APPENDIX 1C

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PRINCIPAL DESIGN CRITERIA

The principal general design criteria given in this section are those resulting from the Atomic Industrial Forum's redraft of the proposed 70 "General Design Criteria for Nuclear Power Plant Construction Permits" (GDC) issued by AEC Press Release No. K-172 on July 10, 1967. It is the intent of CP Co that the design of the Midland Plant Units 1 and 2 meets these design criteria as interpreted herein. The principal safeguards corresponding to each criterion are summarized herein, and reference is made to sections of this report where more detailed information is presented. The numbering of criteria herein is consistent with that of the GDC.

CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required.

Discussion

A. Essential Systems and Components

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

1. Fuel assemblies.
2. Reactor vessel internals.
3. Reactor coolant system.
4. Reactor instrumentation, controls and protection systems.
5. Engineered safeguards.
6. Reactor building.
7. Emergency power sources.

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B. Codes and Standards

The following table references applicable codes and standards for the nuclear units as included in the PSAR:

Item	Codes	Quality Control	Test Procedures
A-1	3.1.2.4.2	(*) App 1B	3.3.3
A-2	3.1.2.4.1 3.2.4.1	3.3.4 App 1B	3.3.4
A-3	4.1 4.1.5	4.1.4.4 4.3.1.1.2 App 1B	4.4 13
A-4	3.1.2.4.4 3.2.4.3.2 3.2.4.3.4 7.1.1.2	3.2.4.3.2 3.2.4.3.4 7.1.1.2 App 1B	3.3.3 7.1.1.2 13
A-5	9 (p 9-1, 9-2) 9.1.2.5 9.3.2.5	9 (p 9-1, 9-2) App 1B	6.1.4 6.2.4 9 (p 9-1, 9-2) 13
A-6	5.1.1 5.1.2 5.1.4 App 5A	5.1.3 App 1B	5.1.2 5.1.4 App 5B App 5H 13
A-7	8.1	App 1B	8.3 13

(*) Fuel assembly production quality control and process procedures are being developed by B&W and vendor manufacturing organizations.

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems

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and components to withstand, without undue risk to the health and safety of the public the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Discussion

A. Essential Systems and Components

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

1. Fuel assemblies.
2. Reactor vessel internals.
3. Reactor coolant system.
4. Reactor instrumentation, controls and protection systems.
5. Engineered safeguards.
6. Reactor building.
7. Emergency power sources.

B. Performance Standards

These essential systems and components have been designed to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena. The designs are based upon the most severe of the natural phenomena, recorded for the vicinity of the site, with an appropriate margin to account for uncertainties in the historical data.

These natural phenomena are listed below with PSAR references:

	Sections	Appendices
25 1. Earthquake	2.7, 5.1.1.2	5A
2. Tornado	2.3, 5.1.1.2	2A, 5A
25 3. Flood and Groundwater	2.4, 2.5, 2.6, 5.1.1.2	2B, 2C, 5A
4. Wind	2.3, 5.1.1.2	2A, 5A
5. Snow and Ice	2.3, 5.1.1.2	2A, 5A
25 6. Other Local Site Effects	5.1.1.2	5A

CRITERION 3 - FIRE PROTECTION (Category A)

A reactor facility shall be designed such that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Discussion

The reactor facility is designed such that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public.

The potential magnitude of a fire in the control room will be limited by the following factors:

a. Materials used in control room construction are nonflammable.

b. Control cables and switchboard wiring are constructed of materials that meet the flame resisting tests described in Insulated Power Cable Engineers Association Publication S 61-402, Part 6.5, and National Electrical Manufacturers Association Publication WC 5-1968.

c. Furniture in the control room is of metal construction.

d. Combustible supplies such as logbooks, records, procedures, manuals, etc, are limited to the amounts required for plant operation.

e. All areas of the control room are readily accessible for fire extinguishing.

f. Adequate fire extinguishers are provided.

g. The control room is occupied at all times by a qualified person who has been trained in fire extinguishing techniques.

The only flammable materials inside the control room are:

a. Paper in the form of logs, records, procedures, manuals, diagrams, etc.

b. Small amounts of combustible materials used in the manufacture of various electronic equipment.

