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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 INTRODUCTION

The Duane Arnold Energy Center (DAEC) Final Safety Analysis Report (FSAR) was submitted in support of an application by Iowa Electric Light and Power Company[1] and by the Central Iowa Power Cooperative and the Corn Belt Power Cooperative for a construction permit and facility license for a nuclear power plant under Section 104 (b) of the Atomic Energy Act of 1954, as amended, and the regulations of the Atomic Energy Commission (AEC) in Title 10, Code of Federal Regulations (10 CFR).

At the time of the initial application, ownership of the plant was divided into the following proportions:

Iowa Electric[1]	70%
Central Iowa Power Cooperative	20%
Corn Belt Power Cooperative	10%

The plant was constructed pursuant to a construction permit dated June 17, 1970. It is located on a site near Palo in Linn County, Iowa, and consists of approximately 500 acres adjacent to the Cedar River. The nuclear steam supply system (NSSS) and the turbine-generator were furnished by the General Electric Company (GE). The balance of plant was designed and constructed by Bechtel Power Corporation (Bechtel) as architect-engineer and constructor. The plant achieved initial criticality on March 23, 1974, and began commercial operation on February 1, 1975.

The unit was originally designed, analyzed, and licensed for a steady-state core power of 1658 MWt. However, the plant Technical Specifications restricted operation to a rated power of 1593 MWt. In 1985, commencing with reload cycle 8, the Technical Specifications were amended to allow the DAEC to operate at a steady-state power level of 1658 MWt (License Amendment #115). Rated (100%) power became 1658 MWt at that time. The net electric output of the plant at that level is 541 MWe. Heat balance and safety analyses were performed at 102% of the rated power which corresponds to 1691 MWt.

Then in 2001, the rated power level was increased again, by the Extended Power Uprate (EPU) Project, to 1912 MWt (License Amendment #243). Extensive engineering studies and safety analyses were performed at 102% of the new licensed power level, i.e., 1950 MWt.

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[1] Iowa Electric Light and Power Company has undergone several mergers since 1994. Some historical references to Iowa Electric Light and Power Company, or Iowa Electric, are maintained as appropriate.

Over the operating history of the DAEC, ownership and operating authority have changed on multiple occasions through company mergers, acquisitions, divestitures, and other corporate activities. The following is the timeline of those changes:

> The original primary owner (70%) and licensed operator of the DAEC was Iowa Electric Light & Power Company (Iowa Electric, IELP). As noted previously, Central Iowa Power and Corn Belt Power Cooperatives are minority owners, 20% and 10%, respectively, and while they have certain regulatory financial responsibilities, they do not have any operating authority for the DAEC.
>
> In 1986, IELP became a wholly-owned subsidiary of IE Industries, Inc. and received NRC approval by letter dated June 30, 1986. No changes to the DAEC operating license were made.
>
> By NRC letter dated May 6, 1991, the merger between IE Industries, Inc. and Iowa Southern, Inc. to become a new corporation, IES Industries, Inc., was acknowledged. There were no changes to the DAEC operating license as a result of this merger, as IELP was a wholly-owned subsidiary of the new corporation.
>
> In January 1994, IELP was merged with Iowa Southern Utilities Company to become, IES Utilities Inc., a wholly-owned subsidiary of IES Industries, and the associated operating license change was approved by NRC in Amendment # 198.
>
> By NRC order dated August 28, 1997, the merger of IES Industries with two other utilities that ultimately become known as Alliant Energy, was approved. IES Utilities became a wholly-owned subsidiary of Alliant Energy. No operating license change was made.
>
> By NRC order dated May 15, 2000 and license amendment # 232, dated August 7, 2000, the operating authority was transferred from IES Utilities Inc. to the Nuclear Management Company (NMC). IES Utilities Inc. retained their 70% ownership of DAEC.
>
> By license amendment # 244, dated January 2, 2002, the DAEC operating license was revised to reflect the name change in the corporation from IES Utilities to Iowa Power and Light Company (IP&L), a wholly-owned subsidiary of Alliant Energy.
>
> By NRC order, dated December 23, 2005, and license amendment # 260, dated January 27, 2006, IP&L's 70% ownership and NMC's operating authority were transferred to FPL Energy Duane Arnold, LLC (hereafter FPL Energy Duane Arnold), a wholly-owned subsidiary of FPL Energy.

On April 16, 2009, the name “FPL Energy Duane Arnold, LLC” was legally changed to “NextEra Energy Duane Arnold, LLC.” The NRC was notified of this name change by letter (NG-09-0242) dated April 17, 2009.

Historical references to the above entities have been retained throughout the UFSAR, where appropriate. When UFSAR Figures require revision, they will be updated with the current company name at that time.

The major systems and components that bear significantly on the acceptability of the site, or on the safety of the plant, including the engineered safeguards systems and the containment, have been evaluated for operation at a power level of 1912 MWt. The plant safety analysis (Chapter 15) has been performed at an assumed power level of 1950 MWt which is rated power plus a 2% uncertainty factor in power level in accordance with Regulatory Guide 1.49.

The purpose of the FSAR was to provide the technical information requested by 10 CFR 50.34 in order to provide information needed for the evaluation of the DAEC plant with respect to the issuance of a construction permit and facility license. This updated FSAR has been prepared in response to 10 CFR 50.71(e).

The principal architectural and engineering criteria that governed the development of the plant design are discussed in Section 1.2. The plant description and analyses that follow explain the various systems and components that are designed to satisfy such criteria. The various sections of the report that describe individual systems or components also provide the functional performance requirements for such systems, together with a summary of the technical and safety evaluation of the system.

Throughout the life of the DAEC 10 CFR 50 license, spent fuel, high-level radioactive waste, and reactor-related greater than Class C waste may be transferred in 10 CFR 72 licensed storage systems from the 10 CFR 50 licensed facility to an Independent Spent Fuel Storage Installation (ISFSI) for interim on-site storage. An ISFSI is licensed under 10 CFR 72. The design and licensing basis for an ISFSI will be contained in design basis documents, separate from this UFSAR, and within each installed 10 CFR 72 licensed storage system’s Certificate of Compliance and Final Safety Analysis Report. An ISFSI requires non-safety related system tie-ins to the 10 CFR 50 licensed facility. As applicable, these system tie-ins will be stated in this UFSAR. An ISFSI interfaces with many 10 CFR 50 established processes and programs (emergency, security, radiation protection, etc.). These process and program interfaces are not discussed in this UFSAR, but will be discussed in ISFSI design basis documents and each specific program. The operation of an ISFSI does not adversely affect the safe operation of the 10 CFR 50 licensed facility.

1.1.1 ORGANIZATIONAL RESPONSIBILITIES

The DAEC organization is discussed in detail in Chapter 13. Some of the basic responsibilities are discussed below.

1.1.1.1 Training

The operating, maintenance, technical, and administrative staff receive extensive training and instruction in academic subjects and practical operations. These instructions are given both within and outside the plant to qualify the staff for their responsibilities and to enable them to obtain operator and senior operator licenses where required. Detailed training plans are described in Section 13.2.

1.1.1.2 Preoperational Test Program

Prior to initial operations, all equipment, components, and systems were tested. These tests were conducted by Iowa Electric with technical assistance from General Electric and Bechtel.

1.1.1.3 Startup and Power Test Program

Following initial fuel loading, a test program progressing from low-power to full-power operation (1593 MWt) was performed with Iowa Electric performing the actual work under the technical direction of the NSSS supplier. Each test was carefully evaluated before proceeding to the next higher power level. Following the cycle 7-8 refueling, the power uprate test program was conducted while bringing reactor power from zero to full power (1658 MWt). Similarly, the EPU Project developed a startup testing program, in several phases, to increase power up to the licensed power level of 1912 MWt. See Chapter 14 for details of the various startup testing programs from original plant construction through EPU.

1.1.1.4 Normal Operations

Following the successful completion of the preliminary tests, the plant was assigned commercial operation status, with all plant functions performed by licensed personnel in accordance with approved procedures and operating license requirements.

1.1.1.5 Emergency Plans

Detailed written procedures have been prepared for emergency situations. The appropriate personnel have been trained in these procedures. Periodic tests and reviews are conducted. Emergency planning is briefly discussed in Section 13.3, and a detailed discussion is contained in the DAEC Emergency Plan. Procedures to handle emergencies have been developed in cooperation with such agencies as the Iowa State Health Department, Iowa State Police, local hospital and medical personnel, officials and appropriate agencies of Linn County, Cedar Rapids, and Palo, Iowa, and the Nuclear Regulatory Commission (NRC).

1.1.1.6 Records

Complete records of all operations, tests, and maintenance activities are maintained to enable the On-site Review Group and the various NRC divisions to review and analyze all operations.

1.1.1.7 Operational Reviews and Audits

Routine review of operations is a responsibility of the plant staff and the On-site Review Group. Independent audits of critical functions of operations, including nuclear safety and NRC license provisions, are performed regularly.

1.1.2 METHODS OF TECHNICAL PRESENTATION

1.1.2.1 Purpose

The purpose of this UFSAR is to provide the technical information required by 10 CFR 50.34 to establish a basis for the evaluation of the plant with respect to the issuance and maintenance of a facility operating license.

1.1.2.2 Radioactive Material Barrier Concept

Because the safety aspects of this report pertain to the relationship between plant behavior under a variety of circumstances and radiological effects on persons off the site, the report is oriented with respect to the radioactive material barriers. This orientation facilitates the evaluation of the radiological effects of the plant on the environment. Thus, the presentation of technical information is considerably different from that which would be expected in an operating manual, maintenance manual, or nuclear engineer's handbook.

The overriding consideration that determines the depth of detailed technical information presented about a system or component is the relationship of the system or component to the radioactive material barriers. Systems that must operate to preserve the radioactive material barriers are described in the greatest detail. Systems that have little relationship to the radioactive material barriers are described with as much detail as is deemed necessary to establish their functional role in the plant.

1.1.2.3 Organization of Contents

The UFSAR is organized into 17 chapters, each of which consists of a number of sections. The principal architectural and engineering criteria, which define the broad frame of reference within which the station is designed, are set forth in Section 1.2. Section 1.2 also presents a brief description of the station in which the nuclear safety systems and engineered safeguards are separated from the other systems, so that those systems essential to safety are clearly identified. Chapters 2 through 14 present detailed information about the design and operation of the plant. In these sections, nuclear safety

systems and engineered safeguards are integrated into sections according to system function (emergency core cooling, control), system type (electrical, mechanical), or according to their relationship to a particular radioactive material barrier. Chapter 4, "Reactor," describes plant components and presents design details that are most pertinent to the fuel barrier. Chapter 5, "Reactor Coolant System and Connected Systems," describes plant components and systems that are most pertinent to the nuclear system process barrier. Chapter 6 describes the engineered safety features, which include the primary and secondary containments. Thus, Chapters 4, 5, and 6 are arranged according to the four radioactive material barriers.

The remainder of the sections group system information according to plant function (radioactive waste control, emergency core cooling, power conversion, control) or system type (electrical, structures). Chapter 15, "Accident Analyses," provides an overall safety evaluation of the plant that demonstrates both the adequacy of equipment designed to protect the radioactive material barriers and the ability of the safeguard features to mitigate the consequences of situations in which one or more radioactive material barriers are assumed damaged.

Chapter 16, "Technical Specifications," is maintained separately from the UFSAR now that the DAEC has an operating license.

Chapter 17, "Quality Assurance," describes the quality assurance program that was used during the design and construction phase. The program now in use during the operations phase is referenced in Section 17.2.

1.1.2.4 Format Organization of UFSAR Sections

Sections are numerically identified by representing their order of appearance by two numbers separated by a period, for example 3.4, the fourth section in Chapter 3. Sections are further subdivided by numbers separated by periods (3.4.1, 3.4.1.1, etc.). Pages within each section are consecutively numbered (3.4-1, 3.4-2, etc.).

Tabulations of data are designated "Tables" and are identified by the section number followed by a dash and the number of the table according to its order of mention in the text, for example, Table 3.4-5 is the fifth table of Section 3.4. Drawings, pictures, sketches, curves, graphs, and engineering diagrams are identified as "Figures" and are numbered in the same manner as tables. Figure 7.1-1 defines the meanings of piping and instrumentation symbols used in the figures of this report.

The general organization of a section describing a system or component is as follows:

1. Objective.
2. Design basis.
3. Description.
4. Evaluation.

5. Inspection and testing.

To clearly distinguish the safety versus operational aspects of a system, the objective, design basis, and evaluation titles are modified by the word "safety" or "power generation," according to the definitions given in Section 1.1.3. Systems that have safety objectives are nuclear safety systems or engineered safeguards. A safety evaluation is included only when the system has a safety design basis. A power generation evaluation is included only when needed to clarify the safety versus operational aspects of a system that has both safety and power generation functions. Sections presenting information on topics other than plant systems or components are arranged individually according to the subject matter so that the relationship between the subject and public safety is emphasized.

Within each section of the text, applicable supporting technical material is referenced. References are cited in a list of references at the end of the text. Most of the references are cited as a particular technical basis for boiling-water reactor (BWR) plant design and analysis. However, a few of the references cited in the report refer to special technical work performed by General Electric that is specifically applicable to the DAEC. The references in this category generally provide a full development and analysis of some aspect of General Electric BWR plant technology. These special references are incorporated by reference into the UFSAR in Section 1.6 in support of the license application and subsequent amendments or reviews.

See Section 1.6 for additional information on material incorporated by reference.

1.1.2.5 Power-Level Basis for Analysis of Abnormal Operational Transients and Accidents

For those abnormal operational transients and accidents for which high-power operation increases the severity of the results, the analyses assume plant operation at 102% of rated power as an initial condition. For those events for which an initial condition of low- or intermediate-power level operation renders the most severe results, the analyses presented in this report represent the most severe case within the operating spectrum.

1.1.3 DEFINITIONS

The following definitions apply to the terms used in this UFSAR.

NOTE: while many of these terms appear in the Technical Specifications, the UFSAR definitions may be different. See Technical Specification Section 1.1 for the definitions used in the Technical Specifications.

Radioactive Material Barrier

A radioactive material barrier includes the systems, structures, or equipment that together physically prevent the uncontrolled release of radioactive materials. The four barriers are identified as follows:

1. Reactor fuel barrier - Uranium dioxide sealed in metal cladding.

2. Nuclear system process barrier - The systems of vessels, pipes, pumps, tubes, and other process equipment that contain the steam, water, gases, and radioactive materials coming from, going to, or in communication with the reactor core. The actual boundaries of the nuclear system process barrier depend on the status of plant operation.

   For example, process system isolation valves, when closed, form part of the barrier. The steam jet air ejector offgas path forms a planned process opening in the barrier during power operation. Because the nuclear system process barrier is designed to be divided by isolation valve action into two major sections under certain conditions, this barrier is considered in two parts as follows:

   a. Nuclear system primary barrier - The reactor vessel and attached piping out to and including the second isolation valve in each attached pipe. In various codes and standards used in the industry, this barrier is sometimes referred to as the "primary system pressure boundary."

   b. Nuclear system secondary barrier - That portion of the nuclear system process barrier not included in the nuclear system primary barrier.

3. Primary containment - The drywell in which the reactor vessel is located, the pressure suppression chamber, and process line reinforcements out to the first isolation valve outside the containment wall. Portions of the nuclear system process barrier may become part of the primary containment, depending on the location of a postulated failure. For example, a closed main steam line isolation valve is part of the primary containment barrier when the postulated failure of the main steam line is inside the primary containment.

4. Secondary containment - The reactor building, which completely encloses the primary containment. The reactor building ventilation system and the standby gas treatment system constitute controlled process openings in this barrier.

Nuclear System

The nuclear system generally includes those systems most closely associated with the reactor vessel that are designed to contain or be in communication with the water and steam coming from or going to the reactor core. The nuclear system includes the following:

1. Reactor vessel.

2. Reactor vessel internals.

3. Reactor core.

4. Main steam lines from reactor vessel to the isolation valves outside the primary containment.

5. Neutron monitoring system.

6. Reactor recirculation system.

7. Control rod drive (CRD) system.

8. Residual heat removal (RHR) system.

9. Reactor core isolation cooling (RCIC) system.

10. Emergency core cooling systems.

11. Reactor water cleanup (RWCU) system.

12. Feedwater system piping between the reactor vessel and the first valve outside the primary containment.

Safety

The word "safety," when used to modify such words as objective, design basis, action, and system, indicates that the objective, design basis, action, or system is related to concerns considered to be of primary safety significance, as opposed to the plant mission--the generation of electrical power. Thus, the word "safety" is used to identify aspects of the plant that are considered to be of primary importance with respect to safety.

Power Generation

The phrase "power generation," when used to modify such words as objective, design basis, action, and system, indicates that the objective, design basis, action, or system is related to the mission of the plant--the generation of electrical power--as opposed to concerns considered to be of primary safety importance. Thus, the phrase "power generation" is used to identify aspects of the plant that are not considered to be of primary importance with respect to safety.

Operational

The word "operational" is used in reference to the working or functioning of the plant, in contrast to the design of the plant.

Scram

Scram is the shutdown of the reactor by rapid insertion of control rods.

Safety Limit

A safety limit is an established limit above normal operational limits on the value of a nuclear system process or analytical variable, or an established limit specifying an allowable degree of barrier damage.

Planned Operation

Planned operation is normal plant operation under planned conditions in the absence of significant abnormalities. Operations subsequent to an incident (transient, accident, or special event) are not considered planned operations until the actions taken in the plant are identical to those that would have been used had the incident not occurred. The established planned operations can be considered as a chronological sequence: refueling → achieving criticality → heatup → power operation → achieving shutdown → cooldown → refueling.

The following planned operations are identified:

1. Refueling - Refueling includes all of the planned operations associated with a normal refueling except those tests in which the reactor is taken out of and returned to the shutdown (more than one rod subcritical) condition. The following operations are included in refueling:

    a. Planned physical movement of core components (fuel, control rods, etc.).

    b. Refueling test operations (except criticality and shutdown margin tests).

    c. Planned maintenance.

2. Achieving criticality - Achieving criticality includes the plant actions that are normally accomplished in bringing the reactor from a shutdown condition to a condition in which nuclear criticality is achieved and maintained.

3. Heatup - Heatup begins where achieving criticality ends and includes plant actions normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through the warmup and synchronization of the turbine-generator.

4. Power operation - Power operation begins where heatup ends and includes continued plant operation at power levels in excess of heatup power.

5. Achieving shutdown - Achieving shutdown begins where power operation ends and includes plant actions normally accomplished in achieving nuclear shutdown (more than one rod subcritical) following power operation.

6. Cooldown - Cooldown begins where achieving shutdown ends and includes plant operations for achieving and maintaining the desired conditions of nuclear system temperature and pressure.

Incident

An incident is any event, that is, abnormal operational transient, accident, special event, or other event, not considered as part of planned operation.

Abnormal Operational Transient

An abnormal operational transient includes the events following a single equipment malfunction or a single operator error that is reasonably expected during the course of plant operations. Power failures, pump trips, and rod withdrawal errors are typical of the single malfunctions or errors initiating the events in this category.

Accident

An accident is a single event, not reasonably expected during the course of plant operations, that has been hypothesized for analysis purposes or postulated from unlikely but possible situations and that causes or threatens a rupture of a radioactive material barrier. A pipe rupture qualifies as an accident; a fuel cladding defect does not.

Design-Basis Accident

A design-basis accident is a hypothesized accident, the characteristics and consequences of which are used in the design of those systems and components pertinent to the preservation of radioactive material barriers and the restriction of radioactive material release from the barriers. The potential radiation exposures resulting from a design-basis accident are greater than any similar accident postulated from the same general accident assumptions. For example, the consequences of a complete severance of a recirculation loop line are more severe than those resulting from any other single pipeline failure inside the primary containment.

Special Event

A special event is an event that qualifies neither as an abnormal operational transient nor an accident, but which is postulated to demonstrate some special capability of the plant or its systems.

Safety Action

A safety action is an ultimate action in the plant that is essential to the avoidance of specified conditions considered to be of primary safety significance. The specified conditions are those that are most directly related to the ultimate limits on the integrity of the radioactive material barriers or the release of radioactive material. There are safety actions associated with planned operation, abnormal operational transients, accidents, and special events. Safety actions include such actions as the indication to the operator of the values of certain process variables, reactor scram, emergency core cooling, and reactor shutdown from outside the control room (see Figures 1.1-1 and 1.1-2).

Protective Action

A protective action is an ultimate action at the system level that contributes to and is essential to the accomplishment of a safety action. System level actions that are essential to accomplishing reactor scram, reactor vessel isolation, containment isolation, pressure relief, automatic depressurization, and emergency core cooling are examples of protective actions (see Figures 1.1-1, 1.1-2, and 1.1-3).

Protective Function

A protective function encompasses the monitoring of one or more plant variables or conditions and the associated initiation of intrasystem actions that eventually result in protective action (see Figure 1.1-2).

Safety System

A safety system is any system, group of systems, component, or group of components the actions of which are essential to accomplishing a safety action (see Figure 1.1-2).

Nuclear Safety System

A nuclear safety system is a safety system the actions of which are essential to a safety action required in response to an abnormal operational transient (see Figure 1.1-2).

Engineered Safeguard

An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents (see Figure 1.1-2).

Protection System

Protection system is a generic term that may be applied to nuclear safety systems and engineered safeguards (see Figure 1.1-2).

Special Safety System

A special safety system is a safety system the actions of which are essential to a safety action required in response to a special event (see Figure 1.1-2).

Safety Objective

A safety objective describes in functional terms the purpose of a system or component as it relates to conditions considered to be of primary significance to the protection of the public. This relationship is stated in terms of radioactive material barriers or radioactive material release. The only systems that have safety objectives are safety systems (see Figure 1.1-2).

Safety Design Basis

The safety design basis for a safety system states in functional terms the unique design requirements that establish the limits within which the safety objective shall be met.

A power generation system may have a safety design basis that states in functional terms the unique design requirements that ensure that neither planned operation nor operational failure of the system results in conditions for which plant safety actions would be inadequate.

Safety Evaluation

A safety evaluation is an evaluation that shows how the system satisfies the safety design basis. A safety evaluation is performed only for those systems having a safety design basis.

Power Generation Evaluation

A power generation evaluation is an evaluation that shows how the system satisfies some or all of the power generation design bases. Because power generation evaluations are not directly pertinent to public safety, they are generally not included. However, where a system or component has both safety and power generation objectives, a power generation evaluation can be used to clarify the safety versus power generation capabilities.

Rated Power

Rated power refers to operation at a reactor power of 1912 MWt; this is also termed 100% power. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power.

Design Power

Design power refers to the power level at which the reactor is producing 102% of reactor vessel rated steam flow. The stated design power in megawatts thermal is the result of a heat balance for a particular plant design. For the DAEC, design power is 1950 MWt. This is the power level at which the design-basis accident analyses are performed and is also referred to as design-basis power.

Single Failure

A single failure is a failure that can be ascribed to a single causal event. Single failures are considered in the design of certain systems and are presumed in the evaluations of incidents to investigate the ability of the plant to respond in the required manner under degraded conditions. The nature of a single causal event to be presumed depends on the risk of the event being evaluated. Reasonably expected single failures are presumed as the causes of abnormal operational transients. Single failures of passive equipment are assumed sometimes to be the causes of accidents. Safety actions essential in response to abnormal operational transients and accidents must be carried out in spite of single failures in active equipment. In any case, a single failure includes the multiple effects resulting from the single causal event.

Operable

A system or component is considered operable when it is capable of performing its required action in its required manner.

Operating

A system or component is operating when it is performing its required action in its required manner.

Shutdown

The reactor is shut down when the effective multiplication factor (k_{eff}) is sufficiently less than 1.0 that the full withdrawal of any one control rod could not produce criticality under the most restrictive potential conditions of temperature, pressure, core age, and fission product concentration.

Shutdown Mode

The reactor is in the shutdown mode when the reactor is shut down, the reactor mode switch is in the shutdown mode position, and all operable control rods are fully inserted.

Cold Shutdown Condition

The reactor is in the cold shutdown condition when the reactor is in the shutdown mode, the reactor coolant is maintained at less than 212°F, and the reactor vessel is vented to the atmosphere.

Place in Shutdown Mode

Place in the shutdown mode means conduct an uninterrupted normal plant shutdown operation until the shutdown mode is attained.

Place in the Cold Shutdown Condition

Place in the cold shutdown condition means conduct an uninterrupted normal plant shutdown operation until the cold shutdown condition is attained.

Refuel Mode

The reactor is in the refuel mode whenever the mode switch is in the refuel mode position.

Startup Mode

The reactor is in the startup mode whenever the reactor mode switch is in the startup mode position.

Run Mode

The reactor is in the run mode whenever the reactor mode switch is in the run mode position.

Place in Isolated Condition

Place in isolated condition means conduct an uninterrupted normal isolation of the reactor from the main condenser, including the closure of the main steam line isolation valves.

Primary Containment Integrity

Primary containment integrity means that the drywell and suppression chamber are closed and all of the following conditions are satisfied:

1. All non-automatic primary containment isolation valves that are not required to be open for plant operation are closed.

2. At least one door in the airlock is closed and sealed.

3. All automatic containment isolation valves are operable or are secured in the closed position.

4. All blind flanges and manways are closed.

Secondary Containment Integrity

Secondary containment integrity means that the reactor building is closed and the following conditions are met:

1. At least one door at each access opening is closed.

2. The standby gas treatment system is operable.

3. All reactor building ventilation system automatic isolation valves are operable or are secured in the closed position.

Refueling (As Applicable to Surveillance Frequency Requirements)

For the purpose of designating the frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the end of the previous refueling outage, the test or surveillance need not be performed until the next regularly scheduled outage.

Core Alteration

A core alteration is the manual addition, removal, relocation, or other manual-physical movement of fuel or reactivity controls in the reactor core. Control rod movement with the CRD system is not defined as a core alteration.

Reliability

Reliability is the probability that an item will perform its specified function without failure for a specified time period in a specified environment.

Availability

Availability is the probability that an item will be operable when called on to perform its specified function.

Test Duration

The test duration is the elapsed time between test initiation and test termination.

Test Interval

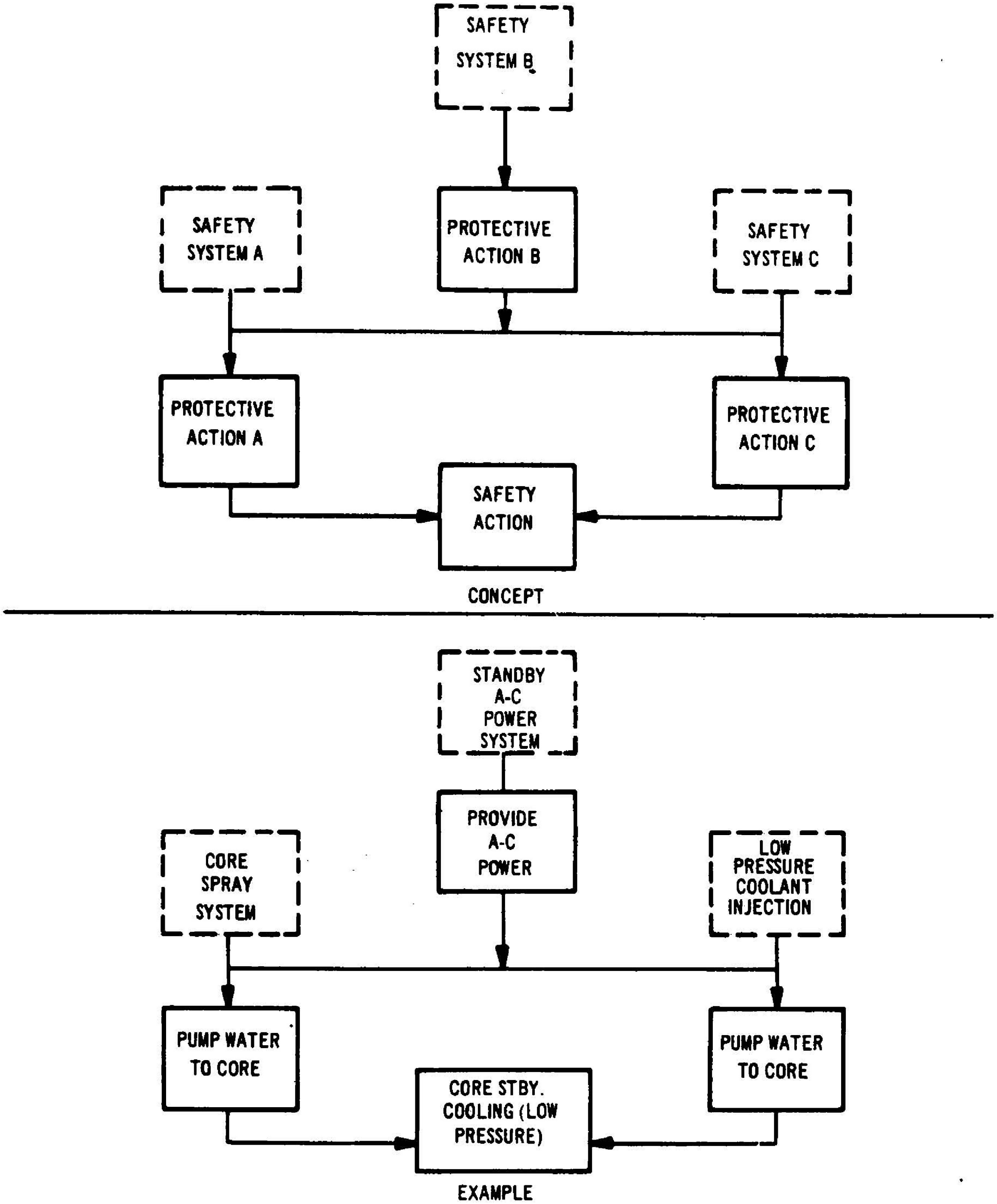
The test interval is the elapsed time between the initiation of identical tests.

Active Component

A device characterized by an expected significant change of state or discernible mechanical motion in response to an imposed design-basis load demand on the system. Examples are switch, relay, valve, pressure switch, turbine, transistor, motor, damper, pump and analog meter.

Passive Component

A device characterized by an expected negligible change of state or negligible mechanical motion in response to an imposed design-basis load demand on the system. Examples are cable, piping, valve in stationary position, resistor, capacitor, fluid filter, indicator lamp, cabinet, and case.



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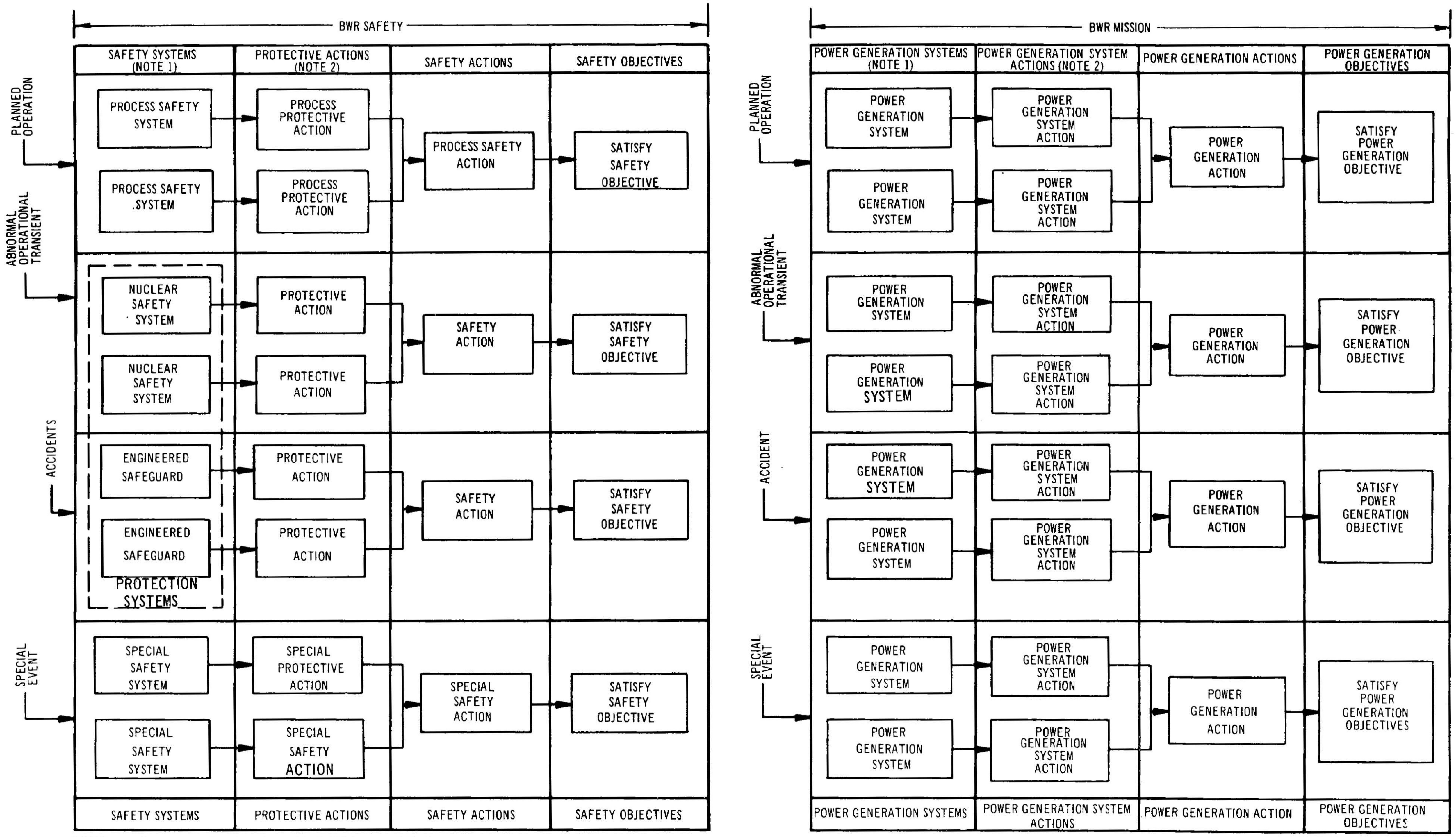
Relationship Between Safety Action and Protective Action

Figure 1.1-1



NOTES: 1. ONLY TWO SYSTEMS OF EACH TYPE ARE SHOWN. THERE MAY BE MORE THAN THIS NUMBER OF SYSTEMS IN ANY CATEGORY.

2. THERE MAY BE CASES WHERE THE SYSTEM LEVEL ACTION IS IDENTICAL TO THE ULTIMATE ACTION IN THE PLANT, IN SUCH A CASE THE INTERMEDIATE SYSTEM LEVEL ACTION NEED NOT BE IDENTIFIED.

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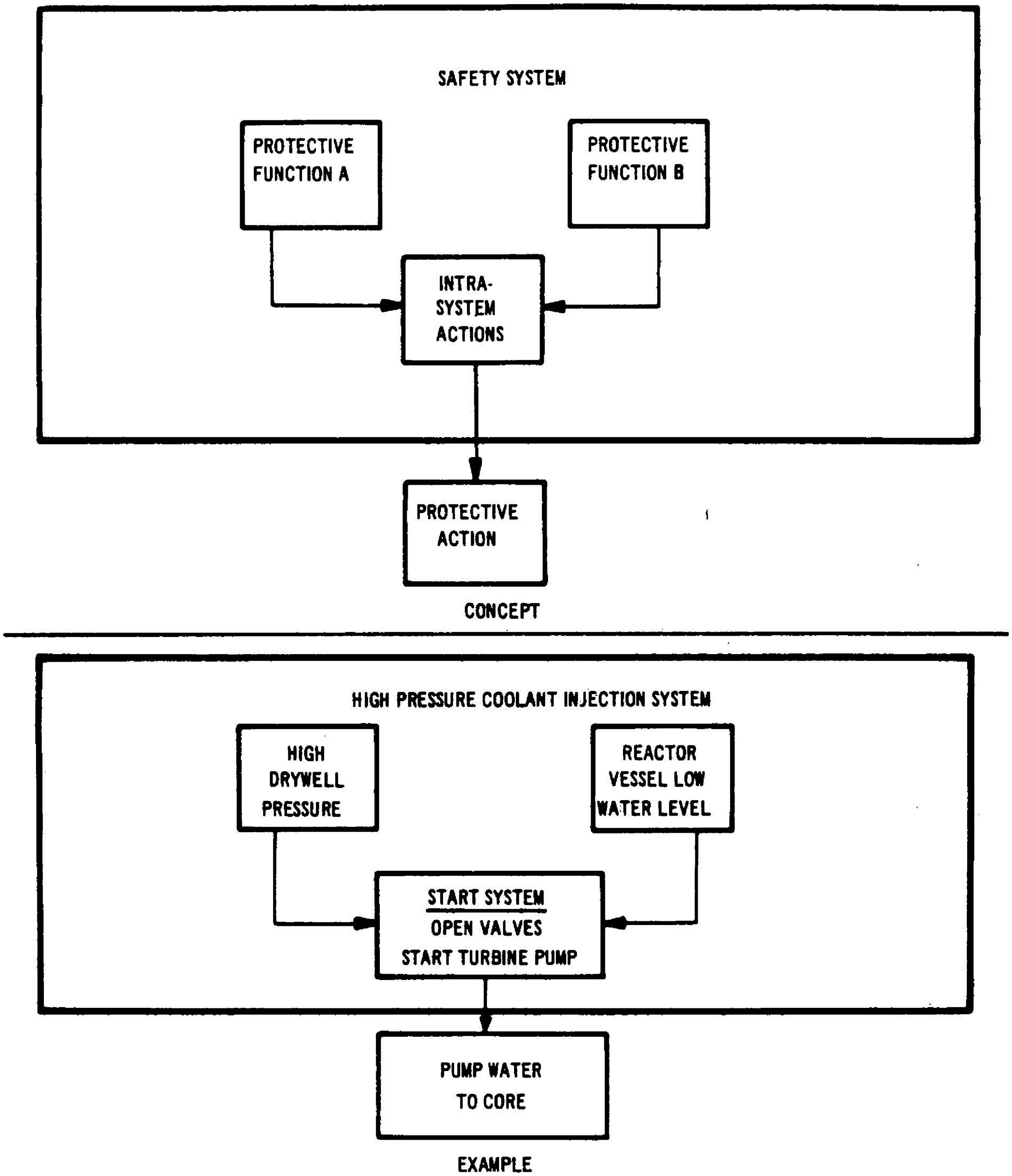
Relationship Between Different Types of Systems, Actions, and Objectives

Figure 1.1-2



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Relationship Between Protective Functions and Protective Actions

Figure 1.1-3

1.2 GENERAL PLANT DESCRIPTION

A cross-reference to the portions of this report applicable to the subject matter is made by the parenthetical number at the end of each descriptive paragraph.

1.2.1 LOCATION AND BOUNDARIES

The plant site is adjacent to the Cedar River approximately 2.5 miles northeast of the Village of Palo, Iowa. The closest city is Cedar Rapids with its outer boundary 8 miles to the southeast. The site containing approximately 500 acres is entirely owned by NextEra Energy Duane Arnold, LLC. [redacted] The boundary of the exclusion area defined in 10 CFR 100.11 is delineated by the property lines shown in Figure 1.2-1. These lines also form the boundary of the unrestricted area referred to in 10 CFR 20. (Note: these definitions were not modified as part of the conversion to the Alternative Source Term (10 CFR 50.67).) Also shown in Figure 1.2-1 are [redacted] In addition, this figure shows the [redacted]. The plant boundaries are also shown in the site aerial photograph (Figure 2.1-1) and the site topographical map (Figure 2.1-2). A paved county road provides access to the site. Figure 2.1-3 indicates the access routes. (Section 2.1)

1.2.2 SITE TOPOGRAPHY

A relatively flat plain at approximately 750 ft. above mean sea level (msl) extends to the site toward the Village of Palo to the southwest. Across the river from the site, the land rises from an elevation of 750 ft. to an elevation of about 900 within a horizontal distance of 2000 ft. These slopes are heavily wooded. To the northwest, land rises to an elevation of 850 ft. Adjacent to the east is a heavily wooded low area. General topographical features in this portion of the Cedar River drainage area consist of broad valleys with relatively narrow floodplains. (Section 2.1)

1.2.3 POPULATION AND LAND USE

[redacted] Population center distance is 8 miles, and distance to the outer boundary of the low-population zone (LPZ) is 6 miles. (Note: these definitions were not modified as part of the conversion to the Alternative Source Term (10 CFR 50.67).) Figures 2.1-4 through 2.1-8 show the present and projected future populations of the surrounding area. (Section 2.1)

1.2.4 FACILITY ARRANGEMENT

The layout of the site provides for locating the plant facilities in such a manner as to provide for ease of access and efficient operation.

The principal buildings and structures that [redacted]

The location and orientation of the buildings on the site are shown in Figure 1.2-2. Figures 1.2-3 through 1.2-19 depict the general arrangement of the buildings.

These buildings and structures are designed and constructed in accordance with codes applicable to the building's use and safety-related functions. For structures important to plant safety, reasonable and appropriate conservatism is incorporated into the design to ensure that the safety-related function is not interrupted.

Shielding and access control are provided to limit radiation exposure to plant personnel. The radiation protection depends on the area in question, its use, and the degree of occupancy.

1.2.4.1 <u>Reactor Building</u>

[redacted] The reactor building is basically a reinforced-concrete structure, consisting of the following major structural components:

1. Reinforced-concrete foundation mat, supported directly on rock.

2. Reinforced-concrete floors supported by precast concrete tee beams at grade and by structural steel framing above grade.

3. Reinforced-concrete or concrete block interior walls.

4. [redacted]

The lining used for the spent-fuel pool and reactor basin cavity is 0.25-in. stainless steel plate, which provides a testable, leaktight membrane for these pools. The dryer-separator pool is coated with 1/8-in. to 0.25-in. epoxy surfacer and a fiberglass-reinforced phenolic topcoat for ease of decontamination. Since this pool is normally dry and accessible for inspection, any postulated cracking of this coating will be readily detectable and repairable before water immersion.

In the event a crack does develop on the inside surface of the dryer-separator pool, repairs may be made easily by locally chipping out the cracked section and replacing the coating system.

Should cracking occur, it is expected that the spacing would be approximately 5 in. and the maximum crack width would be 1.5×10^{-3} in.

