June 24, 1982

Docket No. 50-298

Mr. J. M. Pilant, Director
Licensing & Quality Assurance
Nebraska Public Power District
P. O. Box 499
Columbus, Nebraska 68601

Dear Mr. Pilant:

Re: Cooper Nuclear Station

The Commission has issued the enclosed Amendment No. 80 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. This amendment consists of changes to the Technical Specifications in response to your application dated April 30, 1982 and subsequent discussions between representatives of your staff and the NRC staff.

These revisions to the Technical Specifications include the following: (1) changes to MCPR values and RBM upscale trip level setting; (2) elimination of temporary restrictions no longer applicable; (3) clarification of definitions and bases sections; (4) title changes (i.e., AEC to NRC, and Nebraska Public Power District Management title changes); (5) inclusion of references to Regulatory Guide 10.1 distribution requirements; (6) correction of typographical and grammatical errors and editorial changes; (7) consolidation of blank pages to reduce volume; (8) updating of references; (9) elimination of historical information from bases section (i.e., deletion of initial startup test program); (10) identification of snubbers added or removed from drywell; (11) deletion of references to the containment atmospheric dilution (CAD) system since an air containment atmospheric dilution (ACAD) system was installed and the addition of the ACAD system isolation valves to the list of primary containment testable isolation valves; and (12) the inclusion of two smoke detectors installed in the cable expansion room.

We have reviewed your changes to the MCPR values and RBM upscale trip level setting (identified as Revision 1) and since the proposed changes are bounded by your currently approved Technical Specifications, they are acceptable.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

The identification of snubbers added or removed from the drywell and the inclusion of smoke detectors installed in the cable expansion room (identified as revisions 10 and 12) are required to be included in the Technical Specifications and are therefore acceptable.

The deletion of all references to a CAD system (identified as revision 11), since an ACAD system was installed, is acceptable. In addition, since the ACAD system has never been approved by the NRC, utilization of the isolation valves in this system to isolate it from the primary containment and the inclusion of those isolation valves to the list of primary containment testable isolation valves is also acceptable.

We also reviewed the safety significance of the remaining changes to your Technical Specifications which are primarily to: update the current Technical Specifications; correct typographical and grammatical errors; make editorial corrections; and reflect organizational title changes. Since these proposed changes do not diminish the level of safety provided by the existing Technical Specifications, they are acceptable.

In your submittal, typographical corrections were requested to pages 5a, 94a, 107, 119, 131, 206, 209, and 216b. Since the errors identified were not in our Technical Specifications, these pages were deleted from the enclosed amendment. We have included the deletion of pages 198a and 200a in this amendment. The information on these pages is contained on pages 198 and 200. We have also included corrections to page 193 in this amendment. The issuance of amendments 43 and 45 out of sequence resulted in the need for deletion and inclusion of these pages.

Based on the foregoing, we have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, negative declaration or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

We have further concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Byron L. Siegel, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 80
2. Notice

cc: w/enclosures
See next page

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Mr. J. M. Pilant
Nebraska Public Power District

cc:

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

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 80
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District dated April 30, 1982 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 (i) 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 License No. DPR-46 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, 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

D. B. Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 24, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 80

FACILITY OPERATING LICENSE NO. DPR-46

Docket No. 50-298

Revise Appendix A as follows:

Remove	Insert
ii	ii
iv	iv
1	1
4	4
5	5
7	7
8	8
14	14, 15, 16
15	--
16	--
17	17
18	18
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24	24
25	25
26	26
27	27
31	31
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63	63
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93a	--
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99	99
100	100
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106	--
108	108
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125	125
126	126
129	129, 130
130	--
132	132
133	133
133a	133a
134	134
1371	1371

Remove	Insert
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152	152
162	162
163	163
164	164
165	165
165a	165a
166	166
169	169
173	173
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178a	178a
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187	187, 188, 189, 190, 191, 192
188	--
189	--
190	--
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193	193
195	195
196	196
198a	--
199	199
200	200, 201, 202
200a	--
201	--
202	--
212c	212c
212d	212d
212e	212e
214	214

Remove	Insert
214a	214a
214b	214b
214c	214c
214d	--
214e	--
215b	215b
216k	216k
217	217
219a	219a
220	220
221	221
222	222
223	223
227	227
230	230
231	231
232	232
234	234

Revise Appendix B as follows

Remove	Insert
3	3 through 5
4	--
5	--
31	31 through 40
32	--
33	--
34	--
35	--
36	--
37	--
38	--
39	--
40	--

Remove	Insert
68	68 through 76
69	--
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72	--
73	--
74	--
75	--
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1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Thermal Parameters

1. Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation. (Reference NEDO-10958)

2. Maximum Fraction of Limiting Power Density - The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

3. Minimum Critical Power Ratio (MCPR) - The minimum critical power ratio corresponding to the most limiting fuel assembly in the core.

4. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type. Design LHGR's are 18.5 KW/ft for 7x7 bundles and 13.4 KW/ft for 8x8 bundles.

5. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

B. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation is not defined as a core alteration.

C. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

D. Design Power - Design power means a steady-state power level of 2486 thermal megawatts. This is 104.4% of Rated Power (105% of rated steam flow) and the power to which the safety analysis applies.

E. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required to maintain the consequences of postulated accidents within acceptable limits.

K. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent a margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

L. Mode - The reactor mode is established by the mode selector-switch. The modes include refuel, run, shutdown and startup/hot standby which are defined as follows:

1. Refuel Mode - The reactor is in the refuel mode when the mode switch is in the refuel mode position. When the mode switch is in the refuel position, the refueling interlocks are in service.

2. Run Mode - In this mode the reactor system pressure is at or above 825 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

3. Shutdown Mode - The reactor is in the shutdown mode when the reactor mode switch is in the shutdown mode position.

4. Startup/Hot Standby - In this mode the reactor protection scram trips initiated by the main steam line isolation valve closure are bypassed when reactor pressure is less than 1000 psig, the low pressure main steam line isolation valve closure trip is bypassed, the reactor protection system is energized with APRM (15% SCRAM) and IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

M. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

N. Operating - Operating means that a system or component is performing its intended functions in its required manner.

O. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

P. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

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

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3. All automatic containment isolation valves are operable or de-activated in the isolated position.

4. All blind flanges and manways are closed.

Q. Rated Power - Rated power refers to operation at a reactor power of 2381 megawatts thermal. This is also termed 100% power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, is 104.4% of rated power (105% of rated steam flow), which corresponds to 2486 megawatts thermal.

R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated power.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

T. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling.

U. Safety Limits - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

V. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

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

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

W. Shutdown - The reactor is in a shutdown condition when the mode switch is in the "Shutdown" or "Refuel" position.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F and the reactor vessel vented.

X. Spiral Reload - Pertains to the spiral reloading of the core with fuel, at least 50% of which has previously accumulated a minimum exposure of 1000 MWD/T.

1.1 (Cont'd)

D. Cold Shutdown

Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 18 in. above the top of the normal active fuel zone (top of active fuel is defined in Figure 2.1.1).

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2.1.A (Cont'd)

a. In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%) \left[\frac{FRP}{MFLPD}\right]$$

where,

FRP = fraction of rated thermal power (2381 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7x7 fuel and 13.4 KW/ft for 8x8 fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating values is less than the design value of 1.0, in which case the actual operating value will be used.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

b. APRM Flux Scram Trip Setting (Refuel or Start and Hot Standby Mode)

When the reactor mode switch is in the REFUEL or STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

c. IRM

The IRM flux scram setting shall be $\leq$120/125 of scale.

2.1.A.1 (Cont'd)

d. APRM Rod Block Trip Setting

The APRM rod block trip setting shall be:

$S_{RB} \leq 0.66\ W + 42\%$

where:

S_{RB} = Rod block setting in percent of rated thermal power (2381 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate is that recirculation flow rate which provides 100% coreflow at 100% power)

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S_{RB} \leq (0.66\ W + 42\%) \left[\frac{FRP}{MFLPD}\right]$$

where,

FRP = fraction of rated thermal power (2381 MWt)

MFLPD - maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7x7 fuel and 13.4 KW/ft for 8x8 fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

2. Reactor Water Low Level Scram and Isolation Trip Setting (except MSIV)

$\geq$ +12.5 in. on vessel level instruments.

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2.1 Bases:

The abnormal operational transients applicable to operation of the CNS Unit have been analyzed throughout the spectrum of planned operating conditions up to 105% of rated steam flow. The analyses were based upon plant operation in accordance with Reference 3. In addition, 2381 MWt is the licensed maximum power level of CNS, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greater significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 25% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

This choice of using conservative values of controlling parameters and initiating transients at the design power level produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

2.1 Bases: (Cont'd)

In summary:

i. The abnormal operational transients were analyzed to 105% of rated steam flow.

ii. The licensed maximum power level is 2381 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (2381 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

5. Main Steam Line Isolation Valve Closure on Low Pressure

The low pressure isolation of the main steam lines (Specification 2.1.A.6) was provided to protect against rapid reactor depressurization.

B. Reactor Water Level Trip Settings Which Initiate Core Standby Cooling System (CSCS)

The core standby cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature, to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each Core Standby Cooling System component was established based on the reactor low water level scram set point. To lower the set point of the low water level scram would increase the capacity requirement for each of the CSCS components. Thus, the reactor vessel low water level scram was set low enough to permit margin for operation, yet will not be set lower because of CSCS capacity requirements.

The design for the CSCS components to meet the above guidelines was dependent upon three previously set parameters: The maximum break size, low water level scram set point and the CSCS initiation set point. To lower the set point for initiation of the CSCS may lead to a decrease in effective core cooling. To raise the CSCS initiation set point would be in a safe direction, but it would reduce the margin established to prevent actuation of the CSCS during normal operation or during normally expected transients.

Transient and accident analyses reported in Section 14 of the Final Safety Analyses Report demonstrate that these conditions result in adequate safety margins for the fuel.

C. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10801, Feb., 1973.

2. Station Safety Analysis Report (Section XIV).

3. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (applicable reload document).

1.2 BASES

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The safety limits for reactor coolant system pressure are derived directly from unacceptable safety results 1-3, 2-3, and 3-3 of the Station Nuclear Safety Operational Analysis (Appendix G). These unacceptable results require that applicable code limits for the nuclear system not be exceeded. Thus, the safety limits are direct measures of the unacceptable safety results.

The safety limits for the reactor coolant system pressure have been selected so that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are set high enough that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are selected to be the lowest transient overpressures allowed by the applicable codes. ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The reactor vessel steam dome pressure of 1337 psig is equivalent to a pressure of 1375 psig at the vessel bottom. The design pressure (1250 psig) of the reactor vessel is established so that, when the 10 percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code, Section III, for pressure transients, is added to the design pressure, a transient pressure limit of 1375 psig at the vessel bottom is established. Correspondingly, the suction and discharge design pressures (1148 and 1274 psig) of the reactor coolant system piping are set so that, when the 20 percent allowance (230 and 254 psi) allowed by the USAS Piping Code, Section B31.1 for pressure transients, are added to the design pressures, transient pressure limits of 1378 and 1528 psig are established. Thus, the pressure safety limit for power operation is established at 1375 psig, the lowest transient overpressure allowed by the pertinent codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

Reference 6 provides the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1375 psig, given in Subsection IV2 of the Safety Analysis Report, is well above the peak pressure produced by the pressurization transient described in Reference 6. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result from reasonably expected pressurization transients.

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These pressures create a consistent design with assurance that, if the pressure within the reactor vessel does not exceed 1375 psig, the pressures within the piping cannot exceed their respective transient pressure limits because of static and pump heads.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

REFERENCES

1. Station Safety Analysis (Section XIV)

2. ASME Boiler and Pressure Vessel Code, Section III

3. USAS Piping Code, Section B31.1

4. Reactor Vessel and Appurtenances Mechanical Design (Subsection IV-2)

5. Station Nuclear Safety Operational Analysis (Appendix G)

6. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (applicable reload document).

2.2 BASES

The 8 relief valves and 3 safety valves are sized and set pressures are established in accordance with the requirements of Section III of the ASME Code. The relief valve settings satisfy the Code requirements that the lowest valve set point be at or below the vessel design pressure of 1250 psig. These settings are also sufficiently above the normal operating pressure range to prevent unnecessary cycling caused by minor transients. The postulated transients where inherent relief valve actuation is required are described in Section XIV of the Safety Analysis Report.

Reanalysis in Reference 6 for the case of MSIV-Closure with flux scram transient results in a peak pressure at the vessel bottom which is below the maximum of 110 percent of design pressure allowed by the Code. This is adequate margin to ensure that the 1375 psig pressure safety limit is not exceeded. A sensitivity study on peak vessel pressure to the failure to open of one of the lowest set-point safety valves was performed for a typical high power density BWR (Reference 7). The study is applicable to the Cooper reactor and shows that the sensitivity of a high power density plant to the failure of a safety valve is approximately 20 psi. A plant specific analysis for the Cooper overpressure transient would show results equal to or less than this value.

The design pressure of the shutdown cooling piping of the Residual Heat Removal System is not exceeded with the reactor vessel steam dome less than 75 psig.

REFERENCES

1. Topical Report, "Summary of Results Obtained from a Typical Startup and Power Test Program for a General Electric Boiling Water Reactor", General Electric Company, Atomic Power Equipment Department (APED-5698)

2. Station Nuclear Safety Operational Analysis (Appendix G)

3. Station Safety Analysis (Section XIV)

4. Control and Instrumentation (Section VII)

5. Summary Technical Report of Reactor Vessel Overpressure Protection (Question 4.20, Amendment 11 to SAR)

6. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (applicable reload document).

