1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

The Final Safety Analysis Report was submitted in support of the application of Metropolitan Edison Company, Jersey Central Power & Light Company, and Pennsylvania Electric Company for a license to operate a nuclear generating station designated as Three Mile Island Nuclear Station Unit 1 (TMI-1). After approval by the State and Federal regulatory agencies, Metropolitan Edison, Jersey Central, and Pennsylvania Electric owned the Three Mile Island Nuclear Station Unit 1 on a 50/25/25 percent basis, respectively. Metropolitan Edison Company was responsible for the safe operation of the station. After the Three Mile Island Unit 2 (TMI-2) accident, reorganization included the formation of the GPU Nuclear Corporation. With the post TMI-2 accident reorganization, Metropolitan Edison delegated to GPU Nuclear Corporation the responsibility for all aspects of operations, design, procurement and modifications of TMI-1. Aid in the design, construction, testing, and startup of the unit was supplied principally by Gilbert Associates, Inc.; United Engineers & Constructors, Inc.; and the Babcock & Wilcox Company. Assistance was rendered by other consultants and suppliers as required. See Organization Chart, Figure 1.1-51. The current organization is discussed in Chapter 12.

In 1999, as part of the restructuring of the electric utility industry in the United States, TMI-1 was sold to AmerGen Energy Company, LLC. AmerGen is a limited liability company formed to acquire and operate nuclear power plants in the United States.

The Updated Final Safety Analysis Report is submitted in support of the Exelon license to own and operate the nuclear generating station designated as Three Mile Island Nuclear Station Unit 1. The Unit is located on the northern most section of Three Mile Island near the east shore of the Susquehanna River in Dauphin County, Pennsylvania.

The Three Mile Island Nuclear Station Unit I was initially licensed to operate at a rated power level of 2535 MWt. When the 16 MWt contribution from the reactor coolant pumps is included, the corresponding gross electrical output is 871 MWe. Site parameters, principal structures, engineered safeguards, and certain hypothetical accidents were evaluated at a core power level of 2535 MWt. Most of the design basis analyses including core thermal hydraulics, fuel assembly design, reactor coolant system design, and certain hypothetical accident analyses were performed using a design core power level of 2568 MWt or greater (see Chapters 1, 3, 4 and 14).

License Amendment No. 143 authorized a 1.3 percent increase in the licensed rated power level to 2568 MWt. The basis for the power stretch verified that the UFSAR analyses that had been performed at 2535 MWt, as noted above, remained conservative at the new rated power level.

The nuclear steam supply system is a pressurized water reactor type which is similar to many other Pressurized Water Reactors (PWRs) operating or under construction. It uses chemical shim and control rods for reactivity control and generates steam with a small amount of superheat in Once Through Steam Generators. The original nuclear steam supply system and the fuel for the first core and reloads were supplied by the Babcock & Wilcox Company. The replacement Once Through Steam Generators are supplied by AREVA NP Inc.

The general arrangement of major equipment and structures, including the Reactor, Auxiliary, and Turbine Buildings, is shown on Drawings 1E-120-01-001, 1E-151-02-001 through

1E-151-02-016, 1E-153-02-001 through 1E-153-02-009, 1E-154-02-001 through 1E-154-02-009, 1E-155-02-001 through 1E-155-02-005, 1E-156-02-001 through 1E-156-02-005, 1E-157-02-001 through 1E-157-02-003, 1E-168-02-001 and 1E-168-02-002.

1.2 DESIGN SUMMARY

1.2.1 SITE CHARACTERISTICS

The site is located on the Susquehanna River about 10 miles southeast of Harrisburg, Pennsylvania. It is characterized by a 2,000 foot minimum exclusion area boundary distance; a two mile radius low population zone; sound bedrock as a structural foundation; an ample supply of emergency power and favorable conditions of hydrology, geology, seismology, and meteorology. The land within a 10 mile radius of the site is used primarily for farming.

There are two airports within ten miles of the site. Harrisburg International Airport (formerly Olmsted State Airport) is located approximately 2.5 miles northwest of the site, and Capital City Airport (formerly Harrisburg-York State Airport) is located approximately eight miles west-northwest of the site.

1.2.2 POWER LEVEL

Initially the licensed power for the reactor core was 2535 MWt, and core performance analyses in this report were based on a design power level of 2568 MWt. An additional approximate 16 MWt are available to the core from the contribution of the reactor coolant pumps. The analyses in Chapter 14 of most accidents have been performed at a core power level of 2568 MWt; and some small break LOCA accidents have been analyzed at 2772 MWt power level. However, for purposes of dose calculation and containment evaluation a core power level of 2535 MWt was used. The impact of increased core power levels of 2568 MWt (License Amendment No. 143) on dose calculation, containment evaluation, are described in the appropriate UFSAR sections. The impact of increased core power levels of 2568 MWt on accident analyses are further described in Chapter 14.

1.2.3 LINEAR HEAT RATE

Operation at 2568 MWt results in a nominal linear heat rate of about 5.8 kw per foot of active fuel length (See Table 3.2-11). This value was comparable with other reactors of this size then constructed, and with reactors in the 400-500 MW class such as San Onofre, Ginna, and Connecticut Yankee, and therefore did not represent an extrapolation of technology.

1.2.4 REACTOR BUILDING SYSTEM

The Reactor Building is a reinforced concrete structure composed of cylindrical walls with a flat foundation mat, bearing on sound rock, and a shallow dome roof. The foundation slab is reinforced with conventional mild steel reinforcing.

The cylindrical walls are prestressed with a post-tensioning tendon system in the vertical and horizontal directions. The dome roof is prestressed utilizing a three way post tensioning tendon system. The inside surface of the reactor building is lined with a carbon steel liner to ensure a high degree of leak tightness for containment.

The Reactor Building is similar in design to the containment buildings for the Turkey Point Plant (Docket Nos. 50-250 and 251), the Palisades Plant (Docket No. 50-225), the Point Beach Plant (Docket No. 50-266), the Oconee Nuclear Station (Docket Nos. 50-269, 50-270, and 50-287), and the Crystal River Plant Unit 3 (Docket No. 50-302).

1.2.5 ENGINEERED SAFEGUARDS

The Engineered Safeguards provided for Three Mile Island Nuclear Station Unit 1 have sufficient redundancy of component and power sources such that under the conditions of the worst postulated loss of coolant accident, the system can maintain the integrity of the containment and keep the exposure of the public below the limits of 10CFR100.

The Engineered Safeguards provided for this plant are the following:

a. High pressure injection system prevents uncovering of the core for small coolant piping leaks at high pressure and delays uncovering of the core for intermediate sized leaks. This system is normally operated as part of the makeup and purification system.

b. The core flooding system automatically floods the core when the reactor coolant system pressure reaches a level of approximately 600 psig.

c. The low pressure injection system provides core cooling after the reactor coolant pressure has reached about 100 psig following a loss of coolant accident. This system normally operates as part of the decay heat removal system during shutdowns.

d. The Reactor Building spray system provides a spray of borated water from the BWST. Trisodium phosphate (TSP) contained in baskets on elevation 281'0" of the Reactor Building mixes with the Building Spray solution and the leaking water from the RCS to provide RB Sump pH control and iodine removal for the containment atmosphere at the onset of the recirculation phase. Building Spray also provides a redundant system for cooling of the reactor building atmosphere.

e. The Reactor Building cooling system provides a heat sink to cool the building atmosphere under the conditions of a loss of coolant accident. This system also provides normal building cooling and ventilating requirements.

f. The Reactor Building isolation system provides automatic isolation of all Reactor Building penetrations not required for limiting the consequences of an accident.

g. Deleted

1.2.6 ELECTRICAL SYSTEM AND EMERGENCY POWER

The Three Mile Island Nuclear Station Unit 1 has the following sources of electric power:

a. Three transmission lines from the 230-kV grid system terminating at the station from two different directions and one transmission line from the 500-KV system.

b. Unit 1 generator, may continue to supply the auxiliary loads after a trip that separates the station substation from the transmission system.

c. Two quick starting 3000-kW diesel generator units connected to the engineered safeguards buses.

d. An alternative AC (AAC) power source is utilized at TMI-1. The AAC meets the criteria specified in Appendix B to NUMARC 87-00.

This AAC capability is provided by the Station Black Out (SBO) diesel generator (what once was one of the TMI-2 Emergency Diesel Generators). See Sections 8.5.1 and 8.5.2 for details.

Within the station there are multiple redundant buses and ties supplying power to loads, instruments, and controls. The engineered safeguards are supplied from two separate safeguards power buses, each of which can be supplied from any of the principal sources of power.

The sources of power and associated electric equipment ensure safe functioning of the station and its engineered safeguards.

1.2.7 HYPOTHETICAL AIRCRAFT INCIDENT SUMMARY

Vital areas of the Three Mile Island Nuclear Station Unit 1 were designed to withstand a hypothetical aircraft incident as described in Chapters 5 and 9; Section 2.4 provides an analysis of the airport-to-site relationship as well as a probability study of an aircraft striking the station.

The following presents a summary description of the probability of an aircraft striking the Three Mile Island Nuclear Station Unit 1 and the capability of vital areas of the station to safely withstand a hypothetical aircraft incident.

1.2.7.1 Aircraft Strike Probability

The respective locations of the station and the airport and its runway are described in Section 2.4.1 of the FSAR. The probability of strikes to the station has been studied using as a basis ten years of records in the annual statistical summaries of U.S. air carrier accidents and individual aircraft accident reports available from the Bureau of Safety of the Civil Aeronautics Board. The results of this evaluation show that the probability of an aircraft strike on the unit is very low (see Section 2.4.2).

1.2.7.2 Capability Of Structures To Withstand An Aircraft Strike

(b) (5) [redacted]

Items (1) and (2) above were analyzed utilizing a dynamic elastic analysis developed by Franklin Institute Research Laboratory. This analysis is based upon time-dependent forces acting on the structure. These time-dependent forces are described in Appendix 5A. The results indicate that the dome of the Reactor Building will not collapse as a result of such loadings.

Items (3) and (4) above were analyzed for the vital structures of Three Mile Island Nuclear Station Unit 1 as listed in Section 5.1.3. The analysis included the use of the displacement bound theorem for a rigid-plastic continuum. This analysis is based upon an inelastic collision.

These analyses produce conservative results which support the conclusion that the stability of the structures is not jeopardized.

In addition, the Reactor Building was analyzed to verify that local penetration due to the prescribed loads would not occur.

The load-time curve for an aircraft strike is derived from the angle of impact, velocity, mass distribution, contact area, and structural characteristics of the aircraft. Therefore, different aircraft will have a different time curve depending upon the above variables. It can be concluded that with a favorable load-time curve, it is possible that the Reactor Building could withstand the impact of an aircraft larger than that described above.

1.2.7.3 Fire Protection

In order that the station can be maintained in a safe condition following a fire which might result from a hypothetical aircraft incident, design provisions were made to assure the protection of personnel and equipment in vital structures. These provisions include the following:

a. An air intake tunnel, which brings air into the Auxiliary, Control, and Fuel Handling Buildings, with a remote intake located 125 feet from the plant.

b. Exhaust openings from ventilation systems, such as the control tower complex, are provided with protective shields designed to withstand hypothetical aircraft incident.

c. Vapor detectors are located in critical areas of the ventilation system to detect vapors of liquid fuel which have not ignited. Fire detectors are also installed in the ventilation system in the event that burning fuel should enter the intake structure. Activation of any of these detectors will cause the ventilating fan to stop, fire dampers to close, and halon 1301 and/or deluge water spray systems to operate.

d. Piping which passes through the top protected level of the Auxiliary Building will be protected with a loop seal approximately six feet deep to prevent the passage of fuel in the event of a ruptured aircraft fuel line.

1.2.8 SHARED COMPONENTS WITH UNIT 2

Three Mile Island Nuclear Station Unit 1 will make use of some facilities on a shared basis with Unit 2. A partial list of the shared components include:

a. Fire Protection System

b. Fuel Handling BuildingCrane

c. Industrial Waste Treatment Facility

d. Sewage Treatment Plant

e. Liquid and Solid Radwaste Processing

None of the shared components are connected with safety features or control systems of either nuclear steam supply system.

1.3 DESIGN CHARACTERISTICS

1.3.1 DESIGN CHARACTERISTICS

The important as-built design and operating characteristics of the nuclear steam supply system for TMI-1 are summarized in Table 1.3-1. For details of the latest parameters, see Chapters 3 and 4.

1.3.2 SIGNIFICANT DESIGN REVISIONS

The more significant design revisions made to the unit are listed below.

1.3.2.1 Fuel Assembly

The fuel assembly structure consists of Zircaloy interior control rod guide tubes fastened to stainless steel upper and lower end fittings. The assembly also utilizes an Inconel or Zircaloy top spacer grid, Inconel bottom spacer grid, and Zircaloy intermediate spacer grids. The spacer grids are supported either by a grid restraint system around the center instrument tube, or by welding to the control rod guide tubes. All fuel rods are internally pre-pressurized with helium to minimize clad fatigue due to power and pressure cycling.

1.3.2.2 Axial Power Shaping Rod Assemblies

In addition to the 61 full-length control rods, eight control rods contained a neutron absorber for a portion of their length to aid in controlling axial imbalance. Starting with Cycle 6, gray APSRAs, which are longer and use a weaker absorber (Inconel), replaced the silver-indium-cadmium APSRAs used in all previous cycles. In Cycle 19, the APSRAs were determined to be unnecessary and were removed from service.

Axial power distribution control during power operation with the APSRAs removed is accomplished by adjusting regulating rod group position, as required, to prevent or damp xenon oscillations.

1.3.2.3 Burnable Poison Rod Assembly

Burnable poison rod assemblies are used to reduce power peaking in fresh fuel as well as to reduce the magnitude of the beginning of life positive moderator temperature coefficient. As of Cycle 10, gadolinia poison integral to the UO_2 pellets in selected fuel rods also is used.

1.3.2.4 Control Rod Drives

The unit will utilize sealed roller nut and leadscrew type control rod drives rather than shaft seal rack and pinion drives. Beginning with Cycle 4, the reactor is operated in a rods-out, feed and bleed mode. The core reactivity control is supplied mainly by soluble boron and supplemented by 61 full length Ag-In-Cd control rod assemblies.

1.3.2.5 InCore Instrumentation Readout

Auxiliary readout of selected incore detectors is recorded in the Control Room. The total number of neutron detectors monitored is 36. In addition there are 50 of 52 incore thermocouples to provide for detection of inadequate core cooling (the remaining 2 thermocouples are used for input to RCITS).