The properties of the epoxy surfacer are listed below:

Compressive strength, 7 days (ASTM C579-68)	10,000 psi
Tensile strength, 7 days (ASTM C307-61)	2200 psi
Modules of elasticity, 7 days (ASTM C580-68)	1.6×10^{6} psi
Flexural strength, 7 days (ASTM C580-68)	4000 psi
Coefficient of thermal expansion (ASTM C531-68)	19.33×10^{6} in./in. °F
Linear shrinkage, 7 days (ASTM C531-68)	0.14%
Taber abrasion, 7 days (Modified ASTM C501-62)	19.7 index
Absorption (ASTM C413-66)	0.1%
Shear bond to steel, 7 days	1500 psi
Bond strength to 3000 psi concrete (ASTM C321-64)	304 psi (concrete failure)

5. [redacted] It is shaped similarly to the drywell, which consists of a light-bulb-shaped steel vessel.

6. The reactor support is a reinforced-concrete pedestal. It also serves as a support for the sacrificial shield and two main steel service platforms.

7. Exterior walls up to the refueling floor consist of exterior precast concrete panels, attached to steel columns, which form the exterior face of cast-in-place concrete-bearing walls.

8. The structure above the refueling floor consists of a steel rigid frame that acts as support for built-up roofing over insulated metal decking. The rigid frame supports a 100-ton traveling bridge crane. Insulated metal siding is used above the refueling floor.

9. [redacted] The base, a slab for this structure, is at the same level and is a continuation of the reactor building base slab. The structure is constructed of reinforced concrete.

1.2.4.2 Turbine Building

[redacted] The turbine building is a steel and concrete structure consisting of the following major structural components:

1. Reinforced-concrete foundation mat.

2. Reinforced-concrete floor supported by structural steel framing.

3. Interior walls of reinforced concrete or concrete block.

4. Exterior walls of reinforced concrete below grade and precast concrete panels above grade to the operating floor. Insulated metal siding above the operating floor.

5. Steel superstructure of rigid frame construction in the main turbine house and of braced framed construction in the auxiliary bay. Rigid frame supports runway for a 125-ton crane.

6. Roof of built-up roofing over insulation on metal decking covered by insulation and a single film vinyl cover.

7. Turbine pedestal of reinforced concrete supported on the foundation mat.

1.2.4.3 Offgas Stack

The 100-m offgas stack discharges gases to the atmosphere from the standby gas treatment and offgas systems. It is provided with required appurtenances, such as aviation obstruction lights and radiation monitoring instruments, in accordance with applicable codes and regulations. It is designed in accordance with the criteria for

Seismic Category I structures. However, the tornado design criteria are excluded because a collapse of the stack would not prevent safe plant shutdown. It is more than 100 m from the nearest Seismic Category I structure or equipment.

1.2.4.4 Radwaste Building

[redacted]

The building is a steel and concrete structure consisting of the following major structural components:

1. Reinforced-concrete foundation mat.

2. Reinforced-concrete floors supported by structural steel framing.

3. Reinforced-concrete (poured and precast) exterior walls.

4. Reinforced-concrete or concrete block interior walls.

5. Reinforced-concrete slab on metal roof deck system supported by steel framing.

1.2.4.5 Control Building

[redacted] The building is a steel and concrete structure consisting of the following major structural components:

1. Reinforced-concrete foundation mat.

2. Reinforced-concrete floors supported by structural steel framing.

3. Reinforced-concrete (poured and precast) exterior walls.

4. Reinforced-concrete or concrete block interior walls.

5. Reinforced-concrete slab on metal roof deck system supported by steel framing.

1.2.4.6 Pump House

[redacted] The pump house is a reinforced-concrete structure.

1.2.4.7 Intake Structure

[redacted] The structure is of reinforced-concrete construction. [redacted]

1.2.4.8 Cooling Towers

Two induced-draft cooling towers are used in the closed loop condenser circulating water system to remove heat rejected by the main condenser.

This system is described in Section 10.4.5.

1.2.4.9 Administration Building

The administration building houses the office facilities for plant management and related functions. [redacted]
2016-004 structure composed of structural steel framework supported on reinforced-concrete grade beams. Exterior walls consist of precast concrete panels and metal and glass curtain walls. Floor slabs are of cast-in-place concrete supported by structural steel framing with metal decking. [redacted]

1.2.4.10 Machine Shop

[redacted] It is divided into general shop areas, tool room, maintenance office, toilet room, and decontamination area with all facilities serviced by a 5-ton overhead bridge crane.

1.2.4.11 Guard Facility

[redacted]

1.2.4.12 Offgas Retention Building

[redacted]

1.2.4.13 Technical Support Center

[redacted]

1.2.4.14 Low-Level Radwaste Processing And Storage Facility

[redacted]

1.2.4.15 Training Center

The Training Center provides on-site classrooms for training DAEC personnel. It also contains offices and administrative areas.

1.2.4.16 Data Acquisition Center

[redacted] It provides office space for several plant and engineering support personnel.

1.2.4.17 Plant Support Center

The Plant Support Center provides office space for plant and engineering support personnel.

1.2.4.18 Air Compressor Building

[redacted]

1.2.4.19 Independent Spent Fuel Storage Installation (ISFSI)

An ISFSI is a 10 CFR 72 licensed facility. [redacted]

1.2.4.20 Plant Access Building (PAB)

[redacted]

1.2.5 SYSTEM DESCRIPTIONS

1.2.5.1 Nuclear System

The nuclear system includes a single-cycle, forced-circulation, General Electric BWR producing steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for rated power is shown in Figure 15.0-3.

1.2.5.1.1 Reactor Core and Control Rods

The fuel for the reactor core consists of slightly enriched uranium dioxide pellets contained in sealed Zircaloy-2 tubes. These fuel rods are assembled into individual fuel assemblies. The number of [redacted] (Chapter 4)

[redacted] The control rods are of cruciform shape and are dispersed throughout the lattice of fuel assemblies. The rods are controlled by a hydraulic system. (Sections 7.7 and 3.9.4)

1.2.5.1.2 Reactor Vessel and Internals

The reactor vessel contains the core and supporting structure; the steam separators and dryers; the jet pumps; the control rod guide tubes; distribution lines for the feedwater, core spray, and standby liquid control systems; the incore instrumentation; and other components. The main connections to the vessel include the steam lines, the coolant recirculation lines, feedwater lines, CRD housings, and other emergency core cooling lines.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig. The nominal operating pressure is 1025 psig in the steam space above the separators. The vessel is fabricated of carbon steel and is clad internally with stainless steel, with the exception of the top head.

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 reactor vessel. The steam is then directed to the turbine through the main steam lines. Each steam line is provided with two isolation valves in series--one on each side of the primary containment barrier. (Sections 3.9 and 5.3)

1.2.5.1.3 Reactor Recirculation System

The reactor recirculation system pumps reactor coolant through the core to remove the energy generated in the fuel. [redacted] Each loop has one motor-driven recirculation pump. Recirculation pump speed can be varied to allow some control of the reactor power level through the effects of coolant flow rate on moderator void content. (Section 5.4)

1.2.5.1.4 Residual Heat Removal System

The RHR system is a system of pumps, heat exchangers, and piping that fulfills the following functions:

1. Removal of decay heat during and after plant shutdown.

2. Injection of water into the reactor vessel following a loss-of-coolant accident (LOCA) rapidly enough to reflood the core and prevent excessive fuel clad temperatures. This is discussed in Section 1.2.5.6.

3. Removal of heat from the primary containment following a LOCA to limit the increase in primary containment pressure. This is accomplished by cooling and recirculating the water inside the primary containment. The redundancy of the equipment provided for containment cooling is further extended by a separate part of the RHR system that sprays cooling water into the drywell and the suppression pool air space. This latter capability is discussed in Section 1.2.5.6.7. (Section 5.4)

1.2.5.1.5 Reactor Core Isolation Cooling System

The RCIC system operates automatically to supply the reactor vessel with enough water so that the operation of the emergency core cooling systems (engineered safeguards) is not actuated. The system uses a steam-turbine-driven pump. (Section 5.4)

1.2.5.1.6 Reactor Water Cleanup System

The RWCU system provides a continuous purification of a portion of the reactor recirculation flow. The system can be operated at any time during planned operations.

The major equipment of this system [redacted] and consists of pumps, heat exchangers (both regenerative and nonregenerative), two filter-demineralizers, and the associated valves, piping, and instrumentation. Reactor coolant is removed from the reactor coolant recirculation system and is cooled in the regenerative and nonregenerative heat exchangers. After cooling, the circulated water is filtered and demineralized to reduce the amount of activated corrosion products in the water and returned to the feedwater system.
(Section 5.4)

1.2.5.2 <u>Power Conversion System</u>

1.2.5.2.1 Turbine-Generator

The turbine is an 1800-rpm condensing turbine consisting of a single-flow high-pressure shell and two double-flow low-pressure shells. The generator rating is 715,225 KVA 0.95 power factor, 0.58 short-circuit ratio, 45-psig hydrogen pressure, 1800 rpm, 60 cycles, 22,000 V, 17,412 A. (Section 10.2)

1.2.5.2.2 Turbine Bypass System

A bypass system is provided that passes steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load permitted to pass to the turbine-generator (such as during startup and generator synchronization or following a large electrical load rejection). (Section 10.4)

1.2.5.2.3 Main Condenser

The main condenser is a single-pass, divided-water-box type, dual-pressure deaerating design. It is mounted on rigid foundations, and flexible expansion joints are provided between the condenser necks and turbine exhaust connections. The divided water boxes permit the operation of either half of the condenser individually. Baffles are provided in the hotwell to hold up condensate for 2 min. The condenser will accept the required bypass flow during startup and operation and not exceed turbine exhaust temperature or pressure limitations. (Section 10.4)

1.2.5.2.4 Main Condenser Gas Removal and Turbine Sealing System

Two full-capacity steam jet air ejectors, with inter and after condensers, are provided to remove air and noncondensibles from the condenser. The noncondensibles,

containing hydrogen and oxygen, are delivered to the offgas system for holdup before delivery to the stack.

A mechanical vacuum pump is provided to evacuate the turbine and condenser during startups and shutdowns when steam pressure is insufficient to operate the steam jet air ejectors or when the steam jet air ejectors are no longer required. The vapors are discharged to the offgas stack.

The turbine sealing system provides steam to the turbine glands to control air leakage into the turbine and external steam leakage from the turbine. The system collects and condenses sealing steam, returns it to the condenser, and discharges the noncondensibles to the gland seal holdup system for radioactive decay before release from the stack. (Section 10.4)

1.2.5.2.5 Circulating Water System

The circulating water system consists of two cooling towers, two circulating pumps, facilities for water makeup from the river, chemical treatment facilities, and blowdown to control water concentrations. (Section 10.4)

1.2.5.2.6 Condensate Demineralizer System

The condensate demineralizer system is composed of five filter demineralizer vessels (only four are normally used, although system improvements currently allow all five to be frequently inservice), associated piping, and instrumentation and controls to enable continuous processing of design flow condensate and disposal of wastes to the radwaste system. (Section 10.4)

1.2.5.2.7 Condensate and Feedwater System

Two half-size motor-driven centrifugal condensate pumps circulate water through the steam packing exhauster condenser, steam jet air ejector inter and after condensers, condensate demineralizer, and five low-pressure feedwater heaters in two parallel strings to the suction of the reactor feedwater pumps.

Two half-size motor-driven centrifugal feedwater pumps supply feedwater through a single high-pressure heater arranged in two parallel strings through feedwater control valves to the reactor. (Section 10.4)

The feedwater heaters are of shell and tube construction. They each have integral drain coolers with the exception of the lowest pressure heater, which has a separate shell for the drain cooler. Two low-pressure heaters are located in the condenser neck. All heaters have stainless steel tubes. The heaters also are equipped with stainless steel baffles at the steam entrance and drain connections to minimize erosion.

All heater shells are vented directly to the main condenser. The heater drains flow by pressure differential to the next lower stage heater and from the last stage drain cooler to the main condenser. (Section 10.4)

1.2.5.3 Electrical Power System

Normal Auxiliary ac power is supplied by a startup transformer during periods when the main generator is not producing power. The power source for the startup transformer is the 161-KV bus in the main substation. When the main generator has been synchronized to the system, auxiliary power is supplied by an auxiliary power transformer connected to the generator isolated phase bus.

The plant auxiliary power bus is made up of four sections. [redacted] Each of the four sections of the auxiliary power bus has [redacted]
2017-015 energized from the 161kV busses. (Chapter 8)

1.2.5.4 Radioactive Waste Systems

1.2.5.4.1 Liquid Radwaste System

The liquid radioactive waste control system collects, treats, stores, and disposes all radioactive liquid wastes. These wastes are collected in sumps and drain tanks at various locations throughout the plant and then transferred to the appropriate collection tanks in the radwaste building for treatment, storage, and disposal. Wastes to be discharged from the system are processed on a batch basis with each batch being processed by such method or methods appropriate for the quality and quantity of materials determined to be present. Processed liquid wastes are normally returned to the condensate system but may be discharged to the environs through the effluent line, depending on plant water inventory and water-quality considerations. The liquid wastes discharged to the effluent line are diluted with river water.
(Section 11.2)

1.2.5.4.2 Solid Radwaste System

The solid radwaste system collects, processes, and packages potentially radioactive wet and dry solid wastes. Generally these wastes are stored on the site until the short half-lived activities are insignificant. The wet solid wastes are the spent-demineralizer resins and filter sludges that are a by-product of station water treating processes including radwaste. The dry solid radwastes consist of other miscellaneous radioactive or contaminated solid wastes. Because of differences in radioactivity or

contamination levels of the many wastes, various methods are employed for processing and packaging.

Standard 55-gal steel drums and 4 ft. x 6 ft. x 4 ft. metal containers (approved by the Department of Transportation for radioactive materials) are used for the packaging of dry solid wastes during temporary onsite storage, shipment, and permanent offsite storage. Solid wet wastes are transferred to an 85 ft^3 steel liner wherein they are dewatered, solidified, and transported offsite or dewatered and packaged in high integrity containers and transferred offsite. (Section 11.4)

1.2.5.4.3 Gaseous Radwaste (Offgas) System

The gaseous radwaste system collects gaseous discharges from the main condenser air ejector(s) and gland seal condenser. In addition, this system processes and delivers the gases to the main stack for elevated release to the atmosphere.

Noncondensible radioactive gases are removed from the main condenser by the air ejector. These gases then enter a high-temperature catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen from the air ejector system. This scheme further minimizes any explosion potential in the offgas system. The gas is then chilled to extract condensibles and reduce the volume. Remaining noncondensibles (mainly kryptons, xenons, and air) are delayed in the 30-min holdup subsystem for radioactive decay. Gaseous effluent subsequently has additional moisture removed, is reheated, passes through a prefilter, and proceeds to the charcoal adsorption bed area where the charcoal beds operating in a constant-temperature vault selectively adsorb and delay the xenons and kryptons from the bulk carrier gas (principally air). This delay of the gas on the charcoal allows the xenon and krypton to decay further and undergo dynamic adsorption by activated carbon. After residence in the charcoal beds, the gas effluent passes through a high-efficiency after-filter and proceeds to the elevated release point.

The system is designed to accommodate the possible explosive hazard due to the hydrogen and oxygen present from the radiolytic decomposition of reactor coolant.

The gland seal condenser is exhausted by a blower into shielded piping that provides approximately 2-min holdup to reduce the activity of short-lived radioactive gases (Nitrogen-16 and Oxygen-19), which are then discharged to the main stack. (Section 11.3)

1.2.5.5 <u>Nuclear Safety Systems</u>

1.2.5.5.1 Neutron Monitoring System

Although not all of the neutron monitoring system qualifies as a nuclear safety system, those [redacted]. The intermediate range monitors and average power range monitors, which

monitor neutron flux via incore detectors, signal the reactor protection system to scram in time to prevent fuel cladding damage as a result of overpower transients. (Section 7.6)

1.2.5.5.2 Control Rod Drive System

When a scram is initiated by the reactor protection system, the CRD system inserts the control rods that provide the negative reactivity necessary to shut down the reactor. Each control rod is controlled individually by a hydraulic control unit. When a scram signal is received, high-pressure water from an accumulator for each rod forces each control rod rapidly into the core. (Section 4.6)

1.2.5.5.3 Nuclear System Pressure Relief System

A pressure relief system consisting of [redacted] inside the nuclear system following either abnormal operational transients or accidents. (Section 5.2)

1.2.5.6 Engineered Safeguards

1.2.5.6.1 Primary Containment

The primary containment system uses a pressure suppression containment to [redacted] The pressure suppression system consists of a drywell and a pressure suppression chamber that are interconnected through a series of vents, isolation valves, cooling systems, and other equipment.

A process system piping failure inside of the drywell will result in reactor water and steam being released to the drywell atmosphere. The higher drywell pressure will force the steam-water mixture through the connecting vents into water contained in the suppression chamber. The steam will condense within the suppression chamber, rapidly reducing drywell pressure. (Section 6.2)

1.2.5.6.2 Primary Containment and Reactor Vessel Isolation Control System

The primary containment and reactor vessel isolation control system automatically initiates the closure of isolation valves to close off all potential leakage paths for radioactive material to the environs. This action is taken on the indication of a potential breach in the nuclear system process barrier. (Section 7.3.1.1)

1.2.5.6.3 Secondary Containment

The reactor building completely encloses the primary containment. The building will provide secondary containment when the primary containment is closed and in service and primary containment during periods when the primary containment is open,

such as during refueling periods. [redacted]

Reactor building design includes provisions for seismic load resistance and low infiltration and exfiltration rates. [redacted] (Section 6.2)

1.2.5.6.4 Main Steam Line Isolation Valves

The main steam lines, because of their large size and large mass flow rates, are given special isolation consideration. Two automatic isolation valves, each powered by both nitrogen pressure and spring force, are provided in each main steam line. (Section 5.4) These valves fulfill the following objectives:

1. Limit the loss of reactor coolant from the reactor vessel resulting from either a major leak from the steam piping outside the primary containment or a malfunction of the pressure control system resulting in excessive steam flow from the reactor vessel.

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

1.2.5.6.5 Main Steam Line Flow Restrictors

A venturi-type [redacted]. These devices limit the loss of water from the reactor vessel before the main steam line isolation valves are closed in case of a main steam line break outside the primary containment and prevent uncovering of the core. (Section 5.4)

1.2.5.6.6 Emergency Core Cooling Systems

A number of emergency core cooling systems are provided to prevent excessive fuel clad temperatures in the event of a breach in the nuclear system process barrier that results in a loss of reactor coolant. (Section 6.3) The four emergency core cooling systems are:

1. HPCI system.

2. Automatic depressurization system.

3. Core spray system.

4. Low-pressure coolant injection (LPCI) (an operating mode of the RHR system).

HPCI System

The HPCI system provides and maintains an adequate coolant inventory inside the reactor vessel to prevent excessive fuel clad temperatures as a result of small breaks in the nuclear system process barrier. A high-pressure system is needed for small breaks because the reactor vessel depressurizes slowly, preventing low-pressure systems from injecting coolant. 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.

Automatic Depressurization System

The automatic depressurization system acts to rapidly reduce reactor vessel pressure in a LOCA situation in which the HPCI system fails to provide adequate cooling-water flow. The depressurization provided by the system enables the low-pressure emergency core cooling systems to deliver cooling water to the reactor vessel. The automatic depressurization system 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 both that a break in the nuclear system process barrier has occurred and that the HPCI system is not delivering adequate cooling water flow to the reactor vessel, provided that either a core spray or LPCI pump is operating and only after a preset time delay.

Core Spray System

The core spray system consists of two independent pump loops that deliver cooling water to spray spargers over the core. The system is actuated by conditions indicating that a breach exists in the nuclear system process barrier, but water is delivered to the core only after reactor vessel pressure is reduced. This system provides the capability to cool the fuel by spraying water onto the core. Either core spray loop is capable of preventing excessive fuel clad temperatures following a LOCA.

Low-Pressure Coolant Injection

Low-pressure coolant injection is an operating mode of the RHR system but is discussed here because the LPCI mode acts as an engineered safeguard in conjunction with the other emergency core cooling systems. Low-pressure coolant injection uses the pump loops of the RHR system to inject cooling water at low pressure into an undamaged reactor recirculation loop. Low-pressure coolant injection is actuated by conditions indicating a breach in the nuclear system process barrier, but water is delivered to the core only after reactor vessel pressure is reduced. LPCI operation, in combination with Core Spray operation, together with the core shroud and jet pump arrangement, provides the capability of core reflooding following a LOCA in time to prevent excessive fuel clad temperatures. It should be noted for long-term core cooling, when there is 2/3 or less core submergence and the fuel axial flux profile peak is above 2/3 core height, that excessive fuel clad temperatures may occur without core spray in operation.

1.2.5.6.7 Residual Heat Removal System (Containment Cooling)

A portion of the RHR system is provided to spray 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 energy removal capability and can be placed into service at the discretion of the operator. (Section 5.4)

1.2.5.6.8 Standby Liquid Control System

The standby liquid control system provides a redundant, independent, and functionally diverse method of bringing the nuclear fission reaction to subcriticality and maintaining subcriticality as the reactor cools. The system is not intended to provide prompt reactor shutdown, but rather makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect as the nuclear system cools down in the normal manner. (Section 9.3)

1.2.5.6.9 Control Rod Velocity Limiter

A control rod velocity limiter is attached to each control rod to limit the velocity at which a control rod can fall out of the core should it become detached from its control rod drive. The rate of reactivity insertion resulting from a rod-drop accident is limited by this action. The limiters contain no moving parts. (Section 4.6)

1.2.5.6.10 Control Rod Drive Housing Supports

CRD housing supports are located underneath the reactor vessel near the control rod housings. The supports limit the travel of a control rod in the event that a control rod housing is ruptured. The supports prevent a nuclear excursion as a result of a housing failure, thus protecting the fuel barrier. (Section 3.9)

1.2.5.6.11 Standby Gas Treatment System

The standby gas treatment system consists of two identical processing streams, either of which is capable of exchanging the reactor building volume once in a 24-hr period.

The system contains demisters, prefilters, electric air heaters, high-efficiency filters, activated carbon iodine absorbers, fans, and a flow control system to maintain the system design flow.

The system maintains a slight negative internal building pressure and will process all gaseous effluent before discharge from the elevated release point. (Section 6.2)

1.2.5.6.12 Standby AC Power Supply

Two automatic starting full-capacity diesel-generators provide standby ac power. Each engine is equipped with two, independent starting air supply systems.. Each starting air system has the capability of providing a minimum of five normal diesel starts per air receiver without recharging. Fuel supply for the engines consists of one main fuel storage tank and a day tank for each engine. Each diesel-generator is capable of supplying the power required to shut down and maintain the plant in a safe condition in the event of total loss of normal power sources. (Section 8.3)

1.2.5.6.13 DC Power Supply

The plant dc power supply system consists of two 125-V batteries and one 250-V battery, each with its own battery charger, circuit breakers, and buses. One spare 125-V charger may be switched to either 125-V bus to allow for servicing and backup of the normal power supply charger. One spare charger is also provided for the 250-V battery. Each charger has sufficient capacity to restore its battery to full charge within 12 hr while carrying normal steady-state dc loads. Two separate 125-V dc power panel boards are supplied from each bus.

In addition, two independent ±24-V buses are provided, each supplied by a center grounded 48-V battery and a charger. These dc buses supply certain instrumentation such as source and intermediate range monitors. (Section 8.3)

1.2.5.6.14 Residual Heat Removal and Emergency Service Water Systems

Four RHR service water pumps supply water to the RHR service water systems. Two emergency service water pumps supply water to the emergency service water systems. The RHR service water pumps and the emergency service water pumps are located in the pump house taking their suction from the service water wet pits.

The RHR service water pumps in each loop are connected electrically to the same bus as the diesel-generator in their emergency loop to ensure the emergency equipment service water supply in the event the offsite ac power supply is lost. (Section 9.2)

1.2.5.6.15 Main Steam Line Radiation Monitoring System

The main steam line radiation monitoring system consists of four gamma radiation monitors located external to the main steam lines just outside of the primary containment. The monitors are designed to detect a gross release of fission products from the fuel. (Section 11.5)

1.2.5.7 Nuclear System Process Control

1.2.5.7.1 Reactor Manual Control System

The reactor manual control system provides the means by which control rods are manipulated from the control room for gross power control. The system controls valves in the CRD hydraulic system. Only one control rod can be manipulated at a time. The reactor manual control system includes the controls that restrict control rod movement (rod block) under certain conditions as a backup to procedural controls. (Section 7.7)

1.2.5.7.2 Recirculation Flow Control System

The recirculation flow control system controls the speed of the reactor recirculation pumps. Adjusting the pump speed changes the coolant flow rate through the core. This affects changes in core power level. The system is arranged to adjust reactor power output to the load demand by adjusting the frequency of the electrical power supply for the reactor recirculation pumps. (Section 7.7)

1.2.5.7.3 Neutron Monitoring System

The neutron monitoring system is a system of incore neutron detectors and out-of-core electronic monitoring equipment. The system provides an indication of neutron flux, which can be correlated to thermal power level, for the entire range of flux conditions that may exist in the core. The source range monitors and the intermediate range monitors provide flux level indications during reactor startup and low-power operation. The local power range monitors and average power range monitors allow the assessment of local and overall flux conditions during power range operation. Rod block monitors are provided to prevent rod withdrawal when the change in reactor power would exceed a predetermined value that is based on the original power level. The rod block monitors prevent local fuel damage as a backup to procedural power flow restrictions. The flux mapping and calibration subsystem provides a means to calibrate individual monitors with traveling incore probes. (Section 7.6)

1.2.5.7.4 Refueling Interlocks

A system of interlocks that restrict the movements of refueling equipment and control rods when the reactor is in the refuel mode is provided to prevent an inadvertent criticality during refueling operations. The interlocks backup procedural controls that have the same objective. The interlocks affect the refueling bridge, the refueling bridge hoists, the fuel grapple, control rods, and the service platform hoist. (Section 7.6)

1.2.5.7.5 Reactor Vessel Instrumentation

In addition to instrumentation provided for the nuclear safety systems and engineered safeguards, instrumentation is provided to monitor and transmit information that can be used to assess conditions existing inside the reactor vessel and the physical condition of the vessel itself. The instrumentation provided monitors reactor vessel pressure, water level, surface temperature, internal differential pressures and coolant flow
2017-002 rates, and top head flange leakage. (Section 7.6)

1.2.5.7.6 Plant Process Computer System

An online plant process computer is provided to monitor and log process variables, and to make certain analytical computations. The plant process computer provides core fuel performance analysis and display, and display of plant data in remote locations. (Section 7.7)

1.2.5.7.7 Rod Worth Minimizer System

The rod worth minimizer is implemented on a stand-alone microcomputer system that interfaces to the plant process computer. The rod worth minimizer functions to prevent rod withdrawal if the rod to be withdrawn is not in accordance with a preplanned pattern. The effect of the rod block is to limit the reactivity worth of the control rods by enforcing adherence to the preplanned rod pattern.
(Section 7.7).

1.2.5.8 <u>Power Conversion System Process Control</u>

1.2.5.8.1 Pressure Regulator and Turbine-Generator Control

The pressure regulator and the integrated turbine-generator control system work together to allow proper generator and reactor response to load demand changes. The pressure regulator maintains nuclear system pressure essentially constant eliminating the possibility of pressure-induced core reactivity changes. The pressure regulator adjusts turbine control valves or turbine bypass valves while the turbine-generator controls maintain a constant turbine speed. The speed and load controls initiate rapid closure of the turbine control valves and fast opening of the bypass valves in case of loss of generator electrical load. (Section 7.7)

1.2.5.8.2 Feedwater System Control

A three-element controller is used to regulate the feedwater system so that proper water level is maintained in the reactor vessel. The controller uses main steam flow, reactor vessel water level, and feedwater flow to control feedwater. (Section 7.7)

1.2.5.9 Electrical Power System Protection

The electrical power system protection incorporates modern high-speed relaying to protect the main generator, transformers, buses, and transmission lines while providing a highly reliable source of generation and transmission. The system includes generator voltage and load frequency control and metering to monitor and record unit output.

1.2.5.10 Radiation Monitoring and Control

1.2.5.10.1 Process Radiation Monitoring

Radiation monitors are provided on various lines, some to monitor for radioactive materials released to the environs via process liquids and gases, and others to monitor for process system malfunctions (Section 11.5).

1.2.5.10.2 Area Radiation Monitors

A number of radiation monitors are provided to monitor for abnormal radiation at various locations in the plant. These monitors provide alarms when abnormal radiation levels are detected. (Section 12.3)

1.2.5.10.3 Liquid Radwaste Control

Liquid wastes to be discharged are handled on a batch basis with protection against accidental discharge provided by procedural controls. Instrumentation with alarms to detect abnormal concentration of the radwastes is provided. (Section 11.2)

1.2.5.10.4 Solid Radwaste Control

The solid radwaste system collects, treats, and stores solid radioactive wastes for offsite shipment. Wastes are handled on a batch basis. Radiation levels of the various batches are determined by the operator. (Section 11.4)

1.2.5.10.5 Gaseous Radwaste Control

The gaseous radwaste system is continuously monitored by the main stack radiation monitor and the air ejector offgas radiation monitor. A high-level signal from the air ejector offgas radiation monitoring system will, after an appropriate time delay, automatically isolate the offgas system by closing the isolation valves between the air ejector system and the stack. (Section 11.3)

1.2.5.11 Auxiliary Systems

1.2.5.11.1 Reactor Building Cooling Water System

A closed cooling water system using inhibited demineralized water as the heat transfer medium cools reactor auxiliaries. The system is designed to prevent reactor water contamination and is monitored to detect radioactive leakage into the system. Heat rejection is to the general service water system. (Section 9.2)

1.2.5.11.2 General Service Water System

The general service water system provides water to meet cooling requirements of the reactor building cooling water system and equipment in the turbine building.

The cooling water used is filtered, chlorinated river water that is supplied from the circulating water system and is returned to this system for recycling after being cooled by passage over the cooling towers. (Section 9.2)

1.2.5.11.3 Fire Protection System

The plant fire protection system is designed to supply river water to all areas of the plant and site where the use of water is permissible in extinguishing fires. An electric-driven pump and a diesel-driven pump are provided. In areas where water is unsuitable, chemical fire-fighting equipment is provided. (Section 9.5)

1.2.5.11.4 Heating, Ventilation, and Air Conditioning Systems

Air conditioning systems or fresh air makeup systems supply filtered air to main areas of the plant. The reactor building, turbine building, and radwaste building for the most part receive 100% fresh air makeup. The primary containment air conditioning system uses 100% recirculation during normal operation. The administration building, control room, technical support center, and low-level radwaste processing and storage facility (LLRPSF) use varying amounts of fresh air depending on the season. (Section 9.4)

Normal air flow in the reactor building, turbine building, LLRPSF, and radwaste building is from areas of lesser contamination potential to areas of progressively greater potential before exhaust.

High radiation in any section of the reactor building automatically shuts down the ventilation system. Exhaust is then routed through the standby gas treatment system for processing before release.

A fossil-fueled boiler is provided to supply heating steam for the plant during shutdown of the DAEC. (Section 9.5)

1.2.5.11.5 New- and Spent-Fuel Storage

[redacted] Fuel transfer during refueling is conducted underwater. [redacted]

1.2.5.11.6 Fuel Pool Cooling and Cleanup System

A fuel pool cooling and cleanup system is provided to remove decay heat from spent fuel stored in the fuel pool and to maintain a specified water temperature, purity, clarity, and level. (Section 9.1)

1.2.5.11.7 Instrument, Service and Breathing Air System

Service and instrument air are provided by four motor-driven, nonlubricated compressors, three of which are standby compressors. Each compressor discharges through a separate aftercooler into either or both of two common connected air receivers. Instrument air then passes through an air dryer and filter before entering the instrument air header. Service air is supplied directly from the receivers. Each compressor has adequate capacity to service the plant compressed air requirements.

The breathing air system consists of six-man stations (manifolds), located throughout the power block, supplied by the instrument air system through a cross-tie.

Breathing air for personnel use can also be obtained from the instrument air mains or service air mains, both of which are oil free, wherever necessary throughout the plant. Portable air purifiers are used to meet the Compressed Gas Association Air Specification Grade D requirements. (Section 9.3.1) The instrument air system meets the Grade D requirements and therefore the use of portable filters is not necessary.

1.2.5.11.8 Makeup Water Treatment System

Makeup water is supplied by water from wells on the site. The water is processed through the makeup demineralizers and stored. The storage tank and piping are constructed of materials to prevent metallic contamination of the water. (Section 9.2)

1.2.5.11.9 Potable and Sanitary Water

Water for drinking and sanitary use is supplied from wells on the plant site. The water is filtered and purified as necessary to meet drinking water standards. Shower and lavatory wastewater that does not contain radioactive material is directed to sewage disposal facilities. (Section 9.2)

1.2.5.11.10 Equipment and Floor Drainage Systems

Plant equipment and floor drainage systems handle both radioactive and nonradioactive drains. Radioactive drains contain potentially radioactive materials. Radioactive drains are pumped to the radwaste system for a determination of radioactivity before cleanup and reuse or discharge.

Nonradioactive drains are pumped or drained to the storm drain system. (Section 9.3)

1.2.5.11.11 Process Sampling System

Fluids are sampled continuously or periodically from equipment and systems to determine plant and equipment performance. Samples are taken continuously or obtained as grab samples for laboratory analysis, as required. (Section 9.3)

1.2.5.12 Shielding

The shielding materials generally used are normal and high-density concrete, water, lead, steel, and hydrogenous materials such as masonite. In areas where shielding is not sufficient to permit work, temporary shielding may be used. (Section 12.3)

1.2.5.13 Implementation of Loading Criteria

Loading criteria as affected by site characteristics as well as normal operating loads and postulated accident condition loads are used in the design of all plant buildings, structures and systems, and parts thereof, in accordance with their classification as described below. (Chapter 3)

1.2.5.13.1 Seismic Category I Systems and Components

All Seismic Category I systems and components are designed such that they will function after the design-basis seismic event.

1.2.5.13.2 Seismic Category I Structures

All Seismic Category I structures are designed such that they will function after the design-basis seismic event.

1.2.5.13.3 Nonseismic Structures, Systems, and Components

All Nonseismic items that have no safety considerations and do not degrade the integrity of any Seismic Category I item are designed for normal operating loads and loads normally affected by site characteristics.

If, however a seismic analysis determines that failure of the Category II system, structure, or component will adversely affect the ability of a seismic Category I item to perform its intended function, the Category II system, structure, or component shall be installed to seismic criteria.

1.2.6 PRINCIPAL DESIGN CRITERIA

The principal criteria for design and construction of the DAEC are summarized below.

1.2.6.1 General

1. The plant is designed, fabricated, erected, and will be operated to produce electric power in a safe and reliable manner and, as a minimum, in accordance with applicable codes and regulations.

2. The design of nuclear safety systems and engineered safeguards includes allowances for environmental phenomena at the plant site.

3. Nuclear safety systems and engineered safeguards are designed to permit safe plant operation and to accommodate postulated serious accidents without endangering the health and safety of the public.

1.2.6.2 Reactor Core

1. The nuclear system employs a General Electric BWR to produce steam for direct use in a turbine-generator.

2. The nuclear system, in conjunction with other design parameters, is designed so there is no inherent tendency for sudden divergent oscillation of operating characteristics in any mode of operation.

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

4. Power excursions that could result from any credible reactivity addition accident will not cause damage, either by motion or rupture, to the reactor vessel or impair operation of nuclear safety systems and engineered safeguards.

5. 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 and accidents. For accidents in which one breach in the nuclear system process barrier is postulated, such breach shall not cause additional breaches in the nuclear system process barrier.

6. The reactor core and reactivity control system are designed so that control rod action, with the maximum worth control rod fully withdrawn and unavailable for use, is capable of bringing the core subcritical and maintaining it as such from any power level in the operating cycle.

7. Backup reactivity shutdown capability is provided independent of normal reactivity provisions. This system has the capability, with adequate margin, to shut down the reactor from any operating condition.

8. The fuel rod cladding is designed to contain the fission gas released from the fuel material throughout the design life of the fuel rods.

9. Thermal characteristics of the reactor core are adequate to prevent any fuel cladding integrity safety limits from being exceeded in the design power range and during any abnormal operational transient and to prevent excessive fuel cladding perforations as a result of any design-basis accident.

10. The reactor core and associated systems are designed to accommodate plant transients and maneuvers that might be expected without compromising safety and without fuel damage.

1.2.6.3 Reactor Core Cooling

1. Emergency core cooling systems are provided to prevent the fuel clad temperature from exceeding 2200°F as a result of a LOCA. (Section 6.3)

2. The emergency core cooling systems provide for the continuity of core cooling over the complete range of postulated break sizes in the nuclear system process barrier.

3. The emergency core cooling systems are diverse, reliable, and redundant.

4. The operation of the emergency core cooling systems is initiated automatically when required regardless of the availability of off-site power supplies and the normal generating system of the plant.

5. Heat removal systems are provided to remove heat generated in the reactor core to cover the full range of normal operational conditions from plant shutdown to design power and for any abnormal operational transient. The capacity of such systems is adequate to prevent any fuel cladding integrity safety limits from being exceeded

6. Heat removal systems are provided to remove decay heat generated in the reactor core under circumstances wherein the normal operational heat removal systems become in operative. The capacity of such systems is adequate to prevent any fuel cladding integrity safety limits from being exceeded.

7. Independent means are provided to prevent overpressure conditions that could jeopardize the integrity of the reactor coolant system or the reactor emergency and auxiliary core cooling systems.

1.2.6.4 Plant Containment

1. The primary containment is designed, fabricated, and erected to accommodate, without failure, the pressures and temperatures resulting from or subsequent to the double-ended, or equivalent, failure of any coolant pipe within the primary containment.

2. Provision is made for the removal of energy from within the primary containment and to maintain the integrity of the containment and the containment system as long as necessary following the various postulated design-basis accidents.

3. The reactor building, encompassing the primary containment, provides secondary containment when the primary containment is closed and in service and provides for primary containment when the primary containment is open.

4. Provisions are made for preoperational pressure and leak rate testing of the primary containment system and for leak testing at periodic intervals after the plant has commenced operation. Provision is also made for leak testing selected penetrations and for demonstrating the functional integrity of secondary containment.

5. The integrity of the complete containment system and such other associated engineered safety systems as may be necessary is designed and maintained so that offsite doses resulting from postulated design-basis accidents will be below the values stated in 10 CFR 50.67.

1.2.6.5 Plant Instrumentation and Control

1. The plant is provided with a control room having adequate shielding and air conditioning facilities to permit occupancy for normal plant operation as well as during all DBA situations. In the unlikely event control room must be abandoned because of fire, the plant is provided with an alternate shutdown capability system permitting shutdown of the plant from outside the control room.

2. Interlocks or other protective devices are provided so that procedural controls are not the only means of preventing serious accidents.

3. A reliable reactor protection system, independent from the reactor process control system, is provided to automatically initiate appropriate action whenever plant conditions approach pre-established limits. Periodic testing capability is provided. Sufficient redundancy is provided so that failure or removal from service of any one component or portion of the system will not preclude appropriate actuation of the reactor protection systems when required.

1.2.6.6 Plant Electrical Power

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. The capacity of the power sources is adequate to accomplish all required engineered safety functions under postulated DBA conditions.

1.2.6.7 Plant Radioactive Waste Disposal

1. Gaseous, liquid, and solid waste disposal systems are designed so that the discharge of effluents is in accordance with 10 CFR 20 and other applicable regulations.

2. Process and discharge streams are appropriately monitored and such features are incorporated as may be necessary to maintain releases below the permissible limits of 10 CFR 20.

1.2.6.8 Plant Shielding and Access Control

The radiation shielding in the plant and the plant access control patterns are such that the personnel doses shall be less than the limits of 10 CFR 20.

1.2.6.9 Plant Fuel Handling and Storage

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality and to provide sufficient shielding and cooling for spent fuel.

1.2.7 SUMMARY OF RADIATION EFFECTS

1.2.7.1 Normal Operations

The gaseous and liquid radioactive waste systems are designed such that radioactive releases to the environment will be limited to the as low as is reasonably achievable level.

1.2.7.2 Abnormal Operational Transients

A design objective is to avoid fuel damage as a result of abnormal operational transients. Analyses of these events, which are described in Chapter 15, show that abnormal operational transients do not result in any significant increase of radioactive material release to the environs over that experienced during normal operation.

1.2.7.3 Accidents

The ability of the plant to withstand the consequences of accidents without posing a hazard to the health and safety of the public is evaluated by analyzing a variety of postulated accidents. The calculated consequences of design-basis accidents, which result in the greatest potential offsite radiation exposures, are discussed in Section 15.2. These doses are substantially below the guideline doses given in 10 CFR 50.67.

[redacted]

[redacted]

1.3 COMPARISON TABLES

1.3.1 COMPARISONS WITH SIMILAR FACILITY DESIGNS

This section provides a comparison of major plant features with other BWR facilities at the time that the DAEC operating license was issued.

Tables 1.3-1 through 1.3-5 provide a comparison of the major features of the DAEC with Browns Ferry Units 1 and 2, Monticello Unit 1, Vermont Yankee Unit 1, and Cooper, all of which have been reviewed by the NRC and received construction permits. The DAEC is a GE Model 183T368. **The tables are provided for historical purposes. They have not been updated and do not reflect the current design of the DAEC.**

1.3.2 COMPARISON OF FINAL AND PRELIMINARY INFORMATION

1.3.2.1 Introduction

Section 1.3.2 is provided for historical purposes and has not been updated. In particular, this information does not reflect the Extended Power Uprate Project.

This section lists significant changes from the design of the safety-related structures and equipment of the DAEC as described in the PSAR that took place during the plant design evolution up to the submittal of the initial FSAR.

Changes are listed in sections corresponding to the section numbers of the initial FSAR where the system, component, structure, or analysis is described.

For each item, a justification of the change is provided and appropriate amplifying actions of the initial FSAR are referenced.

1.3.2.2 Auxiliary Transformer Arrangement

The auxiliary transformer arrangement was incorrectly described in PSAR Section 1.5.5.1 as consisting of two auxiliary transformers. Initial FSAR Section 1.5.1.5 correctly describes the single, three-winding transformer.

