CHAPTER 1: INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1. INTRODUCTION

The original Final Safety Analysis Report (FSAR) was submitted in support of the Detroit Edison Company's (Edison) application for a license to operate a 3293-MWt (rated) nuclear power plant at the Enrico Fermi Atomic Power Plant site on the western shore of Lake Erie, at Lagoona Beach, Monroe County, Michigan. This Updated Final Safety Analysis Report (UFSAR) was prepared in response to 10 CFR 50.71(e).

The power plant is designated as Fermi 2. The Fermi 2 PSAR (CP Application) was filed in April 1969 and a construction permit CPPR-87 was issued in September 1972. The original FSAR was filed in April 1975. The plant received its license for fuel loading and low-power testing (5 percent power) on March 20, 1985, and its full-power operating license on July 15, 1985.

Fermi 2 uses a General Electric Company (GE) single-cycle, forced-circulation BWR of the BWR 4 Class, with a pressure-suppression Mark I containment. Fermi 2 is similar in design to these nuclear power plants: Browns Ferry Nuclear Plant Units 1, 2, and 3; Cooper Nuclear Station; Edwin I. Hatch Unit No. 1; and Brunswick Steam Electric Plant Units 1 and 2. The design power rating (emergency core cooling system [ECCS] design basis) for Fermi 2 is 3486 MWt, with a turbine-generator design gross electrical output at the generator terminals of 1235 MWe and a net electrical output of 1170 MWe.

On September 9, 1992, the NRC issued Amendment 87 to the Fermi 2 operating license authorizing a change in the thermal power limit from 3293 MWt to 3430 MWt, a 4.2 percent increase in the thermal power and a 5 percent increase in steam flow. This changed the net electrical capacity from 1093 MWe to 1139 MWe, or an increase of 46 MWe.

During RF05 the LP Steam Path was replaced by a GE designed LP Steam Path with a higher efficiency. This changed the designed net electrical capacity from 1139 MWe to 1150 MWe, or an increase of 11 MWe.

During RF07 the HP Steam Path was replaced by a GE designed HP Steam Path with a higher efficiency. However, the gross generator output will not exceed the present 1217 MWe.

During RF11, the Moisture Separator Reheaters (MSRs) were replaced. The gross generator output will not exceed MWe noted above.

The Fermi Power Uprate Program followed the GE Nuclear Energy generic guidelines and evaluations for BWR power plants.[1,2]

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[1] GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31897P-1, Class III, (Proprietary), June 1991

[2] GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Volumes I and II, Class III, (Proprietary), July 1991.

On February 10, 2014, the NRC issued Amendment 196 to the Fermi 2 operating license authorizing a change in the thermal power limit from 3430 MWt to 3486 MWt, a 1.64 percent increase in thermal power and a 1.88 percent increase in steam flow. This changed the net electrical capacity from 1150 MWe to approximately 1170 MWe. This power uprate was performed in accordance with 10 CFR 50, Appendix K and reflects the improvement in feedwater flow measurement. The Fermi 2 Measurement Uncertainty Recapture (MUR) power uprate followed the GE generic guidelines and evaluations for BWR plants provided in GEH Topical Report NEDC-32938P-A, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," Revision 2, May 2003.

Fermi 2 specific analyses and evaluations were performed, consistent with the generic guidelines, for systems and components that might be affected to ensure their capability to support the increase in power output and steam flow. Since data is described in detail in the UFSAR, revisions were made to this data to reflect the power uprates, as appropriate. The analyses and evaluations resulted in determinations that the systems and components were either not affected by power uprate or had sufficient design capacity to accommodate uprate conditions.

In addition to the above, the effect of the uprates on the environment was assessed to verify that operation of Fermi 2 at uprated power was environmentally acceptable with established NRC requirements and that consistency was maintained with Federal, State, and local regulations. As a result, no changes to the Environmental Protection Plan or to any of the non-NRC permits are required.

The Detroit Edison Company changed its name to DTE Electric Company as of January 1, 2013. The name change to DTE Electric Company was purely administrative in nature; the legal entity remained the same and the name change did not involve a transfer of control or of an interest in the license for Fermi 2. DTE Electric Company continues to be a wholly owned subsidiary of DTE Energy Company. For the purposes of the Fermi 2 UFSAR, except for UFSAR sections of historical context, all DTE Energy Company designations referenced throughout the UFSAR (e.g. DTE Electric, Edison, Detroit Edison, DECo, etc.) are synonymous.

DTE Electric submitted an application for renewal of the operating license for an additional 20 years on April 24, 2014 by letter NRC-14-0028. The application documented the technical and environmental reviews performed to support extension of the license to March 20, 2045. The NRC performed an in-depth review, including audits, an inspection, and multiple requests for additional information. The NRC issued the final Safety Evaluation Report on the License Renewal of Fermi 2 on July 12, 2016. The Safety Evaluation Report was re-issued as NUREG-2210 in October 2016. NUREG-1437, Supplement 56, the Generic Environmental Impact Statement for License Renewal of Nuclear Plants Regarding Fermi 2 Nuclear Power Plant, was published in September 2016.

Appendix A of the License Renewal Application (LRA) included a supplement to be inserted into the UFSAR following approval of the renewed license. That appendix, including changes submitted in response to NRC requests for additional information, is added to the UFSAR as Appendix B. The appendix addresses the aging management programs that will be implemented per the commitments in the License Renewal Application, a summary of how time limited aging analyses were addressed, and a list of commitments made in the

LRA. Changes to Appendix B may be made per the process for UFSAR revisions under the auspices of 10 CFR 50.59.

The renewed license was issued December 15, 2016.

1.2 GENERAL PLANT DESCRIPTION

1.2.1 General Design Criteria

The general architectural and engineering criteria for the design, construction, and operation of Fermi 2 are summarized in this subsection. For specific NRC General Design Criteria (GDC) conformance description, see Section 3.1.

The discussion of the GDC that follows is divided into three sections. First, the overall requirements criteria are presented for the plant and for the nuclear safety systems and engineered safety features (ESFs). Then the GDC are presented in two ways. First, the criteria are considered in a classification-by-classification approach. Second, the criteria are considered in a system-by-system or system group approach.

1.2.1.1 Overall Requirements Criteria

1.2.1.1.1 Plant Criteria

The plant is designed, fabricated, erected, and operated to generate electricity in a safe and reliable manner. Plant design conforms with applicable codes and regulations and complies with regulatory guides to the extent described in Appendix A.

The plant is also designed, fabricated, erected, and operated in such a way that the release of radioactive materials to the environment is less than the limits of 10 CFR 20 and 10 CFR 50, pertaining to the release of radioactive materials, during normal operation and abnormal events.

Components and structures are provided with appropriate safety factors and adequate strength and stiffness so that a hazardous release of radioactive material will not occur.

Careful consideration is given to all known environmental conditions associated with maintenance, testing, and postulated accidents, including LOCAs, that could result in unplanned releases of radioactive material from the plant. Pollution control equipment and specific design provisions are incorporated in the plant for the specific purpose of protecting public health and safety from the release of radioactive material under both normal and abnormal conditions.

1.2.1.1.2 Nuclear Safety Systems and Engineered Safety Features Criteria

Design margins for the nuclear safety systems and ESFs are conservative.

Nuclear safety systems are designed to respond to abnormal operational transients to limit fuel damage so that, should the freed fission products be released to the environs via the normal discharge paths for radioactive material, the limits of 10 CFR 20 and 10 CFR 50 will not be exceeded.

Nuclear safety systems and ESFs act to preclude damage to the nuclear system process barrier as a result of internal pressures caused by abnormal operational transients or accidents.

When positive and precise action is immediately required in response to accidents, such action is automatic, requiring no decision or manipulation of controls by plant operations personnel.

The reactor core and reactivity control systems are designed so that the control rod action is capable of making the core subcritical and maintaining it so, even when the rod of highest worth is fully withdrawn and unavailable for reinsertion.

Essential safety actions are carried out by equipment in sufficient redundance and independence so that a single failure of active components will not prevent the required actions.

Provision has been made for control of active components of nuclear safety systems and ESFs from the main control room.

Nuclear safety systems and ESFs are designed to permit demonstration of their compliance with functional performance requirements.

Nuclear safety systems and ESFs are designed to maintain operability under all plant-related and site-related events (e.g., earthquakes, tornadoes, floods, fires, etc.).

Features of the plant essential to the mitigation of accident consequences are designed for fabrication and erection to quality standards that reflect the importance of the safety function to be performed. A quality assurance program has been established and implemented.

1.2.1.2 Classification-by-Classification Approach

In this approach, three classifications are considered: (1) power generation; (2) safety; and (3) plant radiation zones. The corresponding GDC are discussed below.

1.2.1.2.1 Power Generation Classification Criteria

The GDC for the power generation classification are further subdivided into criteria for planned operations and for operational transients.

1.2.1.2.1.1 Planned Operations

Power generation design criteria for planned operations are as follows:

a. Fuel cladding is designed to retain integrity as a radioactive material barrier throughout the design power range. The fuel cladding accommodates, without loss of integrity, the pressures generated by fission gases released from fuel material throughout the design life of the fuel

b. Heat removal systems are provided in sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions from plant shutdown to design power. The capacity of such systems is adequate to prevent fuel cladding damage

c. Control equipment is provided to allow the reactor to respond to small load changes

d. Reactor power level is manually controllable

e. Control of the nuclear system is possible from a single location

f. Nuclear system process controls, including alarms, are arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions

g. Fuel handling and storage facilities are designed to maintain adequate subcriticality, shielding, and cooling for spent fuel

h. Interlocks or other automatic equipment are provided as backup to procedural controls to avoid conditions requiring unnecessary functioning of nuclear safety systems or ESFs

1.2.1.2.1.2 Operational Transients

Power generation design criteria for operational transients are as follows:

a. The fuel cladding, in conjunction with other plant systems, is designed to retain integrity throughout any abnormal operational transient

b. Heat removal systems are provided in sufficient capacity and operational adequacy to remove heat generated in the reactor core for any abnormal operational transient. The capacity of such systems is adequate to prevent fuel cladding damage

c. Control equipment is provided to allow the reactor to respond automatically to normal operational transients, such as major load changes, and to abnormal operational transients, including bringing the reactor to a hot-shutdown condition when appropriate

d. Backup heat removal systems are provided to remove decay heat generated in the core when the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage

e. Onsite standby electrical power sources are provided to allow removal of decay heat when normal offsite auxiliary power is not available.

1.2.1.2.2 Safety Classification Criteria

The design criteria for the safety classification are further subdivided into criteria for planned operations, operational transients, and accidents.

1.2.1.2.2.1 Planned Operations

Safety design criteria for planned operations are as follows:

a. The plant is designed, fabricated, erected, and operated in such a way that the normal release of radioactive materials to the environment is within the requirements of 10 CFR 20 and 10 CFR 50

b. The reactor core is designed so that its nuclear characteristics do not contribute to a divergent power transient

c. The nuclear system is designed such that there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate plant systems

d. Gaseous, liquid, and solid waste disposal facilities are designed such that the discharge and offsite shipment of radioactive effluents are in accordance with applicable federal, state, and local regulations

e. The design provides a means by which plant operators are informed when limits on the release of radioactive material are approached

f. Sufficient indication is provided to allow determination that the reactor is operating within the range of conditions considered in the plant safety analysis

g. Radiation shielding and access control procedures are provided to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal plant operation

h. Procedures for fuel handling and design of fuel storage facilities prevent inadvertent criticality.

1.2.1.2.2.2 Operational Transients

Safety design criteria for operational transients are as follows:

a. The plant is designed, fabricated, erected, and will be operated in such a way that the release of radioactive materials to the environment is within the requirements of 10 CFR 20 and 10 CFR 50

b. Those portions of the nuclear system that form part of the nuclear system process barrier are designed to retain integrity as a radioactive-material barrier following abnormal operational transients

c. Nuclear safety systems act to ensure that no damage to the nuclear system process barrier results from internal pressures caused by abnormal operational transients

d. When positive and precise action is immediately required in response to abnormal operational transients, such action is automatic, requiring no decision or manipulation of controls by plant operations personnel

e. Essential safety actions are carried out by equipment of sufficient redundancy and independence that a single failure of any active component cannot prevent the required actions

f. Provision is made for control of the active components of nuclear safety systems from the main control room

g. Nuclear safety systems are designed to demonstrate their functional performance requirements

h. Nuclear safety systems are designed to maintain their function under all plant-related and site-related events (e.g., earthquakes, floods, tornadoes, and fires)

i. Standby electrical power sources have sufficient capacity to power all nuclear safety systems requiring electrical power

j. Onsite standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal offsite auxiliary power is not available.

1.2.1.2.2.3 Accidents

Safety design criteria for accidents are as follows:

Fermi 2 has reanalyzed the DBA-LOCA, the control rod drop accident, and the fuel handling accidents in accordance with the methodology in Regulatory Guide 1.183. The release of radioactive materials to the environment is evaluated per the criteria of 10 CFR 50.67 for these accidents only. All other existing accidents are evaluated per the criteria in 10 CFR 100.

a. The plant is designed, fabricated, erected, and will be operated in such a way that the release of radioactive materials to the environment is within the requirements of 10 CFR 100 or 10 CFR 50.67, as applicable

b. Those portions of the nuclear system that form part of the nuclear system process barrier are designed to retain integrity as a radioactive material barrier following accidents. For accidents in which one breach in the nuclear system process barrier is postulated, such a breach does not propagate additional failures in the nuclear system process barrier

c. The ESFs act to ensure that no damage to the nuclear system process barrier results from internal pressures caused by an accident

d. When positive, precise action is immediately required in response to accidents, such action is automatic, requiring no decision or manipulation of controls by plant operating personnel

e. Essential safety actions are carried out by equipment of sufficient redundance and independence that a single failure of any active component cannot prevent the required actions

f. Provision is made for control of active components of the ESFs from the main control room

g. The ESFs are designed to permit demonstration of their functional performance requirements

h. The ESFs are designed to maintain their function under all plant-related and site-related events (e.g., earthquakes, floods, tornadoes, fires, etc.)

i. Onsite standby electrical power sources have sufficient capacity to power the nuclear safety systems and ESFs requiring electrical power during accident conditions

j. Features of the plant essential to the mitigation of accident consequences are designed to be fabricated and erected to quality standards that reflect the importance of the safety actions to be performed

k. The primary containment is designed to retain integrity as a radioactive material barrier during and following accidents that release radioactive material into the primary containment volume

l. The primary containment is designed to permit integrity and leaktightness testing at periodic intervals

m. A secondary barrier (containment) is provided that completely encloses both the primary containment and the fuel storage areas. The secondary barrier design incorporates systems and equipment for controlling the rate of release of radioactive materials from the barrier, and further includes a capability for filtering radioactive materials within the barrier. In the event of a design-basis tornado, the secondary containment barrier above the refueling floor will be breached. See Section 3.3 for additional discussion regarding tornado design

n. The secondary barrier is designed to act as a radioactive material barrier under the same conditions that require the primary containment to act as a radioactive material barrier

o. The secondary barrier is designed to act as a radioactive material barrier, if required, when the primary containment is open for expected operational purposes

p. The primary containment and secondary containment barrier constitute pollution control facilities which, in conjunction with other ESFs, limit radiological effects of accidents resulting in the release of radioactive material to the containment volumes to within the 10 CFR 100 limits or 10 CFR 50.67 limits, as applicable

q. Provisions are made for removing energy from within the primary containment, as necessary, to maintain the integrity of the containment system following accidents that release energy to the primary containment so as to ensure continuing air pollution control functional capability

r. Piping that penetrates the primary containment structure, and which could serve as a path for the uncontrolled release of radioactive material to the environs, is automatically isolated whenever such uncontrolled radioactive material release is threatened. Such isolation is accomplished in time to limit radiological effects to within the 10 CFR l00 limits or 10 CFR 50.67 limits, as applicable

s. The ECCS is provided to limit fuel cladding temperature to 2200 F as a result of a LOCA

t. The ECCS provides for continuity of core cooling over the complete range of postulated break sizes in the nuclear system process barrier in order to minimize the release of radioactive material and to ensure the continuous functional capability of the containment facilities

u. The ECCS is diverse, reliable, and redundant

v. Operation of the ECCS is initiated automatically when required, regardless of the availability of offsite power supplies and the normal generating system of the plant

w. The main control room is shielded against radiation so that occupancy under accident conditions is possible

x. For a special event such as loss of habitability of the main control room, it is possible to bring the reactor from power range operation to a hot-shutdown condition, from outside the main control room, as well as to bring the reactor to a cold-shutdown condition from the hot-shutdown condition

y. For a special event, such as inability to shut down the reactor with control rods, backup reactor shutdown capability is provided, independent of normal reactivity control provisions. This backup system has the capability to shut down the reactor from any normal operating condition and to maintain the shutdown condition.

1.2.1.2.3 Plant Radiation Zone Classification

Radiation zones are identified as a means of classifying the occupancy restrictions on various areas within the plant site boundary. The criteria for each zone are described in Section 12.1.

1.2.1.3 System-by-System Approach

In this approach, the following systems are considered: (1) nuclear system; (2) power conversion systems; (3) electrical power systems; (4) radwaste systems; (5) auxiliary systems; (6) shielding and access control system; (7) nuclear safety and ESFs; and (8) process control systems.

The design criteria are presented below for each one of these systems.

1.2.1.3.1 Nuclear System Criteria

Design criteria for the nuclear system are given below, divided in three groups: mechanical, thermal, and nuclear.

1.2.1.3.1.1 Mechanical

The fuel cladding is designed to retain integrity as a radioactive-material barrier throughout the design power range. The fuel cladding is designed to accommodate, without loss of integrity, the pressures generated by the fission gases released from the fuel material throughout the design life of the fuel.

The fuel cladding, in conjunction with other plant systems, is designed to retain integrity throughout any abnormal operational transient.

Those portions of the nuclear system that form part of the nuclear system process barrier are designed to retain integrity as a radioactive material barrier following operational transients and accidents. For accidents in which one breach in the nuclear system process barrier is postulated, such a breach does not cause additional breaches in the nuclear system process barrier.

1.2.1.3.1.2 Thermal

Heat removal systems are provided in sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions, from plant shutdown to design power, and for any abnormal operational transients. The capacity of such systems is adequate to prevent fuel cladding damage.

Heat removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage. Following loss of operation of the normal heat removal systems, the reactor can be automatically shut down fast enough to permit decay heat removal systems to become effective.

1.2.1.3.1.3 Nuclear

The reactor core and the reactivity control system are designed such that the control rod action is capable of bringing the core subcritical, and maintaining it so, even when the rod of highest reactivity worth is fully withdrawn and unavailable for reinsertion.

The reactor core is designed so that its nuclear characteristics do not contribute to a divergent power transient.

The nuclear system is designed so that there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate plant systems.

1.2.1.3.2 Power Conversion Systems Criteria

The power conversion systems are designed to meet the following criteria:

a. Produce electrical power from the steam coming from the reactor, condense the steam into water, and return the water to the reactor as heated feedwater, with the major portion of its gases and particulate impurities removed

b. Ensure that any fission products or radioactivity associated with the steam and condensate during normal operation are safely contained inside the system, or are released under controlled conditions.

1.2.1.3.3 Electrical Power Systems Criteria

The electrical power systems are designed to meet the following criteria:

a. Sufficient normal and standby auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances

b. The power sources are adequate to accomplish all required ESF functions under postulated design-basis accident (DBA) conditions.

1.2.1.3.4 Radwaste Systems Criteria

The radwaste systems are designed to meet the following criteria:

a. The radwaste systems are designed to limit release of radioactive materials from the plant during normal operation to within the requirements of 10 CFR 20 and 10 CFR 50

b. Gaseous, liquid, and solid waste disposal systems are designed so that discharge of effluents and offsite shipments are in accordance with applicable regulations, including 10 CFR 50, 10 CFR 71, and 49 CFR 171 through 49 CFR 179, as appropriate.

The design provides a means by which plant operations personnel can be informed whenever operational limits on the release of radioactive material are approached.

1.2.1.3.5 Auxiliary Systems Criteria

Design criteria for each one of the auxiliary systems are presented below. The auxiliary systems considered are: (1) fuel handling and storage systems; (2) water systems; (3) process auxiliaries systems; (4) heating, ventilation, and air conditioning (HVAC) systems; and (5) other auxiliary systems.

1.2.1.3.5.1 Fuel Handling and Storage Facilities

Fuel handling and storage facilities are designed to prevent criticality and maintain adequate shielding and cooling for spent fuel.

1.2.1.3.5.2 Water Systems

The condenser circulating water system is designed to condense the steam discharged from the low-pressure turbines into the condenser.

The general service water (GSW) system is designed to remove heat from the reactor and turbine building closed cooling water (TBCCW) loops and selected equipment to maintain proper equipment temperatures during changing ambient conditions and plant operating modes.

The turbine building closed cooling water system (TBCCWS) is designed to transfer heat from the auxiliary equipment housed in the turbine building to the GSW system to maintain proper equipment temperatures, considering variations in the service water temperatures and plant operating conditions.

The reactor building closed cooling water system (RBCCWS) is designed to transfer heat from reactor auxiliary equipment to the GSW system to maintain proper equipment temperatures, considering variations in service water temperature and plant operating conditions.

The emergency equipment cooling water system (EECWS) provides a backup to the RBCCWS to cool essential equipment by transferring heat to the ultimate heat sink through the emergency equipment service water system (EESWS). It is designed to maintain this function in the event of seismic disturbance, loss of offsite power, or other site- or plant-related events.

The supplemental cooling chilled water system assists the RBCCW system in the RBCCW supplemental cooling mode of operation. RBCCW supplemental cooling is a loop within

RBCCW that provides water cooled by chilled water from SCCW to the EECW loops. The SCCW system and the RBCCW supplemental cooling loops are non-safety-related and are intended to operate during normal plant operation when GSW inlet temperatures are greater than 60°F (nominal).

The demineralized water makeup system is designed to provide water of the required purity in quantities sufficient for plant needs.

The potable water system is designed to provide drinking-quality water, according to state and local standards, in sufficient quantity for the use of plant personnel.

The sanitary wastewater system is designed to dispose of nonradioactive plant sewage liquid waste in accordance with state and local regulations.

The ultimate heat sink (residual heat removal [RHR] complex) is designed to provide cooling to the reactor system and essential auxiliaries under emergency conditions when the normal heat sinks are not available.

The condensate storage facilities are designed to provide retention of condensate to meet the requirements of plant systems, particularly primary system makeup to the condenser and water supply for selected ECCS. The facilities are designed with due regard for radioactive contamination of the condensate.