50 of 52 thermocouples are monitored by the plant computer. Sixteen of these thermocouples, four from each core quadrant, can be monitored in the main Control Room independently of the plant computer and its power sources. These thermocouples constitute the Backup Incore Thermocouple Readout (BIRO) System.

1.3.2.6 Control Rod Trip Signal

The intermediate range rate trip has been deleted.

1.3.2.7 Nuclear Service Cooling Water Piping

The piping from the shell side of the nuclear services coolers enters the Auxiliary Building via an underground tunnel which is designed for the hypothetical aircraft incident. This is a change from the preliminary design where redundant lines were contemplated. The addition of the tunnel for personnel access to the vault made available a protected pipe chase, and a single pipeline is therefore utilized.

1.3.2.8 Power-Operated RCS High Point Vents

Addition of power-operated RCS high point vents are provided for the loops, Presssurizer and Reactor Vessel Head for removing noncondensable gases following an accident.

1.3.2.9 PORV And Safety Valve Instrumentation

Addition of delta-pressure and accelerometer for the pressurizer power operated (electromatic) relief valve has been provided for supplying the Control Room with information on the status of the PORV and safety valves.

1.3.2.10 Postaccident Sampling

Note: Technical Specifications Amendment #253 eliminated the requirements to maintain a Post Accident Sampling System. The Post Accident Sampling System will be maintained for contingency actions and long term post accident recovery operations. The specific parameters that must be maintained as commitments to Tech Spec Amendment #253 are as follows:

1. Containment hydrogen monitors are maintained to support SAMGs per NRC SER for BAW-2387, Item 4.1.5, dated November 14, 2002, and TS amendment #253 for PASS elimination.
2. TMI-1 chemistry procedures provide contingency plans for obtaining and analyzing containment atmosphere hydrogen after plant conditions have stabilized post-accident (long term application) to support NRC SER for BAW-2387, Item 3.9, dated November 14, 2002, and TS amendment #253 for PASS elimination.
3. TMI-1 chemistry procedures provide contingency plans for obtaining and analyzing highly radioactive samples of containment atmosphere after plant conditions have stabilized to support NRC SER for BAW-2387, Item 3.11, dated November 14, 2002, and the TS amendment #253 for PASS elimination.
4. TMI-1 chemistry procedures provide contingency plans for obtaining and analyzing highly radioactive samples of containment sump after plant conditions have stabilized to

support NRC SER for BAW-2387, Item 3.12, dated November 14, 2002, and the TS amendment #253 for PASS elimination.

5. Capability for monitoring and assessing iodines released to offsite environs through onsite and offsite surveys is maintained within the TMI-1 Emergency Plan and Emergency Plan Implementing Procedures to support the NRC SER for BAW-2387, Licensee Required Action 4.1.4, dated November 14, 2002, and the TS amendment #253 for PASS elimination.

Addition of postaccident sampling capabilities for analysis of reactor coolant samples and containment atmosphere.

1.3.2.11 Engineered Safeguards System

The high pressure injection and loading sequence will be initiated by Reactor Building pressure greater than 4 psig or reactor coolant pressure less than 1600 psig and also by reactor coolant pressure less than 500 psig when in shutdown bypass mode.

Protective action is initiated by de-energizing output relays (except Reactor Building spray).

The ES signal duplicates manual start, close, or open action. No ES signal is used to block manual operations. There are no ES components that receive an automatic signal calling for an action inverse to the desired action of the ES actuation signal.

The majority of air-operated valves do not require power for ES actuation.

Bistables have been replaced by pressure switches for Reactor Building pressure signal greater than 30 psig.

Bistables have been added to allow bypassing of core injection signals before automatic actuation for normal cooldown and startup modes.

The Reactor Building spray valves are actuated before Reactor Building pressure exceeds 30 psig.

1.3.2.12 (Not Used)

1.3.2.13 Containment Isolation System

The containment isolation system has been redesigned to include the new features of containment isolation on reactor trip, containment isolation on 30 psig building pressure, specific line isolation on high radiation, 1600 psig reactor coolant pressure, and removal of the 4 psig containment isolation signal from RCP essential services.

1.3.2.14 High Pressure Injection Cross-Connect And Cavitating Venturis

The high pressure injection system was changed by cross-connecting the "A" and "C" High Pressure Injection (HPI) legs and the "B" and "D" HPI legs to improve the ability of TMI-1 to withstand the consequences of a small break LOCA.

Cavitating venturis were added to the high pressure injection lines to limit flow to less than run-out flow of a single high pressure injection pump and therefore eliminate the need for operator action on a HPI line break.

1.3.2.15 Emergency Core Cooling System Actuation

The Emergency Core Cooling System is actuated on Reactor Building pressure as well as low Reactor Coolant System pressure.

1.3.2.16 Axial Flux Peaking Protection

A power trip based on imbalance and flow functions has been incorporated, which protects the core in the event of excessive axial flux peaking by tripping the reactor before thermal limits are exceeded.

1.3.2.17 Reactor Building Pressure Trip

The reactor will trip when Reactor Building pressure exceeds a fixed maximum limit.

1.3.2.18 Deleted

1.3.2.19 Reactor Building Spray System

The RBSS has had two significant design revisions.

First, the system was modified to accommodate the replacement of Sodium Thiosulfate with Sodium Hydroxide as the chemical buffer used for iodine scrubbing of the post-LOCA RB atmosphere and establishment and long-term control of the Reactor Building sump pH.

Second, the system was modified again to establish Trisodium phosphate (TSP) as the chemical buffer for iodine scrubbing of the post-LOCA RB atmosphere and control of the Reactor Building sump pH at the onset of recirculation.

1.3.2.20 Emergency Feedwater

The EFW system takes suction from two condensate storage tanks. Alternatively, the EFW system can take suction from the hotwell or the emergency RB cooling water system.

1.3.2.21 Emergency Feedwater System

Modification of the emergency feedwater system to include the following provisions has been provided: a) For both of the motor-driven emergency feedwater (EFW) pumps to automatically start upon loss of both main feedwater pumps or loss of four reactor coolant pumps; b) For all three EFW pumps to automatically start upon: low steam generator water level or high Reactor Building pressure; c) Limit the mass and energy release to the Reactor Building through a steam line break using cavitating venturis in each line to the Steam Generators; d) Blocking the pump minimum recirculation valves open; e) Redundant flow control valves in each line to the Stem Generators; f) Automatic loading of motor-driven EFW pumps on the diesel generator during loss of offsite power; g) Indication in the Control Room of EFW flow to each steam generator; h) Manual control capability to the Control Room of the EFW flow to each steam generator independent of the Integrated Control System (ICS); i) Control Room annunciation for all auto start conditions of the EFW system; j) Removal of wall in alligator pit to provide flooding capacity (25 min).

1.3.2.22 Emergency Feed Pump Cooling Water

The cooling water services for the emergency feed pumps and turbine have been removed from the secondary services cooling system and are self-cooled.

1.3.2.23 Condenser Off-Gas Vent

A change reflected in the final design as compared with the preliminary design is the discharge of condenser air off-gas through its own vent pipe, rather than through the physically far removed vent. The discharge is monitored and will alarm on high radiation.

1.3.2.24 Electrical

The auxiliary transformers have been replaced with new transformers rated for 39/52/65 MVA equipped with Automatic Tap Changers on the 4kV transformer windings.

The diesel generator loading sequence has been changed.

The intake screen and pumphouse feeders have been relocated to 4160-V buses 1D and 1E from buses 1A and 1C so that they can be energized by the engineered safeguards diesel generators since the screenhouse buses now serve some engineered safeguards equipment.

1.3.2.25 Electrical Loads

Provisions have been made for the transfer of pressurizer heater loads from normal to backup power suppliers through the use of a kirk key system.

1.3.2.26 Postaccident Monitoring

Postaccident monitoring capability with inclusion of continuous containment pressure indication, containment water level indication, containment hydrogen indication, high range containment radiation monitor is provided. In addition, high range effluent monitors for each of gas release point, i.e., containment purge exhaust, "A" and "B" OSTG 12" steam lines (atmospheric dump valves, bypass valves, and EF-P-1), Auxiliary and Fuel Handling Buildings exhaust, condenser off-gas exhaust, is also provided.

Additional postaccident monitoring instrumentation provisions, satisfying the requirements of Regulatory Guide 1.97, Revision 3, are described in FSAR Section 7.3.2.2.

1.3.2.27 Inadequate Core Cooling Instrumentation

Additional and modified instrumentation for detection of inadequate core cooling includes connecting incore thermocouples to the plant computer, installing a diverse and separate system to the plant computer for monitoring incore coolant temperature, providing a wide range reactor outlet temperature measurement, and redundant Control Room indication of reactor coolant saturation margin, and installation of a Reactor Coolant Inventory Trending System (RCITS) to monitor coolant inventory in the reactor vessel head and hot legs.

1.3.2.28 Seismic Classification

Changes have been made in the list of Class I structures, components, and systems as to seismic classification. Additional areas of the Auxiliary Building are now Class I and portions of the makeup and purification system are now Class II. For additional details, refer to FSAR Sections 5.1.1.1 and 5.1.1.2.

1.3.2.29 Tendon Ducts

The tendon ducts are now galvanized conduit.

1.3.2.30 Liner Specifications

The exterior surface of the liner will not be painted. Penetrations will be soap bubble tested instead of "sniff-tested." Electrical penetrations will be continuously pressurized with nitrogen instead of air for leak monitoring purposes.

1.3.2.31 Piping

Originally, non-nuclear power piping was designed, fabricated, tested and inspected in accordance with USAS B31.1.0 - 1967. Originally, nuclear piping was designed in accordance with USAS B31.1.0 - 1967, but it was fabricated, tested and inspected to USAS B31.7 - February 1968 Draft, including June 1968 Errata.

USAS B31.7, February 1968 Draft, including June 1968 Errata (which replaced and improved on the USAS B31.1.0 Nuclear Code Cases), was issued after receipt of the construction permit. USAS B31.1.0 - 1967 was enhanced by specifying fabrication, testing and inspection to USAS B31.7 - February 1968 Draft, including June 1968 Errata for pipe classification N_1, N_2 and N_3.

"Nuclear" Piping is piping that normally contains a radioactive substance.

Nuclear valves were specified to be tested and inspected to USAS B31.7 - February 1968 Draft, including June 1968 Errata.

1.3.2.32 Radiation Monitoring System

The range of effluent radiation monitoring system monitors RM-A2P, RM-A8, RM-A9, RM-A5 was increased and additional monitoring capability with radiation monitors RM-G24, RM-G25, RM-G26 and RM-G27 was provided.

These radiation monitors provide extended ranges and readout capability for post-accident monitoring for the Auxiliary Building, Fuel Handling Building, Reactor Building ventilation exhausts, discharges from the condenser vacuum pumps, and "A" and "B" OTSG 12" steam lines (atmospheric dump valves, bypass valves, and EF-P-1)."

1.3.2.33 Liquid Waste Releases

The minimum average annual effluent flow rate from the mechanical draft cooling tower basin was increased from 2000 gpm to 5000 gpm.

1.3.2.34 Liquid Radioactive Waste Processing

The Liquid Radioactive Waste Processing System (LRWPS) has the following modifications:

a. The (LRWPS) piping interconnections between Unit 1 and Unit 2 have been reestablished such that Unit 1 miscellaneous liquid waste can be stored in the Unit 2 Miscellaneous Waste Holdup Tank or the Auxiliary Building Sump Tank and Unit 2 miscellaneous waste can be processed in Unit 1.

b. Ground water sump discharges only to the miscellaneous waste tank.

c. Modification allows treatment of laundry waste by the miscellaneous waste evaporator when the activity exceeds 1.0×10^{-7} microcurie per cubic centimeter.

d. Connections have been established to enable Unit 1 miscellaneous liquid waste to be processed in the Chemical Cleaning Building.

1.3.2.35 Waste Solidification System

The urea formaldehyde solidification system has been replaced by a Hittman Nuclear mobile cement solidification system. The system utilizes cement to immobilize plant wastes in either individual waste packages or in a truck mounted cask. All interface with Unit 2 has been eliminated.

1.3.2.36 Waste Storage

Waste storage for the unit is divided into two areas to provide covered storage. High activity spent resin waste will be stored in a storage cell within the Solid Waste Staging Facility (SWSF). Low activity solid waste will be stored in the Interim Solid Waste Staging Facility (ISWSF) prior to shipment offsite to a licensed burial facility or volume reduction vendor prior to disposal.

1.3.2.37 Technical Support Center

The Technical Support Center is located in an area of the 1st floor of the Operational Support Facility that was specifically designed to support this function. It contains equipment capable of providing displays of vital plant parameters. This facility provides engineering support for emergency operations.

1.3.2.38 Emergency Operations Facility

The Emergency Operations Facility is located in Coatesville, PA. The Emergency Operations Facility serves as the primary location for management of the corporation's overall emergency response. This facility is equipped for and staffed by the Emergency Support Organization to coordinate emergency response with offsite support agencies and assessment of the environmental impact of the emergency.

1.3.2.39 Steam Generator and Hot Leg Replacement

The once through steam generators and hot legs were replaced at the end of Cycle 17. The tubes for the new steam generators are fabricated from thermally treated Ni-Cr-Fe Alloy 690 which exhibits superior resistance to PWSCC. AREVA NP Inc. supplied the replacement

OTSGs and was responsible for ensuring the quality of the equipment within its scope of supply. (References: 1.9.17 and 1.9.18)

1.3.2.40 Oil Spill Control

A supplement to existing structures and facilities that prevent oil spills from reaching the Susquehanna River has been provided. The provision includes an oil containment curbing around the diesel- driven fire pump fuel oil tanks of TMI-1 and 2, the grading of oil tank truck delivery areas, and modification of the southeast dam at the discharge point for the site runoff to the river.

1.3.2.41 Fuel Handling Building Environmental Barrier

Isolation modifications are provided to separate TMI-1 and TMI-2 Fuel Handling and Auxiliary Building, and Supply and Exhaust Ventilation System. Also to prevent potential leakage paths between buildings or systems, modifications were provided to isolate the Unit 1 refueling floor from the Unit 1 Auxiliary Building and from the Control Building.

Modifications to the ventilation system include: 1) addition of a leak tight damper in the discharge of the FHB supply fan, 2) blanking off the supply duct and a branch duct to the FHB general area, and 3) providing an equivalent opening in the FHB supply duct to discharge air to the south side of the elevator shaft that is exhausted at the refueling floor. Other modifications include blanking off a branch duct in the spent fuel pool cooler area with the required air supplied from the Auxiliary Building through a wall opening at elevation 305 ft 0 inch. Also a leaktight damper is provided in the exhaust duct main as it leaves the FHB upstream of the connection with Auxiliary Building Main with an addition of leak tight dampers added to the FHB supply and exhaust ducts.

