIT

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 27, 1982
NYR 82-236

Docket No. 50-29
LS05-82-09-078

ATTACHMENT 430.15-2

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: DEGRADED GRID VOLTAGE - YANKEE NUCLEAR POWER STATION (YANKEE)

The staff is continuing its evaluation of the issue of degraded grid voltage protection for Class 1E power systems at Yankee. We have reviewed your submittals and have determined that your approach to the resolution of this issue by the use of operator action under degraded grid conditions without an accident is acceptable. However, final approval of your proposal is subject to the completion of all proposed modifications and the institution of procedures which address the actions to be taken by the operators during a degraded grid condition.

We therefore request that, within 30 days of your receipt of this letter, you provide a schedule for the submission of the operator procedures, the design details for diesel generator load shedding, and the final Technical Specifications, as described in the enclosed Safety Evaluation and Contractor's Report. After the staff has reviewed this information, we will issue a final safety evaluation.

The reporting requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Safety Evaluation
2. Contractor's Report

cc w/enclosures:
See next page

~~8210040132~~
PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

RECEIVED
OCT 4 1982
YANKEE ATOMIC

September 27, 1982

Docket No. 50-29
LS05-82-09-078

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: DEGRADED GRID VOLTAGE - YANKEE NUCLEAR POWER STATION (YANKEE)

The staff is continuing its evaluation of the issue of degraded grid voltage protection for Class 1E power systems at Yankee. We have reviewed your submittals and have determined that your approach to the resolution of this issue by the use of operator action under degraded grid conditions without an accident is acceptable. However, final approval of your proposal is subject to the completion of all proposed modifications and the institution of procedures which address the actions to be taken by the operators during a degraded grid condition.

We therefore request that, within 30 days of your receipt of this letter, you

as described in the enclosed Safety Evaluation and Contractor's Report. After the staff has reviewed this information, we will issue a final safety evaluation.

The reporting requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Safety Evaluation
2. Contractor's Report

cc w/enclosures:
See next page

ENCLOSU... 1

SAFETY EVALUATION
YANKEE ROWE NUCLEAR POWER STATION
DOCKET NO. 50-029
DEGRADED GRID VOLTAGE PROTECTION FOR
THE CLASS 1E SYSTEM

INTRODUCTION AND SUMMARY

The criteria and staff positions pertaining to degraded grid voltage protection were transmitted to Yankee Atomic Power Company (YAEC) by NRC Generic Letter dated June 3, 1977. In response to this, by letters dated July 18, 1977, March 29, 1978, July 24, 1980, May 5, 1981, May 19, 1982, June 24, 1982 and July 2, 1982, the licensee proposed certain design modifications and changes to the Technical Specifications. A detailed review and technical evaluation of these proposed modifications and changes to the Technical Specifications was perfomed by LLL, under contract to the NRC, and with general supervision by NRC staff. This work is reported by LLL in "Degraded Grid Protection for Class 1E Power Systems Yankee Rowe Nuclear Power Station" (attached). We have reviewed this technical evaluation report and concur in the conclusion that the proposed electrical design modifications and Technical Specification changes are acceptable.

EVALUATION CRITERIA

The criteria used by LLL in its technical evaluation of the proposed changes include GDC-17 ("Electric Power Systems") of Appendix A to 10 CFR 50; IEEE Standard 279-1971 ("Criteria for Protection Systems for Nuclear Power Generating Stations"); IEEE Standard 308-1977 ("Voltage Ratings for Electrical Power Systems and Equipment - 60 Hz"); and staff positions defined in NRC Generic Letter to YAECO dated June 3, 1977.

~~8210040159~~ PDR

cc
Mr. James E. Tribble, President
Yankee Atomic Electric Company
25 Research Drive
Westborough, Massachusetts 01581

Chairman
Board of Selectmen
Town of Rowe
Rowe, Massachusetts 01367

Energy Facilities Siting Council
14th Floor
One Ashburton Place
Boston, Massachusetts 02108

U. S. Environmental Protection
Agency
Region I Office
ATTN: Regional Radiation Representative
JFK Federal Building
Boston, Massachusetts 02203

Resident Inspector
Yankee Rowe Nuclear Power Station
c/o U.S. NRC
Post Office Box 28
Monroe Bridge, Massachusetts 01350

Ronald C. Haynes, Regional Administrator
Nuclear Regulatory Commission, Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

PROPOSED CHANGES, MODIFICATIONS AND DISCUSSION

The existing undervoltage protection at Yankee Rowe consist of the following:

1. One loss of voltage inverse time relay on each 480 volt Class 1E bus. These relays (induction disc type) are set to actuate in 1.8 seconds on a complete loss of power, or 3.0 seconds at 277 volts (58%), or 7.0 seconds at 370 volts (77%) with a tap setting of 105 volts. This trip setting corresponds to 399 volts or 83.25% of nominal. Actuation of this relay will energize an auxiliary relay which initiates the following actions.

   a) Actuates a lock out relay which isolates the 480 volt Class 1E bus and starts the diesel generator,

   b) Trips the high pressure safety injection (HPSI) pump,

   c) Once the diesel generator attains satisfactory voltage and frequency, its output breaker will close. This will deenergize the auxiliary relay which will allow the start of the low pressure safety injection (LPSI) pump and remove the HPSI trip.

   d) The start of the HPSI is delayed until ten seconds after the start of the LPSI pump to allow the voltage to recover to normal.