1.3.2.3 Site and Environs

1.3.2.3.1 Population Data

The population figures presented in initial FSAR Section 2.3, "Population and Land Use," reflect data from the 1970 Census. The population figures presented in the same section of the PSAR were based on 1960 Census data.

1.3.2.3.2 Bedrock Grouting Program I

In bedrock grouting program I, a neat cement grout was occasionally used in place of the sand-cement mixture as stated in PSAR Sections 2.6.3.6.2.3 and G.1, page G.1-5.

A technical review of this change by appropriate specialists indicated that the modification was acceptable. The modification, which is allowed by project specifications, was necessary to meet the field characteristics of certain holes as revealed by the drilling, pressure testing, and grouting operations. This change is covered in Section 2.6.3.6.2.2 of the initial FSAR.

1.3.2.3.3 Bedrock Grouting Program II

In bedrock grouting program II, solution cavities or channels at the bedrock surface were cleaned and filled with a two part sand, one part cement and water mixture to a minimum depth of two times the width of the channel in lieu of lean concrete fill as stated in PSAR Section G.1, page G.1-1.

A technical review of this change by appropriate specialists indicated that the modification was acceptable. The modification was necessary to meet existing field conditions. This change is covered in Section 2.6.3.6.2.2 of the initial FSAR.

1.3.2.3.4 Bedrock Grouting Program III

In bedrock grouting program III, grouting operations commenced at the center holes and worked outward rather than following the procedure in PSAR Section G.1, page G.1-4.

A technical review of this change by appropriate specialists indicated that the adoption of this procedure would have no adverse effects on the results of the grouting program. The procedure in the PSAR has been deleted from the initial FSAR.

1.3.2.4 Reactor

1.3.2.4.1 Gadolinia Core Design

Supplementary reactivity control for the initial core loading at the PSAR stage was in the form of boron-steel curtains. However, the curtains are to be replaced by burnable poison (gadolinia) dispersed in the uranium dioxide fuel.

Gadolinia dispersed in the fuel not only gives better performance, but also avoids the fuel-cycle penalty associated with the residual steel and boron in the curtains. It also eliminates the inconvenience of the curtain removal. This design is described in initial FSAR Section 3.2.

1.3.2.4.2 Control Rod Drive Filter Relocation

The filter assembly in the control rod drive unit was to be mounted near the moving coupling spud according to PSAR design. The FSAR design requires that the filter assembly be mounted on the stationary stop piston.

A moving filter when clogged may affect the control rod insertion time. The stationary filter assembly eliminated this possibility and is, therefore, a more efficient design. This change is described in initial FSAR Section 3.5.

1.3.2.4.3 Control Rod Drive Scram Discharge Volume Increase

The control rod drive scram discharge volume has been increased from 1.1 gal per drive as required at the PSAR stage to 3.34 gal per drive.

The requirement to increase the scram discharge volume to 3.34 gal per drive is derived from BWR operating experience.

1.3.2.4.4 Core Mass Flow Increase

The core mass flow was changed from 48.5 million lb/hr in the PSAR to 50.5 million lb/hr in the initial FSAR.

The PSAR analysis used approximately 7% bypass leakage flow. However, subsequent investigation revealed that approximately 10% bypass leakage flow was necessary. The core mass flow was thus increased and is shown in initial FSAR Section 3.7.

1.3.2.4.5 Vessel Seismic Analysis Model

The analytical model of the pressure vessel and internals used for seismic analysis in PSAR Section 3.3 was modified. There are more "nodal points" in the FSAR model.

The FSAR model is more refined than the PSAR model. Therefore, more representative responses were obtained. This model is described in Section 3.3 of the initial FSAR.

1.3.2.5 Reactor Coolant System

1.3.2.5.1 Main Steam Line Seismic Design

In the amended PSAR, it was stated that the main steam line would be designed to Seismic Category I requirements only up to and including the outboard isolation valve. To provide a decreased probability of failure of these lines, they have been designed to

Seismic Category I requirements up to but not including the turbine stop and control valves as described in initial FSAR Section 4.11.

1.3.2.5.2 RHR/Core Spray Discharge Line Fill System

Subsequent to issuing PSAR Figure 4.8.2, the RHR/core spray pump discharge line fill system was added to maintain these lines sufficiently full of water to prevent water hammer and consequent piping damage. This fill system is shown in initial FSAR Figure 4.8-2 and described in initial FSAR Section 4.8.5.

1.3.2.5.3 Standoff Vessel Insulation

A summary description of the reactor vessel description was given in PSAR Section 4.2.4.9. The significant change in the application of vessel insulation from this description to the final design is to improve access to the outside surface of the vessel and its nozzles for inspection throughout the life of the vessel. The cylindrical portion of the vessel insulation from the juncture of the vessel skirt to the top of the biological shield wall will be installed with a gap or space between the inside of the insulation and the vessel wall. This space is for inspection access to the vessel surface. Removable panels will be provided at appropriate locations on this portion of the insulation. Special design of the insulation at all of the vessel nozzles will allow access for inspection and reassembly at the nozzle areas.

This change was implemented to be responsive to Section XI of the ASME B&PV Code "Rules for In-Service Inspection of Nuclear Reactor Coolant Systems, January 1, 1970," issued for use since the submittal of the DAEC PSAR.

1.3.2.5.4 Leak Detection System

The following changes and additions have been made to the leak detection system:

1. Differential temperature sensors have been added to the reactor water cleanup system. The high differential temperature signal is used to provide an alarm and initiate isolation.

2. Flow elements and other necessary instrumentation have been added to the reactor water cleanup system to provide a differential flow signal used to provide an alarm and initiate isolation.

3. A time delay was added to the high differential flow isolation signal to allow for the startup of the system.

4. Differential temperature sensors were added to the RHR equipment area inlet and outlet ventilation ducts to provide a high differential temperature alarm.

5. Differential temperature sensors were added to the main steam line tunnel to provide an alarm.

6. Differential temperature sensors were added to the HPCI equipment room and the ambient temperature sensors were located near the emergency area cooler. High differential and high ambient temperature signals provide alarms and initiate the isolation of the HPCI steam supply isolation valves.

7. Differential temperature sensors were added to the RCIC equipment room and the ambient temperature sensors were located near the emergency area cooler. High differential and high ambient temperature signals provide alarms and initiate the isolation of the RCIC steam supply isolation valves.

8. Differential temperature sensors were added to the RCIC and HPCI steam line area near the suppression pool. The high differential temperature signal provides alarm and initiates the isolation of the RCIC and HPCI steam supply isolation valves after a time delay. This delay is included to allow time for the operator to take action to determine the source of the high temperature and isolate only the source.

9. A three-channel drywell atmosphere radioactivity detector was added to provide a sensitive and rapid indication of increased nuclear system leakage.

The changes in the leak detection system have been made to improve the capability of the system to detect leakage from the nuclear system process barrier. The addition of differential temperature and flow-sensing instrumentation increases the sensitivity of the leak detection and thereby acts to prevent excessive loss of reactor coolant and the release of significant amounts of radioactive material. The drywell atmosphere radioactivity monitor provides greater sensitivity of leak detection within the primary containment. These changes are discussed in Sections 4.10 and 7.3 of the initial FSAR.

1.3.2.5.5 Automatic Depressurization System Permissive Signal

The automatic depressurization system (ADS) permissive signal as described in PSAR Section 4.4.5 was based on simultaneous signals from high drywell pressure, low reactor water level, and HPCI system low flow. Further evaluation determined that there were postulated instances when breaks in the HPCI system downstream of the sensing system would not be detected and the automatic depressurization system would be blocked out, thinking that HPCI was still an integral system. This is an unacceptable

condition, and the ADS permissive is based on simultaneous signals from high drywell pressure and low reactor water level. In addition, to ensure that adequate cooling water flow is available, the ADS permissive is also based on a discharge pressure permissive in any low-pressure cooling system (LPCI or core spray) after a 2 min delay. This delay provides time for the operator to cancel the ADS signal if control room information indicates the signal is false or is not needed. This system is described in initial FSAR Sections 4.4.5 and 6.4.2.

1.3.2.6 Containment

1.3.2.6.1 BWR Equipment Environmental Requirements

The environmental requirements within the primary containment under which BWR essential components and safety systems are designed to operate were changed. These requirements are delineated in Table 1 of Item 7, Information Guide 2, initial FSAR Appendix M.

Changes were made in the environmental requirements within the primary containment, such as temperature, pressure, time duration at temperature, and pressure, under which BWR essential components and safety systems are required to operate. These changes result from BWR operating experience, including the experiences at Dresden 2 in June 1970.

1.3.2.6.2 Primary Containment Design Parameters

Primary containment design parameters used to generate primary containment accident response analysis in the PSAR were changed, resulting in a reduction in the drywell net free volume and other minor changes to Table 5.2-1 of the PSAR.

Justification consists of a revised accident response analysis. Refer to Section 14.6, Section 0.14.1, and Appendix P of the initial FSAR for the incorporation of these changes.

1.3.2.6.3 Electrical Penetrations

The 12-in.-diameter electrical penetration sleeves in the primary containment described in PSAR Section 5.2.3.4.3 have been changed to 10-in. in diameter because of a change in vendor standard penetration size. This new diameter is not specifically stated in the initial FSAR, but the penetrations are described in Section 5.2.3.4.3 and shown in Figure 5.2-7 of the initial FSAR.

1.3.2.7 Emergency Core Cooling System.

The basic ECCS analytical model in PSAR Chapter 6 was modified. The PSAR model is termed to be a "level swell and interim" model while the FSAR model is termed

to be a "level swell and critical heat flux" model. The old and new models are described in GE topical report NEDO-10329.

The major difference between the PSAR model and FSAR model is attributed to GE's improvements in analytical techniques and to the incorporation of the most up-to-date test data. Therefore, the FSAR model is an improvement of the earlier PSAR model. However, it should be noted that ECCS analysis and performance must conform to AEC's interim acceptance criteria dated June 19, 1971, regardless of the said change. This conformance is demonstrated in Section 6.7 of the initial FSAR.

1.3.2.8 Control and Instrumentation

1.3.2.8.1 Runback of Recirculation Pump on Feedwater Pump Trip

The recirculation logic has been changed to run back the recirculation pumps to approximately 45% flow when a feedwater pump is tripped.

The automatic runback of the recirculation pumps on a feedwater pump trip results in a reactor power reduction. The correction for the loss of one feedwater pump does not allow vessel level to recover fast enough and a reactor scram occurs when level reaches the Level 3 trip point.

1.3.2.8.2 High Reactor Level Isolation

A high-level reactor signal was added to the main steam line isolation logic as an initiating function.

The high-level reactor initiating signal was added to the main steam isolation logic to prevent a rapid depressurization of the reactor vessel during startup or hot standby on the malfunction of the pressure regulator for the main turbine bypass valves. A rapid depressurization of the reactor vessel could cause the decrease of the nuclear steam saturation temperature that would exceed the design rate of the change of the vessel temperature. A depressurization rapid enough to cause the vessel temperature to exceed the design rate of change would cause the water within the reactor to swell and trip the high-level reactor isolation signal. See initial FSAR Section 7.3.4.7.

1.3.2.8.3 Low Reactor Water Level Confirmed for Automatic Depressurization System

A second low reactor water level permissive to the ADS logic was added. This signal is from separate instrumentation and is at a higher water level than the first reactor vessel low water level permissive that has always been in the ADS logic.

The second reactor vessel low water level permissive was added to confirm that the water level in the vessel is low to provide protection against inadvertent depressurization should an instrument line fail. See initial FSAR Section 7.4.3.3.2.

1.3.2.8.4 Gland Seal Condenser Isolation

The PSAR indicated that the gland seal condenser was to be isolated on the steam line high radiation signal following a control rod drop accident. This isolation is deleted from the FSAR design.

If the gland seal condenser is not isolated during a control rod drop accident, the thyroid dose will be increased by 0.09 rem (design-basis accident dose in Table 14.9-1 of the initial FSAR is 11.1 rem and the whole-body dose will be increased by 0.0187 rem (design-basis accident dose in Table 14.9-1 of the initial FSAR is 0.317 rem). This increase in dose is clearly negligible even using AEC assumptions.

1.3.2.8.5 Containment Spray - RHR Interlock

The logic for the containment spray valves has been changed to include a permissive when the drywell pressure is not low (above 2 psig). The permissive logic is one-out-of-two-twice.

It is possible, though highly improbable, that conditions as follows may be present in the containment:

1. Containment atmosphere temperature near maximum allowable.

2. Containment humidity near maximum allowable.

3. Containment pressure less than 1 psig.

4. Suppression pool temperature relatively low.

With the above conditions, if the containment spray is initiated, it is possible to reduce the containment pressure to a negative pressure. The high drywell permissive has been added to the containment spray logic to preclude the initiation of containment spray when containment pressure is low. The permissive pressure setpoint is set high enough to prevent the operation of containment spray during conditions when it is possible to reduce the containment pressure excessively, but low enough to have containment spray available when conditions are such to make containment spray desirable.

1.3.2.8.6 Average Power Range Monitor 15 Percent Power Scram

In the startup mode, an extra average power range monitor (APRM) upscale trip has been added with the setting fixed at 15% of rated power as described in Section 7.5.7 of the initial FSAR.

This APRM trip is designed for low-power no-flow operation and to provide minimum critical heat flux rate protection at low flow. The inclusion of this trip system gives more accurate monitoring without the necessity of repeated calibration that is required by the existing intermediate range monitor trip system.

1.3.2.8.7 Nuclear Steam Supply System Electrical Equipment Separation

Electrical separation changes to the nuclear steam supply systems have been made as follows:

1. The logic for the cleanup system isolation valves was changed to close only the outboard isolation valve on high temperature nonregenerative heat exchange outlet or the start of standby liquid control system.

2. Temperature sensors and other necessary instrumentation have been added to provide separation to the isolation signals for reactor water cleanup system high ambient and high differential temperature. Two separate channels of instrumentation have been provided. One channel provides isolation initiation for only one isolation valve. The other channel provides isolation initiation for the other isolation valve.

3. A flow-sensing device on the HPCI steam supply line has been added to provide a total of two flow devices in the line. The redundant instruments for pressure and differential pressure (flow) previously connected to the one flow device have been equally divided between the flow devices.

4. A flow-sensing device on the RCIC steam supply line has been added to provide a total of two flow devices in the line. The redundant instruments for pressure and differential pressure (flow) previously connected to the one flow device have been equally divided between the flow devices.

5. A remote manual switch has been added to the inboard and outboard isolation valves for the RHR process sampling line so each valve will have a separate remote manual switch instead of using one switch for both valves.

6. A remote manual switch has been added to the inboard and outboard isolation valves for the pump discharge line from the drywell floor drain and drywell equipment drain sump pumps in the radwaste system. This

will provide a separate remote manual switch for each isolation valve instead of using one switch for both isolation valves in each line.

7. An isolation signal reset push button has been added for the HPCI system logic B. This will provide separate isolation reset push buttons for logic A and logic B.

8. An isolation signal reset push button for the RCIC system logic B has been added. This will provide separate isolation reset push buttons for logic A and logic B.

9. The isolation logic for the RHR shutdown cooling pump suction has been changed to separate the initiating signals to the inboard and outboard valves. The instrumentation used to initiate the isolation of the inboard isolation valves is separate from that used for the outboard isolation valves.

10. The isolation logic for the RHR LPCI inboard isolation valves has been changed to separate the isolation signal to each of the valves.

11. The isolation initiation logic from reactor low level or high drywell pressure for the radwaste, reactor water cleanup, and the traversing incore probe systems has been changed to provide separate signals for the inboard and outboard isolation valves.

12. The isolation initiation logic for reactor water sample and main steam line drain isolation valves has been changed to provide separate signals for the inboard and outboard isolation valves.

13. Dual sensing lines to each main steam line flow element have been added. The connection of the redundant instruments for differential pressure (flow) has been changed; these instruments were previously connected to the one set of sensing lines to divide them equally between the dual lines.

The changes in the electrical separation have been made to update the design to comply with the requirements of IEEE 308-1971.

1.3.2.9 Electrical Power Systems

1.3.2.9.1 Auxiliary Power System

The design of the auxiliary power system as described in PSAR Section 8.3.3 and Figure 8.3-1 has been revised to delete the ties between normal and essential buses. The number of essential 480-V load centers was also changed from three to two, and the tie breakers were deleted.

The resultant design is simpler and more reliable and is described in initial FSAR Section 8.3.5 and Figure 8.3-1.

1.3.2.9.2 250-V Battery

A 250-V battery was added to the dc power supply and distribution system described in PSAR Section 8.5.3. The increased size and quantity of battery loads made the addition of this battery necessary.

This system is described in Section 8.5.3 of the initial FSAR and shown in Figure 8.5-2.

1.3.2.9.3 Power Cable Ratings

The power cables have been selected with the ratings satisfying Insulated Power Cable Engineers Association (IPCEA) ampacity values as a minimum requirement rather than National Electric Code (NEC) requirements as stated in PSAR Section 8.3.4.1.

This change was made because IPCEA ratings are more realistic, especially for cables in trays that predominate in power plants. This is described in initial FSAR Section 8.3.6.1.

1.3.2.9.4 Cable Tray Design

The cable tray cross-member spacing and loading described in PSAR Section 7.8 were changed from 4 to 6 in. and 50 lb/ft + 200 lb concentrated midspan load to 9 in. and 60 lb/ft + 200 lb concentrated midspan load.

This change was due to the change from a National Electrical Manufacturers Association (NEMA) II to NEMA III design tray system and is described in initial FSAR Section 8.3.6.1.

1.3.2.9.5 Automatic Transfer Switches

The automatic transfer switches in the dc power supply and distribution system described in PSAR Section 8.5.3 and shown in Figure 8.5.1 have been deleted. This deletion was necessary to achieve redundant, isolated systems in accordance with IEEE-308.

The present system is described in initial FSAR Section 8.5.3 and shown in Figures 8.5-1 and 8.5-2.

1.3.2.9.6 Cable Markings

[redacted]

1.3.2.9.7 Uninterruptible AC Power Supply

The power supply for uninterruptible AC power is a static inverter system as described in UFSAR Section 8.3.1.

1.3.2.10 Radioactive Waste Systems

1.3.2.10.1 Revised Source Terms

The radionuclide sources as used in the PSAR have been revised. This revision is reflected in all related initial FSAR sections, such as Chapter 9 and 14.

A more comprehensive and up-to-date list of radionuclide source terms has been assembled to incorporate the results of extensive measurements at operating BWRs over the past several years.

1.3.2.10.2 Recombiner/Charcoal Offgas System

A catalytic recombiner/ambient charcoal adsorption system, described in initial FSAR Section 9.4, was added to the gaseous radwaste system.

A catalytic recombiner/ambient charcoal adsorption system was added to the DAEC to supplement the previous 30-min holdup design to provide a gaseous radwaste system that limits offsite doses from routine plant releases to the lowest practicable level. This system is described in Section 9.4 of the initial FSAR.

1.3.2.10.3 Liquid Radwaste System

Significant items added to the liquid radwaste system were a floor drain demineralizer and an evaporator. See Section 9.2 (initial FSAR) for the liquid radwaste system description.

The floor drain demineralizer was added to the liquid radwaste system to provide the capability of recycling liquid wastes that do not meet plant water-quality requirements on the basis of activity and/or conductivity. This liquid effluent can then be returned to condensate storage for reuse to the extent practicable.

An evaporator was added to the liquid radwaste system to provide treatment capability for those waste liquids whose chemical quality is such that demineralization is not possible.

The addition of floor drain demineralization and chemical waste evaporation, along with other less significant changes to the liquid radwaste system, has been effected to ensure that levels of radioactive materials in liquid effluents released from the DAEC will be maintained at the lowest practicable level.

1.3.2.11 Auxiliary Systems

1.3.2.11.1 Makeup Water Treatment

The makeup water treatment system, as described in PSAR Section 10.12 and Figure 10.12.1, has been changed from a carboxyl resin/primary cation/degasifier/primary anion resin/mixed-bed resin system to a cation (weak and strong acid)/decarbonator/strong-base system. The new design better suits the well water being used for makeup. This system is described in initial FSAR Section 10.13.3.

1.3.2.11.2 General Service Water System

The general service water system and turbine building cooling water system, as described in PSAR Sections 10.6.1 and 10.9.1 and Figure 10.6.1, have been changed so they no longer supply makeup water to the circulating water system.

To provide a more direct and reliable source of makeup water, makeup to the pump house sumps is presently obtained directly from the Cedar River by the river water supply system. Makeup is then supplied to the circulating water system and general service water system as described in the initial FSAR Section 10.6.5 and is supplemented by the well water system as described in initial FSAR Section 10.10.

1.3.2.11.3 RHR Service Water System

The four full-size RHR service water pumps described in PSAR Section 10.7 have been changed to four half-size pumps as described in initial FSAR Section 10.7.5. This change corrects an error in the PSAR and is in accordance with the original GE design.

1.3.2.11.4 Emergency Lighting System

The emergency lighting system described in PSAR Section 10.18.3 has been changed from full dc distribution to partial dc distribution with the remainder provided by fluorescent fixtures with self-contained batteries, chargers, and inverters.

This change was made to provide a more reliable emergency lighting system and is described in initial FSAR Section 10.19.3.

1.3.2.11.5 Spent-Fuel Racks

The spent-fuel rack design was changed by stiffening the configuration, thereby increasing its natural frequency.

Additional seismic analyses, conducted after the initial design configuration of the spent-fuel rack was established, indicated that in certain cases of partially loaded spent-fuel racks the structural integrity of these racks could be violated. Subsequent to these analyses, the racks were redesigned to increase their natural frequency so that the racks or the fuel assemblies stored within them will not be damaged under forces resulting from the design-basis earthquake.

1.3.2.12 Plant Structures and Shielding

1.3.2.12.1 Reinforcing Steel

Reinforcing steel from several heats that did not meet the yield strength requirements of ASTM A615-68, Grade 60, during testing was installed in portions of the reactor building foundation (see PSAR Section 12.4.8.7). Refer to the response to Safety Guide 15, initial FSAR Appendix G, for a more complete discussion of this item.

1.3.2.12.2 Emergency Diesel-Generator Layout

The emergency diesel-generator layout concept, as discussed in PSAR Section 12.2.6, was revised. [redacted]

[redacted] This is discussed in Section 12.2.2 of the initial FSAR.

1.3.2.12.3 Control Facilities Layout

The control facilities layout concept, as discussed in PSAR Section 12.2.5, was revised. [redacted]

[redacted]

1.3.2.12.4 Pump House Layout

[redacted]

[redacted]

1.3.2.12.5 Curing of Structural Concrete

The curing of structural concrete did not conform to the requirements of ACI 301-66, Paragraphs 1405 (a) and (c), referenced in PSAR Section 12.4.8.2.

A technical review of project curing requirements by appropriate specialists indicated that the procedures being followed with respect to form removal and time of curing were adequate to ensure the production of concrete meeting applicable standards. The requirements of American Concrete Institute (ACI) 301-66 have been deleted from the PSAR as indicated in Section 12.5.2 of the initial FSAR.

1.3.2.12.6 Radwaste Building Seismic Design

In PSAR Section 12.2.5, it was stated that the radwaste building would be a Seismic Category I Structure. An analysis of radioactive liquids that could be released to the environment as a result of postulated natural phenomena indicated that the annual limits of 10 CFR 20 would not be exceeded in the Cedar River, thereby justifying the Nonseismic design. For a discussion of this analysis, refer to Appendix N, Section N.9.3 of the initial FSAR.

1.3.2.12.7 Vertical Construction Joints

Vertical construction joints were not slushed with a coat of neat cement grout immediately before the placing of new concrete as required by ACI 318-63, Section 704 (a), referenced n PSAR Section 12.4.8.2.

A technical review of project requirements in this area by appropriate specialists resulted in the recommendation that a layer of grout not be used on vertical construction joints before the placement of new concrete. The placing conditions that necessitate the use of the grout layer on horizontal joints are not present on vertical joints, and the omission of the grout layer on vertical joints does not have adverse effects on the concrete. This item pertains to Section 12.5.2 of the initial FSAR.

1.3.2.12.8 Keyways

Keyways were omitted in vertical and horizontal construction joints rather than providing keys in certain joints as required by ACI 301-66, referenced in PSAR Section 12.4.8.2.

Justification for this is recommended by appropriate specialists to omit keys in construction joints, in conjunction with proper preparation of joint surfaces. The requirements of ACI 301-66 have been deleted from the PSAR as indicated in Section 12.5.2 of the initial FSAR.

1.3.2.13 Containment Response Analysis

The containment response analytical model in PSAR Section 14.6 was modified. The initial FSAR model and the analytical procedure are discussed in detail in GE Topical Report NEDO-10320.

The containment pressure response model and analytical procedure incorporated in the FSAR were implemented to provide a more conservative prediction of containment pressure following a postulated design-basis LOCA.

1.3.2.14 Pressure Integrity of Piping and Equipment Pressure Parts

Appendix A of the PSAR has been extensively rewritten to include changes in areas of code requirements; design and system classification requirements; materials and brittle fracture control; and fabrication, assembly, and erection.

Several areas of PSAR Appendix A have been rewritten and new tables on code requirements and systems classification have been included. The new table on codes is included to clarify which codes and their respective issue dates apply to various classifications of equipment as a function of the purchase order dates for the various components. The new classification tables and definitions are included to update this area of the appendix with recent BWR classification agreements between AEC, GE, architect-engineers, and various utility companies. In addition, changes are made to other Appendix A areas such as design requirements, materials and brittle fracture control, and fabrication to describe more specific and later GE requirements, latest applicable code requirements, and latest applicable code cases. Finally, this appendix is now written to

apply to all nuclear energy systems components depicted in Figures A.2-1 and A.2-2 and Tables A.2-2 and A.2-3 as opposed to the original PSAR Appendix A, which generally addressed the GE-supplied nuclear steam supply systems.

COMPARISON OF NUCLEAR SYSTEM DESIGN CHARACTERISTICS[a]

Parameter	[redacted]	[redacted]	DAEC[b]	[redacted]	[redacted]
Thermal and Hydraulic Design					
Reference design thermal output, MWt	1593	1464	1593	2381	3293
Maximum anticipated thermal output, MWt	1665	1670	1658	2486	3440
Steam flow rate, lb/hr	6.43×10^6	6.77×10^6	6.843×10^6	9.56×10^6	13.38×10^6
Core coolant flow rate, lb/hr	48.0×10^6	57.6×10^6	50.5×10^6	73.5×10^6	102.5×10^6
Feedwater flow rate, lb/hr	6.43×10^6	6.75×10^6	6.822×10^6	9.52×10^6	13.38×10^6
Feedwater temperature, °F	372	376.3	420	367	376.1
System pressure, nominal in steam dome, psia	1020	1040	1020	1020	1020
Average power density, kW/liter	50.8	40.6	51.0	51.1	50.7
Maximum thermal output	18.37	17.5	18.5	18.5	18.4
Average thermal output, kW/ft	7.1	5.7	7.067	7.094	7.049
Maximum heat flux, Btu/hr-ft^2	426,210	405,000	427,600	428,300	428,400
Average heat flux, Btu/hr-ft^2	163,900	131,346	163,600	164,220	163,250
Maximum UO_2 temperature, °F	4380	4450	4430	4380	4430
Average volumetric fuel temperature, °F	1210	900	1210	1100	1210
Average fuel-rod surface temperature, °F	558	558	560	558	560
Minimum critical heat flux ratio	≥1.9	≥1.9	≥1.9	≥1.9	≥1.9
Core maximum exit voids within assemblies, %	79	76	76	75	79

[a] Parameters related to reference design thermal output for a single unit unless otherwise noted.
[b] Historical Information. Not updated for Extended Power Uprate.

COMPARISON OF NUCLEAR SYSTEM DESIGN CHARACTERISTICS[a]

Parameter	[redacted]	[redacted]	DAEC[b]	[redacted]	[redacted]
Thermal and Hydraulic Design (continued)					
Core average exit quality, % steam	13.6	12.0	14.3	12.9	13.2
Design power peaking factor					
Transverse peaking factor	1.410	1.47	1.405	1.4	1.4
Local peaking factor	1.24	1.24	1.24	1.24	1.24
Axial peaking factor	1.5	1.57	1.5	1.4	1.4
Total peaking factor	2.6	3.08	2.6	2.43	2.43
Nuclear Design (First Core)					
Water: U_2 volume ratio (cold)	2.41	2.42 (undished)	2.41	2.41	Type 1: 2.43 Types II & III: 2.53
Reactivity with strongest control rod out, k_{eff}	<0.99	<0.99	<0.99	<0.99	<0.99
Moderator temperature coefficient at 68°F, Δ K/K-°F water	-5.0×10^{-5}	-8.0×10^{-5}	-3.5×10^{-5}	-3.5×10^{-5}	-3.5×10^{-5}
Moderator void coefficient, hot, no voids, Δ K/K-% void	-1.0×10^{-3}	-1.0×10^{-3}	-1.05×10^{-3}	-0.87×10^{-3}	-0.87×10^{-3}
At rated output, Δ K/K-% void	-1.5×10^{-3}	-1.4×10^{-3}	-1.05×10^{-3}	-1.05×10^{-3}	-1.05×10^{-3}
Fuel temperature doppler coefficient at 68°F, Δ K/K-°F fuel	-1.3×10^{-5}	-1.2×10^{-5}	-0.94×10^{-5}	-0.947×10^{-5}	-0.9×10^{-5}
Hot, no voids, Δ K/K-°F fuel	-1.2×10^{-5}	-1.2×10^{-5}	-0.98×10^{-5}	-0.977×10^{-5}	-1.0×10^{-5}
At rated output, Δ K/K-°F fuel	$\leq (-1.3 \times 10^{-5})$	$\leq (-1.3 \times 10^{-5})$	$\leq (-1.3 \times 10^{-5})$	$\leq (-1.3 \times 10^{-5})$	$\leq (-1.3 \times 10^{-5})$
Initial average U-235 enrichment, wt %	2.50	2.25	1.9	2.17	2.19
Fuel average discharge exposure MWd/ton	19,000	19,000	14,400	19,000	19,000

[a] Parameters related to reference design thermal output for a single unit unless otherwise noted.
[b] Historical Information. Not updated for Extended Power Uprate.

COMPARISON OF NUCLEAR SYSTEM DESIGN CHARACTERISTICS[a]

Parameter	[redacted]	[redacted]	DAEC[b]	[redacted]	[redacted]
		Nuclear Design (First Core) (Continued)			
Fuel pellets					
Material	Uranium dioxide	Uranium dioxide	Uranium dioxide	Uranium dioxide	Uranium dioxide
Density, % of theoretical	93	93	93	93	93
Diameter, in.	0.487	0.487	0.477	Type I: 0.487, Types II &III: 0.477	0.477
Length, in.	0.75	0.75	0.5	Type I: 0.75, Types II &III: 0.5	0.5
Fuel channel					
Overall dimension, in.					
Length	166.875	166.875	166.875	166.875	166.875
Thickness	0.080	0.080	0.080	0.080	0.080
Cross-section dimensions (outside), in.	5.438 x 5.438	5.438 x 5.438	5.438 x 5.438	5.438 x 5.438	5.438 x 5.438
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4
Core assembly					
Fuel weight as UO_2, lb	179,370	238,270	173,500	261,940	361,837
Zirconium weight (Z-2 + Z-4 + Spacers), lb	~63,300	~80,993	~63,300	~94,300	~140,397
Core diameter (equivalent), in.	129.9	149	129.9	158.5	187.1

[a] Parameters related to reference design thermal output for a single unit unless otherwise noted.
[b] Historical Information. Not updated for Extended Power Uprate.

COMPARISON OF NUCLEAR SYSTEM DESIGN CHARACTERISTICS[a]

Parameter	[redacted]	[redacted]	DAEC[b]	[redacted]	[redacted]
			Nuclear Design (First Core) (Continued)		
Core diameter (active fuel), in.	144	144	144	Type I: 144 Types II &III: 146	144
			Core Mechanical Design		
Fuel assembly					
Number of fuel assemblies	368	484	368	548	764
Fuel-rod array	7 x 7	7 x 7	7 x 7	7 x 7	7 x 7
Overall dimensions, in.	175.88	174.88	175.88	175.88	175.88
Weight of UO_2 per assembly, lb	Undished: 487.4	Undished: 492.5 Dished: 481.7	Type I: 469.2 Type II: 468.7 Type III: 468	Type I: 490.4 Type II: 474.4 Type III: 474.1	Type I: 490.9 Type II: 468.9 Type III: 468.8
Weight of fuel assembly, lb	Undished: 682	Undished: 678.9 Dished: 668	675	681.3	Type I: 682.49 Type II & III: 674.75
Fuel rods					
Number per fuelassembly	49	49	49	49	49
Outside diameter, in.	0.563	0.563	0.563	0.563	0.563
Clad thickness, in.	0.032	0.032	0.037	Type I: 0.032, Types II &III: 0.037	Type I: 0.032 Types II & III: 0.037
Pellet-to-clad gap, in.	0.005	0.005	0.006	0.006	0.005

[a] Parameters related to reference design thermal output for a single unit unless otherwise noted.
[b] Historical Information. Not updated for Extended Power Uprate.

COMPARISON OF NUCLEAR SYSTEM DESIGN CHARACTERISTICS[a]

Parameter	[redacted]	[redacted]	DAEC[b]	[redacted]	[redacted]
Core Mechanical Design (Continued)					
Length of gas plenum, in.	16	11.24	16	Type I: 16 Types II &III: 14	16
Clad material	Zircaloy - 2	Zircaloy - 2	Zircaloy - 2	Zircaloy - 2	Zircaloy - 2
Cladding process	Free-standing, loaded tubes	Free-standing, loaded tubes	Free-standing, loaded tubes	Free-standing, loaded tubes	Free-standing, loaded tubes
Reactor control system					
Method of variation of nuclear power	Movable control rods and variable coolant pumping	Movable control rods and variable coolant pumping	Movable control rods and variable coolant pumping	Movable control rods and variable coolant pumping	Movable control rods and variable coolant pumping
Number of movable control rods	89	121	89	137	185
Shape of movable control rods	Cruciform	Cruciform	Cruciform	Cruciform	Cruciform
Pitch of movable control rods, in.	12.0	12.0	12.0	12.0	12.0
Control material in movable rods	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)
Type of control-rod drives	Bottom-entry, locking piston	Bottom-entry, locking piston	Bottom-entry, locking piston	Bottom-entry, locking piston	Bottom-entry, locking piston
Number of temporary control curtains	156	216	None	None	None
Curtain material	Boron SS, flat	Boron SS, flat	- -	- -	- -

[a] Parameters related to reference design thermal output for a single unit unless otherwise noted.
[b] Historical Information. Not updated for Extended Power Uprate.

COMPARISON OF NUCLEAR SYSTEM DESIGN CHARACTERISTICS[a]

Parameter	[redacted]	[redacted]	DAEC[b]	[redacted]	[redacted]
		Core Mechanical Design (Continued)			
Incore neutron instrumentation					
Number of incore neutron detectors	80	96	80	124	172
Number of incore detector strings	20	24	20	31	43
Number of detectors per string	4	4	4	4	4
Number of flux mapping neutron detectors	3	3	3	4	5
Range (and number) of detectors					
Source-range monitor	Source to 0.01% power (4)	Source to 10% power (4)	Source to 0.001% power (4)	Source to 10% power (4)	Source to 10% power (4)
Intermediate-range monitor	0.0001% to 10% power (6)	1% to 10% power (8)	0.0001% to 10% power (6)	1% to10% power (8)	1% to 10% power (8)
Local-power-range monitor	5% to 125% power (80)	5% to 125% power (96)	5% to 125% power (80)	5% to 125% power (124)	5% to 125% power (172)
Average-power-range monitor	5% to 125% power (6)	5% to 125% power (6)	2.5% to 125% power (6)	5% to 125% power (6)	5% to 125% power (6)
Number and type of incore neutron sources	8 Sb-Be	5 Sb-Be	4 Sb-Be	5 Sb-Be	7 Sb-Be
		Reactor Vessel Design			
Material	Carbon steel clad stainless steel (ASME SA-336 & SA-302B)				
Design pressure, psia	1265	1265	1265	1265	1265

[a] Parameters related to reference design thermal output for a single unit unless otherwise noted.
[b] Historical Information. Not updated for Extended Power Uprate.

COMPARISON OF NUCLEAR SYSTEM DESIGN CHARACTERISTICS[a]

Parameter	[redacted]	[redacted]	DAEC[b]	[redacted]	[redacted]
			Reactor Vessel Design (Continued)		
Design temperature, °F	575	575	575	575	575
Inside diameter, ft-in.	17-1	17-2	15-3	18-2	20-11
Inside height, ft-in.	63-1.5	63-2	66-5	69-4	72-0
Side thickness (including clad)	5.187	5.187	4.593	5.531	6.313
Minimum clad thickness, in.	1/8	1/8	1/8	1/8	1/8
			Reactor Coolant Recirculation System Design		
Number of recirculation loops	2	2	2	2	2
Design pressure					
Inlet leg, psig	1175	1148	1150	1148	1148
Outlet leg, psig	1274	1248	1325	1274	1326
Design temperature, °F	562	562	562	562	562
Pipe diameter, in.	28	28	22	28	28
Pipe material	304/316	304	304/316	304/316	304/316
Recirculation pump flow rate	32,500	32,500	27,100	45,200	45,200
Number of jet pumps in reactor	20	20	16	20	20
			Main Steam Lines		
Number of steam lines	4	4	4	4	4
Design pressure, psig	1146	1146	1146	1146	1146
Design temperature, °F	563	563	563	563	563

[a] Parameters related to reference design thermal output for a single unit unless otherwise noted.
[b] Historical Information. Not updated for Extended Power Uprate.

COMPARISON OF NUCLEAR SYSTEM DESIGN CHARACTERISTICS[a]

Parameter	[redacted]	[redacted]	DAEC[b]	[redacted]	[redacted]
Main Steam Lines (Continued)					
Pipe diameter, in.	20	18	20	24	26
Pipe material	Carbon steel (ASTM A115 KC70 or ASTM A106 Grade B)				
Emergency Core Cooling System[c]					
Core spray steam					
Number of loops	2	2	2	2	2
Flow rate, gpm	3000 at 136 psid	3020 at 307 psid	3120 at 113 psid	4500 at 115 psid	6250 at 122 psid
High-pressure coolant injection system					
Number	1	1	1	1	1
Number of loops	1	1	1	1	1
Flow rate, gpm	4250	3000	3000	4250	5000
Automatice depressurization system					
Number	1	1	1	1	1
Low-pressure coolant injection system					
Number	1	1	1	1	1
Number of pumps	4	4	4	4	4
Pump flow rate, gpm per pump	7,200 at 20 psid	4,000 at 20 psid	4,800 at 20 psid	7,700 at 20 psid	10,000 at 20 psid
Residual heat removal system					
Reator shutdown cooling pumps					
Number	4	4	4	4	4

[a] Parameters related to reference design thermal output for a single unit unless otherwise noted.
[b] Historical Information. Not updated for Extended for Power Uprate.
[c] Sized on maximum anticipated thermal output.

COMPARISON OF NUCLEAR SYSTEM DESIGN CHARACTERISTICS[a]

Parameter	[redacted]	[redacted]	DAEC[b]	[redacted]	[redacted]
Emergency Core Cooling Systems[c] (Continued)					
Flow rate (gpm per pump)[d]	7200	3600	4800	7700	10,000
Heat exchangers					
Number	2	2	2	2	4
Capacity (Btu/hr per heat exchanger)[e]	57.5×10^6	57.5×10^6	35×10^6	70×10^6	70×10^6
Primary containment cooling pumps					
Flow rate, gpm	28,000	16,000	19,200	30,800	40,000
Service water system					
Flow rate, gpm per pump	2700	3500	2500	4000	4500
Number of pumps	4	4	4	4	4
Reactor core isolation cooling system					
Flow rate, gpm	400	400	416	416	616
Fuel pool cooling and cleanup system					
Capacity, Btu/hr	2.37×10^6	2.87×10^6	2.37×10^6	3.4×10^6	8.8×10^6

[a] Parameters related to reference design thermal output for a single unit unless otherwise noted.
[b] Historical Information. Not updated for Extended Power Uprate.
[c] Sized on maximum anticipated thermal output.
[d] Capacity during reactor-flooding mode with three of four pumps operating.
[e] Capacity during postaccident cooling mode with 165°F shell-side inlet temperature, maximum service water temperature, and one residual heat removal pump and two service water pumps in operation.

Table 1.3-2

COMPARISON OF POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS

Parameter	[redacted]	[redacted]	DAEC[a]	[redacted]	[redacted]
Turbine Generator					
Design power, MWt	1665	1670	1658	2487	3440
Design power, MWe	564	545	589	836	1152
Generator speed, rpm	1800	1800	1800	1800	1800
Design steam flow, lb/hr	6.423×10^6	6.77×10^6	6.835×10^6	10.049×10^6	14.049×10^6
Turbine inlet pressure, psig	950	950	950	970	965
Turbine Bypass System					
Capacity, % of turbine design steam flow	100	15	25	25	25
Main Condenser					
Heat removal capacity, Btu/hr	3500×10^6	3760×10^6	3520×10^6	5376.6×10^6	7770×10^6
Circulating Water System					
Number of pumps	3	2	2	4	3
Flow rate, gpm per pump	117,000	140,000	141,500	162,500	200,000
Condensate and Feedwater Systems					
Design flow rate, lb/hr	6.4×10^6	6.77×10^6	7.143×10^6	9.773×10^6	13.999×10^6
Number condensate pumps	2	2	2	3	3
Number feedwater pumps	2	2	2	2	3
Condensate pump drive	Ac power	Ac power	Ac power	Ac power	Ac power
Feedwater pump drive	Ac power	Ac power	Ac power	Turbine	Turbine

[a] Historical Information. Not updated for Extended Power Uprate.