Physical barriers include: a) enclosure of a personnel passage between Unit 1 Control Building and Auxiliary Building at elevation 305 ft 0 inch with two main pressure resistant doors and one pair of pressure resistant equipment doors, b) addition of a removable wall at the east end of the truck bay at elevation 305 ft 0 inch and a security fence at the west end of the dock adjacent to a new enclosure at elevation 305 ft 0 inch. Also there is modification of the stair tower between elevations 299 ft 2-1/4 inches and 311 ft 0 inch and the following: a) addition of pressure resistant doors for the new fuel storage room at elevation 329 ft 0 inch, b) addition of a door for the stair tower at elevation 331 ft 0 inch, and c) addition of an enclosure with door at the elevator entrance at elevation 348 ft 0 inch.

1.3.2.42 Changes In Component Location Due To Hypothetical Aircraft Incident

With the emphasis applied to the hypothetical aircraft incident and the subsequent "hardening" of various areas of the plant, certain components had to be located in the conventional construction of the turbine hall. Components so located are main steam leads, main feed pumps and piping, main feed pump turbines and piping, and steam bypass (dump) valves located at the main condenser. The emergency feed pumps, the main steam leads up to the isolation stop check the valves, the steam header supplying the turbine-driven emergency pump, the suction valving to the emergency pumps, and the discharge piping are in the Intermediate Building or the adjacent Reactor Building, both of which were designed for the hypothetical aircraft incident. The emergency feed pumps discharge only to the emergency nozzles on the steam generators and not to the main feedwater system, and the steam supply to the main feedwater pump turbines has been removed from the emergency steam header and is now supplied downstream of the main steam isolation valves, adjacent to the turbine generator steam chest.

1.3.2.43 Pressurizer Heaters

One bank of pressurizer heaters (126kw) were connected to be fed from each diesel. Each pressurizer heater bank can be manually connected through key interlocked breakers to its dedicated diesel as required and only when diesel capacity is available and loads are stabilized. An under voltage trip is also provided.

The purpose of the pressurizer heaters is to provide an alternate method of maintaining the pressure in the pressurizer during an accident.

1.3.2.44 Pressurizer Code Safety Valve Inlet Piping

Testing performed by EPRI to qualify safety valves per NUREG-0737 Item II.D.1 indicated the safety valves installed on long inlet piping exhibited unstable performance during anticipated operating conditions. The testing also indicated that the safety valves exhibited stable performance when installed on a short inlet pipe and with suitable ring settings. TMI-1 removed the existing loop seal configuration and installed the valves on nozzles on the pressurizer. The ring settings were appropriately adjusted based on testing and analysis.

1.3.2.45 Anticipatory Reactor Trip

In order to reduce challenges to the protection systems the reactor will receive trip signals on loss of the main steam turbine or both steam driven main feedwater pumps.

1.3.2.46 Pressure Locking or Thermal Binding of Safety Related Gate Valves

Systems operation and valve design have been reviewed and modified as required to ensure that pressure locking or thermal binding as described in GL 95-07 will not effect the safety function of engineered safeguards remote operated gate valves. DH-V-4A/B and DH-V-1 & 2 have pressure relief paths to the system to eliminate pressure locking.

1.3.2.47 CRD Control System

The CRD Control System was replaced with a digital control system that does not include controls or position indication for Group 8 (APSRs) since APSRs are no longer utilized in core design. The Reactor Trip Breaker configuration was changed to accommodate the new control system.

1.3.2.48 FLEX Storage Facility

The TMI Unit 2 River Water Pump House (RWPH) was modified to provide a robust structure for storage of equipment used in the mitigation of Beyond-Design-Basis External Events (FLEX).

TABLE 1.3-1
(Sheet 1 of 7)

DESIGN PARAMETERS - THREE MILE ISLAND UNIT 1

The following design parameters are similar to those of Oconee Nuclear Station. Oconee parameters that are not similar are enclosed ( ) preceded by an *.

1. Hydraulic and Thermal Design Parameters

Reference design rated heat output (core), MWt	2,568
Reference design rated heat output (core), Btu/h	$8,765 \times 10^6$
Design overpower, %	112
System pressure (nominal), psia	2,200
System pressure (minimum steady state), psia	2,135
Power Distribution Factors	
Heating Generated in Fuel and Cladding, %	97.3
$F_{delta-h}$ (nuclear)	1.800
Fq (nuclear)	2.970
Hot Channel Factors	
Fq (nuclear and mech.)	See Table 3.2-11
DNB ratio at rated conditions	See Table 3.2-11
Minimum DNB ratio at design overpower	See Table 3.2-11

2. Core Mechanical Design Parameters

Fuel Assemblies	
Number	177
Design	CRA canless
Rod pitch, inches	0.568

TABLE 1.3-1
(Sheet 2 of 7)
DESIGN PARAMETERS - THREE MILE ISLAND UNIT 1

Overall dimensions, inches	8.536 x 8.536
Total weight, lb/assembly	1,561
Number of spacer grids per assembly	8
Fuel Rods	
Number	36,816
Outside diameter, inches	0.430
Clad thickness, inches	0.025
Clad material	M5
Control Rod Assemblies (CRA)	
Neutron absorber	5% Cd-15% In-80% Ag
Number of assemblies	61
Number of control rods per assembly	16
Orifice Rod Assemblies (ORA)	
Rod material	304 SS, annealed
Number of orifice rods per assembly	16
Core Structure	
Core barrel ID/OD, inches	141/145
Thermal shield ID/OD, inches	141/151

TABLE 1.3-1
(Sheet 3 of 7)

DESIGN PARAMETERS - THREE MILE ISLAND UNIT 1

3. Nuclear Design Data	
Structural Characteristics	
Fuel weight (as $U0_2$), metric tons	98.4
Core diameter, inches (equivalent)	128.9
Core height, inches (active fuel)	See Table 3.2-11
Number of fuel assemblies	177
Fuel rods/fuel assembly	208
Performance Characteristics	
Loading technique	Very low leakage
Core average burnup, MWd/Mtu	See Table 3.2-2
4. Principal Design Parameters of the Reactor Coolant System	
Reference design system heat output, MWt	2,584
Operating pressure, psig	2,185
Reactor inlet temperature, °F	555.6
Reactor outlet temperature, °F	602.4
Number of loops	2
Design pressure, psig	2,500
Design temperature, °F	650
Hydrostatic test pressure (cold), psig	3,125
Coolant volume, including pressurizer, ft^3	11,245*

*For specific componenet volumes, refer to Tables 4.2-1 through 4.2-5

TABLE 1.3-1
(Sheet 4 of 7)

DESIGN PARAMETERS - THREE MILE ISLAND UNIT 1

5. Reactor Coolant System Code Requirements

Reactor vessel and closure head	ASME SECTION III, Class A
Steam Generator	
Tube side	ASME SECTION III, Class 1
Shell side	ASME SECTION III, Class 1
Pressurizer	ASME SECTION III, Class A
Pressurizer safety valves	ASME SECTION III, Art. 9
Reactor coolant piping	USAS B31.7

6. Principal Design Parameters of Reactor Vessel

Material	SA-533, Grade B, Clad With 18-8 Stainless Steel
Design pressure, psig	2,500
Design temperature, °F	650
Operating pressure, psig	2,185

TABLE 1.3-1
(Sheet 5 of 7)

DESIGN PARAMETERS - THREE MILE ISLAND UNIT 1

7. Principal Design Parameters of the Steam Generators

Number of units	2
Type	Vertical, once-through with Integral superheater
Tube material	Alloy 690 TT
Shell material	Carbon Steel
Tube side design pressure, psig	2,500
Tube side design temperature, °F	650
Tube side design flow, lb/hr	65.66×10^6
Shell side design pressure, psig	1150
Shell side design temperature, °F	605
Operating pressure, tube side normal, psig	2,185
Operating pressure, shell side, nominal, psig	910

8. Principal Design Parameters of the Reactor Coolant Pumps

Number of units	4
Type	Vertical, single stage
Design pressure, psig	2,500
Design temperature, °F	650
Operating pressure, nominal, psig	2,185
Design capacity, gpm	88,000
Motor type	a-c Induction, single speed
Motor rating (nameplate), hp	9,000

TABLE 1.3-1
(Sheet 6 of 7)

DESIGN PARAMETERS - THREE MILE ISLAND UNIT 1

9. Principal Design Parameters of the Reactor Coolant Piping

Material	Carbon Steel Clad With SS
Hot leg (ID), inches	36
Cold leg (ID), inches	28

10. Engineered Safety Features

Safety Injection System	
Number of high head pumps	3
Number of low head pumps	2
Reactor Building Coolers	
Type	Finned Tube
Number of units	3
Capacity, each, at accident condition, Btu/hr	80×10^6
Core Flooding System	
Number of tanks	2
Total volume, Each, ft^3	1,410
Reactor Building Spray	
Number of pumps	2
Capacity, each, gpm	1,500
Spray additive for iodine removal	Trisodium phosphate

TABLE 1.3-1
(Sheet 7 of 7)

DESIGN PARAMETERS - THREE MILE ISLAND UNIT 1

Emergency Power		
Type	diesel	*(Hydro Units)
Quantity/capacity	2/3,000 kW each	*(2/87, 500 kVA each)
Exclusion radius, feet	2000	*(1 mile)
Low population	2 miles	*(6 miles)
Reactor Building		
Type	Prestressed, post-tensioned concrete structure	
Leak rate	0.1%	*(.5%)
Volume ft^3	$> 2 \times 10^6$	*(1,910,000)
Pressure psig	55 psig	*(59)
Emergency Feedwater Pump		
Type	1 Steam driven 2 Motor driven Any 2 of 3 pumps delivering 550 gpm total to both OTSGs at 1050 psig	*(1 Steam driven 7-1/2 percent capacity)

1.4 PRINCIPAL ARCHITECTURAL AND DESIGN CRITERIA

The Three Mile Island Nuclear Station Unit 1 has been designed and constructed taking into consideration the general criteria for nuclear power plant construction permits as listed in the proposed AEC General Design Criteria, dated July 1967 which are applicable to this Unit. In the discussion of each criterion, references are made to sections of this report where more detailed information is presented. The principal safety features that meet each criterion are summarized as follows:

1.4.1 CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality Assurance programs, test procedures, and inspection acceptance levels used shall be identified. A showing of sufficiency and applicability of codes, standards, Quality Assurance programs, test procedures, and inspection acceptance levels used is required.

DISCUSSION

a. Essential Systems and Components

The integrity of systems, structures, and components essential to accident prevention and to mitigation of accident consequences has been included in the reactor design evaluations. These systems, structures, and components are:

1) Fuel assemblies

2) Reactor Vessel internals

3) Reactor Coolant System

4) Reactor instrumentation, control, and protection system

5) Engineered safeguards

6) Radioactive materials handling systems

7) Reactor Building

8) Electric power sources

b. Codes and Standards

Applicable codes and standards for the nuclear unit as included in the applicable section of FSAR.

c. Quality Assurance Programs

The Initial design and construction QA Program is described in Section 1.6. The Quality Assurance Topical Report is described in Chapter 12.

1.4.2 CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of Reactor Building facilities which are essential to the prevention of accidents, which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornados, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (1) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area, and (2) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

DISCUSSION

The systems and components identified in Section 1.4.1 (Criterion 1) have been designed to performance standards that enable the facility to withstand, without loss of capability to protect the public, the additional forces or effects which might be imposed by natural phenomena. The designs are based upon the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in the historical data, or upon the most severe conditions which are susceptible to synthetic analyses.

The following conditions are discussed in Chapters 2 and 5:

a. Earthquakes

b. Tornadoes

c. Floods

d. Winds

e. Ice

f. Other local site effects

1.4.3 CRITERION 3 - FIRE PROTECTION (Category A)

The reactor facility shall be designed: (1) to minimize the probability of events such as fires and explosions, and (2) to minimize the potential effects of such events on safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, Control Room, and components of engineered safety features.

DISCUSSION

The reactor facility is designed to minimize the probability of fire and explosion. Noncombustible and fire resistant materials are used throughout the facility as indicated in the following referenced chapters of the FSAR:

a.	Reactor Building	Chapter 5
b.	Control Room	Chapter 7
c.	Electrical distribution equipment	Chapter 8

1.4.4 CRITERION 4 - SHARING OF SYSTEMS (Category A)

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

DISCUSSION

Three Mile Island Nuclear Station Unit 1 makes use of some facilities on a shared basis with Unit 2. None of the shared components are connected with safety features or control systems of either nuclear steam supply systems.

1.4.5 CRITERION 5 - RECORDS REQUIREMENTS (Category A)

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the licensee or under corporate control throughout the life of the reactor.

DISCUSSION

The following records are maintained by the licensee:

a. A complete set of as-built facility plans and system diagrams which include arrangement plans, system diagrams, major structural plans, and technical manuals of major installed equipment. These are maintained in accordance with our procedures governing Configuration Control.

b. A set of completed test procedures for all plant testing outlined in Chapter 13, Tables 13.1-2, 13.1-3 and 13.1-4.

c. The inspection records and test data as required by the specifications for the essential components of the plant are maintained in a quality control history file. The systems and components for which such files are maintained include as a minimum:

    1) Fuel assemblies

    2) Reactor vessel internals

    3) Reactor Coolant System

4) Reactor instrumentation, control, and protection systems

5) Engineered safeguards

6) Reactor containment

1.4.6 CRITERION 6 - REACTOR CORE DESIGN (Category A)

The reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

DISCUSSION

The reactor core is designed with the necessary margins to accommodate, without fuel damage, expected transients from steady-state operation including the transients given in the criterion. Fuel clad integrity is assured by avoiding clad overstressing and overheating. The evaluation of clad stresses includes the effects of internal and external pressures, temperature gradients and changes, clad-fuel interactions, vibrations, and earthquake effects. Clad fatigue due to power and pressure cycling is minimized by prepressurizing all fuel rods with helium. The fuel rod design prevents collapse at the end volume region of the fuel rod and provides sufficient radial and end void volume to accommodate clad-fuel interactions and internal gas pressures.

Clad overheating is prevented by satisfying the core thermal and hydraulic criteria.

a. At the design overpower, no fuel melting will occur.

b. A 95 percent confidence exists that at least 95 percent of the fuel rods in the core will be in no jeopardy of experiencing a DNB during continuous operation at the design overpower of 112 percent, based upon a reference core design of 2568 MWt.

The design margins allow for deviations of temperature, pressure, flow, reactor power, and reactor-turbine power mismatch. Above 22 percent power, the reactor is operated at a constant average coolant temperature and has a negative power coefficient to damp the effects of power transients. The reactor control system maintains the reactor operating parameters within preset limits, and the reactor protection system shuts down the reactor if normal operating limits are exceeded by preset amounts.