1.2.1.3.5.3 Process Auxiliary Systems

The compressed air system (instrument and service air) is designed to provide air of required quality at pressures and quantities sufficient to meet plant needs for various operating conditions.

The process sampling system is designed to enable the plant personnel to determine the composition and properties of process fluids in a safe and efficient manner.

The equipment and floor drain systems are designed to conduct drain fluids from general plant areas and equipment to the appropriate radwaste processing facilities.

1.2.1.3.5.4 Heating, Ventilation, and Air Conditioning Systems

The HVAC systems are designed to provide the required ambient environment for plant equipment, to provide a comfortable working environment for plant personnel, and to control airborne radioactivity.

1.2.1.3.5.5 Diesel Generator Auxiliaries

The onsite standby power system (diesel generator) auxiliaries are designed to provide the services required by the diesel generators. Each diesel generator is provided with its own auxiliaries, independent of all other units.

1.2.1.3.5.6 Other Auxiliary Systems

The fire protection system (FPS) is designed to adequately protect the plant from special hazards in accordance with national standards and insurance requirements.

The communications system is designed to provide contact between the main control room and various plant areas. Provisions are made for maintaining communications between essential areas in the event of loss of power.

The lighting systems are designed to provide adequate illumination for work in all plant areas. Provisions are made for emergency lighting in essential areas in the event of loss of power.

1.2.1.3.6 Shielding and Access Control Systems Criteria

The plant radiation shielding is designed to minimize the exposure of plant operating personnel and the general public to radiation due to the reactor, power conversion, auxiliary, and waste processing systems during normal operation, anticipated operational occurrences, postulated accident conditions, and maintenance. Radiation shielding is provided and access control patterns are established to limit radiation doses to the plant staff. The main control room and the technical support center are shielded against radiation so that occupancy is possible under accident conditions.

1.2.1.3.7 Nuclear Safety Systems and Engineered Safety Features Criteria

Design criteria for the nuclear safety systems and ESFs, in the system-by-system approach; have already been listed in various other paragraphs. They are as follows:

a. Design margins for the nuclear safety systems and ESFs are conservative

b. Nuclear safety systems are designed to respond to abnormal operational transients to limit fuel damage so that, should the freed fission products be released to the environs via the normal discharge paths for radioactive material, the limits of 10 CFR 20 and 10 CFR 50 will not be exceeded

c. Nuclear safety systems and ESFs act to preclude damage to the nuclear system process barrier as a result of internal pressures caused by abnormal operational transients or accidents

d. When positive and precise action is immediately required in response to accidents, such action is automatic, requiring no decision or manipulation of controls by plant operating personnel

e. The reactor core and reactivity control systems are designed so that the control rod action is capable of making the core subcritical and maintaining it so, even when the rod of highest reactivity worth is fully withdrawn and unavailable for reinsertion

f. Essential safety actions are carried out by equipment in sufficient redundancy and independence so that a single failure of active components will not prevent the required actions

g. Provision has been made for control of active components of nuclear safety systems and ESFs from the main control room

h. Nuclear safety systems and ESFs are designed to permit demonstration of their compliance with functional performance requirements

i. Nuclear safety systems and ESFs are designed to maintain operability under all plant-related and site-related events (e.g., earthquakes, tornadoes, floods, fires)

j. Features of the plant essential to the mitigation of accident consequences are designed for fabrication and erection to quality standards that reflect the importance of the safety function to be performed. A quality assurance program has been established and implemented

k. Onsite standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal offsite auxiliary power is not available

l. The plant is designed, fabricated, erected, and will be operated in such a way that under accident conditions the release of radioactive materials to the environment is within the requirements of 10 CFR l00 or 10 CFR 50.67 as applicable

m. Those portions of the nuclear system that form part of the nuclear system process barrier are designed to retain integrity as a radioactive material barrier following accidents. For accidents in which one breach in the nuclear system process barrier is postulated, such a breach does not propagate additional failures in the nuclear system process barrier

n. Onsite standby electrical power sources have sufficient capacity to power the nuclear safety systems and ESFs requiring electrical power during accident conditions

o. The primary containment is designed to retain integrity as a radioactive material barrier during and following accidents that release radioactive material into the primary containment volume

p. The primary containment is designed to permit integrity and leaktightness testing at periodic intervals

q. A secondary barrier (containment) is provided that completely encloses both the primary containment and the fuel storage areas. The secondary barrier design includes a method for controlling the rate of release of radioactive materials from the barrier, and further includes a capability for filtering radioactive materials within the barrier. In the event of a design-basis tornado, the secondary containment barrier above the refueling floor will be breached. See Section 3.3 for additional discussion regarding tornado design

r. The secondary barrier is designed to act as a radioactive material barrier under the same conditions that require the primary containment to act as a radioactive material barrier

s. For a special event such as loss of habitability of the main control room, it is possible to bring the reactor from power range operation to a hot-shutdown condition from outside the main control room, as well as to bring the reactor to a cold-shutdown condition from the hot- shutdown condition

t. For a special event, such as inability to shut down the reactor with control rods, backup reactor shutdown capability is provided, independent of normal reactivity control provisions. This backup system has the capability to shut down the reactor from any normal operating condition and to maintain the shutdown condition.

1.2.1.3.8 Process Control Systems Criteria

Design criteria for the various process control systems are listed below. The systems under consideration are as follows: (l) nuclear systems; (2) power conversion systems; and (3) electrical power systems.

1.2.1.3.8.1 Nuclear System Process Control

Design criteria for nuclear system process control are as follows:

a. Control equipment is provided to allow the reactor to respond to load changes

b. It is possible to control the reactor power level manually

c. Control of the nuclear system is possible from a single location

d. Nuclear system process controls and alarms are arranged to allow the operator to assess the condition of the nuclear system rapidly and locate process system malfunctions

e. Interlocks, or other automatic equipment, are provided as a backup to plant procedural controls to avoid conditions requiring the actuation of nuclear safety systems or ESFs.

1.2.1.3.8.2 Power Conversion Systems Process Control

Design criteria for power conversion systems process control are as follows:

a. Control equipment is provided to control the reactor pressure throughout its operating range

b. The turbine is able to respond automatically to minor changes in load

c. Control equipment in the feedwater system maintains the water level in the reactor pressure vessel (RPV) at the optimum level required by steam separators

d. Control of the power conversion equipment is possible from one location

e. Interlocks or other automatic equipment are provided, in addition to procedural controls, to avoid conditions requiring unnecessary actuation of nuclear safety systems or ESFs.

1.2.1.3.8.3 Electrical Power System Process Control

Design criteria for electrical power system process control are as follows:

a. The electrical power system is designed as a split bus system, with either system being adequate to safely shut down the unit

b. Protective relaying is used to detect and isolate faulted equipment from the system with a minimum of disturbance in the event of equipment failure

c. Undervoltage relays are used on the emergency equipment buses to isolate them from the normal electrical system in the event of loss of offsite power, and to initiate starting the onsite standby power system diesel generators

d. The standby emergency power diesel generators are started by automatically initiated control relays. The generators are also loaded by a programmed control system to meet the existing emergency conditions

e. All 4160-V and 480-V electrically operated breakers are controllable from the main control room

f. Metering for essential generators, transformers, and circuits is monitored in the main control room.

1.2.2 Plant Description

Fermi 2 contains a GE BWR nuclear steam supply system (NSSS) that delivers at rated flow approximately 14,864,000 lb/hr of 991-psia steam to the turbine generator and auxiliary equipment, which produces (at rated steam flow) 1217 MWe of gross electrical output at the generator terminals. The main condenser circulating water is cooled by two wet-type, natural-draft, hyperbolic cooling towers. The plant is equipped with auxiliary systems for control of radioactive contamination, nuclear safety assurance, and operation of the NSSS and turbine generator. The plant is located southwest of Detroit, Michigan and is intended to supply electrical power to the Edison service area.

1.2.2.1 Location and Size of Site

The Fermi 2 site is located on the shore of the western end of Lake Erie, at Lagoona Beach in Frenchtown Township, Monroe County, Michigan. The site is approximately 6 miles northeast of Monroe, Michigan, 30 miles southwest of downtown Detroit, Michigan, and 25 miles northeast of Toledo, Ohio. Reactor centerline coordinates are latitude 41 57'48"N., and longitude 83 15'31"W. The site consists of approximately 1260 acres.

On the same site is Fermi 1, originally a fast breeder reactor, and later also a conventional oil-fired power plant. Both are decommissioned. Also on the site are four oil-fired combustion turbine peaking units rated at 62.4-MWe total capacity. In addition, there is the Independent Spent Fuel Storage Installation for dry storage of Fermi 2 spent fuel. Figures 1.2-1, 1.2-2, and 1.2-3 show the relationship of the site to the surrounding areas. Figure 1.2-4 shows the site boundary and general site location of Fermi 1 and Fermi 2. Figure 1.2-5 is the Fermi site plan.

Transportation facilities are readily available. Interstate Highways 75 and 275 are approximately 5 miles west of the site. More immediate access to the site is available from the Dixie Highway, which runs north and south approximately 2 miles west of the site. From the Dixie Highway, Enrico Fermi Drive (a paved private access road) enters the site on the western boundary where it serves as the main entrance. Rail service to the site is furnished by a spur line from the main line which is 4 miles west of the site.

1.2.2.2 Description of Plant Environs

1.2.2.2.1 General

The site is bounded on the north by Swan Creek, on the east by Lake Erie, on the south by Pointe Aux Peaux Road, and on the west by Toll Road. Entrance to the site is from the west by way of Enrico Fermi Drive, a private road owned by Edison, and from the south via Pointe Aux Peaux Road to Quarry Lake Road, also owned by Edison.

The northern and southern areas of the site are dominated by large lagoons. The western areas are dominated by several woodlots and a series of quarry lakes. Site elevation ranges from approximately 25 ft above the lake level on the western edge of the site to lake level on the eastern edge.

1.2.2.2.2 Population

The area within a 10-mile radius of the site has an estimated total population of 86,214 (1980 data). The only substantially populated community within this radius is the city of Monroe, Michigan, approximately 6 miles southwest, whose 1980 population was 22,995.

Downtown Detroit, Michigan, is located approximately 30 miles northeast of the Fermi site. Downtown Toledo, Ohio, is located about 25 miles southwest.

1.2.2.2.3 Land Use

Approximately 70 percent of Monroe County, in which the plant is located, is farmland. Most of the industrial activity in the county is concentrated in the city of Monroe. Within a 50-mile radius of the site are all, or portions of, eight counties in Michigan, nine counties in Ohio, and two counties in Ontario, Canada. A large number and variety of manufacturing industries are found in this area. However, according to 1974 data, more than 50 percent of the land within the 50-mile radius is farmland, except for the area in the six counties located around metropolitan Detroit and Toledo.

1.2.2.3 Design Bases Dependent On the Site Environs

1.2.2.3.1 Offgas System

A rooftop plant vent is provided for the discharge of gaseous effluent to the atmosphere. Gaseous releases will be in compliance with 10 CFR 20 and 10 CFR 50.

1.2.2.3.2 Liquid Waste Effluents

Liquid waste will be released so that concentrations at the point of discharge will be in compliance with 10 CFR 20 and 10 CFR 50.

1.2.2.3.3 Wind Loading Design

The primary containment, reactor systems, and structures that contain equipment necessary for safe shutdown are designed with a wind load consideration for a sustained high wind (90 mph) and a transient condition imposed by a postulated tornado (300-mph rotation, 60-mph translation, 3-psi external pressure drop at 1 psi/sec).

1.2.2.3.4 Seismic Design

The design of Category I structures is for a maximum horizontal ground acceleration of 0.15g. The maximum vertical ground acceleration is considered to occur simultaneously, and is equal to 0.67 times the horizontal ground acceleration. The combined stresses resulting from functional loadings and a safe-shutdown earthquake (SSE) having a horizontal ground acceleration of 0.15g will be such that a safe shutdown can be achieved.

1.2.2.3.5 Flooding

A comprehensive study has established a maximum stillwater elevation of 586.9 ft (New York Mean Tide, 1935) for the plant site, based on the probable maximum meteorological event (PMME).

The site grade is 583.0 ft (New York Mean Tide, 1935) along the periphery of the power block (reactor/auxiliary building, RHR complex, turbine house, radwaste building, service building, etc.). From this reference elevation, the site has been graded for proper drainage.

Fermi 2 Category I structures and components are conservatively flood protected (waterproofed) to an elevation of 588 ft.

The shoreline of that portion of the site occupied by the plant is protected from erosion resulting from wave action through the use of a specially constructed shore barrier.

1.2.2.3.6 Loss of Normal Heat Sink

The natural-draft cooling towers provide the normal heat sink for the once-through-type main unit condenser and auxiliary systems. Should this heat sink be lost, the reactor can be safely shut down and maintained using the mechanical-draft cooling towers and the RHR reservoir as a heat sink.

1.2.2.3.7 Environmental Radiation Monitoring Program

An environmental monitoring program has been under way at the Fermi site since 1958 when Fermi 1 was being constructed. The present program, which has been specific for Fermi 2 since 1978, is referenced in UFSAR Section 11.6.

1.2.2.4 General Arrangement of Structures and Equipment

The principal structures located on the plant site are the following:

a. The reactor building, which houses the drywell, the suppression pool, the NSSS, the ESFs, some auxiliary systems equipment, and the fuel storage and shipping area

b. The turbine building, which houses the power conversion equipment, the offgas system, and the plant auxiliaries

c. The auxiliary building, which houses the main control room, the computer facility, electrical equipment, and HVAC equipment

d. The radwaste building, which houses the radioactive waste treatment facilities for liquid and solid waste

e. The switchyard

f. The condensate storage tanks and fuel-oil storage tanks

g. The RHR complex, which houses the emergency diesel generators (EDGs), the RHR cooling towers, the RHR service water (RHRSW) reservoir, and the RHRSW, EESWS, and EDG service water pumps

h. Two natural-draft hyperbolic circulating water cooling towers, and corresponding intake conduits, intake structures, and discharge structures

i. The GSW house, and corresponding intake conduits, intake structures, and discharge structures

j. The circulating water pump house, and corresponding intake conduits, intake structures, and discharge structures

k. A reservoir pond

l. The auxiliary boiler house

m. The meteorological towers

n. The office service building and annex

o. The Fermi 1 plant complex

p. The nuclear operations center

q. Technical assistance center

r. Availability improvement center.

s. Hydrogen/Oxygen supply facility for hydrogen water chemistry

t. Nuclear training center

u. The Independent Spent Fuel Storage Installation (ISFSI) Equipment Storage Building

v. The Independent Spent Fuel Storage Installation (ISFSI) Pad

w. ISFSI Fabrication Pad

x. ISFSI Transfer Pad

y. ISFSI Cask Transfer Facility

z. FLEX Storage Facility #1

aa. FLEX Storage Facility #2

The arrangement of these structures on the plant site is shown in Figure 1.2-5. Figures 1.2-6 through 1.2-31 show the equipment arrangement in the principal buildings.

1.2.2.5 Nuclear System (Chapter 4)

The nuclear system includes a single-cycle, forced-circulation GE BWR that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for the rated power conditions is shown in Figure 1.2-32.

1.2.2.5.1 Reactor Core and Control Rods (Section 4.5)

Fuel for the reactor core consists of enriched uranium dioxide (UO_2) pellets sealed in Zircaloy-2 tubes. These tubes (or fuel rods) are assembled into individual fuel assemblies. Gross control of the core reactivity is achieved by cruciform-shaped, movable, bottom-entry control rods dispersed throughout the lattice of fuel assemblies. These rods are controlled by individual hydraulic systems.

Each fuel assembly has several fuel rods with gadolinia (Gd_2O_3) mixed in solid solution with the UO_2. Gadolinia is a burnable poison that diminishes the reactivity of the fresh fuel and is depleted as the fuel reaches the end of its first cycle.

A conservative limit of plastic strain is used for the design criterion for fuel rod cladding failure. The peak linear heat generation for steady-state operation is well below the damage limit, even late in life. Experience has shown that the control rods are not susceptible to distortion and have an average life expectancy many times greater than the residence time of a fuel loading.

1.2.2.5.2 Reactor Pressure Vessel and Internals (Section 4.5)

The RPV contains the following:

a. Core and supporting structures
b. Steam separators and dryers
c. Jet pumps
d. Control rod guide tubes
e. Distribution lines for the feedwater, core sprays, and standby liquid control
f. In-core instrumentation
g. Other components.

The main connections to the RPV include the steam lines, the coolant recirculation lines, feedwater lines, control rod drive (CRD) housings, and ECCS lines.

The RPV is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig. The nominal rated operating pressure in the steam space above the separators is 1045 psia. The RPV is fabricated of carbon steel and is clad internally (except for the top head) with stainless steel.

The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the RPV. The steam is then directed to the turbine through four 24-in.-diameter main steam lines. Each steam line is provided with three isolation valves in series, one inside the primary containment, and two outside the primary containment.

1.2.2.5.3 Reactor Recirculation System (Subsection 5.5.1)

The reactor recirculation system pumps reactor coolant through the core to remove energy generated in the fuel. This is accomplished by two recirculation loops external to the RPV but inside the primary containment. Each external loop has one motor-driven recirculation pump. Recirculation pump speed can be varied to allow control of reactor power level through the effects of coolant flow rate on moderator void content. The internal portion of the loop consists of the jet pumps, which contain no moving parts, but have high-velocity nozzles to provide a continuous internal circulation path for the core coolant flow. The jet pumps are located in the annular region between the core shroud and the vessel inner wall, and any recirculation line break would still allow core flooding to approximately two-thirds of the core height: the level of the top of the jet pumps.

1.2.2.5.4 Residual Heat Removal System (Subsection 5.5.7)

The RHR system consists of pumps, heat exchangers, and piping that fulfill the following functions:

a. Remove decay heat during and after plant shutdown

b. Remove heat from the primary containment following a LOCA.

1.2.2.5.5 Reactor Water Cleanup System (Subsection 5.5.8)

The reactor water cleanup (RWCU) system recirculates a portion of reactor coolant through a filter-demineralizer to remove particulate and dissolved impurities from the reactor coolant. It also removes excess coolant from the reactor system under controlled conditions.

1.2.2.6 Power Conversion System (Chapter 10)

The megawatt output of the generator is a function of the reactor steam power input to the turbine. Turbine control is achieved by an integrated speed and pressure control system. After the turbine has been brought to the synchronous speed of the power grid system and the generator breakers are closed to lock the machine into the system, the turbine is on pressure control. The turbine acts as a pressure-control device, maintaining the reactor pressure at its particular pressure setpoint level by varying control and/or bypass valve opening. The steam admitted to the turbine is controlled by a pressure regulator that senses the pressure just before the turbine inlet, thus controlling RPV pressure. Figure 1.2-33 shows the turbine-generator heat balance at rated flow.

Feedwater into the reactor is governed by a three-element control system that senses water level, main stream flow rate, and feedwater flow rate. Each of the signals combines in a three-element controller to control the speed of the two turbine-driven reactor feed pumps, thereby regulating feedwater requirements.

1.2.2.7 Electrical Power Systems (Chapter 8)

Power output from the unit is from a nominally rated 1350-MVA turbine generator. Generator output voltage is 22 kV. It is stepped up to 345 kV through two parallel main power transformers, then fed to the 345-kV switchyard and then to the system grid.

Offsite power available for the plant auxiliary system is from both the 345-kV switchyard, just west of the plant, and the 120-kV switchyard located at Fermi 1. Normal auxiliary power is provided from two system service transformers. One transformer is connected to the Fermi 2 345-kV switchyard, which is arranged in a nominal double breaker-double bus design. The remaining system service transformer is energized from the 120-kV switchyard through the 120/13.2-kV transformer 1 with an alternate through 120/13.8/13.8-kV transformer CTG II at the Fermi 1 site.

Onsite standby emergency power is provided from a four-diesel split-bus arrangement that is located in the RHR complex Category I structure near the reactor building. The diesel generators are sized to adequately carry the load necessary to shut down the reactor during a LOCA coincident with a complete loss of offsite power. Battery power is available for loads through two sets of 260/130-V dc Category I station batteries. The batteries furnish power to redundant essential loads. A highly reliable source of 48/24-V dc power is available for neutron monitoring and certain other instrumentation. In addition, a balance-of-plant (BOP) 260/130-V dc battery provides dc power for BOP loads. The batteries are sized to provide adequate power to those loads for a period of not less than 4 hr without battery charger availability. The chargers are full sized and capable of handling the load requirements, while still providing the required float charge for the battery.

1.2.2.8 Radwaste Systems (Chapter 11)

The radioactive waste disposal systems and the radiation monitoring systems (RMS) are designed so that liquid, solid, and gaseous effluents are considerably below those specified in 10 CFR 20.

1.2.2.9 Nuclear Safety Systems and Engineered Safety Features

1.2.2.9.1 Reactor Protection System (Section 7.2)

The reactor protection system (RPS) initiates a rapid, automatic shutdown (scram) of the reactor. It acts in time to prevent fuel cladding damage and any nuclear system process barrier damage following operational transients. The RPS overrides all operator actions and process controls and is based on a fail-safe design philosophy that allows appropriate protective action even if a single failure occurs.

1.2.2.9.2 Neutron Monitoring System (Subsection 7.6.l)

Those portions of the neutron monitoring system (NMS) that provide high neutron flux signals to the RPS qualify as a nuclear safety system. The intermediate range monitors (IRMs) and average power range monitors (APRMs), which monitor neutron flux via in-core detectors, signal the RPS to scram in time to prevent fuel cladding damage as a result of overpower transients.

1.2.2.9.3 Control Rod Drive System (Subsection 4.5.2)

When a scram is initiated by the RPS, the CRD system inserts the negative reactivity necessary to shut down the reactor. Each rod is individually controlled by a hydraulic control unit (HCU). When a scram signal is received, high-pressure water, stored in an accumulator in the HCU, forces its control rod into the core.

1.2.2.9.4 Nuclear System Pressure Relief System (Subsection 5.2.2)

A pressure relief system, consisting of safety/relief valves mounted on the main steam lines, prevents excessive pressure inside the nuclear system following either abnormal operational transients or accidents.

1.2.2.9.5 Reactor Core Isolation Cooling System (Subsection 5.5.6)

The reactor core isolation cooling (RCIC) system provides makeup water to the RPV when the vessel is isolated. The RCIC system uses a steam-driven turbine pump unit and operates automatically, with sufficient coolant flow in time to maintain adequate water levels in the RPV.

