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November 14, 1986

Posted
Amdt. 80
to DPR-42

Docket Nos. 50-282
and 50-306

Mr. D. M. Musolf, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Midland Square, 4th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has issued the enclosed Amendment Nos. 80 and 73 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, in response to your application dated July 15, 1986. The amendments change the technical specifications (TS) by revising the following:

1. The effective limitation of existing reactor coolant system heatup and cooldown curves are revised from ten to 15 effective full power years.

2. The Radiation Environmental Monitoring Program is revised in areas dealing with radionuclides, corn sampling, lower limit of detection of iodine 131 and airborne sampling locations of radioiodine and particulates. These revisions consist of corrections, clarifications and the elimination of potential inconsistencies.

3. Changes are made to Design Features, Section 5 of the technical specifications reflecting changes associated with 10 CFR Part 50, Appendix I involving method for processing radioactive wastes and other administrative changes.

4. Changes are made clarifying the management review of the Security Plans implementing procedures.

5. The Administrative, Section 6 is changed by deleting the requirements of environmental qualification of electrical equipment and other editorial changes.

The issuance of these amendments completes our work effort under TAC Nos. 62137 and 62138.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/S/

Dominic C. DiIanni, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Enclosures:
1. Amendment No. 80 to DPR-42
2. Amendment No. 73 to DPR-60
3. Safety Evaluation

cc's w/enclosures:
See next page

no legal objection

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Date:	10/31/86	10/31/86	11/03/86	10/14/86

Mr. D. M. Musolf
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 80
License No. DPR-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated July 15, 1986 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;

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

   C. There is reasonable assurance (1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 80 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. DiIanni

Dominic C. DiIanni, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 14, 1986

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.73
License No. DPR-60

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated July 15, 1986 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;

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

   C. There is reasonable assurance (1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 73 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. DiIanni

Dominic C. DiIanni, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 14, 1986

ATTACHMENT TO LICENSE AMENDMENT NOS. 80 AND 73

TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of changes.

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TS-viii	TS-viii
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TS.3.1-8	------
TS.3.1-11	TS.3.1-11
TS.3.1-12	TS.3.1-12
TS.3.1-13	TS.3.1-13
Table TS.3.1-1	------
Table TS.3.1-2	------
Figure TS.3.1-1	Figure TS.3.1-1
Figure TS.3.1-2	Figure TS.3.1-2
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Table TS.4.10-1 (Page 1 of 4)	Table TS.4.10-1 (Page 1 of 4)
Table TS.4.10-1 (Page 4 of 4)	Table TS.4.10-1 (Page 4 of 4)
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5. Reactor Coolant Vent System

a. A reactor shall not be made or maintained critical nor shall it be heated or maintained above 200°F unless reactor coolant vent system paths from both the reactor vessel head and pressurizer steam space are operable and closed except as specified in 3.1.A.5.b and 3.1.A.5.c below.

b. During Startup Operation or Power Operation, any one of the following conditions of inoperability may exist for each unit until operability is restored:

   1. Both of the parallel vent valves in the reactor vessel head vent path are inoperable.

   2. Both of the parallel vent valves in the pressurizer vent path are inoperable.

   3. The vent valve to the pressurizer relief tank discharge line is inoperable.

   4. The vent valve to the containment atmospheric discharge line is inoperable.

   If during Startup Operation or Power Operation any one of these conditions is not restored to an operable status within 30 days, the reactor shall be placed in Hot Shutdown within 6 hours and in Cold Shutdown within the following 30 hours.

c. With no reactor coolant vent system path operable, restore at least one vent path to operable status within 72 hours or be in the Hot Shutdown condition within 6 hours and the Cold Shutdown condition within the following 30 hours.

Basis

When the boron concentration of the reactor coolant system is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

"Steam Generator Tube Surveillance", Technical Specification 4.12, identifies steam generator tube imperfections having a depth $\geq$ 50% of the 0.050-inch tube wall thickness as being unacceptable for power operation. The results of steam generator burst and tube collapse tests submitted to the staff have demonstrated that tubes having a wall thickness greater than 0.025-inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents.[2]

Part A of the specification requires that both reactor coolant pumps be operating when the reactor is critical to provide core cooling in the event that a loss of flow occurs. In the event of the worst credible coolant flow loss (loss of both pumps from 100% power) the minimum calculated DNBR remains well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Critical operation, except for low power physics tests, with less than two pumps is not planned. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost. Below 10% power, a shutdown under administrative control will be made if flow from either pump is lost.

The pressurizer is needed to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients. Each of the pressurizer safety valves is designed to relieve 325,000 lbs per hour of saturated steam at the valve set point. Below 350°F and 450 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization of the reactor coolant system for reactor coolant temperatures less than 350°F. The combined capacity of both safety valves is greater than the maximum surge rate resulting from complete loss of load.[1]

B. HEATUP AND COOLDOWN

Specification:

1. The Unit 1 and Unit 2 reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS.3.1-1 and TS.3.1-2.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b. Figures TS.3.1-1 and TS.3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figures TS.3.1-1, TS.3.1-2 shall be recalculated periodically using methods discussed in the Bases section.

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

4. The pressurizer heatup rate shall not exceed 100°F/hr and the pressurizer cooldown rate shall not exceed 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Bases

Pressure/Temperature Limits

Appendix G of 10 CFR Part 50, and the ASME Code require that the reactor coolant pressure boundary be designed with sufficient margin to insure that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner, the probability of rapidly propagating fracture is minimized and the design reflects the uncertainties in determining the effects of irradiation on material properties. Figures TS.3.1.-1 and 2 have been developed (Reference 1) in accordance with these regulations. The curves are based on the properties of the most limiting material in either unit's reactor vessel (Unit 1 reactor vessel weld W-3) and are effective to 15 EFPY. The curves have been adjusted for possible errors in the pressure and temperature sensing instruments.

The curves define a region where brittle fracture will not occur and are determined from the material characteristics, irradiation effects, pressure stresses and stresses due to thermal gradients across the vessel wall.

Heatup Curves

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. At the inner wall of the vessel, the thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. For the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis. The heatup limit curve is a composite curve prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour.

Bases (continued)

Cooldown Curves

During cooldown, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from tensile at the inner wall to compressive at the outer wall. The thermal induced tensile stresses at the inner wall are additive to the pressure induced tensile stresses which are already present. Therefore, the controlling location is always the inside wall.

The cooldown limit curves were prepared utilizing the same type of analysis used to calculate the heatup curve except that the controlling location is always the inside wall.

Criticality Limits

Appendix G of 10 CFR Part 50 requires that for a given pressure, the reactor must not be made critical unless the temperature of the reactor vessel is 40°F above the minimum permissible temperature specified on the heatup curve and above the minimum permissible temperature for the inservice hydrostatic pressure test. For Prairie Island the curves were prepared, requiring that criticality must occur above the maximum permissible temperature for the inservice hydrostatic pressure test.

ASME Code Section XI Inservice Test Limits

The pressure temperature limits for the ASME Code Section XI Inservice Test Limits (hydrostatic pressure test) are less restrictive than the heatup and cooldown curves to allow for the periodic inservice hydrostatic test. These limits are allowed to be less restrictive because the hydrostatic test is based on a 1.5 safety factor versus the 2.0 safety factor built into the heatup and cooldown curves and because the test is run at a constant temperature so the thermal stresses in the vessel are minimal.

Reference

1. USAR, Section 4.2

D. MAXIMUM COOLANT ACTIVITY

Specification:

1. The specific activity of the primary coolant shall be limited to:

   (a) Less than or equal to 1.0 microcuries per gram DOSE EQUIVALENT I-131, and

   (b) Less than or equal to $100/\bar{E}$ microcuries per gram.

2. In Specification 3.1.D.1 the following definitions apply:

   (a) DOSE EQUIVALENT I-131 is that concentration of I-131 (uCi/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 those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

   (b) $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

3. If a reactor is above hot shutdown and RCS temperature is greater than or equal to 500°F:

   (a) With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure TS.3.1-3, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6-month period, a special report to the Commission shall be submitted within 30 days indicating the number of hours above this limit.

   (b) With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure TS.3.1-3, the affected reactor shall be shutdown and RCS temperature cooled to 500°F or less within 6 hours.