Reactor decay heat is removed through the steam generators until the reactor coolant system is cooled to approximately 280F. Steam generated by decay heat supplies the steam-driven main feedwater pump turbine, and can also be vented to atmosphere and/or bypassed to the condenser. The steam generators are supplied feedwater from either the main steam-driven feedwater pumps, or from the emergency feedwater pumps. Main and emergency feedwater pump capacities are described in Chapter 10.

The main feedwater pumps supply the steam generators with water contained in the feedwater train and the condensate storage tanks. The emergency feed pumps take suction from the condensate storage tanks or from the condenser hotwell. These sources provide sufficient coolant to remove decay heat for at least one day after reactor shutdown with primary heat sink (condenser) isolated. The condenser is normally available so that water inventory is not depleted.

The reactor coolant pumps are provided with sufficient inertia to maintain adequate flow during coastdown to prevent fuel damage if power to all pumps is lost. Natural circulation coolant flow provides adequate core cooling after the pump energy has dissipated.

1.4.7 CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (Category B)

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

DISCUSSION

Power oscillations resulting from variations of coolant temperature are minimized by constant average coolant temperature. Power oscillations from spatial xenon effects are minimized by the large negative power coefficient. Features have been provided in the design to allow control of axial oscillations and to stabilize the core with respect to azimuthal oscillations. Analysis has shown the core to be stable with respect to radial oscillations.

The ability of the reactor control and protection system to control the oscillations resulting from variation of coolant temperature within the control system deadband and from spatial xenon oscillations has been analyzed.

Burnable poison rod assemblies and fuel rods containing gadolinia as an integral burnable poison are provided to assure a suitable moderator temperature coefficient and satisfactory radial power peaking.

Beginning with cycle 4, the reactor is operated in a rods-out, feed and bleed mode. The core reactivity control is supplied mainly by soluble boron and supplemented by CRAs. Prior to Cycle 19, axial power-shaping rod assemblies (APSRAs) were available during operation to maintain an acceptable power distribution in the core and to control any tendency towards axial oscillations. The APSRAs were determined to be unnecessary and axial power distribution control during operation with the ASPRAs removed is accomplished by adjusting regulating rod group position, as required, to prevent or damp xenon oscillations.

1.4.8 CRITERION 8 - OVERALL POWER COEFFICIENT (Category B)

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

DISCUSSION

The overall power coefficient is negative in the power operating range.

1.4.9 CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

DISCUSSION

The reactor coolant system pressure boundary meets the criterion through the following:

a. Material selection, design, fabrication, inspection, testing, and certification in accordance with ASME and USAS codes.

b. Manufacture and erection in accordance with approved procedures.

c. Inspection in accordance with ASME and USAS code requirements plus additional requirements imposed by the manufacturer.

d. System analysis to account for cyclic effects of thermal transients, mechanical shock, seismic loadings, and vibratory loadings.

e. Selection of reactor vessel material properties to give due consideration to neutron flux effects and the resultant increase of the nil-ductility transition temperature.

f. Quality Assurance program described in Chapters 1 and 12.

The materials, codes, cyclic loadings, and non-destructive testing are discussed further in Chapter 4.

1.4.10 CRITERION 10 - CONTAINMENT (Category A)

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires, the functional capability to protect the public.

DISCUSSION

The containment structure is designed to provide adequate protection and safety to the public under all normal and accident conditions. Considerations which are used in the design of the containment to assure compliance with the above criterion can be found in Chapters 5 and 6.

1.4.11 CRITERION 11 - CONTROL ROOM (Category B)

The facility shall be provided with a Control Room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the Control Room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposure of personnel in excess of 10CFR20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the Control Room is lost due to fire or other cause.

DISCUSSION

Safe occupancy of the Control Room during abnormal conditions has been provided for in the design. The Control Room is located in a Class I structure which is designed for the hypothetical aircraft incident. Adequate shielding has been provided to maintain tolerable radiation levels in the Control Room even in the event of a maximum hypothetical accident. The control building ventilation system has redundant fans and chillers and is provided with radiation detectors and smoke detectors with appropriate alarms and interlocks. Provisions have been made for the control building air to be recirculated through HEPA and charcoal filters. Fresh air is drawn through an underground ventilation tunnel which has been provided with protection against combustible vapors, incipient explosions, or fires. The tunnel is also designed for the hypothetical aircraft incident.

Denial of access to the Control Room is considered to be very improbable; nevertheless, capability to shut the reactor down and maintain it in a safe condition while access to the Control Room is denied is provided.

1.4.12.1 CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (Category B)

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

DISCUSSION

Reactor regulation is based upon the use of movable control rods and a chemical neutron absorber (boron in the form of boric acid) dissolved in the reactor coolant. Input signals to the reactor controls include reactor coolant average temperature, megawatt demand, and reactor power. The reactor controls are designed to maintain a constant average reactor coolant temperature over the load range from 22 to 100 percent of rated power. The steam system operates at constant pressure at all loads. Adequate instrumentation and controls are provided to maintain operating variables within their prescribed ranges.

The non-nuclear instrumentation measures temperatures, pressures, flows, and levels in the reactor coolant system, steam system, and auxiliary reactor systems, and maintains these variables within prescribed limits.

1.4.12.2 CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (Category B)

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

DISCUSSION

This criterion is met by means of reactivity control and Control Room display. Reactivity control is by movable control rods and by chemical neutron absorber (in the form of boric acid) dissolved in the reactor coolant. The position of each control rod is displayed in the Control Room. Changes in the reactivity status due to soluble boron is indicated by changes in the position of the control rods. Actual boron concentration in the reactor coolant is determined periodically by sampling and analysis.

1.4.14 CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

DISCUSSION

The reactor design meets this criterion by reactor trip provisions and engineered safeguards. The reactor protection system is designed to limit reactor power which might result from unexpected reactivity changes and provides an automatic reactor trip to prevent exceeding acceptable fuel damage limits. In a loss of coolant accident, the engineered safeguards actuation system automatically actuates the high pressure and low pressure injection (makeup and purification and decay heat removal) systems. The core flooding tanks are self- actuating. Certain long term operations in the emergency core cooling systems which do not require immediate actuation, such as remote switching of the low pressure injection pumps to the recirculation mode and sampling of the recirculated coolant, are performed manually by the operator.

1.4.15 CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (Category B)

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

DISCUSSION

The safeguards actuation system senses reactor coolant system pressure and Reactor Building pressure and initiates emergency core coolant injection, Reactor Building isolation, and Reactor Building cooling.

1.4.16 CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY (Category B)

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

DISCUSSION

Reactor coolant pressure boundary integrity can be continuously monitored in the Control Room by surveillance variation from normal conditions for the following:

a. Reactor Building sump level

b. Reactor Building radioactivity levels

c. Condenser offgas radioactivity levels (to detect steam generator tube leakage)

d. Decreasing makeup tank water level (indicating system leakage)

Gross leakage from the reactor coolant boundary has also been indicated by a decrease in pressurizer water level and rapid increase in the Reactor Building sump water level.

1.4.17 CRITERION 17 - MONITORING RADIOACTIVITY RELEASE (Category B)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

DISCUSSION

Monitors are provided for the containment atmosphere, the facility effluent discharge paths, and the facility environs as required to monitor activity that is released as a result of normal operation, anticipated transient conditions, and postulated accidents. The monitors provided for the unit and their functions are described in Chapter 11.

1.4.18 CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (Category B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

DISCUSSION

Radiation monitors and alarms are provided in the reactor, fuel handling, and auxiliary buildings as required to warn personnel of impending excessive levels of radiation or airborne activity or of conditions that might contribute to loss of continuity in decay heat removal. The radiation monitors provided for the unit and their functions are described in Chapter 11.

1.4.19 CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category B)

Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

DISCUSSION

The protection system design meets this criterion by specific instrument location, component redundancy, and inservice testing capability. The major design criteria stated below have been applied to the design of the instrumentation.

a. No single component failure shall prevent the protection systems from fulfilling their protective function when action is required.

b. No single component failure shall initiate unnecessary protection system action, provided implementation does not conflict with the criterion above.

Test connections and capabilities are built into the protection systems to provide for:

a. Pre-operational testing to give assurance that the protection systems can fulfill their required functions.

b. On-line testing to assure availability and operability.

1.4.20 CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (Category B)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from services of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used, where necessary, to achieve true independence of redundant instrumentation components.

DISCUSSION

Reactor protection is by four channels with 2 out of 4 coincidence, and engineered safeguards are by three channels with 2 out of 3 coincidence. All protection system functions are implemented by redundant sensors, instrument strings, logic, and action devices that combine to form the protection channels. Redundant protection channels and their associated elements are electrically independent and packaged to provide physical separation. The reactor protection system initiates a trip of the channel involved when modules or equipment is removed.

1.4.21 CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

Multiple failures resulting from a single event shall be treated as a single failure.

DISCUSSION

The protection systems meet this criterion in that the instrumentation is designed so that a single event cannot result in multiple failures that would prevent the required protective action.

1.4.22 CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (Category B)

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

DISCUSSION

The protection systems' instrument strings are electrically and physically independent. Shared instrumentation for protection and control functions satisfies the single failure criteria by the employment of isolation techniques to the multiple outputs of various instrument strings.

1.4.23 CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (Category B)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in a loss of the protection function.

DISCUSSION

The protection systems are designed to extreme ambient conditions. The protection systems' instrumentation in the Reactor Building will operate from 40F to 140F and withstand the loss of coolant building environmental conditions, including 100 percent relative humidity, without loss of operability. Out-of-core neutron detectors, however, will withstand only 90 percent relative humidity. The protective systems' instrumentation was subject to environmental (qualification) testing as required by IEEE Standard 279 (see 7.5, Reference 2).

Further evaluation was performed for environmental qualification of Class 1E electrical equipment in response to NRC I&E Bulletin 79-01B as described in Appendix 6B of the FSAR.

1.4.24 CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (Category B)

In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

DISCUSSION

The design of this unit conforms to the criterion. In the event of loss of offsite power, power will be supplied from two automatic fast startup diesel engine generators. These are sized so that either one can carry the required engineered safeguards load. The nameplate rating of each emergency generator is 3000 kW at 0.8 power factor for 2000 hours. Each emergency generator will feed one of the engineered safeguards 4160-V buses. Each generator is capable of feeding the required safeguards load of one 4160-V bus plus selected balance of plant emergency loads following any LOCA.

1.4.25 CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (Category B)

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

DISCUSSION

Test circuits are supplied which utilize the redundant, independent, and coincidence features of the protection systems. This makes it possible to manually initiate on-line trip signals in any single protection channel in order to test trip capability in each channel without affecting the other channels.

1.4.26 CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (Category B)

The protection systems shall be designed to fail into a safe state or into a state established as tolerable or a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

DISCUSSION

The reactor protection system will trip the reactor on loss of power. The engineered safeguards systems are supplied with multiple sources of electric power for control and valve action. A total loss of electrical power to the engineered safeguards actuation system will cause it to assume a tripped position with the exception of the Reactor Building spray actuation.

The system is designed for continuous operation under adverse environments, as described in the discussion of Criterion 23.

Redundant instrument channels are provided for the reactor protection and engineered safeguards actuation systems. Loss of power to each individual reactor protection channel will trip that individual channel. Loss of all instrument power will trip the reactor protection system and activate the safeguards actuation system instrumentation.

Manual reactor trip is designed so that failure of the automatic reactor trip circuitry will not prohibit or negate the manual trip. The same is true with respect to manual operation of the engineered safeguards equipment.

1.4.27 CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

At least two independent reactivity control systems, preferably of different principles, shall be provided.

DISCUSSION

This criterion is met by movable control rods and soluble boron poison injection.

1.4.28 CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

At least two of the reactivity control systems provided shall be independently capable of making and holding the core subcritical from any hot standby or hot operating conditions, including those resulting from power changes sufficiently fast to prevent exceeding acceptable fuel damage limits.

DISCUSSION

A single reactivity control system consisting of 61 control rods is provided to protect the core from damage due to the effects of any operating transient. The soluble absorber reactivity control system can add negative reactivity to make the reactor subcritical. However, its action is slow, and its ability to protect the core from damage, which might result from rapid load changes such as a full load turbine trip, is not a design criterion for this system. The high degree of redundancy in the control rod system is considered sufficient to meet the intent of this criterion.

1.4.29 CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (Category A)

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients), sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins, greater than the maximum worth of the most effective control rod when fully withdrawn, shall be provided.

DISCUSSION

The reactor design meets this criterion both under normal operating conditions and under the accident conditions set forth in Chapter 14. The reactor is designed with the capability of providing a shutdown margin of at least 1 percent delta-k/k with the single most reactive control rod fully withdrawn at any point in core life with the reactor at a hot, zero power condition. Shutdown margin data for the current core is described in Chapter 3.2.2.

1.4.30 CRITERION 30 - REACTIVITY HOLDDOWN CAPABILITY (Category B)

At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

DISCUSSION

The reactor meets this criterion with control rods for hot shutdown under normal operating conditions and for shutdown under the accident conditions set forth in Chapter 14. Reactor subcritical margin is maintained during cooldown by changes in soluble boron concentration. The rate of reactivity compensation from boron addition is greater than the reactivity change associated with the reactor cooldown rate of 100 °F/hour. Thus, subcriticality is assured during cooldown with the most reactive control rod totally unavailable.

1.4.31 CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (Category B)

The reactivity control systems shall be capable of sustaining any single malfunction, such as unplanned continuous withdrawal (not ejection) of a control rod without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

<u>DISCUSSION</u>

The reactor design meets this criterion. A reactor trip will protect against continuous withdrawal of a control rod.

1.4.32 CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (Category A)

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements, and on rates at which reactivity can be increased to insure that the potential effects of a sudden or large change of reactivity cannot: (1) rupture the reactor coolant pressure boundary, or (2) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

<u>DISCUSSION</u>

The reactor design meets this criterion by safety features which limit the maximum reactivity insertion rate. These include rod-group withdrawal interlocks, soluble boron concentration reduction interlock, maximum rate of dilution water addition, and dilution-time cutoff. In addition, the rod drives and their controls have an inherent feature that limits overspeed in the event of malfunctions. Ejection of the maximum-worth control rod will not lead to further coolant boundary rupture or to internals damage which would interfere with emergency core cooling.

1.4.33 CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (Category A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

<u>DISCUSSION</u>

The reactor design meets this criterion. There are no credible mechanisms whereby damaging energy releases are liberated to the reactor coolant. Ejection of the maximum-worth control rod will not lead to further coolant boundary rupture.