1.2.2.9.6 Primary Containment (Section 6.2)

The primary containment (Mark I containment) is a steel plate pressure vessel consisting of a light bulb-shaped drywell and a torus-shaped pressure suppression chamber. The primary containment is designed in accordance with the 1968 ASME Boiler and Pressure Vessel Code, Class B Vessel, including the 1969 summer addenda. The basic objective of the primary containment is to provide the capability, in the event of a postulated LOCA, of limiting the release of fission products within the values specified in 10 CFR 50.67 or 10 CFR 100.

1.2.2.9.7 Primary Containment and Reactor Isolation System (Subsection 6.2.4)

The containment isolation system consists of the isolation valves and controls required for the timely isolation of the containment in the event of incidents when the free release of containment contents cannot be permitted. The reactor isolation system consists of the isolation valves and controls required for the timely isolation of the RPV in the event of incidents when the fuel must be prevented from failing.

1.2.2.9.8 Secondary Containment (Section 6.2)

The reactor building, in conjunction with the reactor building heating and ventilation system and the standby gas treatment system (SGTS), constitutes the secondary containment. The primary purpose of the secondary containment is to minimize the ground-level release of airborne radioactive materials and provide means for a controlled release of the building atmosphere.

The reactor building is a cast-in-place reinforced-concrete structure enclosing the primary containment. The superstructure of the reactor building is composed of structural steel and steel siding.

1.2.2.9.9 Main Steam Line Isolation Valves (Subsection 5.5.5)

All pipelines that penetrate the primary containment, offering a potential release path for radioactive material, are provided with redundant isolation capabilities. The main steam lines, because of their large size and large mass flow rates, are given special isolation consideration. The automatic isolation valves in each main steam line, immediately inside and outside the primary containment, are powered by both pneumatic pressure and spring force. These valves fulfill the following objectives:

a. Prevent excessive damage to the fuel barrier by limiting the loss of reactor coolant from the RPV as a result of (1) a major leak in the steam piping outside the primary containment, or (2) a malfunction of the pressure control system causing excessive steam flow from the RPV

b. Limit the release of radioactive materials by closing the nuclear system process barrier in the event of a gross release of radioactive materials from the fuel to the reactor cooling water and steam

c. Limit the release of radioactive materials by closing the primary containment barrier in the event of a major leak from the nuclear system inside the primary containment.

A third, motor-operated, main steam isolation valve (MSIV) is provided in each main steam line to limit postulated leakage. See Subsection 6.2.6.

1.2.2.9.10 Main Steam Line Flow Restrictors (Subsection 5.5.4)

A venturi-type flow restrictor is installed in each steam line. These devices limit the loss of coolant from the RPV before the MSIVs are closed, in case of a main steam line break outside the primary containment.

1.2.2.9.11 Emergency Core Cooling System (Section 6.3)

A number of functions of the ECCS are provided to limit fuel cladding temperatures to minimize the release of radioactive material and to ensure the continued functional capability of the containment facility if a breach in the nuclear system process barrier results in a loss of reactor coolant. The four functions of the ECCS are presented in the following paragraphs.

1.2.2.9.11.1 High Pressure Coolant Injection System

The high pressure coolant injection (HPCI) system provides and maintains an adequate coolant inventory inside the RPV. This limits fuel cladding temperature, which may result from postulated small breaks in the nuclear system process barrier. A high-pressure system is needed for small breaks because the RPV depressurizes slowly, preventing low-pressure systems from injecting coolant. Also, the HPCI system reduces RPV pressure rapidly, permitting operation of the low-pressure systems. The HPCI system includes a turbine-driven pump powered by reactor steam. The system is designed to accomplish its function on a short-term basis, without reliance on plant auxiliary power supplies other than the dc power supply.

1.2.2.9.11.2 Automatic Depressurization System

The automatic depressurization system (ADS) rapidly reduces RPV pressure in a LOCA situation in which the HPCI system fails to maintain the RPV water level. The depressurization provided by the system enables the low-pressure ECCS to deliver cooling water to the RPV. The ADS uses some of the relief valves that are part of the nuclear system pressure relief system. The automatic relief valves are arranged to open on conditions indicating that a break in the nuclear system process barrier has occurred, and that the HPCI system is not delivering sufficient cooling water to the RPV to maintain the water level above a preselected value. The ADS will not be activated unless either the core spray or low pressure coolant injection (LPCI) system pumps are operating. This ensures that adequate cooling will be available so that boiling will not occur at the reduced pressure.

1.2.2.9.11.3 Core Spray System

The core spray system consists of two independent pump loops that deliver cooling water to independent spray spargers over the core. The system is actuated by conditions indicating that a breach exists in the nuclear system process barrier. Water is delivered to the core after RPV pressure is reduced. This system provides the capability of cooling the fuel by spraying water onto the core. Either of the core spray loops is capable of limiting fuel cladding temperature to less than 2200°F following a LOCA.

1.2.2.9.11.4 Residual Heat Removal - Low Pressure Coolant Injection Mode

The LPCI is an operating mode of the RHR system, but is discussed here because the LPCI mode acts as an ESF in conjunction with the other functions of the ECCS. The LPCI system uses the pump loops of the RHR system to inject cooling water at low pressure into an undamaged reactor recirculation loop. The LPCI is actuated by conditions indicating a breach in the nuclear system process barrier. Water is delivered to the core after RPV pressure is reduced. The LPCI operation, together with the core shroud and jet pump arrangement, provides the capability of core reflooding, following a LOCA, in time to prevent fuel cladding temperature from exceeding 2200°F.

1.2.2.9.12 Residual Heat Removal System - Containment Cooling Mode (Section 6.3)

The containment cooling subsystem is placed in operation to limit the temperature of the water in the suppression pool following a design-basis LOCA. In the containment cooling mode of operation, the RHR pumps take suction from the suppression pool and pump the water through the RHR system heat exchangers. Cooling takes place by transferring heat to the RHRSW system. The primary coolant is then discharged back to the suppression pool.

Another portion of the RHR system sprays water into the primary containment as an augmented means of removing energy from the containment following a LOCA. This capability is in excess of the required emergency heat removal capability and can be placed in service at the discretion of the operator.

1.2.2.9.13 Control Rod Velocity Limiter (Subsection 4.5.2.1)

A control rod velocity limiter is attached to each control rod to limit the velocity at which it can fall out of the core should it become detached from its CRD. This action limits the rate of reactivity insertion resulting from a control rod drop accident. The limiters contain no moving parts.

1.2.2.9.14 Control Rod Drive Housing Supports (Subsection 4.5.3)

The CRD housing supports are located underneath the RPV near the control rod housings. The supports limit the travel of a control rod should a control rod housing become ruptured. The supports prevent a nuclear excursion as a result of a housing failure and thus protect the fuel barrier.

1.2.2.9.15 Standby Gas Treatment System (Subsection 6.2.3)

The SGTS consists of two identical 100 percent equipment and filter trains for the plant. On detection of radioactivity or conditions that could lead to a release of radioactivity, the SGTS functions to minimize the release-related offsite dose rates by permitting the venting and purging of both the primary and secondary containment atmospheres under accident or abnormal conditions, and at the same time containing any airborne particulate or halogen contamination that might be present. Either train may be considered as an installed spare, with the other train being capable of passing the required amount of air. Either train alone is capable of exchanging the total reactor building air volume once in a 24-hr period.

Each equipment train contains an electric heater, a prefilter, a high-efficiency particulate filter (water and fire resistant), an iodine filter (fire resistant), a fan, and associated instrumentation.

The primary containment can be purged through the SGTS.

1.2.2.9.16 Onsite AC Power Supply (Subsection 8.3.1)

The onsite ac power supply provides sufficient power to those devices necessary to produce a safe shutdown with subsequent reactor decay heat removal should normal offsite power not be available. Power is derived from four EDGs housed in a Category I structure (RHR complex) located near the reactor building. The EDGs are installed in division pairs. Either division pair is capable of completely maintaining itself and the safety loads it supplies for 7 days. The entire standby power supply system is independent of offsite power.

1.2.2.9.17 DC Power Supply (Subsection 8.3.2)

The dc power supply provides power to those safety devices receiving their motive and/or control power from the station battery systems. The batteries are redundant and each has a battery charger capable of providing the full load capacity and maintaining the float charge on the battery.

1.2.2.9.18 Ultimate Heat Sink (Residual Heat Removal Complex) Section 6.3 and Subsection 9.2.5)

The RHR complex provides cooling for the RHR system, EESW, and EDGs. The RHR complex consists of mechanical-draft cooling towers, cooling water reservoirs, RHR, and emergency equipment cooling and EDG cooling service water pumps. The RHR complex also contains the EDGs. (See Figures 1.2-25 through 1.2-31.)

1.2.2.9.19 Main Steam Line Radiation Monitor System (Subsection 11.4.3.8.2.3)

The main steam line radiation monitor system consists of four gamma radiation channels located external to the main steam lines just outside the primary containment. The monitors are designed to detect a gross release of fission products from the fuel. On detection of high radiation, the trip signals generated by the monitors are used to isolate the reactor water sample system, trip condenser mechanical vacuum pumps, and trip glad seal exhausters.

1.2.2.9.20 Fuel Pool Ventilation Exhaust Radiation Monitor System (Subsection 11.4.3.8.2.11)

The fuel pool ventilation exhaust radiation monitor system consists of four radiation monitors arranged to monitor the activity level of the ventilation exhaust from the fuel pool area. On detection of high radiation, the SGTS is automatically started, the primary containment vent valves are closed, the reactor building vent system is isolated, the control center is isolated, and control center emergency recirculation is initiated.

1.2.2.9.21 Emergency Equipment Cooling Water System (Subsection 9.2.2)

Equipment required for a safe shutdown of the reactor is cooled by the EECWS, which is cross connected to the RBCCWS for normal operation. The EECW is isolated and is cooled by the ultimate heat sink (RHR complex) for emergency operation. The EECWS is designed to Category I requirements.

1.2.2.9.22 Combustible Gas Control (Subsections 6.2.5 and 9.3.6)

The NRC amended 10 CFR 50.44, “Standards for combustible gas control system in light-water-cooled power reactors” on October 16, 2003 to eliminate the requirements for hydrogen recombiners. The hydrogen recombiner Technical Specification requirements were subsequently removed by License Amendment 159, dated March 15, 2004. Regulatory Guide 1.7 was revised in March 2007 to reflect the amended 10 CFR 50.44. The Combustible Gas Control System (CGCS) has been retired in place with its electrical circuits de-energized and fluid process piping isolated from primary containment with redundant locked-closed isolation valves. Combustible gas control of the primary containment is provided by inerting the primary containment with nitrogen.

1.2.2.9.23 Instrumentation and Control Power Supply System Subsection 8.3.1)

The purpose of the instrumentation and control power supply system is to provide a reliable source of 120-V ac regulated power where necessary, for analog instrumentation, solenoid valves, and logic relaying for certain specific systems. These systems include: core spray, RHR, radwaste control, and NSSS process instrumentation.

1.2.2.9.24 Main Control Room Emergency Ventilation System (Section 6.4)

A main control room emergency ventilation system is provided to protect the main control room operators against radiation, smoke, or any noxious chemical release. It consists of an emergency makeup (pressurizing) and a control center recirculation filter train with 100 percent redundant active components.

1.2.2.9.25 Engineered Safety Features Ventilation Cooling System (Subsection 6.2.l.2)

All ESF equipment is provided with ventilation fans and/or cooling units to maintain design temperatures if the normal ventilation system is isolated. Redundant divisional ESF equipment is supplied with its own independent ventilation equipment powered by the corresponding division of the ESF bus.

1.2.2.10 Special Safety Systems

1.2.2.10.1 Standby Liquid Control System (Subsection 4.5.2.4)

Although not intended to provide prompt reactor shutdown, like the control rods, the standby liquid control system (SLCS) provides a redundant, independent, and different way to bring the nuclear fission reaction to subcriticality and maintain subcriticality as the reactor cools. The system permits an orderly and safe shutdown in the event that control rods cannot be inserted into the reactor core in sufficient number to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect in decreasing power from rated power to the cold-shutdown condition.

The SLCS is also credited for injecting sodium pentaborate into the reactor coolant system after a design basis LOCA in order to control ECCS water pH to prevent iodine re-evolution. The SLCS can be manually initiated to provide this function.

1.2.2.10.2 Plant Equipment Outside the Main Control Room To Effect Reactor Shutdown (Section 7.5)

Instrumentation and controls necessary to meet the requirements of 10 CFR 50, Appendix A, Criterion 19, have been provided on a remote shutdown panel located outside the main control room. Details of the instruments and controls provided on the shutdown panels and the procedures required for carrying out a safe and orderly shutdown are described fully in Subsection 7.5.1.5.

Additionally, local shutdown panels are provided to meet the requirements of 10 CFR 50, Appendix R. These panels are provided in the event a fire causes a loss of control from the main control room. Details on achieving reactor shutdown in this event are provided in Subsection 7.5.2.5.

1.2.2.11 Nuclear System Process Control and Instrumentation

1.2.2.11.1 Reactor Manual Control System (Subsection 7.7.1.1)

The reactor manual control system (RMCS) provides the means by which control rods are positioned from the main control room for gross power control. The system operates valves in each HCU to change control rod position. Only one control rod can be manipulated at a time. The RMCS includes the logic that restricts control rod movement (rod block), under certain conditions, as a secondary control.

1.2.2.11.2 Recirculation Flow Control System (Subsection 7.7.1.2)

The recirculation flow control system (RFCS) controls the speed of the reactor recirculation pumps. Adjusting the pump speed changes the coolant flow rate through the core, thereby changing the core power level.

1.2.2.11.3 Neutron Monitoring System (Subsection 7.6.1.13)

The NMS is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level, for the entire range of flux conditions that can exist in the core. The source range monitors SRMs and the IRMs provide flux level indications during reactor startup and low power operation. The local power range monitors (LPRMs) and APRMs allow assessment of local and overall flux conditions during power range operation. Rod block monitors (RBMs) are provided to prevent rod withdrawal when reactor power should not be increased at the existing reactor coolant flow rate and also function to prevent local fuel damage. The flux mapping and calibration subsystem provides a means to calibrate individual monitors with traveling in-core probes.

1.2.2.11.4 Refueling Interlocks (Section 7.6.1.1 and Subsection 9.1.4)

A system of interlocks that restricts movement of refueling equipment and control rods when the reactor is in the refueling mode prevents an inadvertent criticality during refueling operations. The interlocks back up procedural controls that have the same objective. The interlocks affect the refueling bridge, refueling bridge hoists, fuel grapple, and control rods.

1.2.2.11.5 Reactor Pressure Vessel Instrumentation (Section 5.6)

In addition to instrumentation for the nuclear safety systems and ESFs, instrumentation is provided to monitor and transmit information that can be used to assess both the condition existing inside the RPV and the physical condition of the vessel itself. This instrumentation monitors RPV parameters such as pressure, water level, surface temperature, internal differential pressures, coolant flow rates, and top head flange leakage.

1.2.2.11.6 Integrated Plant Computer System (Subsection 7.6.1.9)

The Integrated Plant Computer System (IPCS) includes the following process monitoring functions:

a. Scan, Log and Alarm (SLA)

b. Man-Machine Interface (MMI)

c. Data Archival

d. Nuclear Steam Supply System (NSSS)

e. Balance of Plant (BOP)

f. Emergency Response

   1. Safety Parameter Display System (SPDS)

   2. Emergency Response Data System (ERDS)

g. Meteorological (MET)

h. Transient Recording and Analysis (TRA)

i. External System Interfaces

1.2.2.11.7 Reactor Coolant Pressure Boundary Leakage Detection System (Subsection 5.2.7)

The nuclear leak detection system consists of temperature, pressure, flow, and fission product sensors with associated instrumentation and alarms. This system detects and annunciates leakage in the following systems:

a. Main steam lines

b. Reactor water cleanup

c. Residual heat removal

d. Reactor core isolation cooling

e. High pressure coolant injection

f. Instrument lines.

Small leaks are generally detected by temperature and pressure changes, fillup rate of drain sumps, and fission product concentration inside the primary containment. Large leaks are also detected by changes in reactor water level and changes in process lines.

1.2.2.11.8 Emergency Core Cooling System Suction Piping Leakage Detection (Subsections 6.3.2.2.7and 7.6.1.8.12)

The ECCS leak detection system (LDS) uses the sump level and torus water level monitors to identify any failed line in the reactor building subbasement area and, thereby, prevents a loss of ECCS pump suction head.

1.2.2.11.9 Primary Containment Monitor System (Subsections 6.2.1.5 and 7.6.1.12)

The NRC amended 10 CFR 50.44, “Standards for combustible gas control system in light-water-cooled power reactors” on October 16, 2003 to eliminate the requirements for hydrogen recombiners. The hydrogen recombiner Technical Specification requirements were subsequently removed by License Amendment 159, dated March 15, 2004. Regulatory

Guide 1.7 was revised in March 2007 to reflect the amended 10 CFR 50.44. The Combustible Gas Control System (CGCS) has been retired in place with its electrical circuits de-energized and fluid process piping isolated from primary containment with redundant locked-closed isolation valves.

The primary containment monitor system (PCMS) is an advisory system only, which consists of measurements of hydrogen and oxygen concentration, particulate and gaseous radiation level, pressure, temperature, and water level in the drywell and suppression chamber. Hydrogen and oxygen monitors provide an operator with necessary information for the effective control of the nitrogen inerting system. The radiation monitor supplies information necessary for effective control of the SGTS as a primary containment atmospheric cleanup system and is a part of a redundant leak detection system, operating in conjunction with the drywell floor drain sump level indicating system. Hydrogen and radiation monitors also yield vital information regarding personnel access to the primary containment. The remaining instruments supply information on the overall conditions of the atmosphere in the drywell and suppression chamber and on water level and temperature in the suppression chamber.

1.2.2.11.10 Rodworth Minimizer Computer (Subsection 7.6.1.20)

The rodworth minimizer microcomputer system is a stand alone microcomputer-based system with an RWM operator display and a continuously operating self-test feature that enforces adherence to established startup, shutdown, and low power control rod procedures. The RWM prevents rod motion under low power conditions if the rod being moved is not moved in accordance with a preplanned pattern. The effect of the RWM is to limit the reactivity worth of the control rods by enforcing adherence to the preplanned rod pattern.

1.2.2.12 Power Conversion System Process Control and Instrumentation

1.2.2.12.1 Pressure Regulator and Turbine Generator Control (Subsection 10.4.4)

The pressure regulator maintains control of the turbine control and bypass valves to allow proper generator and reactor response to system load demand changes while also maintaining the nuclear system pressure essentially constant.

The turbine-generator speed-load controls act to maintain the turbine (generator frequency) at constant speed.

1.2.2.12.2 Feedwater Control System (Subsection 7.7.1.3)

A three-element controller is used to regulate the feedwater system so that the proper water level is maintained in the RPV. The control system uses main steam flow rate, RPV water level, and feedwater flow rate signals. The feedwater control signals are used to control the two turbine-driven feedwater pumps.

1.2.2.12.3 Turbine Generator Overspeed Trip System (Subsection 10.2.2)

The turbine generator overspeed trip system protects the turbine generator on overspeed. The system has overspeed trip mechanisms (four magnetic speed pickups and two overspeed trip rings), which will shut down the turbine, closing all valves (turbine high-pressure stop valves, control valves, low-pressure intercept valves, and low-pressure stop valves), on detection of the overspeed condition.

1.2.2.13 Electrical Power System Control and Instrumentation (Chapter 8)

The electrical power system is monitored by indicating and/or recording devices to account for the power generated at the plant, and to determine the auxiliary power usage required to achieve this level of generation. System requirements will govern the generator excitation level needed for the desired megavar output from the generator at the required terminal voltage. Wattmeters, ammeters, varmeters, etc., will be used to indicate electrical conditions. Selected inputs to the IPCS will record conditions for later comparison or record purposes.

1.2.2.14 Radiation Monitoring and Control (Chapters 11 and 12)

1.2.2.14.1 Process and Effluent Radiological Monitoring System (Section 11.4)

Radiation monitors are provided on various lines to monitor for either radioactive materials, released to the environs via process liquids and gases, or process system malfunctions. Subsection 11.4.1 provides the complete listing of all radiation monitoring systems.

1.2.2.14.2 Area Radiation Monitoring (Subsection 12.1.4)

The area radiation monitoring system (ARMS) provides indication in the relay room and recording and alarm in the main control room of abnormal radiation levels in plant work areas where radioactive material may be stored, handled, or inadvertently introduced. In addition, selected local areas have local alarm and/or indication, where necessary, to warn personnel of a substantial rapid increase in radiation levels.

1.2.2.14.3 Site Environs Radiation Monitoring (Section 11.6)

The site environs radiation monitoring program includes the use of passive dosimeters for direct radiation measurement and the orderly collection of samples for laboratory analyses. These analyses include airborne, aquatic, and terrestrial radiological measurements.

The program is designed to document: (1) background levels of direct radiation and concentrations of radionuclides that exist in aquatic and terrestrial ecosystems before and during plant operation; and (2) the concentrations of radionuclides that could be attributable to the operation of Fermi 2.

1.2.2.14.4 Liquid Radwaste Control (Section 11.2)

The liquid radwaste system is designed to segregate, collect, and process waste generated throughout the plant. Processing of the waste is normally sufficient to allow recycling of the wastewater. Ties exist among all of the liquid radwaste subsystems to provide backup processing in the event of failure of one subsystem.

1.2.2.14.5 Solid Radwaste Control (Section 11.5)

The solid radwaste system is designed to handle and package solid waste produced by the plant. The waste, depending on its radioactivity and type, will be packaged for offsite shipment in accordance with applicable regulations.

1.2.2.14.6 Gaseous Radwaste Control (Section 11.3)

The gaseous radwaste system processes and controls the release of gaseous radioactive wastes to the site environs so that the total radiation exposure to persons outside the controlled area does not exceed the limits of 10 CFR 20 and 10 CFR 50.

Continuous radiation monitors provide indications of radioactive release from the reactor by monitoring the offgas equipment trains. The offgas system radiation monitors are used to monitor and alarm on indication of high radioactivity.

1.2.2.15 Auxiliary Systems

1.2.2.15.1 New and Spent-Fuel Storage (Subsections 9.1.1 and 9.1.2)

New fuel may be stored in a dry vault in the reactor building subject to the restrictions discussed in Section 9.1.1.2.1. Irradiated (spent) fuel is stored underwater in the reactor building in the spent fuel pool or in dry storage casks at the Independent Spent Fuel Storage Installation.

1.2.2.15.2 Fuel Pool Cooling and Cleanup System (Subsection 9.1.3)

A fuel pool cooling and cleanup system (FPCCS) removes decay heat from spent fuel stored in the fuel pool and maintains a specified water temperature, purity, clarity, and level.