Table 1.3-3

COMPARISON OF ELECTRICAL POWER SYSTEMS DESIGN CHARACTERISTICS

Parameter	[redacted]	[redacted]	DAEC[a]	[redacted]	[redacted]
			Transmission System		
Outgoing lines (number/rating)	2/345 kV	2/345 kV	2/345 kV	4/345 kV	4/500 kV
			Normal Auxiliary AC Power		
Incoming lines (number/rating)	2/345 kV 1/230 kV 1/115 kV	2/345 kV 2/230 kV 1/115 kV	2/345 kV 3/161 kV	1/115 kV 1/69 kV	2/161 kV
Auxiliary transformers	1	1	1	1	3
Startup transformers	1	1	1	1	2
Shutdown transformers	- -	- -	- -	1	0
			Standby AC Power Supply		
Number of diesel generators	2	2	2	2	4
Number of 4160-V standby buses	2	2	2	2	4
Number of 480-V standby buses	3	3	3	3	8
			DC Power Supply		
Number of 125- or 250-V batteries	2	3	2	2	4
Number of 125- or 250-V buses	4	3	2	4	4

[a] Historical Information. Not updated for Extended Power Uprate.

COMPARISON OF CONTAINMENT DESIGN CHARACTERISTICS

Parameter	[redacted]	[redacted]	DAEC[b]	[redacted]	[redacted]
			Primary Containment [a]		
Type	Pressure suppression	Pressure suppression	Pressure suppression	Pressure suppression	Pressure suppression
Construction					
Drywell	Light bulb shape, steel vessel	Light bulb shape, steel vessel	Light bulb shape, steel vessel	Light bulb shape, steel vessel	Light bulb shape, steel vessel
Pressure suppression chamber	Torus, steel vessel	Torus, steel vessel	Torus, steel vessel	Torus, steel vessel	Torus, steel vessel
Pressure suppression chamber internal design pressure, psig	+56	+56	+56	+56	+56
Pressure suppression chamber external design pressure, psi	+2	+2	+2	+2	+1
Drywell internal design pressure, psig	+56	+56	+56	+56	+56
Drywell free volume, ft^3	134,000	134,200	109,450	145,430	159,000
Pressure suppression chamber free volume, ft^3	99,000	98,280	91,670	109,810	119,000
Pressure suppression chamber pool water volume, ft^3	78,000	68,000	58,000	87,660	135,000
Submergence of vent pipe below pressure suppression chamber pool surface, ft	+4	+4	+4	+4	+4
Design temperature of drywell, °F	281	281	281	281	281
Design temperature of pressure suppression chamber, °F	281	281	281	281	281

[b] Historical Information. Not updated for Extended Power Uprate.
[a] When applicable, containment parameters are based on maximum anticipated power output.

COMPARISON OF CONTAINMENT DESIGN CHARACTERISTICS

Parameter	[redacted]	[redacted]	DAEC[b]	[redacted]	[redacted]
Primary Containment [a] (continued)					
Downcomer vent pressure loss factor	6.21	6.21	4.4	6.21	4.1
Break area/total vent area	0.019	0.019	0.019	0.019	0.017
Calculated maximum pressure after blowdown, psig					
Drywell	35	41	54	46	49.1
Pressure suppression chamber	22	26	25	28	27
Initial pressure suppression chamber temperature rise, °F	35	50	50	50	40
Leakage rate, % free volume/day at 56 psig and 281°F	0.5	0.5	0.5	9.5	0.5
Secondary Containment					
Type	Controlled leakage, elevated 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 superstructure and siding	Steel superstructure and siding	Steel superstructure and siding	Steel superstructure and siding	Steel superstructure and siding
Roof	Steel sheeting	Steel sheeting	Steel sheeting	Steel sheeting	Steel sheeting

[b] Historical Information. Not updated for Extended Power Uprate.
[a] When applicable, containment parameters are based on maximum anticipated power output.

Table 1.3-4

COMPARISON OF CONTAINMENT DESIGN CHARACTERISTICS

Parameter	[redacted]	[redacted]	DAEC[a]	[redacted]	[redacted]
			Secondary Containment (Continued)		
Internal design pressure, psig	0.25	0.25	0.25	0.25	2 in. water
Design inleakage rate, % free volume/day at 0.25 in. H_20	100	100	100	100	100
			Elevated Release Point		
Type	Stack	Stack	Stack	Stack	Stack
Construction	Reinforced concrete	Reinforced concrete	To be determined	Steel	Steel
Height (above ground)	97 m	89 m	100 m	100 m	200 m

[a] Historical Information. Not updated for Extended Power Uprate.

Table 1.3-5

COMPARISON OF STRUCTURAL DESIGN CHARACTERISTICS

Parameter	[redacted]	[redacted]	DAEC	[redacted]	[redacted]
Seismic Design					
Operating-basis earthquake (horizontal g)	0.07	0.06	0.06[a]	0.10	0.10
Design-basis earthquake (horizontal g)	0.14	0.12	0.12[a]	0.20	0.20
Wind Design					
Maximum sustained wind, mph	80	100	105	100	100
Tornados, mph	300	300	300	300	300

[a] On rock.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 ARCHITECT-ENGINEER AND CONSTRUCTOR

Iowa Electric retained Bechtel to provide engineering and construction services for the design and construction of the plant and to integrate the items furnished by General Electric with complete balance-of-plant items. Bechtel was also responsible for the procurement of all equipment other than the nuclear steam supply system and the turbine-generator.

Bechtel has been engaged in construction or engineering activities since 1898. Bechtel has been active in the fields of pipelines, petroleum, power generation and distribution, harbor development, mining and metallurgy, and chemical and industrial processing. The Bechtel organization has grown progressively to become one of the world's largest engineer-constructors for industrial facilities. Since the close of World War II, Bechtel has been responsible for the design of many thermal power generating units representing considerable generating capacity, including nuclear and nonnuclear. Bechtel is qualified to provide required services for station design, equipment procurement, construction, and startup.

1.4.2 NUCLEAR STEAM SUPPLY SYSTEM SUPPLIER

General Electric was awarded the contract to design, fabricate, and deliver the nuclear steam supply system and nuclear fuel for the plant, as well as to provide technical direction for the startup of this equipment. General Electric has been engaged in the development, design, construction, and operation of BWRs since 1955. Operating BWRs designed and built by General Electric include, in part, [redacted]. Thus, General Electric has substantial experience, as well as the knowledge and capability, to design and manufacture the reactor and to furnish technical direction for its startup.

1.4.3 TURBINE-GENERATOR SUPPLIER

The contract to design, fabricate, and deliver the turbine-generator for the plant, and to provide technical assistance for the installation and startup of this equipment was awarded to General Electric. General Electric's application of turbine-generators in nuclear power stations dates back to the inception of nuclear facilities for the production of electric power. General Electric furnishes the turbine-generator units for most stations equipped with its BWR nuclear steam supply systems. General Electric also supplies turbine-generator units for various

other nuclear power plants. The inlet pressures of these units generally vary between 750 and 1500 psig, and temperatures generally vary from saturation to approximately 40°F superheat.

The ratings of these units generally range from 500,000 to 1,090,000 kW. Therefore, General Electric is capable of designing, fabricating, and delivering turbine-generator units and of providing technical direction for the startup of this equipment.

1.4.4 REACTOR VESSEL AND CONTAINMENT SUPPLIER

The Chicago Bridge and Iron Company supplied and erected the reactor vessel. The same company furnished and erected the primary containment. The Chicago Bridge and Iron Company has been engaged in the construction of heavy wall pressure vessels for many years. [redacted]

1.4.5 CONSULTANTS

The information in this section is historical and has not been updated. It was valid at the time of submission of the initial FSAR.

The following consultants were retained by Iowa Electric to provide specialized services:

1. Dames and Moore--geology, seismology, foundation investigation, demography.

2. John A. Blume Associates--dynamic analysis of plant structures.

3. TRC of New England--meteorology.

4. NUS Corporation--environmental radiation analysis and specialized environmental studies.

5. Biotest Laboratories--environmental radiation program specifications.

6. University of Iowa--ecological consulting and environmental impact studies.

7. Nuclear Services Corporation--design review and related engineering services.

8. Pickard, Lowe and Associates--fuel management.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

The design of this facility, and the design of features related to its safe operation, are based on proven technological concepts derived from the development, design, and operating experience of numerous similar facilities. Both the Advisory Committee on Reactor Safeguards (ACRS) and AEC staff identified, during the construction permit stage of the DAEC, certain areas requiring resolution or supporting information during the construction stage or at the time of license review. Also, the ACRS, in their review of other reactor projects, identified an interest in additional technical areas that may be related to the DAEC.

Tables giving the preoperational status of these areas indicating the planned or accomplished resolution or response are included in this section, which, it should be noted, is primarily of historical interest. Tables 1.5-1 through 1.5-3 cover the following areas of interest:

1. Areas specified in the ACRS construction permit letter for the DAEC. (Refer to Table 1.5-1.)

2. Areas specified in the DAEC staff construction permit safety evaluation report. (Refer to Table 1.5-2.)

3. Related areas specified in the ACRS construction permit and operating license reports for other facilities. (Refer to Table 1.5-3.)

Table 1.5-1

AREAS SPECIFIED IN ACRS CONSTRUCTION PERMIT LETTER FOR THE DAEC

Item Description	Location of Information in FSAR
Emergency cooling water system	9.2.1
Secondary containment bypass leakage	6.2.6
Instrument line isolation	1.8
Failure to scram	7.2
Hydrogen control	4.6.3
Fuel cask drop	9.1.4

Table 1.5-2

AREAS SPECIFIED IN THE DAEC STAFF CONSTRUCTION PERMIT
SAFETY EVALUATION REPORT

Item Description	Location of Information in FSAR
Startup vibration testing	14.2.5
Leak detection	5.2
MSIV closure time	5.4.5
Main condenser and turbine isolation	7.6
Effect of low-quality steam on HPCI turbine	6.3.4
NSSS allowable loading	Chapters 3 & 6
Drywell support	3.8
Fire protection	9.5
Fuel damage limit criterion	4.4.5
Effects of fuel bundle flow blockage	Chapter 15
Effects of fuel clad failure on emergency core cooling	6.3.3
Core spray effectiveness	6.3.4
Steam line isolation valve testing	Topical report APED-5750
HPCI system, depressurization	Topical report
Hydrogen control	1.8

Table 1.5-3

RELATED AREAS SPECIFIED IN ACRS CONSTRUCTION PERMIT AND OPERATING LICENSE REPORTS FOR OTHER FACILITIES

Item Description	Location of Information in FSAR
Offsite emergency plans	13.5.2
Plant startup program	14.2.12
Reactor pressure-temperature relationships	5.3
Sensitized stainless steel	5.2
Quality assurance and inspection of the reactor primary system	Chapter 17
Quality assurance and inspection of the reactor vessel	Chapter 17
Recirculation pump, motor runaway during LOCA	Chapter 15
AEC general design criterion 35 (now 31)	3.1
Misorientation of fuel assemblies	4.4.2
Standby gas treatment system	6.5
Steam line isolation valve testing	Topical report APED-5750
Containment inerting	6.2.1
Quality, leak detection, isolation, and surveillance of the ECCS	6.3.4
Venting nonflow lines	5.4
Pressure relief valves, common-mode failure	5.4.13

Table 1.5-3

RELATED AREAS SPECIFIED IN ACRS CONSTRUCTION PERMIT AND OPERATING LICENSE REPORTS FOR OTHER FACILITIES

Item Description	Location of Information in FSAR
Containment environment effects on vital components	Chapter 3
Degeneration of ECCS components	6.3.3
ECCS thermal effects on the reactor vessel and internals	Topical report NEDO-10029; 6.3.3
Effects of blowdown forces on reactor vessel components	3.9
Seismic design and analysis models	Chapter 3
Seismic design of piping systems	3.9
"As-built" stress analysis	3.9
Adequacy of biological shield	3.6.1
Control rod guide tube collapse	3.9.4
Startup vibration testing	14.2.5
Turbine-generated missile damage	10.2.4
Separation of control and electrical system functions	7.2.2
LPCI system - initiation logic	6.3.5
Control systems for emergency power	8.3

Table 1.5-3

Sheet 3 of 3

RELATED AREAS SPECIFIED IN ACRS CONSTRUCTION PERMIT AND OPERATING LICENSE REPORTS FOR OTHER FACILITIES

Item Description	Location of Information in FSAR
Automatic depressurization system, single-failure capability	6.3
Prompt detection of gross fuel failure	7.5.1
Vibration and loose parts detection	7.7.1
Testability of engineered safety features	7.3
Automatic depressurization system, initiation interlock	7.3.1
Diesel-generator synchronization	8.3.2
Performance testing of the standby diesel-generator system	8.3.2
Liquid radwaste system design	11.2.2
Design basis of engineered safety features	Chapter 6
Structural material, water contamination, LOCA	Chapter 15
Fuel damage limit criterion	4.4.5
Effects of fuel bundle flow blockage	Chapter 15
Effects of fuel clad failure on emergency core cooling	6.3.3
Clad temperature, ECCS performance for a LOCA	6.3.4
HPCI system, depressurization capability	Topical report APED-5447

1.6 MATERIAL INCORPORATED BY REFERENCE

Table 1.6-1 is a list of General Electric topical reports that provide information in support of this UFSAR and that have been submitted to the NRC. Certain reports have been designated as General Electric Company proprietary documents and it has been requested that they be withheld from public disclosure.

Table 1.6-1 Sheet 1 of 8

DAEC TOPICAL REPORTS SUBMITTED TO THE NRC

GENERAL ELECTRIC REPORT NO.	TITLE	UFSAR SECTION
APED-4784	Design and Operating Experience of the ESADA Vallecitos Experimental Experimental Superheat Reactor (EVESR) (February 1965)	15.0
APED-4827[a]	Maximum Two-Phase Vessel Blowdown from Pipes (1965)	5.2.5.2.3
APED-5177	Liquid/Vapor Action in a Vessel During Blowdown (June 1966)	15.0
APED-5286[a]	Design Basis for Critical Heat Flux in Boiling Water Reactors (September 1966)	
APED-5296[a]	RIP-2, A Computer Program for Calculation of Reactor Internal Pressure During Accident Conditions (1966)	3.9.5.2.1
APED-5446	Control Rod Velocity Limiter (March 1967)	4.6.1.2.5.3
APED-5447	Depressurization Performance of the General Electric Boiling Water Reactor High-Pressure Coolant Injection System (June 1969)	6.2.1.3.3.5
APED-5448	Analysis Methods of Hypothetical Super-Prompt Critical Reactivity Transients in Large Power Reactors (April 1968)	
APED-5449[a]	Control Rod Worth Minimizer (March 1967)	
APED-5453	Vibration Analysis and Testing of Reactor Internals (April 1967)	
APED-5454	Metal Water Reactions - Effects on Core Standby Cooling and Containment (March 1968)	
APED-5455	The Mechanical Effects of Reactivity Transients (January 1968)	
APED-5458[a]	Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors (March 1968)	
APED-5460	Design and Performance of General Electric Boiling Water Reactor Jet Pumps (September 1968)	5.4.1
APED-5528	Nuclear Excursion Technology (August 1967)	
APED-5555	Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RDB144A (November 1967)	4.6.2.2
APED-5608	General Electric Company Analytical and Experimental Program for Resolution of ACRS Safety Concerns (April 1968) (Not Class I)	
APED-5640	Xenon Considerations in Design of Large Boiling Water Reactors (June 1968)	4.3.2.7.1

[a] Historical Reference Document only.

Table 1.6-1 Sheet 2 of 8

DAEC TOPICAL REPORTS SUBMITTED TO THE NRC

GENERAL ELECTRIC REPORT NO.	TITLE	UFSAR SECTION
APED-5652	Stability and Dynamic Performance of the General Electric Boiling Water Reactor (April 1969)	
APED-5654	Considerations Pertaining to Containment Inerting (August 1968)	
APED-5696	Tornado Protection for the Spent Fuel Storage Pool (November 1968)	1.8.13
APED-5698	Summary of Results Obtained from a Typical Startup and Power Test Program for a General Electric Boiling Water Reactor (February 1969)	14.2.1.3
APED-5703	Design and Analysis of Control Rod Drive Reactor Vessel Penetrations (November 1968)	
APED-5706	In-Core Neutron Monitoring System for General Electric Boiling Water Reactors, Revision 1 (April 1969)	
APED-5736	Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards (April 1969)	
APED-5750	Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves (March 1969)	5.4.5.4
APED-5756[a]	Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor (March 1969)[b]	15.0
NEDC-20989	Mark I Containment - Short Term Program Report (1975)	6.2.1.6.1
NEDC-20989	Addendum 1, Mark I Containment - Short Term Program Report (December 1975)	6.2.1.6.1
NEDC-22082-P[a]	Duane Arnold Energy Center Suppression Pool Temperature Response (March 1982)	6.2.1.3.3.3, 9.2.3.2.1
NEDC-22204[a]	Evaluation of Mark I S/RV Load Cases C-3.1, C-3.2, C-3.3, for the DAEC (September 1982)	5.4.13.2
NEDC-23677	Duane Arnold Feedwater Nozzle Temperature Cycling (1977)	5.4.9.2.3
NEDC-30603-P-1[a]	Duane Arnold Energy Center Power Uprate (December 1984)	

[a] Historical Reference Document only.
[b] Historical Information only. Methods for Radiological Consequences of Design Basis Accidents were revised under Amendments 237 and 240 to follow RG1.183 Revision 0.

Table 1.6-1 Sheet 3 of 8

DAEC TOPICAL REPORTS SUBMITTED TO THE NRC

GENERAL ELECTRIC REPORT NO.	TITLE	UFSAR SECTION
NEDC-30626[a]	General Electric Boiling Water Reactor Extended Load Line Limit Analyses for Duane Arnold Energy Center, Cycle 8	4.4.3.3, 15.0.10
NEDC-30813-P[a]	Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for the DAEC (December 1984)	7.6.1.8.3, 7.6.1.8.4
NEDC-30839	DAEC Reactor Pressure Vessel Fracture Toughness Analysis to 10 CFR 50, Appendix G (May 1983)	5.3.2.1
NEDE-20566-P-A[a]	General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K (November 1975)	
NEDE-20606-P-A	Creep Collapse Analysis of BWR Fuel Using SAFE-COLAPS Model (August 1976)	4.2.1.2.2
NEDE-21354-P	BWR Fuel Channel Mechanical Design and Deflection (September 1976)	4.2.1.1.5, 4.2.2
NEDE-21480	BWR Feedwater Nozzle/Sparger Interim Program Report (July 1977)	5.4.9.2.3
NEDE-21855-P	FSCRD Manufacturing Qualification Test Final Report (August 1978)	6.2.1.6.2.2
NEDE-21864-P	In-Plant Testing of T-Quencher Device at Monticello Nuclear Generating Plant (July 1978)	6.2.1.6.2.2
NEDE-23898-P	Analytical Model for Computing Water Rise in a Safety/Relief Valve Discharge Line Following Valve Closure	5.4.13.2
NEDE-24011-P-A	General Electric Standard Application for Reactor Fuel (latest appicable version)	4.2, 4.3, 4.4
NEDE-24284-P	Assessment of Fuel Rod Bowing in GE BWRs (December 1980)	4.2.3.1.5
NEDE-24343-P	Experience with BWR Fuel Through January 1981 (September 1976)	4.2.4
NEDE-24988-P	BWR Owners Group SRV Test Program (October 1981)	5.4.13.4
NEDE-30021-P[a]	Low-Low Set Relief Logic System and Lower MSIV Water Level Trip for the Duane Arnold Energy Center (January 1983)	5.4.13.2, 15.0
NEDE-30051[a]	Analysis of Reduced RHR Service Water Flow at the DAEC (January 1983)	9.2.3.2.1
NEDO-10017	Field Testing Requirements for Fuel, Curtains, and Control Rods (June 1969)	

[a] Historical Reference Document only.

Table 1.6-1 Sheet 4 of 8

DAEC TOPICAL REPORTS SUBMITTED TO THE NRC

GENERAL ELECTRIC REPORT NO.	TITLE	UFSAR SECTION
NEDE-32417	GE12 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR-II)	4.2.3, 4.3.2, 15.0
NEDO-10029	An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design-Basis Accident (July 1969)	5.3.1.5, 18.3.1.6
NEDO-10045[a]	Consequences of a Steam Line Break for a General Electric Boiling Water Reactor (October 1969)	5.4.5.3
NEDO-10115	Mechanical Property Surveillance of General Electric BWR Vessels (July 1969)	5.3.1.6, 5.3.2.1, 5.3.3.1
NEDO-10139	Compliance of Protection Systems to Industry Criteria; GE BWR NSSS (June 1970)	1.8.22, 6.3.5.4, 7.2.1.2.4
NEDO-10173	Current State of Knowledge - High Performance BWR Zircaloy- Clad UO_2 Fuel (May 1970)	
NEDO-10174	Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor (May 1970)	
NEDO-10179	Effects of Cladding Temperature and Material on ECCS Performance (June 1970)	
NEDO-10189	An Analysis of Functional Common-Mode Failures in General Electric BWR Protection and Control Instrumentation (July 1970)	
NEDO-10208	Effects of Fuel Rod Failure on ECCS Performance (August 1970)	
NEDO-10299	Core Flow Distribution in a Modern BWR as Measured at Monticello (1971)	3.9.4.3.2
NEDO-10320[a]	General Electric Pressure Suppression Containment Analytical Model (April 1971)	3.1.2.5.1, 6.2.1.3.3.1, 15.0
NEDO-10329[a]	Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors (April 1971)	3.9.1.2
NEDO-10349[a]	Analysis of Anticipated Transients without Scram (March 1971)	4.6.2.1
NEDO-10527[a]	Rod Drop Accident Analysis for Large Boiling Water Reactors (March 1972 and supplements)	4.2.1.2.6, 4.6.3.1.1, 4.6.2.4

[a] Historical Reference Document only.

Table 1.6-1 Sheet 5 of 8

DAEC TOPICAL REPORTS SUBMITTED TO THE NRC

GENERAL ELECTRIC REPORT NO.	**TITLE**	**UFSAR SECTION**
NEDO-10677	Analysis of Recirculation Pump Overspeed in a Typical General Electric BWR (October 30, 1972)	7.7.5.4.4
NEDO-10739	Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems (January 1973)	6.3.4.2.1
NEDO-10801-A	Core Spray and Bottom Flooding Effectiveness in the BWR-6 (February 1977)	6.3.4.1.1
NEDO-20377	8 x 8 Fuel Development Support (February 1975)	4.2.3.1.3, 4.2.3.1.5
NEDO-20566-A	General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K (August 1975)	Superceded by NEDC-23785-P-A
NEDO-20606-A	General Electric Creep Collapse Analysis for BWR Fuel Using SAFE-COLARS Model (1976)	4.2.1.2.2
NEDO-21052	Maximum Discharge Rate of Liquid-Vapor Mixtures From Vessels (March 1982)	6.2.1.3.3.3
NEDO-21082-03[a]	Loss-of-Coolant Accident Analysis Report for Duane Arnold Energy Center (Lead Plant) (June 1984)	
NEDO-21082-03[a]	Appendix A, Loss-of-Coolant Accident Analysis Report for Duane Arnold Energy Center (Lead Plant) (June 1984)	
NEDO-21354	BWR Fuel Channel Mechanical Design and Deflection (1976)	4.2.1.1.5, 4.2.2
NEDO-21888	Mark I Containment Program Load Definition Report (November 1981)	3.1.2.5.1 6.2.1.3.3.3
NEDO-22082-P[a]	DAEC Suppression Pool Temperature Response (March 1982)	6.2.1.3.3.3
NEDO-22155	Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containments (August 1982)	6.2.5.1
NEDO-24011-P-A-US	General Electric Standard Application for Reactor Fuel - United States Supplement (latest approved revision)	6.3.1.2, 6.3.3, 15.0
NEDO-24087-3[a]	General Electric Boiling Water Reactor Reload 3 (Cycle 4) Licensing Amendment for Duane Arnold Energy Center, Supplement 3: Application of Measured Scram Insertion Times (June 1978)	

[a] Historical Reference Document only.

Table 1.6-1 Sheet 6 of 8
DAEC TOPICAL REPORTS SUBMITTED TO THE NRC

GENERAL ELECTRIC REPORT NO.	TITLE	UFSAR SECTION
NEDO-24087-6[a]	General Electric Boiling Water Reactor Reload 3 Cycle 4) Licensing Amendment for Duane Arnold Energy Center, Supplement 6: Load Line Limit Analysis (September 1978)	4.4.3.3
NEDO-24134-1	General Electric Process Specification for Heat Sink Welding of Austenitic Stainless Steel (1978)	5.2.3.4
NEDO-24220	Basis for Installation of Recirculation Pump Trip System (September 1979)	7.2.1.2.3
NEDO-24226	Evaluation of Control Blade Lifetime with Potential Loss of B_4C (December 1979)	4.2.1.1.8
NEDO-24232	Control Blade Lifetime Evaluation Accounting for Potential Loss of B_4C (1980)	4.6.1.2.5, 4.6.1.2.5.2
NEDO-24272[a,c]	Duane Arnold Energy Center Single-Loop Operation (July 1980)	4.4, 5.4.3.3, 15.0
NEDO-24571	Mark I Containment Program Plant Unique Load Definition, Duane Arnold Energy Center, Unit 1 (March 1982)	15.2.1
NEDO-24708a[a]	Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors (Rev 1, December 1980)	15.0
NEDO-30603-P[a]	General Electric Company, Duane Arnold Energy Center Power Uprate (May 1984)	
NEDO-30603-1[a]	General Electric Company, Duane Arnold Energy Center Power Uprate (Revision 1) December 1984	
NEDO-30606[a]	General Electric Company Safety Relief Valve Simmer Margin Analysis for the Duane Arnold Energy Center (May 1984)	

[a] Historical Reference Document only.
[c] This report has been supplemented by MDL# APED LI2-003 [DAEC Supplement to NEDC-32915P, Duane Arnold Energy Center GE12 Fuel Upgrade Project, Rev. 0, March 2000], which was not submitted to the NRC, but done as part of the Core Modification Package for Cycle 17.

Table 1.6-1 Sheet 7 of 8
DAEC TOPICAL REPORTS SUBMITTED TO THE NRC

GENERAL ELECTRIC REPORT NO.	TITLE	UFSAR SECTION
NEDO-30813[a]	General Electric BWR Licensing Report: Average Power Range Monitor, Rod Block Monitor, and Technical Specification Improvement (ARTS) Program for the Duane Arnold Energy Center (March 1985)	4.4.3.3
NEDO-31908	General Electric Report, Licensing Criteria for Fuel Designs (Amendment 22 to NEDE-24011-P-A and Corresponding NRC Staff Safety Evaluation (January 1991).	4.2.1, 4.3.2
NEDC-31310-P[a]	Duane Arnold Energy Center, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (August 1986)	4.2.3.2.8, 4.2.3.3.3, 6.3.3, 15.0
NEDC-31310-P[a] Supplement 1, Revision 1	Duane Arnold Energy Center, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (September 1993)	4.2.3.2.8, 4.2.3.3.3, 6.3.3, 15.0
NEDC-32721P	General Electric Licensing Topical Report: Application Methodology for GE Stacked Disc ECCS Suction Strainer (November 1997)	1.8.1, Figure 5.4-15, 6.2.1.6.2.5, 6.3.2.2, 6.3.2.2.8, Table 6.3-3, Table 6.3-4
NEDO-32686-A	General Electric BWROG Topical Report: Utility Resolution Guide for ECCS Suction Strainer Blockage (October 1998)	1.8.1, Figure 5.4-15, 6.2.1.6.2.5, 6.3.2.2, 6.3.2.2.8, Table 6.3-3, Table 6.3-4
NEDC-32868P	GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR-II)	4.2.3 4.3.2.1 4.3.3 4.3.4 15.2.4 15.2.5
NEDO-32980/ NEDC-32980	Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate (November 2000)	15.0
NEDC-32992P[a]	ODYSY Application for Stability Licensing Calulations	4.4.4.6

[a] Historical Reference Document only.

Table 1.6-1

DAEC TOPICAL REPORTS SUBMITTED TO THE NRC

	GENERAL ELECTRIC REPORT NO.	**TITLE**	**UFSAR SECTION**
	NEDE-33213P-A	ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions (April 2009)	4.4.4.6
2019-001	NEDE-33766P-A	GEH Simplified Stability Solution (GS3) (March 2015)	4.4.4.6
2012-020	NEDC-33270P	GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II), NEDC-33270P, Revision 7 (October 2016)	4.2.3, 4.3.3, 4.3.4, 15.0, 15.2
	NEDC-33256P-A	The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 1 - Technical Bases (September 2010)	4.2.3, 6.3.3, 15.0
	NEDC-33257P-A	The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 2 - Qualification (September 2010)	4.2.3, 6.3.3, 15.0
	NEDC-33258P-A	The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 3 - Application Methodology (September 2010)	4.2.3, 6.3.3, 15.0
	WCAP-16182-P-A	Westinghouse BWR Control Rod CR 99 Licensing Report (March 2005)	4.1.2.1.1, 4.2.1.1.8, 4.6.1.2.5

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

1.7.1 ELECTRICAL, INSTRUMENTATION, AND CONTROL DRAWINGS

EI&C drawings are referenced throughout the FSAR text and listed in the List of Figures at the beginning of each chapter.

1.7.2 PIPING AND INSTRUMENTATION DIAGRAMS

A current list of piping and instrumentation diagrams (P&IDs) submitted to the NRC is given in Table 1.7-1. The P&IDs were initially submitted to the NRC, Region III, in 1980 by Reference 1.

1.7.3 OTHER DETAILED INFORMATION

Responses to NRC Staff requests for detailed information have been incorporated in the appropriate sections throughout the updated FSAR and have not been duplicated in this section.

REFERENCES FOR SECTION 1.7

1. Letter from P. D. Ward, Iowa Electric, to J. G. Keppler, NRC, Subject: Document Revision, dated March 31, 1980.

Table 1.7-1

PIPING AND INSTRUMENTATION DIAGRAMS

P&ID No.	Description
M-100	Piping Symbols
M-101	Instrument Symbols
M-102	Instrument Identification
M-103	Main Steam
M-104	Turbine Extraction Steam and Drains, Sheet 1
M-105	Turbine Extraction Steam and Drains, Sheet 2
M-106	Condensate and Feedwater, Sheet 1
M-107	Condensate and Feedwater, Sheet 2
M-108	Condensate Demineralizer System
M-109	Condensate and Demineralized Water System
M-110	Makeup Demineralizer System
M-111	General Service Water System
M-112	Reactor Building Cooling Water System
M-113	RHR Service Water and Emergency Service Water Systems
M-114	Nuclear Boiler System
M-115	Reactor Vessel Instrumentation
M-116	Reactor Recirculation System
M-117	Control Rod Drive Hydraulic System, Sheet 1
M-118	Control Rod Drive Hydraulic System, Sheet 2
M-119	Residual Heat Removal System, Sheet 1

Table 1.7-1

PIPING AND INSTRUMENTATION DIAGRAMS

P&ID No.	Description
M-120	Residual Heat Removal System, Sheet 2
M-121	Core Spray System
M-122	High-Pressure Coolant Injection System, Steam Side, Sheet 1
M-123	High-Pressure Coolant Injection System, Water Side, Sheet 2
M-124	Reactor Core Isolation Cooling System, Steam Side, Sheet 1
M-125	Reactor Core Isolation Cooling System, Water Side, Sheet 2
M-126	Standby Liquid Control System
M-127	Reactor Water Cleanup System
M-128	Reactor Water Filter-Demineralizer
M-129	River Water Supply System Intake Structure
M-130	Compressed Air System
M-131	Turbine Lube-oil System
M-132	Diesel-Generator Systems
M-133	Fire Protection
M-134	Fuel Pool Cooling and Cleanup System
M-135	Fuel Pool Filter-Demineralizer System
M-136	Service Condensate System
M-137	Radwaste Sump System
M-138	Equipment Radwaste System (Closed)
M-139	Floor Drain Radwaste System (Open)

Table 1.7-1

PIPING AND INSTRUMENTATION DIAGRAMS

P&ID No.	Description
M-140	Radwaste Solids Handling System
M-141	Offgas System
M-142	Circulating Water System
M-143	Containment Atmosphere Control System
M-144	Well Cooling Water System
M-145	Miscellaneous Turbo Generator System
M-146	Service Water System Pump House
M-147	Turbine Building Sample System
M-148	Area Radiation Monitoring System
M-149	Offgas Recombiner
M-150	Heating, Ventilating, and Air Conditioning Plant Air Flow Diagram
M-151	Control Building Air Flow Diagram
M-152	Reactor Building Air Flow Diagram
M-153	Turbine Building Air Flow Diagram
M-154	Heating, Ventilating, and Air Conditioning Radwaste Building Flow Diagram
M-155	Administration Building Air Flow Diagram
M-156	Drywell Air Flow Diagram
M-157	Drywell Cooling Water System
M-158	Heating, Ventilating, and Air Conditioning P&ID and Air Flow Diagram, Standby Gas Treatment System

Table 1.7-1

PIPING AND INSTRUMENTATION DIAGRAMS

P&ID No.	Description
M-159	Ventilation Systems, Turbine Building
M-160	Auxiliary Heating System, Boiler and Main Loop
M-161	Air Conditioning System, Control Building
M-162	Auxiliary Heating System, Reactor Building
M-163	Auxiliary Heating System, Turbine Building
M-164	Ventilation System, Radwaste Building
M-165	Main Plant Air Intake and Motor-Generator Room
M-166	Cooling and Heating System, Plant Air supply
M-167	Administration Building Heating and Cooling System
M-168	Administration Building Heating and Cooling System
M-169	Control Building Heating System, Plant Chilled Water System
M-170	Heating, Ventilating, and Air Conditioning Miscellaneous Control System
M-171	Reactor Building HVAC Cooling Systems
M-172	P&ID and Air Flow Diagram, Heating and Cooling Systems, Machine Shop and Offgas Retention Building
M-173	P&ID and Air Flow Diagram, Standby Filter Unit Control Building

Table 1.7-1

Sheet 5 of 5

PIPING AND INSTRUMENTATION DIAGRAMS

P&ID No.	Description
M-174	Dry Well Heating and Ventilating Fan System
M-175	P&ID and Air Flow Diagram, Pump House
M-176	Ventilation System and Offgas Stack, Reactor Building
M-177	Intake Structure Heating and Ventilating Control Diagram
M-179	Heating, Ventilating, and Air Conditioning Symbols and Abbreviations
M-180	Chlorination and Acid Feed Systems
M-181	Containment Atmosphere Monitoring System
M-182	Radwaste Evaporator System
M-183	Radwaste Sample System
M-184	Main Steam Isolation Valve Leakage
M-185	Cardox System
M-186	Radwaste Liquid Waste Storage and Handling System
M-187	Postaccident Sampling
M-188	Low-level Radwaste Processing and Storage Facility Heating and Ventilation System
M-189	Hydrogen Water Chemistry

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

The information in this section represents either the original or an updated position with respect to AEC Safety Guides, which have since been redesignated as NRC Regulatory Guides. Where the original DAEC position has been updated, that fact is so noted.

Only those guides addressed in the original FSAR are included in this section of the updated FSAR. Guides published after the original FSAR was written may in some cases be addressed elsewhere in the updated FSAR.

1.8.1 SAFETY GUIDE 1 (REGULATORY GUIDE 1.1), NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS

This Section has been updated since the initial submittal of the DAEC FSAR.

Regulatory Position

Emergency core cooling and containment heat removal systems should be designed so that adequate net positive suction head (NPSH) is provided to system pumps assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present prior to postulated LOCAs.

Response

The emergency core cooling and containment heat removal functions are accomplished by the emergency core cooling systems. The entire spectrum of possible operating modes of the emergency core cooling systems has been examined for adequacy with regard to net positive suction head at the residual heat removal (RHR), core spray, and high pressure coolant injection (HPCI) pumps. Under no circumstance would there be insufficient net positive suction head at any of the pumps at any time.

These pumps are located below the water level of the suppression pool and/or condensate storage tanks. To demonstrate that net positive suction head would be available at all times, the various modes of operation were examined, and the most limiting for NPSH requirements was determined to occur during the long term transient following a design basis LOCA when core spray and one RHR pump will be running continuously. In this operating condition, the NPSH requirements for the core spray pump are most limiting.

The analysis of this situation demonstrated that available containment pressure was greater than the containment pressure required for adequate net positive suction head to the core spray pump, even though assumptions were used to minimize the containment pressure and maximize the temperature of the suppression pool water. Figure 5.4-15 <Sheet 1> indicates the margin

available between actual suppression pool pressure and that pressure required for adequate core spray and RHR pump net positive suction head. The major assumptions are listed below:

1. Offsite power is assumed lost at the time of the accident and is not restored.

2. One of the onsite diesel-generators fails to start and remains out of service during the entire transient.

3. The service water temperature remains at the Technical Specification limit of 95°F throughout the transient. Normally, service water temperature would be at least 10°F less than this value.

4. The service water flow to the RHR heat exchanger is assumed to be maximized so the containment spray water temperature is lowered; this minimizes the containment pressure.

5. Before the accident, the Technical Specification temperature limit of 135°F exists in the drywell together with 100% humidity. Normal operating conditions would typically be 125° F with 20% humidity.

6. The minimum preaccident containment pressure is 0.5 psig (nominal value); normal operating pressure is typically ≈ 1.0 psig.

7. A containment gas leakage rate of 5.0% per day; this is 2.5 times the maximum allowable leakage rate (L_a) of 2.0% per day incorporated in the Technical Specifications.

8. The discharge of the RHR pump(s) is directed to the containment atmosphere via the broken recirculation loop (short-term), and via the drywell and torus sprays (long-term); this minimizes the containment pressure.

Although the safety guide requirement of no increase in containment pressure is not met exactly, the use of the pressure within the containment to provide additional suction head to the pumps is not unreasonable. This factor would exist in reality and would be greater than calculated due to the conservatism of the analysis.

However, restrictions on additional reliance on overpressure, in either the absolute magnitude or duration, have been added to ensure that adequate NPSH remains available (Reference 5). Figure 5.4-15(b) represents a typical long-term accident response, with the currently-approved overpressure limits, which are the more restrictive of the following:

a) Containment (wetwell) pressure available, based upon the above accident analysis, is at least 2.7 psi greater than the pressure required for adequate NPSH, when the required containment pressure is above atmosphere (14.7 psia).
b) Maximum containment (wetwell) pressure that can be credited for NPSH is 22 psia (7.3 psig).
c) Containment (wetwell) overpressure may not be credited beyond 36 hours after reactor shutdown.

Revision of the above limits in overpressure for demonstrating adequate NPSH requires prior NRC review and approval.

1.8.2 SAFETY GUIDE 2 (REGULATORY GUIDE 1.2), THERMAL SHOCK TO REACTOR PRESSURE VESSELS

This Section has been updated since the initial submittal of the DAEC FSAR. This Regulatory Guide was withdrawn by the NRC in July 1991.

1.8.3 SAFETY GUIDE 3 (REGULATORY GUIDE 1.3), ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR BOILING WATER REACTORS

NOTE: Information that follows is HISTORICAL only for personnel dose and doses to the public. The assumptions used for evaluating the potential Radiological Consequences of a Loss of Coolant Accident for the DAEC were changed to comply with RG 1.183 in License Amendment 237 and 240.

Regulatory Position

1. The assumptions related to the release of radioactive material from the fuel and containment are:

   a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core is immediately available for leakage from the primary reactor containment. Eighty-seven percent of this twenty-five percent is in the form of elemental iodine, five percent of this twenty-five percent is in the form of particulate iodine, and eight percent of this twenty-five percent is in the form of organic iodides.

   b. One-hundred percent of the equilibrium radioactive noble gas inventory developed from maximum full power operation of the core is immediately available for leakage from the reactor containment.

   c. The effect of radiological decay during holdup in the containment or other buildings is taken into account.

   d. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials is evaluated on an individual case basis.

   e. The primary containment leaks at the leak rate incorporated or to be incorporated in the technical specifications for the duration of the accident. The leakage passes directly to

the emergency exhaust system without mixing in the surrounding reactor building atmosphere and is then released as an elevated plume for those facilities with stacks.

f. No credit is given for retention of iodine in the suppression pool.

2. The assumptions for atmospheric diffusion and dose conversion are:

a. Elevated releases are considered to be at a height equal to no more than the actual stack height. Certain site dependent conditions may exist, such as surrounding elevated topography or nearby structures which will have the effect of reducing the actual stack height. The degree of stack height reduction is evaluated on an individual case basis. Also, special meteorological and geographic conditions may exist which can contribute to greater ground level concentrations in the immediate neighborhood of a stack. For example, fumigation is always assumed to occur; however, the length of time that a fumigation condition exists is strongly dependent on geographical and season factors and will be evaluated on a case-by-case basis.

b. No correction is made for depletion of the effluent plume of radioactive iodine due to deposition on the ground, or for the radiological decay of iodine in transit.

c. For the first eight hours, the breathing rate of persons offsite is assumed to be 3.47×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate is assumed to be 1.75×10^{-4} cubic meters per second. After that, until the end of the accident, the rate is assumed to be 2.32×10^{-4} cubic meters per second. (These values were developed from and correspond to the average daily breathing rate [2×10^{7} cm^3/day] assumed in the report of ICRP, Committee II-1959.)

d. The iodine conversion factors are given in ICRP Publication 2, Report of Committee II, Permissible Dose for Internal Radiation, 1959.

e. External whole body doses are calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. "Such a cloud would be considered an infinite cloud for a receptor at the center because any additional gamma and beta-emitting material beyond the cloud dimensions would not alter the flux of gamma rays and beta particles to the receptor" (Meteorology and Atomic Energy, Section 7.4.1.1 - editorial additions made so that gamma and beta emitting material could be considered). Under these conditions the rate of energy absorption per unit volume is equal to the rate of energy released per unit volume.

The following specific assumptions are used:

(i) The dose at any distance from the reactor is calculated based on the maximum concentration in the plume at that distance taking into account special meteorological, topographical, and other characteristics which may affect the maximum plume concentration. These site related characteristics must be

evaluated on an individual case basis. In the case of beta radiation, the receptor is assumed to be exposed to an infinite cloud at the maximum ground level concentration at that distance from the reactor. In the case of gamma radiation, the receptor is assumed to be exposed to only one-half the cloud owing to the presence of the ground. The maximum cloud concentration is always assumed to be at ground level.

(ii) The appropriate average beta and gamma energies emitted per disintegration, as given in the Table of Isotopes, Sixth Edition, by C. M. Lederer, J. M. Hollander, I. Perlman, University of California, Berkeley, Lawrence Radiation Laboratory, are used.