1.4.34 CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (Category B)

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given: (1) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (2) to the state of stress of materials under static and transient loadings, (3) to the quality control specified for materials and component fabrication to limit flaw sizes, and (4) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

DISCUSSION

The reactor coolant pressure boundary design meets this criterion by the following means:

a. Development of reactor vessel plate material properties opposite the core to a specified Charpy-V-notch test result of 30 ft-lb or greater at a nominal low NDTT.

b. Determination of the fatigue usage factor resulting from expected static and transient loading during detailed design and stress analysis.

c. Quality control procedures including permanent identification of materials and non-destructive testing.

d. Operating restrictions to prevent failure towards the end of design vessel life resulting from increase in the nil-ductility transition temperature (NDTT) due to neutron irradiation, as predicted by a material irradiation surveillance program.

In accordance with 10CFR50.61, the projected values of material properties for fracture toughness requirements for protection against pressurized thermal shock events have been calculated as described in FSAR Section 4.3.3.e.

1.4.35 CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (Category A)

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120F above the nil-ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation, or 60F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

DISCUSSION

The reactor vessel is the only reactor coolant system component exposed to a significant level of neutron irradiation and is, therefore, the only component subject to material irradiation damage. However, sufficient testing and analysis of ferritic materials in reactor coolant system pressure boundary components will be performed to assure that the required NDT limits specified in the criterion are met. Unit operating procedures limit the operating pressure to 20 percent of the design pressure when the reactor coolant system temperature is below NDTT + 60F throughout unit life.

Analysis has shown no potential reactivity-induced conditions which will result in energy release to the primary system in the range expected to be absorbed by plastic deformation.

1.4.36 CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

DISCUSSION

The reactor coolant pressure boundary components meet this criterion. Schedule time is provided for non-destructive testing during plant shutdown. A reactor pressure vessel material surveillance program conforming to ASTM-E-185-66 has been established. An integrated reactor vessel material surveillance program has been established as described in Section 4.4.5 of the FSAR, which complies with the requirements of BAW Topical Report BAW-1543 and 10CFR50 Appendix H.

1.4.37 CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

DISCUSSION

The reactor design meets this criterion. The emergency core cooling systems can protect the reactor for any size leak up to and including the circumferential rupture of the largest reactor coolant pipe.

1.4.38 CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (Category A)

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

DISCUSSION

All engineered safeguards systems are designed so that a single failure of an active component in a system will not prevent operation of that system or reduce its capacity below that required to maintain a safe condition. Two independent Reactor Building cooling systems, each having full heat removal capacity, are provided to prevent overpressurization.

The high pressure injection, core flooding, and low pressure injection systems have separate equipment and instrumentation strings to ensure availability of capacity.

Some portions of the engineered safeguards systems have both a normal and an emergency function. During operation, the standby and operating units can be rotated into service.

Engineered safeguards systems equipment piping that is not fully protected against LOCA missile damage utilizes dual lines to preclude loss of the protective function as a result of the secondary failure.

Testing and inspection of the engineered safeguards systems is further described in Chapter 6.

1.4.39 CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES (Category A)

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

DISCUSSION

The electrical systems conform to this criterion. The systems have been designed with sufficient power sources, redundant buses, and required switching to provide reliable electrical power during all modes of operation and shutdown conditions. Engineered safeguards auxiliaries are arranged so that loss of any emergency generator or a single safeguards bus for any reason will still leave sufficient auxiliaries to safely perform the required function.

1.4.40 CRITERION 40 - MISSILE PROTECTION (Category A)

Protection for engineered safeguards shall be provided against dynamic effects and missiles that might result from plant equipment failures.

DISCUSSION

Active engineered safeguards are protected against dynamic effects and missiles hypothesized to result from plant equipment failure. This is accomplished by shielding and/or separation of redundant components.

1.4.41 CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (Category A)

Engineered safety features, such as emergency core cooling and containment heat removal systems, shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

DISCUSSION

All engineered safeguards systems are designed so that a single failure of an active component will not prevent operation of that system or reduce the system capacity below that required to maintain a safe condition. Redundancy is provided in equipment and pipelines so that the failure of a single active component of any system will not impair the required safety function of that system.

1.4.42 CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (Category A)

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss of coolant accident.

DISCUSSION

The engineered safeguards system design meets this criterion. A single-failure analysis of the emergency core cooling systems, and Reactor Building heat removal systems, demonstrates that the systems have sufficient redundancy to perform their design functions.

The core flooding tanks contain check valves which operate to permit flow of emergency coolant from the tanks to the reactor vessel. These valves are self-actuating and need no external signal or external supplied energy to make them operate. Accordingly, it is not considered credible that they would fail to operate when needed.

The engineered safeguards features are designed to function in the unlikely event of a loss of coolant accident with no impairment of function due to the effects of the accident.

1.4.43 CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

DISCUSSION

The engineered safeguards systems are designed to meet this criterion. The water injected to ensure core cooling is sufficiently borated to ensure core subcriticality. Nonessential sources of water inside the Reactor Building are automatically isolated to prevent dilution of the borated coolant. Essential sources of postaccident cooling water are monitored to detect leakage which may lead to dilution of boron content. An analysis has been made to demonstrate that the injection of cold water on the hot reactor coolant system surfaces will not lead to further failure (BAW 1715, Doc No. 77-1130658-00, June 1982). The design of the equipment and its actuating system ensures that water injection will occur in a sufficiently short time period to preclude significant metal-water reactions and consequent energy release to the Reactor Building.

1.4.44 CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY (Category A)

At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function, and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that: (1) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (2) failure of the shared feature or component does not initiate a loss of coolant accident, and (3) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss of coolant accident and is not lost during the entire period this function is required following the accident.

DISCUSSION

Emergency core cooling is provided by pumped injection and pressurized core flooding tanks. Pumped injection is subdivided in such a way that there are two separate and independent strings, each including both high pressure and low pressure coolant injection, and each capable of providing 100 percent of the necessary core injection with the core flooding tanks. There is no sharing of active components between the two subsystems in the postaccident operating mode. The core flooding tanks are passive components which are needed for only a short period of time after the accident, thereby assuring 100 percent availability when needed. This equipment prevents clad melting for the entire spectrum of the reactor coolant system failures ranging from the smallest leak to the complete severance of the largest reactor coolant pipe.

1.4.45 CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems including reactor vessel internals and water injection nozzles.

DISCUSSION

All critical parts of the emergency core cooling system, including the reactor vessel internals and water injection nozzles can be inspected during plant shutdown.

1.4.46 CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEMS COMPONENTS (Category A)

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

DISCUSSION

The design of emergency core cooling systems and components has incorporated adequate test and operational features to permit periodic testing of active components to assure operability and functional capability.

1.4.47 CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

DISCUSSION

The high pressure (makeup and purification) and low pressure injection (decay heat removal) systems are included as part of normal service systems. Consequently, the active components can be tested periodically for delivery capability. The core flooding system delivery capability was demonstrated during startup testing. In addition, all valves are periodically cycled to ensure operability. With these provisions, the delivery capability of the emergency core cooling systems can be periodically demonstrated.

1.4.48 CRITERION 48 - TESTING OF OPERATIONAL SEQUENCY OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test, under conditions as close to design as practical, the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

DISCUSSION

The operational sequence that would bring the emergency core cooling systems into action, including transfer to alternate power sources, can be tested in parts.

1.4.49 CRITERION 49 - CONTAINMENT DESIGN BASIS (Category A)

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a loss of coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

DISCUSSION

The containment structure, access openings, penetrations, and necessary containment heat removal systems are designed to accommodate the loads specified in Section 5. The design is based upon the factored loads and load combinations as specified in Section 5 and will limit the leakage rate from containment following the MHA, to the design value.

1.4.50 CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (Category A)

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30°F above nil-ductility transition temperature (NDTT).

DISCUSSION

Consideration of NDTT requirements for ferritic materials is described in Chapter 5.

1.4.51 CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (Category A)

If part of the reactor coolant pressure boundary is outside the containment, appropriate features, as necessary, shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features, such as isolation valves and additional containment, shall include consideration of the environmental and population conditions surrounding the site.

DISCUSSION

The reactor coolant pressure boundary is defined as those piping systems or components which contain reactor coolant at high pressure and temperature. With the exception of the reactor coolant sampling lines, the reactor coolant pressure boundary, as defined above, is located entirely within the Reactor Building. The sampling lines are provided with remotely operated valves for isolation in the unlikely event of a failure. These lines are normally isolated and are used only during actual sampling operations. All other piping and components which may contain reactor coolant are at low temperatures such that any leakage would be collected by the waste disposal system. No significant environmental dose would result from these sources.

1.4.52 CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (Category A)

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

DISCUSSION

Two systems of different principles are provided to remove heat from the Reactor Building following an accident in order to maintain the pressure below the containment design pressure. The Reactor Building Spray and the Reactor Building Emergency Cooling systems are each capable of removing sufficient energy to maintain the pressure below the containment design pressure.

One half capacity (one spray path) of the Reactor Building Spray System and one cooler of the Reactor Building Emergency Cooling System when operated together have sufficient heat removal capacity to limit containment pressure.

1.4.53 CRITERION 53 - CONTAINMENT ISOLATION VALVES (Category A)

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

DISCUSSION

This unit is in full compliance with this criterion. Each line that penetrates the Reactor Building liner is valved according to the following criterion:

Leakage through all fluid penetrations not serving accident consequence limiting systems is minimized by a double barrier so that no single credible failure or malfunction of an active component can result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the Reactor Building, and various types of isolation valves.

The detailed implementation of this criterion is described in Chapter 5 and presented in Table 5.3-2.

1.4.54 CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING (Category A)

Containment shall be designed so that an integrated leakage rate test can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with the required performance.

DISCUSSION

The containment system is designed and constructed and the necessary equipment is provided to permit the conduct of an initial integrated leakage rate test which will verify that the leakage rate does not exceed the leakage rate criteria detailed in the Technical Specifications. The equipment provided for integrated leak rate testing preoperational leak monitoring and the initial integrated leak rate test is described in Chapter 5 and the Technical Specifications.

1.4.55 CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING (Category A)

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

DISCUSSION

The containment system is designed and constructed and the necessary equipment is provided to permit the conduct of periodic integrated leakage rate tests at design pressure during plant lifetime. The equipment provided for periodic integrated leakage rate testing of containment is described in Chapter 5. Details concerning the conduct of periodic integrated leakage rate tests are presented in Chapter 5 and Technical Specifications.

1.4.56 CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (Category A)

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time.

DISCUSSION

The penetration pressurization system provides continuous pressurization and a means of continuously monitoring the leakage rate from the equipment hatch resilient seals. Continuous leakage rate monitoring of the electrical penetrations, although not required, is conducted at 30 psig. For penetrations having resilient seals that are not continuously pressurized (the purge isolation valves, the doors of the two personnel access airlocks, and the fuel transfer tube flange “O” rings), there are special provisions for conducting individual leakage rate tests at design pressure at any time. Chapter 5 and the Technical Specifications indicate details of periodic leakage rate testing of penetrations.

1.4.57 CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES (Category A)

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

DISCUSSION

This unit is in full compliance with this criterion.

All power operated valves essential to the containment isolation function have controls to permit remote manual operation of the valves from the Control Room. Test operations of these valves are performed in accordance with the test schedule in Technical Specifications.

Fittings are provided in the piping upstream and downstream of all isolation valves, which are subject to Class C containment leakage tests, to permit periodic leakage rate testing with air pressure applied across each valve.

1.4.58 CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

DISCUSSION

Containment pressure reducing systems are the Reactor Building spray system and the Reactor Building emergency cooling system. The Reactor Building cooling units, the Reactor Building sump, and Reactor Building spray pumps are so located that physical inspection of these items is possible during normal plant operation. The spray rings and nozzles of the Reactor Building spray system are located in the dome of the Reactor Building. An air connection is provided on the supply piping to the spray rings from each spray pump for testing the spray nozzles.

Functional operability of each nozzle is tested by blowing air or smoke into the spray rings and observing tell-tale devices such as streamers or balloons.

1.4.59 CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEM COMPONENTS (Category A)

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves can be tested periodically for operability and required functional performance.

DISCUSSION

The containment pressure-reducing systems have the capability of being periodically tested as follows:

a. Reactor Building Cooling System

   1) The three Reactor Building Recirculation Fans can be individually tested for low speed operations.

   2) The emergency cooling coils service water valves can be operated through their full travel.

   3) The emergency cooling river water pumps can be tested for automatic starting.

b. Reactor Building Spray System

   1) The operation of the spray pumps can be tested by recirculating to the borated water storage tank through a test line.

   2) The building spray isolation valves can be operated through their full travel.

1.4.60 CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (Category A)

A capability shall be provided to periodically test the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

DISCUSSION

The delivery capability of the spray nozzles are tested by blowing low pressure air or smoke through the system and verifying flow through the nozzles.

The delivery capability of the pumps are tested by recirculating to the borated water storage tank and monitoring the resultant flow.

1.4.61 CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to the design as practical, the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

DISCUSSION

Provisions to test the operational sequence of the containment pressure reducing systems (Reactor Building spray system and Reactor Building cooling system) are included in the design of Three Mile Island Nuclear Station. As shown on Drawing 302831, provisions are available whereby the emergency cooling coils of the Reactor Building cooling system can be subjected to nuclear services cooling water flow. The operational sequence test is performed: 1) by operating the Reactor Building cooling fan and the emergency river water pump from the normal source of power, 2) by automatically transferring the operation of the above mentioned equipment from their normal source of power to the emergency power source, and 3) by simultaneously operating from the emergency power source required to establish flow through the emergency cooling coil.

Test connections are provided for testing the building spray nozzles and the Reactor Building spray pumps. One test connection provides for the air test of the containment building spray nozzles. The other test connection provides for an operational test of the spray pumps by recirculating back to the borated water storage tank.

A description of the action of switching the components of these systems to alternate power sources is presented in Chapter 8.

1.4.62 CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as ducts, filters, fans, and dampers.

DISCUSSION

The containment purge system is utilized intermittently during normal plant operation to replace the atmosphere within containment with fresh air. Under an accident condition, the purge system is isolated. Containment air cleanup for postaccident iodine removal is accomplished by use of a chemical spray system and this criterion, therefore, is not considered applicable for this unit. (Reference Criterion 58.)

1.4.63 CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS (Category A)

Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

DISCUSSION

Refer to Criterion 62.

1.4.64 CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS (Category A)

A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure: (1) filter bypass paths have not developed, and (2) filter and trapping materials have not deteriorated beyond acceptable limits.

DISCUSSION

Refer to Criterion 62.

1.4.65 CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

DISCUSSION

Refer to Criterion 62.