The above list indicates that the flammable materials are distributed to the extent that a fire would be unlikely to spread. Therefore, a fire, if started, would be of such a small magnitude that it could be extinguished by the operator using a hand fire extinguisher. The resulting smoke and vapors would be removed by the control room ventilation system.

The safety of the nuclear unit is not affected by fires due to these materials. The operation of the engineered safeguards is not impaired due to the effects of fires.

CRITERION 4 - SHARING OF SYSTEMS (Category A)

Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public.

Discussion

Systems and components will be shared only to the extent that it can be demonstrated that such sharing does not affect the functional capability of the system or component to perform adequately in the separate reactor facilities. Shared systems and components are described in 1.2.8.

CRITERION 5 - RECORDS REQUIREMENTS (Category A)

The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public.

Discussion

The Applicant assures the maintenance through the life of the reactor of records of the design, fabrication, construction and tests of major components of the plant essential to avoid undue risk to the health and safety of the public.

CRITERION 6 - REACTOR CORE DESIGN (Categories A & B)

The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.

Discussion

The reactor is designed with the necessary margins to accommodate, without fuel damage, expected transients from steady-state operation including the

transients given in the criterion. Fuel clad integrity is insured under all normal and abnormal modes of anticipated operation by avoiding clad overstressing and overheating. The evaluation of clad stresses includes the effects of internal and external pressures, temperature gradients and changes, clad-fuel interactions, vibrations, and earthquake effects. The freestanding clad design prevents collapse at the end volume region of the fuel rod and provides sufficient radial and end void volume to accommodate clad-fuel interactions and internal gas pressures (3.2.4.2).

Clad overheating is prevented by satisfying the following core thermal and hydraulic criteria (3.1.2.3 and 3.2.3.1.1):

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

b. A 99 percent confidence exists that at least 99.5 percent of the fuel rods in the core will be in no jeopardy of experiencing a DNB during continuous operation at the design overpower of 114 percent.

The design margins allow for deviations of temperature, pressure, flow, reactor power, and reactor-turbine power mismatch. Above 15 percent power, the reactor is operated at a constant average coolant temperature and has a negative power coefficient to damp the effects of power transients. The reactor control system will maintain the reactor operating parameters within preset limits, and the reactor protection system will shut down the reactor if normal operating limits are exceeded by preset amounts (Sections 7.1 and 14).

CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (Category B)

The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

Discussion

Power oscillations resulting from variation of coolant temperature are minimized by constant average coolant temperature above 15 percent power. Power oscillations from spatial xenon effects are minimized by the large negative power coefficient. Features have been provided in the design that will allow control of axial oscillations and will make the core stable in regard to azimuthal oscillations. The reactor is shown by analysis to be stable to radial oscillations. Reactor trip prevents fuel clad damage resulting from DNB.

The ability of the reactor cortrol and protection system to control the oscillations resulting from variation of coolant temperature within the control system dead band and from spatial xenon oscillations has been analyzed. With regard to axial oscillations, certain of the control rod assemblies will contain poison only in a portion of their lengths and will be positioned to

maintain an acceptable power distribution in the core and to control any tendency towards axial oscillation. Azimuthal oscillation tendencies will be minimized in the design by including fixed burnable poison in the core design (3.2.2.2.3).

CRITERION 8 - OVERALL POWER COEFFICIENT (Category B)

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

Discussion

The overall power coefficient is negative in the operating range (3.2.2.1.4).

CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

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

Discussion

The reactor coolant pressure boundary will be designed and constructed to meet these criteria:

a. Material selection, design, fabrication, inspection, testing, and certification will be in keeping with the ASME (Section III) and USASI (B31.7) Codes.

b. Quality manufacture will include weld qualification test plates, permanent identification of materials, welder qualification tests, and extensive production nondestructive testing.

c. Service life of the reactor vessel and other coolant boundary materials will be chosen to retain metallurgical stability of the material, to account for cyclic effects of mechanical shock and vibratory loadings, and to give due consideration to radiation effects and the amount of increase in the nil ductility transition temperature as a result of neutron irradiation (Section 4.1).