1.2.2.15.3 Nitrogen Inerting System (Containment) (Subsection 9.3.6)

The nitrogen inerting system is provided primarily to maintain a nitrogen atmosphere (inerted) inside the primary containment, and also to supply pressurized nitrogen for pneumatic service inside the primary containment and distribution throughout the plant.

1.2.2.15.4 Heating, Ventilation, and Air Conditioning Systems (Sections 6.4 and 9.4)

The objective of the plant HVAC systems is to provide a thermal environment and air quality to ensure personnel comfort, health, and safety and efficient equipment operation and integrity. In addition, the HVAC system for the main control room and the RHR ventilation systems and the fan-coil cooling units located in the reactor/auxiliary building have the further objective to operate under postulated accident conditions.

The HVAC systems provide individual air supply and exhaust systems as described in Section 9.4 for each system. Normally airflow will be routed from areas of lesser to areas of progressively greater potential contamination prior to being exhausted from the building. The ventilation arrangement will protect personnel and equipment from airborne contaminants and temperature extremes. The ventilation air exhaust from each ventilation system is located in such a manner as to minimize the possibility of the same air as was exhausted being drawn into a fresh air intake. Exhaust of potentially radioactive gases will be monitored. If the radioactivity in the exhaust systems exceeds a predetermined level, the ventilation system is shut down and the system intake and exhaust dampers are closed.

1.2.2.15.5 Normal Auxiliary AC Power (Section 8.3)

Normal auxiliary power is provided from two system service transformers. One transformer is connected to the Fermi 2 345-kV switchyard, which is arranged as a highly reliable double breaker-double bus design. The remaining system service transformer is energized from the 120-kV switchyard through the 120/13.2-kV transformer at the Fermi 1 site.

1.2.2.15.6 Reactor Building Closed Cooling Water System (Subsection 9.2.2)

The RBCCWS is a closed-loop system that provides parallel flow cooling to auxiliary equipment in the drywell and the reactor building. The closed loop provides a barrier between contaminated systems and the GSW discharged to the circulating water reservoir. Heat is removed from the closed loop by the GSW system.

1.2.2.15.6.1 RBCCW Supplemental Cooling (Subsection 9.2.2)

RBCCW is designed with two RBCCW supplemental cooling loops. These loops operate using separate pumps and heat exchangers using chilled water from the supplemental cooling chilled water system to cool the RBCCW supplied to the EECW loops during normal plant operation. RBCCW supplemental cooling operation is optional, intended for use when the GSW supply temperature exceeds approximately 60°F.

1.2.2.15.6.2 Supplemental Cooling Chilled Water (Subsection 9.2.9)

The supplemental cooling chilled water (SCCW) system is a chilled water closed loop system designed to cool the water that is supplied to EECW by the RBCCW supplemental cooling loops.

The SCCW system transfers the heat it has removed from the RBCCW via the supplemental RBCCW system to the GSW system via mechanical chillers. The chillers are designed to operate using GSW supply water at 60°F or greater.

1.2.2.15.7 Turbine Building Closed Cooling Water System (Subsection 9.2.7)

The TBCCWS is designed to cool the auxiliary plant equipment associated with the power conversion systems over the full range of normal plant operation.

1.2.2.15.8 Water Systems

1.2.2.15.8.1 Circulating Water System (Subsection 10.4.5)

The circulating water system is a closed-loop system designed to condense steam exhausting into the main condenser from the main turbine. The system consists of five circulating water pumps, two vertical natural-draft cooling towers, piping, and a cooling reservoir. The circulating water pumps are located in a circulating-water pump house adjacent to the reservoir.

1.2.2.15.8.2 General Service Water (Subsection 9.2.1)

The GSW system is designed to cool various non-safety-related plant auxiliary systems such as the RBCCW and the TBCCW during all normal plant operating modes. The GSW system also provides the source of makeup water for the plant FPS and serves as a source of makeup water for the RHR complex. The once-through GSW discharges into the station's circulating water system where its heat load is rejected in the two natural-draft cooling towers. The GSW thus serves as cooling tower makeup.

1.2.2.15.9 Compressed Air Systems (Subsection 9.3.1)

The service and instrument air systems provide a continuous supply of compressed air of suitable quality and pressure for instrument control and general plant use. The service air compressor and the instrument air compressors discharge into their respective air receivers. The air is then distributed throughout the plant. Instrument air is additionally filtered and dried prior to distribution throughout the plant.

1.2.2.15.10 Makeup Demineralized Water System(Subsection 9.2.3)

Potable water is demineralized by the makeup demineralizer system and is stored in the demineralized water storage tank.

1.2.2.15.11 Potable Water System (Subsection 9.2.4)

The potable water system provides the necessary supply of domestic water for the plant. The potable water is supplied by the Frenchtown Township Water Supply System to meet drinking water standards.

1.2.2.15.12 Plant Equipment and Floor Drainage(Subsection 9.3.3)

The equipment and floor drainage system is designed to collect liquid waste throughout the plant and discharge the radioactive waste to the radwaste system for processing. Separate drainage facilities are provided for nonradioactive waste.

The drainage system is also used to detect abnormal leakage in the ESF rooms.

1.2.2.15.13 Process Sampling Systems (Subsection 9.3.2)

The process sampling system provides process information that is required to monitor plant conditions and equipment performance. Representative liquid and gas samples are taken automatically and/or manually during normal plant operation for laboratory or on-line analyses.

1.2.2.15.14 Plant Communication Systems (Subsection 9.5.2)

Plant communications consist of a Hi-Comm system of loudspeakers and hand sets, two-way radio units on a unique wavelength, and main control room phones (hard-wired units) that use local phone jack connections at instrument panels and other selected areas.

1.2.2.15.15 Fire Protection System (Subsection 9.5.1 and Appendix 9A)

The FPS is designed to provide an adequate supply of water, CO_2, Halon, or chemicals to points throughout the plant area where fire protection may be required. Diversified fire alarm and fire suppression types are selected to suit the particular areas or hazards being protected. The water for the system is taken from Lake Erie, and constant pressure is provided by the FPS jockey pump. One electrically driven pump, one diesel-engine-driven pump, and the associated piping, valves, and hydrants are provided.

Chemical fire-fighting systems (portable extinguishers) are also provided as additions to, or in lieu of, the water fire-fighting system and the CO_2 and Halon flooding systems. The necessary instrumentation and controls are provided for the proper operation of the fire-fighting systems and for fire detection and annunciation.

1.2.2.15.16 Auxiliary Steam Boilers (Subsection 9.4.8)

The two auxiliary steam boilers are designed to provide low pressure steam for plant heating and to the radwaste evaporators. The boilers and their associated auxiliary equipment are located in the auxiliary boiler house. The boilers may be operated from the main control room.

Each boiler is designed to provide 50,000 lb/hr of l20-psia steam. Combined capacity of the two boilers will provide sufficient heating and radwaste evaporator steam during a shutdown for refueling.

1.2.2.15.17 Condensate Storage and Transfer System (Subsection 9.2.6)

The condensate storage and transfer system (CSTS) is designed to store and distribute condensate and demineralized water throughout the plant during normal and shutdown plant conditions. The condensate storage and return tanks are arranged to permit gravity feed to the condensate supply pumps and to the HPCI, RCIC, CRD, standby feedwater (SBFW), and core spray systems. During normal station operation, hotwell level is raised as necessary by vacuum dragging water to the hotwell from the CST or CRT. When the plant is shutdown, or when a greater flow is required, the normal, or if necessary the emergency, hotwell supply pumps will start and stop automatically depending on hotwell level.

The makeup demineralized storage tank feeds demineralized water transfer pumps, which supply water to the demineralized water service risers and the condensate storage tank.

1.2.2.15.18 Primary Containment Air Cooling and Handling System (Subsection 9.4.5)

The drywell cooling system's primary function is to maintain the temperature of the drywell atmosphere within design conditions. The system uses air-to-water cooling coils with water being supplied by the RBCCW system during normal operating conditions and by the EECW system during abnormal conditions. However, high drywell pressure will automatically close the EECW supply line outboard containment isolation valves.

1.2.2.16 Shielding (Section 12.1)

Shielding is designed so that the dose to personnel manning the main control room and the technical support center during the course of a postulated LOCA is less than 5 rem to the whole body, or its equivalent to any part of the body. For those Design Basis Accidents that are reanalyzed in accordance with Regulatory Guide 1.183, the shielding is shown to limit dose to the Control Room and TSC personnel to less than 5 rem TEDE. In addition, the shielding ensures that, during normal operation and plant shutdown for refueling and maintenance, the dose to personnel and the dose at the site boundary will be as low as reasonably achievable (ALARA) and within the limits specified in 10 CFR 20.

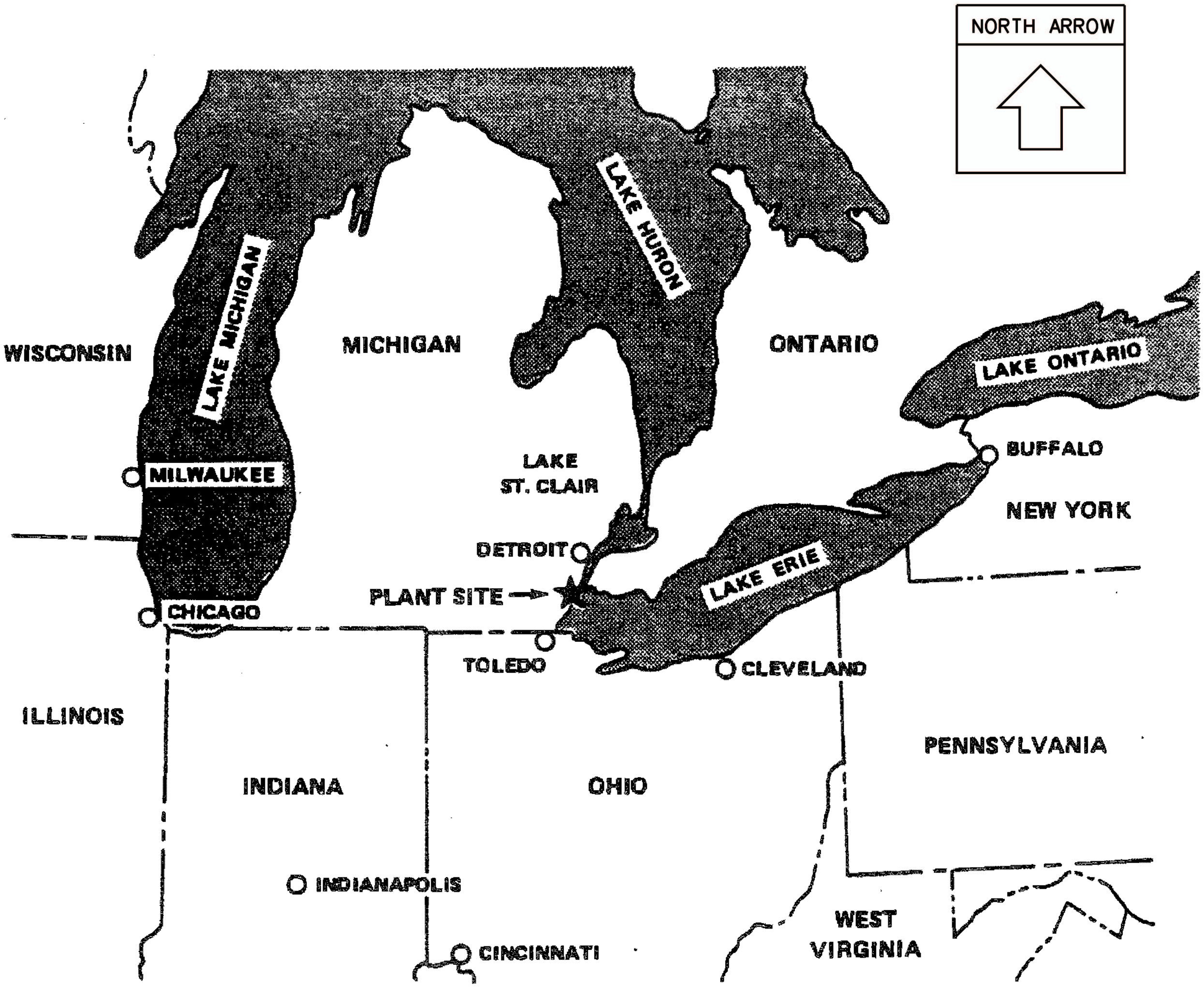

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FIGURE 1.2-1

PLANT SITE LOCATION

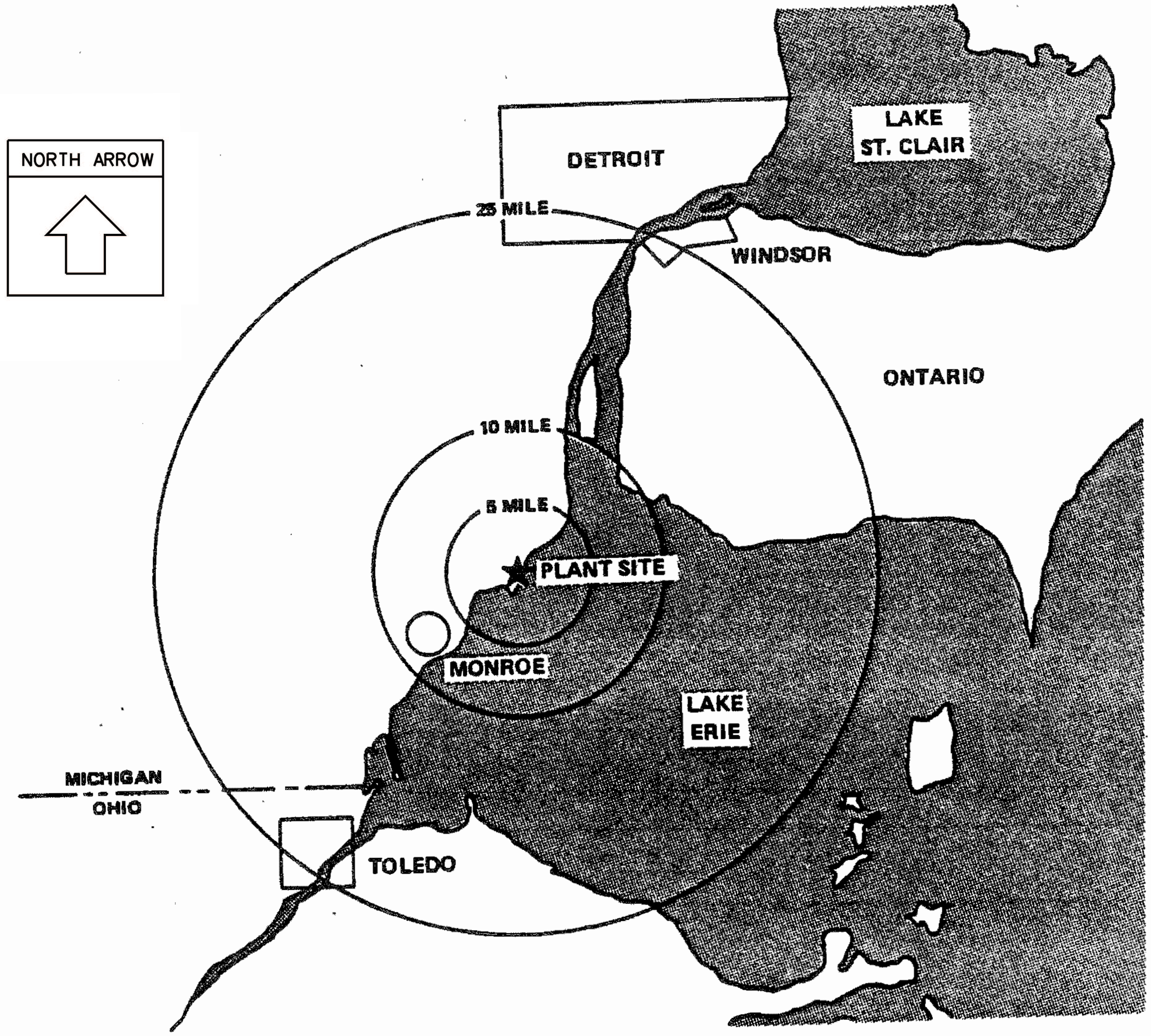

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FIGURE 1.2-2

GENERAL REGION OF THE FERMI SITE

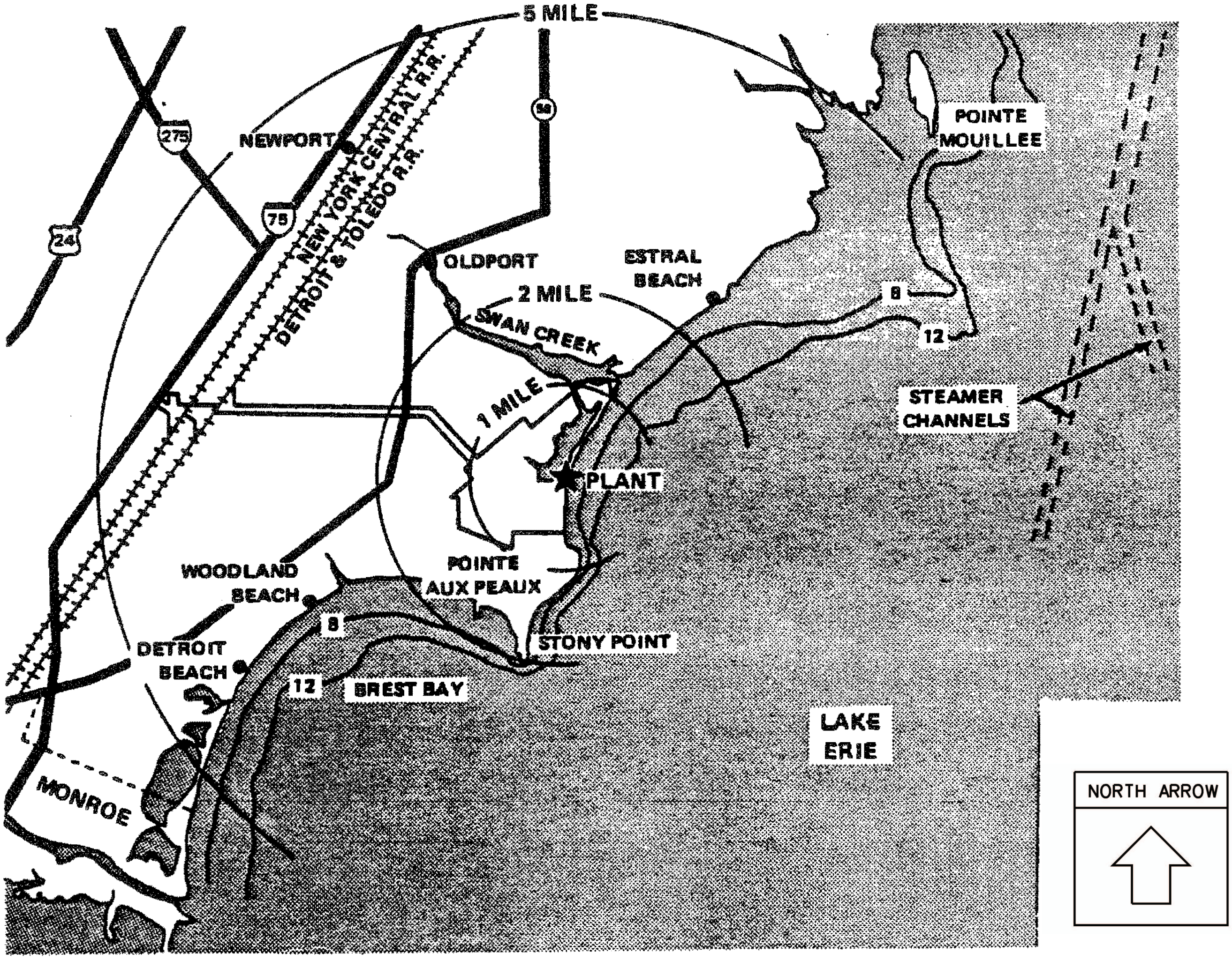

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FIGURE 1.2-3

IMMEDIATE SITE AREA

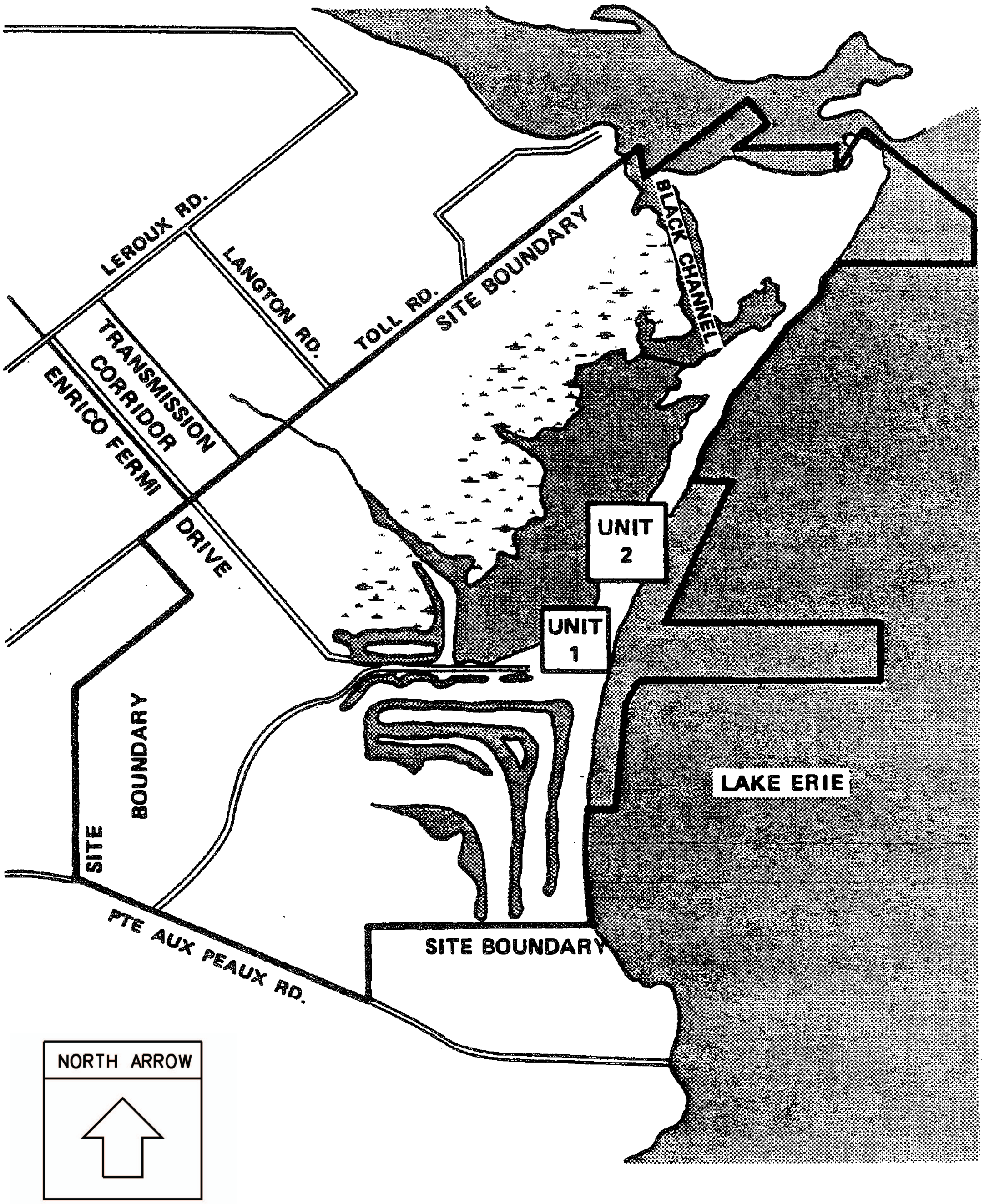

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FIGURE 1.2-4

SITE BOUNDARY

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Refer to Plant Drawing A-2102

