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Committed to Nuclear Excellence

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Kewaunee / Point Beach Nuclear
Operated by Nuclear Management Company, LLC

NRC-02-052

June 7, 2002

10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Proposed Amendment 184 to the Kewaunee Nuclear Power Plant Technical Specifications

The Nuclear Management Company (NMC) is submitting this proposed amendment (PA) to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) to convert four sections to the WORD format and correct minor typographical and format inconsistencies. The affected sections are: 1) Section 1 - "Definitions," 2) Section 2 - "Safety Limits and Limiting Safety System Settings," 3) Section 5 - "Design Features," and 4) Section 6 - "Administrative Controls." The administrative changes include capitalizing defined words, formatting section titles, renumbering pages and correcting miscellaneous grammar and punctuation errors. The conversion is part of an ongoing effort to revise each TS section to a consistent format. In addition, the Table of Contents will be renumbered to correspond with the conversion of each section. The Facility Operating License will be submitted to revise punctuation errors. No technical changes are being proposed to the TS pages.

Attachment 1 to this letter contains a description, a safety evaluation, a significant hazards determination and environmental considerations for the proposed changes. Attachment 2 contains the strike-out Technical Specification pages for the Table of Contents, Facility Operating License, TS Section 1.0, TS Section 2.0, TS Section 5.0, and TS Section 6.0 in their entirety, including the bases, tables and figures associated with the sections. Attachment 3 contains the affected Technical Specification pages as revised for these sections.

A001

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects, these statements are not based entirely on my personal knowledge, but on information furnished by cognizant NMC employees and consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on June 7, 2002.

Mark E. Warner
Site Vice President

DAK

Attachments

cc - US NRC - Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

ATTACHMENT 1

Letter from M. E. Warner (NMC)

to

Document Control Desk (NRC)

Dated

June 7, 2002

Proposed Amendment 184

Description of Proposed Changes
Safety Evaluation
Significant Hazards Determination
Environmental Considerations

Description of Proposed Changes

Technical Specifications (TS) Sections 1.0, 2.0, 5.0, and 6.0 are being converted to WORD format to conform to the standard software used at KNPP. The conversion will include all the bases sections, tables and figures associated with these sections. In addition to the conversion, minor typographical, grammatical and formatting inconsistencies will be corrected for these sections. These corrections include capitalizing defined words, formatting section titles, renumbering pages and standardizing lists and punctuation. The proposed changes will also include renumbering section pages in the Table of Contents and punctuation revisions to the Facility Operating License. No technical changes are being proposed to the TS pages

Safety Evaluation for Proposed Changes

The proposed changes to convert the TS to WORD format and correct typographical, grammatical and format errors are administrative and strictly for a standardized format of the TS. The intent or interpretation of these specifications is not changed and therefore has no safety significance. The revisions will not lower the safety or effectiveness of the organization. Administrative changes have no effect on public health and safety.

Significant Hazards Determination for Proposed Changes

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes will not alter the intent of the TS. Reformatting the TS sections and correcting typographical, grammatical and format inconsistencies are administrative in nature. There is no impact on accident initiators or plant equipment, and therefore does not affect the probability or consequences of an accident

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes do not involve a change to the physical plant or operations. Since these are administrative changes they do not contribute to accident initiation. Therefore, the proposed changes do not produce a new accident scenario or produce a new type of equipment malfunction.

3. Involve a significant reduction in the margin of safety.

   Since these are administrative changes, they do not involve a significant reduction in the margin of safety. The proposed changes do not affect plant equipment or operation. Safety limits and limiting safety system settings are not affected by this change.

Environmental Considerations

NMC has determined that this proposed amendment involves no significant hazards considerations and no significant change in the types of any effluents that may be released off-site and that there is no significant rise in individual or cumulative occupational radiation exposure. Accordingly, this proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with this amendment.

ATTACHMENT 2

Letter from M. E. Warner (NMC)

to

Document Control Desk (NRC)

Dated

June 7, 2002

Proposed Amendment 184

Strike-Out TS Sections:

Table of Contents
Facility Operating License
TS Section 1.0
TS Section 2.0
TS Section 5.0
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Note:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

NUCLEAR MANAGEMENT COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

FACILITY OPERATING LICENSE AS AMENDED

License No. DPR-43

1. The Atomic Energy Commission (the Commission) having found that:

   A. The application for license filed by Wisconsin Public Service Corporation and Wisconsin Power and Light Company (the licensees) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Kewaunee Nuclear Power Plant (facility) has been substantially completed in conformity with Provisional Construction Permit No. CPPR-50, as amended, and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. The Nuclear Management Company, LLC (NMC) is technically qualified and the licensees are financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

   F. The licensees and NMC have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements~~",~~," of the Commission's regulations;

   G. The issuance of this operating license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of Facility Operating License No. DPR-43, subject to the condition for protection of the environment set forth herein, is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied; and

I. The receipt, possession, and use of byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30 and 70, including 10 CFR Section 30.33, 70.23 and 70.31.

2. Facility Operating License No. DPR-43 is hereby issued to NMC, Wisconsin Public Service Corporation and Wisconsin Power and Light Company, to read as follows:

A. This license applies to the Kewaunee Nuclear Power Plant, a pressurized water nuclear reactor and associated equipment (the facility), owned by Wisconsin Public Service Corporation and Wisconsin Power and Light Company. The facility is located in Kewaunee County, Wisconsin, and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 7 through 31) and the Environmental Report as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities~~",~~," Wisconsin Public Service Corporation and Wisconsin Power and Light Company to possess, and the NMC to use and operate the facility at the designated location in Kewaunee County, Wisconsin, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, NMC to receive, possess, and use at any time special nuclear material as reactor fuel in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NMC to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation, and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NMC to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or associated with radioactive apparatus or components~~:~~;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility~~"~~.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: (1) Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; (2) is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and (3) is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of 1650 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. [redacted] are hereby incorporated in the license. The NMC shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

The NMC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the KNPP Fire Plan, and as referenced in the Updated Safety Analysis Report, and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplement dated February 13, 1981) subject to the following provision:

The NMC may make changes to the approved Fire Protection Program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(4) Physical Protection

The NMC shall fully implement and maintain in effect all provisions of the Commission-approved "Kewaunee Nuclear Power Plant Security Manual," Rev. 1, approved by the NRC on December 15, 1989; the "Kewaunee Nuclear Power Plant Security Force Training and Qualification Manual," Rev. 7, approved by the NRC on November 17, 1987; and the "Kewaunee Nuclear Power Plant Security Contingency Plan," Rev. 1, approved by the NRC on September 1, 1983. These manuals include amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p).

(5) Fuel Burnup

The maximum rod average burnup for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit.

(6) Steam Generator Upper Lateral Supports

The design of the steam generator upper lateral supports may be modified by reducing the number of snubbers from four (4) to one (1) per steam generator.

(7) License Transfer

(A) WPSC shall take all necessary steps to ensure that the decommissioning trusts are maintained in accordance with the application for approval of the transfer of MG&E's ownership interest in KNPP to WPSC and the requirements of the Order approving the transfer, and consistent with the safety evaluation supporting the Order. Additionally, if the MG&E nonqualified fund is not transferred to WPSC, WPSC, or NMC acting on WPSC's behalf, shall explicitly include the status of the MG&E nonqualified fund in all future decommissioning funding status reports that WPSC, or NMC, submit in accordance with 10 CFR 50.75(f)(1).

(B) On the closing date of the transfer of MG&E's interests in KNPP to WPSC, MG&E shall transfer to WPSC all of MG&E's accumulated qualified decommissioning trust funds for KNPP. Immediately following such transfer, the amounts for radiological decommissioning of KNPP in WPSC's decommissioning trusts must, with respect to the interests in KNPP that WPSC would then hold, be at a level no less than the formula amounts under 10 CFR Section 50.75.

D. The NMC shall comply with applicable effluent limitations and other limitations and monitoring requirements, if any, specified pursuant to Section 401(d) of the Federal Water Pollution Control Act Amendments of 1972.

E. This license is effective as of the date of issuance, and shall expire at midnight on December 21, 2013.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:

Appendices A and B - Technical Specifications

Date of Issuance: December 21, 1973

## 1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

a. QUADRANT-TO-AVERAGE POWER TILT RATIO

The QUADRANT-TO-AVERAGE POWER TILT RATIO is defined as the ratio of maximum-to-average of the upper excore detector currents or that of the lower excore detector currents, whichever is greater. If one excore detector is out-of-service, then the three in-service units are used in computing the average.

b. SAFETY LIMITS

SAFETY LIMITS are the necessary quantitative restrictions placed upon those process variables that must be controlled in order to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

c. LIMITING SAFETY SYSTEM SETTINGS

LIMITING SAFETY SYSTEM SETTINGS are setpoints for automatic protective devices responsive to the variables on which SAFETY LIMITS have been placed. These setpoints are so chosen that automatic protective actions will correct the most severe, anticipated abnormal situation so that a SAFETY LIMIT is not exceeded.

d. LIMITING CONDITIONS FOR OPERATION

LIMITING CONDITIONS FOR OPERATION are those restrictions on reactor operation, resulting from equipment performance capability, that must be enforced to ensure safe operation of the facility.

e. OPERABLE-OPERABILITY

A system or component is OPERABLE or has OPERABILITY when it is capable of performing its intended function within the required range. The system or component shall be considered to have this capability when: (1) it satisfies the LIMITING CONDITIONS FOR OPERATION defined in TS 3.0, and (2) it has been tested periodically in accordance with TS 4.0 and has met its performance requirements.

Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that is required for the system or component to perform its intended function is also capable of performing their related support functions.

f. OPERATING

A system or component is considered to be OPERATING when it is performing the intended function in the intended manner.

g. CONTAINMENT SYSTEM INTEGRITY

CONTAINMENT SYSTEM INTEGRITY is defined to exist when:

1. The non-automatic Containment System isolation valves and blind flanges are closed, except as provided in TS 3.6.b.

2. The ~~R~~reactor ~~C~~containment ~~V~~vessel and ~~S~~shield ~~B~~building equipment hatches are properly closed.

3. At least ~~ONE~~ one door in both the personnel and the emergency airlocks is properly closed.

4. The required automatic Containment System isolation valves are OPERABLE, except as provided in TS 3.6.b.

5. All requirements of TS 4.4 with regard to Containment System leakage and test frequency are satisfied.

6. The Shield Building Ventilation System and the Auxiliary Building Special Ventilation System satisfy the requirements of TS 3.6.c.

h. PROTECTIVE INSTRUMENTATION LOGIC

1. PROTECTION SYSTEM CHANNEL

A PROTECTION SYSTEM CHANNEL is an arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. The channel loses its identity where single action signals are combined.

2. LOGIC CHANNEL

A LOGIC CHANNEL is a matrix of relay contacts which operate in response to PROTECTIVE SYSTEM CHANNEL signals to generate a protective action signal.