CRITERION 10 - REACTOR CONTAINMENT (Category A)

Reactor containment shall be provided. The containment structure shall be designed (a) to sustain without undue risk to the health and safety of the

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public the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public.

Discussion

The reactor building, a continuous, prestressed concrete structure, with a welded steel liner to provide leak tightness, completely encloses the entire reactor and reactor coolant system, to insure with certain engineered safeguards (see Criterion 37) that an acceptable upper limit for leakage of radioactive materials to the environment will not be exceeded, even if gross failure of the reactor coolant system were to occur (Section 5). It is the design goal to maintain the integrity of the reactor building under both normal and accident conditions.

CRITERION 11 - CONTROL ROOM (Category B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposures of personnel.

Discussion

The facility is provided with a control room (Section 7.6). It is a design objective that occupancy in this control room can be maintained at all times. The reactor can be tripped from the control room. As is discussed in 5.4.2.1, the whole body dose to plant personnel during the 30-day period following a design basis loss-of-coolant accident (MHA) will not exceed 5 Rem, and the thyroid dose will not exceed 30 Rem.

CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (Category B)

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

Discussion

Reactor regulation is based upon the use of movable control rods and a chemical neutron absorber (boron in the form of boric acid), dissolved in the reactor coolant. Input signals to the reactor controls include reactor coolant average

temperature, megawatt demand, and reactor power. The reactor controls are designed to maintain a constant average reactor coolant temperature over the load range from 15 to 100 percent of rated power. The steam system operates on constant pressure at all loads. Adequate instrumentation and controls are provided to maintain operating variables within their prescribed ranges (Section 7.2).

The nonnuclear instrumentation measures temperatures, pressures, flows, and levels in the reactor coolant system, steam system, and auxiliary reactor systems, and maintains these variables within prescribed limits (7.3.2).

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

Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core.

Discussion

This criterion is met by reactivity control means and control room display. Reactivity control is by movable control rods, movable xenon control rods to facilitate the control of axial power maldistribution, fixed burnable poison distributed in the core, and by chemical neutron absorber (in the form of boric acid), dissolved in the reactor coolant. The position of each control rod will be displayed in the control room. Changes in the reactivity status due to soluble boron will be indicated by changes in the position of the control rods. Actual boron concentration in the reactor coolant is determined periodically by the sampling system and is reported to the reactor operator (Section 7.2).

CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

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

Discussion

The reactor design meets this criterion by reactor trip provisions and engineered safeguards. The reactor protection system is designed to limit reactor power which might result from unexpected reactivity changes, and provides an automatic reactor trip to prevent exceeding acceptable fuel damage limits. In a loss-of-coolant accident, the engineered safeguards protection system automatically actuates the high-pressure and low-pressure coolant injection equipment. The core flooding tanks are self-actuating (Section 7.1).

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

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

Discussion

The reactor protection system senses abnormal reactor power level, reactor outlet temperature, reactor coolant pressure and reactor start-up rate, and trips the reactor for each abnormal condition. The engineered safeguards actuation system senses reactor coolant pressure and reactor building pressure, either of which initiates core coolant injection. The reactor building pressure signal also initiates emergency building cooling, reactor building isolation and fission product removal system. A signal of high radiation in the reactor building initiates isolation of piping open to the reactor building atmosphere. Analyses of all accident situations examined, including the postulated LOCA, indicate that the system monitoring sensors provided in the design act to initiate the operation of necessary engineered safeguards to protect the reactor core and reactor coolant system (Section 7.1).

CRITERION 16 - MONITORING REACTOR COOLANT LEAKAGE (Category B)

Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary.

Discussion

Instrumentation is provided to meet this criterion by measuring fluid volume changes (pressurizer and reactor building sump) and radioactivity levels in the reactor building. An increase in net makeup to the combined reactor coolant system and connected high-pressure injection and purification system will also indicate leakage (4.2.7).

CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (Category B)

Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive.