   (c) With the specific activity of the primary coolant greater than $100/\bar{E}$ microcurie per gram, the affected reactor shall be shutdown and RCS temperature cooled to 500°F or less within 6 hours of detection.

4. If a reactor is at or above cold shutdown:

   (a) With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.1-2B until the specific activity of the primary coolant is restored to within its limits. A special report shall be submitted to the Commission within 30 days. This report shall contain the results of the specific activity analyses together with the following information:

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

      2. Fuel burnup by core region,

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

      4. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

      5. The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie per gram DOSE EQUIVALENT I-131.

Basis

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Prairie Island site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

Specification 3.1.D.2, permitting power operation to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure TS.3.1-3, accommodates possible iodine spiking phenomenon which may occur following changes in thermal power. Operation with specific activity levels exeeding 1.0 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure TS.3.1-3 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure TS.3.1-3 increase the 2 hours thyroid

dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

Reducing RCS temperature to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements in Table TS.4.1-2B provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

FIGURE TS.3.1-1

UNIT 1 and UNIT 2
REACTOR COOLANT SYSTEM HEAT UP LIMITATIONS
(Applicable for First 15 EFPY of Operation)

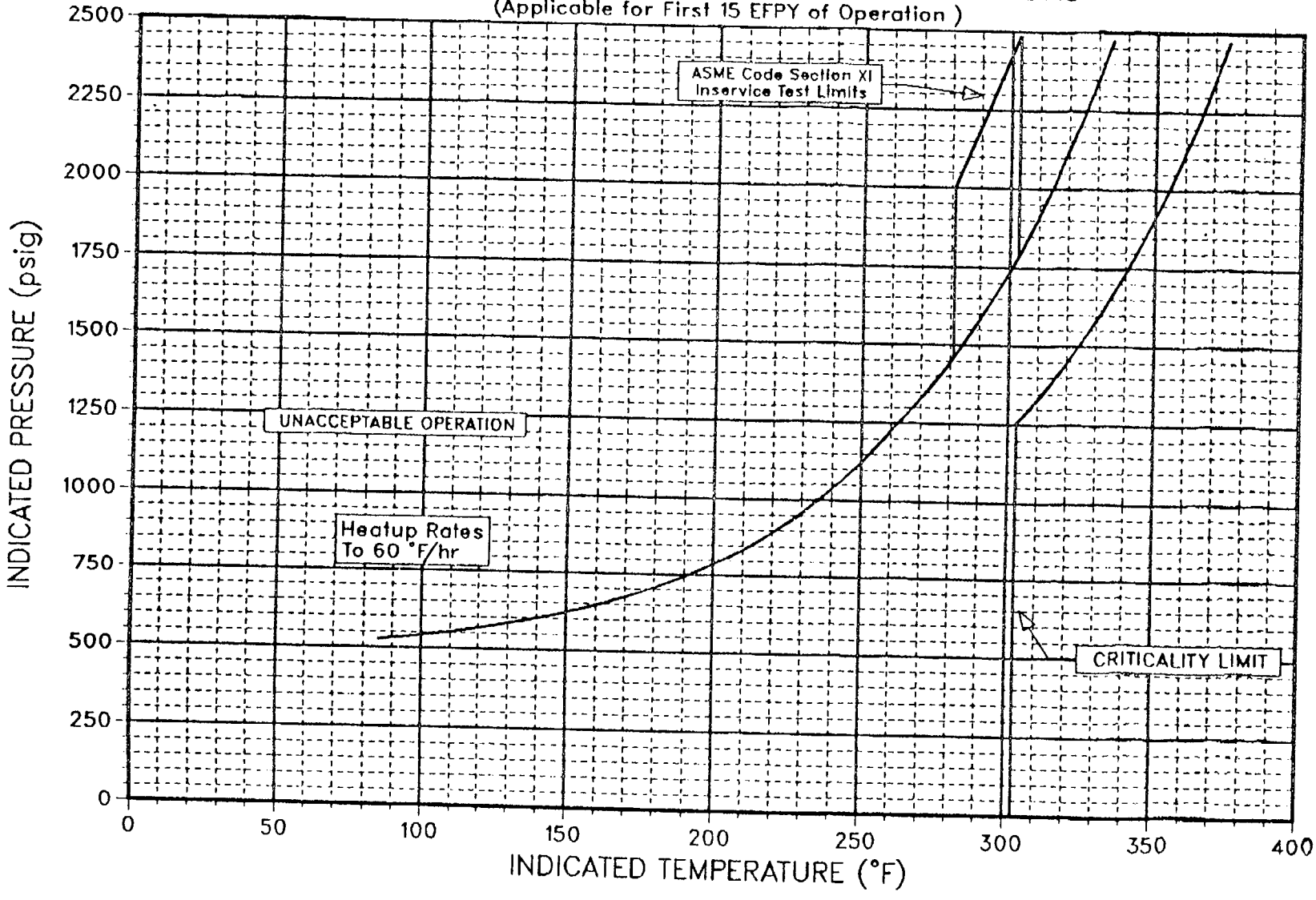
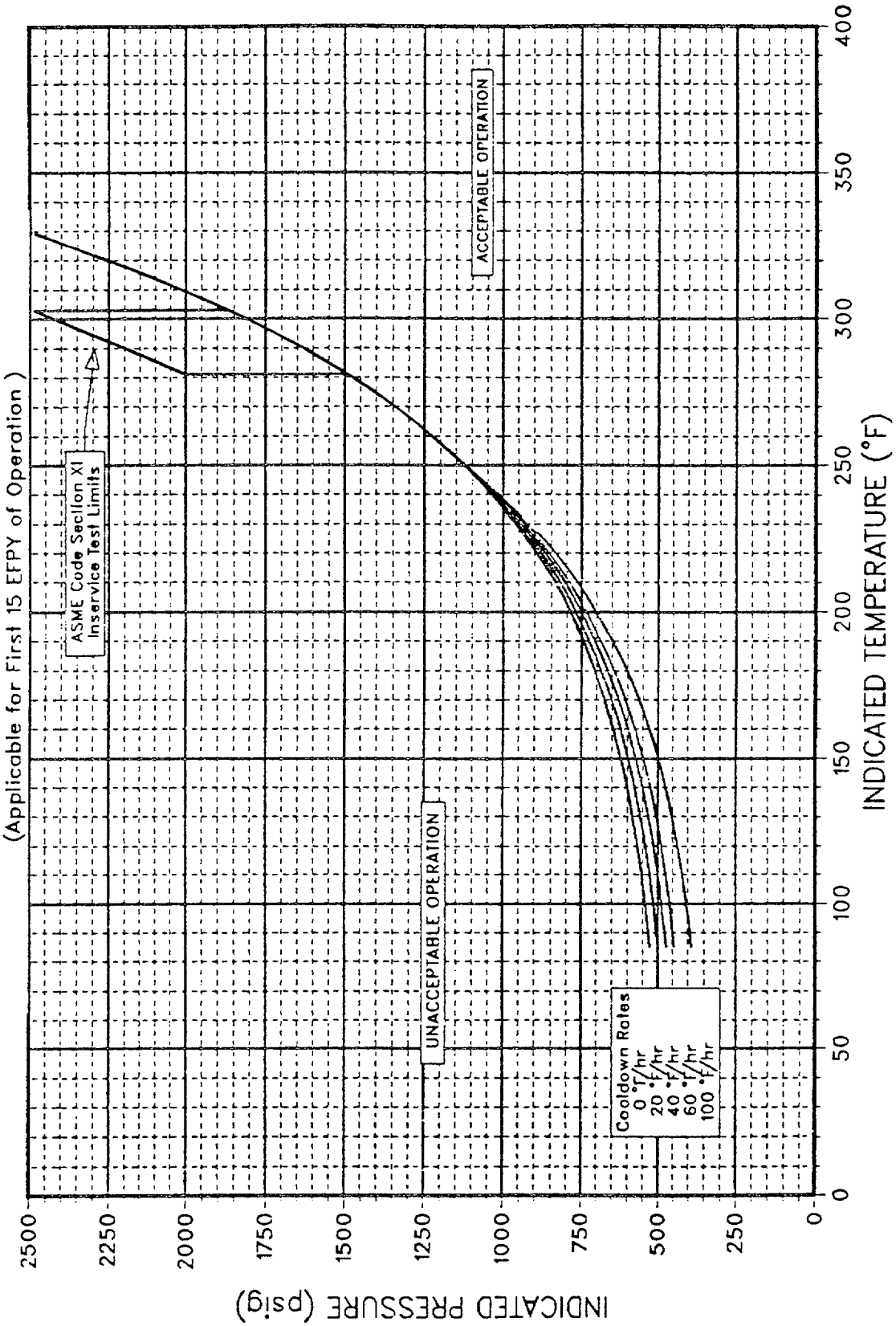


