BACKUP OVERSPEED TRIP

ELECTROHYDRAULIC CONTROL SYSTEM

GENERAL

The BACKUP OVERSPEED TRIP (BUOT) will energize the master trip relay and release the LOCKOUT (if it was picked up) on overspeed at and above its set speed.

It serves primarily as an overspeed trip when the mechanical overspeed trip is being tested while using the LOCKOUT. During normal operation, it is a redundant trip circuit that increases the safety of the system.

DESIGN

See *Schematic Wiring Diagram – Backup Overspeed Trip*, following this publication.

The speed signal for the backup overspeed trip is sensed by a magnetic pickup from a toothed wheel on the turbine shaft and fed to a power amplifier and magacycler circuit whose output is a d-c voltage proportional to speed.

The reference voltage is set so that the voltage comparator will energize its relay coil at and above a speed 0.5 percent higher than the mechanical overspeed trip setting.

Test logic is provided to disconnect the trip circuit and to change the trip reference to 99 percent speed in order to test the circuits at rated speed in normal operation. A 5-second time delay dropout (TDDO) relay will return the circuits to normal approximately 5 seconds after the test button is released.

The test button and associated signal light is located on the monitor panel in the electrohydraulic control (EHC) cabinet.

OPERATION

When the turbine speed rises to the backup overspeed trip (BUOT) speed, the voltage comparator is energized and will:

1. Energize the master trip relay
2. De-energize the LOCKOUT circuit if it was energized for testing of the overspeed trip.

This will trip the emergency trip system redundantly and shut the unit down.

The backup overspeed trip (BUOT) relay will reset at once when the turbine speed returns to a small amount below the trip speed. However, resetting of the emergency trip system should not be attempted until the unit has coasted down to approximately rated speed because the mechanical overspeed trip has also been tripped.

ADJUSTMENTS

The trip speed voltage and the 99 percent speed test voltage can be set with a voltmeter.

While the 99 percent test trip can easily be trimmed with the unit running, it is not advisable to do this at the backup overspeed trip (BUOT) speed, because this involves extended operation at abnormally high speeds. Instead, the trip speed should be preset and checked by Test E as found in *Testing of Overspeed Trip System*, included in this Tab.

TEST

See *Testing of Overspeed Trip System* included in this Tab.

The backup overspeed trip circuits (Test D) should be tested once a week by depressing the test button at the monitor panel in the electrohydraulic control (EHC) cabinet. When this is done the backup overspeed trip (BUOT) (called 12 percent OVERSPEED on some units) light should come on almost immediately.

The actual system trip test (Test E) by overspeeding the unit while the LOCKOUT is depressed, should be made every 12 to 24 months.

See *Testing of Overspeed Trip System*, included in this Tab, for action on unsuccessful tests.

MAINTENANCE

If the backup overspeed trip (BUOT) circuit test (Test D) is negative (test light does not come on), the magnetic pickup voltage input to the BUOT circuit board should be checked. If this voltage is normal and the 99 percent speed reference is also at

the proper value, there must be some defect on the components of the circuit board or the voltage comparator.

The voltage comparator can be replaced by first installing jumpers between the pins that are connected to each closed contact of both voltage comparators on the respective circuit board. Then the board can be removed and replaced and the jumpers removed subsequently.

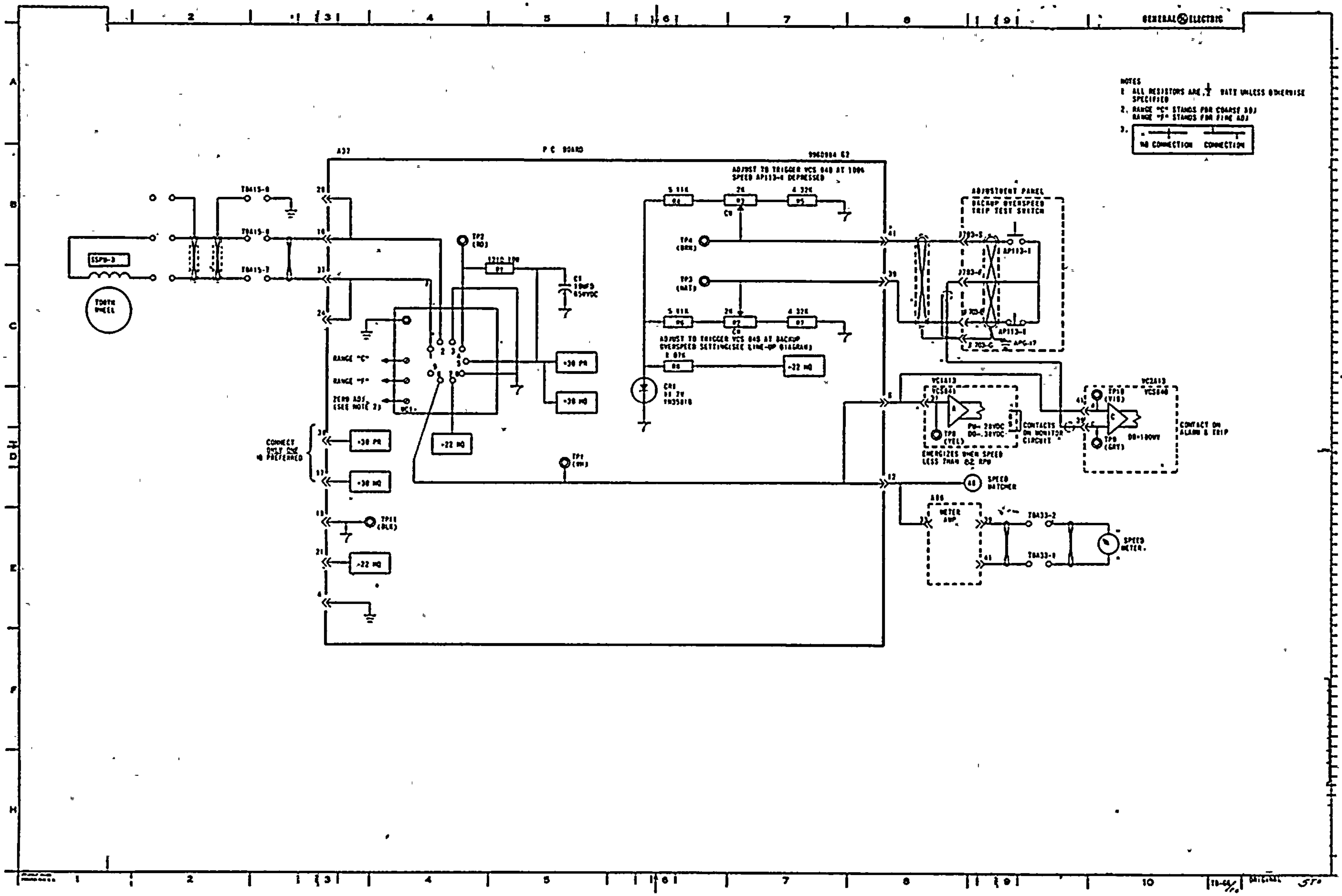
Special care should be used if one of the voltage comparators to be removed is energized. The consequences of the normally closed (NC) contact being closed for a fraction of a second after plugging the card in must be determined and the system should be able to tolerate it. If the consequences are not tolerable or cannot be established, the unit should be shut down before making repair.

If the Magacycler board is defective, it may be removed for repair or replacement provided the voltage comparator is first removed, as previously described.

After a Magacycler board is replaced, the reference voltages must be checked and the circuit test must be successful before tests involving the LOCK-OUT push button can again be performed.

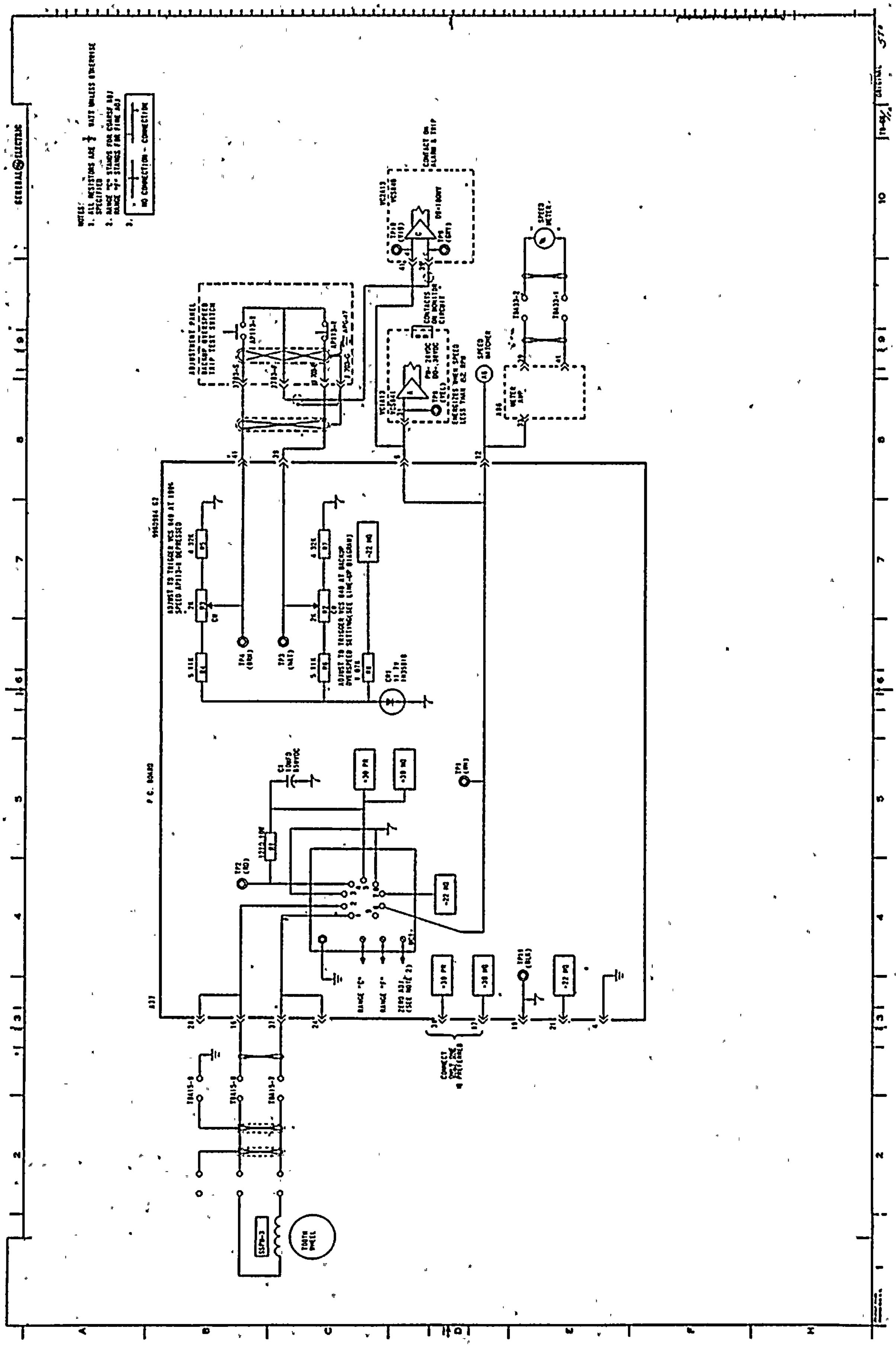
NOTE: On some machines the backup overspeed trip board is also used to provide a speed signal for other switching actions.

GENERAL ELECTRIC



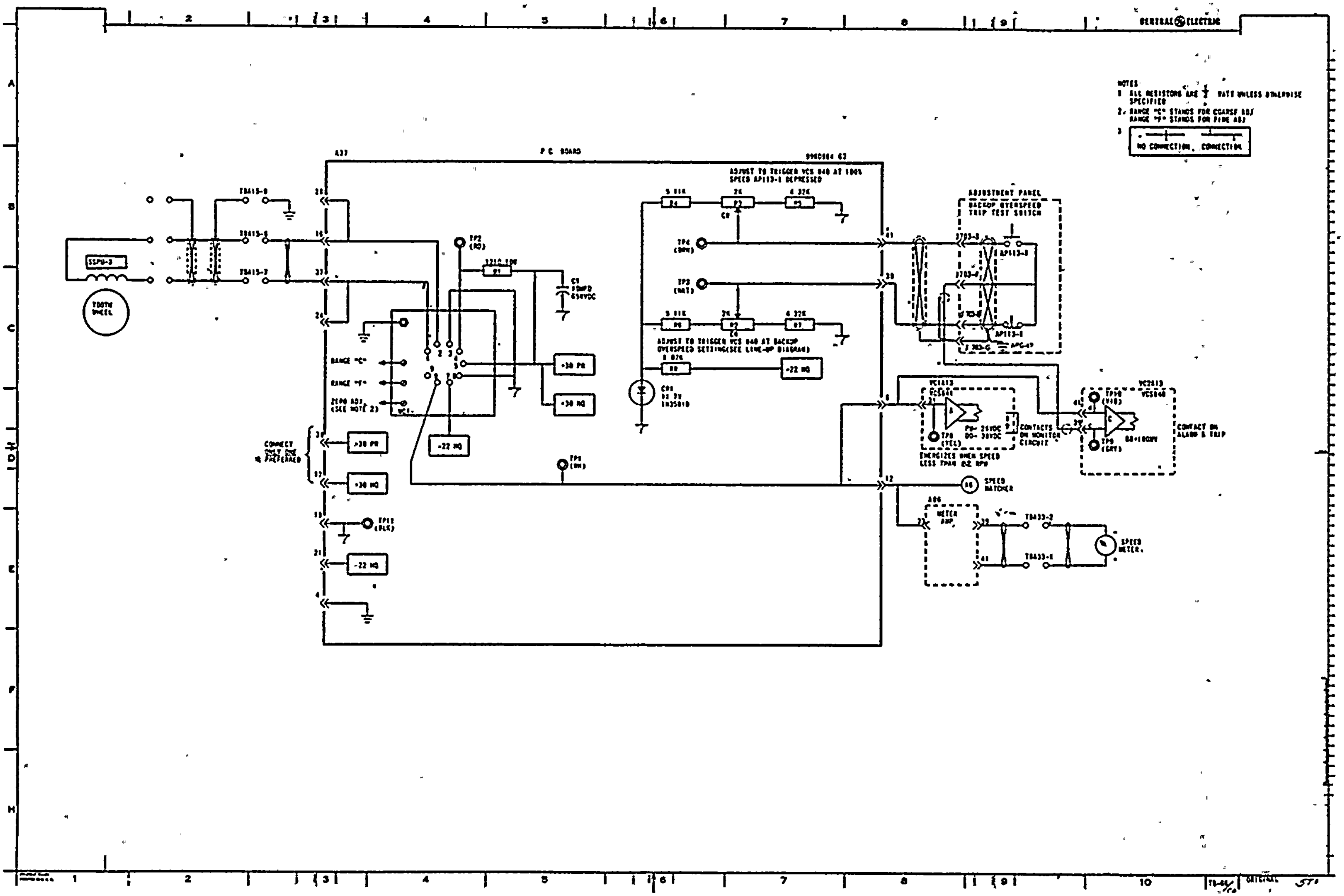
Schematic - Backup Overspeed Trip
Dwg. 125D4408 (Rev. 2)