7. Letter from I. F. Stuart (GE) to v. Stello (NRC) dated December 23, 1975.

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective:

To assure the operability of the reactor protection system.

Specification:

The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1.

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4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2 respectively.

B. Daily during reactor power operation, the peak heat flux and maximum fraction of limiting power density shall be checked and the SCRAM and APRM Rod Block settings given by equations in Specification 2.1.A.1 and 2.1.B shall be calculated if maximum fraction of limiting power density exceeds the fraction of rated power.

C. During reactor power operation with MFLPD > FRP, MCPR shall be calculated at least daily and following any change in power level or distribution that would cause operation with a limiting control rod pattern as defined in Specification 3.3.B.5 and associated bases.

D. When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels.

11. The APRM downscale t[illegible] function is only active when [illegible]he reactor mode switch is in run.

12. The APRM downscale trip is automatically bypassed when the mode switch is not in RUN.

13. An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there is less than 11 operable LPRM detectors to an APRM.

14. W is the recirculation flow in percent of rated flow.

15. This note deleted.

16. The 15% APRM scram is bypassed in the RUN mode.

17. The APRM and IRM instrument channels function in both the Reactor Protection System and Reactor Manual Control System (Control Rod Withdraw Block, Section 3.2.C.). A failure of one channel will affect both of these systems.

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TABLE 4.1.1 (Page 1)
REACTOR PROTECTION SYSTEM (SCRAM INSTRUMENTATION) FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

Instrument Channel	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Once/3 Months
RPS Channel Test Switch (5)	A	Trip Channel and Alarm	Each Refueling Outage
IRM			
High Flux	C	Trip Channel and Alarm (4)	Before each startup and weekly when required to be operable.
Inoperative	C	Trip Channel and Alarm	Before each startup and weekly when required to be operable.
APRM			Before each startup and weekly when required to be operable.
High Flux (15%)	C	Trip Output Relays (4)	
High Flux	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	Trip Output Relays (4)	Once/Month (1)
High Reactor Pressure NBI-PS-55 A,B,C,& D	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure PC-PS-12 A,B,C,& D	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water level (6) NBI-LIS-101 A,B,C, & D	A	Trip Channel and Alarm	Once/Month (1)

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NOTES FOR TABLE 4.1

1. Initially once per month until exposure (M as defined on Figure 4.1.1) is 2.0 x 10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months after review and approval of the NRC. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of CNS.

2. A description of the three groups is included in the Bases of this Specification.

3. Functional tests are not required when the systems are not required to be operable or are tripped. If reactor startups occur more frequently than once per week, the maximum functional test frequency need not exceed once per week.

   If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. Test RPS channel after maintenance.

6. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the monthly functional test program.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The designed system response times from the opening of the sensor contact up to and including the opening of the trip actuator has been shown by start-up testing to not exceed 50 milliseconds.

The reactor protection system is of the dual channel type (Reference subsection VII.2 FSAR). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection

4.1 BASES

A. The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in reference (6). This concept was specifically adapted to the one out of two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failures such as blown fuses, ruptured bourdon tubes, faulted amplifiers, and faulted cables, which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups for functional testing. These are:

A. On-off sensors that provide a scram trip function.

B. Analog devices coupled with bi-stable trips that provide a scram function.

C. Devices which only serve a useful function during some

3.1 BASES (cont'd.)

against short reactor periods in these ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 36 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 36 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram functions (reference paragraph VII.5.4 FSAR). Thus, the IRM and APRM are required in the "Refuel" and "Start/Hot Standby" modes. In the power range the APRM system provides required protection (refer-

4.1 BASES (cont'd.)

revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in the Group (B) devices to calculate their "unsafe" failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 20 X 10^{-6} failures/hour. The bi-stable trip circuits are predicted to have an unsafe failure rate of less than 2 X 10^{-6} failures/hour. Considering the two hour monitoring interval for the analog devices as assumed above, and a weekly test interval for the bi-stable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

The bi-stable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1.1. There are numerous identical bi-stable devices used throughout the plant's instrumentation system. Therefore, significant data on the failure rates for the bi-stable devices should be accumulated rapidly.

The frequency of calibration of the APRM Flow Biasing Network has been established as each refueling outage. The flow biasing network is functionally tested at least once per month and, in addition, cross calibration checks of the flow input to the flow biasing network can be made during the functional test by direct meter reading. There are several instruments which must be calibrated and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a

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LIMITING CONDITIONS FOR OPERATION

3.1 BASES

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SURVEILLANCE REQUIREMENTS

4.1 BASES (Cont'd)

flux is very high. Therefore, with MFLPD $\leq$ FRP there is no technical requirement for calculating MCPR. However, to be consistant with Section 4.11.C, MCPR shall be determined daily during reactor power operation at > 25% rated thermal power. With MFLPD greater than FRP, a daily calculation of MCPR is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached insures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.2 Protective Instrumentation	4.2 Protective Instrumentation
Applicability:	Applicability:
Applies to the plant instrumentation which initiates and controls a protective function.	Applies to the surveillance requirement of the instrumentation that initiates and controls protective function.
Objective:	Objective:
To assure the operability of protective instrumentation.	To specify the type and frequency of surveillance to be applied to protective instrumentation.
Specifications:	Specifications:
A. Primary Containment Isolation Functions	A. Primary Containment Isolation Functions
When primary containment integrity is required, the limiting conditions for operation for the instrumentation that initiates primary containment isolation are given in Table 3.2.A.	Instrumentation shall be functionally tested and calibrated as indicated in Table 4.2.A. System logic shall be functionally tested as indicated in Table 4.2.A.
B. Core and Containment Cooling Systems Initiation and Control	B. Core and Containment Cooling Systems Initiation & Control
The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2.B. This instrumentation must be operable when the system(s) it initiates or controls are required to be operable as specified in Section 3.5.	Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.B. System logic shall be functionally tested as indicated in Table 4.2.B.
C. Control Rod Block Actuation	C. Control Rod Block Actuation
The limiting conditions for operation for the instrumentation that initiates control rod blocks are given in Table 3.2.C.	Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.C. System logic shall be functionally tested as indicated in Table 4.2.C.

LIMITING CONDITION FOR OPERATION

3.2 (cont'd.)

D. Radiation Monitoring Systems - Isolation & Initiation Functions

1. Steam Jet Air Ejector Off-Gas System

a. Except as specified in Specification 2.4.3.a.7 of Appendix B, both steam jet air ejector off-gas system radiation monitors shall be operable.

b. The time delay setting for closure of the steam jet air ejector isolation valves shall not exceed 15 minutes.

c. Other limiting conditions for operation are given on Table 3.2.D and Sections 2.4.3.a.6.b and 2.4.3.a.7 of the Environmental Technical Specifications.

2. Reactor Building Isolation and Standby Gas Treatment Initiation

The limiting conditions for operation are given on Table 3.2.D and Section 2.4.3.a of Appendix B.

3. Liquid Radwaste Discharge Isolation

The limiting conditions for operation are given on Table 3.2.D and Section 2.4.1.b.3 of Appendix B.

4. Main Control Room Ventilation Isolation

The limiting conditions for operation are given on Table 3.2.D and Section 3.12 entitled "Additional Safety Related Plant Capabilities."

SURVEILLANCE REQUIREMENT

4.2 (cont'd)

D. Radiation Monitoring Systems - Isolation & Initiation Functions

1. Steam Jet Air Ejector Off-Gas System

Instrumentation surveillance requirements are given on Table 4.2.D.

2. Reactor Building Isolation and Standby Gas Treatment Initiation

Instrumentation surveillance requirements are given on Table 4.2.D.

3. Liquid Radwaste Discharge Isolation

Instrumentation surveillance requirements are given on Table 4.2.D and Section 3.4.1.b.7 of the Environmental Technical Specifications.

4. Main Control Room Ventilation Isolation

The instrument surveillance requirements are given on Table 4.2.D.

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TABLE 3.2.B (PAGE 5)
HPCI SYSTEM CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability Is Not Assured
Suppression Chamber High Water Level	HPCI-LS-91 A & B	2½" H_2O (5" Above Normal)	1(2)	A
HPCI Gland Seal Cond. Hotwell Level	HPCI-LS-356 B HPCI-LS-356 A	$\geq$18" $\leq$46"	1(3) 1(3)	A A
HPCI Turbine Stop Valve Monitor	HPCI-LMS-4	N.A.	1(2)	B
Suppression Chamber HPCI Suction Valve 23-58	HPCI-LMS-2	N.A.	1(2)	A
HPCI Control Oil Pressure Low	HPCI-PS-2787-H HPCI-PS-2787-L	$\geq$85 psig $\geq$20 psig	1(2)	B
Turbine Conditional Supervisory Alarm Actuation Timer	HPCI-TDR-K14	$13.5 \leq T \leq 16.5$ sec.	1(3)	E
Pump Discharge Line Low Pressure	CM-PS-268	$\geq$10 psig	(3)	D
HPCI Steamline High ΔP Actuation Timer	HPCI-TDR-K33 HPCI-TDR-K43	$2.7 \leq T \leq 3.3$ sec.	1	A

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TABLE 3.2.B (PAGE 6)
REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability Is Not Assured
RCIC High Turbine Exhaust Press.	RCIC-PS-72, A & B	$\leq$25 psig	1(2)	A
RCIC Low Pump Suction Press.	RCIC-PS-67-1	$\leq$-15" Hg	1(2)	
RCIC Steam Line Space Excess Temp.	RCIC-TS-79, A,B,C,&D RCIC-TS-80, A,B,C,&D RCIC-TS-81, A,B,C,&D RCIC-TS-82, A,B,C,&D	$\leq 200^{\circ}F$	2(4)	A
RCIC Steam Line High ΔP	RCIC-dPIS-83 & 84	370" $\leq S \leq$ 620" H_2O	1	A
RCIC Steam Supply Press. Low	RCIC-PS-87, A,B,C,&D	$\geq$50 psig	2(2)	A
RCIC Low Pump Disch. Flow	RCIC-FIS-57	$\geq$40 gpm	1(2)	A
Pump Discharge Line Low Pressure	CM-PS-269	$\geq$10 psig	(3)	D
RCIC Turbine Condition-al Supervisory Alarm Timer	RCIC-TDR-K9	$13.5 \leq T \leq 16.5$	(3)	E
Reactor Low Water Level	10A-K80, A & B 10A-K79, A & B (NBI-LIS-72, A,B,C, & D)	$\geq$-37" Indicated Level	2(2)	A
Reactor High Water Level	NBI-LIS-101, A & C #2	$\leq$+58.5 Indicated Level	2(2)	A
RCIC Steamline High ΔP Actuation Timer	RCIC-TDR-K12 RCIC-TDR-K32	$2.7 \leq T \leq 3.3$ sec	1	A

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TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System(5)
APRM Upscale (Flow Bias)	≤ (0.66W + 42%) $\frac{FRP}{MFLPD}$ (2)	2(1)
APRM Upscale (Startup)	≤ 12%	2(1)
APRM Downscale (9)	≥ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	≤ (0.66W + 40%) (2)	1
RBM Downscale (9)	≥ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	≤ 108/125 of Full Scale	3(1)
IRM Downscale (3)(8)	≥ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	≤ 1×10^5 Counts/Second	1(1)(6)
SRM Detector Not Full In (4)(8)	(≥ 100 cps)	1(1)(6)
SRM Inoperative (8)	(10a)	1(1)(6)
Flow Bias Comparator	≤ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	≤ 110% Recirc. Flow	1
SRM Downscale (8)(7)	≥ 3 Counts/Second (11)	1(1)(6)
SDV Water Level High	≤ 18 gallons	1(12)

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TABLE 3.2.D

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

System	Instrument I. D. No.	Setting Limit	Number of Sensor Channels Provided by Design	Action (1)
Steam Jet Air Ejector Off-Gas System	RMP-RM-150 A & B	$\leq$ 1 ci/sec	2	A
Reactor Building Isolation and Standby Gas Treatment Initiation	RMP-RM-452 A & B	$\leq$ 100 mr/hr	2	B
Liquid Radwaste Discharge Isolation	RMV-RM-2	(2)	1	C
Main Control Room Ventilation Isolation	(RMV-RM-1)	4×10^3 CPM	1	D
Mechanical Vacuum Pump Isolation	RMP-RM-251 A-D	3 times normal full power background. Alarm at 1.5 times normal full power background	4	E

NOTES FOR TABLE 3.2.D

1. Action required when component operability is not assured.

   A. (1) If radiation level exceeds 1.0 ci/sec (prior to 30 min. delay line) for a period greater than 15 consecutive minutes, the off-gas isolation valve shall close and reactor shutdown shall be initiated immediately and the reactor placed in a cold shutdown condition within 24 hours.