FIGURE TS.3.1-2

UNIT 1 and UNIT 2
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
(Applicable for First 15 EFPY of Operation)



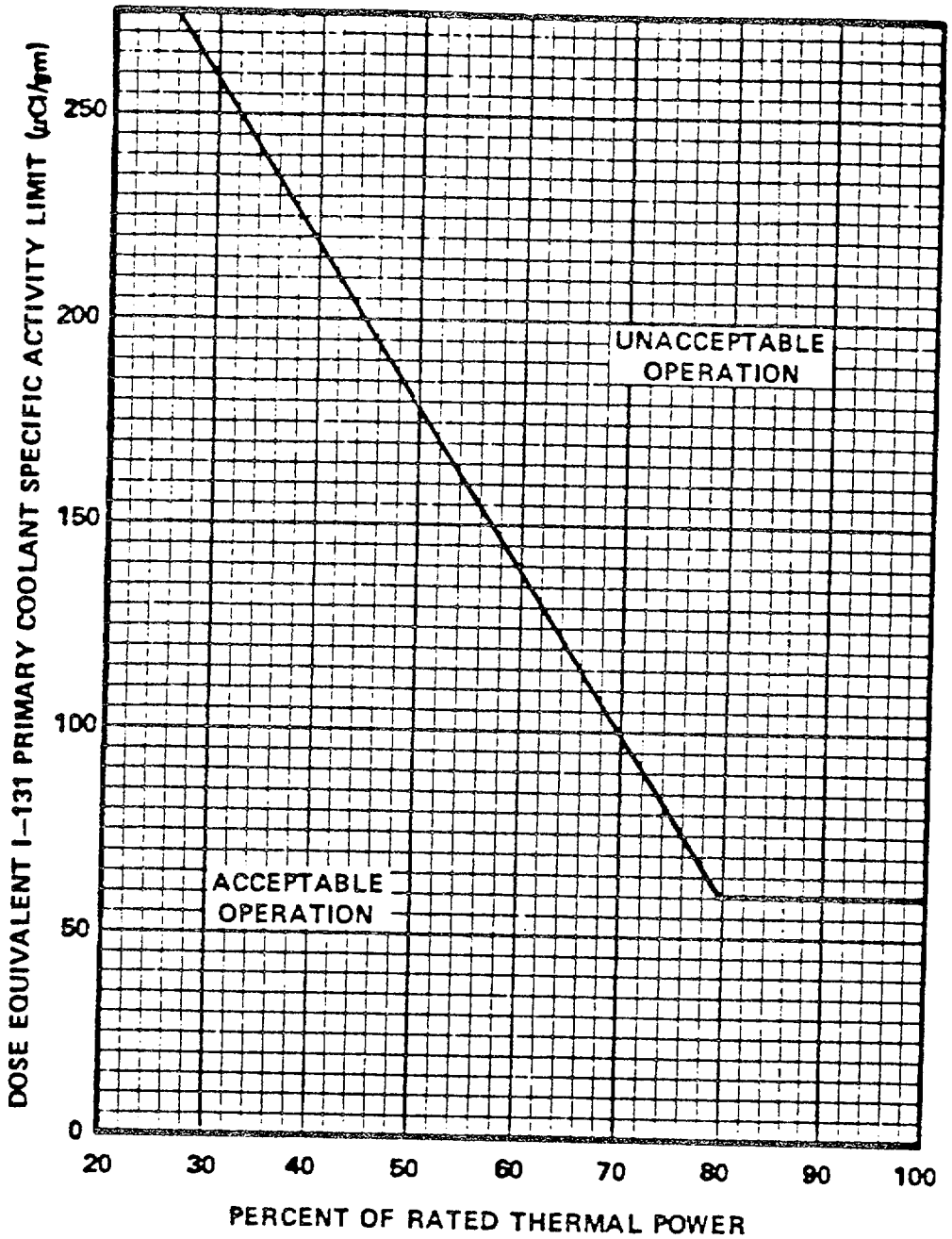


FIGURE TS.3.1-3

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131

4.10 RADIATION ENVIRONMENTAL MONITORING PROGRAM

Applicability

Applies at all times to the periodic monitoring and recording of radioactive effluents found in the plant environs.

Objective

To provide for measurement of radiation levels and radioactivity in the site environs on a continuing basis.

Specification

A. Sample Collection and Analysis

1. The Radiation Environmental Monitoring Program described in Table 4.10-1 shall be conducted. Radioanalysis shall be conducted meeting the requirements of Table TS.4.10-2. A map and a table identifying the locations of the sampling shall be provided in the Offsite Dose Calculation Manual (ODCM).

2. Whenever the Radiation Environmental Monitoring Program is not being conducted as specified in Table TS.4.10-1 the Annual Radiation Environmental Monitoring Report shall include a description of the reasons for not conducting the program as required and plans for preventing a recurrence.

3. Deviations are permitted from the required sampling schedule if samples are unobtainable due to hazardous conditions, seasonable unavailability, or to malfunction of automatic sampling equipment. If the latter occurs, every effort shall be made to complete corrective action prior to the end of the next sampling period.

4. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 4.10-3 when averaged over any calendar quarter, in lieu of any other report, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Report pursuant to Specification 6.7.C.2(a). When more than one of the radionuclides in Table 4.10-3 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots > 1.0$$

When radionuclides other than those in Table 4.10-3 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.9.A.2, 3.9.B.2, or 3.9.B.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Monitoring Report.

TABLE TS. `-1
(Page 1)

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE Radioiodine and Particulates	Samples from 5 locations: 3 samples from offsite locations (in different sectors) of the highest calculated annual average ground level D/Q, 1 sample from the vicinity of a community having the highest calculated annual average ground-level D/Q, and 1 sample from a control location specified in the ODCM	Continuous Sampler operation with sample collection weekly	Radioiodine analysis weekly for I-131 Particulate: Gross beta activity on each filter weekly*. Analyses shall be performed more than 24 hours following filter change. Perform gamma isotopic analysis on composite (by location) sample quarterly.
2. DIRECT RADIATION	32 TLD stations established with duplicate dosimeters placed at the following locations: 1. Using the 16 meteorological wind sectors as guidelines, an inner ring of stations in the general area of the site boundary is established and an outer ring of stations in the 4 to 5 mile distance from the plant site is established. Because of inaccessibility, seven sectors in the inner and outer rings are not covered	Quarterly	Gamma dose quarterly

* If Gross beta activity in any indicator sample exceeds 10 times the yearly average of the control sample, a gamma isotopic analysis is required.

** Sample locations are given on the figure and table in the ODCM.

TABLE TS.4.10-1
(Page 4 of 4)

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
c. Food Products	One sample of corn from any field that is irrigated by water into which liquid plant wastes have been discharged***	At time of harvest	Gamma isotopic analysis of edible portion of each sample
	One sample of broad leaf vegetation from highest D/Q garden and one sample from 10-20 miles	At time of harvest	I-131 analysis of edible portion of each sample

** Sample locations are given on the figure and table in the ODCM.
*** As determined by methods outlined in the ODCM.

TABLE TS.4.10-2

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,e]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4[b]	1×10^{-2}				
^{3}H	2000(1000[b])					
^{54}Mn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			
^{65}Zn	30		260			
^{95}Zr-Nb	15[c]					
^{131}I[d]	1[b]	7×10^{-2}		1	60	-
$^{134,137}Cs$	15(10[b]), 18	1×10^{-2}	130	15	60	150
^{140}Ba-La	15[c]			15[c]		

TABLE TS.4.10-2

TABLE NOTATION

a - The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the apriori lower limit of detection as defined above (as picocurie per unit mass or volume), s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute). In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributing of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt shall be used in the calculations.

E is the counting efficiency (as counts per transformation),

2.22 is the number of transformation per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting.