Figure Intentionally Removed
Refer to Plant Drawing A-2080

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FIGURE 1.2-6

GENERAL ARRANGEMENT DRAWING
SUBBASEMENT, REACTOR BUILDING, AND
HIGH-PRESSURE COOLANT INJECTION ROOM
ELEVATION 540.0 FT

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Refer to Plant Drawing A-2080

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FIGURE 1.2-7

GENERAL ARRANGEMENT DRAWING - BASEMENT
REACTOR BUILDING ELEVATION 562.0 FT,
TURBINE BUILDING ELEVATION 564.0 FT, AND
RADWASTE BUILDING ELEVATION 557.5 FT

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Refer to Plant Drawing A-2081

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FIGURE 1.2-8

GENERAL ARRANGEMENT DRAWIING
FIRST FLOOR, REACTOR BUILDING

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Refer to Plant Drawing A-2081

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FIGURE 1.2-9

GENERAL ARRANGEMENT DRAWING
FIRST FLOOR, TURBINE BUILDING
FLOOR ELEVATION 583.5 FT

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Refer to Plant Drawing A-2082

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FIGURE 1.2-10

GENERAL ARRANGEMENT DRAWING
SECOND AND MEZZANINE LEVELS
REACTOR AND TURBINE BUILDING

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FIGURE 1.2-11

GENERAL ARRANGEMENT DRAWING
SECOND FLOOR, TURBINE BUILDING

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Refer to Plant Drawing A-2082

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FIGURE 1.2-12

GENERAL ARRANGEMENT DRAWING
SECOND FLOOR, MEZZANINES
RADWASTE BUILDING

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Refer to Plant Drawing A-2083

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FIGURE 1.2-13

GENERAL ARRANGEMENT DRAWING
THIRD FLOOR, REACTOR BUILDING
FLOOR ELEVATION 643.5 FT

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FIGURE 1.2-44

GENERAL ARRANGEMENT DRAWING
THIRD FLOOR, TURBINE BUILDING

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FIGURE 1.2-15

GENERAL ARRANGEMENT DRAWING
FOURTH FLOOR, REACTOR BUILDING
FLOOR ELEVATION 659.5 FT

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Refer to Plant Drawing A-2084

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FIGURE 1.2-16

GENERAL ARRANGEMENT DRAWING
FOURTH FLOOR, TURBINE BUILDING
FLOOR ELEVATION 659.5 FT

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Refer to Plant Drawing A-2085

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FIGURE 1.2-17

GENERAL ARRANGEMENT DRAWING
FIFTH FLOOR, REACTOR BUILDING
ELEVATION 677.5 FT AND 684.5 FT

Figure Intentionally Removed
Refer to Plant Drawing A-2085

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FIGURE 1.2-18

GENERAL ARRANGEMENT DRAWING
FIFTH FLOOR, TURBINE BUILDING
ELEVATION 677.5 AND 684.5 FT

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Refer to Plant Drawing A-2086

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FIGURE 1.2-19

GENERAL ARRANGEMENT DRAWING
ROOF PLANS, TURBINE BUILDING

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Refer to Plant Drawing A-2042

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FIGURE 1.2-20

GENERAL ARRANGEMENT DRAWING
TRANSVERSE SECTION

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Refer to Plant Drawing A-2043

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FIGURE 1.2-21

GENERAL ARRANGEMENT DRAWING
LONGITUDINAL SECTION

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Refer to Plant Drawing A-2035

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FIGURE 1.2-22

GENERAL ARRANGEMENT DRAWING
RADWASTE BUILDING, SECTION "A-A"

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Refer to Plant Drawing A-2034

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FIGURE 1.2-23

GENERAL ARRANGEMENT DRAWING
RADWASTE BUILDING. SECTION "B-B"

Figure Intentionally Removed
Refer to Plant Drawing A-2034

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FIGURE 1.2-24

GENERAL ARRANGEMENT DRAWING
RADWASTE BUILDING
SECTIONS "C-C AND D-D"

Figure Intentionally Removed
Refer to Plant Drawing M-N-2026

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FIGURE 1.2-25

GENERAL ARRANGEMENT DRAWING
RESIDUAL HEAT REMOVAL COMPLEX
BASEMENT FLOOR ELEVATION 562.0 FT

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Refer to Plant Drawing M-N-2027

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FIGURE 1.2-26

GENERAL ARRANGEMENT DRAWING
RESIDUAL HEAT REMOVAL COMPLEX, GRADE
FLOOR PLAN

Figure Intentionally Removed
Refer to Plant Drawing M-N-2028

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FIGURE 1.2-27

GENERAL ARRANGEMENT DRAWING
RESIDUAL HEAT REMOVAL COMPLEX
UPPER FLOOR ROOF ELEVATION 617.0 FT

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Refer to Plant Drawing M-N-2029

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FIGURE 1.2-28

GENERAL ARRANGEMENT DRAWING
RESIDUAL HEAT REMOVAL COMPLEX
ROOF PLAN

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Refer to Plant Drawing M-N-2030

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FIGURE 1.2-29

GENERAL ARRANGEMENT DRAWING
RESIDUAL HEAT REMOVAL COMPLEX
SECTIONS "A-A" AND "B-B"

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Refer to Plant Drawing M-N-2031

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FIGURE 1.2-30

GENERAL ARRANGEMENT DRAWING
RESIDUAL HEAT REMOVAL COMPLEX
SECTION "C-C"

Figure Intentionally Removed
Refer to Plant Drawing M-N-2032

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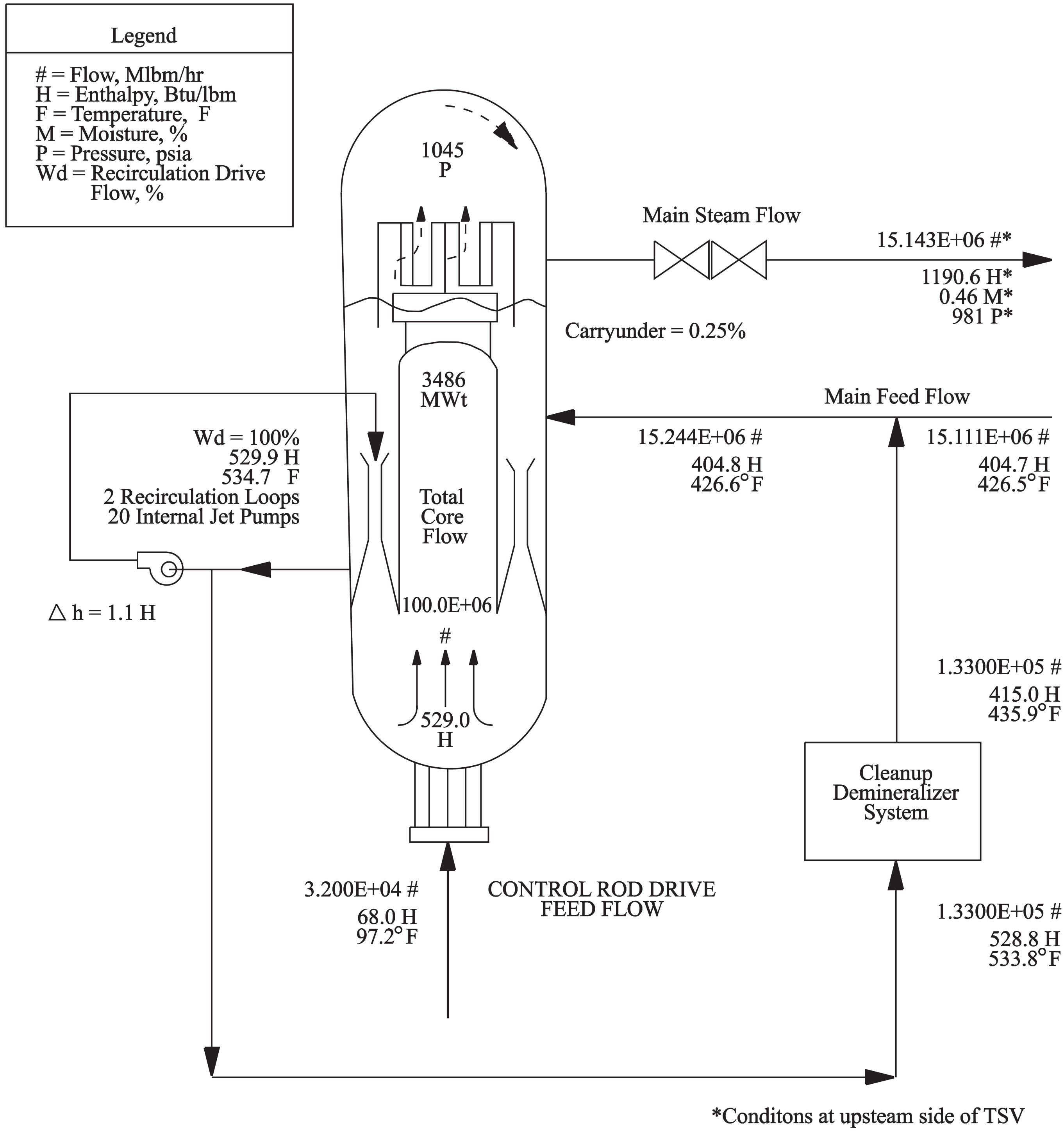
FIGURE 1.2-31

GENERAL ARRANGEMENT DRAWING
RESIDUAL HEAT REMOVAL COMPLEX
SECTION "D-D"



Core Thermal Power	3486.0
Pump Heating	10.6
Cleanup Losses	-4.4
Other System Losses	-1.2
Turbine Cycle Use	3491.0 MWt

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FIGURE 1.2-32

GE REACTOR SYSTEM HEAT BALANCE
RATED PERFORMANCE

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Refer to Plant Drawing C1C OUT

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FIGURE 1.2-33

GEC TURBINE - GENERATOR HEAT BALANCE
AT 100 PERCENT DESIGN FLOW

1.3 COMPARISON TABLES

This section highlights the principal design features of Fermi 2, and provides a comparison of its major features with other BWR facilities for which license applications had been made under 10 CFR 50 at the time of submittal of the original Fermi 2 FSAR.

The design of this facility was based on proven technology attained during the development, design, construction, and operation of BWRs of similar types. The data, performance characteristics, and other information presented herein are subject to revisions as the design of the referenced facilities evolves. However, the information presented is adequate for general comparison purposes and thus will not be subsequently revised.

1.3.1. Comparisons With Similar Facilities Designs

The similar facilities used for comparison are: (l) Brunswick Steam Electric Plant Units 1 and 2; (2) Browns Ferry Nuclear Plant Units 1, 2, and 3; (3) Cooper Nuclear Station; and (4) Edwin I. Hatch Unit No. 1. Of these facilities, Browns Ferry 1, 2, and 3 received operating permits on June 26, 1973, June 28, 1974, and July 2, 1976, respectively. Cooper received an operating permit on January 18, 1974. Hatch received an operating permit on August 6, 1974.

1.3.2. Nuclear System Design Characteristics

Table 1.3-1 summarizes the original design and operating characteristics of Fermi 2, as well as those of the similar facilities discussed in Subsection 1.3.1.

1.3.3. Power Conversion Systems Design Characteristics

Table 1.3-2 compares the original power conversion systems design characteristics of Fermi 2 with those of the similar facilities discussed in Subsection 1.3.1.

TABLE 1.3-1 NUCLEAR PLANTS PRINCIPAL PLANT DESIGN FEATURES COMPARISON[a]

	Fermi 2	Brunswick Units 1 & 2	Browns Ferry Units 1, 2, & 3	Cooper	Edwin I. Hatch Unit 1
Site					
Location	Monroe County, Michigan	Brunswick County, N. Carolina	Limestone County, Alabama	Nemaha County, Nebraska	Appling County, Georgia
Size of site, acres	1120	1200	840	1090	2100
Site ownership	Edison	CP&L	U.S. Government	CPPD	GPC
Plant ownership	Edison	CP&L	TVA	CPPD	GPC
Number of units on site	1	2	3	1	1
Plant-reactor warranted conditions					
Net electrical output, MWe	1093	821/unit	1075/unit	770	786
Gross electrical output, MWe	1154	847/unit	1098/unit	801	813
Turbine heat rate, Btu/kWh	(proprietary)	9816	10,231	10,142	10,218
Gross plant heat rate, Btu/kWh	10,296 net	10,120	10,243	10,187	10,227
Feedwater temperature, °F	420	420	376.1	367	387.4
Reactor pressure vessel					
Inside diameter, in.	251	218	251	218	218
Overall length inside, ft-in.	72-0	69-4	72-0	69-4	69-4
Design pressure, psig	1250	1250	1250	1250	1250
Wall thickness, in. (including clad)	6-7/16	5-17/32	6-5/16	5-17/32	5-17/32
Reactor coolant recirculation loops					
Location of recirculation loops	Primary containment system drywell structure	Primary containment system drywell structure	Primary containment system drywell structure	Primary containment system drywell structure	Primary containment system drywell structure
Number of recirculation loops	2	2	2	2	2
Pipe size, in.	28	28	28	28	28
Pump capacity (each), gpm	45,200	45,200	45,000	45,200	45,200
Number of jet pumps	20	20	20	20	20
Location of jet pumps	Inside reactor primary vessel	Inside reactor primary vessel	Inside reactor primary vessel	Inside reactor primary vessel	Inside reactor primary vessel
Reactor					

TABLE 1.3-1 NUCLEAR PLANTS PRINCIPAL PLANT DESIGN FEATURES COMPARISON[a]

	Fermi 2	Brunswick Units 1 & 2	Browns Ferry Units 1, 2, & 3	Cooper	Edwin I. Hatch Unit 1
Reactor warranted conditions					
Thermal output, MWt	3292	2436	3293	2381	2436
Reactor operating pressure, psig (steam dome)	1005	1005	1005	1005	1005
Total reactor core flow rate, lbs/hr	100.0×10^6	77×10^6	102.5×10^6	73.5×10^6	78.5×10^6
Main steam flow rate, lb/hr (warranted)	14.156×10^6	10.47×10^6	13.36×10^6	9.551×10^6	10.03×10^6
Reactor core description					
Lattice	8 x 8	7 x 7	7 x 7	7 x 7	7 x 7
Pitch of movable control rods, in.	12.0	12.0	12.0	12.0	12.0
Number of fuel assemblies	764	560	764	548	560
Number of movable control rods	185	137	185	137	137
Effective active fuel length, in	150	144	144	146	144
Equivalent reactor core diameter, in.	187.1	160.2	187.1	158.5	160.2
Circumscribed reactor core diameter, in.	198	169.7	197.8	169.7	169.7
Total weight UO_2, lb	348,904	272,850	372,373	267,095	272,850
Reactor fuel description					
Fuel material	UO_2	UO_2	UO_2	UO_2	UO_2
Fuel density, percent of theoretical	95	95	95	95	95
Fuel pellet diameter, in.	0.410	0.487	0.487	0.487	0.487
Fuel rod cladding material	Zircaloy-2	Zircaloy-2	Zircaloy-2	Zircaloy-2	Zircaloy-2
Fuel rod cladding thickness, in.	0.032	0.037	0.032/0.037	0.032/0.037	0.037
Fuel rod cladding process	Freestanding loaded tubes	Freestanding loaded tubes	Freestanding loaded tubes	Freestanding loaded tubes	Freestanding loaded tubes
Fuel rod outside diameter, in.	0.483	0.563	0.563	0.563	0.563
Length of gas plenum, in.	10.0	16.0	16.0	16.0	16.0

TABLE 1.3-1 NUCLEAR PLANTS PRINCIPAL PLANT DESIGN FEATURES COMPARISON[a]

	Fermi 2	Brunswick Units 1 & 2	Browns Ferry Units 1, 2, & 3	Cooper	Edwin I. Hatch Unit 1
Fuel rod pitch, in.	0.640	0.738	0.738	0.738	0.738
Fuel assembly channel material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4
Reactor control					
Control rods					
Number	185	137	185	137	137
Shape	Cruciform	Cruciform	Cruciform	Cruciform	Cruciform
Material	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes
Pitch, in.	12.0	12.0	12.0	12.0	12.0
Poison length, in.	143.0	143.0	143.0	143.0	143.0
Blade span, in.	9.75	9.75	9.75	9.75	9.75
Number of control material tubes for rod	76	76	76	76	76
Tube dimensions, in.	0.188 O.D. x 0.025-wall	0.188 O.D. x 0.025-wall	0.188 O.D. x 0.025-wall	0.188 O.D. x 0.025-wall	0.188 O.D. x 0.025-wall
Stroke, in.	144.0	144.0	144.0	144.0	144.0
Thermal-hydraulic data					
Heat transfer area per assembly, ft^2	97.998	86.513	86.513	86.513	86.513
Reactor core heat transfer area, ft^2	74,871	48,447	66,096	47,409	48,447
Maximum heat flux[b] Btu/hr ft^2	361,590	428,400	428,400	428,400	428,400
Average heat flux[b] Btu/hr ft^2	143,700	164,700	163,310	164,470	164,700
Maximum power per fuel rod unit length[b], kW/ft	13.4	18.5	18.5	18.5	18.5
Average power per fuel rod unit length[b], kW/ft	5.3	7.10	7.04	7.09	7.10
Maximum fuel temperature, °F	3435	4380	4380	4380	4380
Total heat generated in fuel	96	96	96	96	96
Core average exit quality	14.1	13.5	12.9	12.9	12.7
Power distribution - peaking factors (peak/average)					

TABLE 1.3-1 NUCLEAR PLANTS PRINCIPAL PLANT DESIGN FEATURES COMPARISON[a]

	Fermi 2	Brunswick Units 1 & 2	Browns Ferry Units 1, 2, & 3	Cooper	Edwin I. Hatch Unit 1
Axial	1.40	1.50	1.50	1.50	1.50
Radial assembly	1.40	1.40	1.40	1.40	1.40
Local (within assembly)	1.24	1.24	1.24	1.24	1.24
Total peaking factor	2.43	2.6	2.6	2.6	2.6
Nuclear design data					
Average discharge exposure - 1st core, Mwd/ST	16,204	19,000	19,000	19,000	19,000
Moderator to fuel volume ratio at total core H_2O/UO_2 cold	2.74	2.41	2.45	2.41	2.41
In-core neutron instrumentation					
Number of in-core neutron detectors (LPRM)[c]	172	124	172	124	124
Number of in-core detector strings (LPRM)[c]	43	31	43	31	31
Number of detectors per string	4	4	4	4	4
Number of traversing in-core probe detectors	5	4	5	4	4
Range (and number) of detectors					
Source range monitor	Source to 10^{-3}% power (4)	Source to 10^{-3}% power (4)	Source to 10^{-3}% power (4)	Source to 10^{-3}% power (4)	Source to 10^{-3}% power (4)
Intermediate range monitor	10^{-4}% to 10% power (8)	10^{-4}% to 10% power (8)	10^{-4}% to 10% power (8)	10^{-4}% to 10% power (8)	10^{-4}% to 10% power (8)
Local power range monitor	2.5% to 125% power (172)	2.5% to 125% power (124)	2.5% to 125% power (172)	2.5% to 125% power (124)	2.5% to 125% power (124)
Average power range monitor	2.5% to 125% power (6)[d]	2.5% to 125% power (6)[d]	2.5% to 125% power (6)[d]	2.5% to 125% power (6)[d]	2.5% to 125% power (6)[d]
Number and type of in-core neutron sources	7-Sb-Be	5-Sb-Be	7-Sb-Be	5-Sb-Be	5-Sb-Be
Reactivity control					
Approximate effective reactivity of core with all control rods in (cold)	~0.975k	0.96k	0.96k	0.96k	0.96k

TABLE 1.3-1 NUCLEAR PLANTS PRINCIPAL PLANT DESIGN FEATURES COMPARISON[a]

	Fermi 2	Brunswick Units 1 & 2	Browns Ferry Units 1, 2, & 3	Cooper	Edwin I. Hatch Unit 1
Effective reactivity of core with strongest control rod out (cold)	<0.99k	<0.99k	<0.99k	<0.99k	<0.99k
Typical moderator temperature coefficient (Δk/k°F)[e]					
Cold (at 68°F)	-5.0×10^{-5}	-5.0×10^{-5}	-5.0×10^{-5}	-5.0×10^{-5}	-5.0×10^{-5}
Hot (no voids)	-39.0×10^{-5}	-39.0×10^{-5}	-39.0×10^{-5}	-39.0×10^{-5}	-39.0×10^{-5}
Typical moderator void coefficient (Δk/k% void)					
Hot (no voids)	-1.0×10^{-3}	-1.0×10^{-3}	-1.0×10^{-3}	-1.0×10^{-3}	-1.0×10^{-3}
At rated output	-1.6×10^{-3}	-1.6×10^{-3}	-1.6×10^{-3}	-1.6×10^{-3}	-1.6×10^{-3}
Typical fuel temperature (Doppler) coefficient (k/k°F)[e]					
Cold (at 68°F)	-1.3×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}
Hot (no voids)	-1.2×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}
At rated output	$\leq -1.3 \times 10^{-5}$	$\leq -1.3 \times 10^{-5}$	$\leq -1.3 \times 10^{-5}$	$\leq -1.3 \times 10^{-5}$	$\leq -1.3 \times 10^{-5}$
Containment systems					
Primary containment					
Type	Pressure suppression	Pressure suppression	Pressure suppression	Pressure suppression	Pressure suppression
Construction					
Drywell	Light bulb/ steel vessel	Light bulb/ reinforced concrete with steel liner	Light bulb/ steel vessel	Light bulb/ steel vessel	Light bulb/ steel vessel
Pressure suppression chamber	Torus/steel vessel	Torus/reinforced concrete with steel liner	Torus/steel vessel	Torus/steel vessel	Torus/steel vessel
Pressure suppression chamber-internal design pressure, psig	+56	+62	+56	+56	+56
Pressure suppression chamber-external design pressure, psig	+2	+2	+1	+2	+2
Drywell-internal design pressure, psi	+56	+62	+56	+56	+56
Drywell-external design pressure, psig	+2	+2	+1	+2	+2
Drywell free volume, ft^3	163,730	164,100	159,000	145,430	146,240