Discussion

Monitoring of all station solid, liquid, and gaseous releases is accomplished with the appropriate instrumentation (Section 7.5). Releases from the reactor

building ventilation prior to release are monitored systematically. The plant ventilation is similarly monitored and diluted to achieve acceptable concentrations. All liquid effluents are sampled or monitored both prior to and after treatment (Section 11.2). The solid effluents are packaged and checked for radioactivity before shipment off-site. Hence, monitoring of the releases within the facility environs is controlled so that the releases are never more than allowed by 10 CFR 20. Detectors located in selected areas of the station along with operating procedures assure that personnel exposure does not exceed 10 CFR 20 limits (Section 7.5). The environmental program is designed to establish environmental radiation levels and detect any changes which may occur. Sampling points are located on-site and off-site.

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

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels.

Discussion

Monitoring and alarm instrumentation is provided sensitive to the operation of the spent fuel pool cooling system (Section 9.4). Heat is removed from stored radwaste by conduction to the ventilation air. Ventilation air from the radwaste facility is continuously monitored for radioactivity.

CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category B)

Protection systems shall be designed for high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public.

Discussion

The protection systems' design meets this criterion by specific location, ample design capacity, component redundancy, and in-service testability. The major design criteria stated below have been applied to the design of the instrumentation.

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

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

Manual testing facilities are built into the protection systems to provide for:

a. Preoperational testing to give assurance that the protection systems can fulfill their required functions.

b. On-line testing to assure operability and to demonstrate reliability (7.1.1).

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

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served.

Discussion

Reactor protection and engineered safeguards are by four channels with 2/4 coincidence. All protection system functions are implemented by redundant sensors, instrument strings, logic, and action devices that combine to form the protection channels. Redundant protection channels and their associated elements are electrically independent and packaged to provide physical separation. The reactor protection system initiates a trip of the channel involved when modules, equipment, or subassemblies are removed (7.1.1).

CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

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

Discussion

The protection systems meet this criterion by complying with Criterion 23.

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

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation

system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

Discussion

The protection systems and control instrumentation systems meet this criterion by complying with Criterion 20 (Section 7.1).

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

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

Discussion

The protection systems are designed to extreme ambient conditions and with redundancy. Protection systems' instrumentation will operate from 40-140 F and sustain (except for neutron detectors) the loss-of-coolant building environmental conditions of 67 psig and 297 F and still be operable. Protection systems' instrumentation will be subject to environmental (qualification) test as required by the proposed IEEE Standard for Nuclear Power Plant Protection Systems. Protective equipment outside of the reactor building, control room, and relay room is designed for continuous operation in an ambient temperature of 120 F and for 90 percent relative humidity (7.1.1.4).

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

This criterion is deleted since it appears preferable to focus all requirements for emergency power in Criterion 39. Note that "protection systems" is incorporated in Criterion 39 to accommodate this deletion.

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

Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred.

Discussion

Test circuits are supplied which utilize the protection systems' redundancy, independence, and coincidence features. This makes it possible to manually initiate on-line trip signals in any single protection channel in order to test trip capability in each channel without affecting the other channels (7.1.3).

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

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

Discussion

The reactor protection system will trip the reactor on loss of power. The engineered safeguards actuation system is supplied with multiple sources of electric power for control and valve action. A total loss of electrical power to the engineered safeguards actuation system will cause its instrumentation to assume a tripped position with the exception of the final control relays. These relays require power to trip. However, since the engineered safeguards equipment also requires power to operate, this relay need not assume the tripped position upon a total loss of power. The multiple power supplies for the control relays are also backed up by battery power, and therefore are more reliable than the power supply for the engineered safeguards equipment.

The system is designed for continuous operation under adverse environments. The reactor protection system instrumentation within the reactor building is designed for continuous operation in an environment of 140 F, 67 psig, and 100 percent relative humidity. Engineered safeguards equipment and vital instrumentation inside the reactor building are designed for conditions (67 psig, 297 F, and 100 percent RH) which meet the requirements of the loss-of-coolant accident (7.1.1 and 7.1.2).

Redundant instrument channels are provided for the reactor protection and engineered safeguards actuation systems. Loss of power to each individual reactor protection channel will trip that individual channel. Loss of all instrument power will trip the reactor protection system, thereby releasing the control rods, and will activate the engineered safeguards actuation system controls (with the exception of the reactor building spray valves).

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

CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

Two independent reactivity control systems, preferably of different principles, shall be provided.