- LLD for drinking water.
- Total for parent and daughter.
- These LLDs apply only where "^{131}I analysis" is specified.
- Where "Gamma Isotopic Analysis" is specified, the LLD specification applies to the following radionuclides: ^{54}Mn, ^{59}Fe, ^{58}Co, ^{60}Co, ^{65}Zn, ^{95}Zr-Nb, ^{137}Cs, ^{134}Cs, and ^{140}Ba-La. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

5.0 DESIGN FEATURES

5.1 SITE

The Prairie Island Nuclear Generating Plant is located on property owned by Northern States Power (NSP) Company at a site on the west bank of the Mississippi River, approximately 6 miles northwest of the city of Red Wing, Minnesota. The minimum distance from the center line of either reactor to the site exclusion boundary is 715 meters, and the low population zone distance is 1-1/2 miles. The nearest population center of 25,000 or more people is South Saint Paul. These site characteristics comply with definitions in 10CFR100 (Reference 1).

The U. S. Army Corp of Engineers controls the land within the exclusion area that is not owned by NSP. The Corps has made an agreement with NSP to prevent residential construction on this land for the life of the plant (Reference 2).

These specifications use atmospheric diffusion factors based on the NRC staff evaluations. Its evaluation of accidental airborne releases is based on a relative concentration of 9.8×10^{-4} seconds per cubic meter at the site boundary. Its evaluation of routine releases is based on a relative concentration of 1.5×10^{-5} seconds per cubic meter (Reference 3).

The flood of record in 1965 produced a water surface elevation of +688 feet MSL at the site. The calculated probable maximum flood (PMF) level is +703.6 feet mean sea level (MSL), and the estimated wave runup could reach +706.7 feet MSL. (See Section 2.4.2 of this report.) Plant grade level is +695 feet MSL.

Flood protection structures have been provided. The two turbine support facilities, the common auxiliary building, and the two shield buildings have been physically connected by a concrete flood wall, most of the length of which constitutes the concrete foundation walls for the various buildings. The top of this wall supports the metal siding for the buildings at about elevation +705 feet MSL. Fourteen doors through the flood wall, or into the various buildings (including the separate screen house), are provided with receivers for the erection of flood protection panels to prevent flood water from reaching safety related facilities.

The cooling water pumps in the screenhouse are designed to operate up to a flood level of +695 feet MSL without flood protection measures, and up to a level of +707 feet MSL with the erection of flood protection panels. The main transformer foundation is at +695 feet MSL. The transformer will function to a flood level of +698 feet MSL.

The Technical Specification 6.5 A.7. requires an emergency procedure that will necessitate plant shutdown for flood water levels above +692 feet MSL at the plant site. The emergency procedure will assure the proper

erection of flood protection panels and assure an orderly shutdown of the plant and protection of safety related facilities. This procedure will provide for progressive action levels to prevent the possibility of unsafe plant operation and will include requirements for periodic drills to test flood protection measures, such as erecting flood protection panels.

The plant is designed for a design basis earthquake having a horizontal ground acceleration of 0.12g and an operational basis earthquake having a horizontal ground acceleration of 0.06g. An emergency procedure will be prepared in accordance with Specification 6.5.A.7 to define actions required for earthquakes, including plant shutdown and inspection if an operational basis earthquake is measured at the site.

References

1. USAR, Section 2.2.1
2. USAR, Section 3.4.5
3. SER, Sections 2.3.4 and 2.3.5

annulus through certain penetrations in the event of leakage in their isolation valves. Such leakage would escape into a portion of the auxiliary building which is designed for minimum leakage and controlled access. The auxiliary building special ventilation system when actuated will draw all in-leakage air from this special ventilation zone and exhaust it through particulate and charcoal filters.

C. Containment System Functional Design

Functionally, leakage from the primary containment to the annulus between the primary containment and the shield building is processed by recirculating annulus air through either of two redundant fan-filter trains, designated the shield building ventilation system (SBVS), to remove radioactivity prior to exhaust. The vast majority of the leakage that may bypass the annulus through lines penetrating containment is expected to enter the ABSVZ, where it is processed by exhausting zone air through either of two redundant fan-filter trains to remove radioactivity. A minor portion of the total leakage may occur so that it bypasses both the annulus and the ABSVZ.

NSP has used conservative values in evaluating shield building annulus air pressure transients for the most severe loss-of-coolant accident (LOCA) and for a range of shield building leak rates (1% per day to 10% per day) assuming only one shield building fan-filter train operates (Reference 1). During the first 3 minutes following a LOCA, the steel primary containment temperature will increase and the resultant containment expansion and thermal transient will raise the air pressure in the shield building annulus to approximately 7 inches of water above atmospheric pressure. From 3 minutes to 6 minutes after the LOCA, the shield building ventilation system will reduce the annulus air pressure to a negative value relative to air pressure outside the annulus. When the annulus air pressure reaches 2 inches of water below atmospheric pressure, the recirculation damper in the SBVS will start to open. About 24 minutes after the LOCA the air pressure will reach a steady negative value, with the large recirculation fan recirculating air through the filters and the small exhaust fan maintaining a negative pressure in the annulus.

The fan filter trains in the ABSVZ will start automatically and the normal auxiliary building ventilation system will be isolated automatically within 1 minute following the LOCA. The fans in the ABSVZ will maintain a negative pressure in this zone.

The NRC has evaluated the LOCA assuming the total primary containment leakage is 0.25% by weight per day, 0.1 wt%/day of the primary containment leakage will bypass the annulus through lines penetrating containment and enter the ABSVZ, and that 0.01 wt%/day of the leakage will bypass both the annulus and the ABSVZ (References 3, 4, 5, 6). The remainder is assumed to be recirculated through the shield building ventilation system to remove fission products prior to exhaust from the building.

In the NRC dose calculations (References 3, 4, 5, 6) conservative mixing assumptions are used. During the first few minutes when the shield building annulus pressure is positive, direct leakage to the environs without mixing or filtration is assumed. During the time the ventilation system is exhausting without recirculation, primary containment leakage is assumed to enter the SBVS inlet duct without mixing and to be filtered before being released to the environs. During recirculation, leakage is assumed to enter the inlet duct without mixing and recirculated air from the outlet duct is assumed to mix in 50% of the annulus volume. The fraction of primary containment leakage that goes to the ABSVZ is assumed to enter the ventilation system ducts without mixing and to be filtered before discharge to the environs.

The leakage values used in the NRC staff's dose analyses (References 3, 4, 5, 6) form the bases for the limiting containment system leakage for containment system leakage tests in these Technical Specifications.

References

1. USAR, Appendix G
2. USAR, Sections 12.5, 14
3. SER, Section 15
4. SER, Supplement 1, issued later
5. Letter from AEC to NSP dated November 29, 1973
6. Letter from AEC to NSP dated September 16, 1974

5.3 REACTOR

A. Reactor Core

1. The reactor core contains approximately 48 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly contains 179 fuel rods (Reference 1).

2. The average enrichment of the reload core is a nominal 2.90 weight per cent of U-235. The highest Uranium-235 loading is a nominal 39 grams of U-235 per axial centimeter of fuel assembly (average).

3. In the reactor core, there are 29 full-length RCC assemblies that contain a 142-inch length of silver-indium-cadmium alloy clad with stainless steel (Reference 2).

B. Reactor Coolant System

1. The design of the reactor coolant system complies with all applicable code requirements (Reference 3).

2. All high pressure piping, components of the reactor coolant system and their supporting structures are designed to Class I requirements, and have been designed to withstand:

   a. The design seismic ground acceleration, 0.06g acting in the horizontal and 0.04g acting in the vertical planes simultaneously, with stresses maintained within code allowable working stresses.

   b. The maximum potential seismic ground acceleration, 0.12g, acting in the horizontal and 0.08g acting in the vertical planes simultaneously with no loss of function.

3. The nominal liquid volume of the reactor coolant system, at rated operating conditions, is 6100 cubic feet.

C. Protection Systems

The protection systems for the reactor and engineered safety features are designed to applicable codes, including IEEE-279, dated 1968. The design includes a reactor trip for a high negative rate of change of neutron flux as measured by the excore nuclear instruments (Reference 4). The system is intended to trip the reactor upon the abnormal dropping of more than one control rod (Reference 4). If only one control rod is dropped, the core can be operated at full power for a short time, as permitted by Specification 3.10.