Fig. 43-22



Schematic - Backup Overspeed Trip
Dwg. 125D4408 (Rev. 2)

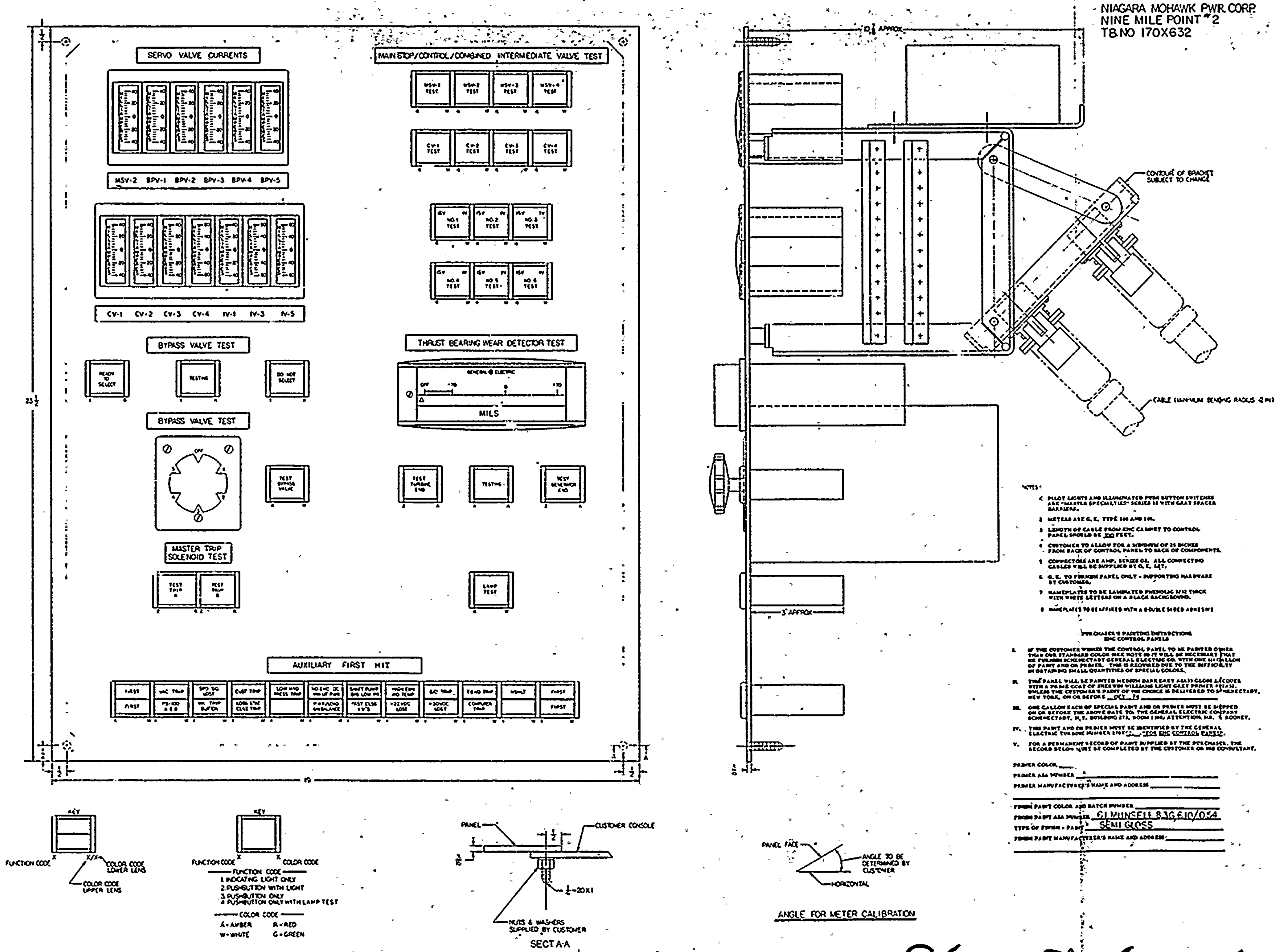
Fig. 43-22



Schematic - Backup Overspeed Trip
Dwg. 125D4408 (Rev. 2)

Fig. 43-22

NIAGARA MOHAWK PWR. CORP.
NINE MILE POINT #2
TB NO 170X632

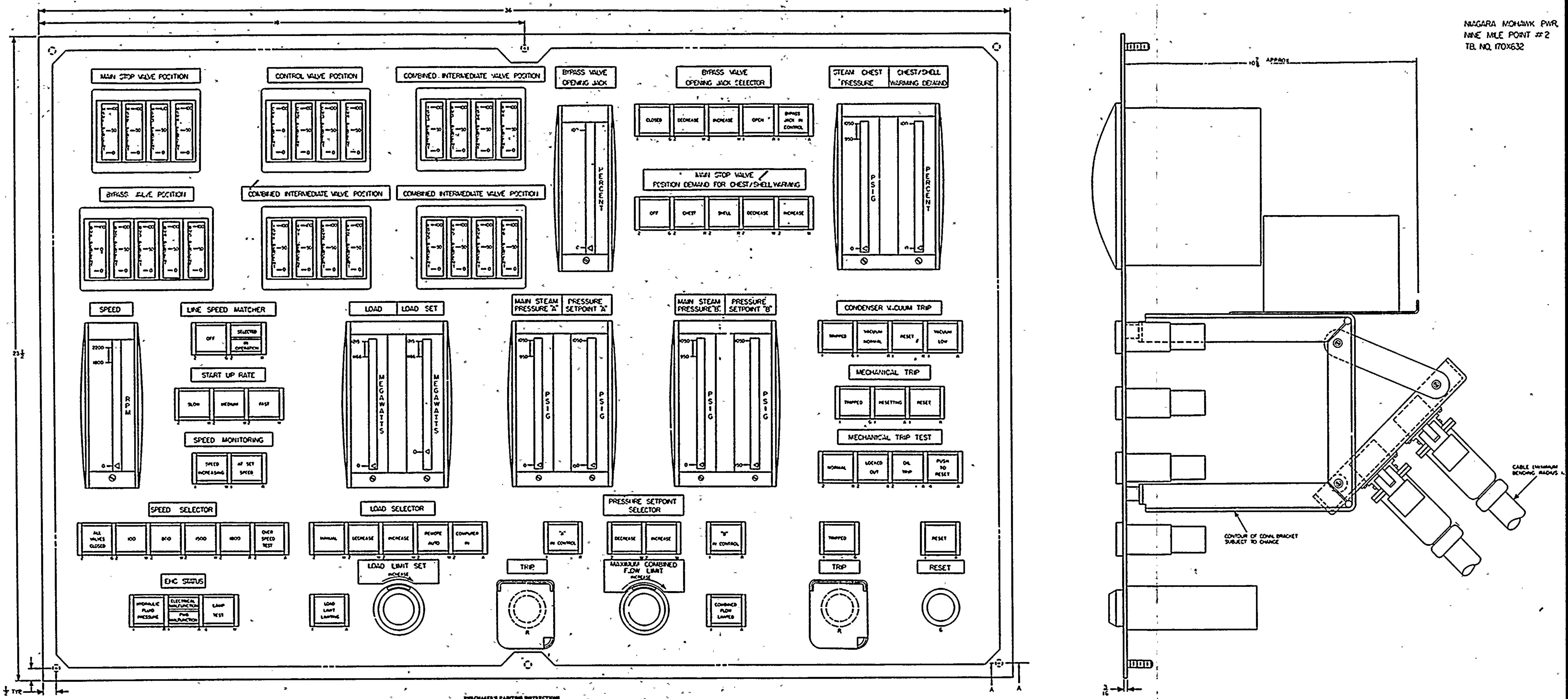


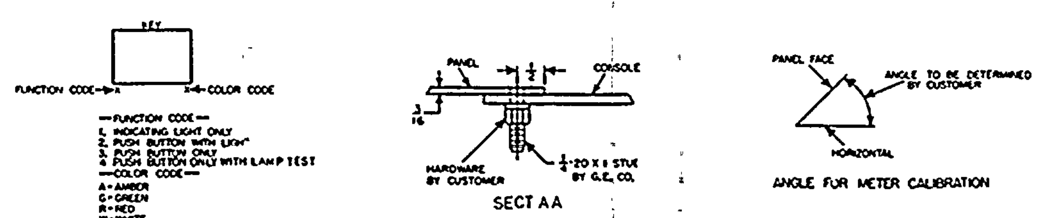
PRC
APERTURE
CARD

8312280499-01

Outline- Control Panel #2
Dwg. 838E186 (Rev. 2)

Fig. 45-2





NOTES:

1. TURBINE TRIP PUSH BUTTON SWITCH IS A G. E. TYPE [illegible] WITH COLOR RING [illegible].
2. TURBINE RESET PUSH BUTTON SWITCH IS A G. E. TYPE [illegible] WITH COLOR RING [illegible].
3. PILOT LIGHTS AND ILLUMINATED PUSH BUTTON SWITCHES ARE "MASTER SPECIALTIES" SERIES 32 WITH GRAY SPACER BARRIERS.
4. METERS ARE G. E. TYPE 180 AND 185.
5. LENGTH OF CABLE FROM EHC CABINET TO CONTROL PANEL SHOULD BE 100 FEET.
6. CUSTOMER TO ALLOW FOR A MINIMUM OF 20 INCHES FROM BACK OF CONTROL PANEL TO BACK OF COMPONENTS.
7. CONNECTORS ARE AMP. SERIES GA. ALL CONNECTING CABLES WILL BE SUPPLIED BY G. E. LST.
8. G. E. TO FURNISH PANEL ONLY - SUPPORTING HARDWARE BY CUSTOMER.
9. NAMEPLATES TO BE LAMINATED PHENOLIC 1/16 THICK WITH WHITE LETTERS ON A BLACK BACKGROUND.
10. NAMEPLATES TO BE AFFIXED WITH A DOUBLE SIDED ADHESIVE.

PURCHASER'S PAINTING INSTRUCTIONS
EHC CONTROL PANELS

I. IF THE CUSTOMER WISHES THE CONTROL PANEL TO BE PAINTED OTHER THAN OUR STANDARD COLOR (SEE NOTE II) IT WILL BE NECESSARY THAT HE FURNISH SCHENECTADY GENERAL ELECTRIC CO. WITH ONE (1) GALLON OF PAINT AND OR PRIMER. THIS IS REQUIRED DUE TO THE DIFFICULTY IN OBTAINING SMALL QUANTITIES OF SPECIAL COLORS.
II. THIS PANEL WILL BE PAINTED MEDIUM DARK GREY ASA [illegible] GLOSS LACQUER WITH A PRIME COAT OF SHERWIN WILLIAMS LIGHT GREY PRIMER [illegible] UNLESS THE CUSTOMER'S PAINT OF HIS CHOICE IS DELIVERED TO SCHENECTADY, NEW YORK, ON OR BEFORE OCT [illegible]
III. ONE GALLON EACH OF SPECIAL PAINT AND OR PRIMER MUST BE SHIPPED ON OR BEFORE THE ABOVE DATE TO: THE GENERAL ELECTRIC COMPANY SCHENECTADY, N. Y. BUILDING 273, ROOM 1300, ATTENTION: MR. J. ROONEY.
IV. THIS PAINT AND OR PRIMER MUST BE IDENTIFIED BY THE GENERAL ELECTRIC TURBINE NUMBER [illegible] "FOR EHC CONTROL PANELS".
V. FOR A PERMANENT RECORD OF PAINT SUPPLIED BY THE PURCHASER, THE RECORD BELOW MUST BE COMPLETED BY THE CUSTOMER OR HIS CONSULTANT.

PRIMER COLOR
PRIMER ASA NUMBER
PRIMER MANUFACTURER'S NAME AND ADDRESS
FINISH PAINT COLOR AND BATCH NUMBER
FINISH PAINT ASA NUMBER 61 MUNSELL 836.610/0.54
TYPE OF FINISH - PAINT SEMI GLOSS
FINISH PAINT MANUFACTURER'S NAME AND ADDRESS

PRC
APERTURE
CARD

8312280499-02

Outline- Control Panel #1
Dwg. 258R335 (Rev. 1)

Fig. 45

SALP INPUT

Plant: Nine Mile Point Unit 2

1. Management Involvement and Control in Assuring Quality: Not Applicable
2. Approach to Resolution of Technical Issues from a Safety Standpoint: Category 2
3. Responsiveness to NRC Initiatives: Category 3
4. Enforcement History: Not Applicable
5. Reporting and Analysis of Reportable Events: Not Applicable
6. Staffing (Including Management): Not Applicable
7. Training and Qualification Effectiveness: Not Applicable

The following is the narrative for Items 2 and 3 above.

The magnitude of the open items is indictive of the applicant's attitude toward the questions which we provided; the larger number of our questions did not receive responses. Where the applicant providedresponses the information was acceptable.

DRAFT SAFETY EVALUATION REPORT
NINE MILE POINT UNIT 2
AUXILIARY SYSTEMS BRANCH

3.4.1 Flood Protection

The design of the facility for flood protection was reviewed in accordance with Section 3.4.1 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the design of the facility for flood protection with respect to the applicable regulations of 10 CFR 50.

In order to assure conformance with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," our review of the overall plant flood protection design included all systems and components whose failure due to flooding could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity.

The design basis flood level (DBFL) for Nine Mile Point, Unit 2 has been determined to be at elevation 260.6 ft based on the flooding of Lake Ontario. The applicant has stated that flooding due to the probable maximum precipitation (PMP) would not exceed el 260.6 ft in the vicinity of the plant buildings. (Refer to Section 2.4 of this SER for further discussion on flood level.) The structures housing safety-related equipment and systems, such as the reactor building, diesel generator building, and control building, are constructed with reinforced concrete walls below grade level to limit inleakage. The personnel entrance and equipment access openings to these buildings are provided at or above elevation 261 ft which is above the DBFL. All penetrations through the exterior walls below the DBFL have watertight penetration sleeves. Underground cables are protected from wetting or flooding by being housed in watertight conduits which are enclosed in reinforced concrete encasements to form electrical ductlines. Joints in safety-related structures are provided with waterstops to prevent inleakage of groundwater or floodwater. The safety-related

[illegible]

components in the screen well structure are completely enclosed by watertight walls to el 261 ft. Therefore, the criteria of Regulatory Guides 1.59, "Design Basis Floods for Nuclear Power Plants," Positions C.1 and C.2, and 1.102, "Flood Protection For Nuclear Power Plants," Position C.1, are satisfied. The sumps pumps and/or drains are provided in all seismic Category I structures. There are no nonseismic or nontornado-protected Category I vessels, pipes, or tanks located outside of buildings. Therefore, there is no potential for flooding of safety-related structures, systems, or components due to the failure of these components during an SSE. (Refer to Section 9.3.3 of this SER for discussion of protection of safety-related equipment from flooding within the plant.)