   A. (2) Refer to Section 2.4.3.a.7 of the Environmental Technical Specifications.

   B. Cease refueling operations, isolate secondary containment and start SBGT.

   C. Refer to Sections 2.4.1.b of the Environmental Technical Specifications

   D. Refer to Section entitled "Additional Safety Related Plant Capabilities."

   E. Refer to Section 3.2.d.5 and the requirements for Primary Containment Isolation on high main steam line radiation. Table 3.2.A

2. Trip setting to correspond to Specification 2.4.1.b.1 of the Environmental Technical Specifications.

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TABLE 4.2.B (Page 4)
HPCI TEST & CALIBRATION FREQUENCIES

	Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
1.	Reactor Low Water Level	NBI-LIS-72, A,B,C, & D, #3	Once/Month (1)	Once/3 Months	Once/Day
2.	Reactor High Water Level	NBI-LIS-101, (B & D #3)	Once/Month (1)	Once/3 Months	Once/Day
3.	High Drywell Pressure	14A - K5 A & B	(7)	(7)	None
		14A - K6 A & B	(7)	(7)	None
4.	HPCI Turbine High Exhaust Press.	HPCI-PS-97 A & B	Once/Month (1)	Once/3 Months	None
5.	HPCI Pump Low Suction Press.	HPCI-PS-84-1	Once/Month (1)	Once/3 Months	None
6.	HPCI Pump Low Discharge Flow	HPCI-FS-78	Once/Month (1)	Once/3 Months	None
7.	HPCI Low Steam Supply Press.	HPCI-PS-68, A,B,C, & D	Once/Month (1)	Once/3 Months	None
8.	HPCI Steam Line High ΔP	HPCI-dPIS-76	Once/Month (1)	Once/3 Months	None
		HPCI-dPIS-77	Once/Month (1)	Once/3 Months	None
9.	HPCI Steam Line Space High Temp.	HPCI-TS-101, A,B,C, & D 102, 103, 104, HPCI-TS-125, 126, 127, 128 RHR-TS-150,151,152,153,154, 155,156,157,158,159,160,161	Once/Month (1)	Once/Oper. Cycle	None
10.	Emergency Cond. Stg. Tk. Low Level	HPCI-LS-74 A & B	Once/Month (1)	Once/3 Months	None
		HPCI-LS-75 A & B	Once/Month (1)	Once/3 Months	None
11.	Suppression Chamber High Water Level	HPCI-LS-91 A & B	Once/Month (1)	Once/3 Months	None
12.	HPCI Gland Seal Cond. Hotwell Level	HPCI-LS-356 B	Once/Month (1)	Once/3 Months	None
		HPCI-LS-356 A	Once/Month (1)	Once/3 Months	None
13.	HPCI Control Oil Pressure Low	HPCI-PS-2787-H	Once/Month (1)	Once/3 Months	None
		HPCI-PS-2787-L	Once/Month (1)	Once/3 Months	None
14.	Turbine Condition Supr. Alarm Actuation Timer	HPCI-TDR-K14	Once/Month (1)	Once/Oper. Cycle	None
15.	Pump Disch. Line Low Press.	CM-PS-268	Once/3 Months	Once/3 Months	None
16.	HPCI Turbine Stop Valve Mon.	HPCI-LMS-4	Once/Month	N.A.	None
17.	Sup. Chamber HPCI Suction Vlv.	HPCI-LMS-2	Once/Month	N.A.	None
18.	HPCI Steam Line High ΔP Actuation Timer	HPCI-TDR-K33,	Once/Month	Once/Oper. Cycle	None
		HPCI-TDR-K43	Once/Month	Once/Oper. Cycle	None

Logic (4)(6)

	Item	Functional Test Freq.	Calibration Freq.
1.	Logic Bus Power Monitor	Once/6 Months	N.A.
2.	HPCI Initiation	Once/6 Months	N.A.
3.	HPCI Turbine Trip	Once/6 Months	N.A.

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TABLE 4.2.B (Page 6)
RCIC TEST & CALIBRATION FREQUENCIES

Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Instrument Channels				
1. Reactor High Water Level	NBI-LIS-101 A & C, #2	Once/Month (1)	Once/3 Months	Once/Day
2. Reactor Low Water Level	10A - K79 A & B 10A-K80 A & B	Once/Month (1)	Once/3 Months	Once/Day
3. RCIC High Turbine Exhaust Press.	RCIC-PS-72, A & B	Once/Month (1)	Once/3 Months	None
4. RCIC Low Pump Suction Press.	RCIC-PS-67-1	Once/Month (1)	Once/3 Months	None
5. RCIC Steam Line Space Excess Temp.	RCIC-TS-79, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-80, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-81, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-82, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
6. RCIC Steam Line High ΔP	RCIC-dPIS-83	Once/Month (1)	Once/3 Months	None
	RCIC-dPIS-84	Once/Month (1)	Once/3 Months	None
7. RCIC Steam Supply Press. Low	RCIC-PS-87, A,B,C, & D	Once/Month (1)	Once/3 Months	None
8. RCIC Low Pump Disch. Flow	RCIC-FIS-57	Once/Month (1)	Once/3 Months	None
9. Pump Disch. Line Low Pressure	CM-PS-269	Once/3 Months	Once/3 Months	None
10. RCIC Turbine Conditional Supv. Alarm Timer	RCIC-TDR - K9	Once/Month (1)	Once/Oper. Cycle	None
11. RCIC Steam Line High ΔP Actuation Timer	RCIC-TDR-K-12	Once/Month	Once/Oper. Cycle	None
	RCIC-TDR-K-32	Once/Month	Once/Oper. Cycle	None
Logic Systems (4)(6)				
1. Logic Buss Power Monitor		Once/6 Months	N.A.	
2. RCIC Initiation		Once/6 Months	N.A.	
3. Turbine Trip		Once/6 Months	N.A.	
4. RCIC Automatic Isolation		Once/6 Months	N.A.	

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TABLE 4.2.C

SURVEILLANCE REQUIREMENTS FOR ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Functional Test Freq.	Calibration Freq.	Instrument Check
APRM Upscale (Flow Bias)	(1) (3)	Once/3 Months	Once/Day
APRM Upscale (Startup Mode)	(1) (3)	Once/3 Months	Once/Day
APRM Downscale	(1) (3)	Once/3 Months	Once/Day
APRM Inoperative	(1) (3)	N.A.	Once/Day
RBM Upscale (Flow Bias)	(1) (3)	Once/6 Months	Once/Day
RBM Downscale	(1) (3)	Once/6 Months	Once/Day
RBM Inoperative	(1) (3)	N.A.	Once/Day
IRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Detector Not Full In	(2) (Once/operating cycle)	Once/Oper. Cycle (10)	Once/Day
IRM Inoperative	(1) (2) (3)	N.A.	N.A.
SRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Detector Not Full In	(2) (Once/operating cycle)	Once/Oper. Cycle (10)	N.A.
SRM Inoperative	(1) (2) (3)	N.A.	N.A.
Flow Bias Comparator	(1) (8)	Once/Oper. Cycle	N.A.
Flow Bias Upscale	(1) (8)	Once/3 Months	N.A.
Rod Block Logic	(9)	N.A.	N.A.
RSCS Bypass	(1)	Once/3 Months	N.A.
SDV High Water Level	Quarterly	Once/Oper. Cycle	N.A.

3.2 BASES (Cont'd)

Trip settings of <100 mr/hr for the monitors in the ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

Flow transmitters are used to record the flow of liquid from the drywell sumps. An air sampling system is also provided to detect leakage inside the primary containment.

For each parameter monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between the two (2) channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

The recirculation pump trip has been added as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events given in General Electric Company Topical Report, NEDO-10349, dated March, 1971.

The liquid radwaste monitor assures that all liquid discharged to the discharge canal does not exceed the limits of Section 2.4.1.b of Environmental Technical Specifications. Upon sensing a high discharge level, an isolation signal is generated which closes the radwaste discharge valve. The set point is adjustable to compensate for variable isotopic discharges and dilution flow rates.

The main control room ventilation isolation is provided by a detector monitoring the intake of the control room ventilation system. Automatic isolation of the normal supply and exhaust and the activation of the emergency filter system is provided by the radiation detector trip function at the predetermined trip level.

The mechanical vacuum pump isolation prevents the exhausting of radioactive gas thru the 1 minute holdup line upon receipt of a main steam line high radiation signal.

The operability of the reactor water level instrumentation in Tables 3/4.2.F ensures that sufficient information is available to monitor and assess accident situations.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3 REACTIVITY CONTROL	4.3 REACTIVITY CONTROL
Applicability:	Applicability:
Applies to the operational status of the control rod system.	Applies to the surveillance requirements to the control rod system.
Objective:	Objective:
To assure the ability of the control rod system to control reactivity.	To verify the ability of the control rod system to control reactivity.
Specification:	Specification:
A. Reactivity Limitations	A. Reactivity Limitations
1. Reactivity margin - core loading	1. Reactivity margin - core loading
A sufficient number of control rods shall be operable so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted.	Sufficient control rods shall be withdrawn following a refueling outage when core alternations were performed to demonstrate, with a margin of 0.38% $\Delta k/k$, that the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest operable control rod fully withdrawn and all other operable rods fully inserted.
2. Reactivity margin - inoperable control rods	2. Reactivity margin inoperable control rods
a. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing. b. The control rod directional control valve for inoperable control rods shall be disarmed electrically. c. Control rods with scram times greater than those permitted by	a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week, when operating above 30% power. This test shall be performed at least once per 24 hours when operating above 30% power in the event power operation is continuing with three or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than three and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod. b. A second licensed operator shall verify the conformance to Specification 3.3.A.2.d before a rod may be bypassed in the Rod Sequence Control System. c. Once per week, check the status of the pressure and level alarms for each accumulator.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.B.3 (cont'd)

e. If Specifications 3.3.B.3a through d cannot be met, the reactor shall not be started, or if the reactor is in the run or startup modes at less than 20% rated power, it shall be brought to a shutdown condition immediately.

f. The sequence restraints imposed on the control rods may be removed by the use of the individual rod position bypass switches for scram testing only those rods which are fully withdrawn in the 100% to 50% rod density range.

4. Control rods shall not be withdrawn for startup unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:

a. Both RBM channels shall be operable: or

b. Control rod withdrawal shall be blocked: or

c. The operating power level shall be limited so that the MCPR will remain above the safety limit assuming a single error that results in complete withdrawal of any single operable control rod.

4.3.B.3.b (cont'd)

1) The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.

2) The RWM computer on line diagnostic test shall be sucessfully performed.

3) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

4) The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.

c. When required, the presence of a second licensed operator or other qualified employee to verify the following of the correct rod program shall be verified.

4. Prior to control rod withdrawal for startup, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).

3.3 (cont'd)

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.50

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.71

4.3 (cont'd)

C. Scram Insertion Times

1. After each refueling outage all operable rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 800 psig and the requirements of Specification 3.3.B.3.a met. This testing shall be completed prior to exceeding 40% power. Below 20% power, only rods in those sequences (A_{12} and A_{34} or B_{12} and B_{34}) which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram time tested. During all scram time testing below 20% power, the Rod Worth Minimizer shall be operable or a second licensed operator or other qualified employee shall verify that the operator at the reactor console is following the control rod program.

2. At 16-week intervals, 10% of the operable control rod drives shall be scram timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.3.C (Cont'd.)

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

D. Reactivity Anomalies

At a specific steady state base condition of the reactor actual control rod inventory will be periodically compared to a normalized computer prediction of the inventory. If the difference between observed and predicted rod inventory reaches the equivalent of 1% Δk reactivity, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

E. Recirculation Pumps

A recirculation pump shall not be started while the reactor is in natural circulation flow and reactor power is greater than 1% of rated thermal power.

F. If Specifications 3.3.A through D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Shutdown condition within 24 hours.

4.3.C (Cont'd.)

D. Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

G. Scram Discharge Volume

1. The scram discharge volume (SDV) vent and drain valves shall be cycled and verified open at least once every 31 days and prior to reactor start-up.

2. The SDV vent and drain valves shall be verified to close within 30 seconds after receipt of a signal for control rod scram once per refueling cycle.

3. SDV vent and drain valve operability shall be verified following any maintenance or modification to any portion (electrical or mechanical) of the SDV which may affect the operation of the vent and drain valves.

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3.3 and 4.3 BASES

A. Reactivity Limitation

1. The requirements for the control rod drive system have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in subsection III.4 of the Final Safety Analysis Report, the control rod system design is intended to provide sufficient control of core reactivity that the core could be made subcritical with the strongest rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance. Compliance with this requirement can be demonstrated conveniently only at the time of initial fuel loading or refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration shall be performed with the reactor core in the cold, xenon-free condition and will show that the reactor is subcritical by at least R + 0.38% $\Delta k/k$ with the analytically determined strongest control rod fully withdrawn.

   The value of "R", in units of %$\Delta k/k$, is the amount by which the core reactivity, in the most reactive condition at any time in the subsequent operating cycle, is calculated to be greater than at the time of the demonstration. "R", therefore, is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of "R" must be positive or zero and must be determined for each fuel cycle.

   The demonstration is performed with a control rod which is calculated to be the strongest rod. In determining this "analytically strongest" rod, it is assumed that every fuel assembly of the same type has identical material properties. In the actual core, however, the control cell material properties vary within allowed manufacturing tolerances, and the strongest rod is determined by a combination of the control cell geometry and local k_∞. Therefore, an additional margin is included in the shutdown margin test to account for the fact that the rod used for the demonstration (the "analytically strongest") is not necessarily the strongest rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38% $\Delta k/k$. When this additional margin is demonstrated, it assures that the reactivity control requirement is met.

2. Reactivity margin - inoperable control rods.

   Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and then disarmed electrically, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shutdown at all times with the remaining control rods assuming the strongest operable control rod does not insert. An allowable pattern for control rods valved out of service, which shall meet this Specification, will be determined and made available to the operator.

In order to perform shutdown margin and control rod drive scram time tests subsequent to any fuel loading operation as required by the Technical Specifications, the relaxation of the following Rod Sequence Control System restraints is required: (a) The sequence restraints imposed on the control rods may be removed by the use of the individual rod position bypass switches for scram testing only those rods which are fully withdrawn in the 100% to 50% rod density range. (b) Verify that subsequent to the use of the rod position bypass switches rod movement in the 50% rod density to preset power level range is restricted to the single notch mode.

If damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

B. Control Rod

1. Control rod drop accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the rod sequence control system and the rod worth minimizer (RWM).

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in subsection III.8.2 of the FSAR and the safety evaluation is given in subsection VIII.8.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to prespecified sequences. These sequences are established such that the drop of any in-sequence control rod or control rod segment (i.e., one or more notches) would not cause the reactor to sustain a power excursion resulting in a peak fuel enthalpy in excess of 280 cal./gm. An enthalpy of 280 cal./gm. is well below the level at which rapid fuel dispersal could occur (i.e., 425 cal./gm.). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Subsections III.6.6, VIII 7.4.5, and XIV.6.2 of the FSAR and Reference 1.

   In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 20% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 20%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

   At power levels below 20% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calories per gram rod drop limit. In this range the RWM and the RSCS constrain the control rod sequences and patterns to those which involve only acceptable rod worths.

   The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. They serve as a backup to procedural control on control rod sequences, which limit the maximum reactivity worth of control rods. In the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator or other qualified technical plant employee whose qualifications have been reviewed by the NRC can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural control to assure conformance.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20%, these devices force adherence to acceptable rod patterns. Above 20% of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20% of rated power are imposed by power distribution requirements as defined in Section 3.3.B.5 of these Technical Specifications. Power level for automatic cutout of the RSCS function is sensed by first stage turbine pressure. Because the instrument has an instrument error of $\pm$ 2% of full power, the nominal instrument setting is 22% of rated power. Power level for automatic cutout of the RWM function is sensed by feedwater and steam flow and is set nominally at 30% of rated power to be consistent with the RSCS setting.

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 20% rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

The RSCS can be functionally tested prior to control rod withdrawal for reactor startup. By selecting, for example, A_{12} and attempting to withdraw, by one notch, a rod or all rods in each other group, it can be determined that the A_{12} group is exclusive. By bypassing to full-out all A_{12} rods, selecting A_{34} and attempting to withdraw, by one notch, a rod or all rods in group B, the A_{34} group is determined exclusive. The same procedure can be repeated for the B groups. After 50% of the control rods have been withdrawn (e.g., groups A_{12} and A_{34}), it is demonstrated that the Group Notch made for the control drives is enforced. This demonstration is made by performing the hardware functional test sequence. The Group Notch restraints are automatically removed above 20% power.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCS occurs and subsequently at appropriate stages of the control rod insertion.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirements of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8}% of rated power used in the analyses of transients cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

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5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

   A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR = 1.07, and LHGR = as defined in 1.0.A.4). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the station superintendent.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the safety limit. The limiting power transient is defined in Reference 3. Analysis of this transient shows that the negative reactivity rates resulting from the scram provide the required protection, and MCPR remains greater than the safety limit.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10% of the control rods at 16-week intervals is adequate for determining the operability of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Cooper Nuclear Station.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives.

In the analytical treatment of the transients which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from the sensor and circuit delays; at this point, the pilot scram solenoid deenergizes. Approximately 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds is conservatively assumed for this time interval in the transient analyses and this is also included in the allowable scram insertion times of Specification 3.3.C. The time to deenergize the pilot valve scram solenoid is measured during the calibration tests required by Specification 4.1.

D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% Δk. Deviations in core reactivity greater than 1% Δk are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

E. Recirculation Pumps

Until analyses are submitted for review and approval by the NRC which prove that recirculation pump startup from natural circulation does not cause a reactivity insertion transient in excess of the most severe coolant flow increase currently analyzed, Specification 3.3.E prevents starting recirculation pumps while the reactor is in natural circulation above 1% of rated thermal power.

G. Scram Discharge Volume

To ensure the Scram Discharge Volume (SDV) does not fill with water, the vent and drain valves shall be verified open at least once every 31 days. This is to preclude establishing a water inventory, which if sufficiently large, could result in slow scram times or only a partial control rod insertion.

The vent and drain valves shut on a scram signal thus providing a contained volume (SDV) capable of receiving the full volume of water discharged by the control rod drives at any reactor vessel pressure. Following a scram the SDV is discharged into the reactor building drain system.

REFERENCES

1. Licensing Topical Report GE-BWR Generic Reload Fuel Application, NEDE-24011-P, (most current approved submittal).

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (applicable reload document).

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3.4

B. Operation with Inoperable Components:

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

C. Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be operable the following conditions shall be met:

1. The net volume versus concentration of the Liquid Control Solution in the liquid control tank shall be maintained as required in Figure 3.4.1.

2. The temperature of the liquid control solution shall be maintained above the curve shown in Figure 3.4.2.

4.4.A.2.c (Cont'd.)

pump demineralized water into the reactor vessel from the test tank.

These tests check the actuation of the explosive charge of the tested loop, proper operation of the valves, and pump operability. The replacement charges to be installed will be selected from the same manufactured batch as the tested charge.

d. Both systems, including both explosive valves, shall be tested in the course of two operating cycles.

B. Surveillance with Inoperable Components:

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

C. Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check and record at least once per day.

2. Temperature: Check and record at least once per day.

3. Concentration: Check and record at least once per month. Also check concentration anytime water or boron is

LIMITING CONDITIONS FOR OPERATION

3.5.A (cont'd.)

2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days provided that during such seven days all active components of the other core spray subsystem and active components of the LPCI subsystem and the diesel generators are operable.

3. Both LPCI subsystems shall be operable:

   (1) prior to reactor startup from a Cold Condition, except as specified in 3.5.F.7, or

   (2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in 3.5.A.4 and 3.5.A.5 below.

4. From and after the date that one of the RHR (LPCI) pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding thirty days provided that during such thirty days the remaining active components of the LPCI subsystem and all active components of both core spray subsystems and the diesel generators are operable.

SURVEILLANCE REQUIREMENTS

4.5.A (cont'd.)

2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem, the LPCI subsystem and the diesel generators shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.

3. LPCI subsystem testing shall be as follows:

	Item	Frequency
a.	Simulated Automatic Actuation Test	Once/Operating Cycle
b.	Pump Operability	Once/month
c.	Motor Operated Valve Operability	Once/month
d.	Pump Flow Rate	Once/3 months

During single pump LPCI, each RHR pump shall deliver at least 7700 GPM but no more than 8400 GPM against a system head equivalent to a reactor vessel pressure of 20 psid above drywell pressure with water level below the jet pumps. At the same conditions, two pump LPCI flow shall be at least 15,000 GPM.

e. Recirculation pump discharge valves shall be tested each refueling outage to verify full open to full closed in $20 \leq t \leq 26$ seconds.

4. When it is determined that one of the RHR (LPCI) pumps is inoperable at a time when it is required to be operating the remaining active components of the LPCI subsystems, the containment cooling subsystem, both core spray systems and the diesel generators shall be demonstrated to be operable immediately and the operable LPCI pumps daily thereafter.

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3.5.F (cont'd)

h. A special flange, capable of sealing a leaking control rod housing, is available for immediate use.

i. The control rod housing is covered with the special flange following the removal of the control rod drive.

j. No work is being performed in the vessel while the housing is open.

6. During a refueling outage, refueling operation may continue with one core spray system or the LPCI system inoperable for a period of thirty days.

7. The LPCI System is required to be operable while performing training startups at atmospheric pressure at power levels less than 1% of rated thermal power with the exception that the RHR system may be aligned in the shutdown cooling mode rather than the LPCI mode.

G. Maintenance of Filled Discharge Pipe

Whenever core spray subsystems, LPCI subsystem, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

4.5.F (cont'd)

G. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to, to assure that the discharge piping of the core spray subsystems, LPCI subsystem, HPCI and RCIC are filled:

1. Whenever the Core Spray, LPCI, HPCI or RCIC systems are made operable, the discharge piping shall be vented from the high point of the system and water flow observed initially and on a monthly basis.

2. The pressure switches which monitor the LPCI, core spray, HPCI and RCIC lines to ensure they are full shall be functionally tested and calibrated every three months.

3.5 BASES

A. Core Spray and LPCI Subsystems

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum required cooling systems. During reactor shutdown when the residual heat removal system is realigned from LPCI to the shutdown cooling mode, the LPCI System is considered operable.

The core spray system is designed to provide emergency cooling to the core by spraying in the event of a loss-of-coolant accident. This system functions in combination with the LPCI system to prevent excessive fuel clad temperature.

The LPCI subsystem is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI subsystem and the core spray subsystem provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high pressure emergency core cooling subsystems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in reference (1). Using the results developed in this reference, the repair period is found to be 1/2 the test interval. This assumes that the

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(1) Jacobs, I.M., "Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards", General Electric Co. A.P.E.D., April, 1969 (APED 5736).

core spray subsystems and LPCI constitute a 1 out of 3 system; however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 is 1 month. Should a subsystem fail, a daily test is called for on the remaining systems to ensure that they will function.

Should one core spray subsystem become inoperable, the remaining core spray and the LPCI system are available should the need for core cooling arise. To assure that the remaining core spray and LPCI subsystems and the diesel generators are available, they are demonstrated to be operable immediately. This demonstration includes a manual initiation of the pumps and associated valves and diesel generators.

Should the loss of one LPCI pump occur, a nearly full complement of core and containment cooling equipment is available. Three LPCI pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, which will be demonstrated to be operable, a thirty day repair period is justified. If the LPCI subsystem is not available, at least 1 LPCI pump must be available to fulfill the containment cooling function. The 7 day repair period is set on this basis.

B. Containment Cooling Subsystem

The containment cooling subsystem for CNS consists of two loops each with 2 RHR (LPCI) pumps serving one side of the RHR heat exchanger and two RHR Service Water Booster Pumps serving the other side. The design of the loops is predicted upon the use of one RHR Service Water Booster Pump and one RHR heat exchanger, for heat removal after a design basis accident. Thus, there are ample spares for margin above design conditions. Loss of margin should be avoided and the equipment maintained in a state of operation. So a 30 day out-of-service time is chosen for this equipment. If one loop is out-of-service reactor operation is permissible for seven days with daily testing of the operable loop after testing the appropriate diesel generator.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventive maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity, the other pumps of this type might be subjected to a capacity test. In any event, surveillance procedures, as required by Section 6 of these specifications, detail the required extent of testing.

The pump capacity test is a comparison of measured pump performance parameters to shop performance tests. Tests during normal operation will be performed by measuring the flow and/or the pump discharge pressure. These parameters and its power requirement will be used to establish flow at that pressure.

C. HPCI

The limiting conditions for operating the HPCI System are derived from the Station Nuclear Safety Operational Analysis (Appendix G) and a detailed functional analysis of the HPCI System (Section VI.).

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss-of-coolant which does not result in rapid depressurization of the reactor vessel. The HPCIS permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray System operation maintains core cooling.

The capacity of the system is selected to provide this required core cooling. The HPCI pump is designed to pump 4250 gpm at reactor pressures between 1120 and 150 psig. Two sources of water are available. Initially, demineralized water from the emergency condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI System. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reaches equilibrium with the flow through the break. Continued depressurization causes the break flow to decrease below the HPCI flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

The analysis in the FSAR, Appendix G, shows that the ADS provides a single failure proof path for depressurization for postulated transients and accidents. The RCIC serves as an alternate to the HPCI only for decay heat removal when feed water is lost. Considering the HPCI and the ADS plus RCIC as redundant paths, reference (1) methods would give an estimated allowable repair time of 15 days based on the one month testing frequency. However, a maximum allowable repair time of 7 days is selected for conservatism. The HPCI and RCIC as well as all other Core Standby Cooling Systems must be operable when starting up from a Cold Condition. It is realized that the HPCI is not designed to operate until reactor pressure exceeds 150 psig and is automatically isolated before the reactor pressure decreases below 100 psig. It is the intent of this speci-

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LIMITING CONDITIONS FOR OPERATION

3.6 Primary System Boundary

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

2. During operation where the core is critical or during heatup by non-nuclear means, the reactor vessel metal and fluid temperatures shall be at or above the temperatures shown on the limiting curves of Figures 3.6.1.a or 3.6.1.b where the curve for the beltline is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

3. The reactor vessel metal temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on the limiting curves of Figure 3.6.2 where the curve for the beltline is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

SURVEILLANCE REQUIREMENTS

4.6 Primary System Boundary

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the following temperatures shall be permanently logged at least every 15 minutes until the difference between any two readings taken over a 45 minute period is less than 50°F.

a. Bottom head drain.

b. Recirculation loops A and B.

2. Reactor vessel temperature and reactor coolant pressure shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220°F and the reactor vessel is not vented.

3. Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 Mev neutrons shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to ASTM E 185-73 to the degree possible.

Selected neutron flux specimens shall be removed during the first refueling

3.6.A (cont'd.)

4. The Reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 80°F.

5. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

6. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

4.6.A (cont'd.)

outage and tested to experimentally verify or adjust the calculated values of integrated neutron flux that are used to determine the RT NDTT for Figure 3.6.1.

If the adjusted reference temperature established in accordance with App. G of Section III of the Code 1972 Summer Addendum does not exceed 100°F over the life of the vessel, the withdrawal schedule should be as follows:

First Capsule: 1/4 service life
Second Capsule: 3/4 service life
Third Capsule: Standby

In the event the surveillance specimens exhibit at 1/4 of the vessel's service life, a shift of the charpy V-notch fracture energy curve greater than predicted by test data the remaining withdrawal schedule shall be modified as follows:

Second Capsule: 1/2 service life
Third Capsule: Standby

4. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

5. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

6. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

3.6 (cont'd.)