Discussion

This criterion is met by control rods and soluble boron addition to, or removal from, the reactor coolant (7.2.2.1).

CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot operating condition.

Discussion

One highly redundant reactivity control system consisting of 49 full-length control rods is provided to rapidly make the core subcritical upon a trip signal and to protect the core from damage due to the effects of any operating transient. The soluble absorber reactivity control system can make the reactor subcritical even from ultimate power. However, its action is slow, and the ability to protect the core from damage which might result from rapid load changes, such as an ultimate load turbine trip, is not a design criterion for this system. The high degree of redundancy in the control rod system is considered sufficient to meet the intent of this criterion (3.2.2.1).

CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (Category A)

One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn.

Discussion

The reactor design meets this criterion both under normal operating conditions, and under the accident conditions set forth in Section 14. The reactor is designed with the capability of providing an adequate shutdown margin with the single most reactive control rod fully withdrawn at any point in core life with the reactor at a hot, zero power condition. The minimum hot shutdown margin occurs at the end of life (3.2.2.1).

CRITERION 30 - REACTIVITY HOLDDOWN CAPABILITY (Category B)

The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

Discussion

The reactor meets this criterion with control rods for hot shutdown under normal operating conditions and for shutdown under the accident conditions set forth in Section 14. Reactor subcritical margin is maintained during cooldown by changes in soluble boron concentration. The rate of reactivity compensation from boron addition is greater than the reactivity change associated with the maximum allowable reactor cooldown rate of 100 F/hour. Thus, subcriticality is assured during cooldown with the most reactive control rod totally unavailable (3.2.2.1).

CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (Category B)

The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

Discussion

The reactor design meets this criterion. A reactor trip will protect against continuous withdrawal of one rod (14.1.2.3).

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

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

Discussion

The reactor design meets this criterion by engineered safeguards which limit the maximum reactivity insertion rate. These include rod-group withdrawal interlocks, soluble boron concentration reduction interlock, maximum rate of

dilution water addition, and dilution-time cutoff (14.1.2.4). In addition, the rod drives and their controls have an inherent feature that limits overspeed in the event of malfunctions (3.2.4.3). Ejection of the maximum worth control rod will not lead to further coolant boundary rupture or internals damage which would interfere with emergency core cooling (14.2.2.2).

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

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

Discussion

The reactor design meets this criterion. There are no credible mechanisms whereby damaging energy releases are liberated to the reactor coolant. Ejection of the maximum worth control rod will not lead to further coolant boundary rupture (14.2.2.2).

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

The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failures. Consideration shall be given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.

Discussion

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

a. The reactor vessel is the only reactor coolant system component exposed to a significant level of neutron irradiation, and is therefore the only component subject to material irradiation damage. The end-of-unit-life NDTT value of the vessel opposite

the core will be not more than 260 F based on an initial value of 10 F. Unit operating procedures will be established to limit the operating pressure to 20 percent of the design pressure when the reactor coolant system temperature is below NDTT + 60 F throughout unit life.

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

c. Quality control procedures including permanent identification of materials and nondestructive testing for flaw identification.

d. Operating restrictions to prevent failure resulting from increase in brittle fracture transition temperature due to neutron irradiation, including a material irradiation surveillance program.

e. The reactor vessel will be manufactured in accordance with the provisions of the ASME Boiler and Pressure Vessel Code, Part III (Nuclear Vessels) (4.1.4).

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

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

Discussion

The reactor coolant pressure boundary meets this criterion by complying with Criterion 34 (4.1.4).

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

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided.

Discussion

The reactor coolant pressure boundary components meet this criterion. Space is provided for nondestructive testing during unit shutdown. A reactor pressure vessel material surveillance program conforming to ASTM-E-185-66 will be established (4.4.3).

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

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

Discussion

The reactor design meets this criterion. The emergency core cooling system can protect the reactor for any size leak up to and including the circumferential rupture of the largest reactor coolant pipe (14.2.2.3).

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

All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public.

Discussion

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

The high-pressure injection, core-flooding, and low-pressure injection components of the emergency core cooling system have separate equipment strings to insure availability of capacity.