References

1. USAR, Section 3.4.2
2. USAR, Section 3.5.2
3. USAR, Table 4.1-11
4. USAR, Section 7.1

5.4 ENGINEERED SAFETY FEATURES

The engineered safety features include the containment system described in Specification 5.2, the emergency core cooling system, the containment air cooling system, the containment spray system, the post-accident combustible gas control system, emergency power supplies, component cooling water system, and the cooling water system. These systems are designed to applicable industry codes, the NRC General Design Criteria in Appendix A to 10CFR50, and NRC Safety Guides. Particular features for the Prairie Island plant include the following:

1. Several of the features are shared between the two units, including the onsite diesel generators, the cooling water system, and the motor-driven pumps of the auxiliary feedwater system. Shared systems are designed to mitigate the effects of an accident in one unit and simultaneously provide for a hot shutdown in the other unit (Reference 1).

2. The emergency cooling water pumps are driven by diesel engines. These diesel engines are designed and will be tested to the same reliability criteria as those for the diesel generators that supply emergency electrical power (Reference 2).

3. The cooling water system is automatically divided into two (2) redundant loops by motor-operated valves which are actuated by a safety injection signal. Branches serving non-safety related equipment in the turbine rooms are isolated from the class I cooling water loops by motor-operated valves. Line breaks in these branches are sensed and the motor-operated valves are actuated by instruments which monitor coincident high flow and low pressure in the class I cooling water loops.

<u>Reference</u>

1. FSAR, Table 1.2-2
2. USAR, Section 10.4

5.5 RADIOACTIVE WASTE SYSTEMS

The design objective for the liquid radwaste system is to process the waste so that discharges will approach essentially zero under normal operating conditions and will meet the requirements of 10 CFR Parts 20 and 70, 10 CFR Part 50 Section 50.36a, Appendices A and I to 10 CFR Part 50, and 40 CFR Part 190 under design basis conditions. The design objective for the gaseous waste system is to release a small fraction of 10 CFR Parts 20 and 70, 10 CFR Part 50 Section 50.36a, Appendices A and I to 10 CFR Part 50, and 40 CFR Part 190 limits. Holdup time for the gaseous waste storage system is designed for the plant lifetime. The design objective of the solid radwaste system is to package solid waste in accordance with applicable government regulations for offsite shipment.

A. Accidental Releases

The auxiliary building is designed as Class I (seismic) in areas containing the auxiliary building special ventilation system and radwaste storage area. The radwaste building is also designed to Class I standards (Reference 1). All radwaste tanks, filters, and equipment are either contained in a Class I (seismic) building or in specially constructed areas to provide a substantial degree of control of the wastes should a liquid radwaste tank rupture.

B. Routine Releases

1. Liquid Wastes

The evaluation of the processing of liquid wastes is described in the USAR (Reference 2). The liquid wastes from the non-aerated and aerated wastes systems (including the laundry, hot shower, and decontamination station system) are normally processed through an evaporator (if required), a demineralizer treatment system, and 100% released to the environment. The boron in the reactor coolant in the chemical and volume control system (CVCS) is essentially recycled. The blowdown treatment system, at a total blowdown rate of 60 to 200 gpm per unit, is treated by demineralization and reclaimed, or discharged. The total activity released is estimated to be 3.2 curies per year (Ci/yr), excluding tritium, and 820 Ci/yr of tritium to be released from the liquid radwaste system for both units. The liquid blowdown system is estimated to release 0.199 Ci/yr excluding tritium.

Non-aerated drains from components within the reactor coolant system and a portion of the coolant letdown stream used for boron management will be processed through the chemical and volume control system (CVCS). Aerated drains from the floor drains, aerated equipment drains and leaks, decontamination drains,

laboratory and sample drains, will be processed through the waste evaporator system or the aerated drain demineralizer treatment system. The laundry-shower water and certain decontamination solutions will be treated in a special coagulation tank facility. All equipment in the liquid radwaste system is common to both units except the steam generator blowdown flash tanks, reactor coolant drain tanks, and drain tank pumps.

2. Gaseous Wastes

The evaluation of the gaseous wastes is described in the USAR (Reference 2). The gas decay tanks are designed to hold gases for the plant lifetime, but it was estimated that the release of one decay tank per year would account for unpredictable influent sources. This tank was estimated at 800 Ci/yr of Kr-85 (other isotopes decayed to negligible amounts) for the two units. The estimated gaseous release from containment purging is 478 Ci/yr of Xe-133 for the two units. The total is thus 1,278 Ci/yr for the two units.

A potential source of radioactivity is the atmospheric steam dump system during large power transients. These releases are expected to be infrequent and small. Specification 3.9 requires an inventory of the activity released from this source to assure compliance with effluent release limits.

3. Solid Wastes

Miscellaneous materials, such as paper, rags, and glassware, will be compacted for disposal when practical (Reference 2). Spent resins and concentrates from the waste evaporators will be solidified if required in 55-gallon drums or other regulation containers and stored in a shielded area prior to shipment offsite for burial. The total annual shipment is estimated to be 4000 cubic feet of solid or solidified concentrates and miscellaneous materials and 250-350 cubic feet of spent resins for the two units.

The storage facilities and packaging and shipping are designed according to 10CFR Part 71 and 49CFR 170-199.

C. Process and Effluent Radiological Monitoring System

The process radiation monitoring system is designed to provide information on radioactive concentrations in certain systems, leakage from one system to another, and radioactive concentrations released to the environment (Reference 3). The monitoring includes: containment or containment purge vent, shield building vent and auxiliary building vent monitoring separately for particulate, gas and iodine; steam generator blowdown liquid monitor; liquid waste disposal system effluent discharge line monitor; condenser air ejector gas monitor;

radwaste treatment building vent monitor; control room ventilation supply monitor; spent fuel pool air exhaust monitor; residual heat removal cubicle air exhaust monitor; containment fan cooling water and component cooling system monitors; and discharge canal effluent monitor.

The area radiation monitoring system is designed to provide information on radiation levels in various areas of the plant for personnel protection and for qualitative information of a systems condition such as a failed demineralizer. Ten channels monitor areas in the control room, containment, radwaste building, and auxiliary building.

These monitoring systems will detect, indicate, annunciate, and/or record the concentrations or levels of activity to verify compliance with 10CFR Part 20 and keep radiation levels as low as practicable.

References

1. USAR, Chapter 12
2. USAR, Section 9.1.1
3. USAR, Section 9.2.3

5.6 FUEL HANDLING

A. Criticality Consideration

The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class I (seismic) structures. The spent fuel pit has a stainless steel liner to ensure against loss of water (Reference 1).

The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The fuel is stored vertically in an array with the center-to-center distance between assemblies sufficient to assure $K_{eff} \leq 0.95$ even if unborated water were used to fill the pit. In addition, fuel in the storage pool shall have a U-235 loading of ≤ 39.0 grams of U-235 per axial centimeter of fuel assembly (average).

The criticality considerations as they relate to the dropping of a spent fuel cask (i.e., heavy load) drop onto the racks has been evaluated. The maximum K_{eff} has been calculated to be 0.949 at a water/UO_2 ratio of a 2.0 with a boron concentration of 1800 ppm.

B. Spent Fuel Storage Structure

The spent fuel storage pool is enclosed with a reinforced concrete building having 12- to 18-inch thick walls and roof (Reference 1). The pool and pool enclosure are Class I (seismic) structures that afford protection against loss of integrity from postulated tornado missiles. The storage compartments and the fuel transfer canal are connected by fuel transfer slots that can be closed off with pneumatically sealed gates. The bottoms of the slots are above the tops of the active fuel in the fuel assemblies which will be stored vertically in specially constructed racks.

The spent fuel pool has a reinforced concrete bottom slab nearly 6 feet thick and has been designed to minimize loss of water due to a dropped cask accident. In addition, the spent fuel cask will have an impact limiter attached or a crash pad will be in place in the pool which will have the capability to absorb energy of impact due to a cask drop. This will result in no structural damage taking place to the pool which would result in significant leakage from the pool. Piping to the pool is arranged so that failure of any pipe cannot drain the pool below the tops of the stored fuel assemblies.

C. Fuel Handling

The fuel handling system provides the means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves after post-irradiation cooling. The system consists of the refueling cavity, the fuel transfer system, the spent fuel storage pit, and the spent fuel cask transfer system.