(i) The basic equation for atmospheric diffusion from an elevated release is:

$$\frac{X}{Q} = \frac{e^{h^2/2\sigma_z^{-2}}}{\pi u \sigma_y \sigma_z}$$

where

X = the short term average centerline value of the ground level concentration (curies/meter^3)

Q = amount of material released (curies/sec)

u = wind speed (meters/sec)

σ_y = the horizontal standard deviation of the plume (meters) (See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.)

σ_z = the vertical standard deviation of the plume (meters) (See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.)

h = effective height of release (meters)

(ii) The equation used for releases with time periods of greater than 8 hours, to determine the atmospheric diffusion from an elevated release assumed to meander and spread uniformly over a 22.5° sector is:

$$\frac{X}{Q} = \frac{2.032}{\sigma_Z u x} \left(e h^2/2\sigma_z^{-2}\right)$$

where

x = distance from the release point (meters) and other variables are as given in f (i) above.

(iii) The atmospheric diffusion model, for an elevated release as a function of the distance from the reactor, shall be based on the following:

TIME FOLLOWING ACCIDENT	ATMOSPHERIC CONDITIONS
0-8 hours	Envelope of Pasquill diffusion categories based on Figure A7, Meteorology and Atomic Energy-1968, assuming various stack heights, wind speed 1 meter/sec, uniform direction.
8-24 hours	Envelope of Pasquill diffusion categories, wind speed 1 meter/sec; variable direction within a 22.5° sector.
1-4 days	Envelope of Pasquill diffusion categories with the following relationship used to represent maximum plume concentrations as a function of distance:
Case 1*	40% Pasquill A 60% Pasquill C
Case 2*	50% Pasquill C 50% Pasquill D
Case 3*	33.3% Pasquill C 33.3% Pasquill D 33.3% Pasquill E
Case 4*	33.3% Pasquill D 33.3% Pasquill E 33.3% Pasquill F
Case 5*	50% Pasquill D 50% Pasquill F
4-30 days	Same diffusion relations as given above; wind speed variable dependent on Pasquill Type used; wind direction 33.3% frequency in a 22.5° sector.

*Atmospheric Condition - windspeed variable (Pasquill Types A, B, E, and F, wind speed 2 meters/sec; Pasquill Types C and D, wind speed 3 meters/sec); variable direction within a 22.5° sector.

Response

The calculations performed for the DAEC conform to this guide.

1.8.4 REGULATORY GUIDE 1.4, ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS

This regulatory guide is not applicable to the DAEC.

1.8.5 SAFETY GUIDE 5 (REGULATORY GUIDE 1.5), ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A STEAMLINE BREAK ACCIDENT FOR BOILING WATER REACTORS

NOTE: Information that follows is HISTORICAL only for personnel doses and doses to the public. The assumptions used for evaluating the potential Radiological Consequences of a Loss of Coolant Accident for the DAEC were changed to comply with RG 1.183 in License Amendments 237 and 240.

Regulatory Position

1. The steamline breaks with the reactor at full power and the reactor scrams.

2. The steamline isolation valves close in the maximum time incorporated or to be incorporated in the technical specifications. This closure time will be verified by suitable periodic testing.

3. The total mass of coolant released is that amount in the steamline and connecting lines at the time break plus the amount that passes through the valves prior to closure. (Ruptures within the turbine complex are not considered in this guide.)

4. The radioactivity in the coolant is assumed to be the maximum amount incorporated or to be incorporated in the technical specifications, provided that no further fuel failures are assumed to occur as a result of delays in valve closure.

5. All of the iodine (no credit for plateout is allowed) and noble gases from the released coolant are released to the atmosphere within two hours at a height of thirty meters with a fumigation condition.

6. The assumptions for dose conversion and atmospheric diffusion are:

   a. No correction is made for depletion from the effluent plume of radioactive iodine due to deposition on the ground, or for the radiological decay of iodine 131 in transit.

   b. The breathing rate of persons offsite is assumed to be 3.47×10^{-4} cubic meters per second. (This value is developed from the average daily breathing rate (2×10^{7} cm^3/day) assumed in the report of ICRP, Committee II-1959.)

c. The iodine dose conversion factors are given in ICRP Publication 2, Report of Committee II, "Permissible Dose for Internal Radiation," 1959.

d. External whole body doses are calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. "Such a cloud would be considered an infinite cloud for a receptor at the center because any additional (gamma and) beta emitting material beyond the cloud dimensions would not alter the flux of (gamma rays and) beta particles to the receptor" (Meteorology and Atomic Energy, Section 7.4.1.1 - editorial additions made so that gamma and beta emitting material could be considered). Under these conditions the rate of energy absorption per unit volume is equal to the rate of energy released per unit volume.

The following specific assumptions are used:

(i) The dose at any distance from the reactor is calculated based on the maximum concentration in the plume at that distance taking into account special meteorological, topographical, and other characteristics which may affect the maximum plume concentration. These site related characteristics must be evaluated on an individual case basis. In the case of beta radiation, the receptor is assumed to be exposed to an infinite cloud at the maximum ground level concentration at that distance from the reactor. In the case of gamma radiation, the receptor is assumed to be exposed to only one-half the cloud owing to the presence of the ground. The maximum cloud concentration is always assumed to be at ground level.

(ii) The appropriate average beta and gamma energies emitted per disintegration, as given in the Table of Isotopes, Sixth Edition, by C. M. Lederer, J. M. Hollander, I. Perlman; University of California, Berkeley, Lawrence Radiation Laboratory, shall be used.

e. (i) The equation used to determine the atmospheric diffusion from an elevated release at 30 meters, uniform wind direction with a fumigation condition existing is:

$$X/Q = \frac{0.0133}{\sigma_y u}$$

where

X = The short term average centerline value of the ground level concentration (curies/meter^3)

Q = Amount of material released (curies/sec)

σ_y = The horizontal standard deviation of the plume (meters)
(See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2; Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.)

u = Wind speed (meters/sec)

(ii) Figure 1 gives elevated release atmospheric diffusion factors assuming a release at thirty meters, and atmospheric conditions assumed to be Pasquill F, wind speed 1 meter/sec.

Response

The calculations performed for the DAEC conform to this guide.

1.8.6 SAFETY GUIDE 6 (REGULATORY GUIDE 1.6), INDEPENDENCE BETWEEN REDUNDANT STANDBY (ONSITE) POWER SOURCES AND BETWEEN THEIR DISTRIBUTION SYSTEMS

This Section has been updated since the initial submittal of the DAEC FSAR.

Regulatory Position

1. The electrically powered safety loads (a-c and d-c) should be separated into redundant load groups such that the loss of any one group will not prevent the minimum safety functions from being performed.

Response

Electrically powered safety loads have been physically and functionally separated into redundant load groups. The functional separation is illustrated and discussed in the UFSAR as follows:

SYSTEM	SECTION
4160-V ac	8.3
480-V ac	8.3
125-V dc	8.3
250-V dc	8.3

Redundant components are also physically separated so that their independence is maintained. The loss of any load group will not prevent required safety functions from being achieved.

Regulatory Position

2. Each ac load group should have a connection to the preferred (offsite) power source and to a standby (onsite) power source (usually a single diesel-generator). The standby power source should have no automatic connection to any other redundant load group. At multiple nuclear unit sites, the standby power source for one load group may have an automatic connection to a load group of a different unit. A preferred power source bus, however, may serve redundant load groups.

Response

Each ac load group (1A3 and 1A4) has a connection to the offsite distribution via the startup and the standby transformers and its own diesel-generator (standby power source). Each load group is indirectly connected to the main generator through the startup and standby transformers via the switchyard. The diesel-generators have no automatic connection to the other redundant load group. This feature is illustrated and discussed in Section 8.3.

Regulatory Position

3. Each dc load group should be energized by a battery and battery charger. The battery charger combination should have no automatic connection to any other redundant dc load group.

Response

125-V DC System. Each 125-V dc distribution panel (1D10 and 1D20) is supplied by its own independent battery (1D1 and 1D2, respectively) and its own independent charger (1D12 and 1D22, respectively). The backup battery charger for the 125-V dc system can be manually placed on either battery via a key interlock. This system is illustrated and discussed in Section 8.3.

250-V DC System. The 250-V dc distribution panel (1D40) is supplied by its own battery (1D4) and two independent battery chargers (1D43 and 1D44), which serve no other dc load group. This is illustrated and discussed in Section 8.3.

24-V DC System. The 24-V dc distribution panels (1D50 and 1D60) are each supplied by an independent battery (1D5 and 1D6, respectively) and by two independent battery chargers (1D51-1D52 and 1D61-1D62, respectively), which serve no other redundant dc load group. This is illustrated and discussed in Section 8.3.

Regulatory Position

4. When operating from the standby sources, redundant load groups and the redundant standby sources should be independent of each other at least to the following extent:

a. The standby source of one load group should not be automatically paralleled with the standby source of another load group under accident conditions.

Response

The diesel-generator breakers are interlocked to prevent closure if any of the other supply breakers to panels lA3 or lA4 are closed. This interlock is overridden for diesel tests.

Regulatory Position

b. No provisions should exist for automatically connecting one load group to another load group.

Response

No provisions exist for automatically connecting one load group to another. Panels 1B1 and 1B2 can be manually connected as can panels 1B5 and 1B6.

Regulatory Position

c. No provisions should exist for automatically transferring loads between redundant power sources.

Response

No provisions exist to automatically transfer loads between diesel-generators except as follows: motor control center 1B34A and lB44A (LPCI isolation valves) are powered by the diesel-generators via an automatic transfer device, and motor control center lB37 is powered from lB34A.

A split bus system is unacceptable for the LPCI system because, for the split bus system, [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Automatic Depressurization System (ADS). Each individual ADS valve has its own automatic transfer device. Since the only common features in the system are common terminal blocks in the control panels, this automatic transfer feature is single-failure proof. Multiple power supplies are required for ADS valves to ensure that, on loss of one power supply, the ADS feature is not lost.

Regulatory Position

d. If means exist for manually connecting redundant load groups together, at least one interlock should be provided to prevent an operator error that would parallel their standby power sources.

Response

No manual means exist whereby the diesel-generators can be paralleled.

Regulatory Position

5. A single generator driven by a prime mover is acceptable as the standby power source for each ac load group of the size and characteristics typical of recent applications. If other arrangements such as multiple diesel generators operated in parallel or multiple prime movers driving a single generator are proposed, the applicant should demonstrate that the proposed arrangement has an equivalent reliability. Common mode failures as well as random single failures should be considered in the analysis.

Response

Two redundant independent diesel-generators provide standby power for the DAEC. Each standby distribution system is adequate to conduct the required safety-related functions of the electrical system. This system is described in Section 8.3.

1.8.7 SAFETY GUIDE 7 (REGULATORY GUIDE 1.7), CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT

This section has been updated since the initial submittal of the DAEC FSAR.

In September 2003, the NRC revised 10CFR50.44, “Standards for Combustible Gas Control for Nuclear Power Reactors.” This rule change reflects the position that only combustible gas generated by a “beyond-design basis accident” (i.e., a severe accident) is a risk-significant threat to containment integrity, provided that Mark I containments are inerted at the start of the accident. The revision to 10CFR50.44 eliminates requirements that previously

pertained to design-basis accidents. The following are the new rule requirements and the DAEC conformance to those new requirements.

Regulatory Position

Each boiling or pressurized water nuclear power reactor must have the capability for ensuring a mixed atmosphere.

Response

Uniform mixing of the containment atmosphere is ensured by diffusion and other driving forces such as natural and forced convection. The one driving force of mixing that can be precisely calculated (i.e., diffusion) is sufficient to ensure that the maximum volume with oxygen concentration more than 0.1% oxygen greater than the average oxygen concentration is less than 10 ft^3.

Regulatory Position

Each boiling water reactor with a Mark I or Mark II type containment must have an inerted atmosphere.

Response

The DAEC containment atmosphere is verified to be inerted, i.e., contain less than 4% oxygen concentration by volume, every seven days, in accordance with Technical Specifications.

Regulatory Position

Each boiling water reactor with Mark III type containment and each pressurized water reactor with an ice condenser containment must have the capability for controlling combustible gas generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region so that there is no loss of containment structural integrity.

Response

The DAEC does not have a Mark III type containment and is not a Pressurized Water Reactor with an ice condenser containment. This requirement does not apply to DAEC.

Regulatory Position

Each boiling water reactor with Mark III type containment and each pressurized water reactor with an ice condenser containment that does not rely upon an inerted atmosphere inside containment to control combustible gasses must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen.

Response

The DAEC does not have a Mark III type containment and is not a Pressurized Water Reactor with an ice condenser containment. This requirements does not apply to DAEC.

Regulatory Position

Equipment must be provided for monitoring oxygen in the containment that uses an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant beyond design basis accident for combustible gas control and accident management, including emergency planning.

Response

Existing oxygen monitoring systems approved by the NRC prior to the effective date of the amendment to 10CFR50.44, are sufficient to meet this criterion.

Regulatory Position

Equipment must be provided for monitoring hydrogen in the containment that uses an inerted atmosphere for combustible gas control. Equipment for monitoring hydrogen must be functional, reliable and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design basis accident for combustible gas control and accident management, including emergency planning. Note: Continuous indication of hydrogen concentration is not required during normal operations. If an indication is not available at all times, continuous indication and recording shall be functioning within 30 minutes of the initiation of safety injection.

Response

Existing hydrogen monitoring systems approved by the NRC prior to the effective date of the amendment to 10CFR50.44, are sufficient to meet this criterion.

1.8.8 SAFETY GUIDE 8 (REGULATORY GUIDE 1.8), PERSONNEL SELECTION AND TRAINING

This section has been updated since the initial submittal of the DAEC FSAR.

Regulatory Position

The criteria for the selection and training of nuclear power plant personnel contained in ANSI N18.1 "Proposed Standard for Selection and Training of Personnel for Nuclear Power Plants," dated June 22, 1970, are generally acceptable and provide an adequate basis for the selection and training of nuclear power plant personnel. In some cases, plant design features or unusual operating conditions may indicate that additional or more specialized expertise is needed. This determination will be made on an individual case basis.

Response

Functional Levels and Assignment of Responsibility. The established functional levels and assignment of responsibility are described in Chapter 13.

Qualifications The DAEC staff will have that combination of education, experience, health, and skills commensurate with their level of responsibility that provides reasonable assurance that decisions and actions during all normal and abnormal conditions will be such that the plant is operated in a safe and efficient manner.

1.8.9 SAFETY GUIDE 9 (REGULATORY GUIDE 1.9), SELECTION OF DIESEL GENERATOR SET CAPACITY FOR STANDBY POWER SUPPLIES

This Section has been updated since the initial submittal of the DAEC FSAR.

Regulatory Position

1. At a time when the characteristics of loads are not accurately known, such as during the construction permit stage of design, each diesel generator set on a standby (onsite) power supply should be selected to have a continuous load rating equal to or greater than the sum of the conservatively estimated loads needed to be powered at any one time. In the absence of fully substantiated performance characteristics for mechanical equipment such as pumps, the electric motor drive ratings should be calculated using conservative estimates of these characteristics, (for example, pump run-out conditions and motor efficiencies of 90% or less).

Response

The initial loading requirements for the standby diesel-generators were conservatively estimated. At that time, a 2850-kW (continuous rating) diesel-generator was selected. This capacity was based on the conservatively estimated loads and an allowance for adding new loads in the future.

Regulatory Position

2. At the operating license stage of review, the predicted loads should not exceed the smaller of the 2000-hour rating, or 90 percent of the 30-minute rating of the set.

Response

At the operating license stage of review, the predicted maximum required load on the diesel-generator falls below the 2000-hr rating of 3000 kW by about 8% and in fact is still below the continuous rating of 2850 kW. The required and optional loads on the diesel-generator are presented in plant design documents. Only the 2000-hr rating is considered here since the highest diesel-generator rating is 3250 kW for 300 hr. A 30-min rating is not available.

Regulatory Position

3. During preoperational testing, the predicted loads should be verified by tests.

Response

Preoperational tests will be performed to verify that the diesel-generator capacity is sufficient to supply power for the predicted plant loads required during the postulated design conditions. These required loads are given in Chapter 8.

In addition to the preshipment shop testing performed on each diesel-generator unit, preoperational testing of the complete standby diesel-generator system will be performed as follows:

1) Verification of starting time capability by applying a start signal to each of the diesel-generator units and measuring the time required for starting and achieving operating (loading) readiness will be done a minimum of 100 times.

2) Concurrent verification of the required capacity of stored starting air and also the recovery rate of the air compressors.

3) Tests for 50%, 75%, 100%, and 110% of rated loads will be performed while recording the power output, fuel consumption, and electric output.

4) Testing that simulates the design loading sequence will be performed.

5) Testing of governor performance in accordance with appropriate DEMA field test code.

Periodic inservice testing will be performed on partial-load and full-load buses, including automatic circuitry, at the intervals stated in the Technical Specifications.

Refer also to Chapter 8.

Regulatory Position

4. Each diesel generator set should be capable of starting and accelerating to rated speed, in the required sequence, all the needed engineered safety feature and emergency shutdown loads. At no time during the loading sequence should the frequency and voltage decrease to less than 95% of nominal and 75% of nominal, respectively. During recovery from transients caused by step load increases or resulting from the disconnection of the largest single load, the speed of the diesel-generator set should not exceed 75% of the difference between nominal speed and the overspeed trip set point or 115% of nominal, whichever is lower.

Voltage should be restored to within 10% of nominal and frequency should be restored to within 2% of nominal in less than 40% of each load sequence time interval.

Note: the use of "within" in the above is intended to mean "at least" and is not intended to imply a +/- window about the nominal value. There is no requirement on overshoot above nominal as a transient condition, only as a steady state requirement after all loads are sequenced onto the diesel-generator.

Response

Each diesel-generator is capable of starting and accelerating all required engineered safety feature and emergency shutdown loads in the required sequence, without causing the voltage to decrease to less than 75% of nominal or the frequency to decrease to less than 95% of nominal, with the exception of the initial diesel loading. Table 8.3-1 indicates that the initial load causes the voltage to dip to approximately 73% of nominal. This dip occurs for approximately 0.7 sec after energizing the load, after which the voltage increases to 100% of rated within 2.0 sec. Since the initial voltage on the bus is zero and the voltage is rapidly increased to 100%, this initial dip has little effect on the acceleration of these initial loads.

The largest single step-load decrease that could occur is the loss of a core spray pump. The diesel-generator overspeed trip is set at 115% of rated speed. A trip of the core spray pump will result in the overspeed of no more than 11.25% (75% of 15%). It should be noted that even for an assumed loss of load equal to the continuous rating of the diesel (2850 kW), no overspeed trip results.

The voltage will recover to at least 90% of rated after each step-load change in 1.3 sec (26.0% of load sequence time interval) or less and to at least 98% of rated frequency within 3.91 sec (78.2% of load sequence time interval). In each case, the frequency is not restored to within at least 2% of nominal until the motors have accelerated to rated speed; the effect of a frequency lower than 2% of rated frequency during the acceleration interval is insignificant.

Regulatory Position

5. The stability of each diesel generator set of the standby power supply should be confirmed by prototype qualification test data and preoperational tests.

Response

The Fairbanks-Morse Model 38TD8-1/8 opposed-piston diesel-generator design has been subjected to specific tests that prove its suitability for nuclear standby service. These tests and the results are presented in IEEE paper 69CP 177-PWR. Fairbanks-Morse has received orders for approximately 30 diesel-generators for standby nuclear service, about eight of which are now in service.

1.8.10 SAFETY GUIDE 10 (REGULATORY GUIDE 1.10), MECHANICAL SPLICES FOR REINFORCING BARS OF CONCRETE CONTAINMENTS

This section has been updated since the initial submittal of the DAEC FSAR.

This regulatory guide was withdrawn by the NRC in July 1981.

1.8.11 SAFETY GUIDE 11 (REGULATORY GUIDE 1.11), INSTRUMENT LINES PENETRATING PRIMARY REACTOR CONTAINMENT

This Section has been updated since the initial submittal of the DAEC FSAR.

Regulatory Position

To implement General Design Criteria 55 and 56 for instrument lines penetrating or connected to primary reactor containment:

1. Sensing lines for instruments that are part of the protection system:

    a. Should satisfy the requirements for redundancy, independence, and testability of the protection system.

Response

Reactor protection system instrument sensing lines have been provided with redundancy commensurate with their function. The independence of the redundant sensing lines has been satisfied by providing maximum possible physical separation, both inside and outside the drywell. Chapter 6 addresses the separation at the penetrations. The redundancy and testability for the protection system are discussed in the following chapters:

1. Reactor Protection System, Chapter 7.

2. Primary Containment Isolation and Nuclear Steam Supply Shutoff System, Chapter 7.

3. Core Standby Cooling Systems Control and Instrumentation, Chapter 7.

4. Reactor Vessel Instrumentation, Chapter 7.

Regulatory Position

b. Should be sized or orificed to assure that in the event of a postulated failure of the piping or of any component (including the postulated rupture of any valve body) in the line outside primary reactor containment during normal reactor operation:

    (i) The leakage is reduced to the maximum extent practical consistent with other safety requirements.

(ii) The rate and extent of coolant loss is within the capability of the reactor coolant makeup system.

(iii) The integrity and functional performance of secondary containment, if provided, and associated safety systems (e.g., filters, standby gas treatment system) will be maintained.

(iv) The potential offsite exposure will be substantially below the guidelines of 10 CFR 100.*

* original Regulatory position. DAEC Licensing basis has been changed to 10 CFR 50.67.

Response

Instrument sensing lines that penetrate the drywell and connect to the reactor coolant pressure boundary are provided with orifices located just inside the drywell. The sizing of the orifices is discussed in Chapter 6.

The orificed instrument lines are designed to minimize coolant leakage without affecting other safety requirements. Instrumentation response is not unacceptably degraded by the inclusion of the orifice as discussed in Chapter 6.

An evaluation of the consequences of an unisolatable rupture of an instrument line that penetrates the primary containment and is connected to the reactor coolant pressure boundary was performed. It was assumed that this highly unlikely rupture would occur external to the primary containment but upstream of the isolation valve. It was conservatively assumed that the line would continuously discharge reactor water to the secondary containment through a 1/4-in. orifice, at a rate equal to the initial discharge rate of 2.7 lb/sec for the duration of the detection and cooldown sequence (3.5 hr total).

Chapter 5 discusses the capability of reactor coolant makeup systems and indicates that the capability is adequate to cope with the calculated discharge from the instrument line.

The orificed instrument lines are designed such that the integrity and functional performance of the secondary containment and its associated safety systems (including the standby gas treatment system) will be maintained. In addition, the environmental effects due to an instrument line break in the reactor building have been shown to be negligible (Section 15.2).

Regulatory Position

c. Should be provided with an isolation valve capable of automatic operation* or remote operation from the control room or from another appropriate location, and located in the line outside the containment as close to the containment as practical. There should be a high degree of assurance that this valve,

(i) Will not close accidentally during normal reactor operation.

(ii) Will close or be closed if the instrument line integrity outside containment is lost during normal reactor operation or under accident conditions.

(iii) Will reopen or can be reopened under the conditions that would prevail when valve reopening is appropriate. Power-operated valves should remain as-is upon loss of power. The status (open or closed) of all such isolation valves should be indicated in the control room. If a remotely operable valve is provided, sufficient information should be available in the control room or other appropriate location to assure timely and proper actions by the operator.

Response

Excess flow check valves specifically designed for the DAEC are provided in each instrument process line that penetrates the drywell and is part of the reactor coolant pressure boundary. This excess flow check valve is designed so that it will not close accidentally during normal operation, will close if a rupture of the instrument line is indicated downstream of the valve, can be reopened when appropriate, and has its status indicated in the control room.

This valve is discussed in Chapter 6.

Instrument process lines that penetrate the drywell and communicate with the drywell atmosphere are discussed in Chapter 6.

Regulatory Position

d. Should be conservatively designed up to and including the isolation valve and of a quality at least equivalent to the containment. These portions of the lines should be located and protected so as to minimize the likelihood of their being damaged accidentally. They should be protected or separated to prevent failure of one line from inducing failure of any other line. Provisions should be included to permit periodic visual in-service

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* A self-actuated excess flow check valve is acceptable as an automatically operated valve provided it has all other features specified in the guide.

inspection particularly of these portions of the lines outside containment up to and including the isolation valve.

Response

Instrument sensing lines are designed to maintain a quality commensurate with their containment functions. The piping code classification of these lines is illustrated in Chapter 3. The seismic classification criteria are also discussed in Chapter 3.

The protection and separation afforded redundant instrument lines have been optimized such that the postulated failure of one line will not induce the failure of another redundant line.

Instrument lines are run with sufficient access to facilitate, to the extent practicable, periodic visual inservice inspection. Excess flow check valves are located very near the containment; nothing will be installed to prevent inservice inspection of the valves or lines from the penetration to the isolation valves.

Regulatory Position

e. Should not be so restricted by components in the lines, such as valves and orifices, that the response time of the connected instrumentation will be increased to an unacceptable degree.

Response

All restrictions have been selected to ensure acceptable instrument response time, as discussed in Chapter 6.

Regulatory Position

2. Sensing lines for instruments that are not part of the protection system:

    a. Should meet the provisions of 1.b, l.c, l.d, and 1.e., above, or

    b. Should be provided with one automatic isolation valve inside and one automatic valve outside containment. The valve outside should be located as close to containment as practical.

Response

All instrument sensing lines that penetrate the primary containment and connect to the reactor coolant pressure boundary satisfy the requirements of l.b, l.c, l.d, and l.e.

1.8.12 SAFETY GUIDE 12 (REGULATORY GUIDE 1.12), INSTRUMENTATION FOR EARTHQUAKES

This Section has been updated since the initial submittal of the DAEC FSAR.

Regulatory Position

1. At each nuclear power plant site, one strong motion triaxial accelerograph in the basement of the reactor containment structure and another strong motion triaxial accelerograph at a higher elevation of the reactor containment structure should be installed to provide data on the frequency, amplitude, and phase relationship of the seismic response of the containment structure and the seismic input for other Category I structures, systems, and components.

Response

Two strong motion triaxial accelerographs, each with its associated seismic trigger, are installed within the reactor building of the DAEC. This equipment will detect and record data on frequency, amplitude, and phase relationship resulting from seismic disturbances. The intended function of the strong-motion triaxial accelerographs data is for post event analytical investigation. There is no playback unit in the control room since the strong-motion triaxial accelerographs output is not relied on as a basis for timely operator action. For this function, a seismic alarm system employing a multilevel acceleration readout is employed as discussed in the response to Regulatory Position 6 of this safety guide.

Each strong-motion triaxial accelerograph has as an integral part a seismic trigger which senses the vertical component of the initial earthquake ground motion and actuates the strong-motion triaxial accelerographs to full operation within less than 0.1 sec. The strong-motion triaxial accelerographs can record a single earthquake or a sequence of earthquake and after-shocks lasting as long as 25 minutes.

One strong-motion triaxial accelerograph is located in the basement and the other strong-motion triaxial accelerograph [redacted] is equipped with an event indicator located in the main control room.

Regulatory Position

2. The accelerographs should be:

   a. Separated from each other by a vertical distance which is a significant fraction of the containment building height.

   b. Oriented such that the three axes of the accelerograph sensors on one accelerograph are pointing in the same directions as the three axes of the other accelerograph.

   c. Located such that one accelerograph is directly over the other.

d. Located such that they are accessible for needed periodic servicing and recovery of the recorded traces following an earthquake.

e. Mounted rigidly to the containment structure or on a structure directly connected to the containment structure such that the accelerograph records can be related to containment structure movement.

<u>Response</u>

The strong-motion triaxial accelerographs are installed in the [redacted] and at the [redacted] These instruments are mounted (located) and oriented as required by this safety guide.

<u>Regulatory Position</u>

3. Other instrumentation, such as peak recording accelerographs and peak deflection recorders, should also be installed on other selected Seismic Category I structures, systems, and components to verify the seismic response determined analytically from the traces recorded by the strong motion accelerographs. The extent to which such other instrumentation need be installed will be evaluated on a case basis.

<u>Response</u>

Peak recording accelerographs are installed in other Seismic Category I structures such as the control building, intake structure, pump house, and in the [redacted] Peak recording accelerographs are also provided on recirculation system piping, main steam piping, and the reactor pressure vessel.

This equipment will detect and record triaxial peak amplitudes of low-frequency acceleration resulting from seismic disturbances. The peak recording accelerograph requires no power, and records on 0.25-in. magnetic tape or a similar recording medium without the danger of excessive radiation destroying the trace. Advantages of the peak recording accelerographs include simplicity of operation and maintenance, and positive recording ability.

<u>Regulatory Position</u>

4. If another Seismic Category I structure, or a structure which contains Seismic Category I systems or components has a foundation independent from that of the reactor containment and is expected to have a significantly different response to an earthquake due to different underlying soil conditions of unique structural dynamic characteristics, instrumentation should also be provided to determine the seismic response of the Seismic Category I structure and the Seismic Category I equipment within the structure. The extent to which such other instrumentation need be installed will be evaluated on a case basis.

<u>Response</u>

The seismic instrumentation discussed in the previous sections of this design guide are considered adequate to monitor this plant site.

Regulatory Position

5. Where more detailed knowledge of soil structure interactions is important to an accurate assessment of potential damage to Category I structures, systems, or components, a free field accelerograph should also be installed.

Response

[redacted] It is set to trigger at 0.01g.

Regulatory Position

6. The value of the peak acceleration level experienced in the basement of the reactor containment structure should be indicated in the control room or available to the control room operator within a few minutes after the earthquake. The indication of the peak acceleration level may be in the form of traces recorded by the accelerograph, either by direct readout or by quick playback of the recorded signals, a multilevel acceleration readout which indicates when predetermined values have been exceeded, or some other suitable means.

Response

A seismic alarm system employing a multilevel acceleration readout in the control room indicates when predetermined values have been exceeded.

Two seismic triggers are located together with a strong-motion accelerograph at the [redacted] basement level. The three orthogonal components of one of the seismic triggers have threshold acceleration levels for switch closure corresponding to the DAEC vertical and horizontal operating-basis earthquake (OBE) accelerations. The other seismic trigger is similarly set for switch closure corresponding to the DAEC vertical and horizontal design-basis earthquake (DBE) accelerations. Switch closure in the OBE case will result in the actuation of a trouble light at a control room location with a distinctive audio annunciation to summon the operator to the seismic alarm display panel. Switch closure in the DBE case will result in the actuation of a trouble light again with accompanying audio annunciation. In addition, the strong-motion triaxial accelerograph starter trigger, which is set at 0.01g vertical acceleration, is wired to the seismic alarm panel to actuate a trouble light with accompanying audio annunciation.

Regulatory Position

7. The instrumentation should be designed to perform its function satisfactorily over the appropriate range of environmental conditions, such as temperature, humidity, pressure, and vibration.

Response

The instrumentation is constructed in such a way that it will perform in a satisfactory manner within the range of environmental conditions expected at the plant site. All of the devices are of compact, rugged construction, sealed to prevent the entrance of moisture and dust, and protected from excessive radiation.

Regulatory Position

8. The Applicant should develop a plan for timely utilization of the data to be obtained from the installed seismic instrumentation.

Response

The multilevel acceleration readout described in the response to Regulatory Position 6 provides prompt indication of seismic accelerations to permit the comparison of such with that used as the design basis. In the event that the seismic alarm display indicated that acceleration values are in excess of OBE accelerations, the plant will be shut down.

To help predict the capability of the plant for resuming operations, field inspection of safety-related items will be implemented, and the measured response from both the peak recording and strong motion accelerographs will be compared with that used in the design to confirm the adequacy of the structure or equipment for further use. If the measured responses are less than the values used in the design for the DBE, the structure or equipment is considered still adequate for future operation. Otherwise a new analysis will be made to check the adequacy of these items for future use.

1.8.13 SAFETY GUIDE 13 (REGULATORY GUIDE 1.13), FUEL STORAGE FACILITY DESIGN BASIS

This Section has been updated since the initial submittal of the DAEC FSAR.

Regulatory Position

1. The spent fuel storage facility (including its structures and equipment except as noted in Section 6, below) should be designed to Seismic Category I requirements.

Response

This requirement has been met in the design of the DAEC.

[redacted] are discussed in Chapter 3. The seismic classification of the spent-fuel storage facility is also discussed in Chapter 3.

Regulatory Position

2. The facility should be designed to prevent cyclonic winds and missiles generated by these winds from causing significant loss of watertight integrity of the fuel storage pool and to prevent missiles generated by cyclonic winds from contacting fuel within the pool.

Response

[redacted]

[redacted]

Regulatory Position

3. Interlocks should be provided to prevent cranes from passing over stored fuel (or near stored fuel, in a manner that could result in tipping the load over on stored fuel in the event of crane failure) when fuel handling is not in progress. During fuel handling operations, the interlocks may be bypassed and administrative control used to prevent the crane from carrying loads that are not necessary for fuel handling over the stored fuel or other prohibited areas. The facility should be designed to minimize the need for bypassing such interlocks.

Response

Interlocks are provided to prevent movements of cranes over stored fuel, but these interlocks may be bypassed and administrative control used as necessary during fuel-handling operations. The design of the facility minimizes the need for bypassing these interlocks. These interlocks are described in Chapters 7 and 9. In Chapter 9, there is also a description of the separate cask pool that has been provided to prevent the positioning of the spent-fuel cask over the spent-fuel pool.

Regulatory Position

4. A controlled leakage building should be provided enclosing the fuel pool. The building should be equipped with an appropriate ventilation and filtration system to limit the potential release of radioactive iodine and other radioactive materials. The building need not be designed to withstand extremely high winds but leakage should be suitably controlled during refueling operations. The design of the ventilation and filtration system should be based on the assumption that the cladding of all of the fuel rods in one fuel bundle might be breached. The inventory of radioactive materials available for leakage from the building should be based on the assumptions given in the safety guide entitled, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident for Boiling and Pressurized Water Reactors."

Response

This requirement has been met in the design of the DAEC. The fundamental reactor building design is discussed in Chapter 1. The reactor building ventilation and filtration system is discussed in Chapters 6 and 9. The assumptions for fuel rod failure during a refueling accident are discussed in Chapter 15; the number of failed rods exceeds the number of rods in one fuel assembly.

The inventory of radioactive materials is discussed in Chapter 15.

The refueling operations are carried out entirely within the confines of the reactor building, which is a controlled leakage structure enclosing the fuel pool. The design of the reactor building and its low leakage characteristics are described in Chapter 6.

The reactor building is equipped with an appropriate ventilation and filtration system to limit the potential release of radioactive iodine and other radioactive materials.

During refueling activities the leakage characteristics of the reactor building and the operability of the reactor building ventilation and filtration systems may be relaxed. The relaxation was based on analysis of the radiological consequences of a fuel handling accident which showed that consequences remained within regulatory guidelines without crediting secondary containment or operation of the standby gas treatment system. In Reference 1, DAEC committed to implementing administrative means to facilitate restoration of the secondary containment should a fuel handling accident occur. In Reference 2, the NRC approved the

relaxation based on the DAEC commitment. In Reference 3, DAEC implemented guidelines to provide normal or contingency methods to restore secondary containment, to reduce potential dose, and to avoid unmonitored releases of radioactivity.

The design-basis fuel-handling accident is discussed in Chapter 15.

Regulatory Position

5. The spent fuel storage facility should have the following provisions with respect to the handling of heavy loads, including the refueling cask:

    a. Cranes capable of carrying heavy loads should be prevented, preferably by design rather than by interlocks, from moving into the vicinity of the pool, or

    b. The fuel pool should be designed to withstand, without leakage which could uncover the fuel, the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted. If this latter approach is followed, design provisions should be made to prevent this crane, when carrying heavy loads, from moving in the vicinity of stored fuel.

Response

The fuel pool is designed to withstand, without leakage that could uncover the fuel, the impact of the heaviest load (refueling cask) to be carried by the crane from the maximum height to which it can be lifted. This is accomplished by providing in the design of the spent-fuel storage facility, a separate cask pool cell with a suitable gating arrangement to facilitate underwater transfer of fuel from the storage area to the cask. This arrangement is shown in Chapter 9. [redacted] The refueling crane, when carrying heavy loads, is prevented from traversing the fuel storage portion of the fuel pool by means of interlocks as discussed in Chapters 7 and 9.

Regulatory Position

6. Drains, permanently connected systems, and other features that by maloperation or failure could cause loss of coolant that would uncover fuel should not be installed or included in the design. Systems for maintaining water quality and quantity should be designed so that any maloperation or failure in such systems (including failures resulting from the DBE) will not cause fuel to be uncovered. These systems need not otherwise meet Seismic Category I requirements.

Response

This requirement has been met in the design of the DAEC and is discussed in Chapter 9.

Regulatory Position

7. Reliable and frequently tested monitoring equipment should be provided that will alarm both locally and in a continuously manned location if the water level in the fuel storage pool falls below a determined level or if high local radiation levels are experienced. (The high radiation level instrumentation should also actuate the filtration system.)

Response

Fuel storage pool level is monitored by level switches mounted on skimmer surge tanks. Level is also monitored for the reactor well water level. This system is described in Chapter 9. Since fuel pool water is continuously recirculated, there is no need to have a high-radiation initiation. Local radiation is monitored by the area radiation monitoring system (Chapter 7). Radiation monitors are also located in the fuel pool ventilation exhaust to initiate secondary containment isolation if required; this is discussed in Chapter 6.

Regulatory Position

8. A Seismic Category I makeup system should be provided to add coolant to the pool. Appropriate redundancy or a backup system for filling the pool from a reliable source such as a lake, river, or onsite Seismic Category I water storage facility should be provided. If a backup system is used, it need not be a permanently installed system. The capacity of the makeup systems should be such that water can be supplied at a rate determined by consideration of the leakage rate that would be expected as the result of damage to the fuel storage pool from the dropping of loads, from earthquakes, or from missiles originating in high winds.

Additional guidance concerning protection against missiles that might be generated by plant failures, such as turbine failures, is being considered. For the present, the protection of the fuel pool against such missiles will be evaluated on an individual case basis.

Response

The RHR system is used as the source of makeup water and is classified as Seismic Category I. The Seismic Category I piping is extended into the fuel pool cooling system as far as necessary to ensure that makeup water will get into the fuel pool. The sources of water are the torus, the condensate storage tank, or the river water system (see Amendment 16 to the PSAR). The torus and river water system are classified as Seismic Category I. Service condensate is available to provide makeup for leakage and evaporative losses.

Subsequent review indicates continued correspondence between IES and the NRC on this issue. The RHR system is classified as Seismic Category I. However, the Seismic Category I piping is not extended into the Fuel Pool Cooling and CleanUp (FPCCU) system. The FPCCU system piping was constructed to ANSI B31.1 and is Seismic Category II. Emergency Service

Water (ESW) is the Seismic Category I water makeup source to the spent fuel pool as documented in Section 9.1 and on Figure 9.2-5.

Since the fuel cask movement will be controlled to prevent approaching the fuel storage pool and since missile damage is not considered plausible (see response to item 2), no determination has been made as to the expected leak rate due to a specific missile. Therefore, makeup capacity has not been made contingent on a missile-generated leak. At least 100 gpm is available from the condensate service system in accordance with GE specification 27Al423.

1.8.14 SAFETY GUIDE 14 (REGULATORY GUIDE 1.14), REACTOR COOLANT PUMP FLYWHEEL INTEGRITY

Regulatory Position

1. The flywheel material should be produced by a process that minimizes flaws in the material and improves its fracture toughness properties, such as the vacuum-melt and degassing process. The material should be examined and tested to meet the following criteria:

   a. The Nil Ductility Transition (NDT) temperature of the flywheel material, as obtained from the dropweight tests (DWT) performed in accordance with the specification ASTM E-208, should be no higher than 10°F.

   b. The Charpy V-notch (C_V) upper-shelf energy level in the "weak" direction (WR orientation in plates) of the flywheel material should be at least 50 ft-lb. A minimum of three C_V specimens should be tested from each plate or forging, in accordance with the specification ASTM A-370.

   c. The minimum fracture toughness of the material at the normal operating temperature of the flywheel should be equivalent to a dynamic stress intensity factor (K_{Ic} dynamic) of at least 100 ksi in. Compliance can be demonstrated by any of the following:

      (i) Testing of the actual material of the flywheel to establish the K_{Ic} (dynamic) value at the normal operating temperature.

      (ii) Testing of the actual material of the flywheel by means of C_V specimens oriented with respect to the "weak" direction (WR orientation in plates). The C_V impact tests should be conducted to define the C_V test curve up to at least 50 ft-lb fracture energy value. The C_V curve should then be adjusted for the NDT temperature and size effect, as described in the proposed AEC "Fracture Toughness Requirements," 10 CFR 50.55a, Appendix G, Section III.B. The adjusted fracture energy, as read from the adjusted C_V curve at the normal operating temperature of the flywheel, should be demonstrated to be equivalent to a K_{Ic} (dynamic) value of at least 100 ksi in. by using appropriate correlation data. The test data and the correlations used should be submitted to the regulatory staff for review.

(iii) Use of a lower bound fracture toughness curve obtained from tests on the same type of material. Such a curve should be translated along the temperature coordinate until the K_{Ic} (dynamic) value of 45 ksi in. is indicated at the NDT temperature of the material, as obtained from the DWT tests. The proposed lower bound fracture toughness curve should be submitted to the regulatory staff for review.

d. Each finished flywheel should be subjected to a 100% volumetric ultrasonic examination using procedures and acceptance criteria equivalent to those specified for Class 1 vessels in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components.

e. If the flywheel is flame cut from a plate or forging, at least 0.5 inch of stock should be left on the outer and bore radii for machining to final dimensions.

f. Finished machined bores, keyways and drilled holes should be subjected to magnetic particle or liquid penetrant examination.

2. The flywheel should be designed to withstand normal conditions, anticipated transients, the design basis loss of coolant accident, and the design basis earthquake without loss of structural integrity. The design of the pump flywheel should meet the following criteria:

   a. The combined primary stresses at the normal operating speed, due to centrifugal forces and the interference fit of the wheel on the shaft, should not exceed 1/3 of the minimum specified yield strength, or 1/3 of the measured yield strength in the weak direction of the material if appropriate tensile tests have been performed on the actual material of the flywheel.

   b. The design overspeed of the flywheel should be at least 10 percent above the highest anticipated overspeed. The anticipated overspeed should include consideration of the maximum rotational speed of the flywheel if a break occurs in the reactor coolant piping in either the suction or discharge side of the pump. The basis for the assumed design overspeed should be submitted to the regulatory staff for review.

   c. The combined primary stresses at the design overspeed, due to centrifugal forces and the interference fit, should not exceed 2/3 of the minimum specified yield strength, or 2/3 of the measured yield strength in the weak direction if appropriate tensile tests have been performed on the actual material of the flywheel.

   d. The shaft and the bearings supporting the flywheel should be able to withstand any combination of the normal operating loads, anticipated transients, the design-basis LOCA and the DBE loads.