Based on our review of the design criteria and bases, and safety classification of safety-related systems, structures, and components necessary for a safe plant shutdown during and following flood conditions, we conclude that the design of the station for flood protection conforms with the requirements of General Design Criterion 2 with respect to protection against natural phenomena and conforms with the guidelines of Regulatory Guides 1.59, Positions C.1 and C.2, and 1.102, Position C.1, concerning flood protection and is, therefore, acceptable.

The design of the facility for flood protection meets the acceptance criteria of SRP Section 3.4.1.

3.5.1.1 Internally Generated Missiles (Outside Containment)

The design of the facility for providing protection from internally generated missiles outside containment was reviewed in accordance with Section 3.5.1.1 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below formed the basis for our evaluation of the design of the facility for providing protection from internally generated missiles outside containment with respect to the applicable regulations of 10 CFR Part 50.

The acceptance criteria for the design of the facility for providing missile protection includes meeting Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles." The review of turbine missiles is discussed separately in Section 3.5.1.3.

General Design Criterion 4, "Environmental and Missile Design Bases," requires protection of plant structures, systems, and components, whose failure could lead to offsite radiological consequences or that are required for safe plant shutdown, against postulated missiles associated with plant operation. The missiles considered in this evaluation include those missiles generated by rotating or pressurized (high-energy fluid system) equipment.

Protection is provided by any one or a combination of compartmentalization, barriers, separation, orientation, and equipment design. The primary means of providing protection to safety-related equipment from damage resulting from internally generated missiles is through the use of plant physical arrangement. Safety-related systems are physically separated from nonsafety-related systems and components of safety-related systems are physically separated from their redundant components. Stored spent fuel in the reactor building is protected by the fuel pool walls from damage by internal missiles which could result in radioactive release and by not locating any high-energy piping system or rotating machinery in the vicinity of the pool.

The applicant has provided an evaluation of potential missile sources from rotating equipment failures and high-energy systems on the basis that a single failure in the system could result in potential missiles. This evaluation included typical internal missile sources such as valve stems, valve bonnets, instrument wells, RCIC turbine blades, fan blades, and pump impellers. Based on the conservative design of these components, the applicant has stated that none of these are credible missiles sources. Valve stems have insufficient stored energy to damage equipment. A single bolt failure will not cause valve bonnets to become missiles. A complete failure of a circumferential weld is required to cause an instrument well missile. Pump casings, fan casings, and RCIC turbine casing are of sufficiently heavy construction to retain potential impeller or blade missiles. Remote location and separation of safety-related system trains provides further protection against the effects of potential

internally generated missiles. Further, the applicant has demonstrated that adequate protection is provided to the safety-related equipment from missiles generated by nonsafety-related equipment by separation of safety and nonsafety related equipment. We concur with the applicant's assumptions and evaluation for potential missiles outside containment.

We have reviewed the adequacy of the applicant's design to maintain the capability for a safe plant shutdown and to prevent unacceptable radiological release in the event of internally generated missiles outside the containment.

Based on the above, we conclude that the design is in conformance with the requirements of General Design Criterion 4 as it relates to protection against internally generated missiles and is, therefore, acceptable. The design of the facility providing protection from internally generated missiles meets the applicable acceptance criteria of SRP Section 3.5.1.1.

3.5.1.2 Internally Generated Missiles (Inside Containment)

The design of the facility for providing protection from internally generated missiles inside containment was reviewed in accordance with Section 3.5.1.2 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the design of the facility for providing protection from internally generated missiles with respect to the applicable regulations of 10 CFR Part 50.

All plant structures, systems, and components (SSC) inside containment whose failure could lead to offsite radiological consequences or that are required for safe plant shutdown must be protected against the effects of internally generated missiles in accordance with the requirements of General Design Criterion 4, "Environmental and Missile Design Bases." Potential missiles that could be generated inside the containment are from failures of rotating components, pressurized component (high-energy fluid system) failures and gravitational effects.

The applicant's analysis of rotating equipment (pumps and fans) failures indicates that conservative equipment design will prevent such components from becoming sources of potential missiles. Pump and fan casings are of sufficiently heavy construction to retain potential impeller and fan blade missiles.

The applicant considered the following for potential missiles from pressurized high-energy fluid systems: high-pressure gas bottles and accumulators, valves, temperature and pressure sensor assemblies. The applicant performed analyses which demonstrate that the design of the above components either prevents the generation of missiles as a result of a single failure, or if generated, the missiles either have insufficient energy to cause unacceptable damage or else adequate compartmentalization, separation, or barriers provide protection for safety-related equipment. Since there are no credible primary missiles sources, secondary missiles are not postulated.

In addition, the applicant evaluated the potential for gravitational missiles inside containment. The nonsafety-related components are supported to prevent their collapse in an SSE.

We have reviewed the adequacy of the applicant's design to maintain the capability for a safe plant shutdown and prevent unacceptable radiological release in the event of internally generated missiles inside containment.

Based on our review, we conclude that the design is in conformance with General Design Criterion 4 as it relates to protection against internally generated missiles inside the containment and is, therefore, acceptable. The design of the facility providing protection from internally generated missiles meets the applicable acceptance criteria of SRP Section 3.5.1.2.

3.5.1.4 Missiles Generated by Natural Phenomena

The tornado missile spectrum was reviewed in accordance with Section 3.5.1.4 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section except as noted below. Conformance with the acceptance criteria

formed the basis for our evaluation of the tornado missile spectrum with respect to the applicable regulations of 10 CFR Part 50.

The portions of the "Review Procedures" concerning the probability per year of damage to safety-related systems due to missiles were not used in our review. Our review for this section of the SRP is concerned with establishing the missile spectrum, not with calculating the probability of damage.

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems, and components essential to safety be designed to withstand the effects of natural phenomena, and General Design Criterion 4, "Environmental and Missile Design Bases," requires that these same plant features be protected against missiles. The missiles generated by natural phenomena that are of concern are those resulting from tornadoes. The applicant has utilized missile spectrum A of SRP 3.5.1.4 for a tornado Region I site as identified in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," Positions C.1 and C.2. We have evaluated this spectrum and conclude that it is representative of missiles at the site and is, therefore, acceptable. A discussion of the protection afforded safety-related equipment from the identified tornado missiles including compliance with the guidelines of Regulatory Guide 1.117, "Tornado Design Classification," is provided in Section 3.5.2 of this SER. A discussion of the adequacy of barriers and structures designed to withstand the effects of the identified tornado missiles is provided in Section 3.5.3 of this SER.

Based upon our review of the tornado missile spectrum, we conclude that the spectrum was properly selected and meets the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena and missiles and the guidelines of Regulatory Guides 1.76, Positions C.1 and C.2, and 1.117 with respect to identification of missiles generated by natural phenomena and is, therefore, acceptable. The tornado missile spectrum meets the acceptance criteria of SRP Section 3.5.1.4.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

The design of the facility for providing protection from tornado-generated missiles was reviewed in accordance with Section 3.5.2 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the design of the facility for providing protection from tornado generated missiles with respect to the applicable regulation of 10 CFR Part 50.

General Design Criterion 2, "Design Basis for Protection Against Natural Phenomena," requires that all structures, systems, and components important to the safety of the plant be protected from the effects of natural phenomena, and General Design Criterion 4, "Environmental and Missile Design Bases," requires that all structures, systems, and components important to the safety of the plant be protected from the effects of externally generated missiles. The Nine Mile Point Unit 2 site is located in Tornado Region I as identified in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." The tornado missile spectrum is discussed in Section 3.5.1.4 of this SER.

The applicant has identified all safety-related structures, systems, and components requiring protection from externally generated missiles. All safety-related structures including the containment and reactor building are designed to withstand postulated tornado generated missiles without damage to safety-related equipment. However, the applicant has not provided the sufficient information for us to evaluate the adequacy of tornado missile protection for the diesel generator exhaust, outside air intakes for HVAC systems and safety-related buried piping. The applicant should provide the necessary assurance that the safety functions performed by these components is assured in the event of a tornado. All safety-related systems and components and stored fuel are located within tornado-missile-protected structures or are provided with tornado missile barriers. The ultimate heat sink for Nine Mile Point Unit 2 is Lake Ontario which has inherent protection against natural phenomena. Therefore, the requirements of General Design Criteria 2 and 4 with respect to missiles

protection are not met. The guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Positions C.2 and C.3, and 1.117, "Tornado Design Classification," Positions C.1 through C.3, concerning tornado missile protection for safety-related structures, systems, and components including stored fuel and the ultimate heat sink are met. Protection from low-trajectory turbine missiles including compliance with Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles," is discussed in Section 3.5.1.3 of this SER.

Based on the above, we cannot conclude that the applicant's list of safety-related structures, systems, and components to be protected from externally generated missiles and the provisions in the plant design providing this protection are in accordance with the requirements of General Design Criteria 2 and 4 with respect to missile and environmental efforts are satisfied. The guidelines of Regulatory Guides 1.13, 1.27, 1.115, and 1.117, concerning protection of safety-related plant features including stored fuel and the ultimate heat sink from tornado missiles. The tornado missile protection does not meet the acceptance criteria of SRP Section 3.5.2. We will report resolution of the above concerns in a supplement to this SER.

3.6.1 Plant Design for Protection Against Postulated Failures in Fluid Systems Outside Containment

The design of the facility for providing protection against postulated piping failures outside containment was reviewed in accordance with Section 3.6.1 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the design of the facility for providing protection against postulated piping failures outside containment with respect to the applicable regulations of 10 CFR 50.

The guidelines for meeting the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," concerning protection against postulated piping failure in high energy and moderate energy fluid systems

outside containment are contained in Branch Technical Position ASB 3-1, "Protection Against Postulated Failures in Fluid Systems Outside Containment."

The applicant has identified all high and moderate energy piping systems in accordance with these guidelines and also has identified those systems requiring protection from postulated piping failures. The applicant has performed a subcompartment analysis for the steam tunnel and main steam lines. The results of the analysis indicate that the resulting jet impingement and environmental effects from a postulated circumferential pipe break in one of these lines will not have any adverse consequences on safety-related equipment or structures. Thus, we conclude that the General Design Criterion 4, "Environmental and Missile Design Basis," as it relates to the plant design accommodates the effects of postulated pipe breaks in high energy fluid piping systems outside containment with respect to pipe whip, jet impingement, resulting reactive forces, and environmental effects, and the effects of postulated cracks in moderate energy fluid systems outside containment with respect to jet impingement, flooding, and other environmental effects including a concurrent single active failure. The means used to protect safety-related systems and components throughout the plant are physical separation, enclosure in suitable designed structures, and restraints where separation and enclosure approaches are not feasible. The applicant has used the guidance in SRP Section 3.6.1 and Branch Techncial Position ASB 3-1 in evaluating the effects of high and moderate energy pipe breaks. The applicant has stated that the main feedwater lines are designed as seismic Category I. The turbine building walls and within the turbine building is seismically supported near the steam line. Further, there is no safety-related equipment in the area of concern within the turbine building.

However, the applicant's analyses for all pipe break locations, jet impingement, and environment effects of postulated pipe breaks in response to our question no. 410.15 is not complete. The applicant has stated that this information will be supplied at a later date. Further discussion of safety-related equipment is contained in Section 3.11 of this SER. Pending receipt of acceptable information as discussed above, we find that the applicant has adequately designed and protected areas and systems required for safe plant shutdown following postulated events, including the combination of pipe failure and

single active failure. We cannot conclude that the plant design meets the requirements of General Design Criterion 4, and the criteria set forth in Branch Technical Position 3-1 with regard to the protection of safety-related systems and components from a postulated high energy line break and with regard to the protection of safety-related systems and components from a postulated moderate-energy line failure. The design of the facility for providing protection from high and moderate energy pipe failures and effects does not meet the applicable acceptance criteria of SRP Section 3.6.1.

4.6 Functional Design of Reactivity Control Systems

The control rod drive system was reviewed in accordance with Section 4.6 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP Section. Conformance with the acceptance criteria formed the basis for our evaluation of the control rod drive system with respect to the applicable regulations of 10 CFR 50.

The control rod drive system (CRDS) and recirculation flow control system (RFCS) are designed to control reactivity during power operation. Reactivity is controlled in the event of fast transients by automatic rod insertion. In the event the reactor cannot be shut down with the control rods, the operator can actuate the standby liquid control system which pumps a solution of sodium pentaborate into the primary system. The evaluation of the functional design of the standby liquid control system can be found in Section 9.3.5 of this report.

Reactivity in the core is controlled by the CRDS via movable control rods interspersed throughout the core. These rods control the overall reactor power level and provide the principal means of quickly and safety shutting down the reactor. This is the normal method of making large changes in reactor power, such as daily or weekly load shifts requiring reductions and increases of power.

Each control rod is moved by a separate hydraulic control unit. A supply pump provides the hydraulic control units with water from the condensate storage

tank for cooling the rods and for moving them into and out of the core, with a spare pump on standby. The pump also provides water to a scram accumulator in each hydraulic control unit to maintain the desired water inventory. When necessary, the accumulator forces water into the drive system to scram the control rod connected to that hydraulic control unit; at lower pressures the volume of water in the scram accumulator is sufficient to scram the rod. At higher pressures, most of the water to scram is provided from the reactor vessel. A single failure in a hydraulic control unit would result in the failure of only one rod.

The CRDS has been designed to permit periodic functional testing during power operation with the capability to test individual scram channels and motion of individual control rods independently. The CRDS is designed so that failure of all electrical power will cause the control rods to scram, thereby protecting the reactor. Based on the above, we conclude that the requirements of General Design Criterion 23, "Protection System Failure Mode," are satisfied.

Preoperational tests of the control rod drive hydraulic system will be conducted to determine capability of the system. Startup tests will be conducted over the range of temperatures and pressures from shutdown to operating conditions in order to determine compliance with applicable technical specifications. Each rod that is partially or fully withdrawn during operation will be exercised one notch at least once each week. Operable control rods will be tested for compliance with scram time criteria, from the fully withdrawn position, after each refueling shutdown.

A malfunction in the CRDS could result in a reactivity change. The applicant demonstrated in Section 15 of the Final Safety Analysis Report (FSAR) that the CRDS limits these postulated transients to within acceptable fuel response limits, as required by General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunction."

The CRDS is designed to provide reactivity control under normal operation and anticipated operational occurrences with an appropriate allowance for a stuck rod. This capability is demonstrated by the safety analyses discussed in

Section 15 of the FSAR. This system is also capable of holding the core subcritical under cold shutdown conditions. The recirculation flow control system is capable of accommodating reactivity changes during normal operation conditions (i.e., power changes and xenon burnout). The standby liquid control system is capable of bringing the reactor subcritical under cold shutdown conditions in the event the control rods cannot be inserted. These systems, taken together, satisfy the requirements of General Design Criteria 26, "Reactivity Control System Redundancy and Capability," 27, "Combined Reactivity Control System Capability," and 29, "Protection Against Anticipated Operational Occurrences."