TABLE 1.3-1 NUCLEAR PLANTS PRINCIPAL PLANT DESIGN FEATURES COMPARISON[a]

	Fermi 2	Brunswick Units 1 & 2	Browns Ferry Units 1, 2, & 3	Cooper	Edwin I. Hatch Unit 1
Pressure suppression chamber free volume, ft^3	127,760 (min)	124,000	119,000	109,810	110,950
Pressure suppression pool water volume, ft^3	121,080 (min)	87,600	85,000	87,660	87,660
Submergence of vent pipe below pressure pool surface, ft-in	4-0	4-0	4-0	4-0	3-8
Design temperature of drywell, °F	340	300	281	281	281
Design temperature of pressure suppression chamber, °F	281	220	281	281	281
Downcomer vent pressure loss factor	6.21	6.21	6.21	6.21	6.21
Break area/gross vent area	0.019	0.02	0.019	0.019	0.019
Drywell free volume/pressure suppression chamber free volume	1.25	1.32	1.33	1.4	1.3
Calculated maximum drywell pressure after blowdown with no pre-purge, psig	56.5	49.4	40.0	46.0	46.5
Leakage rate, percent free volume per day	0.5	0.5	0.5	0.5	1.2
Secondary containment					
Type	Controlled leakage, rooftop release	Controlled leakage, elevated release	Controlled leakage, elevated release	Controlled leakage, elevated release	Controlled leakage, elevated release
Construction					
Lower levels	Reinforced concrete	Reinforced concrete	Reinforced concrete	Reinforced concrete	Reinforced concrete
Upper levels	Steel super-structure and siding	Steel super-structure and siding	Steel super- structure and siding	Steel super-structure and siding	Steel super- structure and siding
Roof	Metal decking with built-up roofing	Metal decking with built-up roofing	Steel sheeting	Steel sheeting	Steel sheeting
Internal design pressure, psig	0.25	0.25	0.25	0.25	0.25

TABLE 1.3-1 NUCLEAR PLANTS PRINCIPAL PLANT DESIGN FEATURES COMPARISON[a]

	Fermi 2	Brunswick Units 1 & 2	Browns Ferry Units 1, 2, & 3	Cooper	Edwin I. Hatch Unit 1
Design in leakage rate, percent free volume/day at 0.25 in. $H_2 0$	100.0	100.0	100.0	100.0	100.0
Elevated release point					
Type	Rooftop	Stack	Stack	Stack	Stack
Construction	Steel	Reinforced concrete	Steel	Steel	Reinforced concrete
Height (above ground), meters	54.1	100.0	200.0	100.0	150.0
Plant auxiliary systems					
Emergency core cooling systems					
Reactor core spray cooling system	2 loops	2 loops	2 loops	2 loops	2 loops
High pressure coolant injection system	1 pump	1 pump	1 pump	1 pump	1 pump
Auto-relief system	1	1	1	1	1
Residual heat removal system					
Low pressure coolant injection subsystem	4 pumps	4 pumps	4 pumps	4 pumps	4 pumps
Primary containment spray/cooling subsystem	2 redundant loops	2 redundant loops	2 redundant loops	2 redundant loops	2 redundant loops
Reactor shutdown cooling subsystem	1	1	1	1	1
Reactor auxiliary systems					
Spent fuel pool cooling and demineralizing system	1	1	1	1	1
Reactor cleanup demineralizer system	1	1	1	1	1
Reactor core isolation cooling system	1	1	1	1	1
Plant electrical power systems					
Transmission system					
Outgoing lines (number-rating)	2-345 kV	8-230 kV	4-500 kV	4-345 kV	5-230 kV
Auxiliary power systems					

TABLE 1.3-1 NUCLEAR PLANTS PRINCIPAL PLANT DESIGN FEATURES COMPARISON[a]

	Fermi 2	Brunswick Units 1 & 2	Browns Ferry Units 1, 2, & 3	Cooper	Edwin I. Hatch Unit 1
Incoming lines (number-rating)	3-120 kV 4-345 kV	8-230 kV	2-161 kV	1-69 kV 1-115 kV	5-230 kV
Onsite Sources					
Auxiliary transformers	2	2	3	1	1
Startup transformers	0	2	2	2	2
Shutdown transformers	0	0	0	1	1
Emergency diesel generator system					
Number of diesel generators	4	4	4	4	3

[a] Original design information provided for comparison purposes only. Not intended to be updated. For current Fermi 2 information, refer to main body of UFSAR.
[b] Items are shown at design limits rather than design points.
[c] Local power range monitor.
[d] Represents six channels.
[e] Beginning of core life.

TABLE 1.3-2 COMPARISON OF POWER CONVERSION SYSTEMS DESIGN CHARACTERISTICS[a]

	Fermi 2	Brunswick Units 1 & 2	Browns Ferry Units 1, 2, & 3	Cooper	Edwin I. Hatch Unit 1
Turbine generator					
Rated generator output, MWe	1154 Tandem compound 6-flow/46 1 high pressure, 3 low pressure	849 Tandem compound 4-flow/43 1 high pressure, 2 low pressure	1152 Tandem compound 6-flow/43 1 high pressure, 3 low pressure	836 Tandem compound 2-flow/44 1 high pressure, 2 low pressure	819 Tandem compound 2-flow/43 1 high pressure, 2 low pressure
Steam conditions at throttle valve					
Flow, lb/hr	14.156×10^6	10.46×10^6	13.38×10^6	9.81×10^6	10.03×10^6
Pressure, psia	965	965	965	970	970
Temperature, °F	540.3	540.3	540.3	540.9	540.9
Moisture content, percent	0.41	0.41	0.28	0.32	0.32
Turbine cylinder arrangement					
Steam reheat stages, no.	1	2	0	0	1
Feedwater heating stages, no.	6	5	5	5	5
Strings of feedwater heaters, no.	2/3	2	2	2	2
Heaters in condenser necks, no.	2	2	2	2	2
Heater drain system	Pumped forward	Pumped forward	Pressure differential	Pumped forward	Pressure differential
Condensate pumps, no.	3	3	3	3	3
Heater feed pumps, no.	3	3	3	3	3
Header drain pumps, no	3	2	0	3	0
Reactor feed pumps, no.	2	2	3	2	2
Main Steam Lines					
Steam lines, no.	4	4	4	4	4
Design pressure, psig	1250	1146	1146	1146	1146
Design Temperature, °F	575	563	563	563	563
Pipe Diameter, in.	24	24	26	24	24
Pipe material	Carbon steel	Carbon steel	Carbon steel	Carbon steel	Carbon steel
Main steam line bypass capacity, percent	25	25 (unit 1) 105 (unit 2)	25	25	25
Final feedwater temperature, °F	420	420	376.1	367	387.4
Condenser					
Type	Single pressure	Single pressure	Single pressure	Single pressure	Single pressure
Condenser shells, no.	2	2	3	2	2
Design pressure, in. Hg abs	1.5	1.5	2.0	2.0	3.37
Total condenser duty, Btu/hr	7.547×10^9	5.6×10^9	7.77×10^9	5.6×10^9	5.8×10^9
Circulating water system					
Type	Closed/ND cooling towers (2)	Open	Open	Open	Closed/ND cooling Towers (2)
Flow, gpm	9×10^5	6.24×10^5	6.3×10^5		5.55×10^5
Circulating water pumps, no.	5	4	3	4	3

[a] Original design information provided for comparison purposes only. Not intended to be updated. For current Fermi 2 information, refer to main body of UFSAR.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 The Detroit Edison Company

The Detroit Edison Company changed its name to DTE Electric Company as of January 1, 2013. The name change to DTE Electric Company was purely administrative in nature; the legal entity remained the same and the name change did not involve a transfer of control or of an interest in the license for Fermi 2. DTE Electric Company continues to be a wholly owned subsidiary of DTE Energy Company. For the purposes of the Fermi 2 UFSAR, except for UFSAR sections of historical context, all DTE Energy Company designations referenced throughout the UFSAR (e.g. DTE Electric, Edison, Detroit Edison, DECo, etc.) are synonymous.

Edison is the sole owner of Fermi 2 and, as such, is responsible for the design, construction, and operation of the facility. Edison is the architect-engineer for Fermi 2.

Edison employed an engineering, design, and construction supervision staff. Many of the key engineering personnel had had previous nuclear experience, primarily on the design, construction, and operation of fast breeder reactor Fermi 1, and, subsequently, in the design and construction of Fermi 2.

Edison has extensive power plant design and development experience, having acted as architect-engineer on the majority of its own power generating facilities.

To ensure competence in all areas of Fermi 2 design and construction, Edison retained various principal agents and contractors.

1.4.2 Sargent & Lundy

Sargent & Lundy (S&L) was retained for the civil, structural, and architectural design of the reactor building and other areas of the plant where that firm's experience was especially appropriate. These include preparation of the specifications for the primary containment vessel, certain electrical design tools, and piping system analyses. By a separate contract, S&L was responsible for the design of the residual heat removal (RHR) complex.

Sargent & Lundy had specialized in consulting and design engineering for the generation, transmission, and distribution of electric power for three-quarters of a century. They had provided engineering services for l5 percent of the nation's investor-owned electric generating capacity. More than 650 turbine generator units with a total capacity of more than 70,000 MWe had been put in operation or were on order; of this total more than 21,800 MWe was nuclear generating capacity, the majority of which was of the water reactor type. Sargent & Lundy had been actively engaged in the nuclear power plant field since its inception.

1.4.3 Stone & Webster Engineering Corporation

Stone & Webster Michigan, Incorporated (S&W), a wholly-owned subsidiary of Stone & Webster, Incorporated, was retained and assigned responsibility for completion of certain engineering and design tasks commencing in January 1978. Some of the major tasks included design of the plant security system, high density fuel racks, pipe hanger design assistance, nonnuclear steam supply, integrated leak-rate testing, and review of seismic requirements. Stone

& Webster also provided assistance in the general areas of licensing requirements, advisory operations, and various electrical, mechanical, and instrument and control activities.

Stone & Webster is an engineering and construction firm serving the electric utility industry in the design and construction of all types of power stations. Stone & Webster had provided engineering services related to generating capacity in excess of 70,000,000 kW. Stone & Webster had been actively engaged in engineering and construction of nuclear power plants since 1954. Over 26,000,000 kW of generating capacity had been associated with S&W's nuclear engineering services.

1.4.4 General Electric Company

General Electric (GE) was contracted to design, fabricate, and deliver the single-cycle boiling water nuclear steam supply system (NSSS), fabricate the first core of nuclear fuel, and provide technical direction for installation and startup of this equipment. General Electric had been engaged in the development, design, construction, and operation of BWRs since l955. Thus, GE had substantial experience, knowledge, and capability to design, manufacture, and furnish technical advice for the installation and startup of the reactor.

GE was later contracted to design, fabricate and deliver a replacement for the LP Turbine Steam Path installed during RF05 and the HP Turbine System Path installed during RF07.

1.4.5 General Electric Company Turbine-Generator, Ltd.

General Electric Company (GEC) Turbine-Generator, Ltd. of Rugby, England, was responsible for the design, fabrication, and delivery of the turbine generator as well as for providing technical assistance for installation and startup of this equipment. General Electric Company Turbine-Generator, Ltd. had had a long history of fabrication and application of turbine generators in electrical power production facilities.

The LP Turbine Steam Path was replaced during RF05 with GE designed components. The major components replaced were the rotors, diaphragms, associated seals and steam flow guides, including the internal exhaust hood spray piping and nozzles.

The HP Turbine Steam Path was replaced during RF07 with GE designed components. The major components replaced were the rotor, diaphragms, associated seals, and coupling spacers. An inlet snout was added to provide the steam flow path into the first stage diaphragm nozzles.

1.4.6 Other Consultants

1.4.6.1 Dames & Moore

The independent consulting firm of Dames & Moore (D&M) was retained to do hydrology, geology, and seismology studies for Fermi 2. Having performed environmental studies for approximately 50 nuclear power plant sites, D&M was active in the field of environmental engineering related to nuclear power plant construction.

1.4.6.2 NUS Corporation

NUS Corporation was retained to provide software for startup and operation of Fermi 2, and to prepare the environmental report and other environmental and licensing consulting services.

Software for Fermi 2 included administrative documents to govern startup, system descriptions, and preoperational test procedures. NUS was also responsible for preparation of the plant operating manual. NUS also provided environmental consulting services in the areas of aquatic ecology, land and water use, thermal and chemical effects, alternatives, radiological effects, and miscellaneous licensing consulting services as required.

NUS had provided consulting services throughout the world for a wide range of utilities, industries, and governmental organization.

1.4.6.3 Ralph M. Parsons Company

Ralph M. Parsons Company of Michigan (Parsons) was engaged as the general contractor for Fermi 2 with responsibility for overall construction management of the entire facility, and with direct contractual responsibility for field fabrication of small diameter piping, and installation of the plant piping systems and mechanical equipment. Parsons was terminated as general contractor in November 1974.

Under a separate contract, Ralph M. Parsons of Los Angeles was engaged to help establish the initial Quality Assurance (QA) and Quality Control (QC) Organization at the site to work in conjunction with Edison to provide work surveillance, inspection, and documentation services which ensure conformance to the codes and standards applicable to nuclear construction and the design specifications. In addition, Ralph M. Parsons of Los Angeles provided support in seismic and pipe structure analyses and specific engineering assignments.

Parsons was one of the world's largest architectural, engineering, and construction firms. Its world headquarters were located in Los Angeles, California, with principal offices in several foreign countries. The company had demonstrated its total engineering and construction capability in a variety of foreign and domestic industrial, technical, and scientific projects completed for the petroleum refining, metallurgical processing, power generation, aerospace, chemical processing, shipbuilding, commercial transportation, and nuclear industries.

Projects included engineering and construction of rapid transit facilities, transportation systems, water and sewage treatment, desalination plant, petroleum and petrochemical plants, gas processing facilities, marine and port complex, automated shipyard, airports and air terminals, mining and metallurgical facilities, environmental process development, fast breeder nuclear reactor installation, nuclear power plant installation, and many others.

1.4.6.4 Daniel Construction Company

Daniel Construction Company was retained and assigned responsibility for site construction management commencing in November 1974. It maintained that responsibility throughout construction until systems and structures nearing completion were transferred to Edison. Commencing in January 1984, Daniel assisted the Fermi 2 Project Management Organization as needed and was responsible for the day-to-day management of Wismer & Becker, API, and Chicago Bridge and Iron Company. Daniel Construction Company, a division of Daniel International Corporation, of Greenville, S.C., had a wide variety of engineering and construction assignments being completed in many parts of the world. A recent survey of the nation's 400 largest contractors rates Daniel fourth in contract awards, twelfth in international contract awards, and thirty-second in design awards.

Daniel had acquired extensive construction and project management experience in major industrial complexes for the chemical, paper, rubber, textile, aluminum, and power generation industries. These construction services involved the ability to meet precise tolerances and specifications on erection, fabrication, and equipment installation, and required a thorough knowledge of heavy construction, mechanical, electrical, and instrumentation techniques and methods. This experience and the developed capabilities were applicable to the construction of nuclear power facilities.

The Daniel Construction Company Quality Assurance Program for ASME nuclear code construction was evaluated and accepted by an ASME survey team, and the certificate of authorization to perform code construction ("N" stamp) was awarded Daniel following the ASME team audit of field implementation and enforcement.

Daniel's experience included construction of nuclear and fossil fueled power plants. Daniel's first project of this nature was construction of the nuclear power Carolina-Virginia Tube Reactor at Parr, South Carolina. This facility operated several years as a prototype plant. Nuclear power plant construction projects included the following:

a. Joseph M. Farley Nuclear Plant, Unit No. 1 and Unit No. 2, 829-MW PWR each, for Alabama Power Company

b. Virgil Summer Nuclear Power Plant, a 920-MW nuclear power generating plant of the Westinghouse pressurized-water type, for South Carolina Electric and Gas Company

c. Shearon Harris Nuclear Power Plant for Carolina Power & Light Company.

1.4.6.5 EG&G, Inc.

EG&G was engaged to provide site meteorological programs. EG&G has performed a variety of marine, meteorological, biological, hydrological, and climatological analyses, instrumentation selection and application, and a full range of services including field installation, maintenance, data gathering and processing, diffusion modeling, and report preparation for many clients.

1.4.6.6 Bechtel Power Corporation

Bechtel Power Corporation was the general services contractor for the Fermi 2 power plant. Bechtel provided engineering, construction, maintenance, startup assistance, and plant operational support services as mutually agreed to by Edison and Bechtel. The work was performed on Quality Assurance Level 1 or non-quality-related systems within the plant. The governing quality assurance program, either Edison's or Bechtel's, was adhered to depending on the kind and nature of the work for which the services are rendered.

Bechtel had demonstrated its ability in successfully performing construction management, engineering, and other functions in accordance with quality assurance programs under the jurisdiction of the NRC over past years. As such, Bechtel was deemed fully qualified to perform any safety-related work that may be assigned to it by Edison.

1.4.6.7 L. K. Comstock

L. K. Comstock was responsible for furnishing labor, tools, equipment, and materials as required to complete the electrical installation at Fermi 2. Comstock's work included electrical installation at Fermi 2. Comstock's work also included receiving, storing, installing, connecting, and readying for service all electrical equipment as well as providing electrical QA/QC services and design engineering services.

Comstock had extensive experience in the nuclear power field and understood the QA requirements. It had provided construction services on the BWR units at Dresden and Quad Cities and had completed the electrical erection contracts at the Kewaunee, Prairie Island, Cook, and FitzPatrick nuclear projects.

1.4.6.8 Commonwealth Associates, Inc., of Gilbert Commonwealth

Commonwealth Associates, Inc., of Gilbert Commonwealth, was retained in 1981 to provide technical personnel to assist during the construction of Fermi 2 in the Field Engineering, Startup, Nuclear Production, and Quality Assurance Departments.

The personnel provided by Commonwealth had the expertise, gained from work at other utilities, required during Fermi 2 construction and the startup operations.

1.4.6.9 NUTECH Engineers

NUTECH was retained to provide technical assistance to Edison's Engineering Department, on an as-required basis. Subsequently, it provided services to the Nuclear Production Department as well as other areas.

Areas of service provided included (a) In-Service Inspection Program development, (b) In-Service Inspection staff augmentation, (c) Computer Program development, (d) Radiation Emergency Preparedness Program development, and (e) Plant Unique Analysis Program addressing hydrodynamic loads in the containment.

1.4.6.10 Wismer & Becker

Wismer & Becker was responsible for furnishing labor, materials, tools, equipment, and technical and professional services as necessary for the installation of piping and mechanical equipment at Fermi 2. Support provided included QA/QC work and pressure testing on piping systems and equipment as required by the applicable codes and specifications.

For over 30 years, Wismer & Becker had been involved in all phases of power plant construction. Previous nuclear experience from the Council Bluffs and Diablo Canyon nuclear power plants had proved that Wismer & Becker had a thorough understanding of ASME Code Section III work and QA requirements

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

This section is included for historical purposes and will not be further updated. It includes a discussion of Advisory Committee on Reactor Safeguards (ACRS) and AEC staff concerns regarding BWRs, Fermi 2 in particular. These concerns were expressed prior to and during the Fermi 2 Construction Permit period and were required to be resolved prior to or during construction.

1.5.1 Resolved Concerns

The ACRS has voiced various concerns about the development of BWRs. Specific concerns resolved during the development of the BWR, and specific Fermi 2 ACRS concerns and the documents in which each specific concern is resolved, were presented in Appendix B of the original Fermi 2 FSAR.

Although some of the concerns expressed by the ACRS did not directly apply to Fermi 2, they were included in Appendix B as evidence of the refinements and degree of analysis included in the design of the Fermi 2 BWR.

Specific GE development programs to improve the safety and performance of the BWR, and the status as applicable to Fermi 2, are discussed in Subsection 1.5.2.

Additionally, the AEC staff enumerated a number of concerns during the Fermi 2 Construction Permit review that were documented in Appendix D to the original FSAR. Appendix D also included the status of the NRC review and resolution of these Fermi 2 specific items.

1.5.2 General Electric Development Programs

1.5.2.1 Instrumentation for Vibration and Loose Parts Detection

System has been abandoned.

1.5.2.2 Core Spray Distribution

Because of the slight changes in core dimensions and spray sparger geometry from plant to plant, a series of tests was conducted. The purpose of these tests was to ensure that the core spray flow distribution for the Fermi 2 header design would supply adequate cooling water from the core spray system to each fuel assembly within the reactor core in the event of a LOCA. The tests demonstrated that each fuel assembly receives adequate cooling water flow for required spray flow rates between rated flow and runout flow conditions. Details of this test program were very similar to those described in Amendment 30 (December 1967) to the Oyster Creek FSAR, NRC Docket No. 50-219.

1.5.2.3 Vibration Testing of Reactor Internals

The major reactor components within the reactor pressure vessel have been subjected to extensive testing and dynamic analysis to properly describe any flow-induced vibration incurred during normal reactor operation and anticipated operational transients. Extensive prototype testing on BWR 4 plants has been reported in GE Topical Report NEDO-24057. Testing provisions for Fermi 2 invoke this prototype test program as stipulated by Regulatory Guide

1.20, Revision 2. An approved preoperational test was conducted prior to fuel load for flow-induced vibration of reactor internals. Refer to Subsection 3.9.1 for details.

1.5.2.4 Pipe Whip Inside Containment

Dynamic restraint tests have been performed on the plastic design restraints to demonstrate the adequacy of the piping restraint concept. The concept provides clearances that allow for normal thermal movements of the pipe but limit motion in the event of a postulated rupture.

Edison has extensively analyzed the dynamic effects of pipe ruptures inside containment and has installed design provisions including pipe whip restraints to prevent damage caused by pipe whip. Refer to Section 3.6 for details.