3. Each flywheel assembly should be tested at the design overspeed of the flywheel.

4. The inservice inspection program for each flywheel should include the following:

a. An in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and keyway at approximately 3-year intervals, during the refueling or maintenance shutdown coinciding with the in-service inspection schedule as required by the ASME Boiler and Pressure Vessel Code, Section XI.

b. A surface examination of all exposed surfaces and complete ultrasonic volumetric examination at approximately 10-year intervals, during the plant shutdown coinciding with the in-service inspection schedule as required by the ASME Boiler and Pressure Vessel Code, Section XI. Removal of the flywheel is not required to perform these examinations.

c. Examination procedure and acceptance criteria in conformance with the requirements specified in C.1.d.

Response

This safety guide is specifically directed to "flywheel integrity." It does not apply to the DAEC because flywheels are not used on the recirculation pump motors. Sufficient flywheel effect is available in the motors to satisfy pump coastdown requirements without the addition of a flywheel.

Suggested requirements in item 1 of the above regulatory position are directed to heavy plates and forgings while the DAEC recirculation pump motor rotors use thin sheet metal laminations.

General Electric has performed work to define the maximum recirculation pump overspeed that can occur during the recirculation line break accident. Refer to the Fermi 2 docket.

1.8.15 SAFETY GUIDE 15 (REGULATORY GUIDE 1.15), TESTING OF REINFORCING BARS FOR CONCRETE STRUCTURES

This section has been updated since the initial submittal of the DAEC FSAR.

This Regulatory Guide was withdrawn by the NRC in July of 1981.

1.8.16 SAFETY GUIDE 16 (REGULATORY GUIDE 1.16), REPORTING OF OPERATING INFORMATION

This section has been updated since the initial submittal of the DAEC FSAR.

Regulatory Position

The following reporting program should be used to implement the reporting requirements of 10 CFR Parts 20, 40, 50, 70, 73 and reporting required as license conditions including those reports required by the technical specifications.

1. Routine Reports

    a. Operations Reports

        (1) Startup Report

        (2) First Year Operation Report

        (3) Semiannual Operating Reports

    b. Personnel Exposure and Monitoring

        (1) Personnel Exposure and Monitoring Reports

        (2) Personnel Exposure on Termination of Employment

    c. Materials Status

        (1) Special Nuclear Material Status

        (2) Special Nuclear Material Transfer

        (3) Source Material Transfer and Status

2. Non-Routine Reports

    a. Reporting of Abnormal Events

        (1) Abnormal Occurrence Reports

        (2) Reporting of Unusual Events

    b. Radiation Exposure and Monitoring

        (1) Overexposure and Excessive Radiation Levels

    c. Loss of Licensed Material

        (1) Theft or Loss of Licensed Material

        (2) Loss of Special Nuclear Material

(3) Special Nuclear Material Unaccounted for

d. Accidents Involving Licensed Material

(1) Notification of Incidents

(2) Accidental Criticality

3. Special Reports

a. Authorization of Changes, Tests, and Experiments

b. Containment Leak Rate

Response

Iowa Electric intends to comply with the reporting program as stated in the Technical Specifications.

A summary report of plant startup and power escalation testing shall be submitted to the NRC Regional Office following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in UFSAR Section 14.2 and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specified details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operations, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

1.8.17 SAFETY GUIDE 17 (REGULATORY GUIDE 1.17), PROTECTION AGAINST INDUSTRIAL SABOTAGE

This Section has been updated since the initial submittal of the DAEC FSAR.

This regulatory guide was withdrawn by the NRC in July, 1991.

1.8.18 SAFETY GUIDE 18 (REGULATORY GUIDE 1.18), STRUCTURAL ACCEPTANCE TEST FOR CONCRETE PRIMARY REACTOR CONTAINMENTS

Discussion of this guide was not contained in the initial DAEC FSAR. This Regulatory Guide was withdrawn by the NRC in July 1981.

1.8.19 SAFETY GUIDE 19 (REGULATORY GUIDE 1.19), NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS

Discussion of this guide was not contained in the initial DAEC FSAR.

This regulatory guide was withdrawn by the NRC in July 1981.

1.8.20 SAFETY GUIDE 20 (REGULATORY GUIDE 1.20), VIBRATION MEASUREMENTS ON REACTOR INTERNALS

This section has been updated since the initial submittal of the DAEC FSAR.

Regulatory Position

A vibration analysis and test program should be developed. The test program should be submitted for review by the Commission prior to the performance of the scheduled preoperational functional tests.

The vibration testing should be conducted with the fuel elements in the core structure of the reactor internals (or with dummy elements which provide equivalent mass and flow characteristics).

Testing may also be conducted with the core structure not loaded with fuel elements provided such conditions can be demonstrated to result in vibrational characteristics which, for the purposes of the test, will yield conservative results.

The test program should include:

1. A brief description of the vibration test program including instrumentation types and diagrams of their location, which will be used for measurement of vibration responses and those parameters which define the input forcing functions.

2. The planned duration of the test for normal operating modes to assure that all critical components are subjected to at least 10^7 cycles of vibration.

3. The additional test duration for other than normal operating modes to assure that the number of cycles imposed on the critical components is sufficient to analyze their adequacy to withstand vibrations under these operating modes.

4. The description of the different flow modes of operation and transients to which the internals will be subjected during the test.

5. The predominant response mode shapes and the estimated range of numerical values of the response of the major components of the reactor internals in terms of amplitudes and, where appropriate, the anticipated values of the parameters which may influence the input forcing function, under those flow modes of reactor operation which are shown by the analyses to be the most critical.

6. The test acceptance criteria and the permissible deviations from these criteria, and the bases upon which these criteria were established.

7. A description of the inspection program which will be followed after the completion of the vibration tests, including the areas of reactor internals subject to examination, the method of examination, the design access provisions in the reactor internals, and the specialized equipment to be employed for performing such examinations.

Response

The DAEC reactor vessel internals vibration program includes provisions for both prototype and confirmatory vibration tests as suggested in Safety Guide 20. The DAEC is a prototype reactor size as defined by paragraph B of Safety Guide 20, and as such will be provided with a BWR prototype vibrational testing program.

It is anticipated that the DAEC prototype vibration test program will fully conform with the intent of Safety Guide 20 as suggested in paragraph C, Regulatory Position for Prototype Reactor Internals, except for paragraphs C.1.b, C.1.g, and C.4.

A vibration analysis and test program will be finalized and submitted to the AEC for review 2 to 3 weeks prior to the performance of the vibration test. The instrumented hot vibration test will be conducted with fuel elements in the core structure.

Iowa Electric's position on details of the test program is as follows:

1. The vibration test program conducted on the DAEC included measurements of vibratory motions of the shroud assembly, guide tube, and the jet pump assembly, including the jet pump riser brace. Shroud assembly motions were measured by using displacement sensors located on the shroud-to-shroud head flange at positions 180 degrees apart, by using strain gauges mounted on the shroud support legs near the juncture to the shroud at positions of 6, 108, and 168 degrees azimuth, and by using accelerometers mounted on the upper bolt guide ring at the separator assembly at three positions 120 degrees apart. The major forcing parameters for motions of the shroud assembly are considered to be mass flow through the steam separators. The guide tube motions were measured by using strain gauges mounted in pairs (in a horizontal plane at ±45 degrees off radial) near the juncture of the guide tube to vessel bottom head. The major forcing parameter for motions of the guide tubes is considered to be the jet pump diffuser exit velocity, which manifests itself as cross-flow in

the lower plenum region. Motions of the jet pump assembly including the jet pump riser braces were measured by using displacement sensors located on the top of the jet pump assembly (on the "rams head") and on the diffusers of adjacent jet pumps at a position near the slip joint and by using strain gauges mounted on each face of a leg of the jet pump riser brace. The major forcing parameters for motions of the jet pump assembly are considered to be the jet pump nozzle velocity and the direction of mass flow caused by unbalanced recirculation pump speed operation.

2. The significance of 10^7 cycles is not fully understood. The BWR vibration acceptance criteria establish allowable sensor motions for continuous cyclic operation of the reactor for approximately 10^{10} cycles. The vibration tests are intended to ensure that these acceptance criteria will not be violated for normal steady-state or transient modes of plant operation.

3. The conditions for other than normal operating modes are initially measured against the steady-state criteria. If vibration levels meet these criteria, then no duration limits are required. If they do not, as in the case of the [redacted] riser brace, then a fatigue usage analysis is performed to ensure that total transient and steady-state operation over plant life will not consume the acceptable fatigue usage.

4. The following description is representative of the different flow modes of operation and transients to which the internals were subjected during vibration testing:

   a. 50% thermal power line (approximately 20% thermal power to 50% thermal power)

      (1) Approximately equally spaced flow points from minimum flow to 100% flow.

      (2) With 100% core flow trip pump A.

      (3) With 100% core flow trip pump B.

      (4) With 100% core flow trip both pumps simultaneously.

   b. 75% thermal power line

      (1) Approximately equally spaced flow points from minimum flow to 100% flow.

      (2) With 100% core flow trip pump A.

      (3) With 100% core flow trip pump B.

      (4) With 100% core flow trip both pumps simultaneously.

   c. 100% thermal power line

(1) Approximately equally spaced flow points from minimum flow to 100% flow.

(2) With 100% core flow trip pump A.

(3) With 100% core flow trip pump B.

(4) With 100% core flow trip both pumps simultaneously.

5. Iowa Electric complies with the intent of this regulatory position.

6. Iowa Electric complies with the intent of this regulatory position.

7. The preoperational cold flow test and inspection program described in this response is intended to fully satisfy the requirements of AEC Safety Guide 20 regarding the inspection of prototype reactor internals after preoperational flow testing (i.e., paragraphs C.l.g and C.4 of Safety Guide 20). The test and inspection are to be conducted with all core support structures in place. The fuel assemblies may or may not be installed in the core structure during the test depending upon considerations of construction schedule and timing at the DAEC.

   The operating test conditions established for the flow test are listed below:

   Test Point 0 - Both pumps deenergized

   Test Point 1 - Minimum flow in loop A, pump in loop B deenergized

   Test Point 2 - 25% of maximum flow in loop A, pump in loop B deenergized

   Test Point 3 - 50% of maximum flow in loop A, pump in loop B deenergized

   Test Point 4 - 75% of maximum flow in loop A, pump in loop B deenergized

   Test Point 5 - Maximum flow in loop A, pump in loop B deenergized

   Test Point 6 - Maximum flow in loop A, minimum flow in loop B

   Test Point 7 - Maximum flow in loop A, 25% of maximum flow in loop B

   Test Point 8 - Maximum flow in loop A, 50% of maximum flow in loop B

   Test Point 9 - Maximum flow in loop A, 75% of maximum flow in loop B

   Test Point 10 - Maximum flow in both loops

Test Point 11 - Maximum flow in both loops, followed by trip of pump in loop B

Test Point 12 - Maximum flow in both loops, followed by trip of pump in loop A

Test Point 13 - Both pumps deenergized

These operating test conditions ensure that the core structures will be subjected to the normal operating flow modes and other than normal operating flow modes (i.e., pump trips) expected during plant operation. The "maximum flow" listed in the table will be as close to rated flow as possible within the limits of pump cavitation, pump temperature, etc. Because the coolant density is greater in the cold condition, the cold test recirculation mass flow rates are expected to be about 20% higher for equal volumetric flow rates, and therefore the cold flow dynamic forces (i.e., pV^2 forces) should also be 20% higher than during hot flow tests.

Measurements of the vibration motion of selected components will be made during the flow test and compared to motions measured during hot (nuclear) steaming conditions to verify that there is not a significant difference in measured response during the two tests.

Prior to conducting the inspection, the flow conditions will be maintained for sufficient time to accumulate in excess of 10^6 vibration cycles on the core structure components. This provides assurance that the core structure components will be exercised to the cyclic endurance limit as defined by ASME Codes and should fully satisfy the recommendations for test duration suggested by paragraphs C.1.b and C.1.e of Safety Guide 20.

After completion of all flow conditions, accessible areas of the following listed core components will be given a visual inspection on a selected basis to confirm that there are no loose or failed parts. Where there are several similar components in a category (i.e., peripheral control rod drive guide tubes), at least one component in that category will be inspected. For categories having instrumented components, the instrumented component exhibiting the greatest vibration response will be inspected.

1. Peripheral control rod drive and incore guide tubes, housings, and their lower joints.

2. Incore guide tube stabilizer connections and stabilizer bars. Examine plenum region for evidence of loose and/or failed parts.

3. Inside surfaces of the jet pump adapter to shroud support welds and jet pump diffuser to jet pump adapter welds.

4. Liquid control and delta pressure line and bracket welds.

5. The shroud-to-shroud support weld.

6. Jet pump instrument lines and brackets.

7. Jet pump annulus for evidence of loose parts.

8. Jet pump beams, beam bolts, wedges and locator screws.

9. Jet pump riser braces and welds.

10. Shroud head and shroud bolt lug welds.

11. Shroud and shroud head flange locating pins for evidence of deleterious motion marks other than those caused from normal installation.

12. Core support plate bolt keepers.

13. Steam separators and standpipes, shroud head bolt support ring brackets and supports.

14. Feedwater sparger and bracket attachments.

15. Core spray line, brackets, and core spray spargers.

Iowa Electric could see no benefit commensurate with the delay (3 to 7 days) in plant startup or operation to be derived from the removal of reactor vessel internals and their subsequent nondestructive examination following the completion of the vibration testing.

An inspection of the reactor internals following the accumulation of 10^7 vibration cycles would not have provided a meaningful demonstration that the reactor internals of a prototype plant are capable of continued operation for 10^{10} cycles. Such a test and inspection would have simply shown that the reactor internals did not fail as a result of the limited number of accumulated cycles but would imply nothing relative to safe plant operation for 10^{10} cycles. It could not be inferred from this information alone that prototype plant operation could be safely extended beyond the time of the inspection.

The best demonstration that prototype reactor internals are capable of extended safe operation is to provide good quality control during all phases of construction, to instrument at selected important core support areas for vibration measurements, to record the vibration response of these areas during reactor preoperational testing, and to follow-up these measures with periodic inservice inspections during the lifetime of the plant. All of these procedures were followed at the DAEC.

Adequate quality control procedures and inspections were imposed on all component parts of the reactor and reactor internals during all stages of fabrication and handling. In addition, detailed installation, inspection, and testing requirements ensured the structural and functional integrity of the reactor system. Adequate quality control procedures and inspections were imposed during all stages of installation to ensure strict compliance with the specifications and instructions.

An instrumented vibration test, as described in the response to regulatory position (1) above, was conducted on the DAEC using very conservative vibration acceptance criteria to

demonstrate that the DAEC prototype core support structure design is acceptable from the standpoint of measured vibration response.

An inspection program was conducted during the first scheduled refueling outages on the DAEC reactor internals according to the requirements of Chapter 17. This approach to inspection, as compared to the suggestions of Safety Guide 20, was justified since the BWR has no single major load-carrying component that is relied on during normal operations to support a significant portion of the core. The 89 individual control rod guide tubes independently support the 368 fuel elements in the DAEC reactor with no more than four elements supported by any single guide tube.

Periodic examination of the reactor internals during subsequent refueling outages in accordance with the inservice inspection program will provide insurance against any gradual changes in the structural character of the reactor internals that might ultimately have an adverse effect on the safety of the system.

The results of BWR vibration testing programs have shown that very little vibration occurs in the BWR. This is to be expected since flow velocities in most key areas of the reactor are quite low. There was no evidence that the inspection requirements of Safety Guide 20 as applied to prototype BWR plants would materially improve confidence in the continued safe operation of a plant, and, therefore, the benefit of such an inspection on prototype plants was considered to be minimal and unjustified.

Prior to initial plant operation, Iowa Electric was aware of reactor vessel vibration testing and inspection programs planned on reactor vessel internals scheduled to be completed before the completion of the DAEC. Iowa Electric worked with General Electric and closely monitored the results of these programs. Iowa Electric also reviewed the results of these testing programs to ensure that the proposed vibration testing program for the DAEC was indeed adequate.

Regulatory Position

A vibration test program should be implemented during the preoperational functional testing program to measure the response (frequency and amplitudes of vibration, in terms of velocities, accelerations and displacements or strains), of the reactor internals and, where appropriate, the values of those parameters which will define the input forcing functions for the more critical modes of reactor operation. The data obtained by these measurements on reactor internals should be sufficient to verify that the cyclic stresses in the components, as determined by analyses of these data, are within the acceptable design stress limits set forth in the design specifications and applicable code requirements and that the results meet the acceptance criteria of the vibration test program.

Response

Iowa Electric complied with the intent of this regulatory position.

Regulatory Position

The extent of the measurements should be determined, for each individual case, on the basis of the design and configuration of those structural elements of the reactor internals important to safety and their predicted behavior as determined from the vibration analyses used in their design. The type of vibration test instrumentation used, the number of measurements taken, and the distribution of measuring devices within the reactor should be adequate to detect the presence of lateral, vertical, and torsional amplitudes of vibration (e.g., beam, column, and shell modes of vibrations, as applicable to the geometry of the internals) and at sufficient locations to determine the points of predominant maximum vibratory oscillations.

Response

Iowa Electric complied with the intent of this regulatory position.

Regulatory Position

After the reactor internals have been subjected to the significant flow modes expected during service lifetime under normal reactor operation, and other modes of reactor operation, visual and nondestructive surface examinations of reactor internals should be conducted to detect any evidence of the effects of vibrations. These examinations should be conducted preferably following removal of the internals from the reactor vessel.

Where removal is not feasible, the examinations should be performed by means of examination equipment appropriate for in situ examination. The areas examined should include all major load-bearing elements of the reactor internals which are relied upon to retain the core structure in place, the lateral, vertical, and torsional restraints provided within the reactor vessel, those locking and bolting devices whose failure could adversely affect the structural integrity of the internals, and those critical locations on reactor internal components as identified from the vibration analyses.

Response

The response to this regulatory position is found in the response to item 7 in the first regulatory position of this safety guide.

Regulatory Position

In the event either the inspections of reactor internals reveal unacceptable defects or the results of the vibration test program fail to meet the specified acceptance criteria, a report should be prepared and submitted to the Commission for review, which includes an evaluation and a description of the corrective actions planned in order to justify the adequacy of the reactor internals design to withstand the vibrations expected in service.

Response

Iowa Electric complied with the intent of this regulatory position.

Regulatory Position

If the test and examination program is acceptable, a summary of the results obtained from the vibration tests and inspection should be submitted to the Commission after completion of the tests. The summary should include:

1. A description of any differences from the specified vibration test program, instrumentation reading anomalies and instrument failures.

2. A comparison between the measured values of vibration responses including the parameters from which input forcing functions are determined and the predicted values from the analysis (where measurements to determine forcing functions cannot be obtained practically in all areas by means of pressure transducers or other instruments, such values may be estimated from measured responses and from analytical and empirical results). This comparison should be made for those components of the reactor internals for which the acceptance criteria under C.f. have been established with respect to the different modes of vibration.

3. An evaluation of measurements that exceeded acceptable limits or of observation that were unanticipated, and the disposition of such deviations.

Response

Iowa Electric complied with the intent of this regulatory position.

1.8.21 SAFETY GUIDE 21 (REGULATORY GUIDE 1.21), MEASURING AND REPORTING OF EFFLUENTS FROM NUCLEAR POWER PLANTS

Regulatory Position

All normal and potential paths for release of radioactive material during normal reactor operation, including anticipated operational occurrences, should be monitored. The sensitivity and calibration of each monitor should be determined periodically. Measurement and reporting of releases of radioactive material should be conducted in accordance with the following provisions. The frequencies of sampling and analysis given are considered to be minimum. In the event effluent levels approach technical specification limits the frequencies of sampling and analysis should be increased.

1. Noble Gas and Tritium Releases to the Atmosphere

   a. Measurements should be made continuously and station records of the quantity of radioactive gases released should be retained (Reference 1, References are at the end of Section 1.8.21). The isotopic composition of the gases released should be determined as in Section 1.b below. For the period of release, the station records should also indicate the existing meteorological conditions on an hourly basis (i.e., wind speed, wind direction, and atmospheric stability, which are representative of conditions at principal points of release). For some nuclear power plants releasing continuously at low effluent levels, hourly meteorological measurements may only be necessary until meaningful average meteorological parameters are established. Additional measurements should be made during the plant lifetime to confirm these meteorological parameters.

Response

The stack and reactor building ventilation radiation monitoring system provide continuous monitoring of the release rate of radioactive gases. Records of radioactive gaseous releases are retained by the plant in accordance with the requirements of the plant Technical Specifications. A meteorological station is installed at the site as described in Chapter 2 and will indicate existing meteorological conditions that will be representative of conditions at principal points of release. The chart recording will be considered as the record.

Regulatory Position

   b. For reactors which release gases continuously, within one month after the date of initial criticality of the reactor, at least monthly thereafter, and following each refueling, process change or other occurrence which could alter the mixture of radionuclides, an isotopic analysis should be made of a sample of the gaseous activity being released (Ref. 2). This analysis should provide the identity and quantity of the principal radionuclides, except tritium, released each month. The sensitivity (Ref. 3) of the analysis should be such that at least 10 μCi/sec of each nuclide released continuously to the atmosphere is measurable. (See Appendix A, Section II, for a list of suggested nuclides.)

Response

Following each refueling process change or other occurrence that could alter the mixture of radionuclides, an isotopic analysis is made of the sample of the gaseous activity being released. Within the limits of sensitivity, this analysis provides the identity and quantity of the principal radionuclides, except tritium, released each month. Reasonable efforts are made to achieve the suggested sensitivity limits; however, only those radionuclides that can be reliably measured are reported.

Regulatory Position

c. For reactors which release gases intermittently, a representative sample of each release should be analyzed isotopically. The sensitivity (Ref. 3) of the analysis should be such that a concentration of at least 10^{-4}μCi/cc is measurable in waste tanks or containment vessels which are discharged intermittently. (See Appendix A, Section II, for a list of suggested nuclides.)

Response

This regulatory position is not applicable to the DAEC.

Regulatory Position

d. For reactors releasing gases intermittently, tritium in a representative sample of each release should be determined. For reactors releasing gases continuously, the release rate of tritium to the atmosphere should be determined at least quarterly. The sensitivities (Ref. 3) of the analyses should be such that at least 10^{-2} μCi/sec released continuously to the atmosphere and a concentration of 10^{-6} μCi/sec in waste tanks or containment vessels which are discharged intermittently to the atmosphere are measurable.

Response

The release rate of tritium to the atmosphere is determined at least quarterly. Reasonable efforts are made to achieve the suggested sensitivity limits of Safety Guide 21.

Regulatory Position

2. Iodine Releases to the Atmosphere

   For releases which contain or potentially contain iodines, a sample should be drawn continuously through an iodine sampling device. A determination should be made of the quantity of radioiodine released. The device should be analyzed at least weekly for iodine-131. An analysis should also be performed of a weekly sample at least quarterly for the radionuclides I-133 and I-135. The sensitivity (Ref. 3) of the analysis for radioiodines should be such that at least 10^{-4}μCi/sec released continuously to the atmosphere is

measurable in waste tanks or containment vessels which are discharged to the atmosphere intermittently.

Response

Releases that contain or potentially contain radioiodines are sampled by continuously drawing a sample through an iodine sampling device. After a week's collection, the sampling device is allowed to decay to allow short-lived noble gas decay, and then it is analyzed on a gamma spectrometer for radioiodines. Reasonable efforts are made to achieve the suggested sensitivity limits; however, experience to date indicates that, while they can be met for iodine-131 and iodine-135, only those radioiodines that can be reliably measured need be reported.

Regulatory Position

3. Particulate Releases to the Atmosphere

   a. For releases of radioactive material in particulate form, a sample should be drawn continuously through a particulate filter. Measurements should be made on these filters to determine the quantities of nuclides with half-lives greater than 8 days in particulate form that are released to the environment. The sensitivities (Reference 3) of the analyses should be such that at least 10^{-4} μCi/sec of each gamma emitting nuclide (10^{-5} μCi/sec each for gross alpha radioactivity, gross radioactivity (β,γ), (Sr-89 and Sr-90) released continuously to the atmosphere is measurable and at least a concentration of 10^{-10} μCi/cc of each gamma emitting nuclide (10^{-11} μCi/cc each for gross alpha radioactivity, gross radio activity (β,γ), (Sr-89 and Sr-90) is measurable in waste tanks or the containment vessel which are discharged to the environment intermittently.

   b. The particulate filters should be changed and analyzed at least weekly for gross radioactivity (β, γ), (Reference 4) and an analysis for at least the radionuclides Ba-140, La-140, and I-131 should be made.

   c. A monthly composite of the weekly samples should be analyzed for the principal gamma emitting nuclides.

Response

A sample of releases containing radioactive material in particulate form is drawn continuously through a particulate filter. The filter is analyzed as follows.

After a week's collection, the filter is allowed to decay for 48 hr. It is then counted in a sodium iodide well crystal for gross gamma activity, with an end window proportional counter for gross beta activity, and with a gamma spectrometer for nuclide identification. A gamma balance is performed quarterly using the well crystal data and the data obtained from the gamma spectrometer and radiochemical separations. The counting efficiency for the beta counter can then be determined for the mixture. Reasonable efforts are made to achieve the recommended sensitivity limits; however, only nuclides that can be reliably measured are reported.

Regulatory Position

d. Analyses for Sr-89 and Sr-90 should be made on a composite of a month's duration of filters at least quarterly.

Response

At least quarterly, a composite of a month's duration is analyzed for Sr-89 and Sr-90.

Regulatory Position

e. An analysis for gross alpha radioactivity on a sample of a week's duration should be made at least quarterly.

Response

It is recognized that the determination of the counting efficiency for alpha radioactivity on a filter is questionable; therefore, the accuracy of the analyses is questionable. Reasonable efforts are made to analyze the filter for alpha radioactivity.

Regulatory Position

4. Liquid Releases

a. Measurements should be made on a representative sample of each batch (Ref. 5) released and station records retained of the quantity and concentration of radioactive materials and volume of each batch of liquid effluent released and estimates made of the average water flow used to dilute the liquid effluent prior to release from the restricted area.

Response

Measurements are made on a representative sample of each batch released and station records retained of the quantity and concentration of radioactive materials and volume of each batch of liquid effluent released and estimates made of the average water flow used to dilute the liquid effluent prior to release from the restricted area.

Regulatory Position

b. Each batch of liquid effluent released should be analyzed for gross radioactivity (β,γ), (Ref. 4). At least one batch per month should be analyzed for dissolved fission and activation gases. The batch(s) should be typical of average releases of radioactivity. The sensitivities (Ref. 3) of analyses for gross radioactivity (β,γ) and dissolved gas radioactivity should be such that at least concentrations of 10^{-7} μCi/ml and 10^{-5} μCi/ml are measurable, respectively.

Response

Each batch of liquid effluent released is analyzed for gross radioactivity (β,γ). Each batch sample in its original sample form is first scanned with a gamma spectrometer.

Aliquots are then taken for gross gamma counting using a sodium iodide well counter and for gross beta counting using an end window beta counter. Iodine carrier and sufficient Na_2SO_4 are added to the beta sample to adjust the total sample weight to 100 mg. The beta sample or a larger sample is used for gamma spectrum analysis. The beta sample is evaporated to dryness under a heat lamp. The gross counting efficiencies of both the beta and gamma well counters are determined monthly via activity balances using the gamma spectrometer and radiochemically separated fractions. If the gamma spectrum or the beta-to-gamma ratio changes significantly from previous sample spectra, the activity balance is repeated. The counting efficiency for the known fraction is applied to the unknown fraction for reporting radioactivity values. Reasonable efforts are made to achieve the suggested sensitivity limits.

Regulatory Position

c. A weekly proportional composite sample (Ref. 6), including an aliquot of each batch released during a week, should be analyzed for Ba-La-140 and I-131.

Response

Radioactivity values for Ba-La-140 and I-131 are obtained for each batch of liquid radioactivity released.

Regulatory Position

A monthly proportional composite sample (Ref. 6), including an aliquot of each batch released during a month, should be analyzed for the principal gamma emitting fission and activation products. In addition, the sample should be analyzed for tritium, Sr-89, and gross alpha radioactivity. The sensitivities (Ref. 3) of analyses of waste tank liquids should be such that there is a capability of measuring concentrations of 10^{-5} μCi/ml of tritium, 10^{-8} μCi/ml each of Sr-89 and Sr-90, 10^{-7} of gross alpha radioactivity and 5×10^{-7} μCi/ml of gamma emitting radionuclides. (see Appendix A, Section I, for a list of suggested radionuclides for analyses.)

Response

Radioactivity values for the principal gamma emitting fission and activation products are obtained for each batch of liquid radioactivity released. A monthly proportional composite sample, including an aliquot of each batch released during a month, are analyzed for tritium. Since the counting of alphas in liquid waste is subject to considerable error, the alpha content of the waste is assumed to be in the same ratio to Cs-134 as it is in the reactor water.

Reasonable efforts will be made to achieve the suggested sensitivity limits.

Regulatory Position

e. A composite proportional sample, including an aliquot of each batch released during a quarter, should be analyzed for Sr-90.

Response

A composite proportional sample, including an aliquot of each batch released during a quarter, is analyzed for Sr-90 and Sr-89.

Regulatory Position

5. Reporting of Effluent Releases

Data should be reported to the Commission in the form given in Appendix A of this Guide (Ref. 7). Except as noted, effluent data should be summarized on a monthly basis, although in some instances more detailed data may be needed. The need for these additional data to be reported to the Commission will be determined on an individual case basis. Where the majority of the activity is released as batches and where there are less than 3 batches per month, each batch should be reported.

Response

Data are reported to the NRC in the form given in Appendix A of Safety Guide 21. Except as noted, effluent data are summarized on a monthly basis and detailed data given as needed. Most of the activity is released as batches, and where there are less than three batches per month, each batch is reported.

Regulatory Position

a. Gaseous releases

(1) Total radioactivity (in curies) releases of noble and activation gases.

(2) Maximum noble gas release rate during any one-hour period.

(3) Total radioactivity (in curies) released, by nuclide, based on representative isotopic analyses performed.

(4) Percent of technical specification limit.

Response

a. Gaseous releases

(1) Reported as suggested.

(2) Reported as suggested.

(3) Reported as suggested. Only those radionuclides that can be reliably measured are reported.

(4) Reported as suggested.

Regulatory Position

b. Iodine releases

(1) Total (I-131, I-133, I-135) radioactivity (in curies) released.

(2) Total radioactivity (in curies) released, by nuclide, based on representative isotopic analyses performed.

(3) Percent of technical specification limit.

Response

b. Iodine releases

(1) Reported as suggested within the limits of reliability. Only those radioiodines that can be reliably measured are reported.

(2) Reported as suggested within the limits of reliability. Only those radioiodines that can be reliably measured are reported.

(3) Reported as suggested within the limits of reliability.

Regulatory Position

c. Particulate releases

(1) Gross radioactivity (β,γ) released (in curies) excluding background radioactivity.

(2) Gross alpha radioactivity released (in curies) excluding background radioactivity.

(3) Total radioactivity released (in curies) of nuclides with half-lives greater than eight days.

(4) Percent of technical specification limit.

Response

c. Particulate releases

(1) Reported as suggested within the limits of reliability.

(2) Reported as suggested within the limits of reliability.

(3) Reported as suggested within the limits of reliability.

(4) Reported as suggested within the limits of reliability.

Regulatory Position

d. Liquid releases

(1) Gross radioactivity (β,γ) released (in curies) and average concentration released to the unrestricted area.

(2) Total tritium and alpha radioactivity (in curies) released and average concentration released to the unrestricted area.

(3) Total dissolved gas radioactivity (in curies) and average concentration released to the unrestricted area.

(4) Total volume (in liters) of liquid waste released.

(5) Total volume (in liters) of dilution water used prior to release from the restricted area.

(6) The maximum concentration of gross radioactivity (β,γ) released to the unrestricted area (averaged over the period of release).

(7) Total radioactivity (in curies) released, by nuclide, based on representative isotopic analyses performed.

(8) Percent of technical specification limit for total activity released.

Response

d. Liquid releases

(1) Reported as suggested within the limits of reliability.

(2) Reported as suggested within the limits of reliability.

(3) Reported as suggested within the limits of reliability.

(4) Reported as suggested within the limits of reliability.

(5) Reported as suggested within the limits of reliability.

(6) Reported as suggested within the limits of reliability.

(7) Reported as suggested within the limits of reliability.

(8) Reported as suggested within the limits of reliability. The percent of Technical Specifications is reported as

$$\left(\frac{\mu Ci/m1A}{\text{MPC of A}} + \frac{\mu Ci/m1B}{\text{MPC of B}} + \ldots \right) 100$$

References for Safety Guide 21

1. All record retention is in accordance with the requirements of the facility Technical Specifications. Records of all isotopic analysis performed are retained.

2. For those processes or other conditions which are changed frequently, an isotopic analysis should be done following each change until a pattern has been established which can be used to predict the isotopic composition of the reactor effluent.

3. The sensitivity limits given for radioactivity analyses in this guide are based on technical feasibility and on the potential significance in the environment of the quantities released. For some nuclides, lower detection limits may be readily achievable and when nuclides are measured below the stated sensitivity limits they should also be reported. For certain mixtures of nuclides of gamma emitters, it may not be possible to measure nuclides in concentrations near their sensitivity limits when other nuclides are present in the sample in much greater concentrations. Under these circumstances, it may be more appropriate to calculate releases of such radionuclides using observed ratios with those nuclides which are readily measurable. In any event, if a sodium iodide detector is used, gamma emitting nuclides should be identified and measured to the extent that no significant peaks are left after spectrum stripping and that the total residual counts after stripping are less than 5% of the total counts in the original spectrum. If a germanium detector is used, all significant peaks should be identified and measured.

4. Gross radioactivity (β, γ) measurements (i.e. gross beta or gross beta in conjunction with gross gamma measurements) should approximate the total activity in the sample. This comment applies to all gross radioactivity (β, γ) measurements suggested in this guide. The details of such approximations should be explained in each report.

5. For plants also releasing radionuclides continuously (for example, steam generator blowdown and secondary leakage in PWRs), a batch should be considered as that volume of liquid released over a period of not more than one week. Data on quantities of gross radioactivity (β, γ), iodine-131 and tritium should be collected and reported separately.

6. A proportional composite sample is one in which the quantity of liquid added to the composite is proportional to the quantity of liquid in each batch that was released. The composite should represent the average concentration prior to release and by multiplying by the total volume released should represent the quantity of radioactivity released during the compositing period.

7. Estimates of the error associated with each six month total should be reported.

1.8.22 SAFETY GUIDE 22 (REGULATORY GUIDE 1.22), PERIODIC TESTING OF PROTECTION SYSTEM ACTUATION FUNCTIONS

This Section has been updated since the initial submittal of the DAEC FSAR.

Regulatory Position

1. The protection system should be designed to permit periodic testing to extend to and include the actuation devices and actuated equipment.

   a. The periodic tests should duplicate, as closely as practicable, the performance that is required for the actuation devices in the event of an accident.

   b. The protection system and the systems whose operation it initiates should be designed to permit testing of the actuation devices during reactor operation.

Response

The protection system for the DAEC is designed to permit periodic testing to extend to and include the actuation devices and actuated equipment with some minor exceptions as shown in the discussion below. The exceptions will be discussed with the responses to Regulatory Positions 2, 3, and/or 4.

The periodic tests, except as noted in the discussion that follows, duplicate as closely as practicable the performance that is required of the actuation devices in the event of an accident. The periodic tests do not duplicate the environmental conditions under which the actuation devices must work during accident conditions.

The protection system and systems whose operation it initiates are designed to permit the testing of the actuation devices during reactor operation except as noted in the discussion that follows.

The systems of the DAEC that are classified as important to safety as defined by IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," and therefore are considered for discussion with respect to the regulatory positions inherent in Safety Guide 22 are the following:

1. Reactor protection system.

2. Primary containment isolation and nuclear steam supply shutoff system.

3. Emergency core cooling systems (formerly termed core standby cooling systems).

4. Standby gas treatment system.

5. Essential ac electrical systems.*
6. Essential dc electrical systems.
7. Essential heating and ventilating systems.
8. Essential service water systems,

The subsystems that are provided process signals to the reactor protection system or receive actuation signals from it for reactor scram are the following:

1. Neutron monitor system.
2. Reactor vessel instrumentation.
3. Nuclear boiler system.
4. Primary containment pressure switches.
5. Control rod drive system (scram functions only).
6. Nuclear steam supply shutoff system.
7. Main turbine control system.

The subsystems that provide process signals for the automatic initiation of the primary containment isolation and nuclear steam supply shutoff system are as follows:

1. Signal inputs
    a. Reactor vessel instrumentation.
    b. Nuclear boiler system.
    c. Nuclear boiler steam leak detection.
    d. Primary containment pressure switches.
    e. Process radiation monitor (main steam line radiation monitor).
2. Actuations
    a. Automatic closure of isolation valves in primary pressure boundary lines that penetrate the primary containment (group A isolation valves).
    b. Automatic closure of isolation valves in lines that penetrate the primary containment and communicate with the atmosphere inside the primary containment (group B isolation valves).
    c. Automatic closure of reactor building (secondary containment) ventilation ducts.
    d. Automatic initiation of standby gas treatment system.

The subsystems that provide process signals for automatic initiation of the emergency core cooling systems are the following:

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* Includes standby diesel-generators and associated diesel oil systems.

1. Reactor vessel instrumentation.

2. Containment pressure switches.

The essential ac electrical systems are the following:

1. The two essential 4160-V ac buses lA3 and lA4 receive power from two independent reliable offsite sources or from two independent diesel-driven generators located within the plant.

   Each diesel-generator automatically starts and phases onto its essential bus on a loss of offsite power.

2. The essential 480-V loads are fed from the 4160-V buses through suitable redundant feeders, transformers, power (load) centers, motor control centers, and other system components.

3. Two independent 125-V dc systems discussed below provide control power for the 4160-V and 480-V breakers.

The essential dc power systems are the following:

1. Two 125-V independent batteries, each with its own battery charger that takes power from the essential ac system (for control and instrumentation of safety systems).

2. One 250-V battery with battery charger (for HPCI heavy dc loads and other dc loads within the plant).

3. Two separate plus and minus 24-V batteries each with two 24-V chargers (for the nuclear instrumentation and portions of the process radiation monitor).

4. The redundant dc buses, feeders, and other components necessary for complete dc distribution systems for the above systems.

The essential heating and ventilation systems serve the following areas:

1. Emergency switchgear and battery rooms.

2. Standby diesel-generator room.

3. Intake structure and RHR/emergency service water pump rooms.

4. Reactor building, RHR, RCIC, HPCI, and core spray pump rooms.

5. Control room.

The essential service water systems are the following:

1. RHR service water.

2. Emergency service water.

3. Intake structure traveling screens (specific portions).

4. River water supply.

A comprehensive description of the method for periodic tests of the reactor protection system appears in Chapter 7. Five tests are described that can be performed during reactor operation. The set of five tests, all of which can be made during reactor operation, verifies the following:

1st Test: Manual trip actuator test, tests half of the manual scram logic and trip actuators at a time. The total test verifies the ability to deenergize all eight groups of scram pilot valve solenoids without scramming the reactor. (Tests actuation devices but not actuated devices.)

2nd Test: Automatic actuator test, verifies the ability of each automatic logic to deenergize the eight groups of scram pilot valves (actuated devices) without scramming the reactor.

3rd Test: Single rod scram test, verifies the capability of each individual control rod (activated device) to scram when both scram pilot valves for that particular rod are deenergized. The conditions for this test are specified in the Technical Specifications.

4th Test: Includes the calibration of the neutron monitoring system as described in Chapter 7.

This test verifies that the core flux sensors associated with the average power range and intermediate range neutron monitor trip systems are accurately measuring the neutron flux in the reactor core and should therefore produce trip at the intended flux.

Other tests and calibration procedures of the neutron monitoring system are discussed in Chapter 7 for the intermediate range monitor and for the average power range monitor. The latter testing may also be performed with the reactor in operation.

5th Test: This test verifies the electrical independence of the channel circuitry as well as the response of the process-type sensing instruments of the reactor protection system subsystems (other than neutron monitor) by applying a test signal through

instrument calibration taps and observing that single channel logic and actuator trip logic circuits trip (tests actuated devices).

Other testability evaluations are as follows:

A general discussion of the design of the reactor protection system and other safeguard systems in relation to IEEE 279-1971 is found in Section 7.2.1.2.4. This general discussion was originally provided in the response to AEC information Guide 2.7 item 1 (FSAR Appendix M.7 subsection M.3.1, subparagraph 3).

General Electric Report 29-GE/NEDO-10139 referenced therein shows that the general testability of the reactor protection systems meets the requirements of IEEE 279-1971.

The primary containment isolation and NSS shutoff system includes the sensors, channels, switches, and remotely activated valve closing mechanisms associated with the valves, which when closed effect the isolation of the primary containment or reactor vessel, or both. It should be noted that the control systems for group A and B isolation valves that close by automatic action are the principal concern in discussion on testability. However, group C remotely operated isolation valves are included because they add to the operator's ability to effect manual isolation. Testable check valves were also included because they provide the operator with an ability to determine that the check valve disk is free to respond to forward flow however, the “testable” function and the position indication features for the testable check valves have been removed per design change. The primary containment isolation and NSS shutoff system is designed to meet the IEEE Proposed Criteria for Nuclear Power Plant Protection Systems (IEEE 279-1971), and this point is also discussed in the NEDO-10139 report referenced above. The primary containment isolation and NSS shutoff system is fully described in Chapter 7. The testability of its control system as well as the activated devices (valves) are also discussed in Chapter 7

The discussions show that all active components of the containment isolation control system and the containment isolation valves, with the exception of the main steam line radiation sensors, can be tested and calibrated during plant operation.