The CRDS is capable of providing reactivity control following postulated accidents with an appropriate margin for a stuck rod. This capability is demonstrated by the loss-of-coolant accident and rod dropout analyses presented by the applicant which, in turn, show that the consequences are acceptable and core cooling is maintained, as required by General Design Criterion 28, "Reactivity Limits."

The design does not utilize a CRDS return line to the reactor pressure vessel. In accordance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," dated November 1980, equalizing valves are installed between the cooling water header and exhaust water header, the flow stabilizer loop is routed to the cooling water header and both the exhaust header and flow stabilizer loop are stainless steel piping.

We have reviewed the extent of conformance of the Scram Discharge Volume (SDV) design with the NRC generic study, "BWR Scram Discharge System Safety Evaluation," dated December 1, 1980. The design provides two separate SDV headers, with an integral instrumented volume (IV) at the end of each header, thus providing close hydraulic coupling. Each IV has redundant and diverse level instrumentation (float sensing and pressure sensing) for the scram function attached directly to the IV. Vent and drain lines are completely separated and contain redundant vent and drain valves equipped with redundant solenoid pilot valves. High point venting is provided. We conlude that the design of the SDV fully meets the recommendations of the above referenced NRC generic SER and is, therefore, acceptable.

The applicant has not fully addressed our generic letter dated May 5, 1981 regarding the report entitled "Safety Concerns Associated With a Pipe Break in the BWR Scram System," NUREG-0803, which we requested in our question 410.16.

Based on our review, we conclude that the functional design of the reactivity control system meets the requirements of General Design Criteria 23, 25, 26, 27, 28, and 29 with respect to demonstrating the ability to reliably control reactivity changes under normal operation, anticipated operational occurrences, and accident conditions including single failures, and the guidelines of NUREG-0619 and the generic document dated December 1, 1980, and is, therefore, acceptable. However, we cannot conclude that Nine Mile Point Unit 2 is in compliance with the guidelines of NUREG-0803 until receipt of further information. We will report resolution of this item in a Supplement to this SER. The functional design of the reactivity control system meets the applicable acceptance criteria of SRP 4.6 except as noted above.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The reactor coolant pressure boundary leakage detection systems were reviewed in accordance with Section 5.2.5 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portions of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the reactor coolant pressure boundary (RCPB) leakage detection system with respect to the applicable regulations of 10 CFR Part 50.

A limited amount of leakage is to be expected from components forming the RCPB. Means are provided for detecting and identifying this leakage in accordance with the requirements of General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary." Leakage is classified into two types - identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges are not completely leaktight. Since this leakage is expected, it is considered identified leakage and is monitored and separated from other leakage (unidentified) by directing it to closed systems as identified in the guidelines of Position C.1 of Regulatory Guide 1.45, "Reactor Coolant Pressure boundary Leakage Detection Systems.

Provisions have been made to monitor intersystem leakage between the RCPB and those systems connected to the RCPB for monitoring and alarming leakage by using radioactivity and differential flow monitors. Thus, the guidelines of Regulatory guide 1.45, Postion C.4 are met. Each leakage detection system has indicators and alarms in the control room, thus meeting the guidelines of Regulatory Guide 1.45, Position C.7.

Unidentified leakage from the RCPB is detected by high pressure and temperature within primary containment, drywell equipment and floor drainage sump level, gaseous radiation level in primary containment, and airborne particulate radioactivity monitoring. The above leakage detection systems are seismic Category I and are designed to be capable of performing their function following an SSE. The leakage detection system is designed to monitor the unidentified leakage flow rate with an accuracy of one gallon per minute within one hour. Thus, the requirements of General Design Criterion 2, "Protection Against Natural Phenomena" and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification" Position C.1 and C.2 and 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" Positions C.2, C.3, C.5, and C.6 are satisfied.

The leakage detection systems are capable of being tested and calibrated during plant operation, thus the design meets the guidelines of Regulatory Guide 1.45, Position C.8.

The applicant has stated that the plant technical specifications would provide limiting conditions for identified and unidentified leakage, thus satisfying the guidelines of Regulatory Guide 1.45, Position C.9.

Based on the above, we conclude that the RCPB leakage detection systems meet the requirements of General Design Criteria 2 and 30 with respect to protection against natural phenomena and provisions for RCPB leak detection and identification and the guidelines of Regulatory Guides 1.29, Positions C.1 and C.2 and 1.45, Positions C.1 through C.9 with respect to seismic design classification and RCPB leakage detection system design are therefore, acceptable. The reactor coolant pressure boundary leakage detection system meets the acceptance criteria of SRP Section 5.2.5.

6.7 Main Steam Isolation Valve Leakage Control System

The applicant has stated that Nine Mile Point Unit 2 will not have a main steam isolation valve leakage control system (MSIVLCS). The applicant believes that the design of the MSIVs will limit leakage to amounts that will not result in unacceptable offsite radiological consequences following a postulated LOCA. We believe that the integrity of the valves can be ascertained by strict inservice inspection and testing requirements. The leakage requirements should be checked in accordance with technical specifications at the same frequency as would be required in a plant with a leakage control system. If the allowable leakage requirements of the technical specifications are exceeded then the valves may require repair, modification or replacement. Continued excessive leakage may require additional measures such as installation of a MSIVLCS. Space should be available in case this is required in the future. We conclude that Nine Mile Point 2 is acceptable at this time without a MSIVLCS.

9 AUXILIARY SYSTEMS

We have reviewed the design of the auxiliary systems necessary for safe reactor operation, shutdown, fuel storage, or whose failure might affect plant safety including their safety-related objectives, as well as the manner in which these objectives are achieved.

The auxiliary systems necessary for safe reactor operation or shutdown include the emergency service water system, emergency closed cooling system, ultimate heat sink, heating, ventilation, and air conditioning systems for the control room and areas housing safety-related equipment, essential portions of the compressed air system, and standby liquid control system.

The auxiliary systems necessary to ensure the safety of the fuel storage facility include new fuel storage, spent fuel storage, spent fuel pool cooling and cleanup system, fuel-handling systems, and fuel-handling area heating, ventilation, and air conditioning system.

We have also reviewed other auxiliary sytems to verify that their failure will not prevent safe shutdown of the plant or result in unacceptable release of radioactivity to the environment. These systems include: the plant service water system, the nuclear closed cooling water system, the demineralized water makeup system, potable and sanitary water system, the condensate storage facilities, the turbine building closed loop cooling water system, nonessential portions of the compressed air system, the equipment and floor drainage system, and heating, ventilation, and air conditioning systems for nonessential portions of the reactor building, and radwaste building, and turbine building.

Nine Mile Point, Unit 2 is independent of Unit 1, and therefore, the requirements of General Design Criterion 5 "Sharing of Structures, Systems and Components" do not apply. Exceptions to this are noted in each applicable section of this SER.

9.1 Fuel Storage Facility

9.1.1 New Fuel Storage

The new fuel storage facility was reviewed in accordance with Section 9.1.1 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below formed the basis for our evaluation of the new fuel storage facility with respect to the applicable regulations of 10 CFR Part 50.

The acceptance criteria for the new fuel storage facility include compliance with the guidelines of ANS 57.1, "Design Requirements for Light-Water Reactor Fuel Handling System," and ANS 57.3, "Design Requirements for New LWR Storage Facilities," as related to prevention of criticality and radioactivity releases. The guidelines contained in the "Review Procedures" portion of SRP Section 9.1.1 were used in lieu of ANS 57.1 and ANS 57.3.

The new fuel storage facility is located in the reactor building. The facility provides dry storage for a maximum of 270 fuel assemblies and includes the new fuel assembly storage racks and the storage vault that contains the storage racks.

The reactor building, which houses the new fuel storage facility, is designed to seismic Category I criteria as are the storage racks and vault. The building is also designed against flooding and tornado missiles (refer to Sections 3.4.1 and 3.5.2 of this SER). Thus, the requirements of General Design Criterion 2, "Design Basis for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.1, are satisfied.

The new-fuel storage vault is not located in the vicinity of any moderate- or high-energy lines or rotating machinery. Physical protection by means of separation is provided for new fuel from internally generated missiles and the effects of pipe breaks (refer to Sections 3.5.1 and 3.6.1 of this SER).

Since Nine Mile Point Unit 2 has its own new fuel storage facility and there is no sharing between units, the requirements of GDC 5, "Sharing of Structures, Systems and Components," are not applicable.

The new fuel storage storage facility is designed to store unirradiated, low emission, fuel assemblies. Accidental damage to the fuel would release relatively minor amounts of radioactivity that would be accommodated by the fuel storage facility ventilation system. The facility is accessible to plant personnel for inspection. Thus, the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," are satisfied.

The new fuel storage racks are designed to store the fuel assemblies in an array, which is sufficient to maintain a K_{eff} of less than 0.95 in the normal dry condition or abnormal completely water flooded condition. The racks are not designed to maintain a K_{eff} of 0.98 or less under optimum moderation (foam, small droplets, spray, or fogging) as identified in the guidelines of Section 9.1.1 of NUREG-0800. The condition of optimum moderation is precluded since the new fuel storage vault is provided with non-combustible covers. Although the covers are not watertight, they are designed to preclude inadvertent admission of optimum moderating fluid into the new fuel storage vault. The applicant will utilize administrative controls to preclude entry of sources of optimum moderation into the new fuel storage area during movement of fuel, thereby reducing the probability of such a condition. In addition, the floor of the vault is sloped to a drain to remove any water introduced into the vault. We find this approach acceptable. The racks are designed to preclude the inadvertent placement of a fuel assembly in other than the prescribed spacing. Thus, the requirements of General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling," are satisfied.

Based on our review, we conclude that the new fuel storage facility is in conformance with the requirements of General Design Criteria 2, 5, 61, and 62 as they relate to new fuel protection against natural phenomena, shared functions, radiation protection and prevention of criticality, and the guidelines of Regulatory Guide 1.29, Position C.1, relating to seismic classification and is, therefore, acceptable. The new fuel storage facility meets the acceptance

criteria of SRP Section 9.1.1 with the exception of the rack design for maintaining a K_{eff} of 0.98 or less under optimum moderation.

9.1.2 Spent Fuel Storage

The spent fuel storage facility was reviewed in accordance with Section 9.1.2 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of Section 9.1.2 was performed according to the guidelines provided in the "Review Procedures" portion of the SRP Section. Conformance with the acceptance criteria, except as noted below, formed the basis for our evaluation of the spent fuel storage facility with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria for the spent fuel storage facility include meeting various portions of the guidelines of ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." The guidelines contained in the "Review Procedures" portion of SRP Section 9.1.2 were used in lieu of ANS 57.2. Additionally, the acceptance criteria include Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles." Compliance with the guidelines of Regulatory Guide 1.115 are evaluated separately in Section 3.5.1.3.

The spent fuel storage facility is located in the reactor building. The spent fuel storage facility provides underwater storage for 4000 fuel assemblies (523 percent of the full core fuel load). The facility includes the storage racks which are high density type of stainless steel construction and the storage pool which is lined with stainless steel. The structure housing the facility is designed to seismic Category I criteria as are the storage racks, pool liner, gates, and the pool. The facility is also designed against flooding and tornado missiles (refer to Sections 3.4.1 and 3.5.2 of this SER). Therefore, we conclude that the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," Position C.3; 1.29, "Seismic Design Classification," Positions C.1 and C.2; and 1.117, "Tornado Design Classification," Positions C.1 through C.3, are satisfied for the spent fuel storage facility.

The spent fuel pool is not located in the vicinity of any high-energy lines or rotating machinery. Therefore, protection of spent fuel from internally generated missiles and the effects of pipe breaks by physical separation is provided (refer to Sections 3.5.1.1 and 3.6.1 of this SER). Thus the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of RG 1.13, Position C.3, concerning missile and environmental protection for spent fuel, are satisfied.

Since Nine Mile Point Unit 2 has its own spent fuel storage facility and there is no sharing with Unit 1, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

The seismic Category I storage rack arrangement provides a fuel storage array adequate to maintain the multiplication factor, K_{eff}, below 0.95 for both normal storage and in case of accidental dropping of a fuel assembly. The stainless steel racks contain Boraflux sheets which serve as a neutron absorbing material. The racks are designed to preclude the inadvertent placement of a fuel assembly in other than the prescribed spacing. The racks can withstand the impact of a dropped fuel assembly without unacceptable damage to the fuel and can withstand the maximum uplift forces exerted by the fuel-handling machine. However, the applicant should provide the following additional information in order for us to perform an independent evaluation of the racks' criticality capability:

1. A dimensional sketch of the fuel rack showing the location of the Boraflux sheets and the center-to-center spacing between assemblies.

2. The boron loading of the Boraflux sheets in terms of grams of Boron-10 per square centimeter between storage locations in sufficient data to permit this quantity to be desired.

3. Verify that the calculations are based on 3.6 w/o U-235 enrichment. Is this equivalent to the K_∞ value of I-131 used for the new fuel racks?

Thus, the requirements of General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control," and 62, "Prevention of Criticality in Fuel Storage and Handling," and the guidelines of Regulatory Guide 1.13, Positions C.1 and

C.4, concerning fuel storage facility design are not satisfied. The design of the storage pools includes an alarmed leakage detection system (for indication of excessive pool liner leakage), water level monitoring systems, and radiation monitoring systems in the control room. Thus, the design meets the requirements of General Design Criterion 63, "Monitoring Fuel and Waste Storage."

Based on our review, we conclude that the spent fuel storage facility is in conformance with the requirements of General Design Criteria 2, 4, and 63 as they relate to protection of spent fuel against natural phenomena, missiles, environmental effects, and performance monitoring, and the guidelines of Regulatory Guides 1.13, Positions C.3 and C.4, 1.29, Positions C.1 and C.2, and 1.117, Positions C.1 through C.3 relating to the facility's design basis, seismic classification, and protection against tornado missiles, and is, therefore, acceptable. However, we cannot conclude that the spent fuel storage facility meets the requirements of General Design Criteria 61 and 62 as they relate to facility's radiation protection and prevention of criticality and the guidelines of Regulatory Guide 1.13, Position C.1. The spent fuel storage facility meets the acceptance criteria of SRP Section 9.1.2 except as noted above. We will report resolution of our concerns in a supplement to this SER.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling and cleanup system was reviewed in accordance with Section 9.1.3 of the Standard Review Plan (SRP) NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below formed the basis for our evaluation of the spent fuel pool cooling and cleanup system with respect to the applicable regulations of 10 CFR Part 50.

The acceptance criteria for the spent fuel pool cooling and cleanup system includes meeting the guidelines of Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plant," and Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational

Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable." Compliance with the guidelines of Regulatory Guide 1.52 is evaluated separately in Section 9.4.2. Compliance with the guidelines of Regulatory Guide 8.8 is evaluated separately in Section 12.

The spent fuel pool cooling and cleanup system (SFPCS) is designed to maintain water quality and clarity and remove decay heat generated by spent fuel assemblies in the pool. The system includes all components and piping from inlet to exit from the storage pools, piping used for fuel pool makeup, and the cleanup filter/demineralizers to the point of discharge to the radwaste system. The design consists of two fuel pool cooling pump/heat exchanger trains and two spent fuel surge tanks. Each fuel pool cooling pump can be powered from redundant divisions of the Class 1E power system.