Some engineered safeguards have both a normal and an emergency function, thereby providing nearly continuous demonstration of operability. During normal operation, the standby and operating units will be rotated into service on a scheduled basis.

00287

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

Testing and inspection of the engineered safeguards is described in the PSAR for each system in the criteria where such information is asked for specifically (6.1.4, 6.2.4, 6.3.4).

CRITERION 39 - EMERGENCY POWER (Category A)

An emergency power source shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems required to avoid undue risk to the health and safety of the public. This power source shall provide this capacity assuming a failure of a single active component.

Discussion

In the event of loss of all off-site power, a drop in plant load to auxiliary load is accomplished by the step load reduction detailed in 14.1.2.8 In addition, emergency power sources provide a dependable supply of power for the critical services in the unlikely event of simultaneous loss of normal and standby power (8.2.3). Two diesel generators supply two 4160 volt buses and station batteries, and will provide the power required for postulated loss-of-coolant accidents for one unit for vital auxiliaries, instrumentation, control equipment and emergency lighting to enable safe shutdown, and provide normal shutdown for the other unit.

CRITERION 40 - MISSILE PROTECTION (Category A)

Adequate protection for those engineered safety features, the failure of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Discussion

Those engineered safeguards, the failure of which could cause an undue risk to the health and safety of the public, are adequately protected against dynamic effects and missiles that might result from plant equipment failures.

00288

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

Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public.

Discussion

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

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

Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public.

Discussion

The reactor design meets this criterion. The engineered safeguards are designed to function in the unlikely event of a loss-of-coolant accident with no impairment of capability due to the effects of the accident.

The core flooding tanks contain check valves which operate to permit flow of emergency coolant from the tanks to the reactor vessel. These valves are self-actuating and need no external signal or external supplied energy to make them operate.

CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Protection against any action of the engineered safety features which would accentuate significantly the adverse after-effects of a loss of normal cooling shall be provided.

Discussion

The engineered safeguards are designed to meet this criterion. The water injected to insure core cooling is sufficiently borated to insure core subcriticality. Nonessential sources of water inside the reactor building are

automatically isolated to prevent dilution of the borated coolant. Essential sources of post-accident cooling waters are monitored to detect leakage which may lead to dilution of boron content. An analysis has been made to demonstrate that the injection of cold water on the hot reactor coolant system surfaces will not lead to further failure. The design of the equipment and its actuating system insures that water injection will occur in a sufficiently short time period to preclude significant metal-water reactions and consequent energy releases to the reactor building (14.2.2.3).

CRITERION 44 - EMERGENCY CORE COOLING SYSTEM CAPABILITY (Category A)

An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty.

Discussion

Emergency core cooling is provided by pumped injection and pressurized core flooding tanks. This equipment prevents clad melting for the entire spectrum of reactor coolant system failures ranging from the smallest leak to the complete severance of the largest reactor coolant pipe. Pumped injection is subdivided in such a way that there are two separate and independent strings, each including both high-pressure and low-pressure coolant injection, and each capable of providing 100 percent of the necessary core injection with the core flooding tanks. The core flooding tanks are passive components which are needed for only a short period of time after the accident, thereby assuring 100 percent availability when needed.

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

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

Discussion

All critical parts of the emergency core cooling system, including the reactor vessel internals and water injection nozzles, can be inspected during unit shutdown (Section 4.4).

00290

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

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

Discussion

The emergency core cooling system design meets this criterion by rotating the active components which are normally in service as components of the engineered safeguards systems. In addition, periodic tests are performed on components not normally in service (6.1.4).

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

A capability shall be provided to test periodically the operability of the emergency core cooling system up to a location as close to the core as is practical.

Discussion

The high-pressure (makeup water) and low-pressure injection (decay-heat removal) strings are included as part of normal service systems. Consequently, the active components can be tested periodically for operability. The core flooding system operability can be tested during shutdown or refueling. In addition, all valves will be periodically cycled to insure operability. With these provisions, the operability of the emergency core cooling system can be periodically demonstrated (6.1.4).

CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEM (Category A)

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

Discussion

The operational sequence that would bring the emergency core cooling system into action, including transfer to alternate power sources, can be tested in parts (6.1.4 and 7.1.3).