1.4.66 CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

DISCUSSION

Refer to Chapter 9.7.2.3 for discussion of criticality safety analysis of new and spent fuel storage.

1.4.67 CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

DISCUSSION

This criterion is met by the spent fuel cooling system which incorporates provisions to maintain water cleanliness, temperature, and water level. Two pumps and two coolers are adequate to maintain the spent fuel pool temperature within acceptable limits. The pumps in this system operate to provide continuous cooling capability in the fuel storage facility (See Section 9.4).

1.4.68 CRITERION 68 - FUEL AND WASTE STORAGE SHIELDING (Category B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10CFR20.

DISCUSSION

Shielding during handling and storage of spent fuel elements is provided by a combination of borated water and reinforced concrete sufficient to meet or exceed the requirements of 10CFR20 for radiation protection. Components and piping located within the reactor, fuel handling, and auxiliary buildings containing primary coolant or other fluids requiring it, are shielded by reinforced concrete sufficient to meet or exceed the requirements of 10CFR20 for radiation protection. Chapter 11 of the FSAR specifies the design criteria for shielding throughout the unit and the design dose rates at various locations. The criteria comply with 10CFR20 limits.

1.4.69 CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

DISCUSSION

All spent fuel storage and radioactive liquid waste processing and storage facilities for this unit are housed in Class I reinforced concrete structures designed and constructed to withstand the hypothetical aircraft incident. The analyses presented in Chapter 14

of hypothetical accidental release of radioactive gaseous wastes demonstrate that the limits of 10CFR100 (at the site boundary) are not exceeded. The analyses cover accidental gas waste releases from the waste gas storage system and gas release from a fuel handling accident.

1.4.70 CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluent, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactivity effluents to the environment. In all cases, the design for radioactivity control shall be justified: (1) on the basis of 10CFR20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur, and (2) on the basis of 10CFR100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

DISCUSSION

The design of the unit incorporates the means necessary to maintain control of releases of radioactive liquid, gas, and solid waste effluents such that these will be within 10CFR20 limits, except tritium, for normal operation and reasonable transient situations and within 10CFR100 limits for accidents of low probability. Ample holdup capacity is provided in the unit for the retention of liquid, gaseous, and solid wastes to ensure that they can always be released in a controlled manner. The liquid waste disposal system provides equipment for extensive decontamination of liquid wastes (if required) prior to their release and capabilities to prepare solid wastes for packaging and offsite shipment. The waste gas system provides storage to permit a design maximum of 90 days storage for radioactive decay of waste gases prior to their release. All normal releases of radioactive liquids and gases are continuously monitored and controlled by the radiation monitoring systems described in Chapter 11. The means provided to control releases of radioactive liquid, gas, and solid waste effluents, under normal and reasonable transient situations, are discussed in Chapters 5, 9, and 11. Chapter 14 presents the results of analyses of releases of radioactive gases which might occur as the result of hypothetical accident situations.

1.5 RESEARCH AND DEVELOPMENT

A number of areas in which research and development would be carried out to finalize design details which were identified during the course of the construction, permit review and later evaluations. A summary of the status of each of those programs follows:

1.5.1 ONCE THROUGH STEAM GENERATOR TEST

Testing necessary to prove the adequacy of the original once through steam generator design for service at the initial power level, and to confirm the size and configuration of the units, has been completed. The results of the tests have been evaluated to the extent necessary to establish that the design criteria have been met and to establish final design characteristics for manufacture of the original steam generators. Reference 1 presents the results of the original once through steam generator R&D. The new once through steam generators were designed to be a like-for-like replacement of the original generators. No additional R&D testing was performed.

1.5.2 CONTROL ROD DRIVE LINE TEST

The test assembly for this program was a full-sized fuel assembly with associated control rod and control rod guide, adjacent internals, and control rod drive. The purpose of this program was to seek out potential material and/or design problems prior to production unit testing.

The test program of the roller nut mechanism was performed by the B&W Research Center in Alliance, Ohio, in sufficient scope and depth to establish that the performance of the mechanism is satisfactory. Reference 2 provides the results and analysis of the test data.

1.5.3 SELF-POWERED DETECTOR TESTS

The self-powered detector tests consisted of qualification testing with sufficient longevity to ensure that both neutron flux and power information can be reliably measured. The self-powered detectors have received an integrated dose which is equivalent to over four full power years at Three Mile Island Unit 1 conditions, and have shown no fault that would prohibit their use in PWR service. The results of the self-powered detector test program are provided in Reference 12.

1.5.4 THERMAL AND HYDRAULIC PROGRAMS

Section 3.3.2 provides a discussion of the Thermal and Hydraulic Tests. The initial core power level for the first core was justifiable on the basis of the well-known W-3 DNB correlation which was the correlation used to develop the design parameters for the initial core. Using the W-3 correlation, the only information required for licensing at the design core power level was that obtained from the reactor vessel model flow tests in which a 1/6 scale model of the vessel and internals was used to measure the flow distribution to the core, fluid mixing in the vessel and core, and the distribution of the pressure drop within the reactor vessel. All of the tests relating to the safety analysis, maximum design power rating, and fabrication of reactor internals was satisfactorily completed. Test data analysis and documentation were conducted and a final report was submitted as Reference 3. As described in Chapter 3 the present core design uses either the BWC or BHTP CHF correlation.

1.5.5 EMERGENCY CORE COOLING AND INTERNALS VENT VALVES

Analytical evaluations of the effects of blowdown forces on the reactor internals and tests of the performance of the internals vent valves have been completed. The results of the analysis of the pressure-time history in the primary system following a LOCA and the resultant stresses and deflections in the reactor internals are reported in Reference 4 "Reactor Internals Stress and Deflection Due to a Loss of Coolant Accident and Maximum Hypothetical Earthquake." A similar investigation was performed to determine the stresses and deflections in the core and was submitted as Reference 5.

A full-sized prototype internals vent valve has been analyzed and experimentally tested to assure that the valve is structurally adequate to withstand hydraulic loadings and, subsequently, perform its steam venting function during a LOCA resulting from a postulated pipe rupture. The results of this investigation are discussed in Chapter 3, Reference 55.

1.5.6 FUEL ROD CLAD FAILURE

A study of clad failure mechanisms associated with a loss of coolant accident was performed. This study included identification of the potential failure mechanisms, a search of the literature to obtain applicable data, evaluation and application of existing data, and scoping tests to obtain data on potential failure mechanisms. The initial results of this study included the identification of the failure mechanisms, an evaluation of the information available in the literature concerning these mechanisms, and an evaluation of the effects of these mechanisms on the reactor system design.

The objective of the study was to assure that there are no potential failure mechanisms that might interfere with the ability of the emergency core cooling system to terminate the core temperature transient and remove decay heat in the event of a loss of coolant accident. These potential failure mechanisms include clad melting, zirconium-water reaction, eutectic formation between the Zircaloy clad and the Inconel 718 spacer grids, the possibility of clad embrittlement as a result of the quenching during core flooding, and clad perforation or deformation accompanying its failure. In the case of clad melting and zirconium-water reaction, our present design limit for peak clad temperature precludes these as possible failure modes. Information available in the literature, along with experimental evidence from tests conducted by B&W, shows that brittle fracture of the cladding will not occur as a result of quenching following a loss of coolant accident, and that eutectic formation between dissimilar core materials will not interfere with the flow of emergency core coolant after the accident. (Note: Intermediate spacer grid design for later fuel assemblies typically not dissimilar to clad, i.e. Zircaloy, etc.).

Preliminary tests showed that clad expansion is localized, and that any significant pressure is relieved by perforation at temperatures of the order of 1000 to 1400F. The force for continued expansion therefore is dissipated. Extensions of these preliminary tests evaluated the effects of other variables in order to verify the conclusion that coolant channels will remain sufficiently open to permit core cooling.

Data available indicated that the cladding deformation would be of a random nature and of small magnitude. The interpretation of these data leads to the conclusion that this phenomenon will not affect ECCS performance significantly. Thus, the testing was of a confirmatory nature to more specifically evaluate the effect of clad swelling on the fuel, and clad temperature during a LOCA. Completion of B&W's program provided confirmation that coolant channel restrictions due to clad swelling will not limit ECCS effectiveness. The results of this work were filed as Reference 6.

1.5.7 XENON OSCILLATIONS

This program was concerned with establishing the stability of the core and with evaluating the effects of part length control rods and burnable poison clusters on core stability. If mechanisms for control of diverging xenon oscillations are required, they were to be developed.

The xenon program consisted of the following:

a. Modal analysis

b. One dimensional digital analysis

c. Two- and three-dimensional digital analysis

The results of the modal analysis were submitted as Reference 7. The results of the one-dimensional digital analysis were submitted as Reference 8. The results of the two- and three-dimensional digital analyses were submitted as Reference 9.

As a result of the analysis the following conclusions were reached:

a. Diverging azimuthal or radial oscillations will not occur.

b. Diverging axial oscillations could occur but can be controlled with the Axial Power Shaping Rod Assemblies if they do.

The forgoing results are valid while the APSRs are in the core. However, current reload designs have been shown to be axially stable (Reference 16) and no longer include APSRAs.

For operation with APSRs removed, the following criterion applies:

Axial power oscillations induced by an axial xenon oscillation shall be naturally damped.

During the core reload or redesign analysis, a design xenon transient is simulated in accordance with the methodology of Reference 13. If the simulation shows that the criterion is not met, the result would be noted in the safety evaluation and regulating rods would be used to damp any induced xenon oscillations.

1.5.8 IODINE REMOVAL SPRAY

In 1982, sodium thiosulfate was deleted. Sodium hydroxide was used to perform the function of iodine scrubbing from a post-LOCA Reactor Building atmosphere and adjusting the Reactor Building long term sump pH. The modified system was designed to perform its iodine scrubbing functions to keep the radiation doses well within the 10CFR100 guidelines, and to raise the pH of the borated water to between 8.0 and 11.0 during the injection phase.

In 2007, trisodium phosphate replaced sodium hydroxide as the Reactor Building sump chemical buffer. The system description in UFSAR Section 6.2, and the Safety Analysis in UFSAR Section 14.2.2.5 reflect the current configuration.

Note:

In response to GSI 191, Trisodium phosphate replaced sodium hydroxide as the Reactor Building sump chemical buffer. The system description in UFSAR Section 6.2, and the Safety Analysis in UFSAR Section 14 (including Appendix 14B, Iodine Removal Capabilities of the TMI-1 Reactor Building Spray System), as well as other parts of the UFSAR have been updated to reflect the current configuration.

1.6 QUALITY ASSURANCE

1.6.1 INTRODUCTION

Metropolitan Edison Company (Met-Ed), as the original operator of Three Mile Island Unit 1, had the responsibility to assure that the unit was fabricated and constructed in accordance with applicable codes and specifications, and to assure that succeeding activities including testing, operating, refueling, modifying, maintaining, and repairing were conducted in accordance with quality assurance practices consistent with those employed during design and construction. Accordingly, Met-Ed had established a comprehensive quality assurance program for the design and construction phase of the project as described in Section 1.6.2. Further, as described in Chapters 12, 13, and Technical Specifications, Met-Ed had instituted administrative controls and quality assurance practices which would assure that succeeding activities were conducted in a controlled manner consistent with the quality assurance practices used during the design and construction phase of the project.

1.6.2 DESIGN AND CONSTRUCTION

1.6.2.1 General

Met-Ed had assigned the design and construction of the Three Mile Island Unit 1 Plant to GPU Service Corporation (GPUSC). GPUSC had established a comprehensive quality assurance program which was carried through all phases of equipment procurement, fabrication, erection, and construction. This program provided for review of specifications and/or associated purchase documents to ensure that necessary quality control requirements were included, and for quality assurance surveillance and auditing to assure that the specified requirements were met.

GPUSC had assigned to its Manager of Quality Assurance the duties of coordination and direction of all quality assurance measures for the plant. Design and construction work were administered by the GPUSC Project Manager, who was responsible for the technical direction and coordination of the nuclear steam supply system provided by B&W, the engineering efforts of the architect-engineer, GAI, and for the technical direction and coordination of all construction and site- related activities. Met-Ed's three main contractors: Babcock and Wilcox, Nuclear Power Generation Division (B&W-NPGD), Gilbert Associates Inc. (GAI); and, United Engineers and Constructors (UE&C), were responsible for developing all necessary quality requirements or procedures and ensuring that these requirements were observed during all phases of shop fabrication and site construction. B&W-NPGD was the Nuclear Steam Supply System supplier and was responsible for ensuring the quality of the equipment within its scope of supply.

The original OTSGs were replaced at the end of Cycle 17. AREVA NP Inc. supplied the replacement OTSGs and was responsible for the quality of the equipment within its scope of supply.

GAI was the Architect-Engineer for the project and was responsible for the quality assurance requirements for all of the areas outside B&W's scope of supply. UE&C was the Construction Manager and was responsible for ensuring that all quality control and inspection requirements were met for all site construction work which was governed by B&W-NPGD and GAI specifications; however, UE&C's Quality Control personnel were under the management direction of GPUSC.

1.6.2.2 Scope And General Approach

A three level quality assurance program applied to the nuclear related portions of the plant, i.e., the reactor core, the reactor coolant system, and its directly associated auxiliary systems, the containment system, the engineered safeguards, the fuel handling system, and the radioactive waste disposal system.

The Met-Ed quality assurance program started at the initial design phase with their three main contractors developing specifications and/or associated purchase documents which contained quality control and inspection requirements, and proceeded through the selection of the supplier, fabrication of the components or systems, and erection and installation.

Specific quality control requirements covered such areas as material and material control, welding requirements, cleanliness requirements, acceptance criteria, provisions for comprehensive auditing of manufacturer's or constructor's efforts, the preparation and retention of complete records, etc. B&W-NPGD and GAI, as Met-Ed's design contractors, were responsible for seeing that the necessary quality control requirements were included in all the component and construction specifications and/or associated purchase documents they prepared. Specifications and/or associated purchase documents were reviewed by MPR and/or by Met-Ed to assure that necessary quality control provisions were incorporated in these documents. Thus, specifications and associated requirement documents pertaining to the nuclear related portions of the plant received at least one independent review for evaluation of quality control requirements.

The selection of a component manufacturer or field erection and installation contractor was made only after it had been ascertained that his organization had the necessary design, manufacturing, and quality control capability and the qualified personnel to provide the level of integrity required for the equipment or construction involved.