The SFPCS is housed in the reactor building which is seismic Category I and tornado protected. The system itself, with the exception of the cleanup portion, is designed to quality group C and seismic Category I criteria. In case of a seismic event, a parallel seismic Category I bypass line and redundant seismic Category I isolation valves have been provided at the cleanup system connections to the fuel pool cooling lines to isolate the nonseismic Category I portions of the system and thereby ensure that failure in that portion of the system has no adverse effect on safety-related equipment. Thus, the design satisfies the requirements of General Design Criterion 2, "Design Bases For Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," Positions C.1, C.2, and C.6, and Regulatory Guide 1.29, "Seismic Design Classification," Positions C.1 and C.2.

The SFPCS components are not located in the vicinity of other moderate or high-energy piping systems. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases" and the guidelines of Regulatory Guide 1.13, Position C.2, are satisfied.

Since Nine Mile Point Unit 2 has its own spent fuel pool cooling and cleanup system, thus the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are not applicable.

The system is accessible for routine visual inspection of the system components. One fuel pump and the heat exchanger, and one filter/demeralizer are in continuous operation. The redundant pump will be operated periodically to verify its operability. Thus, the requirements of General Design Criterion 45, "Inspection of Cooling Water Systems" and General Design Criterion 46, "Testing of Cooling Water Systems," are satisfied.

Either of the two spent fuel pool cooling trains will maintain the pool water temperature at 125° or less under the normal heat load conditions with a heat load of 12 MTBTU/hr. The normal condition assumes one-third core with full irradiation and 7-day decay, with the remainder of the spent fuel pool filled with 12 similar refueling discharges. If one pump and heat exchanger were lost under these conditions, the redundant pump and heat exchanger would be placed in service. In the case of an abnormal heat load of 31 MTBTU/hr when the full core must be unloaded, the residual heat removal (RHR) system would be used for additional spent fuel pool cooling capacity. Should the spent fuel pool cooling system be lost under these conditions, the pool water temperature will rise to 146°F with the RHR system providing pool cooling. The applicant has committed to running the RHR system of a shutdown reactor to maintain the pool water temperature below 150°F until the spent fuel pool cooling system could maintain the temperature below 150°F. We will require a Technical Specification that the reactor of Unit 2 not be started when the RHR system is providing pool cooling. The above pool water temperatures (normal and abnormal) are within the acceptance criteria of the SRP guidelines. Heat loads for the above storage model are based on BTP ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling," and Standard Review Plan Section 9.1.3.

No connections are provided to the spent fuel pool below the normal water level that may cause the pool to be drained, and, therefore, the fuel would not be uncovered should these lines fail. All lines that connect to the pool and extend below the safe shielding level of the pool water are equipped with syphon breakers, check valves, or other means to prevent inadvertent pool drainage. The service water system provides cooling water to the fuel pool heat exchanger under normal conditions. Backup cooling in emergencies is available from the safety-related reactor building closed loop cooling water system. In addition, the residual heat removal system can be utilized to supplement the

fuel pool cooling system by providing additional cooling during shutdown as described above. Based on the above, we conclude that the requirements of General Design Criterion 44, "Cooling Water," are met.

Normally makeup water for the spent fuel pool is provided to the spent fuel pool surge tanks from the condensate storage tank via the condensate makeup and drawoff system to replace losses due to leakage through the liner and evaporation. Emergency makeup water to the spent fuel pool is available from the seismic Category I portion of the service water system. Thus, the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," and the guidelines of Regulatory Guide 1.13, Position C.8, concerning pool makeup are satisfied.

The system incorporates control room alarmed pool water level, water temperature, and building radiation level monitoring systems, thus satisfying the requirements of General Design Criterion 63, "Monitoring Fuel and Waste Storage."

Based on our review, we conclude that the spent fuel pool cooling and cleanup system is in conformance with the requirements of General Design Criteria 2, 4, 44, 45, 46, 61, and 63 as they relate to protection against natural phenomena, missiles and environmental effects, cooling water capability, inservice inspection, functional testing, radiation protection and monitoring provisions, and the guidelines of Regulatory Guides 1.13, Positions C.1, C.2, C.6, and C.8, 1.29, Positions C.1 and C.2, and Branch Technical Position ASB 9-2, relating to the system's design, seismic classification, and design decay heat load, and is, therefore, acceptable. The spent fuel pool cooling and cleanup system meets the acceptance criteria of SRP Section 9.1.3.

9.1.4 Light Load Handling System (Related to Refueling)

The light load handling systems were reviewed in accordance with Section 9.1.4 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria, except as noted below,

formed the basis for our evaluation of the light load handling system with respect to applicable regulations of 10 CFR Part 50.

The acceptance criteria for the light load handling systems include meeting the guidelines of ANS 57.1, "Design Requirements for LWR Fuel Handling Systems." The guidelines contained in the "Review Procedures" were used in lieu of ANS 57.1.

The fuel handling system provides the means of transporting, handling, and storing fuel (both new and spent fuel) in the reactor building. The fuel handling system consists of equipment necessary to facilitate the periodic refueling of the reactor. The transfer of new fuel assemblies between the uncrating area and new fuel storage vault is accomplished using the reactor building polar crane (RBPC). The RBPC auxiliary hoist equipped with a general purpose grapple is used to transfer new fuel from the vault to the fuel storage pool. From this point, the new fuel is handled by the telescoping grapple on the refueling platform.

Since Nine Mile Point Unit 2 has its own independent fuel handling system, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

The entire system is housed within the reactor building and the containment which are seismic Category I, flood and tornado protected (see Sections 3.4.1 and 3.5.2 of this SER). The RBPC and the refueling platform are designed to seismic Category I criteria so that they will not fail in a manner which results in unacceptable consequences such as fuel damage or damage to safety-related equipment. However, fuel-handling systems are not required to function following a safe shutdown earthquake (SSE). The new fuel inspection stand and the jib crane which is used for fuel preparation during refueling are designed to seismic Category I requirements. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Bases," Positions C.1 and C.6, and 1.29, "Seismic Design Classification," Positions C.1 and C.2, relating to protection of safety-related equipment and spent fuel from the effects of earthquakes, are satisfied.

The refueling platform is used to transport fuel and reactor components to and from pool storage and the reactor vessel. The fuel grapple hoist of the refueling platform has redundant load handling components so that no single component failure will result in a fuel bundle drop. Redundant interlocks and limit switches prevent accidental collision with pool walls. The design of the fuel grapple in its fully raised position maintains adequate water shielding. Spent fuel will be handled with telescoping grapples designed to assure adequate water shielding.

[The applicant has not provided the results of an analysis which verifies that the maximum potential kinetic energy resulting from dropping each object of less weight than a fuel bundle and its handling tool which could be handled over spent fuel will not exceed the effects of the fuel handling accident described in Section 15 of the FSAR. A list of such objects has not been provided.]

[Based on the above, we cannot conclude that the requirements of General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control," and 62, "Prevention of Criticality in Fuel Storage and Handling," and the guidelines of Regulatory Guide 1.13, Position C.3, with respect to prevention of unacceptable radioactivity releases and criticality accidents, are satisfied.]

Based on our review, we concude that the fuel handling system is in conformance with the requirements of General Design Criterion 2 and the guidelines of Regulatory Guides 1.13, Positions C.1 and C.6, and 1.29, Positions C.1 and C.2, with respect to protection of safety-related equipment and spent fuel from the effects of earthquakes, and is, therefore, acceptable. [We cannot conclude conformance with the requirements of General Design Criteria 61 and 62 and the guidelines of Regulatory Guide 1.13, Position C.3, with respect to the prevention of unacceptable radioactive releases and criticality accidents until the applicant provides acceptable responses to our concerns. The light load handling system does not meet the acceptance criteria of SRP Section 9.1.4. We will report resolution of this item in a supplement to this SER.

9.1.5 Overhead Heavy Load Handling

The overhead heavy load handling systems were reviewed in accordance with Section 9.1.5 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria, except as noted below, formed the basis for our evaluation of the overhead heavy load handling systems with respect to the applicable regulations of 10 CFR Part 50.

The acceptance criteria for the overhead heavy load handling systems include meeting the guidelines of ANS 57.1, "Design Requirements of Light Water Reactor Fuel Handling Systems," and ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants." The guidelines contained in the "Review Procedures" and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," were used in lieu of ANS 57.1 and ANS 57.2.

The overhead heavy load handling systems consist of equipment necessary for the safe handling of the spent fuel cask and for safe disassembly and reassembly of the reactor vessel head and internals during refueling operations. The reactor building polar crane is used for handling of heavy loads over the refueling floor in the reactor building.

Since Nine Mile Point Unit 2 has its own overhead handling system, the requirements of General Design Criteria 5, "Sharing of Structures, Systems, and Components," are not applicable.

The entire system is housed within the reactor building which is seismic Category I, flood and tornado protected (refer to Sections 3.4.1 and 3.5.2 of this SER). The 120-ton reactor building polar crane (RBPC) is designed to seismic Category I criteria so that it will not fail in a manner which results in unacceptable consequences such as fuel damage or damage to safety-related equipment. However, the crane is not required to function following an SSE. The RBPC used for handling the 120-ton spent fuel shipping cask, which is the maximum critical load, is single failure proof and is designed to the requirements of Crane Manufacturer's Association of America Specification No. 70.

The crane is used to move the reactor vessel head, shroud head/separator, shield plugs, and dryer assembly. Therefore, the design satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Bases," Positions C.1 and C.6, and 1.29, "Seismic Design Classifications," Positions C.1 and C.2.

The spent fuel cask pool is separated from the fuel storage pool by a canal with a seismic Category I gate. Should the cask be dropped and tip after falling on the guard walls surrounding the cask loading area, its center of gravity is such that it will not fall outside the cask-loading area and therefore will not affect spent fuel in the spent fuel storage pool. Further, the RBPC is equipped with interlocks which prevent the cask from being carried over the fuel pool. The crane coverage area does not include any safety-related equipment. A dropped cask cannot, therefore, result in fuel damage or damage safety-related equipment. Thus, we conclude that the requirement of General Design Criteria 4, "Environment and Missile Design Bases," and 61, "Fuel Storage and Handling and Radioactivity Control," and the guidelines of Regulatory Guide 1.13, Positions C.3 and C.5, have been satisfied for handling of the spent fuel cask.

As a result of Generic Task A-36, "Control of Heavy Loads Near Spent Fuel," a set of guidelines was developed to assure safe handling of heavy loads over structures, systems, and components important to safety. These recommendations were documented in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

Following the issuance of NUREG-0612, a generic letter dated December 22, 1980, was sent to all operating licenses and holders of construction permits requesting that responses be prepared to indicate the degree of compliance with the guidelines of NUREG-0612. The responses were to be made in two stages. The first response (Phase I, Section 5.1.1 of NUREG-0612) was to identify the load handling equipment within the scope of NUREG-0612 and to describe the associated general load handling operations such as safe load paths, procedures, operator training, special and general purpose of lifting devices, the maintenance, testing, and repair of equipment, and the handling equipment specifications. The

second response (Phase II) was intended to show that either single-failure-proof handling equipment was not needed or that single-failure-proof equipment has been provided.

In the December 22, 1980 letter, the applicant was requested to review their provisions for handling and control of heavy loads at the facility to determine the extent to which the guidelines of NUREG-0612 are satisfied and to commit to mutally agreeable changes and modifications that would be required in order to fully satisfy these guidelines.

The applicant has not responded to the generic letter and our question no. 410.28, therefore, we will require that the applicant comply with the guidelines of Section 5.1.1 of NUREG-0612 (Phase I - the 6-month response to the NRC generic letter dated December 22, 1980) prior to license issuance. We further require that a condition be placed in the license requiring that following the first refueling outage the applicant shall have made commitments acceptable to the staff regarding the guidelines of Sections 5.1.2 through 5.1.6 of NUREG-0612 (Phase II - 9-month responses to the NRC generic letter dated December 22, 1980).

In respects other than those related to the evaluation of the applicant's response to NUREG-0612, we find that the requirements of General Design Criteria 4 and 61 and the guidelines of Regulatory Guide 1.13, Positions C.3 and C.5, are met.

Based on our review, we conclude that the overhead handling systems are in conformance with the requirements of General Design Criteria 2, 4, and 61 as related to protection against natural phenomena, protection of safety-related equipment from the effects of internal missiles, and safe handling and storage of the fuel, the guidelines of Regulatory Guides 1.13, Positions C.1., C.3, C.5, and C.6, and 1.29, Positions C.1 and C.2, with respect to overhead crane interlocks and maintaining plant safety in a seismic event. The overhead heavy load handling system meets the acceptance criteria of SRP Section 9.1.5 except as noted above. We will report resolution of our concern regarding compliance with NUREG-0612 criteria in a supplement to this SER.

9.2 Water Systems

9.2.1 Station Service Water System

The service water system was reviewed in accordance with Section 9.2.1 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section except as noted below. Conformance with

9.2 Water Systems

9.2.1 Station Service Water System

The service water system was reviewed in accordance with Section 9.2.1 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the service water system with respect to the applicable regulations of 10 CFR Part 50.

The service water system (SWS) supplies cooling water to the plant from Lake Ontario which serves as the ultimate heat sink (UHS) as discussed in Section 9.2.5 of this SER. The SWS operates during hot standby, cold shutdown, and accident conditions. Under these conditions, the SWS provides cooling to the following essential plant components: the residual heat removal (RHR) heat exchangers, the emergency diesel generator coolers, the control building chilled water chillers, RHR pump seal coolers, DBA hydrogen recombiners, reactor building ventilation recirculation cooling coils, and reactor building, control building, diesel generator building, and service water pump bay unit coolers. Additionally, the SWS is capable of supplying water to flood containment for post accident recovery, to provide emergency makeup to the fuel pool.

The SWS is also designed to provide cooling water to the secondary sides of the reactor building closed loop cooling water (RBCLW) and turbine building closed loop cooling water (TBCLW) heat exchangers during normal operation and planned outages. In addition, the system is designed to provide makeup water to the circulating water system (CWS) and cooling water to miscellaneous nonessential turbine and reactor building components during normal plant operation.

The SWS consists of three loops; two are safety-related and one is nonsafety-related. Redundant essential components are fed by separate safety-related loops. During an accident the nonsafety-related loop is isolated from the two

safety-related loops. There are no essential heat loads on the nonsafety-related loop. The water enters the SWS from Lake Ontario through two intake structures, passes through the trash racks and travelling water screens and enters the SWS intake bay. A conservative margin is provided over the required net positive suction head (NPSH) at the design minimum low water level in the intake bay. Service water is pumped from the intake bay through an automatically operated strainer located in the discharge line of each pump. From the strainers, the service water is directed to a common header in the screen well building. Two motor operated isolation valves are provided in the header. These isolation valves separate the SWS into two separate redundant loops, an "A" and "B" loop. All essential components in the A and B loops are powered from separate redundant emergency power supplies. Each loop is provided with three 50% capacity pumps. The isolation vaves which esparate the safety-related sections of the SWS from the nonsafety-related portions close automatically when there is a LOCA signal coincident with either loop or low header pressure in respective loop. These valves are designed to seismic Category I criteria.