00291

CRITERION 49 - REACTOR CONTAINMENT DESIGN BASIS (Category A)

The reactor containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident, including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system, will not result in undue risk to the health and safety of the public.

Discussion

The reactor building, including access openings and penetrations, has a design pressure of 67 psig at 297 F. The greatest transient peak pressure, associated with a postulated rupture of the piping in the reactor coolant system and the calculated effects of a metal-water reaction, does not exceed these values.

The reactor building and engineered safeguards systems have been evaluated for various combinations of credible energy releases. The analysis accounts for system energy and decay heat. The cooling capacity of either reactor building cooling system is adequate to prevent overpressurization of the structure.

The use of ECCS for core flooding limits the reactor building pressure to less than the design pressure.

The high-pressure injection and low-pressure injection systems have redundancy of equipment to insure availability of capacity.

Electric motors and valves, which must function within the reactor building during accident condition, are designed to operate in a steam-air atmosphere at 297 F and 67 psig.

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

The selection and use of containment materials shall be in accordance with applicable engineering codes.

Discussion

The ferritic materials used as load carrying components in the reactor building design are selected in accordance with the appropriate codes, regulations, and testing requirements (Section 5).

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

If part of the reactor coolant pressure boundary is outside the containment, features shall be provided to avoid undue risk to the health and safety of the public in case of an accidental rupture in that part.

Discussion

The reactor design meets this criterion. The reactor coolant pressure boundary is defined as those piping systems or components which contain reactor coolant at design pressure and temperature. With the exception of the pressurizer sampling line, the reactor coolant pressure boundary, as defined above, is located entirely within the reactor building. The sampling line is provided with remotely operated valves for isolation in the event of a failure. This line is normally isolated and is used only during actual sampling operations. All other piping and components which may contain reactor coolant are at low temperatures such that any leakage would be collected by the contaminated drain system. No significant environmental dose would arise from these sources.

CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (Category A)

Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure this system shall perform its required function, assuming failure of any single active component.

Discussion

The reactor building design includes two redundant accident heat removal systems, the reactor building spray system and the reactor building air recirculation and cooling system, each with a full capacity of 200×10^6 Btu/h (Sections 6.2 and 6.3).

CRITERION 53 - CONTAINMENT ISOLATION VALVES (Category A)

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

Discussion

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

CRITERION 54 - INITIAL LEAKAGE RATE TESTING OF CONTAINMENT (Category A)

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

Discussion

The design leakage from the reactor building and penetrations is consistent with the requirements of 10 CFR 100. An integrated leakage rate test at the peak pressure calculated to result from the postulated loss-of-coolant accidents, is possible on the completed reactor building including all penetrations (5.1.4).

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

The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime.

Discussion

The reactor building is designed so that the leakage rate can be periodically checked during plant lifetime.

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

Provisions shall be made to the extent practical for periodically testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at the peak pressure calculated to result from occurrence of the design basis accident.

Discussion

All electrical penetrations have provisions for pressurizing between the double seals periodically during operation or shutdown to allow for leak checking by observing the pressure decay. There are no pipe penetrations to the reactor building which require a bellows seal between the pipe and the reactor building.

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

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

Discussion

Testing of the isolation valves and the associated instrumentation is provided for.

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

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

Discussion

The reactor building pressure-reducing systems are the spray system and the air recirculation and cooling system.

Performance testing of all active components of the reactor building spray system is accomplished as described in Criterion 59. During these tests, the equipment is visually inspected for leaks. Valves and pumps are operated and inspected after any maintenance to insure proper operation.

The equipment, piping, valves and instrumentation of the reactor building air recirculation and cooling units are arranged so that they can be visually inspected. The air recirculation and cooling units and associated piping are located outside the secondary concrete shield around the reactor coolant system loops. Since the air recirculation and cooling system is normally in operation to remove the equipment heat load, its performance can be monitored continuously. The service water piping and valves outside the reactor building are inspectable at all times. Operational tests and inspections are performed prior to initial start-up.

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

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

Discussion

The reactor building air recirculation and cooling system is normally in service. Valving on the coils can be periodically cycled, thus placing the coils into emergency service periodically during operation. The active components of the reactor building spray system are tested periodically as set forth in 6.2.4.