3. DEGREE OF REDUNDANCY

DEGREE OF REDUNDANCY is defined as the difference between the number of OPERATING channels and the minimum number of channels which, when tripped, will cause an automatic shutdown.

4. PROTECTION SYSTEM

The PROTECTION SYSTEM consists of both the Reactor PROTECTION SYSTEM and the Engineered Safety Features System. The PROTECTION SYSTEM encompasses all electric and mechanical devices and circuitry (from sensors through actuated device) which are required to operate in order to produce the required protective function. Tests of the PROTECTION SYSTEM will be considered acceptable when tests are run in part and it can be shown that all parts satisfy the requirements of the system.

i. INSTRUMENTATION SURVEILLANCE

1. CHANNEL CHECK

CHANNEL CHECK is a qualitative determination of acceptable OPERABILITY by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication with other indications derived from independent channels measuring the same variable.

2. CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST consists of injecting a simulated signal into the channel as close to the primary sensor as practicable to verify that it is OPERABLE, including alarm and/or trip initiating action.

3. CHANNEL CALIBRATION

CHANNEL CALIBRATION consists of the adjustment of channel output as necessary, such that it responds~~,~~ with acceptable range and accuracy~~,~~ to known values of the parameter ~~which~~ that the channel monitors. Calibration shall encompass the entire channel, including alarm and/or trip, and shall be deemed to include the CHANNEL FUNCTIONAL TEST.

4. SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

5. FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of surveillance requirements shall correspond to the intervals in Table TS 1.0-1.

j. MODES

| MODE | REACTIVITY Δk/k | COOLANT TEMP $T_{avg}$ °F | FISSION POWER % |
|---|---|---|---|
| REFUELING | ≤ -5% | ≤ 140 | ~0 |
| COLD SHUTDOWN | ≤ -1% | ≤ 200 | ~0 |
| INTERMEDIATE SHUTDOWN | (1) | > 200 < 540 | ~0 |
| HOT SHUTDOWN | (1) | ≥ 540 | ~0 |
| HOT STANDBY | < 0.25% | ~$T_{oper}$ | < 2 |
| OPERATING | < 0.25% | ~$T_{oper}$ | ≥ 2 |
| LOW POWER PHYSICS TESTING | (To be specified by specific tests) | | |
| (1) Refer to Figure TS 3.10-1 | | | |

k. REACTOR CRITICAL

The reactor is said to be critical when the neutron chain reaction is self-sustaining.

l. REFUELING OPERATION

REFUELING OPERATION is any operation involving movement of reactor vessel internal components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

m. RATED POWER

RATED POWER is the steady-state reactor core output of 1,650 MWt.

n. REPORTABLE EVENT

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

o. RADIOLOGICAL EFFLUENTS

1. MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

2. OFF-SITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall contain the current methodology and parameters used in: (1) the calculation of off-site doses due to radioactive gaseous and liquid effluents, ~~and~~ (2) ~~in~~ the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and ~~in~~ (3) the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain: (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by TS 6.16.b, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by TS 6.9.b.1 and TS 6.9.b.2.

3. PROCESS CONTROL PROGRAM (PCP)

The PCP shall contain the current formulae, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes, based on demonstrated processing of actual or simulated wet solid wastes, will be accomplished in such a way as to ~~as~~ensure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71, federal and state regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.

4. SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

5. UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

p. DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 is that concentration of I-131 ($\mu$Ci/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be as listed and calculated with the methodology established in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

| DOSE CONVERSION FACTOR | ISOTOPE |
|---|---|
| 1.0000 | I-131 |
| 0.0361 | I-132 |
| 0.2703 | I-133 |
| 0.0169 | I-134 |
| 0.0838 | I-135 |

## 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

## 2.1 SAFETY LIMITS~~,~~ - REACTOR CORE

### APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during operation.

### OBJECTIVE

To maintain the integrity of the fuel cladding.

### SPECIFICATION

The combination of ~~rated power~~RATED POWER level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure TS 2.1-1. The SAFETY LIMIT is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

## BASIS - Safety Limits, -Reactor Core (TS 2.1)

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to the DNBR limit. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The curves of Figure TS 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNBR is equal to the DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ~~as~~ensured is below these lines.

The curves are based on the nuclear hot channel factor limits of TS 3.10.b.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure TS 3.10-3 ~~i~~ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the DNBR limit.

## 2.2 SAFETY LIMIT, - REACTOR COOLANT SYSTEM PRESSURE

### APPLICABILITY

Applies to the maximum limit on Reactor Coolant System pressure.

### OBJECTIVE

To maintain the integrity of the Reactor Coolant System.

### SPECIFICATION

The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

## BASIS - Safety Limit - Reactor Coolant System Pressure (TS 2.2)

The Reactor Coolant System[1] serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is ensured. The maximum transient pressure allowable in the reactor pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USASI B.31.1.0 is 120% of design pressure. Thus, the SAFETY LIMIT of 2735 psig (110% of design pressure, 2485 psig) has been established.[2]

The settings of the power-operated relief valves, the reactor high pressure trip and the safety valves have been established to prevent exceeding the SAFETY LIMIT of 2735 psig. The initial hydrostatic test was conducted at 3107 psig to ensure the integrity of the Reactor Coolant System.

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[1] USAR Section 4
[2] USAR Section 4.3

## 2.3 LIMITING SAFETY SYSTEM SETTINGS, - PROTECTIVE INSTRUMENTATION

### APPLICABILITY

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection.

### OBJECTIVE

To prevent the principal process variables from exceeding a SAFETY LIMIT.

### SPECIFICATION

a. Reactor trip settings shall be as follows:

1. Nuclear Flux

| | |
|---|---|
| A. Source Range (high setpoint) | within span of source range instrumentation |
| B. Intermediate range (high setpoint) | ≤ 40% of RATED POWER |
| C. Power range (low setpoint) | ≤ 25% of RATED POWER |
| D. Power range (high setpoint) | ≤ 109% of RATED POWER |
| E. Power range fast flux rate trip (positive) | 15%Δq/5 sec |
| F. Power range fast flux rate trip (negative) | 10%Δq/5 sec |

2. Pressurizer

| | |
|---|---|
| A. High pressurizer pressure | ≤ 2385 psig |
| B. Low pressurizer pressure | ≥ 1875 psig |
| C. High pressurizer water level | ≤ 90% of full scale |

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \left[ K_1 - K_2 ( T - T' ) \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 ( P - P' ) - f ( \Delta I ) \right]$$

where

| | | |
|---|---|---|
| $\Delta T_0$ | = | Indicated $\Delta T$ at RATED POWER, °F |
| T | = | Average temperature, °F |
| T' | = | 567.3°F |
| P | = | Pressurizer pressure, psig |
| P' | = | 2235 psig |
| $K_1$ | = | 1.11 |
| $K_2$ | = | 0.0090 |
| $K_3$ | = | 0.000566 |
| $\tau_1$ | = | 30 sec. |
| $\tau_2$ | = | 4 sec. |
| $F^2(\Delta I)$ | = | An even function of the indicated difference between top and bottom detectors of the power- range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where $q_t$ and $q_b$ are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that: |

~~(a)~~1. For $q_t - q_b$ within -12, +9%, f ($\Delta$~~I~~I) = 0.

~~(b)~~2. For each percent that the magnitude of $q_t - q_b$ exceeds +9% the $\Delta T$ trip setpoint shall be automatically reduced by an equivalent of 2.5% of RATED POWER.

~~(c)~~3. For each percent that the magnitude of $q_t - q_b$ exceed -12% the $\Delta T$ trip setpoint shall be automatically reduced by an equivalent of 1.5% of RATED POWER.

B. Overpower

$$\Delta T \leq \Delta T_0 \left[ K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 ( T - T' ) - f ( \Delta I ) \right]$$

where

$\Delta T_0$ = Indicated $\Delta T$ at RATED POWER, °F

T = Average Temperature, °F

T' = 567.3°F

$K_4$ $\leq$ 1.10

$K_5$ $\geq$ 0.0275 for increasing T; 0 for decreasing T

$K_6$ $\geq$ 0.002 for T > T'; 0 for T < T'

$\tau_3$ = 10 sec.

$f(\Delta I)$ = As defined above

4. Reactor Coolant Flow

A. Low reactor coolant flow per loop $\geq$ 90% of normal indicated flow as measured by elbow taps.

B. Reactor coolant pump motor breaker open

1. Low frequency setpoint $\geq$ 55.0 Hz

2. Low voltage setpoint $\geq$ 75% of normal voltage

5. Steam Generators

Low-low steam generator water level $\geq$ 5% of narrow range instrument span.

6. Reactor Trip Interlocks

Protective instrumentation settings for reactor trip interlocks shall be as follows:

A. Above 10% of RATED POWER, the low pressurizer pressure trip, high pressurizer level trip, the low reactor coolant flow trips (for both loops), and the turbine trip-reactor trip are made functional.

B. Above 10% of RATED POWER, the single-loop loss-of-flow trip is made functional.

7. Other Trips

A. Undervoltage ≥ 75% of normal voltage

B. Turbine trip

C. Manual trip

D. Safety injection trip (Refer to Table TS 3.5-1 for trip settings)

## BASIS - Limiting Safety System Settings - Protective Instrumentation (TS 2.3)

### Nuclear Flux

The source range high flux reactor trip prevents a startup accident from subcritical conditions from proceeding into the power range. Any setpoint within its range would prevent an excursion from proceeding to the point at which significant thermal power is generated.

The power range reactor trip low setpoint provides protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis.[1]

The power range reactor trip high setpoint protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis.[2]

Two sustained rate protective trip functions have been incorporated in the Reactor PROTECTION SYSTEM. The positive sustained rate trip provides protection against hypothetical rod ejection accident. The negative sustained rate trip provides protection for the core (low DNBR) in the event two or more rod control cluster assemblies (RCCAs) fall into the core. The circuits are independent and ensure immediate reactor trip independent of the initial OPERATING state of the reactor. These trip functions are the LIMITING SAFETY SYSTEM actions employed in the accident analysis.

### Pressurizer

The high and low pressure trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure trip causes a reactor trip in the unlikely event of a loss-of-coolant accident.[3] The high pressurizer water level trip protects the pressurizer safety valves against water relief. The specified setpoint allows margin for instrument error [2] and transient level overshoot before the reactor trips.

### Reactor Coolant Temperature

The overtemperature $\Delta T$ reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that: 1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 2 seconds), and 2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors[2] is always below the core SAFETY LIMITS shown in Figure TS 2.1-1. If axial peaks are greater than design, as indicated by differences between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.

---

[1] USAR Section 14.1.1
[2] USAR Section 14.0
[3] USAR Section 14.3.1

The overpower $\Delta T$ reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur, and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

The overpower and overtemperature PROTECTION SYSTEM setpoints include the effects of fuel densification and clad flattening on core SAFETY LIMITS.[4]

Reactor Coolant Flow

The low-flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[5]

The undervoltage and low frequency reactor trips provide additional protection against a decrease in flow. The undervoltage setting provides a direct reactor trip and a reactor coolant pump breaker trip. The undervoltage setting ensures a reactor trip signal will be generated before the low-flow trip setting is reached. The low frequency setting provides only a reactor coolant pump breaker trip.