B. Coolant Chemistry

1. The reactor coolant radioactivity concentration shall be maintained within the following limits:

a. Whenever the reactor is critical, the reactor coolant activity shall not exceed the equilibrium value of 3.1 μCi/gm of dose equivalent I-131.

b. The limit of 3.6.B.1.a above may be exceeded by a factor of 10 or less for a maximum of 48 hours following power transients. The reactor shall not be operated more than 5% of its annual power operation under this exception.

c. If the iodine concentration in the coolant exceeds the equilibrium limit by a factor greater than 10, the reactor shall be shutdown in an orderly manner and in the cold shutdown condition within 24 hours, and the steam line isolation valves shall be closed.

4.6 (cont'd.)

B. Coolant Chemistry

1.a. A sample of reactor coolant shall be collected and analyzed for gross gamma activity as follows:

1. At least every 96 hours whenever the reactor is critical.

2. Prior to reactor startup.

3. In the STARTUP mode, at 4-hour intervals following a power change exceeding 5% of rated power in one hour or less.

4. In the RUN mode, at 4-hour intervals followed a power change exceeding 20% of rated power in one hour or less.

5. At 4-hour intervals following an off-gas activity increase of 10,000 μCi/sec measured at the SJAE.

6. At 4-hour intervals whenever measurements indicate the equilibrium iodine concentration limit of 3.6.B.1 is exceeded, until a stable value below the equilibrium limit is established.

The samples required in 4.6.B.1.a.3, 4, and 5 shall be collected for 48 hours but may be discontinued if the reactor coolant concentration is shown to be less than 1% of the equilibrium value specified in 3.6.B.1 or when a stable iodine concentration below the limiting equilibrium value is established. Whereas a single measurement may be used to show an activity level below 1%, at least 3 consecutive samples with the last 2 yielding activities below the equilibrium value are required to establish a stable concentration below the equilibrium limit.

3.6.B. (cont'd)

2. Prior to startup and during the operating of the reactor up to 10% of rated power, and during hot standby, the reactor coolant shall not exceed the following limits:

a.	Conductivity	< 5 μmho/cm at 25°C
b.	Chloride	0.1 ppm

The reactor shall be shut down if pH is <5.6 or >8.6 for a 24-hour period.

3. During reactor operation in excess of 10% of rated power, the reactor coolant shall not exceed the following limits:

a.	Conductivity	1 μmho/cm at 25°C
b.	Chloride	0.2 ppm

4. During the reactor operation in excess of 10% of rated power, the reactor coolant may exceed the limits of Paragraph 3.6.B.3 only for the time limits specified here. If these time limits or the following maximum limits are exceeded, the reactor shall be shutdown and placed in the Cold Shutdown Condition.

a.	Conductivity	Time above 1 μmho/cm at 25°C, 2 weeks/year Maximum limit-10 μmho/cm at 25°C
b.	Chloride	Time above 0.2 ppm, 2 weeks/year Maximum limit-0.5 ppm

The reactor shall be shut down if pH is <5.6 or >8.6 for a 24-hour period.

5. When the reactor is not pressurized (i.e. at or below 212°F), reactor coolant shall be maintained below the following limits:

a.	Conductivity	10μmho/cm at 25°C
b.	Chloride	0.5 ppm

4.6 (cont'd)

b. If the gross activity counts of a sample indicate an activity concentration above 3.1 μCi/gm of dose equivalent I-131, an isotopic analysis shall be performed and quantitative measurements made to determine the dose equivalent I-131 concentration.

c. An isotopic analysis of a reactor coolant sample shall be made at least once per month.

2. Reactor coolant shall be continuously, monitored for conductivity.

3. Prior to startup, during the operation of the reactor and during hot standby, a sample of the reactor coolant shall be analyzed:

a. At least every 80 hours for conductivity and chloride ion content when the continuous conductivity monitor reading is ≤0.7 μmho/cm 25°C.

b. At least every 24 hours for conductivity and chloride ion content when the continuous conductivity monitor reading is >0.7 but ≤2.0 μmho/cm at 25°C.

c. At least every 8 hours for conductivity and chloride ion content when the continuous conductivity monitor reading is >2 but ≤3.5 μmho/cm at 25°C.

d. At least every 4 hours for conductivity, chloride ion content, and pH, when the continuous conductivity monitor reading is >3.5 μmho/cm at 25°C or when the continuous conductivity monitor is inoperable.

4. When the reactor is not pressurized, a sample of the reactor coolant shall be analyzed at least every 80 hours for conductivity and chloride ion content.

Table 3.6.3

INACCESSIBLE SAFETY RELATED MECHANICAL SHOCK SUPPRESSORS (SNUBBERS)
(Cont'd)

Snubber No.	Location	Elevation
VR-H-61D	Drywell	897'
VR-H-62C	Drywell	899'
VR-H-63B	Drywell	897'
VR-H-63C	Drywell	898'
VR-55-9-Y	Drywell	919'
VR-55-9-Z	Drywell	919'
VR-55-23-X	Drywell	906'
VR-55-23-Y	Drywell	907'
VR-55-26-Z	Drywell	906'
VR-56-12-Y	Drywell	913'
VR-56-26-Y	Drywell	916'
VR-56-24-X	Drywell	907'
VR-56-24-Z	Drywell	910'
VR-58-12-Y	Drywell	924'
VR-59-7-X	Drywell	920'
VR-59-7-Z	Drywell	920'
VR-60-7-X	Drywell	920'
VR-60-7-Z	Drywell	920'
VR-61-8-X	Drywell	919'
VR-61-8-Z	Drywell	919'
VR-61-17-X	Drywell	915'
VR-62-8-X	Drywell	922'
VR-62-8-Z	Drywell	915'
VR-62-17-X	Drywell	915'
VR-S-10	Drywell	919'
VR-S-11	Drywell	917'
VR-S-14	Drywell	896'
VR-S-20	Drywell	925'
VR-S-21	Drywell	922'
VR-S-22	Drywell	915'
VR-S-30	Drywell	927'
VR-S-31	Drywell	926'
VR-S-32	Drywell	897'
VR-S-40	Drywell	924'
VR-S-41	Drywell	925'
VR-S-42A	Drywell	924'
VR-S-42B	Drywell	924'
VR-S-43	Drywell	894'
VR-S-50A	Drywell	925'
VR-S-50B	Drywell	925'
VR-S-51	Drywell	896'
VR-S-60	Drywell	927'
VR-S-61	Drywell	925'
VR-S-62A	Drywell	926'
VR-S-62B	Drywell	928'
VR-S-63	Drywell	921'
VR-S-87A	Drywell	894'
VR-S-87B	Drywell	894'
VR-S-88	Drywell	894'
VR-H-62B	Drywell	899'
VR-H-64D	Drywell	899'

Thermal and Pressurization Limitations

The requirements for the reactor vessel have been identified by evaluating the need for its integrity over the full spectrum of plant conditions and events.

This is accomplished through the Station Nuclear Safety Operational Analysis (Appendix G) and a detailed functional analysis of the reactor vessel. The limits expressed in the technical specification for the applicable operating states are taken from the actual Nuclear Safety Operational Requirements for the reactor vessel as given in Subsection IV-2.8 of the Safety Analysis Report.

The components of the nuclear system pressure boundary are constructed so that its initial maximum nil-ductility transition temperature (RT NDT) is not greater than 40°F, as cited in Subsection IV-2.5 of the Safety Analysis Report. The heatup-cooldown and hydrostatic test minimum pressurization temperatures were calculated to comply with the recommendations of Appendix G of Section III, ASME Boiler and Pressure Vessel Code, 1972 Summer Addendum.

The temperature versus pressure limits when critical which are presented in Figure 3.6.1.b assure compliance with Appendix G of 10CFR50.

Tightening the studs on the reactor vessel head flexes it slightly to bring together the entire contact surfaces adjacent to the O-rings of the head and vessel flange. The reactor vessel head flange and head are constructed so that their initial maximum NDTT is 20°F, as cited in Paragraph IV-2.5 of the Safety Analysis Report. Therefore, the initial minimum temperature at which the studs can be placed in tension is established at 80°F (20°F + 60°F). The total integrated neutron flux in the head flange region will be less than that at the core mid-plane level by a factor of 10^{-3} or 10^{-4}, therefore, the maximum calculated fluence in the head flange region will be far below 1×10^{17} nvt. With such a low total integrated neutron flux in the head flange region, there will be no detectable or significant NDTT shift, and the minimum stud tightening temperature remains at 80°F.

The reactor vessel is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, for a pressure of 1250 psig. The pressure limit of 1035 psig represents the maximum expected operating pressure in the steam dome when the station is operating at design thermal power. Observation of this limit assures that the operator remains within the envelope of conditions considered by the Station Analysis (Section XIV).

Stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. The specific conditions analyzed included a maximum of 120 cycles of normal startup and shutdown with a heating and cooling rate of 100°F per hour applied continuously over a temperature range of 100°F to 546°F. The expected number of normal heatup and cooldown cycles to which the vessel will be subjected is 80.

As described in the safety analysis report, detailed stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The maximum calculated neutron fluence of 1 Mev or greater, based on 100 percent rated power and 100 percent availability for 40 years, is given by Figure 3.6.1. The neutron flux wires are removed and tested after approximately one year of operation during the first refueling outage to experimentally verify the calculated values of integrated neutron flux. The RT/NDT is determined by utilizing the value of the fluence measured at the core mid-plane level. This approach is conservative because the fluence level decreases as the point of measurement is removed from the core mid-plane level. In addition, vessel material samples will be located within the vessel to monitor the effect of neutron exposure on these materials. The samples include specimens of base metal, weld zone metal and heat affected zone metal. These samples will receive neutron exposure more rapidly than the vessel wall material and therefore, will lead the vessel in integrated neutron flux exposure. These samples will provide further assurance that the Shift in RT/NDT used in the specification is conservative.

B. Coolant Chemistry

Materials in the primary system are primarily Type-304 stainless steel and Ziracloy cladding. The reactor water chemistry limits are established to provide an environment favorable to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it can be continuously and reliably measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel.

Several investigations have shown that in neutral solutions some oxygen is required to cause stress corrosion cracking of stainless steel, while in the absence of oxygen no cracking occurs. One of these is the chloride-oxygen relationship of Williams[1], where it is shown that at high chloride concentration little oxygen is required to cause stress corrosion cracking of stainless steel, and at high oxygen concentration little chloride is required to cause cracking. These measurements were determined in a wetting and drying situation using alkaline-phosphate-treated boiler water and therefore, are of limited significance to BWR conditions. They are, however, a qualitative indication of trends.

[1] W. L. Williams, Corrosion 13, 1957, p. 539t.

indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks, associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shutdown to allow further investigation and corrective action.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pumps is 50 gpm and the capacity of the drywell equipment sump pumps is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with margin.

Reactor coolant leakage is also sensed by the containment radiation monitoring unit which senses gross beta, gamma particulate and iodine as well as by oxygen and hydrogen analyzers. Leakage can also be detected by area temperature detectors, humidity detectors and pressure instrumentation. Due to the many and varied ways of detecting primary leakage, a 30 day allowable repair time is justified.

D. Safety and Relief Valves

The safety and relief valves are required to be operable above the pressure (113 psig) at which the core spray system is not designed to deliver full flow. The pressure relief system for Cooper Nuclear Station has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protective criteria of the ASME code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis IV.4.2.1 of subsection IV.4 which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME code requirements is presented in subsection IV.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report presented in question 4.20 of Amendment 11 to the FSAR. Results of the overpressure protection analysis are provided in the current reload license document.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations.

jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle riser system failure.

F. Jet Pump Flow Mismatch

Requiring the discharge valve of the lower speed loop to remain closed until the speed of faster pump is equal to or less than 50% of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

A loss-of-coolant accident analysis occurring during operation with one recirculation loop has not been performed. Therefore, operation with a single loop is prohibited except for a limited interval of 24 hours.

G. Structural Integrity

A preservice inspection of accessible components listed in Table 4.6.1 will be conducted before initial fuel loading to assure the system is free of gross defects and as a reference base for later inspections. Construction orientated nondestructive testing is being conducted as systems are fabricated to assure applicable code requirements are met. Prior to operation, the primary system boundary will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout the life of the station. The inspection program given in Table 4.6.1 is based on the requirements of Section IS-242: Table IS-251, Components, Parts and Methods of Examination, and Table IS-251, Examination Categories, all of Section XI of the 1970 ASME Boiler and Pressure Vessel Code, except where accessibility for inspection was not provided. The initial program was revised to update to the summer 1972 Addendum Table IS-261. Modifications were made to vessel nozzle insulation and nozzle blockout removable shielding designs with the intent to make the inspection areas more accessible by reducing the personnel radiation exposure required for inspection utilizing available equipment.

The inspection program and the modifications described above were developed by the Nebraska Public Power District with assistance from its contractors. The services of General Electric were retained to aid in developing the inspection program, provide advice on practical modifications to existing designs for improved inspectability and to perform the preservice inspection. It is not possible, however, to make all changes that might be desired to insure literal compliance with all areas of the current inspection code. The areas of exclusion and reasons for this exclusion are discussed below.

Category A

Accessibility is not provided for these welds. The permanent standoff type insulation was installed on the vessel and then the concrete sacrificial shield was erected. It was not possible to obtain any base line data on these welds. However, Nebraska Public Power District will evaluate new advances in inspection techniques and will inspect these areas when the equipment and techniques become practicable.

Category B

In addition to the exclusion bases stated for Category A welds, at the present time there is no practical way to volumetrically inspect welds in the bottom head because of the combination of insulation and control rod and incore monitor housings configuration on the outside of the vessel and jet pumps and core shroud on the inside of the vessel.