The reactor vessel level, pressure, and main steam line flow instrumentation can each be checked one at a time by the application of simulated process signals directly to the sensing section of the instrument. These instruments have integrally mounted trip units that provide trip signals for reactor trip and NSS system shutoff. Reactor vessel level instrumentation, high drywell pressure instrumentation, RCIC and HPCI steam line flow and pressure instrumentation, and reactor water cleanup system high differential flow instrumentation that provide trip signals to the primary containment isolation and/or secondary containment (reactor building) isolation system and start standby gas treatment system may be tested by the operation of simulated process signals directly to the sensing section of the instrument. All of the above instruments except the reactor water cleanup system have integrally mounted trip units that provide the associated trips. The latter sensing section (flow transmitters) send analog signals through

signal-conditioning equipment (flow comparators) to remote trip units that trip at some preset analog signal level.

In addition, all of the differential pressure indicating switches have a very fine control vent bleed valve installed on the high and low sides of the pressure sensing body to provide the capability for online testing of the instruments. The switch trip signals may be tested while the instrument is removed from service by venting either side of the pressure body depending on the direction the indicator is required to move for the operation of the trip point under test.

The following is a list of typical functions that have this type of testing capability:

1. Reactor vessel level switches.

2. Main steam line high flow switches.

3. Recirculation pumps differential pressure switches (used for RHR loop selection).

4. Jet pump riser differential pressure switches (used for RHR loop selection).

5. Core spray differential pressure from spray ring to core plate.

6. HPCI steam line high flow switches.

7. HPCI pump flow switches.

8. RCIC steam line high flow switches.

9. RCIC pump flow switches.

10. RHR pump flow switches.

The temperature sensors that provide signals for the leak detection systems for HPCI, RCIC, and reactor water cleanup isolation (as described in Chapter 7) are accessible during reactor operation. These sensors and the isolation circuits that they feed are testable during normal plant operation in accordance with paragraphs 4.9(3) and 4.10 of IEEE 279-1971. Each of these temperature sensors consists of dual thermocouple elements that supply analog signals to a control room vertical board. The analog signal from one thermocouple element drives the temperature switch that feeds the isolation logic while the redundant element is available for comparison testing against the active element. The leak detection temperature sensors are sufficient in quantity so that failed sensors may be jumpered-out as permitted by Technical Specifications and replaced later during plant shutdown.

Each temperature sensing loop in the main steam line temperature sensing system consists of a resistance temperature detector wired to:

1. A remotely-located temperature indicator, and

2. A remotely-located temperature switch (electronic type).

The main steam line low-pressure switches that trip the main steam line isolation valves at low reactor pressure when the reactor is in the "Run" mode are accessible for calibration and test during operation, as they are located outside the shielding wall and are provided with instrument valves that allow isolation for testing purposes.

The main steam line radiation sensors cannot be checked during reactor operation because of the high radiation from N-16 activity with steam.

There are four steam line radiation sensors, and each sensor provides a signal for its own independent radiation monitor/trip unit. Each monitor has a radiation indicator. Two of the monitors may be selected for recording.

The radiation sensors are located in very nearly equal radiation source zones. The indicators all show nearly equal readings and the readings vary in proportion to reactor power level.

Any monitor whose indicated radiation level deviates from the average of the other three at steady-state operation or fails to respond properly whenever the reactor power level is varied will immediately be suspect and subjected to test.

Testing will be done by introducing a simulated current signal into the sensor input terminals to check monitor response. A suspect monitor that responds properly to the simulated test will show that the sensor was the cause of the suspected reading. The Technical Specifications specify the minimum number of operable instrument channels per trip system and the required action if the minimum is not available.

The circuit accommodation for and the method of periodic testing of the emergency core cooling systems and automatic depressurization system control logic and trip logic circuit is described in Chapter 7. The general bases for the design and testability of the emergency core cooling system are also presented in Chapter 7.

The discussions show that all active components of the emergency core cooling systems can be tested and calibrated during plant operation. The active components include the initiating logics, the actuation logics, and the actuators actuating devices as well as valves (actuated devices).

The Technical Specifications specify the required testing of the systems to prove adequate cooling capability during operation and the periodic testing required to verify the operability of the core and containment cooling subsystems.

The injection valves of the low-pressure cooling system (core spray and residual heat removal) are testable for the ability to open and close by the operation of the manual switch at the control panel during any mode of reactor operation. The control circuit is arranged such that

at reactor pressures greater than 450 psig one of the two injection valves in each line must be closed before the other can be exercised by means of the manual switch. The low reactor pressure interlock will not allow both valves to be opened at the same time when the reactor pressure is greater than 450 psig.

The low reactor pressure permissives allow both manual or automatic opening of both valves at the same time at reactor pressures lower than 450 psig. The automatic actuation test will be run once each operating cycle as directed by the Technical Specifications. The valve test will normally be run during the reactor startup at reactor pressure lower than 450 psig, but with the associated pump breakers racked out so that water will not actually be delivered to the vessel.

The pump automatic start test may be conducted at any time reactor pressure is greater than 450 psig with rated flow diverted to the torus through the system flow test lines.

The HPCI system has a steam-turbine-driven pump and may be fully tested for automatic actuation at any time during reactor operation when sufficient steam pressure exists to drive the turbine. The tests are described in the Technical Specifications.

The automatic depressurization system automatic initiation logic channels and trip actuation devices may be individually tested during reactor operation as described in Chapter 7. This portion of the test does not include the actuated device (relief valve). The simulated automatic activation of the relief valve can be tested before reactor startup and may be conducted during plant shutdown as described in Chapter 5. With the reactor at pressure, each relief valve can be tested for its ability to be opened by manual operation of its control switch at the control room panel. Proof that the valve has opened to allow steam flow is indicated by an increase in temperature at the valve outlet as sensed by the thermocouple mounted in the discharge line from the valve and by activation of discharge tailpipe pressure switches.

The remaining safety-related systems listed below can be fully tested during any mode of reactor operation except that safeguard system availability must be considered before the tests may be conducted. One example of the availability requirement is as follows:

The Technical Specifications require that redundant core cooling systems be known to be operational before any particular core cooling system can be bypassed for testing. For example, the failure of a rated load discharge test of the 250-V dc system will render the HPCI system unavailable as described in Chapter 8.

The Technical Specifications allow unavailability of the HPCI system providing that all active components of the remaining core cooling systems are operable. Under the specified conditions, testing of the 250-V dc system can be done during any mode of reactor operation.

1. Essential ac electrical system

    a. Auxiliary power systems, Chapter 8.

    b. Standby ac power, Chapter 8.

   c. Diesel-generator fuel supply system, Chapter 9.

2. Essential dc electrical systems - dc power supply and distribution, Chapter 8.

3. Essential service water system

   a. River water supply, Chapter 9 (includes intake structure and traveling screen).

   b. RHR service water and emergency service water, Chapter 9.

4. Essential heating and ventilating systems - all of the following are discussed in Chapter 9:

   a. Emergency switchgear and battery rooms.

   b. Standby diesel-generator room.

   c. Intake structure and RHR/emergency service water pump rooms.

   d. Reactor building, RHR, RCIC, HPCI, and core spray pump rooms.

   e. Control room.

   f. Standby gas treatment system.

Regulatory Position

2. Acceptable methods of including the actuation devices in the periodic tests of the protection system are:

   a. Testing simultaneously all actuation devices and actuated equipment associated with each redundant protection system output signal.

   b. Testing all actuation devices and actuated equipment individually or in judiciously selected groups.

   c. Preventing the operation of certain actuated equipment during a test of their actuation devices.

   d. Providing the actuated equipment with more than one actuation device and testing individually each actuation device.

Method a. set forth above is the preferable method of including the actuation devices in the periodic tests of the protection system. It shall be noted that the acceptability of each of the four above methods is conditioned by the provisions of Regulatory Positions 3 and 4 below.

Response

1. Reactor protection system

   Description. The reactor protection system has the function to initiate

reactor scram. The basic description of the system is found in Chapter 7. Briefly, the system has two trip systems, A and B.

The basic logic arrangement of the system is illustrated in Chapter 7. Each trip system has three logics. Two of the logics are used to produce automatic trip signals. The remaining logic is used for a manual trip signal. Each of the two logics used for automatic trip signals receives input signals from at least one channel for each monitored variable. Thus, two channels are required for each monitored variable to provide independent inputs to the logics of one trip system. At least four channels for each monitored variable are required for the logics of both trip systems.

Each actuator associated with any one logic provides inputs into each of the actuator logics for the associated trip system. Thus, either of the two automatic logics associated with one trip system can produce a trip system trip. The logic is a one-out-of-two arrangement. To produce a scram, the actuator logics of both trip systems must be tripped. The overall logic of the reactor protection system could be termed one-out-of-two taken twice.

The automatic logics of trip system A are logics Al and A2; the manual logic of trip system A is logic A3. Similarly, the logics for trip system B are logics Bl, B2, and B3. The actuators associated with any particular logics are identified by the logic identity (such as actuators B2) and a letter. Channels are identified by the name of the monitored variable and the logic identity with which the channel is associated (such as reactor vessel high-pressure channel B).

During normal operation, all sensor and trip contacts essential to safety are closed; channels, logics, and actuators are energized. In contrast, however, trip bypass channels consist of normally open contact networks, as does the backup scram circuitry.

There are two scram pilot valve solenoids and two scram valves for each control rod, arranged as shown in Chapter 7. The scram pilot valve is solenoid operated, with the solenoids normally energized. The scram pilot valve controls the air supply to the respective scram valves for each control rod. With either scram pilot valve energized, air pressure holds the scram valves closed. The scram valves control the supply and discharge paths for control rod drive water. One of the scram pilot solenoids for each control rod is controlled by actuator logics A, the other solenoid by actuator logics B. There are two dc solenoid-operated backup scram valves that provide a second means of controlling the air supply to the scram valves for all control rods. The dc solenoid for each backup scram valve is normally deenergized. The backup scram valves are energized (initiate scram) when both trip system A and trip system B are tripped.

Evaluation. The actuated equipment (control rod drives) is thus provided with more than one actuation device, and tests 1, 2, and 5 discussed under response to Regulatory Position 1 provide a test of each individual simulated variable or by manual actuation of one of the two scram buttons without causing the control rods to drive in. This is considered by Regulatory Position 2d to be an acceptable method of testing the actuation devices.

The testing of each rod by the third test "single rod" scram test once each operating cycle is considered to be an acceptable test per Regulatory Position 2b. The latter consideration is further strengthened by the high reliability of the control rod drive system to scram the reactor as discussed in Chapter 4.

The simultaneous testing of all actuation devices and actuated equipment associated with each redundant protection system output signal as suggested by Regulatory Position 2a is not planned because to do so would scram the reactor and seriously reduce unit efficiency as a power-producing unit.

Preventing the operation of the actuated equipment during a test of their actuation devices as suggested by Regulatory Position 2c is possible within the design of the system by the operation of certain manual block valves in the control rod drive units, but is not planned for the reactor protection system because Position 2d accomplishes the same results.

2. Containment isolation and NSS shutoff system

Description. The typical logic arrangement for the main steam line isolation valves is shown in Chapter 7. This logic requires one out of two trips to each solenoid pilot valve and requires the trip of both solenoid pilot valves to close the main isolation valve. The typical logic arrangement for motor-operated isolation valves is a two-out-of-two logic to each isolation valve. This logic requires the trip of two logic channels and the trip of two actuation devices to close each isolation valve.

Motor-operated isolation valves have one-out-of-two-twice initiation logic arrangement for each valve that requires the trip of two trip channels and two actuation devices (relays) to close each isolation valve.

Evaluation. Similar to the reactor protection system, the testability of the trip systems up to and including the actuation devices for all containment isolation valves is considered to meet Regulatory Position 2d.

The main steam line isolation valves can be tested by slowly closing each valve individually to the 90% open position and then allowing it to return to the open position to show that the valve stem is free to move. At reduced power levels, each valve can be individually tested for rapid closure on one steam line at a time by the actuation of its manual control switch at the control room panel. Thus, the testability of the main steam line isolation valves is considered to meet Regulatory Position 2b.

All other containment isolation valves may be safely tested during reactor operation as directed by the Technical Specifications without affecting unit power production efficiency or reactor safety and are thus considered to meet Regulatory Position 2b.

Similar to the reasons set forth in the discussion of the reactor protection system, the tests suggested by Regulatory Positions 2a and 2c are not planned.

3. Reactor emergency core cooling systems.

The testing procedure discussed in response to Regulatory Positions la and lb for the emergency core cooling system conform to one or all of the Regulatory Positions 2a, 2b, 2c, or 2d.

The initiating logic for each core cooling system is a dual-channel one-out-of-two-twice logic, which permits the testing of individual channels without changing the state of the actuated equipment.

4. Essential service systems

The following essential systems were shown to be testable as suggested by Regulatory Position 2a, and, therefore, Regulatory Positions 2b, 2c, and 2d although applicable are not generally planned methods of test:

a. Essential ac electrical system

    (1) Auxiliary power systems.

    (2) Standby ac power.

    (3) Diesel-generator fuel supply system.

b. Essential dc electrical systems - dc power supply and distribution.

c. Essential service water systems - river water supply (includes intake structure and traveling screen).

d. Essential heating and ventilating systems

    (1) Emergency switchgear and battery rooms.

    (2) Standby diesel-generator room.

    (3) Intake structure and RHR/emergency water pump rooms.

    (4) Reactor building, RHR, RCIC, HPCI, and core spray pump rooms.

    (5) Control room.

    (6) Standby gas treatment system.

<u>Regulatory Position</u>

3. Where the ability of a system to respond to a bona fide accident signal is intentionally bypassed for the purpose of performing a test during reactor operation:

    a. Positive means should be provided to prevent expansion of the bypass condition to redundant or diverse systems, and

    b. Each bypass condition should be individually and automatically indicated to the reactor operator in the main control room.

Response

1. Reactor protection system

The design criteria for reactor protection system scram bypass design are given in Chapter 7, as is the evaluation of the actual system design against the criteria.

Scram bypasses are provided with the reactor protection system. The bypasses are the following:

a. Scram discharge volume high-water-level scram trip.

b. Main steam line isolation valve closure scram trip.

c. Turbine stop valve and control valve fast closure scram trip.

d. Neutron monitor bypass.

e. Removal of instruments from service.

The item a, b, and c bypasses are evaluated in Chapter 7. A further evaluation against IEEE 279 is presented below to show general compliance. The evaluation shows that, in general, the procedures rely on administrative control of the operator to prevent expansion of the bypass condition to redundant or diverse systems rather than on positive mechanical or electrical interlocks as seems to be suggested by Regulatory Position 3a. All bypasses are individually and automatically indicated to the operator as suggested by Position 3b, except that the position of instrumentation valves and other test valves is relied on to prevent inadvertent bypass conditions of the latter valves as discussed below:

a. Bypass (a)-scram discharge volume high level

   (1) Channel Bypass or Removal from Operation (IEEE 279, paragraph 4.11) - Individual level switches may be removed from service under administrative control of the operator. Since only one level switch associated with reactor scram is valved out-of-service at any given time, and since the test interval to confirm proper level switch response is relatively short, the protective function is maintained by means of the one level switch in service on one of the trip systems and the two level switches in service on the other trip system. Furthermore, the operator can ascertain that volume is empty before the start of any single level switch test.

   (2) Operating bypasses (IEEE 279, paragraph 4.12) - An operating bypass is provided in the control room for the operator to bypass the trip outputs. The control of this bypass is achieved through administrative means, and its only purpose is to permit the reset of the reactor protection system following reactor scram. The discharge volume high-water-level trip channels are bypassable only in the shutdown and refuel modes of reactor operation. The bypass is manually initiated and must be manually removed to commence control rod withdrawal.

Since the bypass is used for reactor protection system reset after a reactor scram, automatic removal of the bypass is not a meaningful design requirement.

(3) Indication of bypasses (IEEE 279, paragraph 4.13) - The control room operator must exercise administrative control over valving one level switch out-of-service at a time during the periodic test of the trip channel level switches. When the level switch is placed in its tripped condition as a result of the test, the operator is informed of the trip by the discharge volume high-water-level trip annunciator and the trip channel identification logged by the process computer.

The discharge volume high-water-level trip-bypassed annunciator provides the operator with indication that one or more operating bypass channels have been placed into effect.

(4) Access to means for bypassing (IEEE 279, paragraph 4.14) - All instrumentation valves associated with the periodic testing of individual level switches are under administrative control of the control room operator.

The operating bypass switch located on the control room panel is a key-lock switch under direct control of the operator.

b. Bypass (b)-main steam line isolation valve closure.

(1) Channel bypass or removal from operation (IEEE 279, paragraph 4.11) - The eight trip channels of this protective function tabulated in Chapter 7 meet the IEEE 279 design requirement.

The use of valve limit switches makes it impossible for the operator to remove a trip channel from service. Limit switch testing is an integral part of the main steam line isolation valve test.

During normal plant operation, the bypass circuit is not in operation, and its circuitry is in a passive deenergized state. The removal of the bypass capability is permitted during plant operation when the reactor mode switch is in the "Run" mode because the bypass has no effect on plant safety. Under plant conditions (startup mode) where the bypass is operable, one channel may be removed from service for calibration or test purposes without causing a reactor scram or influencing any aspect of reactor safety. This removal from service is accomplished by valving one of the permissive reactor pressure switches out of service at any given time. The reactor pressure switches permit a scram trip whenever the reactor is in the startup mode and the reactor pressure is greater than 850 psig as discussed below.

(2) Operating bypasses (IEEE 279, paragraph 4.12) - An operating bypass is provided for the reactor protection system protective function. One portion of the bypass requires that the reactor pressure be below 850 psig, and the other

portion requires that the reactor system mode switch, which is under the direct control of the operator, be placed in the shutdown, refuel, or startup positions. The only purpose of this bypass is to permit the reactor protection system to be placed in its normal energized state for operation at low-power levels with the main steam line isolation valves not fully open.

Whenever permissive conditions for bypass are not met, such as reactor pressure above 850 psig or the mode switch in the run position, the bypass is automatically removed. Four channels are provided for this bypass to ensure compliance with the IEEE 279 requirements.

(3) Indication of bypasses (IEEE 279, paragraph 4.13) - The bypassed annunciator of the main steam line isolation valve closure trip provides the operator with indication that one or more operating bypass channels have been placed into effect for this reactor protection system protective function.

Whenever one of the four bypass channels is placed in the bypass state, a control room annunciator is initiated. At this time, if the associated protective trip channel has been in its tripped state, the process computer will log the return to normal condition for the trip.

(4) Access to means of bypassing (IEEE 279, paragraph 4.14) - The mode switch is under the direct control of the plant operator. Administrative control is imposed on the valving out of service and return for the reactor pressure switches.

c. Bypass (c)-turbine stop valve and control valve closure.

(1) Channel bypass or removal from operation (IEEE 279, paragraph 4.11) - The eight trip channels meet the design requirement.

Because of the use of the valve limit switches, it is not possible for the operator to remove a trip channel from service. Limit switch testing is an integral part of the turbine stop valve test.

During normal plant operation above 26% of rated power, the bypass circuitry is in its passive, deenergized state. At these conditions, removal for periodic tests is permitted since it has no effect on plant safety. Under plant conditions below 26% of rated power, one bypass channel may be removed from service at a time without initiating protective action or affecting plant safety. This removal from service is accomplished under administrative control of the control room operator personnel.

(2) Operating bypasses (IEEE 279, paragraph 4.12) - An operating bypass is provided for this protective function in that the turbine stop valve trip output will not be operable whenever the turbine is operating at an initial power level below 30% of rated power. The only purpose of this bypass is to permit the reactor

protection system to be placed in its normal energized state for operation at low-power levels with the turbine stop valves not fully open. The design of the four bypass channels is compatible with the design of the reactor protection system and the IEEE 279 criteria.

(3) Indication of bypasses (IEEE 279, paragraph 4.13) - Whenever all four bypass channels are placed in the bypass state, a control room annunciator is initiated.

(4) Access to means for bypassing (IEEE 279, paragraph 4.14) - Under normal operating conditions, all four bypass channels are in operation and will be automatically removed from service as reactor power is increased above the 26% (nominal) setpoint and automatically reinstated as reactor power is reduced below this same setpoint. During the periodic test of each bypass channel, one sensor at a time will be removed from service under administrative control of the control room operator.

d) Bypass (d)-neutron monitor trip bypasses

Chapter 7 shows the means of bypassing the individual intermediate range monitor (IRM) and average power range monitor (APRM) channels.

(1) Channel bypass or removal from operation (IEEE 279, paragraph 4.11) - A sufficient number of IRM channels have been provided to permit any one IRM channel in a given trip system to be manually bypassed and still ensure that the remaining operable IRM channels comply with the IEEE 279 design requirements.

One IRM manual bypass switch has been provided for each reactor protection system trip system. The mechanical characteristics of this switch permit only one of the three IRM channels of that trip system to be bypassed at any time. To accommodate a single failure of this bypass switch, electrical interlocks have also been incorporated into the bypass logic to prevent the bypassing of more than one IRM in that trip system at any time. Consequently, with any IRM bypassed in a given trip system, two IRM channels remain in operation to satisfy the protection system requirements.

In a similar manner, one APRM manual bypass switch has been provided for each reactor protection system trip system to permit one average power range monitor to be bypassed at any time. Mechanical interlocks have been provided with the bypass switch and electrical interlocks have been provided in the bypass circuitry to accommodate the possibility of switch failure. With one average power range monitor bypassed in a given reactor protection system trip system, two APRM trip channels remain in operation to provide the necessary protection for the reactor.

At any time, one intermediate range monitor in the A trip system and one intermediate range monitor in the B trip system may be manually bypassed; similarly one average power range monitor in each trip system may also be manually bypassed. Under these circumstances, the remaining IRM and APRM channels provide protective capability in compliance with IEEE 279.

(2) Operating bypass (IEEE 279, paragraph 4.12) - Automatic bypass capability is not provided for the neutron monitoring system trip channels. Operating bypasses are permitted of any IRM or APRM channel during reactor operation and are accomplished by the manual bypass switches discussed above.

(3) Indication of bypasses (IEEE 279, paragraph 4.13) - When any IRM or APRM instrument channel output to the reactor protection system is bypassed, this fact is indicated by lights on the main control room panels.

(4) Access to means of bypassing (IEEE 279, paragraph 4.14) - Manual bypassing of any IRM or APRM channel is accomplished with control room selector switches under the direct control of the control room operator.

e. Bypass (e)-removal of instruments from service

The valving out of level switches that scram the reactor on low water level in the reactor vessel is an example of this type of bypass. The evaluation below is typical for all others of similar function.

(1) Channel bypass or removal from operation (IEEE 279, paragraph 4.11) - During the periodic test of any one trip channel, the level sensor is valved out of service and returned to service under administrative control procedures. Since only one level switch is valved out of service at any given time during the test interval, protective capability is maintained through the one level switch in service in one trip system and the two level switches in service with the other trip system.

Operating bypasses (IEEE 279, paragraph 4.12) - This design requirement is not applicable to this protective function.

(2) Indication of bypasses (IEEE 279, paragraph 4.13) - During the periodic test, a control room annunciator is initiated whenever the instrument setpoint has been exceeded and the output placed in a tripped condition. The process computer also provides a typed log of the channel identification as the channel is tripped.

(3) Access to means for bypassing (IEEE 279, paragraph 4.14) - During the periodic test, administrative control procedures must be followed to valve one level sensor out of service and subsequently return it to service.

Since no operating bypasses are available for this protective function, this design requirement does not apply.

2. Core cooling systems

   Chapter 7 describes bypasses of certain circuit conditions during tests of all core cooling systems and shows that administrative procedures during the tests prevent bypassing more than one initiation channel at a time.

   A trip channel under test is automatically indicated to the operator in the control room by permanently installed lights on the control room panel and by the control room annunciator. Other possible bypass conditions over which the operator has only administrative control are discussed below.

   These latter bypass conditions are not automatically indicated to the operator in the control room. The operator has administrative control over the test procedure, however, and is aware of the bypass condition during the test.

3. Core spray system

   a. Operating bypasses (IEEE 279, paragraph 4.12) - There are four mechanisms by which a core spray subsystem can be rendered inoperative deliberately. These are the following:

      (1) Manually racking-out a pump breaker or tripping the feeder breakers to an emergency switchgear section.

      (2) Manually opening a valve motor starter feeder breaker at a motor control center.

      (3) Manually shutting off instrument line valves in various specific combinations: An A and a B low-pressure permissive switch line; an A and a B, or a C and a D vessel level switch line plus an A and a B, or a C and a D high drywell pressure line.

      (4) Shutting core spray suction valves.

      All of the above items are to be considered under supervisory control and are not intended to be automatically overcome by accident detection signals. Racking-out a pump breaker is indicated by tagging procedures for out-of-service equipment that is considered to be an adequate and accurate indication of equipment status. Tripping the feeder breaker to an emergency bus will give an alarm.

      Manually opening a breaker or racking-out a starter for a specific valve operator will deenergize its control transformer in the motor starter module, and thus deenergize the valve position lights advising the operator of an off-normal condition. The closing of the core spray suction valves is annunciated, and valve position switches operate the valve-closed position lights on the operating console.

Each of the disabling actions specified above would have to be deliberately initiated and repeated for both core spray subsystems before the core spray function could be impaired.

It is concluded that this design is in conformance with the intent of IEEE 279, paragraph 4.12.

b. Indication of bypasses (IEEE 279, paragraph 4.13) - There are no automatic bypasses of any part of the core spray control system. It is not practicable to monitor all elements of system continuity (including all normally deenergized current-carrying parts) and thus give an indication of inoperability in the control room. This would introduce excessive complexity and could adversely affect reliability or cause inadvertent false operation.

   The racking-out of 4160-V breakers is controlled procedurally and access is limited to authorized personnel. Consequently, this is considered equivalent to removing a valve or pump for maintenance. This is a maintenance procedure that should be administered by tagging the remote breaker control switch located in the control room.

c. Access to means for bypassing (IEEE 279, paragraph 4.14) - Access to switchgear, motor control centers, and instrument valves is under the administrative control of the control room operator.

4. RHR/LPCI system

The bypass conditions discussed for the core spray system are applicable to the RHR/LPCI system, except that one automatic bypass occurs as follows:

The only automatic bypass of the LPCI system is the closure of the LPCI inboard injection valve on an isolation signal during the RHR shutdown cooling mode. See Chapter 7 for the logic provided to produce this trip. The indication of this is provided by an indicating light in the main control room.

It is considered that there is no violation of the intent of IEEE 279, paragraph 4.12.

5. HPCI system

a. Channel bypass or removal from operation (IEEE-279, paragraph 4.11) - The calibration of a sensor that introduces a single instrument channel trip will not cause a protective function without the coincident trip of a second channel. There are no instrument channel bypasses as such in the HPCI system. The removal of a sensor from operation during calibration does not prevent the redundant instrument channel from functioning if accident conditions occur. The removal of an instrument channel from service during calibration will be brief.

b. Operating bypasses (IEEE-279, paragraph 4.12)

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

The controller is in the main control room and therefore under the direct supervision of the control room operator.

It is considered that all of the above are items under supervisory control and are not [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

d. Access to means for bypassing (IEEE 279, paragraph 4.14) - Access to motor control centers and instrument valves is administratively controlled. Access to other means of

bypassing is located in the main control room and therefore under the direct control of the operators.

6. Automatic depressurization subsystem of nuclear boiler system

   a. Channel bypass or removal from operation (IEEE 279, paragraph 4.11) - The calibration of each sensor will introduce a single instrument channel trip. This does not cause a protective action without the coincident trip of a second channel. The removal of an instrument channel from service during calibration will be brief and will not significantly increase the probability of failure to operate. There are no channel bypasses as such in the automatic depressurization system. The removal of a sensor from operation during calibration does not prevent the redundant trip circuit from functioning if accident conditions occur because they will be sensed by the redundant sensors. In addition, the HPCI system is always available as an alternate mechanism of depressurization during the testing of an automatic depressurization system sensor. The manual reset buttons can interrupt the automatic depressurization for a limited time. However, releasing either one of the two reset buttons will allow automatic timing and action to resume. The timers can be locked out preventing automatic depressurization system initiation by placing both reset switches in the override position.

      In view of the above, it is determined that the automatic depressurization system alone can fulfill requirements of paragraph 4.11 of IEEE 279.

   b. Operating bypasses (IEEE 279, paragraph 4.12) - The calibration of each sensor will introduce a single instrument channel trip. This does not cause a protective function without coincident operation of a second channel. The removal of an instrument channel from service during calibration will be brief and in compliance with special provision of IEEE 279, paragraph 4.11 for one-out-of-two times two systems. [redacted]

   c. An alarm in the main control room is annunciated when the automatic depressurization system timers are locked out.

   d. Access to means for bypassing (IEEE 279, paragraph 4.14) - Instrument valves cannot be operated without the permission of the responsible authorized personnel.

      Reset buttons are on the control panel in the main control room. Control power breakers are in dc distribution cabinets that are normally locked and under the control of the operator.

7. Primary containment isolation and NSS shutoff system

    a. Operating bypasses (IEEE 279, paragraph 4.12) - The only bypass in the isolation valve control system of the NSS shutoff system is the main steam line low-pressure bypass, which is imposed by means of the mode switch in the other-than-run mode. The mode switch cannot be left in this position with neutron flux measuring power above 15% of rated power without imposing a scram.

       Therefore, the bypass is considered to be removed in accordance with the intent of IEEE 279, although it is a manual action that removes it rather than an automatic one. In the case of the motor-operated valves, automatic or manual closure can be prevented by shutting off electric power to the motor starters. This action will be indicated in the main control room by indicating lights going out. Both lights will be deenergized because their power supply is taken from the same circuit as the valve motor starter.

       Additionally, several bypasses (defeats) have been installed in the Groups 1, 2, 3, 5, 6, and 7 isolation logics. These bypasses were added to allow overriding some or all of the isolation signals under certain specific conditions (support of Emergency Operating Procedures or Operating Instructions). These bypasses are described in detail in Section 7.3.1.1.1.6. As in other engineered safeguards systems, many of the sensors for process variables operate from instrument lines hooked up, by necessity, with root valves and instrument valves. [redacted]

    b. Indication of bypasses (IEEE 279, paragraph 4.13) - The bypass of the main steam line low-pressure isolation signal is not indicated directly in the control room except by the position of the Mode Switch handle. This switch is key-lockable in each position and is under strict operator control. Its specific bypass functions are a matter of operator training and, as such, do not reasonably need to be brought to the operator's attention each time he places the switch in STARTUP mode. Since the bypass is not removed by any automatic action, it is positively in effect any time the Mode Switch is in position to impose it. The bypass of the main condenser low-vacuum isolation signal is indicated directly in the control room by means of two annunicators, one associated with each trip system.

       Each of the defeats in the Groups 1, 2, 3, 5, 6, and 7 isolation logics is keylocked, and is indicated in the control room by annunciator and/or indicating light.

       [redacted]

[redacted]

c. Access to means for bypassing (IEEE 279, paragraph 4.14) - The Mode Switch is one of the bypasses switch affecting the NSS shutoff control system, and it is centrally located on the operator's main control console and is key-lockable. The four keylock switches for the main condenser low-vacuum bypass are located in the control room back panel area.

All of the defeats in the Groups 1, 2, 3, 5, 6, and 7 isolation logics are located in the control room, and are under the control of plant operators.
The bypass discussion for the NSS shutoff system is applicable to the primary containment isolation system except that automatic bypass of the isolation signal to the following valves exists.

The automatic isolation signals that close the vacuum breakers from reactor building to torus are overridden by the torus differential pressure switches when the torus pressure approaches a vacuum.

8. Essential service systems

The bypass conditions that may disable the standby ac power system are fully discussed in Chapter 8. The specific bypasses in the electrical portion of the system and the diesel-generator fuel supply system, starting air supply system, and other supporting features are also discussed. The bypasses listed in the referenced discussion are shown to be indicated to the operator either by alarm, light, or position of a switch at the control panel.

Offsite power sources that feed the essential ac buses may be effectively automatically bypassed by lockout relays, but at the same time, another source is automatically switched into service. The automatic lockout event is alarmed.

Bypasses that might occur in the power system (heating and ventilating, service water) are sufficiently detected by process instrumentation and alarmed in the control room. The process conditions that might result from bypass conditions are low-flow alarms, low pump or fan differential pressures alarm, pump breaker trip alarm, low intake water level alarm, etc.

The manual bypass conditions of valving-out instruments, racking-out breakers, and pull-to-lock switches are generally applicable to the essential systems listed below:

a. Essential ac electrical system

(1) Auxiliary power systems, Chapter 8.

(2) Standby ac power, Chapter 8.

(3) Diesel-generator fuel supply system, Chapter 9.

b. Essential dc electrical systems - dc power supply and distribution, Chapter 8.

c. Essential service water systems

(1) River water supply, Chapter 9 (includes intake structures and traveling screen).

(2) RHR service water and emergency service water, Chapter 9.

d. Essential heating and ventilating systems

The following areas are discussed in the appropriate chapter of the FSAR:

(1) Emergency switchgear and battery rooms.

(2) Standby diesel-generator room.

(3) Intake structure water pump rooms.

(4) Reactor building, RHR, RCIC, HPCI, and core spray pump rooms.

(5) Control room.

(6) Standby gas treatment system.

Regulatory Position

4. Where actuated equipment is not tested during reactor operation, it should be shown that:

a. There is no practicable system design that would permit operation of the actuated equipment without adversely affecting the safety or operability of the plant.

b. The probability that the protection system will fail to initiate the operation of the actuated equipment is, and can be maintained, acceptably low without testing the actuated equipment during reactor operation.

c. The actuated equipment can be routinely tested when the reactor is shut down.

Response

The response to Regulatory Positions 1 and 2 above shows that wherever possible, actuated equipment is testable during reactor operation.

Where testability during operation was not included, it was shown that such tests would adversely affect the operability of the plant.

The reliability of the protective system of the plant is discussed under the sections of the FSAR safety evaluation for each safety-oriented system.

The discussions show that systems having adequate redundancy and testability are such that the probability of protection system failure to initiate will be maintained acceptably low.

The response to Regulatory Positions 1 and 2 shows that the actuated equipment can be routinely tested when the reactor is shut down and refueling is not in progress.

1.8.23 SAFETY GUIDE 23 (REGULATORY GUIDE 1.23), ONSITE METEOROLOGICAL PROGRAMS

This section has been revised since the initial submittal of the DAEC FSAR to correct the height of the meteorological instruments above the base of the meteorological tower to 33 ft and 156 ft from the previously reported 35 ft and 165 ft and to include a meteorological system upgrade completed in 1984.

<u>Regulatory Position</u>

1. Meteorological Parameters

   To obtain the meteorological Information required for a valid estimate of atmospheric diffusion at a particular site, instrumentation should be provided that is capable of measuring wind direction, wind speed, and ambient air temperature at a minimum of two levels on at least one tower or mast. At sites where there is a potential for fogging or icing due to an increase in atmospheric moisture content caused by plant operation, instrumentation should be provided for measuring the dew point (or humidity) on the tower or mast.

<u>Response</u>

The DAEC onsite meteorology program was initiated January 10, 1971. New redundant instrumentation was added in November 1984. In accordance with the regulatory position in Regulatory Guide 1.97, Revision 2, the meteorological system was designed in accordance with proposed Revision 1 to Regulatory Guide 1.23, September 1980. Instrumentation is provided that is capable of measuring wind direction, wind speed, and ambient air temperature at two levels on the DAEC meteorological tower. Instrumentation is also provided for measuring the dewpoint at the 33-ft level. For a discussion of the instrumentation, refer to Section 2.3.3.

Regulatory Position

2. Siting of Meteorological Instruments

The tower or mast should be sited at approximately the same elevation as finished plant grade and in an area where plant structures will have little or no influence on the meteorological measurements. The lower set of instruments should sense wind speed and direction, temperature, and dew point (where required) at an elevation of 10 meters above the ground, and the upper set should sense wind speed and direction and temperature at the height of release of radioactive material (plant vent height) but should be positioned not less than 30 meters above the lower sensor set. For stack releases, another set of sensors should be located at an elevation such that meteorological conditions at stack height can be represented.

Response

As discussed in Chapter 2, plant grade (base of reactor building) has been raised by fill to a level 12 ft higher than the base of the meteorological tower. This fill does not interfere aerodynamically with flow around the tower wind sensor because of the distances involved (fill is 700 ft from base of tower). It does mean the wind sensor is 21 ft higher than reactor grade, but 33 ft above the base of the tower and the flood plain.

The location of the meteorological tower 1700 ft south-southeast of the reactor building (1125 ft southeast of the offgas stack) ensures that plant structures will have an insignificant effect on the meteorological measurements.

As stated above, the lower set of instruments sense wind speed and direction, temperature, and dewpoint at an elevation of 33 ft above the ground.

The upper set of instruments is positioned at 156 ft above the ground. Although the main stack is 300 ft tall, the use of meteorological measurements at 156 ft is conservative in that average wind speeds at 300 ft would be higher than measured with the result that atmospheric diffusion is underpredicted.

Regulatory Position

3. Data Recorders

Either analog (strip chart) or digital recording of data may be used as a basis for analysis. In lieu of providing redundant digital recorders, digital outputs may be supplemented by strip chart recorders to minimize possible loss of data due to instrument malfunction. Recorders (analog or digital) for wind direction and speed and temperature difference (two temperatures or one temperature difference measurement on a tower or mast) should be located in the reactor control room for use during plant operation.

Response

Observations are averaged and recorded on a digital recorder located in the reactor control room and in the computer disk storage associated with the plume model software.

Regulatory Position

4. Instrument Accuracy

   a. Wind direction accuracy for instantaneous recorded values ±5°.

   b. Wind speed accuracy for time averaged values ±0.5 mph. Starting speed of anemometer <1 mph.

   c. Temperature accuracy for time averaged values ±0.5°C. Temperature difference accuracy from either difference between averaged temperatures or average temperature difference ±0.1°C.

   d. Dew point accuracy for time averaged values ±0.5°C.

Response

   a. Wind direction accuracy is ±5°.

   b. Wind speed accuracy is ±0.5 mph for wind speeds to 25 mph and ±1.0 mph for wind speeds greater than 25 mph.

      The starting speed of the wind speed system is <1 mph.

   c. The air temperature accuracy is ±0.5°C. The accuracy of the temperature difference instrumentation is ±0.12°C for the 37.5-meter height interval between instruments.

   d. The accuracy of the dewpoint instrumentation is ±1.5°C for relative humidity greater than 60% and temperature between -30°C and +30°C and ±2.5°C for conditions outside this range.

Regulatory Position

5. Instrument Maintenance and Servicing Schedule

Meteorological instruments should be inspected and serviced at a frequency which will assure at least a 90% data recovery and which will minimize extended periods of instrument outage. The use of redundant sensors and/or recorders may be another acceptable means of achieving the 90% data recovery goal. The instruments should be calibrated at least semiannually.

Response

The system is designed to provide an overall data recovery of at least 90%. The instruments are calibrated semiannually.

Regulatory Position

6. Data Reduction and Compilation

   a. Wind, temperature, and humidity data should be averaged over a period of at least 15 minutes, once each hour.

   b. The basic reduced data should be compiled into monthly or seasonal and annual joint frequency distributions of wind speed and wind direction by atmospheric stability class. Table 1 gives an example of a suitable format for data compilation and reporting purposes. Similar tables of joint frequency distribution should be prepared for each of the other atmospheric stability classes. Atmospheric stability classes are those defined in NRC codes PAVAN and ARCON 96.

   c. To aid in assessing the impact of plant operation on the environment, joint frequency distribution types of data summaries should be compiled which will permit the description of the frequency and extent of fogging and icing conditions caused by plant operation.

   d. When evaluating the acceptability of a site for a nuclear power plant, because of unique meteorological conditions at the site, it is sometimes necessary or desirable to depart from the meteorological assumptions provided in Safety Guides No. 3 and 4.

      In these cases, when reducing the data, it is necessary to analyze the joint frequency of persistent wind direction, wind speed, and atmospheric stability to determine appropriately conservative atmospheric diffusion factors (X/Q) for time periods over which the release is assumed to occur (up to 30 days).

   e. An analysis of meteorological conditions and atmospheric diffusion factors (X/Q) for accidental and annual average releases of effluents should be provided and the assumptions and calculation procedures described. The probability distributions of X/Q estimates for appropriate time periods should be presented.

Response

   a. Wind temperature and humidity data are averaged over a period of 30 min, once each hour.

b. The basic reduced data are compiled into seasonal and annual joint frequency distributions of wind speed and wind direction by atmospheric stability class in Sections 2, 3, 4, and 5 of the Onsite Meteorological Data Supplement.

   The stability classification correlations for sigma theta shown in Table 2 of Safety Guide 23 are essentially the same as those used in the DAEC data reduction that are shown in Table 1.2 of the Onsite Meteorological Data Supplement.

   With regard to the stability classification correlations for ΔT shown in Table 2 of Safety Guide 23, it is noted that there are several differences when compared to the classification used in the DAEC Onsite Meteorological Data Supplement as shown in Table 1.2 of the supplement.

   In order to evaluate the differences between the classification criteria used by Iowa Electric versus those developed by the AEC in Safety Guide 23, the 1 year of onsite ΔT data and the lower level sigma theta data were processed using both sets of criteria.

c. Joint frequency distributions of onsite humidity data are presented in Chapter 2.

d. Section 1.3 of the DAEC Onsite Meteorological Data Supplement "Comparison of Onsite Diffusion Parameters with AEC Safety Guide" shows that DAEC site meteorology compares favorably with respect to AEC safety guides.

e. An analysis of X/Q values for accidental releases is provided in the referenced portion of the Onsite Meteorological Data Supplement mentioned in item d above and includes all assumptions and calculational procedures used.

   X/Q values for annual average releases of effluents are discussed in Section 14 and tabulated in Sections 12 and 13 of the Onsite Meteorological Data Supplement.

   Annual probability distributions of X/Q values have been developed in Section 1.3 of the Onsite Meteorological Data Supplement.

   In support of conversion to the 10 CFR 50.67 Alternate Source Term, DAEC performed a new meteorological data assessment using two years of data collected from January 1, 1997 through December 31, 1999. This assessment is described in the Reference 4 amendment request. The assessment included use of the PAVAN and ARCON96 codes to derive new X/Q values. In Reference 2 the NRC granted a partial scope amendment approving the use of the alternate source term for the fuel handling accident.