A surveillance of the quality control programs of component manufacturers was performed by the B&W-NPGD Quality Assurance Group for components in the B&W scope of supply and by the GPUSC Vendor Surveillance Group for the remaining components, which were procured to GAI specifications. When UE&C subcontracted site work, UE&C personnel under GPUSC management performed a surveillance of the quality control and inspection programs of site construction subcontractors. In those cases where UE&C acted as a site construction contractor, UE&C quality personnel under GPUSC management performed quality control and inspection of UE&C's work. For such cases GAI performed quality assurance surveillance of the construction work and quality control. The surveillance of the component manufacturers' and site construction contractors' quality control programs was to assure that the work was proceeding in accordance with specification requirements.

In addition, to assure that Met-Ed's quality assurance program was functioning as desired, GPUSC and/or MPR audited, on a spot-check basis, the quality programs of B&W-NPGD, GAI, UE&C (including GPUSC management quality control) and their subcontractors and any other Met-Ed contractors involved with the nuclear portion of the plant.

In summary, this Quality Assurance Program can be termed a three level quality program. The first level of the program was quality control and inspection, the second level was quality assurance surveillance, and the third level was quality assurance auditing. Further definition and functions of these three levels are given in Subsection 1.6.2.3.

The GPUSC Project Manager and the GPUSC TMI Quality Assurance Manager, where appropriate, were advised of deficiencies found during fabrication or site construction and were authorized to initiate additional corrective action including the ordering of stoppage of work.

1.6.2.3 Organization And Definition Of The Three Levels Of The Met-Ed Quality Assurance Program

Figure 1.6-1 is the Quality Assurance Organization chart for the design and construction phase of the project showing the working relationships among Met-Ed, GPUSC, Met-Ed's major contractors, and GPUSC's consultants.

As indicated in Subsection 1.6.2.2, during the design and construction phase, the quality assurance program was divided into three levels. These levels are indicated on the left-hand side of Figure 1.6-1. The definition and functions of the three main levels indicated on Figure 1.6-1 were as follows:

a. First Level - Quality Control and Inspection

   The First Level, which was defined as a Quality Control and Inspection function, was performed by component manufacturers and construction contractors. The individual manufacturers and constructors were required by the applicable component, structural, or installation specifications, or appropriate purchase documents prepared by B&W-NPGD, and GAI to have a quality control and inspection system suitable for the end-product which they fabricate or construct. These documents required manufacturers and constructors to be responsible for production of the end-product and for the testing, inspection, and quality control to demonstrate that the final end-product had the specified degree of quality. Contractors in this category were referred to as "First Level Contractors." Examples of contractors in this category were: B&W Mount Vernon - reactor vessel, B&W Barberton - steam generator, Westinghouse - reactor coolant pumps, Chicago Bridge and Iron - containment building liner, Pittsburgh - Des Moines - reactor coolant bleed tanks, etc. When UE&C performed site construction work, UE&C QC engineers and inspectors performed the first level quality control and inspection work under the management direction of GPUSC.

b. Second Level - Quality Assurance Surveillance

   The Second Level, which was defined as a Quality Assurance Surveillance function, was performed by B&W-NPGD on the equipment in their scope of supply and by the GPUSC QA Engineering and Vendor Surveillance Groups for components built to GAI specifications. For site construction work not performed by UE&C, UE&C QC under GPUSC management performed surveillance on the site construction contractors . When UE&C acted as a site construction contractor, UE&C's construction effort was under the surveillance of GAI.

   The Quality Assurance groups who performed this second level function included personnel with technical backgrounds in the electrical, instrumentation and control, mechanical, welding, materials, and structural concrete fields, etc., as appropriate. These quality assurance groups had two major tasks. The first was during the design phase where the Quality Assurance Groups were responsible for assuring that the various specifications and drawings and/or associated purchase documents included applicable codes and quality control requirements. The second major task of the Quality Assurance Groups occurred during the actual fabrication and construction phase of the project. During this phase, these groups performed a quality assurance surveillance on the individual manufacturers' and site constructors' (first level contractors) Quality Control and Inspection Programs to ensure that the design and quality requirements were in fact being met.

c. Third Level - Quality Assurance Auditing

The Third Level, which was defined as a Quality Assurance Auditing function, was performed by GPUSC or its designated agent (e.g., MPR). The purpose of this function was to ensure that this Quality Assurance Program was functioning as planned. To accomplish this, GPUSC and/or its designated agent reviewed the specifications and other requirement documents furnished by the design contractors (B&W-NPGD and GAI) to check that the necessary quality requirements had been incorporated in these documents. In addition, the GPUSC Home Office, Site QA Auditor and/or GPUSC's designated agent performed, on a spot-check monitoring basis, quality assurance audits to ensure that the second level quality assurance surveillance programs of B&W-NPGD, GAI, UE&C and GPUSC and the first level quality control programs of manufacturers and constructors were actually functioning as required.

1.6.2.4 Implementation

Met-Ed had assigned the responsibility for establishing the TMI-1 Quality Assurance Program to GPUSC. The quality assurance program for TMI-1 was implemented under the direction of the GPUSC President, GPUSC Executive Vice-President and the GPUSC Design and Construction Division Vice-President. As discussed in Subsection 1.6.2.1, the GPUSC Manager of Quality Assurance, who reported to the Vice- President of the Design and Construction Division, had the responsibility for coordinating and managing the overall TMI-1 Quality Assurance Program for design and construction. Design and construction work was administered by the GPUSC Project Manager.

During the initial stages of the Three Mile Island Unit 1 project, the quality assurance effort was primarily directed towards developing and specifying the quality standards to be met by the equipment, systems, and structures in the nuclear related portions of the plant. These quality standards covered items such as requirements for non-destructive testing, material specifications, and proof tests. Also, the initial equipment and construction specifications, and related purchase documents contained quality control program requirements, such as those requiring the use of written test procedures. As work on the project progressed, Met-Ed recognized that it would be desirable to contractually invoke additional quality control program requirements of the type described in the April 1969 USAEC Quality Assurance Criteria. Accordingly, beginning in 1968, Met-Ed began to develop and apply quality control program requirements for equipment procurement and site construction work. Implementation of these requirements is described in the sections which follow in which the responsibilities of each of Met-Ed's contractors are described.

The overall responsibilities of the Met-Ed contractors in this quality assurance program can be summarized as follows:

a. Babcock and Wilcox

1) Introduction

The Babcock & Wilcox Company had a program of Quality Assurance for its nuclear steam supply systems. The Quality Assurance program included in its scope the design, fabrication, shipment, erection, and testing of B&W nuclear steam supply systems and fuel. The program was administered by a separate Quality Assurance organization whose responsibility was to assure the implementation of the program.

2) Program Elements

The B&W QA program included the following: organization; planning; design control; procurement control; manufacturing control; test control; and records and audits. These program elements as applied to the safety-related function of reactor coolant system and engineered safeguards components, were considered to meet the intent of the proposed Appendix B to 10CFR50, "Quality Assurance Criteria for Nuclear Power Plants."

3) Organization

The company executed its utility nuclear steam supply system contracts through the Power Generation Group which designed, manufactured, procured, erected, and serviced nuclear equipment including reactor cores. Within this division, the Nuclear Power Generation Division had the overall responsibility for systems and fuel contracts.

The B&W-NPGD Quality Assurance organization administered the B&W Quality Assurance program within the Power Generation Division with direct administrative responsibility to the Vice-President in charge of the Nuclear Power Generation Division. Figure 1.6-2 depicts the relationship among the organizations within the Power Generation Division. The B&W-NPGD Quality Assurance organization is shown in Figure 1.6-3. The NPGD-QA organization provided an independent audit of the quality programs for equipment manufactured by B&W as well as equipment purchased from suppliers.

4) Quality Assurance Program

a) Objective

The objectives of the B&W QA program were to (1) establish quality requirements, (2) document and communicate the established quality level, (3) provide or select manufacturing facilities that would assure achievement of the quality requirements level, (4) monitor the manufacturing as appropriate to assure that quality processes were being used, and (5) provide documentation to demonstrate that the quality level set had been achieved.

b) Planning

B&W implemented its QA program by use of standards and written procedures. Emphasis was placed upon establishment of an audit program and documentation requirements consistent with the nuclear safety related functions of the equipment.

c) Procurement

For equipment manufactured by B&W, B&W prepared both functional and equipment specifications. B&W performed functional and mechanical design and established the material requirements, manufacturing processes, and inspection and test requirements. For equipment purchased by B&W, B&W prepared equipment specifications. Suppliers prepared detailed designs. For selected equipment, a supplier certification audit was performed in accordance with written procedures prior to placement of a purchase order, or prior to release to manufacture.

d) Erection

B&W provided technical direction of erection for the equipment supplied by B&W. B&W requirements for storage, handling, and erection were furnished through an erection quality assurance manual.

e) Plant Test Program

B&W prepared test specifications incorporating B&W procedures and standards for the plant test programs for equipment and systems within the B&W scope of supply. B&W supplied advice and consultation during the test program and reviewed the test results.

5) Design Control

The B&W QA program incorporated provisions to ensure that limits imposed by the "design bases" and applicable regulatory requirements were correctly translated into specifications, drawings, procedures, and instructions.

Designs and drawings prepared for equipment manufactured by B&W and its suppliers were reviewed by B&W-NPGD for conformance to specifications and for compatibility with interface requirements.

Where design standards did not exist or where they were modified in design analysis, design reviews were conducted by personnel who did not perform the original design work.

6) Instrumentation and Control Equipment

The instrumentation and control equipment which must perform safety feature functions such as reactor protection, emergency core cooling, or reactor building isolation were monitored by approved quality control programs which included, as applicable and appropriate for the equipment involved, the following methods and procedures:

a) Control of Raw Materials

Including the procedures for the receiving inspection, identification, and certification of incoming raw materials.

b) Control of the Quality of Purchased Parts

Including inspection by the manufacturer and receipt inspection and testing by the supplier.

c) Control of Fabrication Processes

Including qualification of fabrication processes, control and checking of tools and fixtures, heat treatment procedures, cleanliness procedures, etc.

d) Control of Inspection and Test Equipment

Including calibration standards and recalibration frequency.

e) Control of Packaging and Shipping

Including final inspection releases, inspection of packages, and maintenance of cleanliness.

f) Control of Changes to Documents Affecting Quality

Including drawings, specifications, procedures, and other related documents.

g) Material Identification

Including means used to positively identify all material and to identify the inspection status of all material.

h) Disposition of NonConforming Items

Including repair, rework, and retest procedures and the steps taken to assure that nonconforming material was positively identified and not inadvertently used.

i) Control and Storage of Inspection and Test Records

Including their immediate availability for review by the Company and its Utility Contractor.

j) Customer Site Receiving Inspection

Including the procedures for site receiving inspection, identification, and storage.

k) Erection and Installation Inspection

Including a system for verifying the quality of erection and installation work.

l) Field Operations Audits

Including procedures for conducting audits of field operations to verify their adherence to approved procedures and processes.

b. Gilbert Associates, Inc.

As a design contractor for Met-Ed, GAI was responsible through their projects manager for developing equipment, material, and construction specifications for components and systems other than those supplied by B&W. The GAI project manager was assisted in this effort by the Manager, Quality Assurance Group, of GAI who, however, reported directly to higher management to ensure that he had sufficient organizational freedom to identify problems affecting quality and to insure that solutions were obtained. However, the GAI project manager was responsible for coordinating and planning and scheduling of the efforts of GAI's Quality Assurance Group with other organizations and contractors involved in the TMI-1 project. For those cases where UE&C acted as a site construction contractor rather than as a site construction manager, GAI conducted quality assurance surveillance on such UE&C efforts. Thus, this assured that all site construction efforts were subjected to at least two quality programs. In addition, GAI maintained records of their surveillance effort on site construction items where UE&C was a site constructor. This file was transferred to Met-Ed at completion of the project.

The major design organizations involved in the seismic design of the unit were GAI and B&W. Subordinate to these were their consultants and equipment suppliers. Metropolitan Edison Company and their consultant, MPR Associates, reviewed assignments of seismic classification to structures, components, and systems and the detailing of seismic design criteria in specifications produced by the major design organizations.

GAI, through its consultant, Weston Geophysical Research, Inc., was responsible for reviewing the site seismicity and establishing the ground response spectra described in Section 2.8.

GAI and B&W were responsible for establishing the classification of structures, systems and components within their respective engineering scopes. In the case of GAI, the Department Head and Project Engineer (for the engineering disciplines involved in the design of the unit) jointly established the classification of structures, systems, and components that were in their area of responsibility. Subsequently, the listings developed by all departments were combined with the combined list and reviewed by the other engineering disciplines involved in the design of the unit to establish that the list was all-inclusive and that the classifications were properly assigned. B&W internally established classifications in a basically similar manner.

The equipment specifications developed by the B&W or GAI engineering disciplines were reviewed by both Met-Ed and MPR personnel and the final requirements were mutually resolved. GAI specifications were also initially reviewed in-house by the GAI Quality Assurance Group.

GAI was responsible for the seismic analyses of all safety related structures for one unit and developed the relevant response data which was needed for the design of the components and systems supported in these structures. Seismic response data for the relevant structures were distributed to B&W and the various GAI engineering disciplines involved in the design by letters and internal memoranda, respectively. The B&W and GAI engineering disciplines then utilized this data in developing the seismic design criteria to be met by the designers and manufacturers of equipment and systems where the actual seismic analysis was to be performed by others. Likewise, the structural response data were used directly for those systems and components which were designed in-house by GAI and B&W.

Seismic design criteria included in the specifications for equipment or systems required the supplier to perform a seismic analysis or test. A list of the specific methods employed for the various equipment items is listed in Table 5.4-1. The equipment design reports, when required by the applicable specifications, were reviewed for compliance with specification and code requirements by the engineering discipline within the relevant organization responsible for preparing the equipment specification.

GAI performed the seismic analysis of the primary coolant system to allow consideration of building interaction with the reactor coolant system. The methods employed and partial results are described in Reference 11. This was done in order to expedite development of loading information required for the design of foundations, other structures, components, and balance of plant systems. B&W supplied the analytical models for all major components and the ordinary loop piping. Each mass model was independently checked, using the basic response spectra and GAI supplied foundation spring constants, to be assured that the component was properly modeled so as not to mask out natural frequencies. Also, complete system arrangement drawings along with sufficient detailed information was provided to permit an independent check by GAI. All analytical results of the GAI analysis of the primary loop were provided to B&W for inclusion in the equipment design specifications as required by the applicable codes.

The adequacy of the GAI dynamic analysis for use in the seismic design of the reactor coolant system was checked by B&W. This was accomplished by a review by B&W of the GAI report of the dynamic analysis for accuracy of input information, validity of the dynamic model used, and the validity of the use of the GAI computer code for evaluation of conservative seismic loadings to be applied to the reactor coolant system components. Also, B&W was able to compare the results of the GAI dynamic analysis with the results of an independent dynamic analysis of the reactor coolant system. The individual reactor coolant system components were analyzed for the seismic loadings developed by the GAI dynamic analysis.