Category E-(2)

At the present time there is no practical way to volumetrically or visually inspect the bottom head penetrations or drain nozzle weld because of the combination of insulation and control rod and in-core monitor housings configuration. The combination of hydrostatic test and visual checks to be performed to provide reasonable assurance these examination areas are free of gross defects.

Category L-(2)

It is the intent that no internal examination be performed on the recirculation pumps unless they are disassembled for maintenance because of the high personnel radiation exposures which would be involved.

Category M-(2)

There are several valves in the primary pressure boundary which cannot be inspected unless the reactor fuel is removed and reactor water level lowered to the level of the entrance to the jet pump mixer assembly resulting in high personnel radiation exposures from the loss of shielding from the water. Therefore, those valves which would require the reactor water level to be lowered below the low-low water level protection system trip point are excluded from the requirement of visual inspection of internals.

3.7.A (Cont'd)

4.7.A (cont'd)

repeated provided locally measured leakage reductions, achieved by repairs, reduce the containment's overall measured leakage rate sufficiently to meet the acceptance criteria.

f. *With the exception of main steam isolation valves and main steam line and feedwater line bellows, (see below) local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves at a pressure of 58 psig during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years. Bolted double-gasket seals shall be tested after each opening and during each reactor shutdown for refueling, or other convenient intervals but in no case at intervals greater than two years.

* The main steam isolation valves (MSIV's) shall be tested at a pressure of 29 psig. If a total leakage rate of 11.5 scf/hr for any one MSIV is exceeded, repairs and retest shall be performed to correct the condition.

* Main steam line and feedwater line expansion bellows shall be tested at a pressure of 5 psig.

g. Continuous Leak Rate Monitor

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

h. Drywell Surfaces

The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of

* Exemptions to Appendix J of 10 CFR 50.

LIMITING CONDITIONS FOR OPERATION

3.7.A (cont'd.)

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building air actuated vacuum breakers shall be 0.5 psid. The self actuated vacuum breakers shall open fully when subjected to a force equivalent to 0.5 psid acting on the valve disc.

b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, the vacuum breaker switch shall be secured in the closed position and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be operable at the 0.5 psid setpoint and positioned in the fully closed position as indicated by the position indicating system except during testing and except as specified in 3.7.A.4.b and .c below.

b. Three drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening provided they are secured in the fully closed position or that the requirement of 3.7.A.4.c is demonstrated to be met.

SURVEILLANCE REQUIREMENTS

4.7.A (cont'd.)

torus corrosion or leakage.

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentation, including set points shall be checked for proper operation every three months.

b. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specifications 3.7.A.3.a and each vacuum breaker shall be inspected and verified to meet design requirements.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle every 30 days.

b. When it is determined that a vacuum breaker valve is inoperable for opening at a time when operability is required all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

LIMITING CONDITIONS FOR OPERATION

3.7.A (cont'd.)

c. The total leakage between the drywell and suppression chamber shall be less than the equivalent leakage through a 1" diameter orifice.

d. If specifications 3.7.A.4.a, b or c, cannot be met, the situation shall be corrected within 24 hours or the reactor will be placed in a cold shutdown condition within the subsequent 24 hours.

5. <u>Oxygen Concentration</u>

a. After completion of the startup test program and demonstration of plant electrical output, the primay containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown.

c. When the containment atmosphere oxygen concentration is required to be less than 4%, the minimum quantity of liquid nitrogen in the liquid nitrogen storage tank shall be 500 gallons.

d. If the specifications of 3.7.A.5.a thru c cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

e. The specifications of 3.7.A.5.a thru d are not applicable during a 48 hour continuous period between the dates of March 22, 1982 and March 25, 1982.

SURVEILLANCE REQUIREMENTS

4.7.A (cont'd.)

c. Once each operating cycle, each vacuum breaker valve shall be visually inspected to insure proper maintenance and operation of the position indication switch. The differential pressure setpoint shall be verified.

d. Prior to reactor startup after each refueling, a leak test of the drywell to suppression chamber structure shall be conducted to demonstrate that the requirement of 3.7.A.4.c is met.

5. <u>Oxygen Concentration</u>

a. The primary containment oxygen concentration shall be measured and recorded at least twice weekly.

b. The quantity of liquid nitrogen in the liquid nitrogen storage tank shall be determined twice per week when the volume requirements of 3.7.A.5.c are in effect.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7. (cont'd.)	4.7 (cont'd.)
B. Standby Gas Treatment System	B. Standby Gas Treatment System
1. Except as specified in 3.7.B.3 below, both circuits of the standby gas treatment system and the diesel generators required for operation of such circuits shall be operable at all times when secondary containment integrity is required.	1. At least once per operating cycle the following conditions shall be demonstrated. a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at the system design flow rate. b. Inlet heater input is capable of reducing R.H. from 100 to 70% R.H.
2.a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal.	2.a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per year for standby service or after every 720 hours of system operation and following significant painting fire or chemical release in any ventilation zone communicating with the system.
b. The results of laboratory carbon sample analysis shall show $\geq$95% radioactive methyl iodide removal at a velocity within 20 percent of actual system design, 0.5 to 1.5 mg/m^3 inlet methyl iodide concentration, $\geq$70% R.H. and $\geq$190°F.	b. Cold DOP testing shall be performed afte[r] each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
c. Fans shall be shown to operate within $\pm$10% design flow.	c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber ban[k] or after any structural maintenance on the system housing. d. Each circuit shall be operated with the heaters on at least 10 hours every month e. Test sealing of gaskets for housing door[s] downstream of the HEPA filters and charcoal adsorbers shall be performed at, and in conformance with, each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a.
3. From and after the date that one circuit of the standby gas treatment system is made or found to be inoperable for any reason, reactor operation or fuel handling is permissible only during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other standby gas treatment circuit shall be operable.	3. System drains where present shall be inspected quarterly for adequte water leve[l] in loop-seals.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.B (cont'd)	4.7.B (cont'd)
4. If these conditions cannot be met, procedures shall be initiated immediately to establish reactor conditions for which the standby gas treatment system is not required.	4.a. At least once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.
	b. At least once per operating cycle manual operability of the bypass valve for filter cooling shall be demonstrated.
	c. When one circuit of the standby gas treatment system becomes inoperable the other circuit shall be demonstrated to be operable immediately and daily thereafter.
C. <u>Secondary Containment</u>	C. <u>Secondary Containment</u>
1. Secondary containment integrity shall be maintained during all modes of plant operation except when all of the following conditions are met.	1. Secondary containment surveillance shall be performed as indicated below:

3.7.C (cont'd.)

a. The reactor is subcritical and Specification 3.3.A is met.

b. The reactor water temperature is below 212°F and the reactor coolant system is vented.

c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

d. Irradiated fuel is not being handled in the secondary containment.

e. If secondary containment integrity cannot be maintained, restore secondary containment integrity within 4 hours or;

   a. Be in at least Hot Shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

   b. Suspend irradiated fuel handling operations in the secondary containment and all core alterations and activities which could reduce the shutdown margin. The provisions of Specification 1.0.J are not applicable.

D. Primary Containment Isolation Valves

1. During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all instrument line flow check valves shall be operable except as specified in 3.7.D.2.

4.7.C (cont'd.)

a. A preoperational secondary containment capability test shall be conducted after isolating the reactor building and placing either standby gas treatment system filter train in operation. Such tests shall demonstrate the capability to maintain 1/4 inch of water vacuum under calm wind ($2<\bar{\mu}<5$) conditions with a filter train flow rate of not more than 100% of building volume per day. ($\bar{\mu}$= wind speed)

b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

c. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind ($2<\bar{\mu}<5$ mph) conditions with a filter train flow rate of not more than 100% of building volume per day, shall be demonstrated at each refueling outage prior to refueling.

d. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4 inch of water negative pressure under calm wind conditions.

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

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TABLE 3.7.1 (Page 2)
PRIMARY CONTAINMENT ISOLATION VALVES

Valve & Steam	Number of Power Operated Valves		Maximum Operating Time (Sec) (1)	Normal Position (2)	Action On Initiating Signal (3)
	Inboard	Outboard			
Primary Containment Purge & Vent PC-246AV, PC-231MV		2	15	C	SC
Primary Containment & N_2 Supply PC-238AV, PC-232MV		2	15	C	SC
Suppression Chamber Purge & Vent PC-230MV Bypass (PC-305MV)		1	40	C	SC(4)
Primary Containment Purge & Vent PC-231MV Bypass (PC-306MV)		1	40	C	SC(4)
ACAD Supply MV 1303, MV 1304 MV 1305, MV 1306		 2 2	 15 15	 C C	 SC SC
ACAD Supply MV 1301, MV 1302 MV 1311, MV 1312		 2 2	 15 15	 O O	 GC GC
ACAD Suppression Chamber Bleed Isolation MV 1308		1	15	C	SC
ACAD Drywell Chamber Bleed Isolation MV 1310		1	15	C	SC

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TABLE 3.7.4
PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

PEN. NO.	VALVE NUMBERS	TEST MEDIA
X-7A	MS-AO-80A and MS-AO-86A, Main Steam Isolation Valves	Air
X-7B	MS-AO-80B and MS-AO-86B, Main Steam Isolation Valves	Air
X-7C	MS-AO-80C and MS-AO-86C, Main Steam Isolation Valves	Air
X-7D	MS-AO-80D and MS-AO-86D, Main Steam Isolation valves	Air
X-8	MS-MO-74 and MS-MO-77, Main Steam Line Drain	Air
X-9A	RF-15CV and RF-16CV, Feedwater Check Valve	Water
X-9A	RCIC-AO-22, RCIC-MO-17, and RWCU-15CV, RCIC/RWCU Connection to Feedwater	Water
X-9B	RF-13CV and RF-14CV, Feedwater Check Valves	Water
X-9B	HPCI-AO-18 and HPCI-MO-57, HPCI Connection to Feedwater	Water
X-10	RCIC-MO-15 and RCIC-MO-16, RCIC Steam Line	Air
X-11	HPCI-MO-15 and HPCI-MO-16, RPCI Steam Line	Air
X-12	RHR-MO-17 and RHR-MO-18, RHR Suction Cooling	Air
X-13A	RHR-MO-25A and RHR-MO-27A, RHR Supply to RPV	Air
X-13B	RHR-MO-25B and RHR-MO-27B, RHR Supply to RPV	Air
X-14	RWCU-MO-15 and RWCU-MO-18, Inlet to RWCU System	Air
X-16A	CS-MO-11A and CS-MO-12A, Core Spray to RPV	Air
X-16B	CS-MO-11B and CS-MO-12B, Core Spray to RPV	Air
X-17	RHR-MO-32 and RHR-MO-33, RPV Head Spray	Air
X-18	RW-732AV and RW-733AV, Drywell Equipment Sump Discharge	Air
X-19	RW-765AV and RW-766AV, Drywell Floor Drain Sump Discharge	Air
X-25	PC-232MV and PC-238AV, Purge and Vent Supply to Drywell	Air
X-25	ACAD-1305MV and ACAD-1306MV, Supply to Drywell	Air
X-26	PC-231MV and PC-246AV, Purge and Vent Exhaust from Drywell	Air
X-26	ACAD-1310MV, Bleed from Drywell	Air

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TABLE 3.7.4 (page 2)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

PEN. NO.	VALVE NUMBERS	TEST MEDIA
X-39A	RHR-MO-26A and RHR-MO-31A, Drywell Spray Header Supply	Air
X-39B	RHR-MO-26B and RHR-MO-31B, Drywell Spray Header Supply	Air
X-39B	ACAD-1311MV and ACAD-1312MV, Supply to Drywell	Air
X-41	RRV-740AV and RRV-741AV, Reactor Water Sample Line	Air
X-42	SLC-12CV and SLC-13CV, Standby Liquid Control	Air
X-205	PC-233MV and PC-237AV, Purge and Vent Supply to Torus	Air
X-205	PC-13CV and PC-243AV, Torus Vacuum Relief	Air
X-205	PC-14CV and PC-244AV, Torus Vacuum Relief	Air
X-205	ACAD-1303MV and ACAD-1304MV, Supply to Torus	Air
X-210A	RCIC-MO-27 and RCIC-13CV, RCIC Minimum Flow Line	Air
X-210A	RHR-MO-21A, RHR to Torus	Air
X-210A	RHR-MO-16A, RHR-10CV, and RHR-12CV, RHR Minimum Flow Line	Air
X-210B	RHR-MO-21B, RHR to Torus	Air
X-210B	HPCI-17CV and HPCI-MO-25, HPCI Minimum Flow Line	Air
X-210B	RHR-MO-16B, RHR-11CV, and RHR-13CV, RHR Minimum Flow Line	Air
X-210A and 211A	RHR-MO-34A, RHR-MO-38A, and RHR-MO-39A, RHR to Torus	Air
X-210B and 211B	RHR-MO-34B, RHR-MO-38B, and RHR-MO-39B, RHR to Torus	Air
X-211B	ACAD-1301MV and ACAD-1302MV, Supply to Torus	Air
X-212	RCIC-15CV and RCIC-37, RCIC Turbine Exhaust	Air
X-214	HPCI-15CV and HPCI-44, HPCI Turbine Exhaust	Air
X-214	HPCI-AO-70 and HPCI-AO-71, HPCI Turbine Exhaust Drain	Air
X-214	RHR-MO-166A and RHR-MO-167A RHR Heat Exch. Vent	Air
X-214	RHR-MO-166B and RHR-MO-167B RHR Heat Exch. Vent	Air
X-220	PC-230MV and PC-245AV, Purge and Vent Exhaust from Torus	Air
X-220	ACAD-1308MV, Bleed from Torus	Air
X-221	RCIC-12CV and RCIC-42, RCIC Vacuum Line	Air
X-222	HPCI-50 and HPCI-16CV, HPCI Turbine Drain	Air

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The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance.