The system is housed in seismic Category I, flood- and tornado protected structures. The safety-related portion of the SWS including piping and valves is designed to seismic Category I and Quality Group C. The intake structure and bar racks are class IE electrically heated. Thermostatically controlled electric unit heaters maintain the building above the minimum temperatures to provide freeze protection. The applicant has stated that there is no buried nonseismic Category I pipe in the Unit 2 SWS. However, he has not provided any information regarding other nonseismic pipe buried near the safety-related pipe which could rupture during an SSE and cause potential failure in the SWS due to soil erosion. Therefore, we cannot conclude that the SWS design meets the General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena" and the requirements of Regulatory Guide 1.29, "Seismic Design Classification" Positions C1 and C2.

The design of the SWS described above ensures that system function is not lost assuming a single active component failure coincident with a loss of offsite power under design basis accident heat load conditions. The applicant has stated that biofouling organism do not inhabit Lake Ontario, therefore no treatment

program is necessary. Therefore, we conclude that the requirements of General Design Criterion 44, "Cooling Water" are satisfied.

The operability of. the SWS pumps is demonstrated during normal plant operation. The spare pumps will be cycled periodically to ensure their availability.

The system design also incorporates provisions for accessibility to permit inservice inspection as required. Therefore, we conclude that the design meets the requirements of General Design Criteria 45, "Inspection of Cooling Water System" and 46, "Testing of Cooling Water Systems."

Based on the above, we cannot conclude that the service water system meets the requirements of GDC 2, with respect to the system's protection against natural phenomena and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, with respect to the system's seismic classification. However, the SRP service water system meets the requirements of GDC, 44, 45 and 46 with respect to the system's capability for transferring the required heat loads, inservice inspection and functional testing. The service water system does not meet the acceptance criteria of SRP Section 9.2.1. We will report resolution of our concern in a supplement to the SER.

9.2.2 Reactor Auxiliary Cooling Water System (Reactor Building Closed Loop Cooling Water System)

The reactor building closed loop cooling water system (RBCLCW) system was reviewed in accordance with Section 9.2.2 of NUREG-0800. An audit of review of each of the areas listed in the "Areas of Review" portion of Section 9.2.2 was performed according to the guidelines provided in the "Review Procedures" portion of the SRP Section. Conformance with the acceptance criteria formed the basis for the evaluation of that the reactor building closed loop cooling water system with respect to the applicable regulations of 10 CFR Part 50.

The reactor building closed loop cooling water (RBCLW) system provides cooling to reactor auxiliary system equipment and accessories during normal plant operating conditions. Heat is transferred from the RBCLCW by the Service

Water System to the normal heat sink (Lake Ontario). The RBCLCW system is not required to operate during emergency or accident conditions. However, during emergency or accident conditions, portions of the system serve as a seismic Category I pressure boundary for backup cooling provided from the service water system to the spent fuel pool cooling heat exchangers, RHR pump seal coolers, recirculation pump seal coolers, and ADS air compressor. The RBCLW systems consists of three 50-percent capacity pumps and 3 heat exchangers, one expansion tank, piping and valves. These components do not perform a safety function.

The safety-related RBCLCW supply headers which serve the above indicated equipment in an emergency are isolated from the non-safety related portion of the RBCLW system by means of check valves (one in each header) in series with normally open motor-or solenoid-operated isolation valves. These valves are powered from the Class IE buses. These valves will isolate the safety-related portion of the RBCLCW system on a loss of pressure in the nonsafety-related portion of the system.

The safety-related portion of the RBCLCW system is housed in seismic Category I, flood- and tornado-protected structures (refer to Sections 3.4.1 and 3.5.2 of this report). The safety-related portion of the RBCLCW system is designed to Category I, Quality Group B. Thus the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification" Positions C.1 and C.2 and satisfied.

Since the RBCLCW system safety function is only to serve as a pressure boundary for cooling water supplied by the SWS in emergencies and it does not transfer heat load, the requirements of General Design Criteria 44, "Cooling Water," 45, "Inspection of Cooling Water System" and 46, "Testing of Cooling Water System" are not applicable. (Refer to Section 9.2.1 for discussion of heat transfer from essential equipment during emergencies).

Based on the above, we conclude that the RBCLCW system meet the requirements of General Design Criterion 2 with respect to protection against natural

phenomena, and the guidelines of Regulatory Guide 1.29 Positions C.1 and C.2 with respect to seismic classification and is, therefore acceptable. The RBCLCW system meet the applicable acceptance criteria of SRP Section 9.2.2.

9.2.3 Demineralized Water Makeup System (Makeup Water Treatment System)

The demineralized water makeup system (Makeup Water Treatment System) was reviewed in accordance with Section 9.2.3 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the demineralized water makeup system with respect to the applicable regulations of 10 CFR Part 50.

The nonsafety-related (nonseismic Category I) makeup water treatment system includes all components and piping associated with the system from the demineralized water storage savers to the point of dischage to other systems. The system has no safety related function. Protection from flooding for safety-related equipment resulting from failure of the system is discussed in Sections 3.4.1 and 9.3.3 of this SER. Drains and overflows from various system equipment and waste chemicals are collected in a sump tank. The system is capable of fulfilling the normal operating requirements of the facility for acceptable makeup water with the necessary component redundancy. The applicant stated that the demineralized water storage tanks, which are part of the makeup water system include fill and a supply connection for demineralized water with Unit 1. This assures demineralized water availability should the Unit 2 water treatment system need to be secured for maintenance or other purposes. Entry of potentially radioactive water into the system is precluded by assuring a greater pressure for demineralized makeup water than in the potentially radiaoctive sources to which it discharges. Alarmed instrumentation has been provided to prevent delivery of off-specification water to safety-related systems. Failure of the system will not affect plant safety as described above; thus the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena" and 5, "Sharing of Structures,

Systems, and Components" and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.2, are met.

Based on the above, we conclude the system meets the requirements of General Design Criteria 2 and 5 with respect to the need for protection against natural phenomena and shared functions and meets the guidelines of Regulatory Guide 1.29, Position C.2, concerning the seismic classification and is, therefore, acceptable. The makeup water treatment system meets the acceptance criteria of SRP Section 9.2.3.

9.2.4 Potable and Sanitary Water Systems (Domestic Water and Sanitary Drains and Disposal Systems)

The potable and sanitary water system (domestic water and sanitary drains and disposal systems) was reviewd in accordance with Section 9.2.4 of the Standard Review Plan (SRP) NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the potable and sanitary water systems domestic water and sanitary drains and disposal systems with respect to the applicable regulations of 10 CFR Part 50.

The nonsafety-related (nonseismic Category I) domestic water and sanitary drains and disposal systems provide water for human consumption and sanitary purposes. Oswego city water is the normal source of domestic water. The domestic water and sanitary drains and disposal systems are not safety related. They are not connected to any potentially radioactive process systems. Thus, we conclude that the requirements of General Design Criterion 60, "Control of Releases and Radioactive Materials to the Environment," are satisfied.

Protection from flooding for safety-related equipment resulting from failure of the system is discussed in Sections 3.4.1 and 9.3.3 of this SER. Failure of this system does not affect plant safety as described above.

Based on our review, we conclude that the domestic water, sanitary drains and disposal systems meet the requirements of General Design Criterion 60 with respect to prevention of release of potentially radioactive water, and is, therefore, acceptable. The domestic water, and sanitary drains and disposal systems meet the acceptance criteria of SRP Section 9.2.4.

9.2.5 Ultimate Heat Sink

The ultimate heat sink was reviewed in accordance with Section 9.2.5 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below, formed the basis for our evaluation of the ultimate heat sink with respect to the applicable regulations of 10 CFR Part 50.

The acceptance criteria for the ultimate heat sink includes Regulatory Guide 1.72, "Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin." The ultimate heat sink for Nine Mile Point Unit 2 is Lake Ontario, therefore, this acceptance criterion is not applicable.

The normal heat sink and the ultimate heat sink (UHS) for Nine Mile Point 2 is Lake Ontario (for a further discussion of the UHS (Lake Ontario) refer to Section 2.4 of this SER). Cooling water from the lake is provided to the plant under all operating conditions by the service water system (SWS). For a discussion of the SWS, refer to Section 9.2.1 of this SER.

Two identical intake structures are located approximately 950 and 1050 ft from the existing shoreline of Lake Ontario. These structures are located at lake bottom (el 224.5 ft.). A minimum water depth of approximately 10 ft is maintained over the structures. Each structure is independently connected to the onshore screenwell by a concrete intake encasement. The encasements are located within two tunnels.

Bar racks are provided to prevent large debris from entering the intake system. Electrical heating elements are provided to eliminate the potential for the ice adhesion to the racks. These heating elements receive power from two separate Class IE electrical buses. Trash racks are provided upstream of the travelling water screens to prevent floating debris from entering the flow path to the SWS pumps should the screens become dislodged.

The SWS pumps discharge into a discharge bay located in the screenwell building. From there the warm water is discharged through a tunnel/pipe to which is attached a diffuser oriented away from the SWS intakes located approximately 1500 ft offshore and 12 ft. below the water level of the lake.

The applicant has stated that the maximum recorded ice thickness for Lake Ontario of 20 inches will not affect UHS performance.

The intake structures, discharge pipe/tunner, bar racks, electrical heating elements, intake pipes, screenwell substructure, retangular rotary gates, and trash racks are designed as seismic Category I components. The service water pump house is tornado protected structure within the screenwell building. Other UHS structures are submerged or located below grade, thus providing inherent protection from tornado missiles. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.29, Position C.1 and 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Positions C.2 and C.5 are satisfied.

The applicant used a mathematical model to analyze the effects of the warm SWS discharge on the lake and the potential for recirculation to the SWS intakes. The analysis indicated that the distance between the intake structure and discharge structure will adequately prevent the recirculation of warm water. The applicant's evaluation also indicated that the total heat rejected by the SWS will have only a negligible thermal effect on a localized area of the lake at the discharge structure.

The applicant's analysis also demonstrated the capability of the Lake Ontario and service water system to meet the cooling requirements of the plant. The

applicant utilized BTP ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling," criteria in this analysis. The analysis indicates that sufficient water is available for 30 days for assuring plant cooling water safety functions following a design basis LOCA and concurrent limiting signle failure under the worst 30-day site meteorology. Therefore, we conclude that the design meets the requirements of General Design Criterion 44, "Cooling Water" and the guidelines of Regulatory Guides 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Positions C.2 and C.3 and 1.29, "Seismic Design Classification Position C.1 are satisfied.

The applicant has stated that the Nine Mile Point Unit 1 and the James A. Fitzpatrick Plant also use the Lake Ontario as an UHS. Their analysis of the UHS performance for Nine Mile Point 2 indicates that this sharing will not affect the heat removal capability of Unit 2, and therefore the requirements of Design Criterion 5, "Sharing of Structures, Systems and Components" are satisfied.

General Design Criteria 45, "Inspection of Cooling Water System" and 46, "Testing of Cooling Water System" are not applicable since the UHS is a passive design.

Based on the above, we conclude that the UHS meets the requirements of General Design Criteria 2, 5, and 44, with respect to protection against natural phenomena and missiles, sharing of structures, heat dissipation capability, and the guidelines of Regulatory Guides 1.27, Positions C.2 and C.3 and 1.29 Position C.1 and BTP ASB-9-2 with respect to the capability to remove sufficient decay heat to maintain plant safety and seismic design classification and is, therefore, acceptable. The ultimate heat sink meets the applicable acceptance criteria of SRP Section 9.2.5.

9.2.6 Condensate Storage Facilities

The condensate storage and transfer system was reviewed in accordance with Section 9.2.6 of the Standard Review Plan (SRP) NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the condensate storage and transfer system with respect to the applicable regulations of 10 CFR Part 50.

The nonsafety-related (quality Group D, nonseismic Category I) condensate storage and transfer system includes all components and piping associated with the system from the storage tanks to the points of connection or interfaces with other systems. The condensate storage system serves as a reserve source of water for the high pressure core spray (HPCS) system and reactor core isolation cooling (RCIC) system, as well as for the control rod drive pumps. Additionally, the condensate storage facilities provide water for refueling activities, condenser hotwell makeup, and spent fuel pool makeup.

The system was evaluated and found to have no functions necessary for achieving safe reactor shutdown conditions or for accident prevention or mitigation. a minimum storage capacity of 135,000 gallons of the 450,000 gallons capacity is reserved for the RCIC and HPCS systems. However, the safety-related water source for the HPCI and RCIS systems is the suppression pool. Thus, the requirements of General Design Criteria 44, "Cooling Water System," 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water System" are not applicable.

The condensate storage system is located in two condensate storage building. The condensate storage tanks, necessary piping and pumps are housed in the condensate storage building. The HPCS and RCIC systems condensate storage tank suction and test return lines are provided with seismic Category I, redundant Class 1E powered Quality Group B isolation valves. These valves provide adequate isolation from the nonsafety-related portion of the system.

Protection from flooding for safety-related equipment resulting from failure of the system is discussed in Sections 3.4.1 and 9.3.3 of this SER. Thus, the system meets the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.2.

Based on our review, we conclude the condensate storage and transfer system meets the requirements of General Design Criterion 2 with respect to the need for protection against natural phenomena and the guidelines of Regulatory Guide 1.29, Position C.2, concerning its seismic classification and is, therefore, acceptable. The condensate storage and transfer system meets the acceptance criteria of SRP Section 9.2.6.

9.2.7 Turbine Building Closed Loop Cooling Water System

The turbine building closed loop cooling water (TBCLW) system was reviewed in accordance with section 9.2.1 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP Section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the TBCLW with respect to the applicable regulations of 10 CFR Part 50.

The TBCLW system is designed to remove heat from various nonsafety-related heat exchangers in the turbine building and radwaste building. The system serves as an intermediate cooling distribution loop that transfers heat from equipment served to the service water system. The TBCLW system is not required to function to support safe shutdown of the reactor or to support the operation of any nuclear safety-related equipment. Therefore, the requirements of General Design Criteria 44, "Cooling Water," 45, "Inspection of Cooling Water System" and 46, "Testing of Cooling Water System" are not applicable.

The TBCLW system is located in the turbine building and consists of a single loop with three 50% capacity pumps in parallel (one on standby) feeding three 50%-capacity TBCLW heat exchangers also arranged in parallel (one on standby).

The TBCLW system is not interconnected with any safety-related systems an is separated from safety-related equipment. Therefore, fialure of this system will not effect the safety-related equipment requirements of General Design Criterion 2, and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.2 are satisfied.

Based on the above, we conclude that the TBCLW system meets the applicable requirements of General Design Criteria 2, with respect to the system's protection against natural phenomena and the guidelines of Regulatory Guide 1.29, Position C.2, with respect to the system's seismic classification. The TBCLW system meets the applicable acceptance criteria of SRP Section 9.2.2.