The interchange of required design information, internally between engineering disciplines or between companies, was usually made by means of internal memoranda or design reports, respectively. Engineering information transmitted for design review was usually in the form of preliminary specification, which may have been accompanied by design sketches or engineering drawings. When final design information was to be transmitted by letter form, the letter was properly documented and referenced in the design reports or in the Quality Assurance files of the procurement agency. Design information received by B&W for design of the Nuclear Steam Supply System was tabulated and included in a foundation and component loading specification which, in turn, became a part of the Equipment Design Specification. Design information received by GAI (e.g., seismic calculations) from its suppliers became a permanent part of Met-Ed's records for the Unit. For both B&W and GAI, the engineering discipline responsible for the equipment specification was responsible for the review of the design reports for compliance with the specification.

The cognizant GAI Project Engineer was responsible for monitoring the execution of seismic design routines for GAI designed structures, piping systems, ducts, cable trays, etc., and maintained documentation of this effort. An independent review was made of all such design within GAI which, basically, verified the design assumptions, input, analysis and reasonableness of results. A certification was issued to the effect that a review was completed for each item listed as a Class I structure or system. In essence, the certification was a statement to the effect that the design review was conducted and indicated that the design met the intent of the seismic design criteria. The GAI Project Engineer forwarded in-house certifications of structural, piping system, etc., seismic calculations, together with attachments, to the GAI Project Manager, who maintained files of these for the projects.

1) Preparation of Specifications and Drawings

GAI was responsible for preparing all equipment specifications except for the nuclear components supplied by B&W. GAI was also responsible for preparing site construction specifications and drawings. These documents and drawings incorporate or reference the applicable design requirements, safety criteria, and quality control program requirements. The GAI specifications included formal quality control program requirements for piping and site construction.

The quality control requirements for the plant piping required a quality control program which covered at least:

a) Control of raw materials

b) Control of the quality of purchased parts

c) Control of fabrication processes

d) Control of inspection and test equipment

e) Control of packaging and shipping

f) Control of changes to documents affecting quality

g) Material control and identification

h) Disposition of nonconforming material

i) Control and storage of inspection and test records

In addition, Met-Ed had assured, by review of the piping supplier's procedures, and by audits that items such as organization, planning, program documentation, control of inspection status, etc., were satisfactorily defined.

Quality control system requirements for site construction work were developed and strengthened as the project progressed. In late 1968 a requirement for a complete quality control program, which was consistent with the draft USAEC Quality Assurance Criteria of April 17, 1969, was placed on UE&C for nuclear related site construction work. Similar, but somewhat more limited, quality control programs were used by the major subcontractors in the nuclear area, for the Reactor Building liner and the Reactor Building tendons.

2) Equipment Procurement

With the exception of the nuclear steam supply system, all equipment was procured to requirements established by GAI. This equipment was procured either by Met-Ed or UE&C. When procured by Met-Ed, GAI evaluated possible manufacturers and provided Met-Ed with a proposed bidders' list. Quality assurance surveillance of such contracts was performed by GPUSC. Where required by applicable specifications, GAI approved drawings, performance procedures, etc., prepared by the manufacturers. Review of quality assurance documents such as quality control manuals, welding procedures, etc., prepared by these manufacturers was performed by the GPUSC Quality Assurance Group. However, the GAI Quality Assurance Group also reviewed these documents when requested by GPUSC (e.g., Grinnell quality control documents for piping fabrication). Also, the GAI Quality Assurance Group provided assistance to GPUSC upon request, particularly in the area of specification interpretation.

3) Site Construction

In certain cases UE&C acted as a site construction contractor rather than as site construction manager. In such cases, GAI conducted a quality assurance surveillance on UE&C's site construction work. This site construction surveillance was the responsibility of GAI's Quality Assurance Manager. The GAI Quality Assurance Manager was responsible for establishing the quality assurance procedures that GAI used in the field. The GAI Quality Assurance Representative at the construction site, reporting directly to the GAI Quality Assurance Group in the home office, was responsible for monitoring the materials used, the equipment installation, UE&C's construction and inspection procedures, and all other phases of UE&C's work which was related to quality control. Any deviations from the applicable codes, specifications, and quality control requirements were reported immediately to the GPUSC Field Supervisor-Quality Control. If immediate action to correct such deviations was not taken, GAI reported the deviation to the GPUSC TMI Quality Assurance Manager. The GPUSC TMI Quality Assurance Manager would stop the work until such time as corrective action was taken.

c. United Engineers and Constructors

UE&C had two roles in the Three Mile Island Unit 1 Project. One role was when it was acting as the Project Construction Manager and as such was responsible for managing, planning, coordinating, and monitoring the activities of the various site construction contractors (see Case A on Figure 1.6-1). The second UE&C role was when it acted as a site construction contractor rather than as construction manager (see Case B on Figure 1.6-1).

1) UE&C as Construction Manager

In its role as Construction Manager, UE&C was responsible, through its Project Manager, for ensuring that the quality requirements in GAI and B&W-NPGD specifications were met for all site construction activities. The UE&C Project Manager was assisted in this effort by UE&C quality control personnel (under the management direction of GPUSC) who performed quality assurance surveillance of site contractors. The UE&C procedures which indicated specifically how UE&C implemented its quality assurance surveillance responsibilities required approval by the GPUSC TMI Quality Assurance Manager.

United Engineers and Constructors also maintained a quality control file for the entire project including the quality control file provided by B&W for equipment in B&W's scope of supply. Throughout the course of construction, these records were available for review by GPUSC, regulatory agencies and/or their authorized representatives. At the completion of the project, these records were turned over to Met-Ed.

UE&C's task as the Construction Manager during the equipment procurement and site construction phases were:

a) Equipment Procurement

UE&C acted as Met-Ed's agent in expediting equipment procured to GAI specifications. UE&C also transmitted manufacturer's quality control documents for review by the GPUSC QA engineering staff; UE&C was responsible to assure that vendor documents were acceptable to GPUSC QA engineering prior to use.

b) Site Construction

In its capacity as Construction Manager, UE&C was responsible for administration of construction activities of the various site construction contractors. UE&C quality personnel performed quality assurance surveillance of the site construction contractors under the direction of the GPUSC Field Supervisor-Quality Control. This included quality assurance surveillance of the materials used, the equipment installation, and all phases of the work by the site contractors. Any deviations from the applicable codes, specifications, procedures, and quality control requirements were reported immediately to UE&C site management for correction and to the GPUSC Field Supervisor-Quality Control.

Site construction contractors were selected by UE&C after they had made a thorough evaluation of the contractor's capability to perform the necessary work to the desired level of integrity and determined that he had the necessary experience and an adequate quality control system. The site construction and quality control procedures to be used by the site contractors were reviewed and approved by UE&C, GPUSC and others as appropriate. UE&C also required that all site contractors maintain a complete record of the work they performed. These records were included in the quality control history file for the entire project.

2) UE&C as a Site Construction Contractor

For those instances where UE&C actually performed as a site construction contractor (see Case B on the Quality Assurance Organization Chart, Figure 1.6-1) the UE&C quality control and inspection program for the items they installed, constructed, or procured for their site construction effort was under the management direction of GPUSC, and was subject to the surveillance of GAI (see Item b of this Subsection). As a site constructor, UE&C was required to perform their construction work to written procedures as required by GAI installation and construction specifications. Where appropriate, such UE&C procedures were reviewed by B&W-NPGD, GAI, GPUSC and/or MPR.

d. MPR Associates, Inc.

Met-Ed had retained MPR Associates, Inc. to assist GPUSC in monitoring the overall quality assurance program for the nuclear related portions of the plant. It should be noted that this assistance was in addition to the quality functions that were performed by Met-Ed's main contractors and their subcontractors. Specifically, MPR performed the following, on a spot-check monitoring basis:

1) Review of specifications, drawings, procedures, inspection check lists, and other pertinent documents to assure that the quality control and test requirements for components, systems, and structures were adequate.

2) Periodic quality assurance audits at manufacturer's plants and at the construction site to evaluate the performance of Met-Ed contractors and subcontractors in implementing the specified quality control and testing requirements and maintaining necessary records.

3) Witness, on a spot-check basis, tests on components and systems. Further, upon request, evaluated proposed waivers to specifications to determine their effects on quality.

In the performance of the above work, MPR advised the GPUSC Manager of Quality Assurance or Site Quality Assurance Auditor, as appropriate, of any deviations or deficiencies found during fabrication or site construction.

1.6.2.5 <u>Conclusions</u>

This comprehensive quality assurance program provided reasonable assurance that TMI Unit 1 was being built with an appropriate level of quality and integrity.

1.6.3 QUALITY ASSURANCE TOPICAL REPORT

For details of the operational quality assurance program, see the latest revision of the Quality Assurance Topical Report.

1.7 IDENTIFICATION OF AGENTS AND CONTRACTORS

(Following are the primary Organizations during the construction and licensing stages)

Metropolitan Edison Company had the responsibility to interested regulatory agencies for the engineering, design, purchasing, construction, pre-operational testing, operation, and maintenance of Three Mile Island Nuclear Station Unit 1.

The General Public Utilities Service Corporation had acted as the agent for Metropolitan Edison Company and its co-owners, and was responsible for the engineering, design, construction, quality assurance, and pre-operational testing of Three Mile Island Unit 1.

Gilbert Associates, Inc. had been retained by Metropolitan Edison Company as Architect-Engineer for the project. Gilbert Associates had drawn up specifications for all major equipment and systems, furnished plant layouts and system arrangements, assembled information required for site studies and plot plans, and cooperated with Metropolitan Edison in the evaluation of bids.

United Engineers and Constructors had been retained by Metropolitan Edison as Construction Manager for the plant. United Engineers and Constructors was responsible for the supervision and coordination of construction of Three Mile Island Nuclear Station Unit 1.

Babcock & Wilcox had contracted with Metropolitan Edison to design, manufacture and deliver to the site the complete nuclear steam supply system. Babcock & Wilcox was responsible for the erection consultation and their equipment, and also competent technical and professional coordination required during fuel loading and initial startup of the NSSS.

Pickard, Lowe & Associates had been retained as Nuclear Consultants to aid in the preparation of reports and studies, and furnish guidance to Metropolitan Edison in nuclear related matters associated with the securing of permits for the project.

Sheppard T. Powell and Associates had been retained as Water Chemistry Consultants.

MPR Associates, Inc. had been retained as consultants for Quality Control. Their specific Quality Control responsibilities are described in Item d of Section 1.6.2.4.

Weston Geophysical Research, Inc. was engaged to perform a seismicity study and to develop response spectra for the site. Mr. Richard J. Holt administered this work. Rev. D. Linghan, S.J., directed the seismicity analysis and Professor Robert V. Whitman of Massachusetts Institute of Technology developed the response spectra.

Dr. Marvin E. Kauffman, of Franklin and Marshall College, was engaged to research the regional structural, geologic, and tectonic aspects of the site.

Metropolitan Edison collected the meteorological data at the site. Pickard, Lowe & Associates performed the analysis of the data with consultations from Dr. J. Holitsky of New York University.

General Electric Corp. - Turbine Generator Supplier

General Electric Corporation was responsible for design and delivery of the turbine generator for the project. This responsibility included the provision of criteria, plans and guidelines required by Gilbert Associates for design of turbine support systems, and by United Engineers and Constructors Corporation for preparation of installation procedures and proper installation of the Turbine Generator.

Dr. G. Hoyt Whipple, served as consultant on the Environmental Monitoring Program being performed at Three Mile Island.

Wald, Spritzer Associates had been retained as Medical Radiation Consultants.

Radiation Management Corporation served as consultants in the areas of control of potential accidents involving radiation and also provided transportation facilities for the rapid transit of persons requiring extensive medical care as a result of these potential accidents.

1.8 CONCLUSIONS

The personnel assembled to design, construct, and operate Three Mile Island Nuclear Station Unit 1 are capable of performing their required project function. The health and safety of the public and station operating personnel and reliability of equipment and systems have been among primary considerations in the design and construction of the unit.

The reactor system as installed is a practical design of proven type which will not require fuel exposures or operating conditions beyond those presently achievable with the materials used in construction. The shutdown margin and performance characteristics are comparable to those used in existing reactors. The reactor system is enclosed in a conservatively designed structure which is capable of providing shielding and containment following the worst postulated accident condition. The containment capability of the structure is supplemented by the engineered safeguards which operate to remove energy and fission products from the contained atmosphere.

The radioactive waste systems are designed to process all waste prior to disposal as permitted by Nuclear Regulatory Commission regulations.

In consideration of the above circumstances and the station design as presented herein, it is concluded that the Three Mile Island Nuclear Station Unit 1 can be operated in a safe manner and that the design will provide adequate protection to the public from any natural or mechanical events resulting in disablement of equipment.

1.9 REFERENCES

1. Once Through Steam Generator Research and Development Report, B&W Topical Report BAW-10027.

2. Control Rod Drive Mechanism Test Program, B&W Topical Report BAW-10029, Rev. 1.

3. Reactor Vessel Model Flow Tests, B&W Topical Report BAW-10037, Rev. 2.

4. Reactor Internals Stress and Deflection Due to a Loss-of-Coolant Accident and Maximum Hypothetical Earthquake, B&W Topical Report BAW-10008, Part 1, Rev 1.

5. Fuel Assembly Stress and Deflection Analysis For Loss-of-Coolant Accident and Seismic Excitation, B&W Topical Report BAW-10035, Rev. 1.

6. Effect of Fuel Rod Failure on Emergency Core Cooling Effectiveness, B&W Topical Report BAW-10009.

7. Stability Margin for Xenon Oscillations - Model Analysis, B&W Topical Report BAW-10010, Part 1.

8. Stability Margin for Xenon Oscillations - One Dimensional Digital Analysis, B&W Topical Report BAW-10010, Part 2.

9. Stability Margin for Xenon Oscillations - Two and Three Dimensional Digital Analyses, B&W Topical Report BAW-10010, Part 3, Rev. 1.

10. Deleted.

11. GAI Topical Report 1729.

12. Incore Instrumentation Test Program, B&W Topical Report BAW-10001.

13. Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, B&W Fuel Company Topical Report BAW-10179P-A. (The current revision level shall be specified in the Core Operating Limits Report)

14. Deleted.

15. Deleted.

16. AREVA Document 51-5068286-00, “Reduced APSR Evaluation for TMI-1,” September 20, 2005

17. ECR 06-00935, Rev. 2, "Steam Generator Component ECR."

18. ECR 07-00576, Rev. 2, "Hot Leg Piping and RTD Replacement."