Steam Generators

The low-low steam generator water level reactor trip ensures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting the Auxiliary Feedwater System. [6]

Reactor Trip Interlocks

Specified reactor trips are bypassed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed setpoints above which these trips are made functional ensures their availability in the power range where needed. Confirmation that bypasses are automatically removed at the prescribed setpoints will be determined by periodic testing. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10% of power.

Table TS 3.5-1 lists the various parameters and their setpoints which initiate safety injection signals. A safety injection signal (SIS) also initiates a reactor trip signal. The periodic testing will verify that safety injection signals perform their intended function. Refer to the basis of Section 3.5 of these specifications for details of SIS signals.

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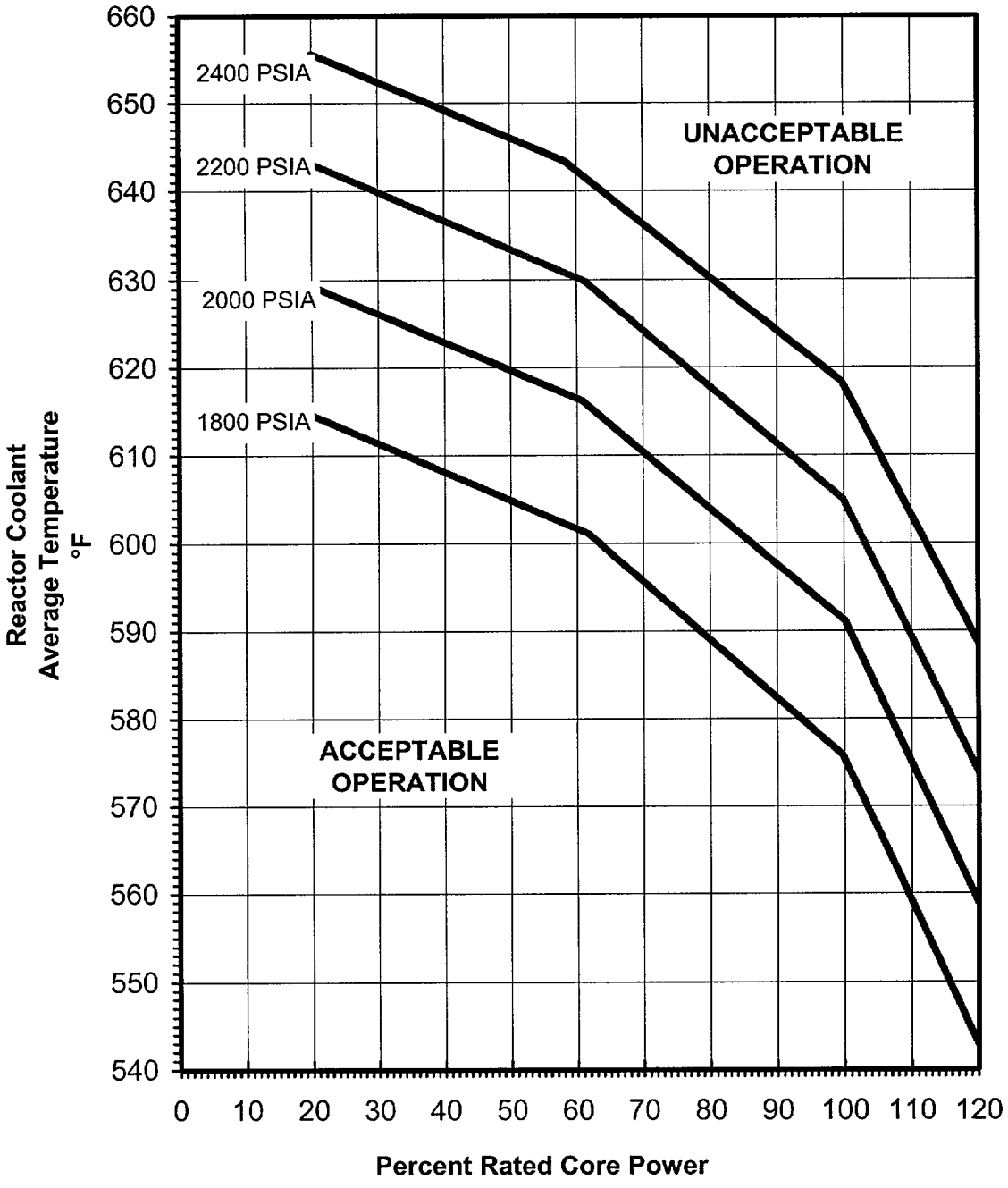
[4] WCAP-8092
[5] USAR Section 14.1.8
[6] USAR Section 14.1.10

FIGURE TS 2.1-1

**Safety Limits Reactor Core - Minimum Coolant System Flow**

**(TS 3.10.m) - Minimum DNBR**



## 5.0 DESIGN FEATURES

## 5.1 SITE

### APPLICABILITY

Applies to the location and extent of the reactor site.

### OBJECTIVE

To define those aspects of the site which affect the overall safety of the installation.

### SPECIFICATION

The Kewaunee Nuclear Power Plant is located on property owned by Wisconsin Public Service Corporation and Wisconsin Power and Light Company at a site on the west shore of Lake Michigan, approximately 30 miles east-southeast of the city of Green Bay, Wisconsin.

The minimum distance from the center line of the reactor containment to the site exclusion radius as defined in 10 CFR 100.3 is 1200 meters.

## 5.2 CONTAINMENT

### APPLICABILITY

Applies to those design features of the Containment System relating to operational and public safety.

### OBJECTIVE

To define the significant design features of the Containment System.

### SPECIFICATION

a. Containment System

1. The Containment System completely encloses the entire reactor and the Reactor Coolant System and ensures that leakage of activity is limited, filtered and delayed such that off-site doses resulting from the ~~D~~design ~~B~~basis ~~A~~accident are within the guidelines of 10 CFR Part 100. The Containment System provides biological shielding for both normal OPERATING conditions and accident situations.

2. The Containment System consists of:

   A. A free-standing steel ~~R~~reactor ~~C~~containment ~~V~~vessel designed for the peak pressure of the ~~D~~design ~~B~~basis ~~A~~accident.

   B. A concrete ~~S~~shield ~~B~~building which surrounds the ~~C~~containment ~~V~~vessel, providing a ~~S~~shield ~~B~~building annulus between the two structures.

   C. A Shield Building Ventilation System ~~which~~ that causes leakage from the ~~R~~reactor ~~C~~containment ~~V~~vessel to be delayed and filtered before its release to the environment.

   D. An Auxiliary Building Special Ventilation System ~~which~~ that serves the ~~S~~special ~~V~~ventilation ~~Z~~zone and supplements the Shield Building Ventilation System during an accident condition by causing any leakage from the Residual Heat Removal System (RHRS) and certain small amounts of leakage ~~which~~ that might be postulated to bypass the Shield Building Ventilation System to be filtered before their release.

b. Reactor Containment Vessel

1. The ~~R~~reactor ~~C~~containment ~~V~~vessel is designed for the peak internal pressure of the ~~D~~design ~~B~~basis ~~A~~accident plus the loads resulting from an earthquake producing 0.06g horizontally and 0.04g vertically. It is also designed to withstand an external pressure 0.8 psi greater than the internal pressure.

2. Penetrations of the ~~C~~containment ~~V~~vessel for piping, electrical conductors, ducts and access hatches are provided with double barriers against leakage.

3. The automatically actuated containment valves are designed to close upon high containment pressure and on a safety injection signal. The actuation system is designed so that no single component failure will prevent containment isolation, if required.

c. Shield Building

The ~~S~~shield ~~B~~building is a reinforced concrete structure with a wall thickness of 2.5 feet and a dome thickness of 2 feet. It is designed for the same seismic conditions as the ~~R~~reactor ~~C~~containment ~~V~~vessel and is designed to resist a 3 psi internal pressure due to tornadoes.

d. Shield Building Ventilation System

In the event of a loss-of-coolant accident, the Shield Building Ventilation System will relieve the initial thermal expansion of air through particulate and charcoal filters and will then cause a vacuum to be produced throughout the ~~S~~shield ~~B~~building annulus. A momentary positive pressure no greater than 0.5 psi will result during the thermal expansion. Once vacuum is achieved, the system causes the air within the annulus to be recirculated through the filters while vacuum is maintained. The filtered mixture of annulus air plus leakage is vented through the Containment System ~~V~~vent by the discharge fan that maintains vacuum at a vent rate determined by in-leakage to the ~~S~~shield ~~B~~building.[1]

e. Auxiliary Building Special Ventilation Zone ~~&~~ and Special Ventilation System

A limited amount of containment leakage could potentially escape through certain penetrations in the event of leakage in the isolation valves, as described in the Basis of TS 3.6. The leakage escaping into that portion of the ~~A~~auxiliary ~~B~~building which is designed for medium leakage and controlled access would be processed by the Auxiliary Building Special Ventilation System. When actuated, the system will draw all in-leakage air from this ~~S~~special ~~V~~ventilation ~~Z~~zone and exhaust it through particulate and charcoal filters to the ~~A~~auxiliary ~~B~~building ~~V~~vent.[2]

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[1] USAR Section 5.5
[2] USAR Section 9.6

## 5.3 REACTOR CORE

### APPLICABILITY

Applies to the reactor core.

### OBJECTIVE

To define those design features which are essential in providing for safe reactor core operations.

### SPECIFICATION

a. Fuel Assemblies

The reactor shall contain 121 fuel assemblies. Each assembly shall consist of a matrix of zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead-test-assemblies that have not completed representative testing may be placed in non-limiting core regions. Lead-test-assemblies shall be of designs approved by the NRC for use in pressurized water reactors and their clad materials shall be the materials approved as part of those designs.

b. Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

## 5.4 FUEL STORAGE

### APPLICABILITY

Applies to the capacity and storage arrays of new and spent fuel.