The 500 gallon conservative limit on the nitrogen storage tank assures that adequate time is available to get the tank refilled assuming normal plant operation. The estimated maximum makeup rate is 1500 SCFD which would require about 160 gallons for a 10 day makeup requirement. The normal leak rate should be about 200 SCFD.

Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain a pressure differential of less than 2 psi, the external design pressure. One valve may be out of service for repairs for a period of 7 days. If repairs cannot be completed within 7 days the reactor coolant system is brought to a condition where vacuum relief is no longer required.

The capacity of the 12 drywell vacuum relief valves are sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to well under the design limit of 2 psi. They are sized on the basis of the Bodega Bay pressure suppression system tests. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows a 2 psi differential; therefore, with three vacuum relief valves secured in the closed position and 9 operable valves, containment integrity is not impaired.

Leak Rate Testing

The maximum allowable test leak rate is 0.635%/day at a pressure of 58 psig, the peak calculated accident pressure. Experience has shown that there is negligible difference between the leakage rates of air at normal temperature and a steam-hot air mixture.

Establishing the test limit of 0.635%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate, La or the allowable test leak rate, Lt by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on the NRC guide for developing leak rate testing and surveillance of reactor containment vessels. Allowing the test intervals to be extended up to 8 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

Table 3.7.4 identifies certain isolation valves that are tested by pressurizing the volume between the inboard and outboard isolation valves. This results in conservative test results since the inboard valve, if a globe valve, will be tested such that the test pressure is tending to lift the globe off its seat. Additionally, the measured leak rate for such a test is conservatively assigned to both of the valves equally and not divided between the two.

The main steam and feedwater testable penetrations consist of a double layered metal bellows. The inboard high pressure side of the bellows is subjected to drywell pressure. Therefore, the bellows is tested in its entirety when the drywell is tested. The bellows layers are tested for the integrity of both layers by pressurizing the void between the layers to 5 psig. Any higher pressure could cause permanent deformation, damage and possible ruptures of the bellows.

The primary containment pre-operational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 58 psig which would rapidly reduce to 29 psig following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay. The design pressure of the drywell and suppression chamber is 56 psig. Based on the calculated containment pressure response discussed above, the primary containment pre-operational test pressure was chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 0.635%/day at 58 psig. Calculations made by the NRC staff with leak rate and a standby gas treatment system filter efficiency of 90% for halogens and assuming the fission product release fractions stated in NRC Regulatory Guide 1.3, show that the maximum total whole body passing cloud dose is about 1.0 REM and the maximum total thyroid dose is about 12 REM at 1100 meters from the stack over an exposure duration of two hours. The resultant doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily

3.7.B & 3.7.C BASES (cont'd)

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 95 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Only one of the two standby gas treatment systems is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is brought to a condition where the standby gas treatment system is not required.

4.7.B & 4.7.C BASES

Standby Gas Treatment System and Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A 7.8 kw heater is capable of maintaining relative humidity below 70%. Heater capacity and pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced

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3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the auxiliary electrical power system.

Objective:

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification:

A. Auxiliary Electrical Equipment

The reactor shall not be made critical from a Cold Shutdown Condition unless all of the following conditions are satisfied:

1. Both off-site sources (345 KV and 69 KV) and the startup transformer and emergency transformer are available and capable of automatically supplying power to the 4160 Volt emergency buses 1F and 1G.

2. Both diesel generators shall be operable and there shall be a minimum of 45,000 gal. of diesel fuel in the fuel oil storage tanks.

3. The 4160V critical buses 1F and 1G and the 480V critical buses 1F and 1G are energized.

   a. The loss of voltage relays and their auxiliary relays are operable.

   b. The undervoltage relays and their auxiliary relays are operable.

4. The four unit 125V/250V batteries and their chargers shall be operable.

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Auxiliary Electrical Equipment

1. Emergency Buses Undervoltage Relays

   a. Loss of voltage relays

      Once every 18 months, loss of voltage on emergency buses is simulated to demonstrate the load shedding from emergency buses and the automatic start of diesel generators.

   b. Undervoltage relays

      Once every 18 months, low voltage on emergency buses is simulated to demonstrate disconnection of the emergency buses from the offsite power source. The undervoltage relays shall be calibrated once every 18 months.

2. Diesel Generators

   a. Each diesel-generator shall be started manually and loaded to not less than 35% of rated load for no less than 2 hours once each month to demonstrate operational readiness.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.A	4.9.A (cont'd.) cell and overall battery voltage shall be measured and logged. b. Every three months the measurements shall be made of the voltage of each cell to nearest 0.1 Volt, specific gravity of each cell, and temperature of every sixth cell. These measurements shall be logged. c. Once each operating cycle, the stated batteries shall be subjected to a rated load discharge test. The specific gravity and voltage of each cell shall be determined after the discharge and logged.
B. Operation with Inoperable Equipment Whenever the reactor is in Run Mode or Startup Mode with the reactor not in a Cold Condition, the availability of electric power shall be as specified in 3.9.A, except as specified in 3.9.B.1. 1. From and after the date incoming power is not available from a startup or emergency transformer, continued reactor operation is permissible under this condition for seven days. At the end of this period, provided the second source of incoming power has not been made immediately available, the NRC must be notified of the event and the plan to restore this second source. During this period, the two diesel generators and associated critical buses must be demonstrated to be operable. 2. From and after the date that incoming power is not available from both start-up and emergency transformers, continued operation is permissible, provided the two diesel generators and associated critical buses are demonstrated to be	

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3.9.B (cont'd.)

operable, all core and containment cooling systems are operable, reactor power level is reduced to 25% of the rated and NRC is notified within 24 hours of the situation, the precautions to be taken during this period and the plans for prompt restoration of incoming power.

3. From and after the date that one of the diesel generators or an associated critical bus is made or found to be inoperable for any reason, continued reactor operation is permissible in accordance with Specification 3.5.F.1 if Specification 3.9.A.1 is satisfied.

4. From and after the date that both diesel generators are made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 24 hours in accordance with Specification 3.5.F.2 if Specification 3.9.A.1 is satisified.

5. From and after the date that one of the diesel generators or associated critical buses and either the emergency or startup transformer power source are made or found to be inoperable for any reason, continued reactor operation is permissible in accordance with Specification 3.5.F.1, provided the other off-site source, startup transformer or emergency transformer is available and capable of automatically supplying power to the 4160V critical buses and the NRC is notified within 24 hours of the occurrence and the plans for restoration of the inoperable components.

4.9.B

4.9 BASES

The monthly test of the diesel generator is conducted to check for equipment failures and deterioration. Testing is conducted up to equilibrium operating conditions to demonstrate proper operation at these conditions. The diesel generator will be manually started, synchronized and connected to the bus and load picked up. The diesel generator should be loaded to at least 35% of rated load to prevent fouling of the engine. It is expected that the diesel generator will be run for at least two hours. Diesel generator experience at other generating stations indicates that the testing frequency is adequate and provides a high reliability of operation should the system be required.

Each diesel generator has two air compressors and two air receivers for starting. It is expected that the air compressors will run only infrequently. During the monthly check of the diesel generator, each receiver in each set of receivers will be drawn down below the point at which the corresponding compressor automatically starts to check operation and the ability of the compressors to recharge the receivers.

The diesel generator fuel consumption rate at full load is approximately 275 gallons per hour. Thus, the monthly load test of the diesel generators will test the operation and the ability of the fuel oil transfer pumps to refill the day tank and will check the operation of these pumps from the emergency source.

The test of the diesel generator during the refueling outage will be more comprehensive in that it will functionally test the system; i.e, it will check diesel generator starting and closure of diesel generator breaker and sequencing of load on the diesel generator. The diesel generator will be started by simulation of a loss-of-coolant accident. In addition, an undervoltage condition will be imposed to simulate a loss of off-site power.

Periodic tests between refueling outages verify the ability of the diesel generator to run at full load and the core and containment cooling pumps to deliver full flow. Periodic testing of the various components, plus a functional test once-a-cycle, is sufficient to maintain adequate reliability.

Although station batteries will deteriorate with time, utility experience indicates there is almost no possibility of precipitous failure. The type of surveillance described in this specification is that which has been demonstrated over the years to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure. In addition, the checks described also provide adequate indication that the batteries have the specified ampere-hour capability.

The diesel fuel oil quality must be checked to ensure proper operation of the diesel generators. Water content should be minimized because water in the fuel could contribute to excessive damage to the diesel engine.

When it is determined that some auxiliary electrical equipment is out of service, the increased surveillance required in Section 4.5.F is deemed adequate to provide assurance that the remaining equipment will be operable.

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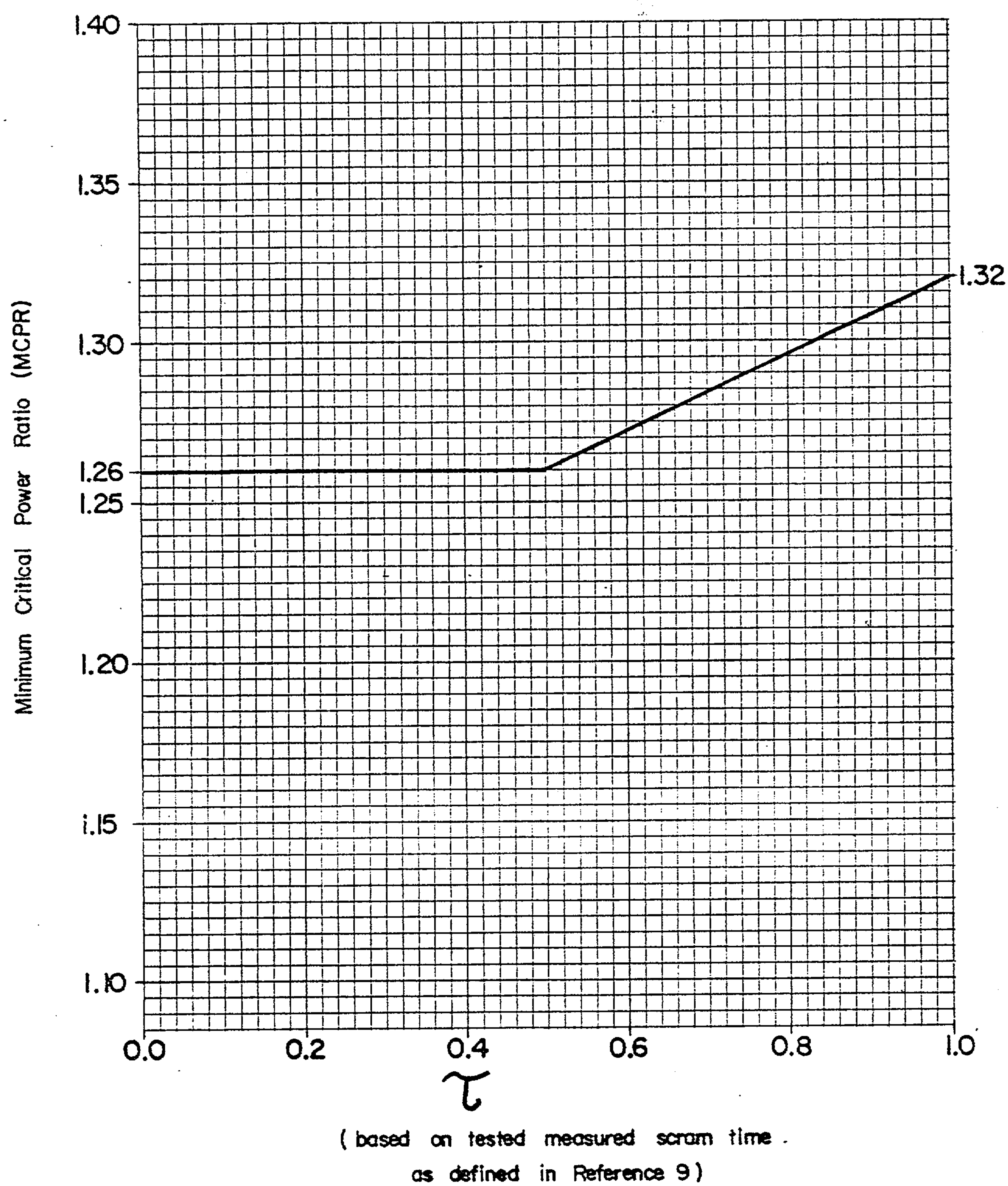