9.2.8 Plant Chilled Water System

The plant chilled water (PCW) system was reviewed in accordance with Section 9.2.2 of the Standard Review Plan (SRP) NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the, "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the plant chilled water system (PCW) with respect to the applicable regulations of 10 CFR 50.

The plant chilled water (PCW) system consists of two subsystems, the control building chilled water (CBCW) subsystem and the ventilation chilled water (VCW) subsystem. These two subsystems are not interconnected.

a. Control building chilled water (CBCW) subsystem

The CBCW sybsystem provides cooling for personnel and equipment in the control room, relay room, remote shutdown room and computer room. The CBCW subsystem is safety-related and seismically qualified with the exception of that portion serving the computer room. The CBCW subsystem is designed to perform during normal operation, plant shutdown, or accident conditions without loss of function.

The CBCW subsystem consists of a closed loop, and contains two redundant trains. Each train consists of a water chiller, a chilled water circulating pump, one expansion tank and associated controls and piping. Each train is capable of meeting the total chilled water demand for the CBCW subsystem.

Water is provided to each chiller from the service water system. (See Section 9.2.1 of the SER for discussion of the service water system). Each chilled water train has separate condenser water connections to the corresponding loops of the service water system. The service water system is capable of supplying water to the chiller condensers during all modes of plant operation. The CBCW subsystem is designed to seismic Category I Quality Group C criteria and conforms to the single-failure criterion. The subsystems receive power from offsite sources during normal operation and from the standby diesel generators if offsite power is lost. The equipment associated with the CBCW subsystem is located in a separate room. This subsystem is designed to operate during accident conditions and nonsafety-related equipment can be isolated from the safety-related equipment with safety-related isolation valves.. The applicant has demonstrated that the safety-related components under postulated accident conditions. Therefore the CBCW subsystem meets the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," and 44, "Cooling Water," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification" Positions C.1 and C.2.

The CBCW subsystem is designed to permit periodic inspection and maintenance of active components. Instruments and controls are provided for testing of the subsystem during normal operation or scheduled shutdown. Therefore, the CBCW subsystem meets the design criteria 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water System."

b. Ventilation Chilled Water (VCW) Subsystem

The VCW subsystem is located in the chiller building and provides cooling for personnel and equipment in the turbine building, the normal switchgear building and the radwaste building. The subsystem has no safety-related function and failure or malfunction of the VCW subsystem will not compromise any safety-related system or component or prevent safe reactor shutdown.

The VCW subsystem consists of three 50 percent capacity hot water absorption liquid chillers, two 100 percent capacity chilled water circulating pumps, one expansion tank controls and associated piping. This subsystem is designed as nonseismic Category I and Qaulity Group D. Water is provided to each chiller from the service water system. (Refer to Section 9.2.1 of this SER for a discussion of the service water system). There is no safety-related equipment in the vicinity of the VCW subsystem. Failure of this system will not have any effect on safety-related systems. Therefore, the VCW subsystem meets the requirements of General Design Criteria 2 and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.2.

Based on the above, we conclude that the safety-related and nonsafety-related portions of the plant chilled water system meet the requirements of General Design Criterion 2, 44, 45 ad 46 with respect to protection against natural phenomena, cooling water, and inspection and testing of cooling water systems, and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2. The plant chilled water system meets the applicable acceptance criteria of SRP Section 9.2.2.

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

The compressed air systems were reviewed in accordance with Section 9.3.1 of the Standard Review Plan (SRP) NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below formed the basis for our evaluation of the compressed air system with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria for the safety-related compressed air systems includes meeting General Design Criterion 1, "Quality Standards and Records." Compliance with the requirements of General Design Criterion 1 are evaluated separately in Section 3.2 of this SER.

The compressed air systems consist of the instrument and service air systems, the breathing air system and an alternate air supply connection to the automatic depressurization system. All compressed air systems are nonseismic Category I, Quality Group D, with the exception of the piping which penetrates containment and the isolation valves which are seismic Category I, Quality Group B and are located in seismic Category I structure.

The instrument air system supplies clean, dry, oil-free air to all air-operated instrumentation and valves in accordance with the air quality criteria of ANSI MC11.1-1976 (IAS-S7.3) "Quality Standard for Instrument Air Systems." the service air system supplies air for various maintenance purposes. The breathing air system supplies clean, oil-free, air of breathable quality to various areas throughout the plant. The above systems are not required to achieve safe reactor shutdown or to mitigate the consequences of an accident. Failure of the service air, instrument air and breathing air systems will not prevent safety-related components or systems from performing their intended safety functions.

The instrument air system consists of three parallel trains of compressors, coolers, moisture separators and air receivers feeding two refrigerant type dryers and filter banks with a common discharge header. The service air system supply header is connected to the common compressed air supply header upstream of the instrument air dryers. The air compressors are operated from offsite power.

An isolation valve on the service air system supply header will close when the common compressed air supply pressure drops to less than 85 psig. A separate instrument air system receiver tank is provided downstream of the refrigerant dryers. The breathing air system consists of separate compressor equipment with an inlet filter and after coolers.

The applicant has stated that air operated valves in safety-related systems are supplied with nitrogen bottles as a backup to the instrument air system. The safety and relief valves, and automatic depressurization system(ADS) valves which are required for safe shutdown under emergency conditions are

included in the backup nitrogen supply system. However, we are unable to confirm that the backup nitrogen supply system is designed to assure operation of air operated valves (including the ADS valves) for sufficient time following loss of the instrument air system in an SSE with concurrent loss of offsite and that it is adequately isolated from the nonsafety-related instrument air system. The applicant should provide drawings of the nitrogen supply system and provide further discussion regarding its design capability.] Therefore, we cannot conclude that the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification" Positions C.1 and C.2 are satisfied.

Based on the above, we cannot conclude that the compressed air systems meet the requirements of General Design Criterion 2 regardin the protection against natural phenomena and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2. We will report resolution of our concerns in a supplement to this SER. The compressed air system does not meet the applicable acceptance criteria of SRP Section 9.3.1. We will report resolution of our concern in a supplement to this SER.

9.3.3 Equipment and Floor Drainage System

The equipment and floor drain system was reviewed in accordance with Section 9.3.3 of the Standard Review Plan (SRP) (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the equipment and floor drain system with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related (Quality Group D, nonseismic Category I) equipment and floor drainage system includes all piping from equipment or floor drains to the sump, sump pumps, and piping necessary to carry potentially radioactive and nonradioactive effluents through separate subsystems. ~~However, all~~ Containment and reactor building penetrations and associated isolation valves are seismic Category I, Quality Group B. Potentially radioactive drainage is

Insert

[The applicant has not committed to perform periodic air quality testing of the air systems to assure maintaining compliance with the air quality standards of ANSI MC11.1-1976]

collected in floor and equipment drain sumps in each building and discharged to the radwaste processing system. Drainage from nonradioactive sources such as plumbing fixtures and roof drains are discharged to the sanitary waste treatment system.

There are no interconnections between the radioactive and nonradioactive drainage systems. Thus, we conclude that the system design meets the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment."

Our review considered those safety systems needed to provide safe plant shutdown and the physical location of those systems with regard to potential in-plant flooding. Because of their location at the lowest elevation in the reactor building, the ECCS equipment rooms which contain components required for safe plant shutdown were considered of particular importance with respect to provisions for prevention of water accumulation.

Redundant ECCS equipment is located in separate cubicles. Each ECCS cubicle is watertight and is equipped with watertight doors, penetration seals, and a separate drain line to the reactor building sump inlet header. Flooding of an ECCS cubicle will only result in the potential failure of one ECCS pump. Backflooding of the ECCS cubicle is prevented by use of nonseismic Category I check valves or drain plugs. [The applicant has not provided an acceptable response to our concern regarding flooding due to a rupture of nonseismic Category D piping, vessels, or tanks, or due to the failure of a backflow prevention device in the drainage system.] The ECCS cubicles have seismic Category I water level instrumentation to alarm in the control room on high water level in the event of flooding caused by blockage in the drains. [The applicant also has not provided a response to our question 410.38 concerning the drainage of leadage water away from safety related components or systems.] Therefore, we cannot conclude that the system design meets the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," and 4, "Environmental and Missile Basis," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Positions C.1 and C.2, with respect to the failure of the drainage system resulting in potentially unacceptable safety-related equipment failure.

Based on our review, we cannot conclude that adequate protection against flooding of safety-related equipment is provided in the event of failure of the drainage system. We cannot, therefore, conclude that the system meets the requirements of General Design Criteria 2 and 4, with respect to the need for protection against natural phenomena, pipe breaks, and environmental effects (flooding), and the guidelines of Regulatory Guide 1.29, Position C.2, with respect to seismic classification. However, we conclude that the requirements of GDC 60 regarding protection against inadvertent release of potentially radioactive liquids to the environment through drainage paths are satisfied. The equipment and floor drain system does not meet the acceptance criteria of SRP Section 9.3.3. We will report resolution of our concern item in a supplement to this SER.

9.3.5 Standby Liquid Control System

The standby liquid control system was reviewed in accordance with Section 9.3.5 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the standby liquid control system with respect to the applicable regulations of 10 CFR 50.

The standby liquid control system (SLCS) is a reactivity control system, its purpose being to inject sodium pentaborate into the reactor to provide an independent means for achieving cold shutdown should the normal reactivity control system become inoperable; thus satisfying the requirements of General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

The system consists of a boron solution tank, a test water tank, two positive displacement pumps, two explosive-actuated valves, associated local valves, piping and controls located in the reactor building. An electrical resistance heating system maintains the solution storage tank and pump suction lines between 75 and 85 degrees Fahrenheit to prevent precipitation of the sodium pentaborate from solution during storage. High and low liquid level and temperature are alarmed in the control room. The explosive-actuated valves

provide assurance that they can be opened when needed. The design of the valve assures that boron will not leak into the reactor even during SLCS pump testing. The two parallel pumps take suction from the storage tank and pump the solution into the reactor vessel via a common injection line. The discharge from each pump is provided with a check-valve. Each pump and its associated valves are powered from separate emergency AC power supplies. They are arranged such that failure of a single pump or explosive valve will not prevent adequate amounts of sodium pentaborate solution from entering the reactor vessel to accomplish shutdown.

System initiation is accomplished by manual actuation of key-locked switches on the control room panel. Changing switch status to "run" starts an injection pump, actuates an explosive valve, and closes reactor water cleanup system isolation valves to prevent loss or dilution of boron. A similar procedure is used to actuate the other SLC5 train should the first train fail.

The SLCS is located in the reactor building and containment which are all seismic Category I, flood and tornado protected. All portions of the SLCS necessary for injection of sodium pentaborate into the reactor are seismic Category I, Quality Group B, or Quality Group A if they are part of the reactor coolant pressure boundary. Thus the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.1, are met.

The SLCS is designed to function in the expected environmental conditions. The protection provided the SLCS from missiles is discussed in Section 3.5.1 of this SER and protection from failure of high- and moderate-energy piping systems is discussed in Section 3.6.1 of this SER.

The SLCS has adequate redundancy such that no single active failure will compromise its functional capability. The injection portion of the system can be functionally tested by injecting demineralized water from a test tank into the reactor. Thus, the requirements of General Design Criterion 27, "Combined Reactivity Control Systems Capability" are satisfied.

Based on the above, we conclude that the standby liquid control system is in conformance with the requirements of General Design Criteria 2, 26 and 27, as they relate to protection against natural phenomena, system functions, and system redundancy and testability, and the guidelines of Regulatory Guide 1.29, Position C.1, relating to the system's seismic classification and is, therefore, acceptable. The standby liquid control system meets the acceptance criteria of SRP Section 9.3.5.

9.4.1 Control Building and Normal Switchgear Building Heating, Ventilating, and Air Conditioning System

The control building (CB) and normal switchgear building (NSB) heating, ventilating and air conditioning (HVAC) systems were reviewed in accordance with Section 9.4.1 of the Standard Review Plan (SRP), NUREG-0800. An audit of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the CB and NSB systems with respect to the applicable regulations of 10 CFR 50.

The CB HVAC system includes the main control room (MCR) HVAC system, relay room (RR), remote shutdown room (RSR), standby switchgear rooms (SSR), computer room (CR), battery rooms (BR), record storage vault (RSV), stairwells (S), basement and electrical tunnels (ET) HVAC systems. (Refer to Section 6.4 of this SER for a discussion of control room habitability). The NSB system serves the non-essential normal switchgear building.

The MCR, RR, RSR, SSR, and BR systems are safety-related and consist of two 100% capacity Class 1E redundant trains. The basement HVAC system consists of safety-related cooler, filter and cooling coil, to which water is supplied from the service water system. The normal switchgear HVAC system is nonsafety related (nonseismic Catetory I) and consists of three 50% capacity trains which

are non-class 1E powered. For those systems with two full capacity trains, one train normally operates with the redundant train on standby. The standby train automatically starts upon failure of the normally operating train.

The control building HVAC systems are located in the control building which is seismic Category I, tornado, and flood protected. Therefore, we conclude that the requirements of General Design Criterion 2, "Design Basis for Protection Against Natural Phenomena," are satisfied. The CB system is seismic Category I except for the smoke removal equipment. The NSB system and CB system smoke removal equipment serve no safety function and their failure after a safe shutdown earthquake will not affect any safety-realted equipment. Thus, we concluded that CB and NB HVAC systems meet the guidelines of Regulatory Guide 1.29," Seismic Design Classification," Positions C.1 and C.2.

The CB HVAC system takes outside air from four 100% capacity, missile and tornado protected air intakes. The air intake for the relay room and standby switchgear/battery rooms are also tornado missile protected. The exhaust for MCR, RR, SS, BR, and ET systems are tornado missile protected. Refer to Section 3.5.2 of this SER for further discussion of tornado missile protection. Thus, we conclude that the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," are satisfied. Area radiation monitors are also provided in the main control room. In the event of increasing radioactivity levels, the outdoor supply air is automatically direrted through emergency air filter trains. Upon receipt of a high radiation alarm in the control room, the operator can isolate the control building outdoor air inlet dampers manually.

The MCR system which potentially handles radioactive material during accident conditions through the air intakes consists of two full capacity redundant emergency filter trains for HEPA and charcoal filters in series. Thus, the requirements of General Design Criteria 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guides 1.52, "Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Plants," Position C.2, and 1.140, "Design, Testing and Maintenance Criteria for Normal

Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Plants," Positions C.1 and C.2, are satisfied with respect to ensuring radiological environmental limits for personnel under normal and accident conditions, including LOCA conditions.

The control room area is normally maintanined at a slightly positive pressure relative to the outdoors by taking makeup air from either or four outside intakes located at different elevation of the control building. The control room HVAC system is designed to maintain the operability of the equipment in the control room. The air intakes have no chlorine monitoring capability but do have radiation monitoring capability. [However, the applicant has not provided sufficient information for us to evaluate the design capability of the CB HVAC system and its operating models. The applicant has not provided a response to our concerns in questions 410.41 and 42 regarding chlorine detection and assuring a proper environment during long term pressurization of the control room.] Thus, we cannot conclude that the requirements of General Design Criterion 19, "Control Room," and the guidelines of Regulatory Guides 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hàzardous Chemical Release," Positions C.3, C.7, and C.14 and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Positions C.4a and C.4d are satisfied.