1.5.2.5 Recirculation Pump-Motor Missiles

An analysis has been performed on the generation of missiles as a result of a recirculation line break. Based on GE analyses, postulated recirculation pump missiles, which may be generated during a design-basis LOCA overspeed condition, are safely contained within the pump casing. Analyses of pump missiles ejected from the open end of the broken pipe have also been performed. Piping restraints were added to prevent the potential missile exit points in the pipe from developing. Further details and references to GE topical reports are provided in Subsections 3.5.1.2 and 5.5.1.4.

1.5.2.6 Standby Gas Treatment System Filter Efficiency Test

A test program to demonstrate the efficiency of the new gasket-less carbon filter was successfully completed by Edison in 1974. NEDC-12431 (Reference 1) concluded that tests on the filter, simulating the Fermi 2 standby gas treatment system (SGTS) carbon filter, successfully demonstrated the ability of the filter to remove greater than 99.99 percent of the iodine processed through the filter. Thus, the Fermi 2 SGTS can be credited (with adequate conservatism) with an iodine removal efficiency of 95 percent. For additional information on this subject, refer to Subsection 6.2.3.

1.5.2.7 Hydrogen Flammability Tests

The NRC amended 10 CFR 50.44, “Standards for combustible gas control system in light-water-cooled power reactors” on October 16, 2003 to eliminate the requirements for hydrogen recombiners. The hydrogen recombiner Technical Specification requirements were subsequently removed by License Amendment 159, dated March 15, 2004. Regulatory Guide 1.7 was revised in March 2007 to reflect the amended 10 CFR 50.44. The Combustible Gas Control System (CGCS) has been retired in place with its electrical circuits de-energized and fluid process piping isolated from primary containment with redundant locked-closed isolation valves. Measures against hydrogen-oxygen combustion are provided by inerting of the primary containment atmosphere during plant operation. Refer to Subsections 6.2.5 and 9.3.6 for details.

1.5.2.8 Water Chemistry Program

Edison has participated extensively in water chemistry development programs and in the application of operating BWR water chemistry findings to the Fermi 2 plant. A water chemistry program with applicable Technical Requirements Manual and operating procedures has been developed in conformance with Regulatory Guide 1.56. General Electric Water Quality Document No. 22A2747 has served as a basis for this program. Refer to Subsections 9.3.2 and 10.4.6 for details.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

REFERENCES

1. NEDC-12431 Class I, January 30, 1974, Subject: Detroit Edison Standby Gas Treatment System Gasketless Filter Test Series, D. P. Siegwarth and M. Siegler, General Electric Company.

1.6 MATERIAL INCORPORATED BY REFERENCE

Table 1.6-1 lists topical reports that are incorporated in whole or in part by reference in this Updated Final Safety Analysis Report (UFSAR); these references are on file with the U.S. Nuclear Regulatory Commission (NRC).

UFSAR Figures that are derived from Edison controlled drawings contain a reference to the Edison drawing number. These figures will be regularly updated or have been removed. Drawings that are not expected to require revision fall into one or more of the following classes:

a. Figures that are typical (e.g., generic) sketches not showing design detail

b. Figures that will not change throughout the life of Fermi 2 (e.g., site geology, site geography, population distribution, and design criteria used during construction)

c. The portion of the drawing referenced from the UFSAR text that is not likely to change.

UFSAR Figures that are based on vendor drawings contain a reference to the vendor drawing number. These drawings may or may not be updated regularly or have been removed.

The Technical Requirements Manual (TRM) Volume 1 provides a central location for requirements relocated from the Fermi Operating License, Appendix A, Technical Specifications. The TRM Volume 1 (except for the Core Operating Limits Report) is incorporated by reference into the UFSAR.

TABLE 1.6-1 REFERENCED REPORTS

General Electric Company Reports

Report Number	Title	UFSAR Sections Where Referenced
APED-555	Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RDB144A (November 1967)	4.5
APED-5458	Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors (March 1968)	5.5
APED-5460	Design and Performance of GE BWR Jet Pumps (July 1968)	4.5
APED-5652	Stability and Dynamic Performance of the General Electric Boiling Water Reactor (April 1969)	4.1
APED-5696	Tornado Protection for the Spent Fuel Storage Pool (November 1968)	3.3, 3.5
APED-5706	In-Core Neutron Monitoring System for General Electric Boiling Water Reactors (November 1968; revised April 1969)	7.6
APED-5750	Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves (March 1969)	5.5
NEDO-10029	An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident (July 1969)	App. A
NEDO-10139	Compliance of Protection Systems to Industry Criteria: General Electric BWR Nuclear Steam Supply System (June 1970)	3.12, 7.1, 7.2, 7.3, 7.6
NEDO-10173	Current State of Knowledge, High Performance BWR Zircaloy-Clad UO_2 Fuel (May 1970)	11.1
NEDO-10299	Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello (January 1971)	4.4
NEDO-10320	The General Electric Pressure Suppression Containment Analytical Model (April 1971), Supplement 1 (May 1971)	6.2
NEDO-10329	Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors (April 1971), Supplement 1 (April 1971), Addenda (May 1971)	6.2

TABLE 1.6-1 REFERENCED REPORTS

General Electric Company Reports

Report Number	Title	UFSAR Sections Where Referenced
NEDO-10505	Experience with BWR Fuel Through September 1971 (May 1972)	11.1
NEDO-10527	Rod Drop Accident Analysis for Large Boiling Water Reactors (March 1972), Supplement 1 (July 1972) and Supplement 2 (January 1973)	4.5, 7.6, 15.4.9
NEDO-10602	Testing of Improved Jet Pump for the BWR/6 Nuclear System (June 1972)	4.5
NEDO-10677	Analysis of Recirculation Pump Overspeed in a Typical GE BWR (October, 1972)	5.5
NEDO-10678	Seismic Qualification of Class I Electric Equipment (November 1972)	3.10, 7.1, 7.3, 7.4, 7.6
NEDO-10698	Environmental Qualification of Class 1 Control and Instrumentation Equipment (November 1972)	3.11, 7.1, 7.2, 7.3, 7.4, 7.6
NEDO-10722A	Core Flow Distribution in a General Electric Boiling Water Reactor as Measured in Quad Cities Unit 1 (August 1976)	4.4
NEDO-10802 NEDO-10802-1	Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor (February 1973), Supplement 1 (April 1973)	4.4
NEDE-10811	Pipe Restraint Testing Program Conducted in Conjunction with the Design of the Enrico Fermi Power Plant Unit No. 1 (April 1973)	3.6
NEDO-10812	Hydrogen Flammability and Burning Characteristics in BWR Containments (July 1973)	1.5
NEDE-10813	PDA - Pipe Dynamic Analysis Program for Pipe Rupture Movement (March 1973)	3.6
NEDO-10871	Technical Derivation of BWR 1971 Design Basis Radioactive Source Terms (March 1973)	11.1
NEDO-10899	Chloride Control in BWR Coolants (June 1973)	5.2

TABLE 1.6-1 REFERENCED REPORTS

General Electric Company Reports

Report Number	Title	UFSAR Sections Where Referenced
NEDO-10958 NEDE-10958	General Electric Company BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application (November 1973)	4.4, 15.1.2
NEDO-10958A	GETAB Data, Correlation, and Design Application (January 1977)	4.4
NEDO-12037	Summary of Gamma and Beta Energy and Intensity Data (January 1970)	15A
NEDC-12431	Detroit Edison SGTS Gasketless Filter Test (July 1973)	1.5, 6.2
NEDE-13296	Pipe Whip Restraint Dynamic Evaluation (August 1972)	3.6
NEDE-13298	Deformation of Piping Due to Combined Bending and Lateral Load Under Pipe Whip Loading (August 1972)	3.6
NEDE-13331	Deformation of Piping Due to Combined Bending and Restraint Lateral Load – Additional Tests of Stainless Steel Pipes (March 1973)	3.6
NEDO-20360	General Electric BWR Generic Reload Application for 8 x8 Fuel	15.4.9
NEDO-20566, NEDE-20566-P	Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR Part 50, Appendix K (December 1975)	4.2, 6.3
NEDO-20566A	General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K (September 1986)	6.3
NEDO-20944, NEDE-20944-P, NEDE-20944-1P	BWR 4 and BWR 5 Fuel Design (October 1976) Proprietary Version (January 1977)	4.1, 4.2, 4.3, 4.4
NEDO-20946-A	BWR Simulator Methods Verification (July 1976)	4.3
NEDC-20994	Peach Bottom Atomic Power Station Units 2 and 3 Safety Analysis Report for Plant Modifications To Eliminate Significant In-Core Vibration (September 1975)	4.4, 4.5

TABLE 1.6-1 REFERENCED REPORTS

General Electric Company Reports

Report Number	Title	UFSAR Sections Where Referenced
NEDO-21143	Conservative Radiological Accident Evaluation – The CONACO1 Code	15.6.7, 15.7.4
NEDE-21156	Supplemental Information for Plant Modification To Eliminate Significant In-Core Vibration (January 1976)	4.4
NEDE-21175P-3	BWR Fuel Assembly Evaluation of Combined SSE and LOCA Loadings (July 1982)	3.9, 4.2, 4.5
NEDO-21291	Group Notch Mode of the Rod Sequence Control System for Cooper Nuclear Station (June 1976)	4.3, 15.4.1
NEDO-21506	Stability and Dynamic Performance of the General Electric Boiling Water Reactor (January 1977)	4.4
NEDO-21617 NEDO-21617-A	Analog Transmitter/Trip Unit System for Engineered Safeguard Sensor Trip Inputs (December 1978)	7.1, 7.2, 7.3, 7.4
NEDO-21778-A	Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors (January 1978)	5.2
NEDE-21821	Boiling Water Reactor Feedwater Nozzle Sparger (March 1978)	5.2
NEDO-21888-2	Mark I Containment Program Load Definition Report (November 1981)	3.8, 6.2
NEDO-22209	Analysis of Scram Discharge Volume System Piping Integrity (August 1982)	3.6
NEDE-23785-PA	The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident – SAFER/GESTR Application (October 1984)	6.3
NEDO-23786-1 NEDO-23786-P	Fuel and Rod Prepressurization (May 1978)	4.2
NEDO-24048	Evaluation of Acoustic Pressure Loads on BWR/6 Internal Components (September 1978)	3.9
NEDO-24057 NEDO-24057-P	Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants (November 1977)	1.5, 3.9

TABLE 1.6-1 REFERENCED REPORTS

General Electric Company Reports

Report Number	Title	UFSAR Sections Where Referenced
NEDO-24154	Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors (October 1978)	5.2, 2.3
NEDO-24342	GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Break (April 1981)	3.6
NEDC-24388-P	Enrico Fermi Atomic Power Plant Unit 2 Suppression Pool Temperature Response (December 1981)	6.2
NEDO-24568-3	Mark I Containment Program Plant Unique Load Definition – Enrico Fermi Atomic Power Plant Unit 2 (April 1982)	3.8, 6.2
NEDO-24708-A	Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors	3.6
GEAP 13197	Emergency Cooling in BWRs Under Simulated Loss-of-Coolant (BWR FLECHT Final Report) (June 1971)	6.2
NEDE-24011-P-A-10	General Electric Standard Application for Reactor Fuel (March 1991)	4.1, 4.2, 4.3, 4.4, 15.0, 15.1, 15.2, 15.4, 15.5
NEDE-24011-P-A-10-US	General Electric Standard Application for Reactor Fuel, United States Supplement (March 1991)	4.1, 4.2, 4.3, 4.4, 15.0, 15.1, 15.2, 15.4, 15.5
NEDE-31096	Anticipated Transients Without Scram Response to NRC ATWS Rule 10 CFR 50.62 (February1987)	15.8
NEDC-33865P	DTE Energy Enrico Fermi 2 SAFER/PRIME-LOCA Loss-of-Coolant Accident Analysis (March 2015)	6.3

1.7 ABBREVIATIONS AND SYMBOLS USED IN THE UFSAR

Abbreviations and symbols used in the UFSAR are contained in this section. Figure 1.7-1 contains the symbols used on Edison and GEC drawings. Figure l.7-2 contains the piping and instrumentation symbols used on GE drawings and figures. Figure 1.7-3 contains the logic symbols used on GE/Edison Functional Control Diagrams. Figure 1.7-4 contains the piping and instrumentation symbols used on Sargent & Lundy drawings and figures.

1.7.1. Abbreviations

A	
Advisory Committee on Reactor Safeguards	ACRS
Alternative Source Term	AST
alternating current	ac
American Concrete Institute	ACI
American Institute of Steel Construction	AISC
American Iron and Steel Institute	AISI
American National Standards Institute	ANSI
American Nuclear Society	ANS
American Petroleum Institute	API
American Society for Testing and Materials	ASTM
American Society of Agricultural Engineers	ASAE
American Society of Civil Engineers	ASCE
American Society of Heating, Refrigerating, and Air-Conditioning Engineers	ASHRAE
American Society of Mechanical Engineers	ASME
American Standards Association	ASA
American Water Works Association	AWWA
American Welding Society	AWS
Ampere	A
as low as reasonably achievable	ALARA
Atomic Energy Commission (see also NRC)	AEC
Atomic Safety and Licensing Board	ASLB
B	
Battelle Memorial Institute	BMI
Branch Technical Position	BTP

C	
Canadian Standards Association	CSA
Charpy V-notch	CVN
Chicago Bridge and Iron (Company)	CBI
Code of Federal Regulations	CFR
critical heat flux	CHF
cubic centimeter	cm^3
cubic feet per minute	cfm
cubic feet per second	cfs
cubic foot	ft^3
cubic meter	m^3
cubic meters per second	m^3/sec
cubic yard	yd^3
curie	Ci
cycles per second	Hz
D	
decibel	dB
degree (plane angle)	---
degree - Centigrade	C
degree - Fahrenheit	F
degree Rankine	R
Department of Transportation	DOT
Diesel Engine Manufacturers Association	DEMA
dioctyl phthalate penetration test	DOP
direct current	dc
Director, Reactor Licensing	DRL
The Detroit Edison Company	Edison
E	
2.7l8 ---, base of Naperian log system	e
Electric Power Research Institute	EPRI
electron volt	eV

electronic data processing	EDP
end of life	EOL
Environmental Protection Agency	EPA
erg	erg
effective neutron multiplication factor of the reactor	k_{eff}
F	
failure modes and effects analysis	FMEA
Federal Power Commission	FPC
Federal Water Pollution Control Act	FWPCA
feet per hour	ft/hr
feet per minute	fpm
feet per second	fps
foot (feet)	ft
foot of water (conventional)	ft H_2O
foot-pound	ft-lb
G	
gallon	gal
gallons per minute	gpm
gallons per second	gps
General Design Criterion (Criteria)	GDC
General Electric - Boiling Water Reactor	GE-BWR
General Electric Company Turbine - Generator, Ltd.	GEC
Geological Society of America	GSA
gigacycles per second	GHz
gigaelectron volt (10^9)	GeV
gram	g
grams per cubic centimeter	g/cm^3
gravitational acceleration factor, (32 ft per sec^2)	g
The General Electric Company	GE
H	
Heat Exchange Institute	HEI
henry	H

hertz	Hz
horsepower	hp
hour	hr
hydrogen-ion concentration	pH
I	
inch	in.
inch per second	in./sec
inch-pound	in.-lb
inches of mercury absolute	in. Hg abs
inches of water (pressure)	in. H_2O
inservice inspection	ISI
inside diameter	I.D.
Institute of Electrical and Electronics Engineers	IEEE
Institute of Nuclear Power Operations	INPO
Instrument Society of America	ISA
Interim Acceptance Criteria (AEC)	IAC
Interstate Commerce Commission	ICC
K	
kilo	k
kilocalorie	kcal
kilocycle per second	kHz
kiloelectron volt	keV
kilogram	kg
kilogram per square centimeter	kg/cm^2
kilojoule	kJ
kilometer	km
kilovolt, 10^3	kV
kilovolt-ampere	kVA
kilowatt	kW
kilowatt-hour	kWh
L	
least significant bit	LSB

licensee event report	LER
linear heat generation rate	LHGR
liter	l
low-population zone	LPZ
M	
maximum permissible concentration	mpc
mean low water datum	MLD
mega (10^6)	M
megacycles per second	MHz
megaelectron volt (10^6)	MeV
megahertz	MHz
megavolt-ampere	MVA
megawatt	MW
megawatt electric	MWe
megawatt thermal	MWt
megawatt-days per metric ton	MWd/t
megawatt-days per short ton	MWd/ST
meter	m
mho	mho
micro (10^{-6})	μ
microampere	μA
microcurie	μCi
microgram	μg
microhenry	μH
micrometer	μm
micromho	μmho
microsecond	μsec
microwatt	μW
mil	mil
miles per hour	mph
Military Specification	MIL
milli (10^{-3})	m

milliampere	mA
millicurie	mCi
milligram	mg
millihenry	mH
millimeter	mm
millimeter of mercury absolute	mm Hg abs
million electron volts	MeV
millirem	mrem
milliroentgen	mR
millisecond	msec
millivolt	mV
milliwatt	mW
Mine Safety Appliance	MSA
minute (time)	minute
molecular power supply unit	MPSU
N	
National Electrical Manufacturers Association	NEMA
National Fire Protection Association	NFPA
National Fire Protection Organization	NFPO
National Institute of Occupational Safety and Health	NIOSH
National Society of Professional Engineers	NSPE
National Weather Records Center	NWRC
neutron density, neutrons per cubic centimeter	n
neutron flux, neutrons per cubic centimeter per second	nv
neutron velocity time	nvt
nil ductility transition temperature	NDTT
nondestructive examination	NDE
nondestructive testing	NDT
Nuclear Energy Property Insurance Association	NEPIA
Nuclear Regulatory Commission (see also AEC)	NRC
O	
Occupational Safety and Health Administration	OSHA
Operating License	OL

P	
parts per billion	ppb
parts per million	ppm
percent	percent
piping and instrumentation drawing	P&ID
Plant Operations Manual	POM
pound	lb
pound mass per second	lbm/sec
pound-foot	lb-ft
pounds per cubic foot	lb/ft^3
pounds per hour	lb/hr
pounds per second	lb/sec
pounds per square inch	psi
pounds per square inch, absolute	psia
pounds per square inch, differential	psid
pounds per square inch, gage	psig
preservice inspection	PSI
probable maximum flood	PMF
probable maximum meteorological event	PMME
probable maximum precipitation	PMP
Q	
quality assurance	QA
quality control	QC
R	
rad, unit of absorbed radiation	rad
radian	radian
Radiological Emergency Response Preparedness	RERP
Radiologically Controlled Area	RCA
revolutions per minute	rpm
revolutions per second	rps
Rock Quality Designation	RQD

Rockwell hardness number	RHN
roentgen equivalent, man	rem
roentgen, unit of radiation exposure	R
root mean square	rms
S	
safe-shutdown earthquake	SSE
Safety Evaluation Report	SER
second (time)	sec
Seismic Qualification Review Team	SQRT
Southeast Michigan Council of Governments	SEMCOG
square centimeter	cm^2
square foot	ft^2
square inch	$in.^2$
square root of the sum of the squares	SRSS
square yard	yd^2
standard cubic feet per minute	scfm
Standard Review Plan	SRP
T	
thousand electron volts	keV
total effective dose equivalent	TEDE
Transient Reactor Analysis Code (GE)	TRACG
Tubular Exchanger Manufacturers Association	TEMA
U	
United States Bureau of Mines	USBM
United States Coast and Geodetic Survey	USC&GS
United States Geological Survey	USGS
V	
volt	V
volt-ampere	VA
volts, alternating current	V ac
volts, direct current	V dc
W	

watt	W
watt-hour	Wh

1.7.2. System, Component, and Process Abbreviations

anticipated transient without scram	ATWS
area radiation monitoring system	ARMS
automatic depressurization system	ADS
automatic gain control	AGC
average power range monitor	APRM
balance of plant	BOP
boiling water reactor	BWR
cathode ray tube	CRT
closed cooling water	CCW
combustible gas control system	CGCS
combustion turbine generator	CTG
condensate storage and transfer system	CSTS
containment and reactor vessel isolation control system	CRVICS
continuous air monitor	CAM
control center air conditioning system	CCACS
control rod drive	CRD
control rod drive return line	CRDRL
core cooling and containment system	CCCS
critical power ratio	CPR
design-basis accident	DBA
dosimeter of legal record	DLR
electro-hydraulic control	EHC
emergency core cooling system	ECCS
emergency diesel generator	EDG
emergency diesel generator service water system	EDGSW
emergency equipment cooling water system	EECWS
emergency equipment service water system	EESWS
emergency response data system	ERDS
engineered safety feature	ESF

excess flow check valve	EFCV
fire protection system	FPS
fuel pool cooling and cleanup system	FPCCS
full length emergency cooling heat transfer	FLECHT
functional control diagram	FCD
GE type of relay	HFA
Geiger-Mueller tubes	G-M tubes
general service water	GSW
heat affected zone	HAZ
heating, ventilation, and air conditioning	HVAC
high pressure coolant injection	HPCI
high-efficiency particulate air	HEPA
hydraulic control unit	HCU
hydrogen water chemistry	HWC
Independent Spent Fuel Storage Installation	ISFSI
induction heating stress improvement	IHSI
integrated plant computer system	IPCS
intergranular stress corrosion cracking	IGSCC
intermediate range monitor	IRM
intermediate-break accident	IBA
leak detection system	LDS
local power range monitor	LPRM
loose parts monitoring system	LPMS
loss-of-coolant accident	LOCA
low pressure coolant injection	LPCI
main steam isolation valve	MSIV
main steam isolation valve leakage control system	MSIVLCS
maximum average planar linear heat generation rate	MAPLHGR
maximum linear heat generation rate	MLHGR
mechanical equipment qualification	MEQ
minimum critical power ratio	MCPR
motor control center	MCC
motor-generator sets	M-G sets

net positive suction head	NPSH
neutron monitoring system	NMS
noninterruptible air supply	NIAS
nuclear boiler system	NBS
nuclear pressure relief system	NPRS
nuclear steam supply system	NSSS
Onsite Review Organization	OSRO
operating-basis earthquake	OBE
oscillation power range monitor	OPRM
pipe whip restraint support system	PWRSS
power range monitor	PRM
pressure control valve	PCV
primary containment monitoring system	PCMS
process and effluent radiation monitor system	PERMS
radiation area protective (clothing)	RAP
radiation monitoring system	RMS
reactor building closed cooling water system	RBCCW
reactor coolant leak detection system	RCLDS
reactor coolant pressure boundary	RCPB
reactor core isolation cooling (system)	RCIC
reactor feed pump	RFP
reactor manual control system	RMCS
reactor pressure vessel	RPV
reactor protection system	RPS
reactor recirculation system	RRS
reactor water cleanup	RWCU
recirculation flow control system	RFCS
recirculation pump trip	RPT
residual heat removal	RHR
residual heat removal service water	RHRSW
rod block monitor	RBM
rod sequence control system	RSCS
rod worth minimizer	RWM

safe-shutdown earthquake	SSE
safety parameter display system	SPDS
safety/relief valve	SRV
scram discharge volume	SDV
small-break accident	SBA
sequence of events	SOE
source range monitor	SRM
standby gas treatment system	SGTS
standby liquid control system	SLCS
steam generation system	SGS
stuck open relief valve	SORV
supplemental cooling chilled water	SCCW
torus water management system	TWMS
traversing in-core probe	TIP
turbine building closed cooling water system	TBCCWS