The load shed feature of the HPSI pump is retained since no other loads are sequenced on following its start.

The following electrical system design modifications and technical specification changes were proposed by YAECO.

1. Installation of an additional loss of voltage relay on each 480 volt Class 1E bus. This will provide a two-out-of-two coincident logic per bus for the first level loss of voltage protection.

2. Installation of a second level of undervoltage relays on each 480 volt Class 1E bus. The second level relays will consist of two relays per 480 volt Class 1E bus arranged in a two-out-of-two coincidence logic with a setpoint of 91.5 ($\pm$1%) of nominal and a time delay of 10 ($\pm$1) seconds. The operation of the second level undervoltage protection is as follows:

If the 480 volt Class 1E bus voltage should degrade to 91.5% of nominal an alarm is initiated in the control room. Upon receipt of this alarm the operator will notify Rhode Island, Eastern Massachusetts and Vermont Energy Control (REMVEC) system dispatcher and request an assessment of the degraded voltage condition. If the (REMVEC) system dispatcher is unable to restore the voltage to an acceptable level within a reasonable time period, the operator will start the diesel generators and disconnect the Class 1E buses from the degraded offsite power system. The Class 1E buses will then be automatically sequenced on the onsite emergency diesel generators.

If a safety injection signal (SI) occurs at any time during level two actuation, the protective relays will automatically disconnect the offsite power source, initiate load shedding, and start the onsite emergency diesel generator. The safety loads will then be sequenced on the emergency diesel generator when acceptable frequency and voltage are achieved. The design will bypass the load shedding feature when the diesel generators are supplying the Class 1E bus. This feature will be automatically reinstated if the diesel generator breaker should trip. The licensee has not supplied the details on how this design will accomplish these features.

The licensee's proposed level two (degraded voltage) design will provide automatic separation of the Class 1E power system from offsite if a degraded grid exists coincident with a safety injection signal (LOCA). This approach provides protection to the Class 1E equipment needed to mitigate the consequences of an accident and is acceptable. For a degraded grid condition without a LOCA an alarm will be actuated and operator action will be taken to restore the grid to an acceptable level. If the grid cannot be restored to an acceptable level within a reasonable time period, the operator will start the emergency diesel generator and disconnect the Class 1E buses from the offsite power system. The Class 1E buses will then be automatically sequenced on the onsite emergency diesel generators. This approach deviates from the staff position that requires automatic isolation of the offsite power system for such undervoltage after a time delay. Acceptability of this alternative approach requires demonstration by the licensee that adequate safety systems will be available for safe shutdown of the reactor for these

conditions and that appropriate plant operating procedures are developed and available to the operator for the required operator action. We recommend that these procedures be reviewed as part of the integrated assessment of Yankee Rowe.

In response to the above concerns, the licensee in a submittal dated June 24, 1982, provided a list of systems that will not be exposed to or rendered inoperative by degraded grid voltage and therefore would be available to place the plant in a safe shutdown status under non-accident conditions. The Reactor Systems Branch (RSB) and Auxiliary Systems Branch (ASB) have reviewed the listing and concurred with the licensee's approach that this equipment provides the capability to place the plant in a safe shutdown condition. This equipment additionally has the capability to maintain the plant in a hot shutdown condition for the time required to reset any overload protective devices, or replace fuses that may have blown as a result of the degraded voltage.

On the basis of the above and that protection devices, i.e., circuit breakers, fuses, relays, etc., are provided to prevent damage to the equipment required for long term plant safe shutdown, and that alarms are provided to alert the operator to this abnormal condition, we find the licensee approach using operator action under degraded grid conditions without an accident acceptable. Acceptability of this approach is subject to the completion of all proposed modifications and institution of adequate procedures covering actions to be taken by the operator during a degraded grid under non-accident conditions.

A draft of the changes and additions to plant technical specifications including the surveillance requirements, allowable limits for the setpoint and time delay, and limiting conditions for operation have been provided by the licensee. The changes and additions to technical specifications have been reviewed and found acceptable. We require a formal submittal of these technical specifications for staff review.

CONCLUSIONS

We have reviewed the licensee submittals and the LLL technical evaluation report and find that:

1. The proposed degraded grid modifications will protect the Class 1E equipment from sustained degraded voltage of the offsite power system during accident conditions and is acceptable.

2. The licensee's proposal to use operator action instead of automatic disconnection of the Class 1E buses from a degraded offsite power source under non-accident conditions does not meet the staff's position. To justify this alternate approach the licensee has shown that adequate safety sysems, which are not exposed to or rendered inoperable by degraded grid voltage, are available to place and maintain the plant in a safe shutdown condition. The Reactor Systems Branch and Auxiliary Systems Branch have reviewed the licensee's shut-down systems and concurred that these systems are adequate to effect a plant safe shutdown under non-accident conditions. Based on the above, we find the licensee's alternate approach acceptable.

3. The licensee is r:quired to provide the following:

   (a) Design details and a description of the operation of the proposed load shedding bypass circuitry and how this feature will be reinstated on a diesel generator breaker trip.

   (b) Technical Specifications to cover the setpoints and tolerances, limiting conditions for operation and surveillance testing for the undervoltage protective relaying system.

   (c) Plant operating procedures to cover operator actions for degraded grid under non-accident conditions.

We therefore find the Yankee Rowe Nuclea: Power Station design acceptable subject to resolution of item 3 above. A,cer resolution of item 3 with YAECO, PSB will issue a supplement to this evaluation report.