### OBJECTIVE

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

### SPECIFICATION

a. Criticality

1. The spent fuel storage racks are designed and shall be maintained with the following:

   a. Fuel assemblies having a maximum enrichment of 56.067 grams Uranium-235 per axial centimeter~~;~~

   b. $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties~~.~~

2. The new fuel storage racks are designed and shall be maintained with:

   a. Fuel assemblies having a maximum enrichment of 56.067 grams Uranium-235 per axial centimeter~~;~~

   b. $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties~~;~~

   c. $k_{eff} < 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties~~.~~

3. The spent fuel pool is filled with borated water at a concentration to match that used in the reactor REFUELING cavity and REFUELING canal during REFUELING OPERATIONS or whenever there is fuel in the pool.

b. Capacity

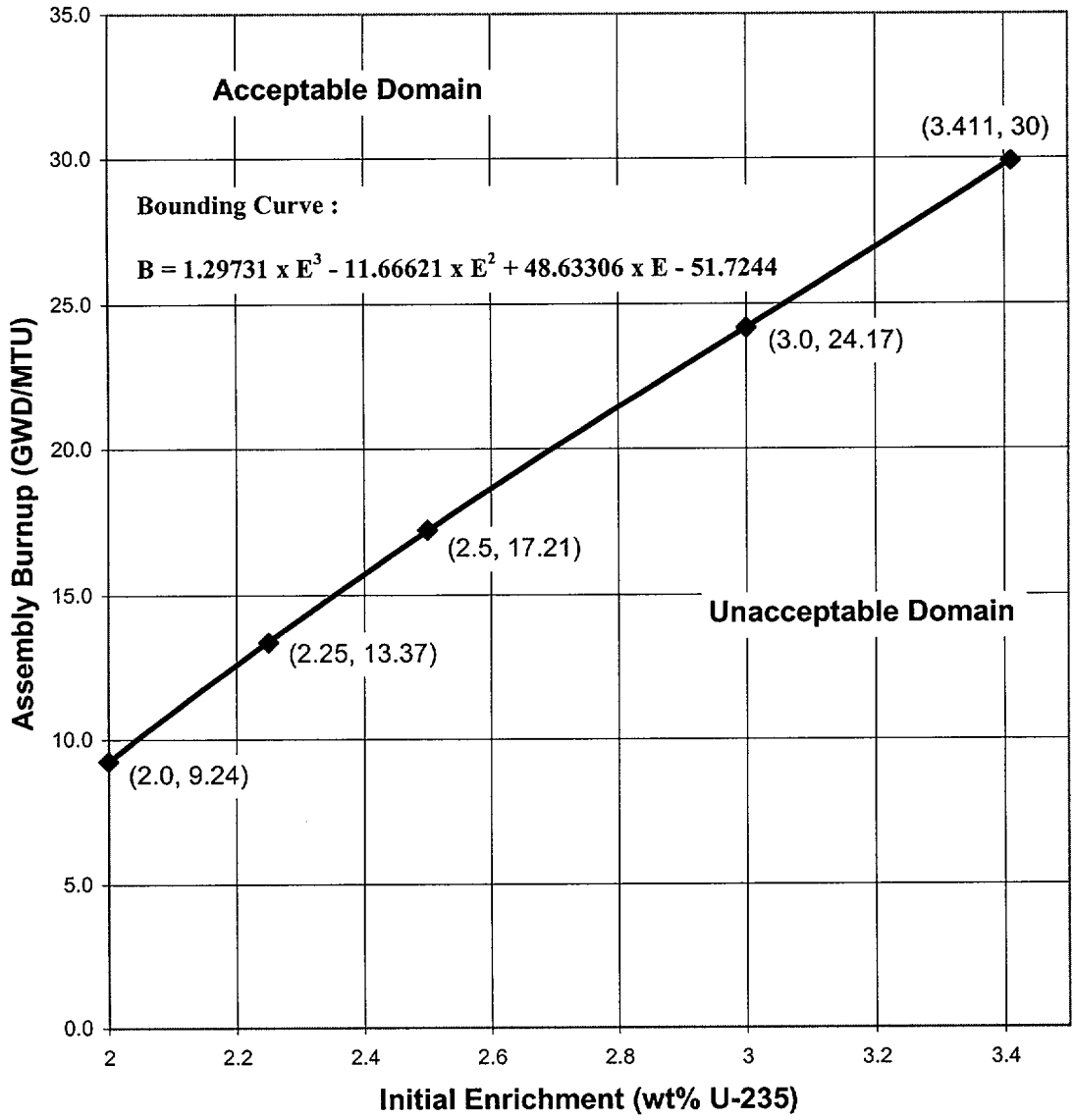
The spent fuel storage pool is designed with a storage capacity of 1205 assemblies and shall be limited to no more than 1205 fuel assemblies.

c. Canal Rack Storage

Fuel assemblies stored in the canal racks shall meet the minimum required fuel assembly burnup as a function of nominal initial enrichment as shown in Figure TS 5.4-1. These assemblies shall also have been discharged prior to or during the 1984 REFUELING outage.

FIGURE TS 5.4-1

MINIMUM REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF NOMINAL INITIAL ENRICHMENT TO PERMIT STORAGE IN THE TRANSFER CANAL



## 6.0 ADMINISTRATIVE CONTROLS

### 6.1 RESPONSIBILITY

a. The Manager - Kewaunee Plant shall be responsible for overall plant operation and shall delegate in writing the succession of this responsibility during his absence.

b. The Manager - Kewaunee Plant, or his designee, shall approve prior to implementation, each proposed test, experiment or modification to structures, systems or components that affect nuclear safety.

## 6.10 RECORD RETENTION

a. The following records shall be retained for at least five years:

1. Records and logs of plant operation, including power levels and periods of operation at each power level.

2. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment pertaining to nuclear safety.

3. Reports of all REPORTABLE EVENTS.

4. Records of periodic checks, inspections, and calibrations required by these Technical Specifications.

5. Records of nuclear safety-related tests or experiments.

6. Records of radioactive shipments.

7. Records of changes to OPERATING procedures.

8. Records of sealed source leak tests and results.

9. Records of annual physical inventory of all source material of record.

10. Records of Quality Assurance activities required by the Operational Quality Assurance Program (OQAP) except where it is determined that the records should be maintained for a longer period of time.

b. The following records shall be retained for the duration of the Plant Operating License.

1. Records of a complete set of as-built drawings for the plant as originally licensed and all print changes showing modifications made to the plant.

2. Records of new and spent fuel inventory, fuel transfers, and assembly burnup histories.

3. Records of plant radiation and contamination surveys.

4. Records of radiation exposure of all plant personnel, and others who enter radiation control areas.

5. Records of radioactivity in liquid and gaseous wastes released to the environment.

6. Records of transient or operational cycles for these facility components.

7. Records of training and qualification for current members of the plant staff.

8. Records of in-service inspections performed pursuant to these Technical Specifications.

9. Records of meetings of the JOSRC and PORC.

10. Records for ~~E~~environmental ~~Q~~qualification.

11. Records of reviews performed for changes made to the ODCM and the PCP.

## 6.11 RADIATION PROTECTION PROGRAM

a. Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

b. Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne in-plant iodine concentrations under accident conditions. This program shall include the following:

1. Training of personnel~~,~~
2. Procedures for monitoring~~, and~~
3. Provisions for maintenance of sampling and analysis equipment~~.~~

## 6.12 SYSTEM INTEGRITY

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

a. Provisions establishing preventive maintenance and periodic visual inspection requirements~~, and~~.

b. Integrated leak test requirements for each system at a frequency not to exceed REFUELING cycle intervals.

## 6.13 HIGH RADIATION AREA

a. In lieu of the "control device" or "alarm signal" required by Paragraph 20.1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is > 100 mrem/hr, but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a ~~R~~radiation ~~W~~work ~~P~~permit (RWP).[1] Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following.

   1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   2. A radiation monitoring device which continuously integrates the radiation dose in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

   3. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the RWP.

b. In addition to the requirements of 6.13.a., areas accessible to personnel with radiation levels such that a major portion of the body could receive in ~~1~~ one hour a dose > 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in ~~1~~ one hour a dose > 1000 mrem[2] that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

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[1] Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

[2] Measurement made at 30 centimeters from source of radioactivity.

## 6.14 DELETED

## 6.15 SECONDARY WATER CHEMISTRY

The licensee shall implement a secondary water chemistry monitoring program. The intent of this program will be to control corrosion thereby inhibiting steam generator tube degradation. The secondary water chemistry program shall act as a guide for the chemistry group in their routine as well as non-routine activities.

## 6.16 RADIOLOGICAL EFFLUENTS

a. Written procedures shall be established, implemented and maintained covering the activities referenced below:

1. Process Control Program (PCP) implementation~~.~~

2. OFF-SITE DOSE CALCULATION MANUAL (ODCM) implementation~~.~~

3. Quality Assurance Program for effluent and environmental monitoring~~.~~

b. The following programs shall be established, implemented, and maintained:

1. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program ~~(1)~~ shall: (1) be contained in the ODCM, (2) ~~shall~~ be implemented by OPERATING procedures, and (3) ~~shall~~ include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

~~(~~A.~~)~~ Limitations on the OPERABILITY of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM.

~~(~~B~~)~~. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2.

~~(~~C~~)~~. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM.

~~(~~D~~)~~. Limitations on the annual and quarterly doses or dose commitment to a MEMBER(S) OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50.

~~(~~E~~)~~. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days.

~~(F)~~. Limitations on the OPERABILITY and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2% of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50.

~~(G)~~. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1.

~~(H)~~. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50.

~~(I)~~. Limitations on the annual and quarterly doses to ~~a~~ MEMBER(S) OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than ~~8~~ eight days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50.

~~(J)~~. Limitations on the annual dose or dose commitment to any MEMBER(S) OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

2. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide: (1) representative measurement of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall: (1) be contained in the ODCM~~,~~ (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

~~(A)~~. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

~~(B)~~. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census~~, and~~.

~~(C)~~. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the Quality Assurance Program for environmental monitoring.

## 6.17 PROCESS CONTROL PROGRAM (PCP)

a. The PCP shall be approved by the Commission prior to implementation.

b. Licensee initiated changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained as required by TS 6.10.b.11. The documentation shall contain:

   ~~(a)~~A. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s)~~, and~~.

   ~~(b)~~B. A determination that the change will maintain the overall conformance of the soldified waste product to existing requirements of federal, state, or other applicable regulations.

2. Shall become effective upon review and acceptance by the PORC.

## 6.18 OFF-SITE DOSE CALCULATION MANUAL (ODCM)

a. The ODCM shall be approved by the Commission prior to implementation.

b. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained as required by TS 6.10.b.11. This documentation shall contain:

   ~~(a)~~A. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change~~(s), and~~.

   ~~(b)~~B. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after review and acceptance by the PORC.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. The date the changes were made shall be indicated. In addition, a method such as redlining should be used to clearly identify the changes.

ATTACHMENT 3

Letter from M. E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

June 7, 2002

Proposed Amendment 184

TS Affected Revised Sections:

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TS Section 2.0
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Note:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

NUCLEAR MANAGEMENT COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

FACILITY OPERATING LICENSE AS AMENDED

License No. DPR-43

1. The Atomic Energy Commission (the Commission) having found that:

   A. The application for license filed by Wisconsin Public Service Corporation and Wisconsin Power and Light Company (the licensees) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Kewaunee Nuclear Power Plant (facility) has been substantially completed in conformity with Provisional Construction Permit No. CPPR-50, as amended, and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. The Nuclear Management Company, LLC (NMC) is technically qualified and the licensees are financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

   F. The licensees and NMC have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this operating license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of Facility Operating License No. DPR-43, subject to the condition for protection of the environment set forth herein, is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied; and

I. The receipt, possession, and use of byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30 and 70, including 10 CFR Section 30.33, 70.23 and 70.31.