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

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

Discussion

The reactor building spray systems are tested on a periodic basis as follows:

Reactor Building Spray Pumps	These pumps are tested singly by opening the valves in the test line, closing the block valves upstream of the spray header isolation valves, and closing the isolation valves. Each pump in turn is manually started and checked for flow.
Reactor Building Spray Nozzles	When the unit is shut down, air or smoke is blown through the test connections with visual observation of the nozzles.

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

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

Discussion

Capability to test under conditions as close to design as practical the full operational sequence that would bring the reactor building pressure-reducing system into action is accomplished with a test line in the spray system (Section 6.2) just prior to the spray system isolation valves. Transfer to emergency power sources is accomplished by tripping the normal source breaker to simulate loss of source. The reactor building air cooling units are continually tested by normal duty.

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

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

Discussion

Since air cleanup immediately following the LOCA or MHA is accomplished by a chemical additive spray system, there is no provision for a post-incident recirculatory air cleanup system within the reactor building.

A filtration system is provided in the reactor building hydrogen vent system which is required only during a post-LOCA or MHA period. All components of the hydrogen venting filtration system including dehumidifiers, ducts, filters, fans and dampers are located outside the enclosure building filtration region. Access for physical inspection of these components is thereby not impeded by reactor operation.

This system is described in Paragraph 9.12.2.1.

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

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

Discussion

Active components of the hydrogen vent and auxiliary building filtration systems are tested periodically for operability and required functional performance.

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

A capability shall be provided to the extent practical for in-situ periodic testing and surveillance of the air cleanup systems to insure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

Discussion

Due to the hydrogen venting and refueling building filtration systems being located outside the reactor building, they may be periodically tested to insure that (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

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

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

Discussion

The full operational sequence that would bring the hydrogen venting and refueling building filtration system into action, including the transfer to an alternate power source and the design airflow delivery capability, can be tested.

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

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

Discussion

This criterion is met by the design of the new and spent fuel assembly storage facilities to maintain a safe condition by storing fuel assemblies in racks having spacing and/or poison sufficient to maintain a k_{eff} of less than 0.90 when wet.

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

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release which would result in undue risk to the health and safety of the public.

Discussion

This decay heat removal is accomplished by the fuel cooling system. In case a maximum of 1-2/3 cores are stored due to complete unloading of one reactor vessel with 2/3 core already in residence, this system plus the decay heat removal system maintains the fuel pool temperature within acceptable limits as described in Section 9.3. The most serious failure would be complete loss of water in the storage pool. This is prevented by placing the cooling connections near or above the water level so that the pool cannot be gravity-drained. Additionally, a

backup water supply is available from the fire system which could be utilized in the unlikely event of a considerable loss of water from the pool. These precautions together with the shielding specified in Section 5.4 will prevent radioactive release to the plant environs.

Cooling is not required for waste storage tanks due to the low level of radioactive heating experienced.

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

Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities.

Discussion

The shielding provided in the spent fuel and waste storage area is in accordance with the radiation zoning described in Section 5.4, enabling the plant to meet the guidelines of 10 CFR 20.

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

Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity.

Discussion

The fuel and waste storage facilities are designed so that accidental releases of radioactivity to the environment resulting from rupture of a waste gas decay tank or from damage to a spent fuel assembly as described in 14.2.2.1 are below the 10 CFR 100 guideline values.

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

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur, and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence.

Discussion

The liquid radioactive waste system collects, treats, stores, recycles, and/or disposes of all radioactive liquid wastes. These are collected in sumps and drain tanks and then transferred to the appropriate tanks 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 methods which are appropriate for the materials present. All processed wastes discharged are monitored prior to release.

Solid radioactive wastes are collected, processed, packaged, and stored temporarily on the site to permit decay or accumulation prior to shipment from the plant for permanent storage.

Equipment is provided to compress and hold up radioactive gases for decay before release through the station stack at a controlled rate, although normally, these gases are monitored and discharged directly to the stack (Section 11). All gaseous and liquid wastes discharged will be in accordance with the guidelines of 10 CFR 20.