Regulatory Position

7. Special Considerations

At some sites, due to complex flow patterns in nonuniform terrain, additional wind and temperature instrumentation and more comprehensive programs may be necessary. Also, measurements of precipitation and/or solar radiation may be desirable at some locations.

Occasionally the unique diffusion characteristics of a particular site may warrant use of special meteorological instrumentation and/or studies. Proposed studies of this nature should be described in the application for a construction permit.

Response

There are no unique topographical features or unique diffusion characteristics for the DAEC site that would warrant the use of special instrumentation or the conduct of special studies.

Regulatory Position

8. Documentation

The onsite meteorological measurements program should be documented in the safety analysis reports, in accordance with Sections 50.34(a)(1) and 50.34(b)(1) of 10 CFR 50.

Response

The onsite meteorological measurements program for the DAEC is documented in Section 2.3.3 and the Onsite Meteorological Data Supplement.

1.8.24 SAFETY GUIDE 24 (REGULATORY GUIDE 1.24), ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS STORAGE TANK FAILURE

This guide is not applicable to the DAEC.

1.8.25 SAFETY GUIDE 25 (REGULATORY GUIDE 1.25), ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS

This issue is discussed in Chapter 15.

1.8.26 SAFETY GUIDE 26 (REGULATORY GUIDE 1.26), QUALITY GROUP CLASSIFICATIONS AND STANDARDS

Regulatory Position

Group B Quality Standards (Reference 1, References are at the end of Section 1.8.26) should be applied to water- and steam-containing pressure vessels (other than turbines and condensers), storage tanks, piping, pumps, and valves that are either (1) part of the reactor coolant pressure boundary defined in Par. 50.2(v) but excluded from the requirements of Par. 50.55a (Ref. 2) pursuant to footnote 1 of that section, or (2) not part of the reactor coolant pressure boundary but part of:

a. Systems or portions of systems (Ref. 3) that are required for (1) emergency core cooling, (2) postaccident containment heat removal, or (3) postaccident containment atmosphere cleanup.

b. Systems or portions of systems (Ref. 3) that are required for (1) reactor shutdown and (2) residual heat removal.

c. Those portions of the steam systems of boiling water reactors extending from the outermost containment isolation valve up to but not including the turbine stop valves, and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation.

d. Those portions of the steam and feedwater systems of pressurized water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping up to and including the outermost containment isolation valves, and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation.

e. Systems or portions of systems that are connected to the reactor coolant pressure boundary and are not capable of being isolated from the boundary during all modes of normal reactor operation by the two valves, each of which is either normally closed or capable of automatic closure.

Response

The systems that are required for emergency core cooling are the HPCI, LPCI, and core spray systems. The automatic depressurization system serves as a backup to high-pressure coolant injection, and as shown in Chapter 3, the automatic depressurization system valves are group A. Those portions of these systems required for emergency core cooling are group B. The containment cooling system and the shutdown cooling system, which is also used for postaccident residual heat removal, are group B. The postaccident containment cleanup system as referenced in this safety guide is not applicable to BWRS. That portion of the feedwater system (inside the second isolation valve) that is necessary for shutdown cooling is group A.

The main steam piping extending from the outermost containment isolation valve up to the main steam turbine stop valves is group D plus QA. The quality provisions applied to this piping are described in Chapter 5. This piping met the plant design and industry code requirements in effect at the time of its purchase. It also complies with ANS-22, "Nuclear Safety Criteria for the Design of Stationary BWR Plants." The piping extending from the outermost isolation valve up to the HPCI and RCIC turbine stop valves is group B. For the DAEC, there are no systems that are connected to the reactor coolant pressure boundary and are not capable of being isolated from the boundary during all modes of normal reactor operation by the two valves, each of which is either normally closed or capable of automatic closure. The closure of check valves is discussed in Chapter 6.

Regulatory Position

2. Group C quality standards (Ref. 1) should be applied to water, steam, and radioactive waste-containing pressure vessels (other than turbines and condensers), storage tanks, piping, pumps, and valves, not part of the reactor coolant pressure boundary nor included in quality Group B but part of:

    a. Cooling water and auxiliary feedwater systems or portions of these systems (Ref. 3) that are required for (1) emergency core cooling, (2) post-accident containment heat removal, (3) post-accident containment atmosphere cleanup, and (4) residual heat removal from the reactor and from the spent fuel storage pool. Portions of these systems required for their safety functions that do not operate during any mode of normal reactor operation or cannot be tested adequately should be classified as Group B.

    b. Cooling water and seal water systems or portions of these systems (Ref. 3) that are required for functioning of reactor coolant system components important to safety, such as reactor coolant pumps.

    c. Systems or portions of systems that are connected to the reactor coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure (Ref. 4).

    d. Radioactive waste treatment, handling and disposal systems, and other systems that contain or may contain radioactive material. Components in these systems (except those included in Regulatory Positions 2.a., 2.b., and 2.c.) whose failure would not result in calculated potential exposures in excess of 0.17 rem whole body (or its equivalent to parts of the body) at the site boundary or beyond may be classified as Group D.

Response

The cooling water piping to the RHR and core spray pumps, which are required for emergency core cooling, is group D plus QA. This piping met the plant design and industry code requirements in effect at the time of purchase. Portions of the RHR service water system are group D, but those portions that are required for postaccident containment heat removal and for

residual heat removal from the reactor are group C. There are no cooling water or auxiliary feedwater systems required for postaccident containment atmosphere cleanup on a BWR. The piping in the fuel pool cooling and cleanup system, which removes residual heat from the spent-fuel storage pool, is group C. Fuel pool equipment, such as valves, pumps, heat exchangers, filter-demineralizer, and tanks, is classified as either group C or group D, depending on the system design classification requirements in effect at the time the given equipment items were purchased.

The reactor building closed cooling water system does not supply water to any component important to safety.

All systems or portions of systems that are connected to the reactor coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure, are group B or group C.

Components in radioactive waste treatment, handling and disposal systems, and other systems that contain or may contain radioactive material (except those included in Regulatory Positions 2.a and 2.c) are classified as group C and group D. A failure of these components would not result in calculated potential exposures in excess of 0.17 rem whole body. Chapter 11 presents an evaluation of the nonmechanistic failure of all liquid radwaste surge, sample, and storage tanks within the radwaste building together with the instantaneous release of their contents to the soil. The resulting calculated total fraction of maximum permissible concentration at the Cedar Rapids municipal water intake was 0.04.

Similarly, the nonmechanistic failure of all charcoal delay tanks and the instantaneous loss of the complete equilibrium accumulation of noble gas and halogen isotopes were evaluated. The resulting whole-body and thyroid doses were 0.050 and 0.5 rem, respectively.

Since the above evaluations were carried out on the basis of total nonmechanistic failure of all components, it is clear that the Safety Guide 26 criteria of 0.17-rem whole body (or its equivalent to parts of the body) are met by a wide margin for individual components within the radwaste system and hence may be classified as group D.

<u>Regulatory Position</u>

3. Group D quality standards (Ref. 1) should be applied to water-and-steam-containing components not part of the reactor coolant pressure boundary nor included in quality Group B or C but part of systems or portions of systems that contain or may contain radioactive material.

<u>Response</u>

Water- and steam-containing components that are not part of the reactor coolant pressure boundary and are not included in quality groups B or C but are part of systems or portions of systems that contain or may contain radioactive material are shown in Chapter 3.

Because the code requirements have changed several times during the purchase of piping and components, the codes to which the piping and components were actually purchased are shown in Chapter 3.

References for Safety Guide 26

1. Quality standards applicable to this classification group are given in Table 1.8-3.

2. Group A quality standards that are required for pressure-containing components that are part of the reactor coolant pressure boundary are specified in Section 50.55a of 10 CFR 50.

3. The system boundary includes those portions of the system required to accomplish the specified safety function and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure when the safety function is required.

4. Components in influent lines may be classified as Group D provided they are capable of being isolated from the reactor coolant pressure boundary by an additional valve which has high leak-tight integrity.

1.8.27 SAFETY GUIDE 27 (REGULATORY GUIDE 1.27), ULTIMATE HEAT SINK

This Section has been updated since the original issue of the DAEC FSAR.

<u>Regulatory Position</u>

1. The ultimate heat sink should be capable of providing sufficient cooling for at least 30 days (a) to permit simultaneous safe shutdown and cooldown of all nuclear reactor units that it serves, and maintain them in a safe shutdown condition, and (b) in the event of an accident in one unit, to permit control of that accident safely and permit simultaneous safe shutdown and cooldown of the remaining units and maintain them in a safe shutdown condition. Procedures for assuring a continued capability after 30 days should be available.

<u>Response</u>

As is discussed in DAEC PSAR Amendment 16, the plant requirement for emergency cooling water is about 13 cfs. The extreme value extrapolated 1000-year instantaneous minimum flow in the Cedar River is 60 cfs. Since the data used to develop this minimum were biased by a brief, extremely low point (53 cfs) resulting from the operation of a now-abandoned run-of-river hydroelectric plant, it is concluded that the data are additionally conservative. Therefore, 60 cfs has been selected as the minimum design criterion of the intake structure for the DAEC. This resulting 60 cfs minimum design criterion is more than four times the 13 cfs required for emergency cooling. The intake has been designed to ensure that the required flow will reach the pump. A water depth of at least 12 in. at the intake is maintained by the Technical
2018-004 Specifications to ensure that the required emergency cooling flows are available (Reference 6). Additionally, the 1000-yr instantaneous low flow would still provide substantial margin beyond emergency cooling requirements, it is concluded that the Cedar River is capable of providing sufficient cooling for an indefinite period of time to permit the control of an accident safely. Accordingly, procedures for ensuring a continued capability after 30 days are not applicable.

<u>Regulatory Position</u>

2. The ultimate heat sink should be capable of withstanding the effects of the most severe natural phenomena associated with its locations, other applicable site related events, reasonably probable combinations of less severe phenomena or events where this is appropriate to provide a consistent level of conservatism, and a single failure of man-made structural features without loss of the capability specified in Regulatory Position 1 above.

<u>Response</u>

As discussed in DAEC PSAR Amendment 16, the formation of an ice jam may cause a temporary blockage in the channel, reducing flows downstream significantly. Ice breakings and resulting jams, however, are caused by an increase in flow, so that it is not possible for a jam to occur at a minimum flow.

Temporary ice dams are never watertight and some flow always passes through them. Furthermore, the temporary storage behind them will fill in a very short time, with the result that either the ice dam is overtopped and spills, or it is washed out by the increased head.

It is not considered possible for critical low flow to result from this cause for more than a few minutes. The most likely cause of low flow would be a prolonged drought. The greatest drought known was that of the decade of the 1930s. The lowest flow in the Cedar River during this drought was a brief 178 cfs on September 25, 1935 (Average flow for the day was 833 cfs.) It is likely that this interruption also resulted from regulation by the hydroelectric plant dam. The drought of the 1930s is generally considered to have a recurrence interval of 200 to 300 years. It is evident that a much more severe drought would be required to produce critical low flows in the Cedar River.

With regard to the effect of an earthquake on the flow in the Cedar River, historically, earthquakes have resulted in only minimal fluctuations of river level and thus only minor variations in river flow. An exception to this effect has occurred when high banks have fallen into the river and caused flow blockage. The banks of the Cedar River are extremely low, and in the relatively wide, flat flood plain there are no topographic features that could significantly change the river flow in the event of an earthquake. Thus, the Cedar River should be capable of withstanding the effects of an earthquake and still supply sufficient cooling water as described in the response to Regulatory Position 1 above. Chapter 2 describes in detail the site topography.

As discussed in DAEC PSAR Amendment 16, there are no man-made structural features that could result in a loss of the river as a heat sink.

The intake structure itself is designed as a Seismic Category I structure in order to ensure
2015-005 its integrity following a design-basis seismic event. As described in Chapter 9, the river water supply piping from the intake structure to the pump house, and that portion of the pump house containing the emergency service water pumps and the RHR service water pumps are designed to Seismic Category I criteria.

2015-005 Calculations have been performed which demonstrate that the failure of the sediment management system called "Iowa Vanes," including the steel sheet guide wall, installed adjacent to the intake structure, would have no adverse impact on the intake structure and river water
2018-004 supply system (References 7 and 8).

<u>Regulatory Position</u>

3. The ultimate heat sink should consist of at least two sources of water, including their retaining structures, each with the capability to perform the safety functions specified in Regulatory Position 1 above unless it can be demonstrated that there is an extremely low probability of losing the capability of a single source. There should be at least two canals or conduits connecting the source(s) with the intake structures of the nuclear power units, unless it can be demonstrated that there is an extremely low probability that a single canal can fail entirely from natural phenomena. All water sources and their associated canals or

conduits should be highly reliable and should be separated and protected such that failure of any one will not induce failure of any other.

Response

As discussed in the responses to Regulatory Positions 1 and 2 above, there is an extremely low probability that the Cedar River will ever fail to provide sufficient amounts of cooling water.

Regulatory Position

4. The technical specifications for the plant should include actions to be taken in the event that conditions threaten partial loss of the capability of the ultimate heat sink or it temporarily does not satisfy Regulatory Positions 1 and 3 above during operation.

Response

Because of the extremely low probability that the flow in the Cedar River will ever drop below that required to provide sufficient emergency cooling water flow, no technical specification on Cedar River flow is deemed necessary.

1.8.28 SAFETY GUIDE 28 (REGULATORY GUIDE 1.28), QUALITY ASSURANCE PROGRAM REQUIREMENTS

NOTE: This section was applicable to the design and construction phase, and is for historical purposes only.

Regulatory Position

The general requirements and guidelines for establishing and executing a quality assurance program during the design and construction phases of nuclear power plants, which are included in ANSI N45.2-1971, "Quality Assurance Program Requirements for Nuclear Power Plants" are generally acceptable and provide an adequate basis for complying with the program requirements of Appendix B to 10 CFR 50.

Response

ANSI N45.2-1971 is a more inclusive and also a more general document than Appendix B to 10 CFR 50 to which the DAEC program is patterned. ANSI N45.2 identifies 19 sections, while the DAEC program contains 18 identified sections. The extra section in ANSI N45.2 is an introduction containing items of scope, responsibility, and definition that are covered in a separate introduction and/or in the various sections throughout the program.

In the following discussion, the ANSI N45.2 section identification is shown first and then the DAEC program (Appendix D) section numbers. This discussion is limited to the compliance of the DAEC program for safety-related components and systems.

1. Introduction, D.1 No exact corresponding section. See above comments.

2. QA Program, D.2.2. The DAEC program complies with these requirements.

3. Organization, D.2.1. The DAEC program complies with these requirements.

4. Design Control, D.2.3. The DAEC program complies with these requirements.

5. Procurement Document Control, D.2.4. The DAEC program complies with these requirements.

6. Instructions, Procedures, & Drawings, D.2.5. The DAEC program complies with these requirements.

7. Document Control, D.2.6. The DAEC program complies with these requirements.

8. Control of Purchased Material, Equipment, and Service, D.2.7. The DAEC program equals or exceeds these requirements. Exceeded areas are those where source evaluation, source inspection, and documentary evidence are concerned.

9. Identification and Control of Material, Parts, and Components, D.2.8. The DAEC program complies with these requirements.

10. Control of Special Processes, D.2.9. The DAEC program complies with these requirements.

11. Inspection, D.2.10. The DAEC program complies with these requirements.

12. Test Control, D.2.11. The DAEC program complies with these requirements.

13. Control of Measuring and Test Equipment, D.2.12. The DAEC program complies with these requirements except in the area of the evaluation of the validity of previous inspections and/or tests due to measuring and test equipment being found out of calibration.

14. Handling Storage and Shipping, D.2.13. The DAEC program complies with these requirements.

15. Inspection, Test, and Operating Status, D.2.14. The DAEC program complies with these requirements.

16. Nonconforming Item, D.2.15. The DAEC program complies with these requirements.

17. Corrective Action, D.2.16. The DAEC program complies with these requirements.

18. QA Records, D.2.17. The DAEC program complies with these requirements.

19. Audits, D.2.18. The DAEC program complies with these requirements.

1.8.29 SAFETY GUIDE 29 (REGULATORY GUIDE 1.29), SEISMIC DESIGN CLASSIFICATION

Regulatory Position

1. The following structures, systems, and components of a nuclear power plant, including their foundations and supports, are designated as Seismic Category I and should be designed to withstand the effects of the DBE and remain functional. (The system boundary includes those portions of the system required to accomplish the specified safety function and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure when the safety function is required.)

   a. The reactor coolant pressure boundary.

   b. The reactor core and reactor vessel internals.

   c. Systems or portions of systems that are required for (1) emergency core cooling, (2) post-accident containment heat removal, or (3) post-accident containment atmosphere cleanup (e.g., hydrogen removal system).

   d. Systems or portions of systems that are required for (1) reactor shutdown, (2) residual heat removal, and (3) cooling the spent fuel storage pool.

   e. Those portions of the steam systems of boiling water reactors extending from the outermost containment isolation valve up to but not including the turbine stop valve, and connected piping of 2-1/2 inches or larger nominal pipe size up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation. The turbine stop valve should be designed to withstand the SSE and maintain its integrity.

   f. Those portions of the steam and feedwater systems of pressurized water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping of 2-1/2 inches or larger nominal pipe size up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation.

   g. Cooling water, component cooling, and auxiliary feedwater systems or portions of these systems that are required for (1) emergency core cooling, (2) post-accident containment heat removal, (3) post-accident containment atmosphere cleanup, (4) residual heat removal from the reactor, and (5) cooling the spent fuel storage pool.

   h. Cooling water and seal water systems or portions of these systems that are required for functioning of reactor coolant system components important to safety, such as reactor coolant pumps.

i. Radioactive waste treatment, handling and disposal systems except those portions of these systems whose postulated simultaneous failure would not result in conservatively calculated potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

j. Systems or portions of systems that are required to supply fuel for emergency equipment.

k. Systems or portions of systems that are required for monitoring and actuation of systems important to safety.

l. The protection system.

m. The spent fuel storage pool structure, including the fuel racks.

n. The reactivity control systems; e.g., control rods, control rod drives, and boron injection system.

o. The control room, including its associated vital equipment and life support systems, and any structures or equipment inside or outside of the control room whose failure could result in incapacitating injury to the operators.

p. Primary and secondary reactor containment.

q. Portions of the onsite electrical power system, including the onsite electrical power sources, that provide the emergency electrical power needed for functioning of plant features included in Items 1.a through 1.p above.

r. Structures, systems, or components whose failure could reduce the functioning of any plant feature included in Items 1.a through 1.q above to an unacceptable safety level.

Response

Those portions of the systems that lie within the definition of Seismic Category I as defined in Chapter 3 have been appropriately designed for seismic response and protected as necessary to withstand the effects of the safe-shutdown earthquake and remain functional.

All of the structures, systems, portions of systems, and components listed in item a through r of this regulatory position are included in Chapter 3, with the exception of items d(3), f, g(3), g(5), h, and i. The turbine stop valves are not designed to Seismic Category I requirements.

The systems or portions of systems listed in items d(3), g(5) and h met the plant design requirements that were based on applicable industry codes in effect at the time of their purchase. The design of the fuel pool cooling and cleanup system is described in detail in Chapter 9. Item

f does not apply to BWRs and item g(3) does not apply to the DAEC since the postaccident containment atmosphere cleanup system does not require cooling water.

As discussed in Chapter 11, simultaneous failure of all portions of the radioactive waste treatment handling and disposal systems that are located in the Nonseismic radwaste building did not result in conservatively calculated potential offsite exposures that would exceed the guideline values of 10 CFR 100.

The P&IDs for each particular system show which portions of those systems are Seismic Category I as discussed in Chapter 3.

Regulatory Position

2. Category I design requirements should extend to the first seismic restraint beyond the defined boundaries. Structures, systems, or components which form interfaces between Seismic Category I and Nonseismic features should be designed to Seismic Category I requirements.

Response

Seismic Category I design requirements at the DAEC do extend to the first seismic restraint beyond the defined boundaries. Structures, systems, or components that form interfaces between Seismic Category I and Nonseismic features are designed to Category I requirements.

1.8.30 REGULATORY GUIDE 1.54, "QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER-COOLED NUCLEAR POWER PLANTS"

Comments and Clarifications:

The Company is not committed to Regulatory Guide 1.54, June 1973. The Company's controls relative to protective coatings are described below.

Special Protective Coatings (Paint):

The application of a special protective coating shall be controlled as a special process when the failure (i.e. peeling or spalling) of the coating to adhere to the substrate can cause the malfunction of a safety-related, important to safety, or selected other structure, system or component. Special process coatings shall be applied by qualified personnel using qualified materials and equipment, and approved procedures. Documentation shall include identification of the following:

1. person applying the coating (and qualification),

2. material used,

3. procedure used (and qualifying procedure if different),

4. tests performed and results,

5. date of application of coating, and

6. traceability of coating location.

1.8.31 REGULATORY GUIDE 1.61, "DAMPING VALUES FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS", OF OCTOBER 1973

Regulatory Position

This regulatory guide delineates damping values acceptable to the AEC regulatory staff to be used in the elastic model dynamic seismic analysis of seismic Category I structures, systems and components.

Response:

The NextEra Energy Duane Arnold commitment to Regulatory Guide 1.61 of October 1973 is limited to the design of PaR spent fuel storage racks only.

1.8.32 REGULATORY GUIDE 1.105, “INSTRUMENT SETPOINTS FOR SAFETY-RELATED SYSTEMS” REVISION 2, FEBRUARY 1986

Regulatory Position

ISA-S67.04-1982, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants," establishes requirements acceptable to the NRC staff for ensuring that instrument setpoints in safety-related systems are initially within and remain within the technical specification limits. The last section of ISA-S67.04-1982 lists additional standards that are referenced in other sections of the standard. Those referenced standards not endorsed by a regulatory guide (or incorporated into the regulations) also contain valuable information and, if used, should be used in a manner consistent with current regulations.

Response

The DAEC conforms to this regulatory position for those instruments with setpoints in the Technical Specifications. It may be used for other instrumentation on a case-by-case basis. The DAEC Setpoint Control Program uses the General Electric (GE) Instrument Setpoint Methodology; NEDC-31336, “General Electric Instrumentation Setpoint Methodology,” which meets this ISA standard. The GE Setpoint Methodology has been given NRC-approval as documented in a Revision to the Safety Evaluation Report transmitted by letter from B. Boger (NRC) to R. Pinelli (BWROG) dated November 6, 1995.

1.8.33 REGULATORY GUIDE 1.143, "DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT WATER-COOLED NUCLEAR POWER PLANTS", REVISION 1, OCTOBER 1979

Regulatory Position C.1, "Systems Handling Radioactive Materials In Liquids"

1.1 The liquid radwaste treatment system, including the steam generator blow down system, downstream of the outermost containment isolation valve should meet the following criteria:

1.1.1 The systems should be designed and tested to requirements set forth in the codes and standards listed in Table 1 supplemented by regulatory positions 1.1.2 and 4 of this guide.

1.1.2 Materials for pressure-retaining components should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical and radioactive environment of specific applications during normal conditions and anticipated operational occurrences. Manufacturers' material certificates of compliance with material specifications such as those contained in the codes referenced in Table 1 may be provided in lieu of certified material test reports.

1.1.3 Foundations and walls of structures that house the liquid radwaste system should be designed to the seismic criteria described in regulatory position 5 of this guide to a height sufficient to contain the maximum liquid inventory expected to be in the building.

1.1.4 Equipment and components used to collect, process, and store liquid radioactive waste need not be designed to the seismic criteria given in regulatory position 5 of this guide.

1.2 All tanks located outside reactor containment and containing radioactive materials in liquids should be designed to prevent uncontrolled releases of radioactive materials due to spillage in buildings or from outdoor tanks. The following design features should be included for such tanks:

1.2.1 All tanks inside and outside the plant, including the condensate storage tanks, should have provisions to monitor liquid levels. Designated high-liquid-level conditions should actuate alarms both locally and in the control room.

1.2.2 All tank overflows, drains, and sample lines should be routed to the liquid radwaste treatment system.

(Retention by an intermediate sump or drain tank design for radioactive materials and having provisions for routing to the liquid radwaste system is acceptable.)

1.2.3 Indoor tanks should have curbs or elevated thresholds with floor drains routed to the liquid radwaste treatment system.

(Retention by an intermediate sump or drain tank design for radioactive materials and having provisions for routing to the liquid radwaste system is acceptable.)

1.2.4 The design should include provisions to prevent leakage from entering unmonitored and nonradioactive systems and ductwork in the area.

1.2.5 Outdoor tanks should have a dike or retention pond capable of preventing runoff in the event of a tank overflow and should have provisions for sampling collected liquids and routing them to the liquid radwaste treatment system.

Response:

NextEra Energy Duane Arnold commits to the content of Regulatory Guide 1.143, Revision 1, October 1979, Position C.1 with the following exceptions:

1.1.1 The liquid radwaste treatment system codes and standards for design, materials, fabrication, inspection and testing are defined in UFSAR Section 3.2.

1.1.2 Materials for pressure retaining components should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, or ASTM material specification, consistent with the codes and standards defined in Section 3.2 of the UFSAR. Manufacturers' material certificates of compliance with material specifications such as those contained in the codes referenced in UFSAR Section 3.2 may be provided in lieu of certified material test reports. As an alternative to the Manufacturers' material certificates of compliance, material compliance to the applicable codes and standards may be verified by independent testing or other verifiable methods.

1.1.3 The NextEra Energy Duane Arnold commitment for seismic design criteria is limited to the seismic methods used in the original radwaste facility construction and not upgraded to Regulatory Guide 1.143 standards. Refer to Regulatory Position C.5 below.

1.1.4 thru 1.2.5 - No exceptions.

<u>Regulatory Position C.2</u>, "Gaseous Radwaste Systems"

2.1 The gaseous radwaste treatment system should meet the following criteria: (For a BWR this includes the system provided for treatment of normal offgas releases from the main condenser vacuum system beginning at the point of discharge from the condenser air removal equipment.)

2.1.1 The systems should be designed and tested to requirements set forth in the codes and standards listed in Table 1 supplemented by regulatory positions 2.1.2 and 4 of this guide.

2.1.2 Materials for pressure-retaining components should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical, and radioactive environment of specific applications during normal conditions and anticipated operational occurrences. If the potential for an explosive mixture of hydrogen and oxygen exists, adequate provisions should be made to preclude buildup of explosive mixtures, or the system should be designed to withstand the effects of an explosion. Manufacturers' material certificates of compliance with material specifications such as those contained in the codes reference in Table 1 may be provided in lieu of certified materials test reports.

2.1.3 Those portions of the gaseous radwaste treatment system that are intended to store or delay the release of gaseous radioactive waste, including portions of structures housing these systems, should be designed to the seismic design criteria given in regulatory position 5 of this guide. For the systems that normally operate at pressures above 1.5 atmospheres (absolute), these criteria should apply to isolation valves, equipment, interconnecting piping, and components located between the upstream and downstream valves used to isolate these components from the rest of the system (e.g. waste gas storage tanks in the PWR) and to the building housing this equipment. For systems that operate near ambient pressure and retain gases on charcoal adsorbers, these criteria should apply to the tank support elements (e.g., charcoal delay tanks in a BWR) and the building housing the tanks.

<u>Response:</u>

NextEra Energy Duane Arnold commits to the content of Regulatory Guide 1.143, Revision 1, October 1979, Position C.2 with the following exceptions:

2.1.1 Refer to response 1.1.1 above.

2.1.2 Refer to response 1.1.2 above.

2.1.3 Refer to response 1.1.3 above.

Regulatory Position C.3, "Solid Radwaste Systems"

3.1 The solid radwaste system consists of slurry waste collection and settling tanks, spent resin storage tanks, phase separators, and components and subsystems used to solidify radwastes prior to offsite shipment. The solid radwaste handling and treatment system should meet the following criteria:

3.1.1 The system should be designed and tested to the requirements set forth in the codes and standards listed in Table 1 supplemented by regulatory positions 3.1.2 and 4 of this guide.

3.1.2 Materials for pressure-retaining components should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical, and radioactive environment of specific applications during normal conditions and anticipated operational occurrences. Manufactures' material certificates of compliance with material specifications such as those contained in the codes referenced in Table 1 may be provided in lieu of certified materials test reports.

3.1.3 Foundations and adjacent walls of structures that house the solid radwaste system should be designed to the seismic criteria given in regulatory position 5 of this guide to a height sufficient to contain the maximum liquid inventory expected to be in the building.

3.1.4 Equipment and components used to collect, process, or store solid radwastes need not be designed to the seismic criteria in regulatory position 5 of this guide.

Response:

NextEra Energy Duane Arnold commits to the content of Regulatory Guide 1.143, Revision 1, October 1979, Position C.3 with the following exceptions:

3.1.1 Refer to response 1.1.1 above.

3.1.2 Refer to response 1.1.2 above.

3.1.3 Refer to response 1.1.3 above.

3.1.4 No exceptions.

Regulatory Position C.4, "Additional Design, Construction, and Testing Criteria"

In addition to the requirements inherent in the codes and standards listed in Table 1, the following criteria, as a minimum, should be implemented for components and systems considered in this guide.

4.1 Radioactive waste management structures, systems, and components should be designed to control leakage and facilitate access, operation, inspection, testing, and maintenance in order to maintain radiation exposures to operating and maintenance personnel as low as is reasonably achievable. Regulatory Guide 8.8 provides guidelines acceptable to the NRC staff on this subject.

4.2 The quality assurance provisions described in regulatory position 6 of this guide should be applied.

4.3 Pressure-retaining components of process systems should use welded construction to the maximum practicable extent. Process systems include the first root valve on sample and instrument lines. Flanged joints or suitable rapid-disconnect fittings should be used only where maintenance or operational requirements clearly indicate that such construction is preferable. Screwed connections in which threads provide the only seal should not be used except for instrumentation and cast pump body drain and vent connections where welded connections are not suitable. Process lines should not be less than 3/4 inch (nominal). Screwed connections backed up by seal welding, mechanical joints, or socket welding may be used on lines 3/4 inches or larger but less than 2-1/2 inches. For lines 2-1/2 inches and above, pipe welds should be of the butt-joint type. Nonconsumable backing rings should not be used in lines carrying resins or other particulate material. All welding constructing the pressure boundary of pressure-retaining components should be performed in accordance with ASME Boiler and Pressure Vessel Code Section IX.

4.4 Piping systems should be hydrostatically tested in their entirety except (1) at atmospheric tanks where no isolation valves exist, (2) where such testing would damage equipment, and (3) where such testing could seriously interfere with other system or component testing. In the case of (2) and (3), pneumatic testing should be performed. Pressure testing should be performed on as large a portion of the in-place systems as practicable. Testing of piping systems should be performed in accordance with applicable ASME or ANSI codes. The system is acceptable if pressure is held for 30 minutes with no leakage indicated.

4.5 Testing provisions should be incorporated to enable periodic evaluation of the operability and required functional performance of active components of the system.

Response:

NextEra Energy Duane Arnold commits to the content of Regulatory Guide 1.143, Revision 1, October 1979, Position C.4, with the following exceptions:

The codes and standards listed in Table 1 are superseded as described in 1.1.1 above.

4.1 The NextEra Energy Duane Arnold position on Regulatory Guide 8.8 is addressed separately.

4.2 Refer to the "Response" on Regulatory Position C.6 regarding the quality assurance provisions applied to radioactive waste management systems, structures and components.

NOTE: The Quality Assurance Program for the Operations Phase supports Regulatory Guide 1.143, Revision 2, in which provisions are contained in Regulatory Position C.7.

4.3 Pressure retaining components and connections will be fabricated in accordance with good operability, maintenance and repairability practices, consistent with good ALARA Practices to the extent practicable.

NOTES: Underlined section added from the original IELP position.

4.4 No exceptions taken.

4.5 No exceptions taken.

Regulatory Position C.5, "Seismic Design for Radwaste Management Systems and Structures Housing Radwaste Management Systems"

5.1 Gaseous Radwaste Management Systems

5.1.1 For the evaluation of the gaseous radwaste system described in regulatory position 2.1.3, a simplified seismic analysis procedure to determine seismic loads may be used. The floor response spectra should be obtained analytically (regulatory position 5.2) from the application of the Regulatory Guide 1.60 design response spectra normalized to the maximum ground acceleration for the operating basis earthquake (OBE), as established in the application, at the foundation of the building housing the gaseous radwaste system.

5.1.2 The allowable stresses to be used for steel system support elements should be those given in "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings", adopted by the American Institute of Steel Construction (AISC) in February 1969. The one-third allowable stress increase provisions for combinations involving earthquake loads, indicated in Section 1.5.6 of the specifications, should be included. For

the design of concrete structures, the use of the American Concrete Institute (ACI) standard ACI 318-77, "Building Code Requirements for Reinforced Concrete", is acceptable.

5.1.3 The construction and inspection requirements for the support elements should comply with those stipulated in AISC or ACI codes as appropriate.

5.2 Buildings Housing Radwaste Systems

5.2.1 Ground motion at the foundation of the building housing the radwaste systems should be defined by normalizing the Regulatory Guide 1.60 spectra to the maximum ground acceleration selected for the plant OBE as established in the application. Damping values to be used in the analysis of the building should be those given for the OBE in Table 1 of Regulatory Guide 1.61. A simplified analysis should be performed to determine appropriate seismic loads and floor response spectra pertinent to the location of the system.

5.2.2 The simplified method for determining seismic loads for the building consists of (a) calculating the first several model frequencies and participation factors for the building, (b) determining model seismic loads using regulatory position 5.2.1 input spectra, and (c) combining modal seismic loads in one of the ways described in Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis".

5.2.3 With regard to generation of floor response spectra for radwaste systems, simplified methods that give approximate floor response spectra without need for performing a time history analysis such as those in References 1, 2, and 3 may be used. Further guidance with respect to floor response spectra can be found in Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components".

5.2.4 The load factors and load combinations to be used for concrete structures should be those given in ACI 318-77. The allowable stresses for steel components should be those given in the AISC specifications. (See regulatory position 5.1.2).

5.2.5 The construction and inspection requirements for the building elements should comply with those stipulated in the appropriate AISC or ACI code.

5.2.6 The foundation media of structures holding the radwaste systems should be selected and designed to prevent liquefaction from the effects of the maximum ground acceleration selected for the plant OBE.

5.3 In lieu of the criteria and procedures defined above, optional shield structures constructed around and supporting the radwaste systems may be erected to protect the radwaste systems from effects of housing structural failure. If this option is adopted, the procedures described in regulatory position 5.2 need only be applied to the shield structures.

Response:

NextEra Energy Duane Arnold commits to the content of Regulatory Guide 1.143, Revision 1, October 1979, Position C.5, with the following exceptions:

5.1 The DAEC Gaseous Radwaste Management Systems are classified non-seismic. \ Reference UFSAR, Section 3.2.

5.1.1 Does not apply to DAEC.

5.1.2 Does not apply to DAEC.

5.1.3 Does not apply to DAEC.

5.2 The DAEC buildings housing radwaste systems are not classified as seismic structures. Refer to UFSAR, Section 3.2.

5.2.1 Does not apply to DAEC.

5.2.2 Does not apply to DAEC.

5.2.3 Does not apply to DAEC.

5.2.4 Does not apply to DAEC.

5.2.5 Does not apply to DAEC.

5.2.6 Does not apply to DAEC.

5.3 The DAEC buildings housing radwaste systems are not classified as seismic structures. Refer to UFSAR Section 3.2.

Regulatory Position C.6, "Quality Assurance for Radwaste Management Systems"

NOTE: The Quality Assurance Program for the Operations Phase supports Regulatory Guide 1.1.43, Revision 2, in which provisions are contained in Regulatory Position C.7.

Since the impact of these systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. To ensure that systems will perform their intended functions, a quality assurance program sufficient to ensure that all design, construction, and

testing provisions are met should be established and documented. The following quality assurance program is acceptable to the NRC staff. It is reprinted by permission of the American Nuclear Society from ANSI N199-1976/ANS-55.2, "Liquid Radioactive Waste Processing System for Pressurized Water Reactor Plants".

6.1 Quality Control

The design, procurement, fabrication and construction activities shall conform to the quality control provisions of the codes and standards specified herein. Acceptable codes and standards are indicated in Table 1 of this guide. In addition, or where not covered by the referenced codes and standards, the following quality control features shall be established:

Acceptable codes and standards are indicated in Table 1 of this guide.

6.1.1 System Designer and Procurer

(1) Design and Procurement Document Control-Design and procurement documents shall be independently verified for conformance with the requirements of this standard by individuals within the design organization who are not the originators of the document. Changes to these documents shall be verified or controlled to maintain conformance to this standard.

(2) Control of Purchased Material, Equipment and Services-Measures to ensure that suppliers of material, equipment and construction services are capable of supplying these items to the quality specified in the procurement documents shall be established. This may be done by an evaluation or a survey of the suppliers' products and facilities.

(3) Handling, Storage, and Shipping-Instructions shall be provided in procurement documents to control the handling, storage, shipping and preservation of material and equipment to prevent damage, deterioration or reduction of cleanness.

6.1.2 System Constructor

Inspection. In addition to required code inspections a program for inspection of activities affecting quality shall be established and executed by, or for, the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. This shall include the visual inspection of components prior to installation for conformance with procurement documents and the visual inspection of items and systems following installation, cleanness and passivation (where applied).

(1) Inspection, Test and Operating Status. Measures should be established to provide for the identification of items which have satisfactorily passed required inspection and tests.

(2) Identification and Corrective Action for Items of Nonconformances. Measures should be established to identify items of nonconformance with regard to the requirement of the procurement documents or applicable codes and standards and to identify the action taken to correct such items.

In Section 4.2.3.2(3), "items of nonconformance" should be interpreted to include failures, malfunctions, deficiencies, deviations, and defective material and equipment.

Sufficient records should be maintained to furnish evidence that the measures identified above are being implemented. The records should include results of reviews and inspections and should be identifiable and retrievable.

Response:

NextEra Energy Duane Arnold commits to the content of Regulatory Guide 1.143, Revision 1, October 1979, Position C.6, with the following exceptions:

NOTE: The Quality Assurance Program for the Operations Phase supports Regulatory Guide 1.1.43, Revision 2, in which provisions are contained in Regulatory Position C.7.

6.1 "Quality Control"

The design, procurement, fabrication and construction activities shall conform to the quality control provisions of the codes and standards as specified in response to Regulatory Position 1.1.1 above.

6.1.1 System Designer and Procurer:

(1) Design and Procurement Document Control

Procurement documents for spare and replacement parts are reviewed by a responsible engineer to ensure conformance with the original specifications. An independent review is not performed for spare and replacement parts.

(2) Control of Purchased Material, Equipment and Services:

Measures are established to assure that purchased material, equipment and services conform to procurement documents. These measures include provisions, as appropriate, for supplier evaluation or a survey, objective

evidence of quality furnished by the supplier, and examination of products upon delivery to ensure the integrity of materials performing a pressure boundary function.

(3) Handling, Storage, and Shipping

No exceptions.

6.1.2 System Constructor

(1) Inspection

These inspections apply to pressure retaining components only.

(2) Inspection, Test and Operating Status

No exceptions taken.

(3) Identification and Corrective Action for Items of Nonconformances

No exceptions taken.

1.8.34 REGULATORY GUIDE 1.155, "STATION BLACKOUT"

Comments and Clarifications:

The Company complies with Appendix A, "Quality Assurance Guidelines for Non-Safety Systems and Equipment," to Regulatory Guide 1.155, Revision 1, August 1988.

REFERENCES FOR SECTION 1.8

1. Letter from Gary Van Middlesworth (NMC DAEC) to Office of Nuclear Reactor Regulation "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-037 – Alternative Source Term. (TAC #MB0347)" dated March 23, 2001.

2. Letter from Darl S. Hood (NRC) to Gary Van Middlesworth (NMC DAEC) "Duane Arnold Energy Center – Issuance of Amendment Regarding Secondary Containment Operability During Movement of Irradiated Fuel and Core Alterations (TAC No. MB1569)" dated April 16, 2001.

3. Outage Risk Management Guidelines OMG-7 Section 6.3.5.(2)

4. Letter from Gary Van Middlesworth (NMC DAEC) to Office of Nuclear Reactor Regulation "Technical Specification Change Request (TSCR-037): Alternative Source Term" dated October 19, 2000.

5. Letter from B. Mozafari (USNRC) to G. Van Middlesworth (NMC), "Duane Arnold Energy Center – Use of Net Positive Suction Head Margin as an Acceptable Criterion (TAC No. MB0543)", September 25, 2001.

6. CAL-M06-012, Rev. 0: Required Water Depth at River Water Intake.

2018-004 7. CAL-M90-030, Iowa Vane Qualification.

8. CAL-C91-006, Qualification of Steel Sheet Piling Guidewall.

TABLE 1.8-1

INCREMENTAL EXPOSURE FROM VENTING

EXPOSURE	WHOLE BODY[a] EXPOSURE (REM)	THYROID[a] EXPOSURE (REM)
Site Boundary		
Venting increment (7-day exposure starting after 30 days)	0.016	2.9
Low Population Zone		
Venting increment (7-day exposure starting after 30 days)	0.001	0.22

[a]Results of LOCA evaluation for the initial operating license application.

TABLE 1.8-2
Deleted

TABLE 1.8-3

QUALITY STANDARDS

COMPONENTS	QUALITY B	QUALITY C	QUALITY D
Pressure vessels	ASME B&PV Code, Section III, Nuclear Power Plant Components,[a] Class 2	ASME B&PV Code, Section III, Nuclear Power Plant Components,[a] Class 3	ASME B&PV Code, Section VIII, Division I
Piping	As above	As above	ANSI B31.1.0, Power Piping
Pumps	As above	As above	ASME B&PV Code, Section VIII, Division I
Valves	As above	As above	ANSI B31.1.0
Atmospheric storage tanks	As above	As above	API-650, AWWA D 100, or ANSI B96.1
0-15 psig storage tanks	As above	As above	API-620

[a]1971 Code and appropriate Addenda. See 10 CFR 50.55a for general guidance relating the Code and Addenda to be applied, the date the component was ordered, and the date of issuance of a construction permit.

1.9 STANDARD DESIGNS

Section 1.9 of Regulatory Guide 1.70, Revision 3, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,” does not apply to the DAEC.