2. Facility Operating License No. DPR-43 is hereby issued to NMC, Wisconsin Public Service Corporation and Wisconsin Power and Light Company, to read as follows:

A. This license applies to the Kewaunee Nuclear Power Plant, a pressurized water nuclear reactor and associated equipment (the facility), owned by Wisconsin Public Service Corporation and Wisconsin Power and Light Company. The facility is located in Kewaunee County, Wisconsin, and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 7 through 31) and the Environmental Report as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," Wisconsin Public Service Corporation and Wisconsin Power and Light Company to possess, and the NMC to use and operate the facility at the designated location in Kewaunee County, Wisconsin, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, NMC to receive, possess, and use at any time special nuclear material as reactor fuel in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NMC to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation, and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NMC to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: (1) Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, (2) is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and (3) is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of 1650 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. [redacted] are hereby incorporated in the license. The NMC shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

The NMC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the KNPP Fire Plan, and as referenced in the Updated Safety Analysis Report, and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplement dated February 13, 1981) subject to the following provision:

The NMC may make changes to the approved Fire Protection Program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(4) Physical Protection

The NMC shall fully implement and maintain in effect all provisions of the Commission-approved "Kewaunee Nuclear Power Plant Security Manual," Rev. 1, approved by the NRC on December 15, 1989, the "Kewaunee Nuclear Power Plant Security Force Training and Qualification Manual," Rev. 7, approved by the NRC on November 17, 1987, and the "Kewaunee Nuclear Power Plant Security Contingency Plan," Rev. 1, approved by the NRC on September 1, 1983. These manuals include amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p).

(5) Fuel Burnup

The maximum rod average burnup for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit.

(6) Steam Generator Upper Lateral Supports

The design of the steam generator upper lateral supports may be modified by reducing the number of snubbers from four (4) to one (1) per steam generator.

(7) License Transfer

(A) WPSC shall take all necessary steps to ensure that the decommissioning trusts are maintained in accordance with the application for approval of the transfer of MG&E's ownership interest in KNPP to WPSC and the requirements of the Order approving the transfer, and consistent with the safety evaluation supporting the Order. Additionally, if the MG&E nonqualified fund is not transferred to WPSC, WPSC, or NMC acting on WPSC's behalf, shall explicitly include the status of the MG&E nonqualified fund in all future decommissioning funding status reports that WPSC, or NMC, submit in accordance with 10 CFR 50.75(f)(1).

(B) On the closing date of the transfer of MG&E's interests in KNPP to WPSC, MG&E shall transfer to WPSC all of MG&E's accumulated qualified decommissioning trust funds for KNPP. Immediately following such transfer, the amounts for radiological decommissioning of KNPP in WPSC's decommissioning trusts must, with respect to the interests in KNPP that WPSC would then hold, be at a level no less than the formula amounts under 10 CFR Section 50.75.

D. The NMC shall comply with applicable effluent limitations and other limitations and monitoring requirements, if any, specified pursuant to Section 401(d) of the Federal Water Pollution Control Act Amendments of 1972.

E. This license is effective as of the date of issuance, and shall expire at midnight on December 21, 2013.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:

Appendices A and B - Technical Specifications

Date of Issuance: December 21, 1973

## 1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

a. QUADRANT-TO-AVERAGE POWER TILT RATIO

The QUADRANT-TO-AVERAGE POWER TILT RATIO is defined as the ratio of maximum-to-average of the upper excore detector currents or that of the lower excore detector currents, whichever is greater. If one excore detector is out-of-service, then the three in-service units are used in computing the average.

b. SAFETY LIMITS

SAFETY LIMITS are the necessary quantitative restrictions placed upon those process variables that must be controlled in order to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

c. LIMITING SAFETY SYSTEM SETTINGS

LIMITING SAFETY SYSTEM SETTINGS are setpoints for automatic protective devices responsive to the variables on which SAFETY LIMITS have been placed. These setpoints are so chosen that automatic protective actions will correct the most severe, anticipated abnormal situation so that a SAFETY LIMIT is not exceeded.

d. LIMITING CONDITIONS FOR OPERATION

LIMITING CONDITIONS FOR OPERATION are those restrictions on reactor operation, resulting from equipment performance capability that must be enforced to ensure safe operation of the facility.

e. OPERABLE-OPERABILITY

A system or component is OPERABLE or has OPERABILITY when it is capable of performing its intended function within the required range. The system or component shall be considered to have this capability when: (1) it satisfies the LIMITING CONDITIONS FOR OPERATION defined in TS 3.0, and (2) it has been tested periodically in accordance with TS 4.0 and has met its performance requirements.

Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that is required for the system or component to perform its intended function is also capable of performing their related support functions.

f. OPERATING

A system or component is considered to be OPERATING when it is performing the intended function in the intended manner.

g. CONTAINMENT SYSTEM INTEGRITY

CONTAINMENT SYSTEM INTEGRITY is defined to exist when:

1. The non-automatic Containment System isolation valves and blind flanges are closed, except as provided in TS 3.6.b.

2. The reactor containment vessel and shield building equipment hatches are properly closed.

3. At least one door in both the personnel and the emergency airlocks is properly closed.

4. The required automatic Containment System isolation valves are OPERABLE, except as provided in TS 3.6.b.

5. All requirements of TS 4.4 with regard to Containment System leakage and test frequency are satisfied.

6. The Shield Building Ventilation System and the Auxiliary Building Special Ventilation System satisfy the requirements of TS 3.6.c.

h. PROTECTIVE INSTRUMENTATION LOGIC

1. PROTECTION SYSTEM CHANNEL

A PROTECTION SYSTEM CHANNEL is an arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. The channel loses its identity where single action signals are combined.

2. LOGIC CHANNEL

A LOGIC CHANNEL is a matrix of relay contacts which operate in response to PROTECTIVE SYSTEM CHANNEL signals to generate a protective action signal.

3. DEGREE OF REDUNDANCY

DEGREE OF REDUNDANCY is defined as the difference between the number of OPERATING channels and the minimum number of channels which, when tripped, will cause an automatic shutdown.

4. PROTECTION SYSTEM

The PROTECTION SYSTEM consists of both the Reactor PROTECTION SYSTEM and the Engineered Safety Features System. The PROTECTION SYSTEM encompasses all electric and mechanical devices and circuitry (from sensors through actuated device) which are required to operate in order to produce the required protective function. Tests of the PROTECTION SYSTEM will be considered acceptable when tests are run in part and it can be shown that all parts satisfy the requirements of the system.

i. INSTRUMENTATION SURVEILLANCE

1. CHANNEL CHECK

CHANNEL CHECK is a qualitative determination of acceptable OPERABILITY by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication with other indications derived from independent channels measuring the same variable.

2. CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST consists of injecting a simulated signal into the channel as close to the primary sensor as practicable to verify that it is OPERABLE, including alarm and/or trip initiating action.

3. CHANNEL CALIBRATION

CHANNEL CALIBRATION consists of the adjustment of channel output as necessary, such that it responds with acceptable range and accuracy to known values of the parameter that the channel monitors. Calibration shall encompass the entire channel, including alarm and/or trip, and shall be deemed to include the CHANNEL FUNCTIONAL TEST.

4. SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

5. FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of surveillance requirements shall correspond to the intervals in Table TS 1.0-1.

j. MODES

| MODE | REACTIVITY $\Delta k/k$ | COOLANT TEMP $T_{avg}$ °F | FISSION POWER % |
|---|---|---|---|
| REFUELING | $\leq$ -5% | $\leq$ 140 | ~0 |
| COLD SHUTDOWN | $\leq$ -1% | $\leq$ 200 | ~0 |
| INTERMEDIATE SHUTDOWN | (1) | > 200 < 540 | ~0 |
| HOT SHUTDOWN | (1) | $\geq$ 540 | ~0 |
| HOT STANDBY | < 0.25% | ~$T_{oper}$ | < 2 |
| OPERATING | < 0.25% | ~$T_{oper}$ | $\geq$ 2 |
| LOW POWER PHYSICS TESTING | (To be specified by specific tests) | | |
| (1) Refer to Figure TS 3.10-1 | | | |

k. REACTOR CRITICAL

The reactor is said to be critical when the neutron chain reaction is self-sustaining.

l. REFUELING OPERATION

REFUELING OPERATION is any operation involving movement of reactor vessel internal components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

m. RATED POWER

RATED POWER is the steady-state reactor core output of 1,650 MWt.

n. REPORTABLE EVENT

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

o. RADIOLOGICAL EFFLUENTS

1. MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

2. OFF-SITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall contain the current methodology and parameters used in: (1) the calculation of off-site doses due to radioactive gaseous and liquid effluents, (2) the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and (3) the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain: (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by TS 6.16.b, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by TS 6.9.b.1 and TS 6.9.b.2.

3. PROCESS CONTROL PROGRAM (PCP)

The PCP shall contain the current formulae, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes, based on demonstrated processing of actual or simulated wet solid wastes, will be accomplished in such a way as to ensure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71, federal and state regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.

4. SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

5. UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

p. DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 is that concentration of I-131 ($\mu$Ci/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be as listed and calculated with the methodology established in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

| DOSE CONVERSION FACTOR | ISOTOPE |
|---|---|
| 1.0000 | I-131 |
| 0.0361 | I-132 |
| 0.2703 | I-133 |
| 0.0169 | I-134 |
| 0.0838 | I-135 |

## TABLE TS 1.0-1

## FREQUENCY NOTATIONS

| NOTATION | FREQUENCY |
|---|---|
| Shift (S) | At least once per 12 hours |
| Daily (D) | At least once per 24 hours |
| Weekly (W) | At least once per 7 days |
| Monthly (M) | At least once per 31 days |
| Quarterly (Q) | At least once per 92 days |
| Semiannual (SA) | At least once per 184 days |
| Refueling (R) | At least once per 18 months |
| Reactor Startup (S/U) | Prior to each reactor startup |
| N.A. | Not Applicable |

## 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

## 2.1 SAFETY LIMITS - REACTOR CORE

### APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during operation.

### OBJECTIVE

To maintain the integrity of the fuel cladding.

### SPECIFICATION

The combination of RATED POWER level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure TS 2.1-1. The SAFETY LIMIT is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

## BASIS - Safety Limits-Reactor Core (TS 2.1)

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to the DNBR limit. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The curves of Figure TS 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNBR is equal to the DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below these lines.

The curves are based on the nuclear hot channel factor limits of TS 3.10.b.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure TS 3.10-3 ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the DNBR limit.

## 2.2 SAFETY LIMIT - REACTOR COOLANT SYSTEM PRESSURE

### APPLICABILITY

Applies to the maximum limit on Reactor Coolant System pressure.