Figure Intentionally Removed
Refer to Plant Drawing M-2001

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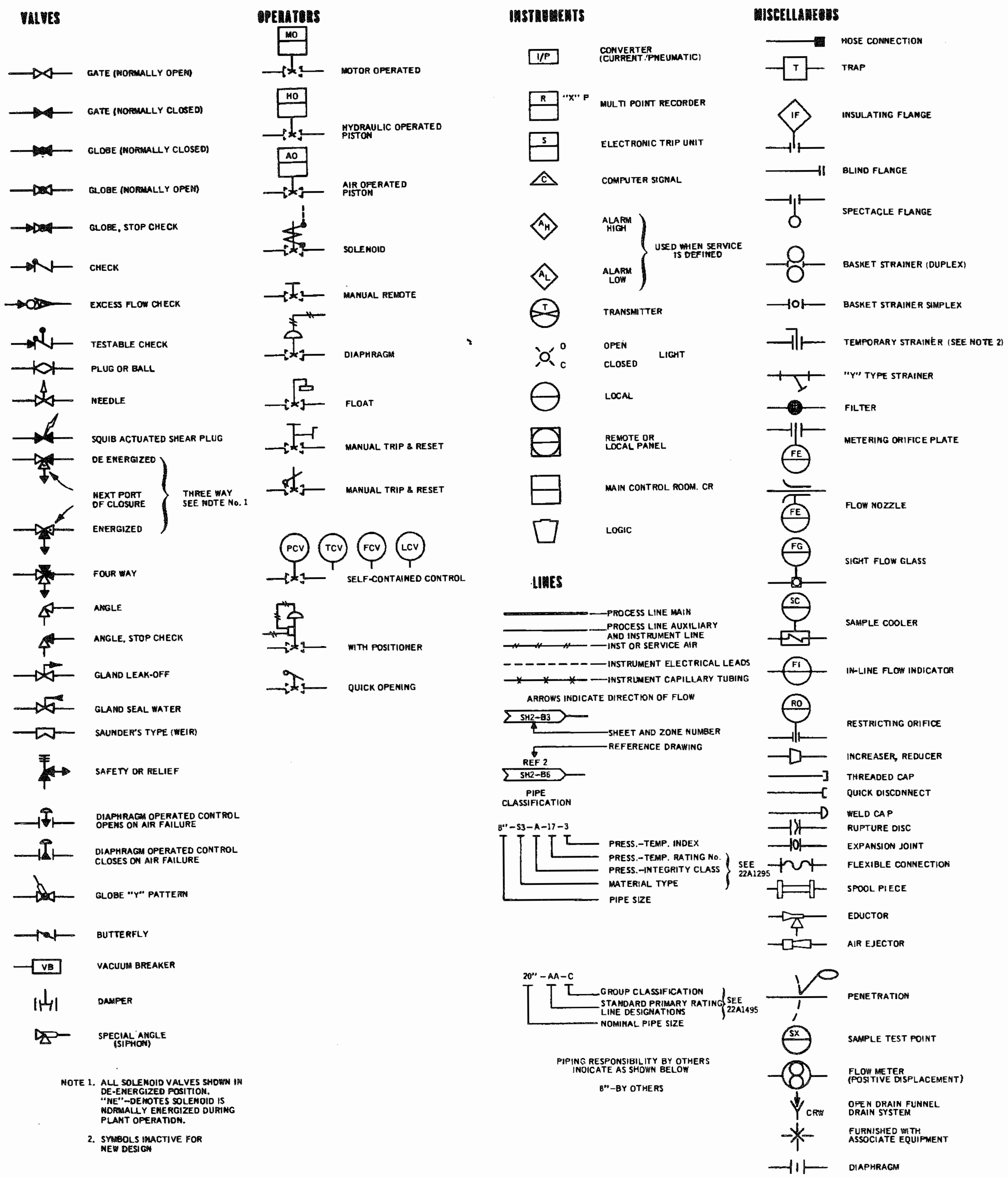
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 1.7-1

SYMBOLS APPLICABLE TO
EDISON AND GEC FIGURES



GENERAL ELECTRIC

PARTS LIST FOR 197R567 CONT ON SHEET F SH NO. 1

REV NO. Ø 1 — PARTS LIST FOR Rev. 1 1-30-70 — CONT ON SHEET — SH NO.

TITLE: PIPING AND INSTRUMENT SYMBOLS

FIRST MADE FOR

INSTRUMENT FUNTION / MEASURED VARIABLE (FIRST LETTER)		CONTROLLING (SECOND and THIRD LETTER)						MEASURING (SECOND and THIRD LETTER)											
		RECORDING	INDICATING	NON-INDICATING	CONTROL VALVE	SUMMER	FUNCTION GENERATOR	RECORDER	INDICATOR	OBSERVATION GLASS	PRIMARY ELEMENT	TEST POINT	TRANSMITTER OR PREAMP	INTEGRATOR	AMPLIFIER	SAMPLER	INDICATING SWITCH	NON INDICATING SWITCH	ALARMS
		–RC	–IC	–C	–CV	–Σ	–f	–R	–I	–G	–E	–X	–T	–Q	Am	Sm	IS	–S	–A
AIR	A	–	–	–	–	–	–	–	–	–	–	–	–	–	–	ASm	–	–	–
CONDUCTIVITY	C	CRC	CIC	–	CCV	–	–	CR	CI	–	CE	CX	CT	–	–	CSm	CIS	CS	CA
DENSITY	D	DRC	DIC	DC	DCV	–	–	DR	DI	–	–	DX	DT	–	–	–	DIS	DS	DA
DIFF. PRESS.	dP	dPRC	dPIC	dPC	dPCV	–	–	dPR	dPI	–	–	–	dPT	–	–	–	dPIS	dPS	dPA
FLOW	F	FRC	FIC	FC	FCV	FΣ	Ff	FR	FI	FG	FE	FX	FT	FQ	–	–	FIS	FS	FA
HYDROGEN ION CONC.	pH	pHRC	pHIC	pHC	pHCV	–	–	pHR	pHI	–	pHE	pHX	–	–	pHAm	pHSm	–	–	–
LEVEL	L	LRC	LIC	LC	LCV	–	–	LR	LI	LG	–	–	LT	–	–	–	LIS	LS	LA
MOISTURE	M	MRC	MIC	–	–	–	–	MR	MI	–	ME	–	–	–	–	–	–	–	–
NEUTRON FLUX	N	NRC	NIC	NC	–	–	–	NR	NI	–	NE	NX	NT	NQ	NAm	–	–	–	NA
OXYGEN	O_2	–	–	–	–	–	–	O_2R	O_2I	–	O_2E	–	–	–	–	–	–	–	–
PRESSURE	P	PRC	PIC	PC	PCV	–	–	PR	PI	–	–	PX	PT	–	–	–	PIS	PS	PA
POSITION	Po	–	–	–	–	–	–	PoR	PoI	–	–	–	PoT	–	–	–	–	PoS	PoA
RADIATION	R	–	–	–	–	–	–	RR	RI	–	RE	RX	–	–	RAm	RSm	–	RS	RA
SPEED	S	SRC	SIC	SC	–	–	–	SR	SI	–	SE	–	–	–	–	–	–	–	–
TEMPERATURE	T	TRC	TIC	TC	TCV	–	–	TR	TI	–	TE	TX	TT	–	–	–	TIS	TS	TA
TIME	t	–	–	tC	–	–	–	–	tI	–	–	–	–	–	–	–	tIS	tS	–
VIBRATION	Vb	–	–	–	–	–	–	VbR	VbI	–	–	–	–	–	–	–	–	VbS	VbA
WEIGHT FACTOR	Wf	WfRC	WfIC	–	WfV	–	–	–	–	–	–	–	–	–	–	–	–	–	–

MISCELLANEOUS ABBREVIATIONS:

A/S	AIR SUPPLY
AW	ACID WASTE (CORROSIVE, CAUSTIC)
CIT	CONDUCTIVITY INDICATOR TRANSMITTER
CRD	CONTROL ROD DRIVE
CRDHS	CONTROL ROD DRIVE HYDRAULIC SYSTEM
CRS	CONDUCTIVITY RECORDING SWITCH
CRW	CLEAN RADWASTE
DRW	DIRTY RADWASTE
dTS	DIFFERENTIAL TEMPERATURE SWITCH
E/P	CONVERTER (VOLTAGE PNEUMATIC)
E/S	SPECIAL ELECTRIC POWER SUPPLY REQUIRED
FAI	FAIL AS IS (SEE NOTE 2)
FC	INDICATES CLOSES ON AIR OR ELECTRICAL FAILURE
FIT	FLOW INDICATOR TRANSMITTER
FO	INDICATES OPENS ON AIR OR ELECTRICAL FAILURE
FRCS	FLOW RECORDING CONTROLLER SWITCH
HCU	HYDRAULIC CONTROL UNIT
HS	HAND SWITCH (SEE NOTE 2)
IP	CONVERTER (CURRENT PNEUMATIC)
LC	LOCK CLOSED
L/DRS	LEVEL & DENSITY RECORDER SWITCH
LIM SW	LIMIT SWITCH
LIRS	LEVEL INDICATOR RECORDING SWITCH
LO	LOCK OPEN
LRS	LEVEL RECORDING SWITCH
MV/I	MILLIVOLT TO CURRENT CONVERTER
NC	NORMALLY CLOSED
ND	NORMALLY DE-ENERGIZED
NE	NORMALLY ENERGIZED
NO	NORMALLY OPEN
NW	NORMAL WASTE (CONVENTIONAL)
RBCCW	REACTOR BUILDING CLOSED COOLING WATER
RBEDT	REACTOR BUILDING EQUIPMENT DRAIN TANK
RM	REMOTE MANUAL
RMC	REMOTE MANUAL CONTROL
RMS	REMOTE MANUAL SWITCH
RPS	REACTOR PROTECTION SYSTEM
RPV	REACTOR PRESSURE VESSEL
SS	SELECTOR SWITCH
SSa	SELECTIVE SWITCH AUTOMATIC
SQ RT or $\sqrt{}$	SQUARE ROOT CONVERTOR
TBCCW	TURBINE BUILDING CLOSED COOLING WATER
tC	CYCLE TIMER
tdS	TIME DELAY SWITCH
TQOS	TORQUE OVERLOAD SWITCH
TQRS	TORQUE RECORDER SWITCH
TQT	TORQUE TRANSMITTER
TRS	TEMPERATURE RECORDER SWITCH

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FIGURE 1.7-2

PIPING AND INSTRUMENTATION SYMBOLS APPLICABLE TO GE FIGURES

Figure Intentionally Removed
Refer to Plant Drawing 209A4756

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 1.7-3

LOGIC SYMBOLS USED ON GE/EDISON
FUNCTIONAL CONTROL DIAGRAMS

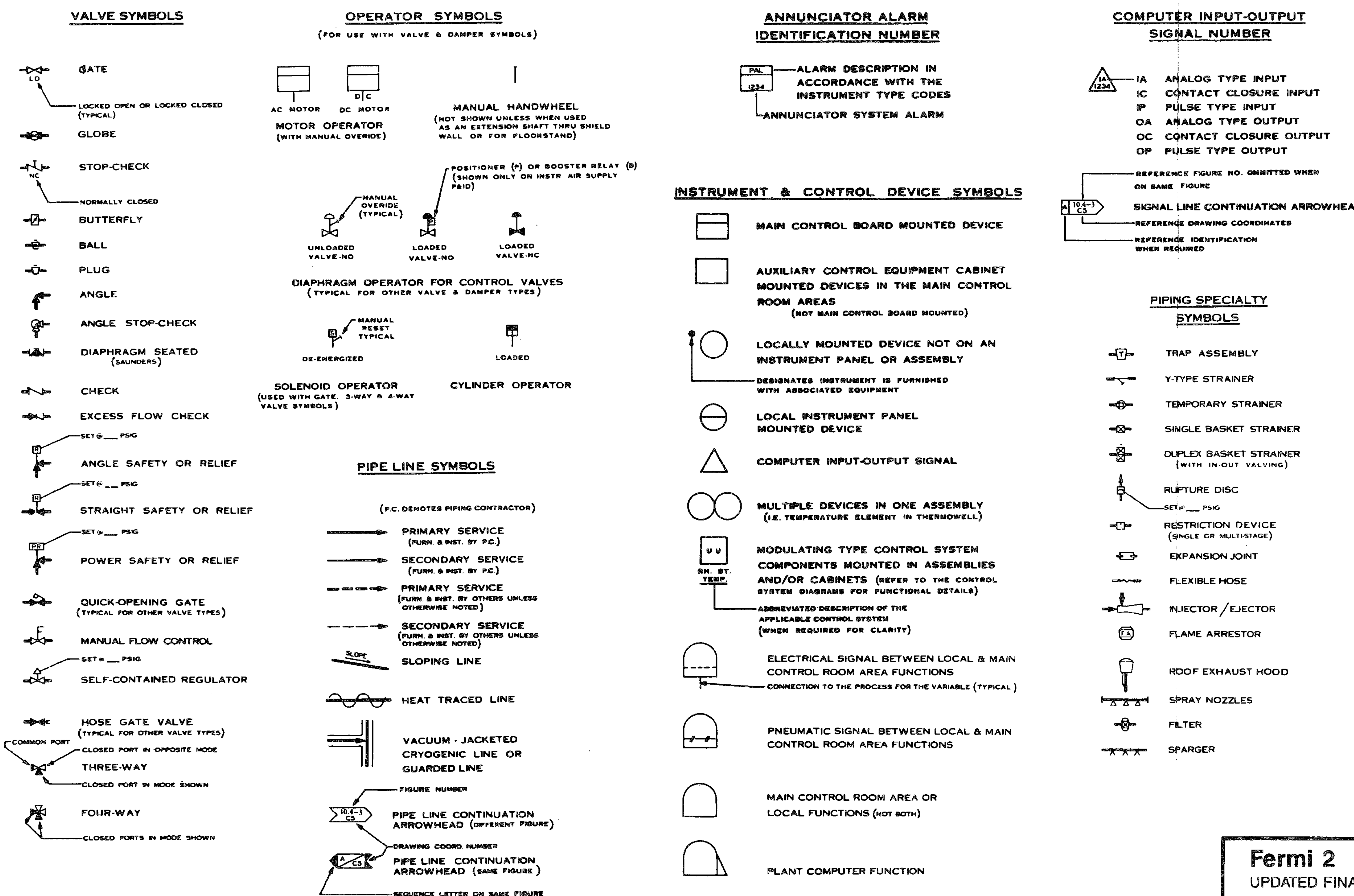

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FIGURE 1.7-4, SHEET 1

SYMBOLS APPLICABLE TO SARGENT & LUNDY FIGURES

INSTRUMENT TYPE CODES

FIRST LETTER
VARIABLE
(MEASURED OR INITIATING)

A	ANALYSIS
B	BURNER-FLAME
C	CONDUCTIVITY
D	DENSITY
E	VOLTAGE
F	FLOW
H	HAND (MANUAL)
I	CURRENT (ELEC.)
J	POWER (KW)
K	TIME
L	LEVEL (LIQUID OR SOLID)
M	MOISTURE (HUMIDITY)
N	NEUTRON FLUX
P	PRESSURE
R	RADIOACTIVITY
S	SPEED OR FREQUENCY
T	TEMPERATURE
U	MULTIVARIABLE
V	VIBRATION
W	WEIGHT
X	SPECIAL
Y	STRAIN
Z	POSITION

SECOND OR SUCCEEDING LETTERS
INSTRUMENT FUNCTION

A	ALARM
AC	ALARM CLOSED
AH	ALARM HIGH
AL	ALARM LOW
AO	ALARM OPEN
C	CONTROLLER (BLIND)
CD	CONTROL DAMPER
CK	CONTROLLER & HAND/AUTO
CV	CONTROL VALVE
D	DAMPER
DC	DIFFERENTIAL CONTROLLER
DI	DIFFERENTIAL INDICATOR
DK	DIFFERENTIAL CONTROL STATION
DR	DIFFERENTIAL RECORDER
DS	DIFFERENTIAL SWITCH
DT	DIFFERENTIAL TRANSMITTER
DY	DIFFERENTIAL RELAY
E	PRIMARY ELEMENT
EW	PRIMARY ELEMENT THERMOWELL
G	GLASS
I	INDICATOR
IC	INDICATING CONTROLLER
IR	INDICATING RECORDER
IS	INDICATING SWITCH
IT	INDICATING TRANSMITTER
IY	INDICATING RELAY
K	CONTROL STATION -VARIABLE TYPE-HAND. HAND-AUTO
L	LIGHT (PILOT)
Q	INTEGRATOR
R	RECORDER
RC	RECORDER CONTROLLER
S	SWITCH
SC	SWITCH - CLOSED DEVICE POSITION
SH	SWITCH HIGH
SI	SWITCH - INTERMEDIATE DEVICE POSITION
SL	SWITCH LOW
SO	SWITCH - OPEN DEVICE POSITION
SV	SOLENOID VALVE
T	TRANSMITTER
U	MULTIFUNCTION
V	VALVE - ON-OFF POWER-OPERATED TYPE
W	THERMOWELL
X	SPECIAL
Y	RELAY - SIGNAL CONVERTER, COMPUTING, ETC.
Z	POWER POSITIONER (EXCEPT VALVE MOUNTED)

CLARIFYING SYMBOL LIST

A	ANALOG SIGNAL	MAX	MAXIMUM
AVG	AVERAGE	MIN	MINIMUM
D	DIGITAL	N_2	NITROGEN
DIFF	SUBTRACT	N_2H_4	HYDRAZINE
DIR	DIRECT ACTING	O_2	OXYGEN
FC	FAIL CLOSED	pH	pH ANALYSIS
FI	FAIL INTERMEDIATE	REV	REVERSE ACTING
FL	FAIL LOCKED	SI	SILICA
FO	FAIL OPEN	SM	SMOKE OR IONIZATION
H_2	HYDROGEN	SP	SET POINT
HL	HIGH LIMIT	SQ. RT	SQUARE ROOT
HP	HIGH PASS	TURB	TURBIDITY
LL	LOW LIMIT	X	MULTIPLY
LP	LOW PASS		

PIPE LINE INSTRUMENT SYMBOLS

FE

FLOW NOZZLE OR VENTURI

FLOW NOZZLE OR VENTURI PIPE SECTION

FLOW ORIFICE (ORIFICE FLANGES BY P.C., UNLESS OTHERWISE NOTED)

FC

UNCLASSIFIED FLOW ELEMENT (MAGNETIC, TARGET, AIR FOIL, LAMINAR, ETC.)

SIGHT FLOW INDICATOR

ROTAMETER

PITOT TUBE

TURBINE METER

FLOW STRAIGHTENING VANES

AIR FILTER-PRV SET

PROCESS RADIATION MONITOR

TW THERMOWELL

FS FLOW SWITCH

AE SAMPLE ELEMENT (ANALYSIS - CELL, PROBE, ETC.)

TEST CONNECTIONS

PRESSURE

GRAB SAMPLE

TEMPERATURE WHEN THERMOWELL IS NOT USED

FLOW

HVAC DUCT TEST FOR PRESSURE, TEMPERATURE AND / OR FLOW

PIPE LINE COMPONENT SYMBOLS

END CLOSURE (WELDED CAP)

END CLOSURE (SCREWED CAP)

END CLOSURE (BLIND FLANGE)

END CLOSURE (HOSE CONN)

REDUCER

OPEN FUNNEL DRAIN

CLOSED FUNNEL DRAIN

FLOOR DRAIN

ATMOSPHERIC VENT (WITH SCREEN)

THERMAL SLEEVE

RELIEF VALVE VENT PIPE SLIP JOINT

PENETRATION OF PRIMARY CONTAINMENT

HOT SIDE

PENETRATION OF OUTER SHIELD WALL

PENETRATION OF INNER SHIELD WALL

HVAC SYMBOLS

DUCTS

DUCTED AIR FLOW (FURN. & INST. BY HVAC CONTRACTOR)

NON-DUCTED AIR FLOW

DAMPERS

OPPOSED BLADE

NO FO — NORMALLY OPEN, FAIL OPEN — TYPICAL

PARALLEL BLADE

NC FC — NORMALLY CLOSED, FAIL CLOSED — TYPICAL

FAN INLET VANES

BUTTERFLY

NORMALLY CLOSED

NORMALLY OPEN

GRAVITY SHUTTER

FIRE DAMPER WITH FUSIBLE LINK

EQUIPMENT

CV CONSTANT VOLUME BOX

RH TERMINAL REHEAT ATTENUATOR

MB MIXING BOX

ABBREVIATIONS

SA	SUPPLY AIR
RA	RETURN AIR
EA	EXHAUST AIR
OA	OUTSIDE AIR
MB	MIXING BOX
CV	CONSTANT VOLUME BOX
RH	TERMINAL REHEAT ATTENUATOR
H	HOT DUCT
C	COLD DUCT

FILTERS

PRE-FILTER

HEPA FILTER

MED EFF FILTER

CHARCOAL FILTER

INSTRUMENT & CONTROL SIGNAL SYMBOLS

INSTRUMENT TRANSMITTED PNEUMATIC SIGNAL
NOT INSTRUMENT AIR SUPPLY (FURN. & INST. BY P.C.)
LINES NOT CONNECTED
LINES CONNECTED

INSTRUMENT TRANSMITTED PNEUMATIC SIGNAL (FURN. & INST. BY OTHERS UNLESS OTHERWISE NOTED)
LINES NOT CONNECTED
LINES CONNECTED

ELECTRICAL CONTROL & INTERLOCKING SIGNAL
LINES NOT CONNECTED
LINES CONNECTED

HYDRAULIC SIGNAL
LINES NOT CONNECTED
LINES CONNECTED

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FIGURE 1.7-4, SHEET 2

SYMBOLS APPLICABLE TO SARGENT & LUNDY FIGURES