[The applicant has not provided sufficient information for us to determine the acceptability of the measures provided for protection against accumulation of hydrogen in the battery rooms.]

[The applicant has not provided sufficient justification for the design basis maximum and minimum outdoor temperatures assumed for sizing of the CB HVAC system. The applicant should confirm that under postulated temperature extremes the HVAC systems serving safety-related equipment and essential occupied spaces can maintain a suitable environment to assure safety functions assuming an accident, loss of offsite power, and concurrent single failure in the HVAC system.]

Based on the above, we conclude that the control building and normal switchgear building HVAC systems are in conformance with the rquirements of the General

Design Criterion 2, 4, and 60 with respect to protection against natural phenomina, tornado missile protection, and radiological release the guidelines of Regulatory Guide 1.29 Positions C.1 and C.2, 1.52 Position C.2 and 1.140 Positions C.1 and C.2. We cannot conclude that the CB system is in conformance with the requirements of General Design Criterion 19 with respect to control room environmental conditions and the guidelines of Regulatory Guides 1.78 relating to the protection against hazardous chemical release Positions C.3, C.7, and C.14 and 1.95 Positions C.4a and C.4d with respect to protection of personnel against chlorine gas release. We will report resolution of our concerns in a supplement to this SER. The HVAC system does not meet the acceptance criteria of SRP Section 9.4.1.

9.4.2 Spent Fuel Pool Area Ventilation System

The spent fuel pool area ventilation system was reviewed in accordance with Section 9.4.2 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP. Conformance with the acceptance criteria formed the basis for our evaluation of the spent fuel pool area ventilation systems with respect to applicable regulations of 10 CFR Part 50.

The applicant has not provided sufficient information regarding the spent fuel pool area ventilation system for us to evaluate the system in accordance with SRP Section 9.4.2. Further, there has been no response to our question 410.43 regarding the need for information on the spent fuel pool area ventilation system. Thus, we cannot conclude that the spent fuel pool area ventilation system meets the acceptance criterion of SRP Section 9.4.2. We will report resolution of our concerns in a supplement to this SER.

9.4.3 Radwaste Building Ventilation System

The radwaste building ventilation system was reviewed in accordance with Section 9.4.3 of the Standard Review Plan (SRP) NUREG-0800. An audit review

of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the radwaste building ventilation system with respect to the applicable regulations of 10 CFR Part 50.

The radwaste building ventilation system is classified as nonsafety-related, nonseismic Category I. The ventilation system is capable of fulfilling the requirements of the facility for providing an environment with controlled temperature and air flow to ensure the integrity of the nonessential equipment and components served. Equipment and instrumentation have been provided with suitable redundancy to ensure normal operation and to prevent release of radioactivity to the environment and thus the system is acceptable for its designed task. Failure of the system does not compromise the operation of any essential systems and does not affect the capability to safely shut down the plant or result in an unacceptable release of radioactivity; thus, the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," and 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guides 1.29, "Seismic Design Classification," Position C.2, and 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Positions C.1 and C.2, are met.

Based on the above, we conclude that the radwaste building ventilation system is in conformance with the requirements of General Design Criteria 2 and 60 as related to protection against natural phenomena and radioactive release and the guidelines of Regulatory Guides 1.29, Position C.2, and 1.140, Positions C.1 and C.2, with respect to seismic classification and normal ventilation exhaust and air filtration and are, therefore, acceptable. The radwaste building ventilation system meets the acceptance criteria of SRP Section 9.4.3.

9.4.4 Turbine Building Ventilation System

The turbine building ventilation system was reviewed in accordance with Section 9.4.4 of the Standard Review Plan (SRP) NUREG-0800. An audit review

of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the turbine area ventilation system with respect to the applicable regulations of 10 CFR Part 50.

The turbine building ventilation system is classified as nonsafety-related, nonseismic Category I. The ventilation system is capable of adequately maintaining an acceptable environment for nonessential equipment served during normal plant operation. Failure of the system does not compromise the operation of any essential systems and does not affect the capability to safely shut down the plant, thus the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.2 are met. The turbine building ventilation system is not required to handle radiological releases and, therefore, the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," are not applicable.

Based on our review, we conclude that the turbine building ventilation system meets the requirements of General Design Criterion 2 with respect to the need for protection against natural phenomena and the guidelines of Regulatory Guide 1.29, Position C.2, concerning its seismic classification and is, therefore, acceptable. The turbine building ventilation system meets the acceptance criteria of SRP Section 9.4.4.

9.4.5 Engineered Safety Feature Ventilation System

The engineered safety feature ventilation systems were reviewed in accordance with Section 9.4.5 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance

criteria, except as noted, formed the basis for our evaluation of the engineered safety feature ventilation systems with respect to the applicable regulations of 10 CFR 50.

The engineered safety feature ventilation system provides cooling for equipment in the service water pump bays (SWPS), ECCS pump rooms, and the standby diesel-generator (SDG) building. The equipment in these areas is not required for control of releases of radioactive materials to the environment, and thus the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment" and the guidelines of Regulatory Guides 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," are not applicable.

(a) Service Water Pump Bay HVAC System

The service water pump bay is located in the screenwell building. Two 100-percent capacity safety-related unit cooler trains are provided for the service water pump bay. Each unit cooler train consists of a supply fan, filter, cooling oil, smoke exhaust fan and unit heaters, dampers and controls to assure a proper ambient environment under all operation modes and are tornado missile protected. (Refer to Section 3.5.2 of this SER for further discussion on tornado missile protection.) Nonseismic Category I electric unit heaters maintain space temperatures in winter to prevent freezing. Each train is powered from a Class IE Source, except the heaters. On loss of offsite power, the SWS pumps will continue to run an emergency power. The operation of these pumps produces sufficient heat to prevent freezing. The HVAC system operates automatically based on the ambient temperature in the area they serve. The above design assures system function in the event of a single failure. Therefore, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," with respect to monitoring a suitable environment for essential equipment are satisfied.

The system is housed in the safety-related portion of the screenwell building which is seismic Category I, flood and tornado protected and the system is designed to seismic Category I, Quality Group C criteria, thereby satisfying the requirement of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.1.

(b) <u>Diesel Generator Building HVAC System</u>

There are three diesel generators which are located in separate rooms within the diesel generator building. Each of the three diesel generator rooms consists of a 100 percent capacity exhaust fan, and associated duct work and motor-operated dampers. Each room is also equipped with a safety-related and seismically qualified unit cooler, and associated duct work, to maintain the room temperature below the maximum temperature for equipment operation. Each diesel generator room HVAC system is powered from its respective emergency bus and is automatically started when its respective diesel is started.

The system is designed to seismic Category I, Quality Group C requirements and is housed in the seismic Category I, flood and tornado protected diesel generator building, thereby satisfying the requirements of General Design Criterion 2 and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2. [The applicant has not provided information about tornado missile protection for the inlet and outlet louvers.] The system is separated from high energy piping systems and internally generated missiles. [We cannot conclude, however, that the requirements of General Design Criterion 4 are satisfied.] [Refer to Section 3.5.2 of this SER for discussion of tornado missile protection.] [The applicant has not specified the height of the inlet louvers about plant grade, thus we cannot conclude that the guidance of item 2, subsection A, of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability" and therefore, the pertinent requirements of General Design Criterion 17, "Electric Power System," relating to the protection of essential electrical components from failure due to the accumulation of dust and particulate material are satisified.]

(c) <u>ECCS Pump Room Cooling Systems</u>

Each ECCS pump cubicle contains two 100% capacity redundant seismic Catetory I, class 1E powered unit coolers. Cooling water to the coolers is supplied from redundant trains of the seismic Catetory I service water system (refer to Section 9.2.1 of this SER). Thus, the requirements of General Design Criterion 4 and the guidelines of the TMI Action Plan II.K.3.24 are satisfied with respect to maintaining a safe environment for equipment operation.

Based on the above, we conclude that the engineered safety feature ventilation system is in conformance with the requirements of General Design Criteria 2 and 4 as they relate to protection against natural phenomena and maintaining a suitable environment for equipment operation and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2 relating to the system's seismic classification and is, therefore, acceptable. [We cannot conclude that the requirements of General Design Criterion 17 and the guidelines of NUREG/CR-0660 relates to providing protection of the diesel generator from dust accumulation are met. We will report on this item in a supplement to this SER.] The engineered safety feature ventilation system does not meet the acceptance criteria of SRP Section 9.4.5.

10.3.1 Main Steam Supply System

The main steam supply system was reviewed in accordance with Section 10.3 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below, formed the basis for our evaluation of the main steam supply system with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria for the main steam supply system includes meeting Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles." Compliance with the guidelines of Regulatory Guide 1.115 are evaluated separately in Section 3.5.1.3.

The steam generated in the reactor vessel is routed to the high-pressure turbine by means of four main steam lines. Each main steam line contains two main steam isolation valves (MSIVs). The main steam isolation valves are designed to close against the maximum steam flow. One MSIV is located immediately inside of the drywell and the other immediately outside of containment. The MSIVs are air operated, fail-closed valves. Operating air is supplied to the valves from the instrument air system, and a seismic Category I air accumulator with bottled nitrogen which provides backup operating gas for each valve in the event of loss of the normal instrument air supply. The MSIVs are designed to withstand the dynamic forces under the postulated steamline break flow conditions.

The main steam supply lines including the outermost MSIVs are seismic Category I and are designed to quality group A criteria. The steam lines, turbine stop valves and shutoff valves inside the turbine building are seismically supported to return their pressure boundary in an SSE. The steam lines in the reactor building (including containment and steam tunnel) are located in seismic Category I, flood- and tornado-protected structures. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, are satisfied for these portions of the main steam supply system.

The MSIVs, which are required to function in order to assure main steam isolation, are protected against the effects of high-energy pipe breaks and are qualified to function in the expected steam environment resulting from a main steam line break. Refer to Sections 3.6.1 and 3.11 of this SER for further discussion on environmental qualification of essential equipment. This equipment is located in tornado missile protected structures and is separated from the effects of internally generated missiles. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.117, "Tornado Design Classification," Positions 2 and 4, are satisfied. Nine Mile Point 2 does not share the main steam supply system with unit 1, therefore, General Design Criterion 5, "Sharing of Structures, Components and Systems," is not applicable.

Based on the above, we conclude that the main steam supply system from the reactor to the turbine building meets the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena, floods, tornadoes, missiles and environmental effects, and the guidelines of Regulatory Guides 1.29, Positions C.1 and C.2, and 1.117, Appendix Positions 2 and 4, relating to the system's seismic classification and protection against tornado missiles and high- and moderate-energy pipe breaks and is, therefore, acceptable. The main steam supply system meets the acceptance criteria of SRP Section 10.3.

10.4.5 Circulating Water System

The circulating water system (CWS) was reviewed in accordance with Section 10.4.5 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the circulating water system with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related (nonseismic Category I, Quality Group D) circulating water system (CWS) is designed to remove the heat rejected from the main condenser to the atmosphere via a natural draft cooling tower. The CWS is normally used to shut down the reactor but is not required for safe shutdown following accident conditions.

The applicant has not provided the results of an analysis of the effects of possible flooding as a result of a postulated failure of a circulating water expansion joint or line failure as a result of an SSE. A failure of this system or any of its components may affect safety-related equipment in areas adjacent to the turbine building. The applicant should indicate the level reached by flood water from a CWS failure, and confirm in the analysis that when the postulated rupture occurres, the escaping water would not accumulate in the vicinity of the safety-related equipment. Also the applicant should verify that all doors and penetrations which interconnect to other structures

are watertight so that safety-related equipment would not be affected by the failure in the CWS. Credit for nonseismic Category I equipment can not be taken for mitigation of the above postulated concern. Thus, we cannot conclude that the requirements of General Design Criterion 4, "Environmental and Missile Design Basis," with respect to protection of safety-related systems from flooding as a result of failure of nonsafety-related systems are satisfied.

Based on our review, we cannot conclude that the circulating water system meets the requirements of General Design Criterion 4 with respect to protection of safety-related systems from failures in nonsafety-related systems. The circulating water system does not meet the acceptance criteria of SRP Section 10.4.5. We will report resolution of our concern in a supplement to this SER.

10.4.7 Condensate and Feedwater System

The condensate and feedwater system was reviewed in accordance with Section 10.4.7 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the condensate and feedwater system with respect to the applicable regulations of 10 CFR 50.

The condensate and feedwater system includes all components and equipment from the condenser outlet to the connection at the reactor vessel and to the heater drain system. The system serves no safety function and is therefore classified as nonsafety related (nonseismic Category I). However, the portion of the system between the reactor vessel and containment is safety related and designed to seismic Category I, Quality Group A criteria from the reactor to the outboard containment isolation valve, and seismic Category I, Quality Group B criteria from the outboard containment isolation valve to the feedwater shutoff valve in order to assure feedwater system isolation under accident conditions. Each main feedwater line contains a motor-operated check valve, a check valve held open by air pressure during normal operation as the outboard containment

isolation valve, an inboard isolation check valve, and a motor-operated shutoff valve. Thus, feedwater isolation is assured in the event of a single failure in any isolation valve. These isolation valves are powered by redundant 1E sources.

The safety-related portion of the system is located in the seismic Category I, flood- and tornado-protected reactor building. The main feedwater piping in the steam tunnel was analyzed for high energy pipe breaks and the resulting effects of pipe whip, jet impingement, and flooding. Pipe whip restraints have been provided to prevent pipe whip. The only equipment which could be adversely affected by jet impingement are the main steam isolation valve (MSIV) operators. Jet impingement barriers have been provided to protect the MSIV operators. No adverse effects have been identified due to flooding. Therefore, a break in the main feedwater pipe will have no adverse effects on safety-related equipment or prevent safe shutdown of the reactor. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Positions C.1 and C.2, are satisified. The essential equipment is separated from the effects of internally generated missiles and is qualified to function in a steam line break environment. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," are satisified. Refer to Sections 3.6.1 and 3.11 of this SER for further discussion of environmental qualification of essential equipment and protection against postulated piping failures.

The feedwater system is not shared between units nor is it required to transfer heat under accident conditions and, therefore, the requirements of General Design Criteria 5, "Sharing Structures Systems and Components," 44, "Cooling Water," 45, "Inspection of Cooling Water Systems," and 46, "Testing of Cooling Water Systems," are not applicable.

Based on the above, we conclude that the safety-related portions of the condensate and feedwater system meet the requirements of General Design Criteria 2 and 4 with respect to its protection against natural phenomena,

missiles and environmental effects, and meets the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, with respect to its seismic classification and is, therefore, acceptable. The condensate and feedwater system meets the acceptance criteria of SRP Section 10.4.7.

missiles and environmental effects, and meets the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, with respect to its seismic classification and is, therefore, acceptable. The condensate and feedwater system meets the acceptance criteria of SRP Section 10.4.7.