### OBJECTIVE

To maintain the integrity of the Reactor Coolant System.

### SPECIFICATION

The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

**BASIS - Safety Limit - Reactor Coolant System Pressure (TS 2.2)**

The Reactor Coolant System[1] serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is ensured. The maximum transient pressure allowable in the reactor pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USASI B.31.1.0 is 120% of design pressure. Thus, the SAFETY LIMIT of 2735 psig (110% of design pressure, 2485 psig) has been established.[2]

The settings of the power-operated relief valves, the reactor high pressure trip and the safety valves have been established to prevent exceeding the SAFETY LIMIT of 2735 psig. The initial hydrostatic test was conducted at 3107 psig to ensure the integrity of the Reactor Coolant System.

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[1] USAR Section 4
[2] USAR Section 4.3

## 2.3 LIMITING SAFETY SYSTEM SETTINGS - PROTECTIVE INSTRUMENTATION

### APPLICABILITY

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection.

### OBJECTIVE

To prevent the principal process variables from exceeding a SAFETY LIMIT.

### SPECIFICATION

a. Reactor trip settings shall be as follows:

1. Nuclear Flux

| | |
|---|---|
| A. Source Range (high setpoint) | within span of source range instrumentation |
| B. Intermediate range (high setpoint) | $\leq$ 40% of RATED POWER |
| C. Power range (low setpoint) | $\leq$ 25% of RATED POWER |
| D. Power range (high setpoint) | $\leq$ 109% of RATED POWER |
| E. Power range fast flux rate trip (positive) | 15%$\Delta$q/5 sec |
| F. Power range fast flux rate trip (negative) | 10%$\Delta$q/5 sec |

2. Pressurizer

| | |
|---|---|
| A. High pressurizer pressure | $\leq$ 2385 psig |
| B. Low pressurizer pressure | $\geq$ 1875 psig |
| C. High pressurizer water level | $\leq$ 90% of full scale |

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \left[ K_1 - K_2 ( T - T' ) \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 ( P - P' ) - f ( \Delta I ) \right]$$

where

| | | |
|---|---|---|
| $\Delta T_0$ | = | Indicated $\Delta T$ at RATED POWER, °F |
| T | = | Average temperature, °F |
| T' | = | 567.3°F |
| P | = | Pressurizer pressure, psig |
| P' | = | 2235 psig |
| $K_1$ | = | 1.11 |
| $K_2$ | = | 0.0090 |
| $K_3$ | = | 0.000566 |
| $\tau_1$ | = | 30 sec. |
| $\tau_2$ | = | 4 sec. |
| $F^2(\Delta I)$ | = | An even function of the indicated difference between top and bottom detectors of the power range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where $q_t$ and $q_b$ are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that: |

1. For $q_t - q_b$ within -12, +9%, f ($\Delta I$) = 0.

2. For each percent that the magnitude of $q_t - q_b$ exceeds +9% the $\Delta T$ trip setpoint shall be automatically reduced by an equivalent of 2.5% of RATED POWER.

3. For each percent that the magnitude of $q_t - q_b$ exceed -12% the $\Delta T$ trip setpoint shall be automatically reduced by an equivalent of 1.5% of RATED POWER.

B. Overpower

$$\Delta T \leq \Delta T_0 \left[ K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 ( T - T' ) - f ( \Delta I ) \right]$$

where

$\Delta T_0$ = Indicated $\Delta T$ at RATED POWER, °F

T = Average Temperature, °F

T' = 567.3°F

$K_4$ $\leq$ 1.10

$K_5$ $\geq$ 0.0275 for increasing T; 0 for decreasing T

$K_6$ $\geq$ 0.002 for T > T'; 0 for T < T'

$\tau_3$ = 10 sec.

$f(\Delta I)$ = As defined above

4. Reactor Coolant Flow

A. Low reactor coolant flow per loop $\geq$ 90% of normal indicated flow as measured by elbow taps.

B. Reactor coolant pump motor breaker open

1. Low frequency setpoint $\geq$ 55.0 Hz

2. Low voltage setpoint $\geq$ 75% of normal voltage

5. Steam Generators

Low-low steam generator water level $\geq$ 5% of narrow range instrument span.

6. Reactor Trip Interlocks

Protective instrumentation settings for reactor trip interlocks shall be as follows:

A. Above 10% of RATED POWER, the low pressurizer pressure trip, high pressurizer level trip, the low reactor coolant flow trips (for both loops), and the turbine trip-reactor trip are made functional.

B. Above 10% of RATED POWER, the single loop loss-of-flow trip is made functional.

7. Other Trips

A. Undervoltage $\geq$ 75% of normal voltage

B. Turbine trip

C. Manual trip

D. Safety injection trip (Refer to Table TS 3.5-1 for trip settings)

## BASIS - Limiting Safety System Settings - Protective Instrumentation (TS 2.3)

### Nuclear Flux

The source range high flux reactor trip prevents a startup accident from subcritical conditions from proceeding into the power range. Any setpoint within its range would prevent an excursion from proceeding to the point at which significant thermal power is generated.

The power range reactor trip low setpoint provides protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis.[1]

The power range reactor trip high setpoint protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis.[2]

Two sustained rate protective trip functions have been incorporated in the Reactor PROTECTION SYSTEM. The positive sustained rate trip provides protection against hypothetical rod ejection accident. The negative sustained rate trip provides protection for the core (low DNBR) in the event two or more rod control cluster assemblies (RCCAs) fall into the core. The circuits are independent and ensure immediate reactor trip independent of the initial OPERATING state of the reactor. These trip functions are the LIMITING SAFETY SYSTEM actions employed in the accident analysis.

### Pressurizer

The high and low pressure trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure trip causes a reactor trip in the unlikely event of a loss-of-coolant accident.[3] The high pressurizer water level trip protects the pressurizer safety valves against water relief. The specified setpoint allows margin for instrument error [2] and transient level overshoot before the reactor trips.

### Reactor Coolant Temperature

The overtemperature $\Delta T$ reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that: 1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 2 seconds), and 2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors[2] is always below the core SAFETY LIMITS shown in Figure TS 2.1-1. If axial peaks are greater than design, as indicated by differences between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.

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[1] USAR Section 14.1.1
[2] USAR Section 14.0
[3] USAR Section 14.3.1

The overpower $\Delta T$ reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur, and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

The overpower and overtemperature PROTECTION SYSTEM setpoints include the effects of fuel densification and clad flattening on core SAFETY LIMITS.[4]

Reactor Coolant Flow

The low-flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[5]

The undervoltage and low frequency reactor trips provide additional protection against a decrease in flow. The undervoltage setting provides a direct reactor trip and a reactor coolant pump breaker trip. The undervoltage setting ensures a reactor trip signal will be generated before the low-flow trip setting is reached. The low frequency setting provides only a reactor coolant pump breaker trip.

Steam Generators

The low-low steam generator water level reactor trip ensures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting the Auxiliary Feedwater System. [6]

Reactor Trip Interlocks

Specified reactor trips are bypassed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed setpoints above which these trips are made functional ensures their availability in the power range where needed. Confirmation that bypasses are automatically removed at the prescribed setpoints will be determined by periodic testing. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10% of power.

Table TS 3.5-1 lists the various parameters and their setpoints which initiate safety injection signals. A safety injection signal (SIS) also initiates a reactor trip signal. The periodic testing will verify that safety injection signals perform their intended function. Refer to the basis of Section 3.5 of these specifications for details of SIS signals.

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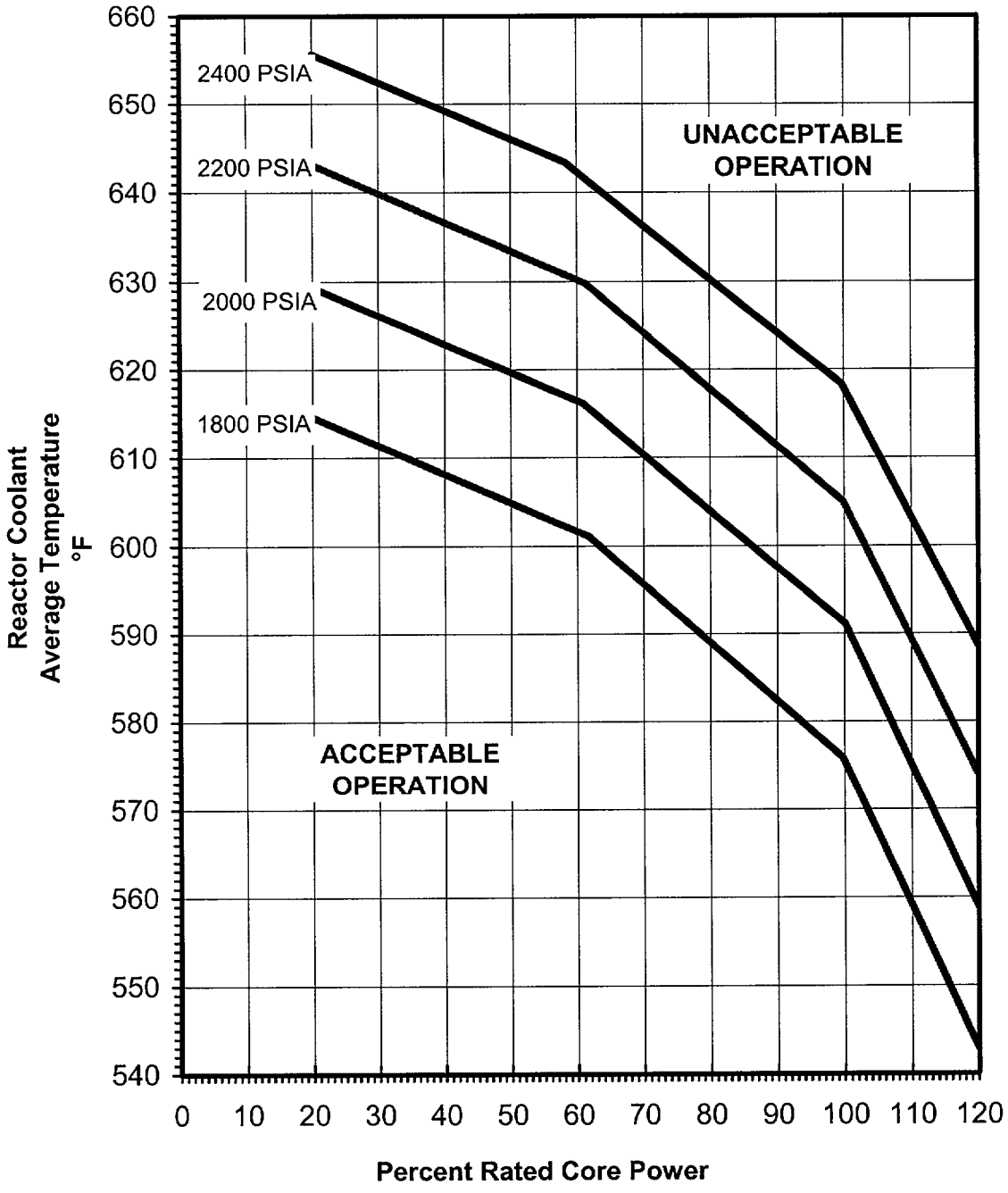
[4] WCAP-8092
[5] USAR Section 14.1.8
[6] USAR Section 14.1.10

FIGURE TS 2.1-1

**Safety Limits Reactor Core - Minimum Coolant System Flow**

**(TS 3.10.m) - Minimum DNBR**



## 5.0 DESIGN FEATURES

## 5.1 SITE

### APPLICABILITY

Applies to the location and extent of the reactor site.