Major components of the fuel handling system are the manipulation crane, the spent fuel pool bridge, the auxiliary building crane, the fuel transfer system, the spent fuel storage racks, the spent fuel cask, and the rod cluster control changing fixture. The reactor vessel stud tensioner, the reactor vessel head lifting device, and the reactor internals lifting device are used for preparing the reactor for refueling and for assembling the reactor after refueling.

Upon arrival in the storage pit, spent fuel will be removed from the transfer system and placed, one assembly at a time, in storage racks using a long-handled manual tool suspended from the spent fuel pit bridge crane. After sufficient decay, the fuel will be loaded into shipping casks for removal from the site. The casks will be handled by the auxiliary building crane.

The load drop consequences of a spent fuel cask for Prairie Island have been evaluated. It is not possible, due to physical constraints, for a cask to be dropped into the large pool (pool no. 2). A load path has been defined which provides for safe movement of the cask. Travel interlocks and mechanical stops prevent cask movement outside of this path. The only safety-related equipment that can be impacted directly during a cask drop along this path is the fuel stored in the small pool (pool no. 1). The consequences of this drop have been evaluated and found to meet the NRC staff criteria contained in NUREG-0612 if at least 50 days have elapsed since reactor shutdown for fission gas release considerations and the pool water contains at least 1800 ppm boron for criticality considerations. While 50 days was determined adequate, a minimum decay period of 5 years has been incorporated into these technical specifications to provide additional margin in meeting the criteria specified in NUREG-0612 for fission gas releases, while not restricting the plant's operational flexibility. A cask impact limiter or crash pad prevents significant structural damage to the pool floor.

The spent fuel cask will be lowered 66 feet from the auxiliary building to the railroad car for offsite transportation. Specification 3.8 will limit this loading operation so that if the cask drops 66 feet, there will not be a significant release of fission products from the fuel in the cask.

D. Spent Fuel Storage Capacity

The spent fuel storage facility is a two-compartment pool that, if completely filled with fuel storage racks, provides up to 1582 storage locations. The southeast corner of the small pool (pool no. 1) also serves as the cask lay down area. During times when the cask is being used, four racks are removed from the small pool. With the four storage racks in the southeast corner of pool 1 removed, a total of 1386 storage locations are provided. To allow insertion of a shipping cask, total storage is limited to 1386 assemblies, not including those assemblies which can be returned to the reactor.

Reference

1. USAR, Section 10.1

f. Investigations of all Reportable Events and events requiring Special Reports to the Commission.

g. Drills on emergency procedures (including plant evacuation) and adequacy of communication with offsite support groups.

h. All procedures required by these Technical Specifications, including implementing procedures of the Emergency Plan, and the Security Plan (except as exempted in Section 6.5.F), shall be reviewed initially and periodically with a frequency commensurate with their safety significance but at an interval of not more than two years.

i. Special reviews and investigations, as requested by the Safety Audit Committee.

j. Review of investigative reports of unplanned releases of radioactive material to the environs.

k. All changes to the Process Control Program (PCP) and the Offsite Dose Calculation Manual (ODCM).

5. Authority

The OC shall be advisory to the Plant Manager. In the event of a disagreement between the recommendations of the OC and the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed. A written summary of the disagreement will be sent to the General Manager Nuclear Plants and the Chairman of the SAC for review.

6. Records

Minutes shall be recorded for all meetings of the OC and shall identify all documentary material reviewed. The minutes shall be distributed to each member of the OC, the Chairman and each member of the Safety Audit Committee, the General Manager Nuclear Plants and others designated by the OC Chairman or Vice Chairman.

7. Procedures

A written charter for the OC shall be prepared that contains:

a. Responsibility and authority of the group

b. Content and method of submission of presentations to the Operations Committee

c. Mechanism for scheduling meetings

d. Provision for meeting agenda

6.5 PLANT OPERATING PROCEDURES

Detailed written procedures, including the applicable checkoff lists and instructions, covering areas listed below shall be prepared and followed. These procedures and changes thereto, except as specified in TS 6.5.F and G, shall be reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager.

A. Plant Operations

1. Integrated and system procedures for normal startup, operation and shutdown of the reactor and all systems and components involving nuclear safety of the facility.

2. Fuel handling operations

3. Actions to be taken to correct specific and foreseen potential or actual malfunction of systems or components including responses to alarms, primary system leaks and abnormal reactivity changes and including follow-up actions required after plant protective system actions have initiated.

4. Surveillance and testing requirements that could have an effect on nuclear safety.

5. Implementing procedures of the Facility Emergency Plan, including procedures for coping with emergency conditions involving potential or actual releases of radioactivity.

6. Implementing procedures of emergency plans for coping with earthquakes and floods. The flood emergency plan shall require plant shutdown for water levels at the site higher than 692 feet above MSL.

7. Implementing procedures of the fire protection program.

8. Implementing procedures for the Process Control Program and Offsite Dose Calculation Manual including quality control measures.

Drills on the procedures specified in A.3. above, shall be conducted as a part of the retraining program.

B. Radiological

Radiation control procedures shall be maintained and made available to all plant personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10CFR20. This radiation protection program shall be organized to meet the requirements of 10CFR20.

E. Offsite Dose Calculation Manual (ODCM)

The ODCM shall be approved by the Commission prior to initial implementation. Changes to the ODCM shall satisfy the following requirements:

1. Shall be submitted to the Commission with the Semi-Annual Radioactive Effluent Report for the period in which the change(s) were made effective. This submittal shall contain:

   a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with a revision date, together with appropriate analyses or evaluations justifying the change(s).

   b. a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. documentation of the fact that the change has been reviewed and found acceptable by the Operations Committee.

2. Shall become effective upon review and acceptance by the Operations Committee.

F. Security

Procedures shall be developed to implement the requirements of the Security Plan and the Security Contingency Plan. These implementing procedures, with the exception of those non-safety related procedures which govern work activities exclusively applicable to or performed by security personnel, shall be reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager. Security procedures not reviewed by the Operations Committee shall be reviewed and approved by the Supervisor, Security and Services.

G. Temporary Changes to Procedures

Temporary changes to Operations Committee reviewed procedures described in A,B,C,D,E and F above, which do not change the intent of the original procedure may be made with the concurrence of two individuals holding senior operator licenses. Such changes shall be documented, reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager within one month.

Temporary changes to security procedures not reviewed by the Operations Committee shall be reviewed by two (2) individuals knowledgeable in the area affected by the procedure.

6. Plant radiation and contamination surveys.

7. Changes made to the plant as it is described in the Final Safety Analysis Report, reflected in updated, corrected and as-built drawings.

8. Cycling beyond normal limits for those components that have been designed to operate safely for a limited number of cycles beyond such limits.

9. Reactor coolant system in-service inspections.

10. Minutes of meetings of the Safety Audit Committee.

11. Records of the service lives of all safety-related snubbers, including the date at which the service life commences and associated installation and maintenance records.

2. Occupational Exposure Report.[1] An annual report of occupational exposure covering the previous calendar year shall be submitted prior to March 1 of each year.

   The report should tabulate on an annual basis the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor opertions and surveillance, inservice 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, or film badge measurements. Small exposures totalling less than 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.

3. Monthly Operating Report. A monthly report of operating statistics and shutdown experience covering the previous month shall be submitted by the 15th of the following month to the Director, Office of Resource Management, US Nuclear Regulatory Commission, Washington, DC 20555.

(1) This report supplements the requirements of 10CFR20, section 20.407. If 10CFR20, Section 20.407 is revised to include such information, this Specification is unnecessary.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 80 AND 73

TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

INTRODUCTION

By letter dated July 15, 1986, Northern States Power Company (NSP), the licensee, requested amendments to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 (PINGP). The amendments would change the technical specifications by revising (1) the existing reactor coolant system heatup and cooldown curves that would increase the effective limitation from 10 to 15 effective full power years, (2) the Radiation Environmental Monitoring Program, (3) design features appearing in Section 5 of the technical specifications, (4) Security Plan Implementing Procedures, and (5) administrative changes appearing in Section 6 of the technical specifications (TS). Specifically, the amendment changes would affect Sections TS 3.1, TS 4.10-1, TS 5.1, TS 5.2, TS 5.3, TS 5.4, TS 5.5, TS 5.6, TS 6.2, TS 6.5, TS 6.6, TS 6.7 TS 6.8, Tables TS 3.1-1, TS 3.1.2, TS 4.10-1 and TS 4.10-2, and Figures TS 3.1-1 thru 5. These changes consider the increased neutron fluence up to 15 effective full power years for the limiting material of the reactor pressure vessel in regard to the heatup and cooldown curves and other changes involving error corrections, clarifications, minor administrative changes and changes to attain consistency throughout different sections of the TS.