### OBJECTIVE

To define those aspects of the site which affect the overall safety of the installation.

### SPECIFICATION

The Kewaunee Nuclear Power Plant is located on property owned by Wisconsin Public Service Corporation and Wisconsin Power and Light Company at a site on the west shore of Lake Michigan, approximately 30 miles east-southeast of the city of Green Bay, Wisconsin.

The minimum distance from the center line of the reactor containment to the site exclusion radius as defined in 10 CFR 100.3 is 1200 meters.

## 5.2 CONTAINMENT

### APPLICABILITY

Applies to those design features of the Containment System relating to operational and public safety.

### OBJECTIVE

To define the significant design features of the Containment System.

### SPECIFICATION

a. Containment System

1. The Containment System completely encloses the entire reactor and the Reactor Coolant System and ensures that leakage of activity is limited, filtered and delayed such that off-site doses resulting from the design basis accident are within the guidelines of 10 CFR Part 100. The Containment System provides biological shielding for both normal OPERATING conditions and accident situations.

2. The Containment System consists of:

   A. A free-standing steel reactor containment vessel designed for the peak pressure of the design basis accident.

   B. A concrete shield building which surrounds the containment vessel, providing a shield building annulus between the two structures.

   C. A Shield Building Ventilation System that causes leakage from the reactor containment vessel to be delayed and filtered before its release to the environment.

   D. An Auxiliary Building Special Ventilation System that serves the special ventilation zone and supplements the Shield Building Ventilation System during an accident condition by causing any leakage from the Residual Heat Removal System (RHRS) and certain small amounts of leakage that might be postulated to bypass the Shield Building Ventilation System to be filtered before their release.

b. Reactor Containment Vessel

1. The reactor containment vessel is designed for the peak internal pressure of the design basis accident plus the loads resulting from an earthquake producing 0.06g horizontally and 0.04g vertically. It is also designed to withstand an external pressure 0.8 psi greater than the internal pressure.

2. Penetrations of the containment vessel for piping, electrical conductors, ducts and access hatches are provided with double barriers against leakage.

3. The automatically actuated containment valves are designed to close upon high containment pressure and on a safety injection signal. The actuation system is designed so that no single component failure will prevent containment isolation, if required.

c. Shield Building

The shield building is a reinforced concrete structure with a wall thickness of 2.5 feet and a dome thickness of 2 feet. It is designed for the same seismic conditions as the reactor containment vessel and is designed to resist a 3 psi internal pressure due to tornadoes.

d. Shield Building Ventilation System

In the event of a loss-of-coolant accident, the Shield Building Ventilation System will relieve the initial thermal expansion of air through particulate and charcoal filters and will then cause a vacuum to be produced throughout the shield building annulus. A momentary positive pressure no greater than 0.5 psi will result during the thermal expansion. Once vacuum is achieved, the system causes the air within the annulus to be recirculated through the filters while vacuum is maintained. The filtered mixture of annulus air plus leakage is vented through the Containment System vent by the discharge fan that maintains vacuum at a vent rate determined by in-leakage to the shield building.[1]

e. Auxiliary Building Special Ventilation Zone and Special Ventilation System

A limited amount of containment leakage could potentially escape through certain penetrations in the event of leakage in the isolation valves, as described in the Basis of TS 3.6. The leakage escaping into that portion of the auxiliary building which is designed for medium leakage and controlled access would be processed by the Auxiliary Building Special Ventilation System. When actuated, the system will draw all in-leakage air from this special ventilation zone and exhaust it through particulate and charcoal filters to the auxiliary building vent.[2]

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[1] USAR Section 5.5
[2] USAR Section 9.6

## 5.3 REACTOR CORE

### APPLICABILITY

Applies to the reactor core.

### OBJECTIVE

To define those design features which are essential in providing for safe reactor core operations.

### SPECIFICATION

a. Fuel Assemblies

The reactor shall contain 121 fuel assemblies. Each assembly shall consist of a matrix of zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead-test-assemblies that have not completed representative testing may be placed in non-limiting core regions. Lead-test-assemblies shall be of designs approved by the NRC for use in pressurized water reactors and their clad materials shall be the materials approved as part of those designs.

b. Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

## 5.4 FUEL STORAGE

### APPLICABILITY

Applies to the capacity and storage arrays of new and spent fuel.

### OBJECTIVE

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

### SPECIFICATION

a. Criticality

1. The spent fuel storage racks are designed and shall be maintained with the following:

   a. Fuel assemblies having a maximum enrichment of 56.067 grams Uranium-235 per axial centimeter

   b. $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties

2. The new fuel storage racks are designed and shall be maintained with:

   a. Fuel assemblies having a maximum enrichment of 56.067 grams Uranium-235 per axial centimeter

   b. $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties

   c. $k_{eff} < 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties

3. The spent fuel pool is filled with borated water at a concentration to match that used in the reactor REFUELING cavity and REFUELING canal during REFUELING OPERATIONS or whenever there is fuel in the pool.

b. Capacity

The spent fuel storage pool is designed with a storage capacity of 1205 assemblies and shall be limited to no more than 1205 fuel assemblies.

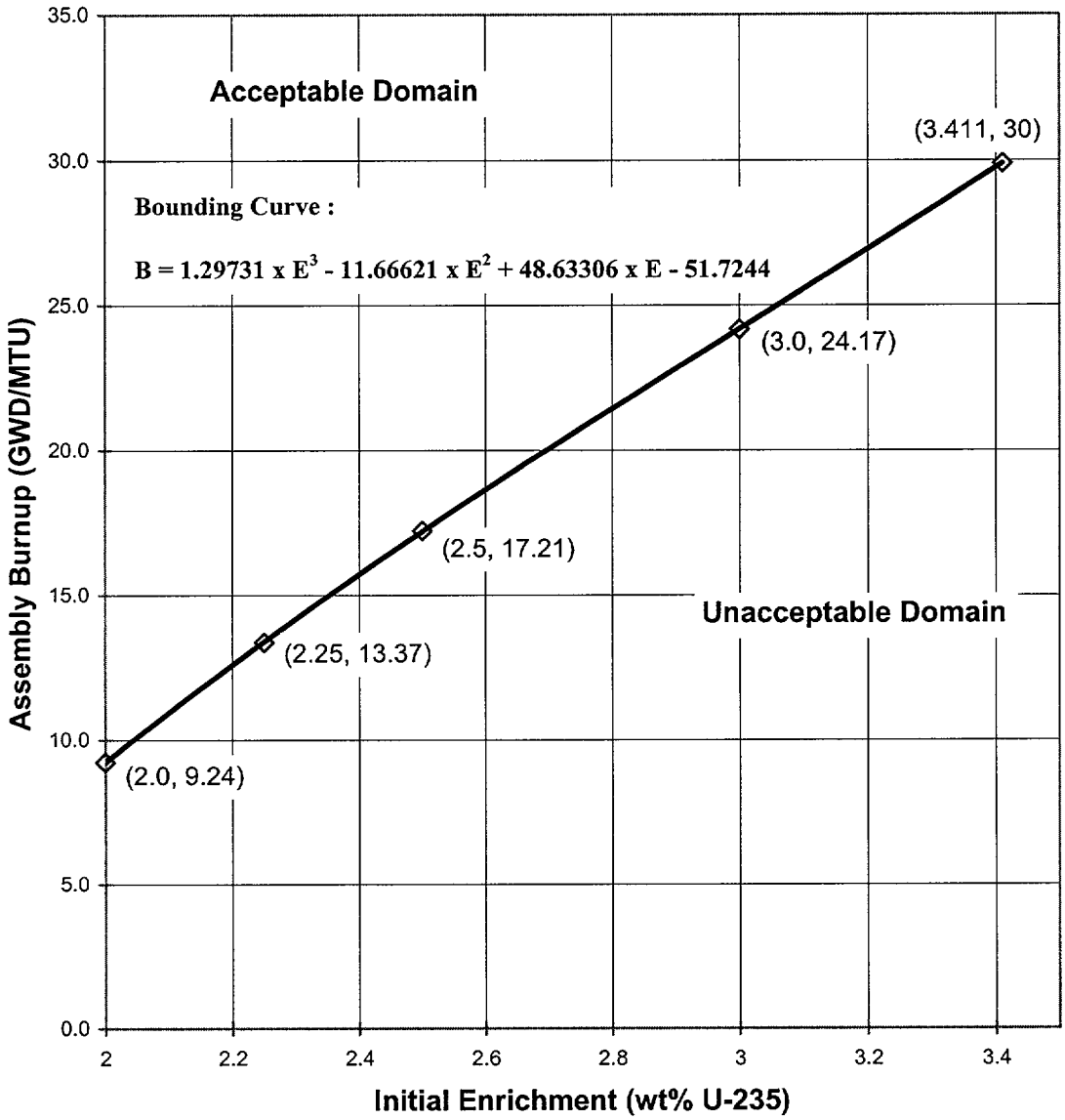
c. Canal Rack Storage

Fuel assemblies stored in the canal racks shall meet the minimum required fuel assembly burnup as a function of nominal initial enrichment as shown in Figure TS 5.4-1. These assemblies shall also have been discharged prior to or during the 1984 REFUELING outage.

FIGURE TS 5.4-1

MINIMUM REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF

NOMINAL INITIAL ENRICHMENT TO PERMIT STORAGE IN THE TRANSFER CANAL



## 6.0 ADMINISTRATIVE CONTROLS

### 6.1 RESPONSIBILITY

a. The Manager - Kewaunee Plant shall be responsible for overall plant operation and shall delegate in writing the succession of this responsibility during his absence.

b. The Manager - Kewaunee Plant, or his designee, shall approve prior to implementation, each proposed test, experiment or modification to structures, systems or components that affect nuclear safety.

## 6.2 ORGANIZATION

a. Off-Site Staff

The off-site organization for plant management and technical support shall be as described in the Operational Quality Assurance Program Description.

b. Facility Staff

The plant organization shall be as described in the Operational Quality Assurance Program Description.

1. Each on-duty shift complement shall consist of at least:

   A. One Shift Manager (SRO)
   B. Two licensed Reactor Operators
   C. Two Nuclear Auxiliary Operators
   D. Deleted
   E. One Radiation Technologist

2. While above COLD SHUTDOWN, the on-duty shift complement shall consist of the personnel required by TS 6.2.b.1 and an additional SRO.