DISCUSSION AND EVALUATION

A. Heatup and Cooldown Curves

Discussion

The licensee requested changes to the pressure temperature limits described in Figures TS 3.1-1 and TS 3.1-2 and deletion of Figures TS 3.1-3 and TS 3.1-4 and Tables TS 3.1-1 and TS 3.1-2 from the technical specifications. The changes in the heatup and cooldown limits are based on the test results from the Prairie Island surveillance program, contained in References 1 through 5. References 1 through 5 were submitted for staff review in letters from the licensee dated September 17, 1977, October 12, 1982, April 25, 1986, December 6, 1978 and May 13, 1981, respectively.

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR Part 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR Part 50 are dependent upon the initial reference temperature (RT_{NDT}) for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material. The Prairie Island reactor vessels were procured to ASME Code requirements, which did specify fracture toughness testing to determine the initial RT_{NDT} for each vessel material. The licensee indicates that the initial RT_{NDT} for the limiting materials in the closure flange and beltline regions of the Prairie Island vessels were estimated using the method recommended by the staff in Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," and from surveillance program data. These methods result in an initial RT_{NDT} for the limiting beltline base metal and weld metal of 14°F and 0°F, respectively, and an initial RT_{NDT} for the limiting closure flange material of -4°F.

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensee using the empirical relationship documented in Regulatory Guide 1.99, Rev. 1, April 1977, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." This method of predicting neutron irradiation damage is dependent upon the predicted amount of neutron fluence and the amounts of residual elements (copper and phosphorus) in the beltline material. The neutron fluence used to predict neutron irradiation damage is based on the calculated neutron fluence at the vessel location with peak flux. These neutron fluence predictions were verified by measurements from passive neutron flux monitors and by analysis, which was made with the DOT Code, a two-dimensional discrete ordinates code. Inputs into the analysis included 47 neutron energy groups, P3 expansion of the scattering cross section, and power distributions representative of time-averaged conditions derived from statistical studies of long-term operation of Westinghouse 2-loop plants. The cross sections used in the analysis were obtained from the SAILOR cross section library. Using this method of analysis, the measured average fast (E > 1.0 MeV) neutron flux derived from the five threshold reaction dosimeters in Capsule U is 1.49×10^{11} η/cm^2-sec. The design basis calculated fast neutron flux is 1.51×10^{11} η/cm^2-sec. Since the calculated flux is greater than the measured flux from the capsule dosimetry, neutron fluence calculations using the calculated flux should conservatively predict neutron fluence. An evaluation of the licensee's neutron fluence estimates and protection against pressurizer thermal shock (PTS) events is elaborated in Enclosure 1, which was discussed in Reference 6.

The prediction curves in Regulatory Guide 1.99, Rev. 1 are dependent upon the amounts of residual elements in the beltline material. The licensee's letter dated January 10, 1986, identified the residual elements in the core region welds and forgings.

Exhibit E, in the January 10, 1986 letter from the licensee, identified the weld metal with the greatest amount of copper. This weld metal was located in the intermediate to lower shell weld in the reactor vessel of Unit 2. However, in a telecon on September 8, 1986 the licensee and Westinghouse informed the staff that weld metal with the greatest amounts of copper in the intermediate to the lower weld of Unit 2 was only used in the root passes of the weld joint. This weld joint's geometry was described as a double U with the root located half way between the inside and outside surfaces. The thru-wall dimension of the root is approximately 1/2 inch. Since the minimum wall thickness of the beltline weld is 6.7 inches, the weld metal in the root is only a small part of the overall weld and will not significantly affect the weld's fracture toughness. In addition, the location of the root at the mid-wall thickness reduces the effect of neutron irradiation damage to the material. Hence, weld metal with the greatest amount of copper will not be the limiting material and will not affect the proposed pressure-temperature limits. Based on the above discussion, the limiting beltline material for both units is weld metal used to fabricate the intermediate to lower shell weld in Unit 1. This weld metal was used in the Unit 1 surveillance program. The licensee indicated that information discussed in the telecon will be included in a future revision to the USAR.

Tables 1 and 2 compare the ΔRT_{NDT} measured from surveillance material in Units 1 and 2, respectively, to the values predicted using Regulatory Guide 1.99, Rev. 1. Except for HSST Plate 02, the values predicted by the regulatory guide exceed the values measured from the surveillance material. HSST Plate 02 is an experimental heat of material, which was not removed from material used to fabricate the Prairie Island reactor vessels. All other materials identified in Tables 1 and 2 are representative of the materials used to fabricate the beltlines of the Prairie Island reactor vessels. Since the values predicted for ΔRT_{NDT} using Regulatory Guide 1.99, Rev. 1, exceed the values measured from surveillance materials that represent the Prairie Island reactor vessel beltlines, the prediction method in Regulatory Guide 1.99, Rev. 1, should conservatively predict the ΔRT_{NDT} for the Prairie Island reactor vessel beltline materials.

The information contained in TS Tables TS 3.1-1 and TS 3.1-2 and Figures TS 3.1-3 and TS 3.1-4 are either reported in the USAR, in regulatory guides issued by the NRC, or in surveillance reports docketed by the licensee. In addition, they are not used during plant operation. Hence, they may be deleted from the plant's TS.

Evaluation

The staff has used the method of calculating pressure-temperature limits in USNRC Standard Review Plan 5.3.1, NUREG-0800, Rev. 1, July 1981 to evaluate the proposed pressure-temperature limits. The amount of neutron irradiation damage was calculated using design basis calculated neutron fluences and the Regulatory Guide 1.99, Rev. 1, prediction curves. Our

conclusion is that the proposed pressure-temperature limits meet the safety margins of Appendix G, 10 CFR Part 50 for 15 EFPY and the changes to TS Section 3.1(B) and bases are acceptable.

B. Radiation Environmental Monitoring Program

Discussion

The licensee proposes the following changes to the Radiation Environmental Monitoring Program portion of the technical specifications.

1. In the specification, TS 4.10.A4, Table TS 3.12-2 is referred to regarding radionuclides. The Table number (3.12-2) is incorrect. The correct Table number should be 4.10-3.

2. The specification Table TS 4.10-1 (Page 4 of 4) requires one sample of corn is to be obtained from a farm having highest D/Q and another sample 10-20 miles away. The corn sampling would be changed by the proposed amendment to allow corn sampling taken from any field that is irrigated by water into which liquid plant wastes have been discharged. The existing specifications assumed agricultural crops of the area would take river water for irrigating the fields in the area. However, over time a uniform process of deep well irrigation has been developed and the use of river water irrigation has been discontinued. Therefore, corn sampling collection under the existing TS methods is no longer a valid monitoring method to determine the impact of liquid releases.

3. The notes on TS Table TS 4.10-2 would be clarified by the proposed change by specifying which isotopes apply in considering the lower limit of detection (footnote e) and clarifying note (d) specifying that lower limits of detection of Iodine-131 only applies when I-131 analysis is specified.

4. The airborne radioiodine and particulates sample locations have been clarified by eliminating a potential inconsistency that could arise between the sample locations specified in the TS (Table TS 4.10-1, Page 1 of 4) and locations specified in ODCM, Figure 5.1-1.

Evaluation

All changes proposed for Section 4 of the TS dealing with the Radiation Environmental Monitoring Program have been reviewed by the staff. This review disclosed that the proposed changes are merely a correction in the case of Table TS 4.10-3, an error in corn sampling requirements, and a clarification for the lower limit of detection of radioiodine and sampling location for monitoring airborne radioiodine and particulates. These changes in no way alter the intent of the TS requirements related to the environmental monitoring program nor do they reduce the level of plant safety or alter the protective level of the environmental conditions in the vicinity of the plant site. Therefore, the proposed changes are judged consistent with regulatory requirements and, therefore, are acceptable.

C. Design Features-Section 5

Discussion and Evaluation

Section 5 of the TS deals with design features of the plant. The licensee's proposed changes reflect changes associated with the implementation of 10 CFR Part 50, Appendix I and methods currently used to process radioactive waste. Other proposed changes are administrative in nature, designed to complement the upgrading of the TS. Section 5.0 of the TS provides a general description of the plant and its systems and does not impose requirements or limiting conditions of operations.