3. In the event that one of the shift members becomes incapacitated due to illness or injury or the Radiation Technologist has to accompany an injured person to the hospital, reactor operations may continue with the reduced complement until a replacement arrives. In all but severe weather conditions, a replacement is required within two hours.

4. At least one licensed operator shall be in the control room when fuel is in the reactor.

5. Two licensed operators, one of which shall be an SRO, shall be present in the control room when the unit is in an operational MODE other than COLD SHUTDOWN or REFUELING.

6. REFUELING OPERATIONS shall be directed by a licensed SRO assigned to the REFUELING OPERATION who has no other concurrent responsibilities during the REFUELING OPERATION.

7. When the reactor is above the COLD SHUTDOWN condition, a qualified Shift Technical Advisor shall be within 10 minutes of the control room.

c. Organizational Changes

Changes not affecting safety may be made to the off-site and facility staff organizations. Such changes that are described in the Technical Specifications shall be reported to the Commission in the form of an application for license amendment within 60 days of the implementation of the change.

## 6.3 PLANT STAFF QUALIFICATIONS

a. Qualification of each member of the Plant Staff shall meet or exceed the minimum acceptable levels of ANSI N18.1-1971 for comparable positions, except for the Superintendent-Plant Radiation Protection who shall meet or exceed the recommendation of Regulatory Guide 1.8, Revision 1-R, September 1975, or their equivalent as further clarified in Attachment 1 to the Safety Evaluation Report enclosed with Amendment No. 46 to Facility Operating License DPR-43.

b. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in the design of the Kewaunee Plant and plant transient and accident analysis.

## 6.4 TRAINING

A retraining and replacement training program for the Plant Staff shall be maintained and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI-N18.1-1971 and 10 CFR Part 55.

## 6.5 DELETED

## 6.6 DELETED

## 6.7 SAFETY LIMIT VIOLATION

The following actions shall be taken in the event a SAFETY LIMIT is violated:

a. The reactor shall be shut down and operation shall not be resumed until authorized by the Commission.

b. The Report shall be prepared in accordance with 10 CFR 50.72 and 10 CFR 50.73.

## 6.8 PROCEDURES

a. Written procedures and administrative policies shall be established, implemented and maintained that meet the requirements and recommendations of Section 5.2.2, 5.2.5, 5.2.15 and 5.3 of ANSI N18.7-1976.

b. Changes to procedures are made in accordance with the provisions of ANSI N18.7-1976 Section 5.2.2, except temporary changes which clearly do not change the intent of the procedure shall, as a minimum, be approved by two individuals knowledgeable in the area affected one of which holds an active SRO license at Kewaunee.

c. Procedures are reviewed in accordance with the provisions of ANSI N18.7-1976, Section 5.2.15. The biennial review requirement is accomplished through alternate programs as described in the OQAPD.

## 6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

a. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted 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 the USAR 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 specific 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 operation, or (3) nine 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.

2. Annual Reporting Requirements

Routine OPERATING reports covering the operation of the unit during the previous calendar year shall be submitted prior to March 1 of each year. Items reported in this category include:

A. Deleted

B. As per applicable, portions of Regulatory Guide 1.16, a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated person rem exposure according to work and job functions,[1] e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and REFUELING. The dose assignment to various duty functions may be estimates based on pocket dosimeter (TLD). Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

C. Challenges to and failures of the pressurizer power operated relief valves and safety valves.[2]

D. This report shall document the results of specific activity analysis in which the reactor coolant exceeded the limits of TS 3.1.c.1.A during the past year. The following information shall be included:

(1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded.

(2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations.

(3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded.

(4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level.

(5) The time duration when the specific activity of the reactor coolant exceeded the radioiodine limit.

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[1] This tabulation supplements the requirements of Section 20.2206(b) of 10 CFR Part 20.
[2] Letter from E. R. Mathews (WPSC) to D. G. Eisenhut (U.S. NRC) dated January 5, 1981.

3. Monthly OPERATING Report

Routine reports of OPERATING statistics and shutdown experience shall be submitted on a monthly basis to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the OFF-SITE DOSE CALCULATION MANUAL (ODCM) and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

2. Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit for the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the PCP, and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

3. Special Reports

A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

(1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

## 6.10 RECORD RETENTION

a. The following records shall be retained for at least five years:

1. Records and logs of plant operation, including power levels and periods of operation at each power level.

2. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment pertaining to nuclear safety.

3. Reports of all REPORTABLE EVENTS.

4. Records of periodic checks, inspections, and calibrations required by these Technical Specifications.

5. Records of nuclear safety-related tests or experiments.

6. Records of radioactive shipments.

7. Records of changes to OPERATING procedures.

8. Records of sealed source leak tests and results.

9. Records of annual physical inventory of all source material of record.

10. Records of Quality Assurance activities required by the Operational Quality Assurance Program (OQAP) except where it is determined that the records should be maintained for a longer period of time.

b. The following records shall be retained for the duration of the Plant Operating License.

1. Records of a complete set of as-built drawings for the plant as originally licensed and all print changes showing modifications made to the plant.

2. Records of new and spent fuel inventory, fuel transfers, and assembly burnup histories.

3. Records of plant radiation and contamination surveys.

4. Records of radiation exposure of all plant personnel, and others who enter radiation control areas.

5. Records of radioactivity in liquid and gaseous wastes released to the environment.

6. Records of transient or operational cycles for these facility components.

7. Records of training and qualification for current members of the plant staff.

8. Records of in-service inspections performed pursuant to these Technical Specifications.

9. Records of meetings of the JOSRC and PORC.

10. Records for environmental qualification.

11. Records of reviews performed for changes made to the ODCM and the PCP.

## 6.11 RADIATION PROTECTION PROGRAM

a. Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

b. Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne in-plant iodine concentrations under accident conditions. This program shall include the following:

1. Training of personnel
2. Procedures for monitoring
3. Provisions for maintenance of sampling and analysis equipment

## 6.12 SYSTEM INTEGRITY

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

a. Provisions establishing preventive maintenance and periodic visual inspection requirements.

b. Integrated leak test requirements for each system at a frequency not to exceed REFUELING cycle intervals.

## 6.13 HIGH RADIATION AREA

a. In lieu of the "control device" or "alarm signal" required by Paragraph 20.1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is > 100 mrem/hr, but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a radiation work permit (RWP).[1] Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following.

   1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   2. A radiation monitoring device which continuously integrates the radiation dose in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

   3. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the RWP.

b. In addition to the requirements of 6.13.a., areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose > 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose > 1000 mrem[2] that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

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[1] Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

[2] Measurement made at 30 centimeters from source of radioactivity.

## 6.14 DELETED

## 6.15 SECONDARY WATER CHEMISTRY

The licensee shall implement a secondary water chemistry monitoring program. The intent of this program will be to control corrosion thereby inhibiting steam generator tube degradation. The secondary water chemistry program shall act as a guide for the chemistry group in their routine as well as non-routine activities.

## 6.16 RADIOLOGICAL EFFLUENTS

a. Written procedures shall be established, implemented and maintained covering the activities referenced below:

   1. Process Control Program (PCP) implementation

   2. OFF-SITE DOSE CALCULATION MANUAL (ODCM) implementation

   3. Quality Assurance Program for effluent and environmental monitoring

b. The following programs shall be established, implemented, and maintained:

   1. Radioactive Effluent Controls Program

      A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program shall: (1) be contained in the ODCM, (2) be implemented by OPERATING procedures, and (3) include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

      A. Limitations on the OPERABILITY of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM.

      B. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2.

      C. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM.

      D. Limitations on the annual and quarterly doses or dose commitment to a MEMBER(S) OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50.

      E. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days.

F. Limitations on the OPERABILITY and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31 day period would exceed 2% of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50.

G. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1.

H. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50.

I. Limitations on the annual and quarterly doses to MEMBER(S) OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than eight days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50.

J. Limitations on the annual dose or dose commitment to any MEMBER(S) OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

2. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide: (1) representative measurement of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall: (1) be contained in the ODCM (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

A. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

B. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census.

C. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the Quality Assurance Program for environmental monitoring.

## 6.17 PROCESS CONTROL PROGRAM (PCP)

a. The PCP shall be approved by the Commission prior to implementation.

b. Licensee initiated changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained as required by TS 6.10.b.11. The documentation shall contain:

   A. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s).

   B. A determination that the change will maintain the overall conformance of the soldified waste product to existing requirements of federal, state, or other applicable regulations.

2. Shall become effective upon review and acceptance by the PORC.

## 6.18 OFF-SITE DOSE CALCULATION MANUAL (ODCM)

a. The ODCM shall be approved by the Commission prior to implementation.

b. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained as required by TS 6.10.b.11. This documentation shall contain:

   A. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change.

   B. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after review and acceptance by the PORC.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. The date the changes were made shall be indicated. In addition, a method such as redlining should be used to clearly identify the changes.

## 6.19 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS[1]

Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

a. Shall be reported to the Commission in the Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PORC. The discussion of each change shall contain:

   1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

   2. Sufficient information to support the reason for the change without benefit of additional or supplemental information.

   3. A description of the equipment, components and processes involved and the interfaces with other plant systems.

   4. An evaluation of the change that shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto.

   5. An evaluation of the change that shows the expected maximum exposures to individuals in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto.

   6. A comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made.

   7. An estimate of the exposure to plant OPERATING personnel as a result of the change.

   8. Documentation of the fact that the change was reviewed and found acceptable by the PORC.

b. Shall become effective upon review and acceptance by the PORC.

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[1] Licensees may choose to submit the information called for in this TS as part of the periodic USAR update.

## 6.20 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. The provisions of TS 4.0.b do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. The provisions of TS 4.0.c are applicable to the Containment Leakage Rate Testing Program.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident is less than the containment internal test pressure, $P_a$. The maximum allowable leakage rate ($L_a$) is 0.5 weight percent of the contained air per 24 hours at the peak test pressure ($P_a$) of 46 psig.

For penetrations which extend into the auxiliary building special ventilation zone, the combined leak rate from these penetrations shall not exceed $0.10L_a$. For penetrations which are exterior to both the shield building and the auxiliary building special ventilation zone, the combined leak rate from these penetrations shall not exceed $0.01L_a$. If leak rates are exceeded, repairs and retest shall be performed to demonstrate reduction of the combined leak rate to these values.

Leakage rate acceptance criteria:

a. The containment leakage rate acceptance criterion is $\leq 1.0L_a$.

b. Prior to unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.6L_a$ for Type B and C tests and $< 0.75L_a$ for the Type A test.

c. The personnel and emergency air lock leakage rates, when combined with the cumulative Type B and C leakage, shall be $< 0.6L_a$. For each air lock door seal, the leakage rate shall be $< 0.005L_a$ when tested to $\geq 10$ psig.