All of these changes were reviewed by the staff. The staff finds that the proposed changes enhance the TS by providing the description of plant systems as they actually exist in the plant and upgrade references to current standards. These proposed changes will not remove or relax any existing requirements that appear in other parts of the TS and, therefore, will have no affect on the probability or consequences of accidents previously considered. In addition, the proposed changes will not remove or relax any existing requirements needed to provide reasonable assurance that the health and safety of the public will be compromised in any way. On this basis, the staff finds the proposed changes acceptable.

D. Security Plan Implementing Procedures

Discussion and Evaluation

Section 6.5 of the TS is concerned with the plant operating procedures that shall be reviewed by the Operations Committee and approved by a member of plant management. The proposed change would clarify that the Operations Committee would not be required to review nonsafety-related Security Plan implementing procedures and this review function would be exclusively performed by the guards. A similar request dealing with the clarification of the reviews of the nonsafety-related Security Plan implementing procedures was found acceptable by the staff for the Monticello Nuclear Generating Plant (Docket No. 50-263) as noted by Amendment No. 25 dated August 15, 1984. It was never the intent of the staff to have the Operations Committee review nonsafety-related Security Plan implementing procedures. Therefore, such an interpretation of this administrative requirement is considered to be in error. The staff has reviewed the licensee's proposed change and finds that it would eliminate this potential misinterpretation in the administrative requirements of the TS. On this basis, we find the change acceptable.

E. Administrative-Section 6

Discussion and Evaluation

The licensee's proposed changes involve establishing a title to the address for submitting the monthly operating report and deleting the record keeping requirements for the environmental qualification records. The proposed changes would impact TS Sections TS 6.7.A.3 and TS 6.8. The change involving the address for submitting the monthly report is editorial in nature and has no affect on the plant safety level. TS 6.8 dealing with environmental qualification of safety-related equipment specified the schedular requirements and the interim guidelines of NUREG-0588 which have become obsolete and does not serve a useful purpose with the issuance of the rule in 10 CFR 50.49. The recent change to 10 CFR Part 50 to include section 50.49 specifies the requirements for environmental qualification of electrical equipment including the record keeping which the licensee must comply. Thus, the deletion of TS 6.8 would eliminate this duplication and potential confusing requirement concerning environmental qualification of electrical equipment. In addition, the deletion of this requirement from the TS will in no way result in a reduction in the plant margin of safety. On the above basis, we find the proposed change deleting section 6.8 and 6.6.B.11 acceptable.

ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

These amendments also involve changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors:

B. Elliot
C. Willis
L. Lois
D. DiIanni

Date: November 14, 1986

References:

(1) Westinghouse Report WCAP 8916, "Analysis of Capsule V From Northern States Power Company Prairie Island Unit No. 1 Reactor Vessel Radiation Surveillance Program", Davidson et al, August 1977.

(2) Westinghouse Report WCAP 10102, "Analysis of Capsule P From Northern States Power Company Prairie Island Unit 1 Reactor Vessel Radiation Surveillance Program", Yanichko et al, May 1982.

(3) Westinghouse Report WCAP 11006, "Analysis of Capsule R From Northern States Power Company Prairie Island Unit 1 Reactor Vessel Radiation Surveillance Program", Boggs et al, February 1986.

(4) Westinghouse Report WCAP 9212, "Analysis of Capsule V From Northern States Power Company Prairie Island Unit No. 2 Reactor Vessel Radiation Surveillance Program", Davidson et al, November 1977.

(5) Westinghouse Report WCAP 9877, "Analysis of Capsule T From Northern States Power Company Prairie Island, Unit No. 2 Reactor Vessel Radiation Surveillance Program", Yanichko et al, March 1981.

(6) Memorandum from C. Berlinger to D. DiIanni dated August 28, 1986.

TABLE I

Comparison of Increase in Reference Temperature (ΔRT_{NDT}) Measured from Unit 1 Surveillance Material to the ΔRT_{NDT} Predicted by Regulatory Guide 1.99, Rev. 1

Capsule/Surveillance Material	Neutron Fluence ($X10^{19}$ n/cm²)	RT_{NDT} Measured From Surveillance Material (°F)	RT_{NDT} Predicted by Regulatory Guide 1.99 Rev. 1 (°F)
Capsule R[a]			
Weld Metal	4.03	117	271
Forging C (Tang. orient.)	4.03	87	131
(Axial orient.)	4.03	80	131
HSST Plate 02	4.03	186	241
Capsule P[b]			
Weld Metal	1.25	42	151
Forging C (Tang. orient.)	1.25	20	73
(Axial orient.)	1.25	37	73
HSST Plate 02	1.25	156	134
Capsule V[c]			
Weld Metal	.546	25	100
Forging C (Tang. orient.)	.546	38	48
(Axial orient.)	.546	24	48
HSST Plate 02	.546	110	89

(a) Data from Reference 3
(b) Data from Reference 2
(c) Data from Reference 1

Table II

Comparison of Increase in Reference Temperature (ΔRT_{NDT}) Measured from Unit 2 Surveillance Material to the ΔRT_{NDT} Predicted by Regulatory Guide 1.99, Rev. 1

Capsule/Surveillance Material	Neutron Fluence ($X10^{19}\eta/cm^2$)	RT_{NDT} Measured From Surveillance Material (°F)	RT_{NDT} Predicted by Regulatory Guide 1.99 Rev. 1 (°F)
Capsule T[a]			
Weld Metal	1.05	60	99
Forging (Tang. orient.)	1.05	55	62
(Axial orient.)	1.05	35	62
HSST Plate 02	1.05	160	123
Capsule V[b]			
Weld Metal	.586	60	74
Forging (Tang orient.)	.586	35	46
(Axial orient.)	.586	30	46
HSST Plate 02	.586	125	92

(a) Data from Reference 5
(b) Data from Reference 4

Enclosure 1

Prairie Island Units 1 and 2, Fast Neutron Fluence Reevaluation for the Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61)

By letter dated July 15, 1986, Northern States Power Company, (the licensee) for the Prairie Island Units 1 and 2 plants, requested a license amendment for the Pressure-Temperature curves revision as required by 10 CFR Part 50, Appendix G (Reference 1). The Pressure-Temperature curves and the Pressure Vessel fast neutron ($E \geq 1.0$ MeV) fluence were reevaluated for 15 effective full power years. The projected estimate of the revised fluence was found to be 21% higher than the original estimate. Then, according to 10 CFR 50.61, the licensee submitted the new projected fluence and RT_{PTS} values. In addition, in the PTS evaluations, the staff requested that the licensee submit the RT_{PTS} reevaluations along with the Appendix G submittals (Reference 2).

The following is the 60 effective full power years RT_{PTS} for Prairie Island Units 1 and 2 with the new values of the fluence. The equation specified in 10 CFR 50.61 as applicable for the Prairie Island Units 1 and 2 is:

$$RT_{PTS} = I+M+(-10+470.Cu+350.Cu.Ni)f^{0.27}$$

where:

	Unit 1	Unit 2
I = Initial RT_{NDT}	= 0°F	0°F
M = Uncertainty Margin	= 59°F	59°F
Cu = w/o Copper in weld W-3	= 0.14	0.19
Ni = w/o Nickel in weld W-3	= 0.17	0.13
f = peak fluence ($E \geq 1.0$ MeV) on weld W-3 x 10^{-19} cm^2/n	= 9.75	9.75

Then: Unit 1

$$RT_{PTS} = 59 + (-10 + 470 \times 0.14 + 350 \times 0.14 \times 0.17) \times 9.75^{0.27}$$
$$= 59 + 64.1 \times 1.85 = 177.6°F$$

Unit 2

$$RT_{PTS} = 59 + (-10 + 470 \times 0.19 + 350 \times 0.19 \times 0.13) \times 9.75^{0.27}$$
$$= 59 + 87.95 \times 1.85 = 221.7°F$$

The applicable criteria specified in 10 CFR 50.61 is 300°F, therefore, the revised values meet the criteria for 60 effective full power years by a large margin.

References

1. Letter from D. Musolf, Northern States Power Company, to Director, NRR, dated July 15, 1986.

2. Memoranda from C. E. Rossi to D. DiIanni, "Fast Neutron Fluence for the Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, dated April 28 and May 12, 1986, for Units 2 and 1, respectively.