Figure 3.11-2b 8 x 8 Fuel

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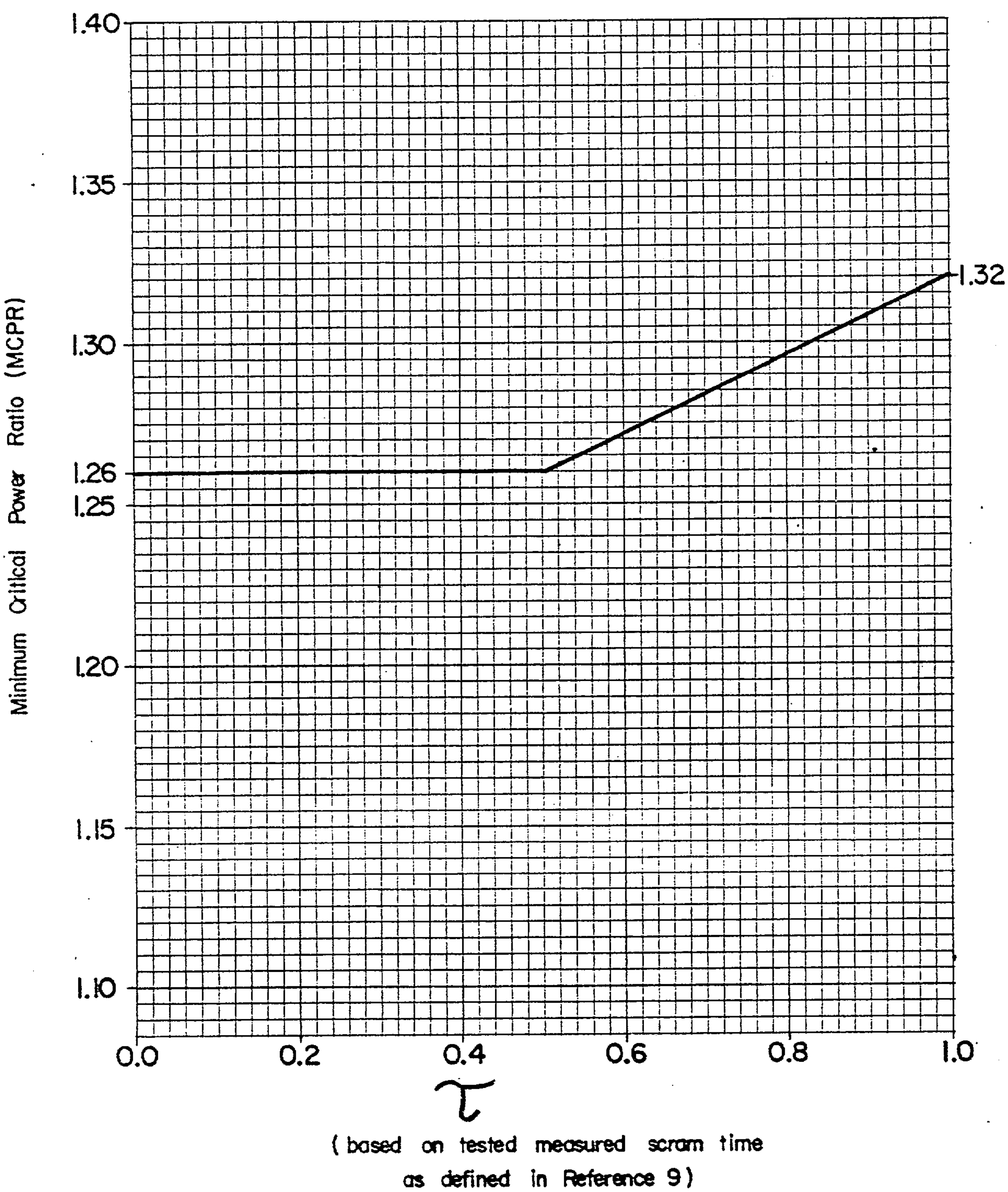

Figure 3.11-2c 8x8R Fuel

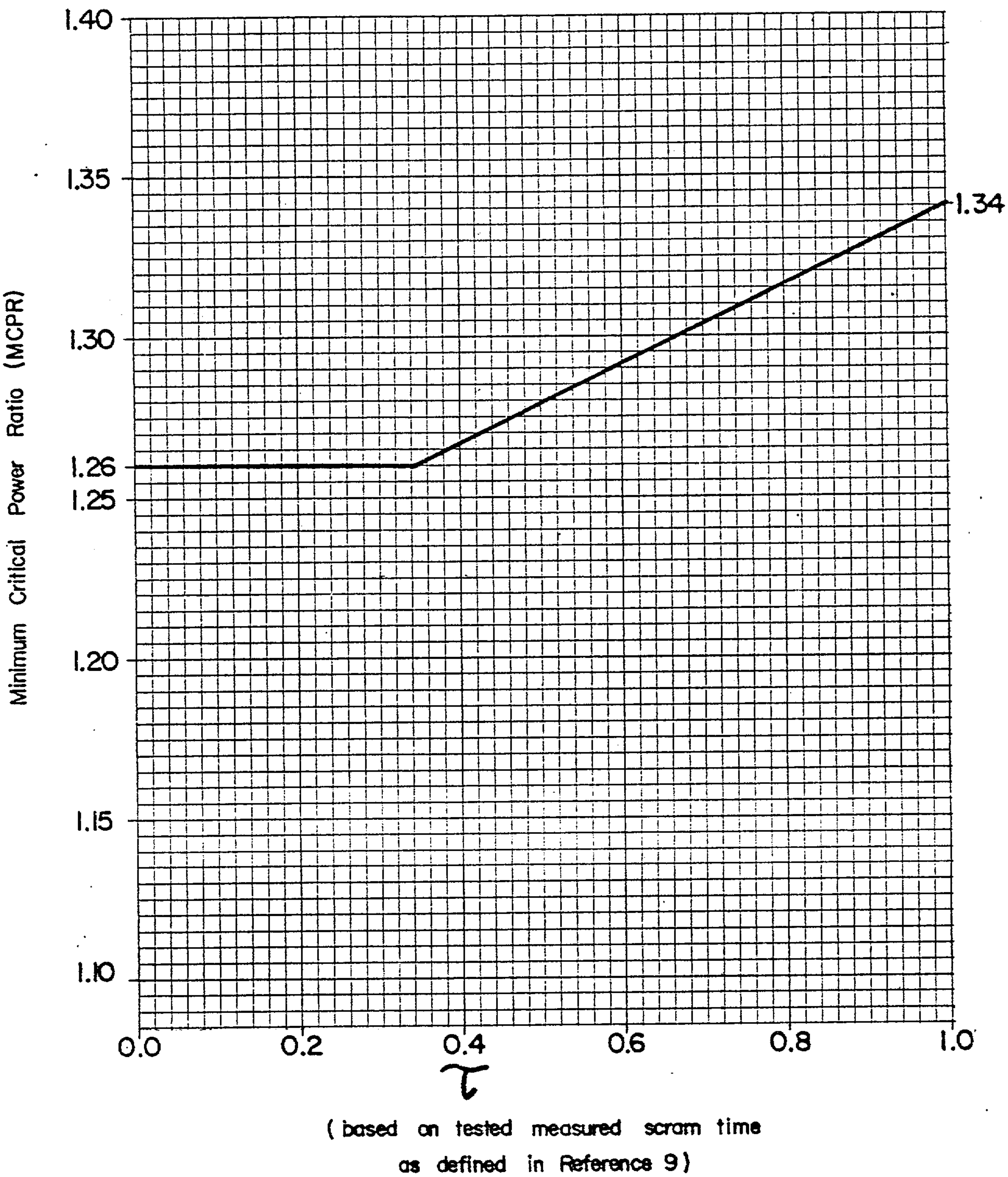

Figure 3.11-2d P8x8R Fuel

3.11 BASES

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20^{o}F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50 Appendix K limit. The limiting value for APLHGR is shown in Figure 3.11-1.

The calculational procedure used to establish the APLHGR shown on Figure 3.11.1 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10CRF50. A complete discussion of each code employed in the analysis is presented in Reference 1.

References for Bases 3.11.A

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566, dated January 1976.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 5 of Reference 2 and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern. Pellet densification power spiking in 8x8 fuel has been accounted for in the safety analysis presented in Reference 5; thus no adjustment to the LHGR limit for densification effects is required for 8x8 fuels.

C. Minimum Critical Power Ratio (MCPR)

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.11C are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07, and an analysis of abnormal operational transients (Reference 5). For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the more limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The limiting transient which determines the required steady state MCPR limit and thus yields the largest ΔCPR is discussed in Reference 5. When added to the safety limit MCPR of 1.07 the deterministic MCPR's are obtained. The required minimum operating limit MCPR's are determine by methods given in References 8 and 9.

Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in Section 4 of NEDO-24011[(2)] and on core parameters shown in Table 5-2 of Reference 2.

The evaluation of a given transient begins with the system initial parameters shown in Table 5-2 of Reference 2 that are input to the GE core dynamic behavior transient computer program described in NEDO-10802[(3)] and NEDO-24154[(7)]. The outputs of the program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDE-20566[(4)]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow, the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

3.11 Bases: (Cont'd)

The K_f factor curves shown in Figure 3.11-3 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.11-3, are conservative for Cooper operation because the operating limit MCPR's are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

D. Thermal-hydraulic Stability

The calculations, regarding reactor core stability, presented in Reference 5 show that the reactor is in compliance with the ultimate performance criteria, including the most responsive condition at natural circulation and rod block power. However, to preclude the possibility of operation under conditions which could result in reactor core instability, the NRC requested the incorporation of a specification limit.

The power level specified results in a decay ratio (X_2/X_0) which is significantly less than the ultimate stability limit of 1.0.

References for Bases 3.11.B, 3.11.C, 3.11.D

1. "Cooper Nuclear Station Channel Inspection and Safety Analyses with Bypass Holes Plugged," NEDO-21072, October 1975.

2. Licensing Topical Report, General Electric Boiling Water Reactor, Generic Reload Fuel Application, (NEDE-24011-P), (most current approved submittal).

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566, dated January 1976.

5. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (applicable reload document).

6. April 18, 1978 letter from J. M. Pilant (NPPD) to G. E. Lear (NRC).

3.11 Bases: (Cont'd)

7. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154, Volumes 1, 2 and 3, October 1978.

8. Letter, R. H. Buckholz (GE) to P. S. Check (NRC), "ODYN Adjustment Methods for Determination of Operating Limits," January 19, 1981.

9. Letter (with attachment) R. H. Buckholz (GE) to P. S. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980.

4.11 Bases:

A&B. Average and Local LHGR

The LHGR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

C. Minimum Critical Power Ratio (MCPR) - (Surveillance Requirement)

At core thermal power levels less than or equal to 25%, the reactor will be operating at less than or equal to minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation was made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin was thus demonstrated such that subsequent MCPR evaluation below this power level was shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION

B. Reactor Building Closed Cooling Water System (REC)

1. Both reactor building closed cooling water loops and their associated pumps shall be operable whenever irradiated fuel is in the vessel or the spent fuel pool, except as specified in 3.12.B.2 and 3.12.B.3 below.

2. From and after the date that any component in one loop becomes inoperable continued reactor operation is permissible during the succeeding thirty days provided that during such thirty days all the components of the other loop and the active components of the engineered safeguards compartment cooling systems, the diesel generator associated with the operable loop are operable.

   The allowable repair time does not apply when the reactor is in the shutdown mode and reactor pressure is less than 75 psig.

3. Both reactor building closed cooling water loops with one pump per loop shall be operable as stated in 3.12. B.1 and 3.12.B.2 above during reactor head-off operations requiring LPCI or Core Spray System availability or service water cooling shall be available.

4. If the requirements of 3.12.B.1 through 3.12.B.3 cannot be met, the reactor shall be shutdown in an orderly manner and in the Cold Shutdown condition within 24 hours or operations requiring LPCI or core spray system availability shall be halted.

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SURVEILLANCE REQUIREMENTS

B. Reactor Building Closed Cooling Water System (REC)

1. REC System Testing

	Item	Frequency
a.	Pump Operability	Once/Month
b.	Motor operated Valve Operability	Once/Month
c.	Pump flow rate Each pump shall deliver 1175 gpm at 65 psid.	Once/3 months and after pump maintenance
d.	System head tank level shall be monitored.	Daily

2. When it is determined that any active component in an REC loop is inoperable, all components in the other loop shall be demonstrated operable immediately and weekly thereafter.

INSTRUMENT LOCATION		INSTRUMENT ID NO.
2	<u>Control Room</u>	FP-SD-17-1 FP-SD-17-2 FP-SD-17-3
3	<u>Cable Spreading Room</u>	FP-SD-16-1 FP-SD-16-2 FP-SD-16-3 FP-SD-16-4 FP-SD-16-5 FP-SD-16-6
	<u>Cable Expansion Room</u>	FP-SD-16-7 FP-SD-16-8
4	<u>Switchgear Rooms</u>	
	DC Switchgear Rooms	FP-SD-15-2 FP-SD-15-3
	Critical Switchgear Room	FP-SD-22-1 FP-SD-22-2
5	<u>Station Battery Rooms</u>	FP-SD-15-1 FP-SD-15-4 FP-SD-15-1A FP-SD-15-4A
6	<u>Diesel Generator Rooms</u>	FP-SD-10-1 FP-SD-10-2 FP-SD-10-3 FP-SD-10-4 CO2-SD-DG-1A CO2-SD-DG-1B CO2-SD-DG-1C CO2-SD-DG-1D CO2-SD-DG-2A CO2-SD-DG-2B CO2-SD-DG-2C CO2-SD-DG-2D
7	<u>Diesel Fuel Storage Rooms</u>	CO2-TD-DG-1A CO2-TD-DG-1B
8	<u>Safety Related Equipment not in Reactor Building</u>	
	RHR Service Water Booster Pumps	FP-SD-14-3
	Emergency Condensate Storage Tanks	FP-SD-14-1
	Fire Water Pumps & Service Water Pumps	FP-FD-32-1 FP-FD-32-2
9	<u>Auxiliary Relay Room & Reactor Protection System Rooms</u>	
	Auxiliary Relay Room	FP-SD-15-9
	Reactor Protection System Room 1A	FP-SD-15-7
	Reactor Protection System Room 1B	FP-SD-15-8

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5.0 MAJOR DESIGN FEATURES

5.1 Site Features

The Cooper Nuclear Station site is located in Nemaha County, Nebraska, on the west bank of the Missouri River, at river mile 532.5. This part of the river is referred to by the Corps of Engineers as the Lower Brownville Bend. Site coordinates are approximately 40° 21' north latitude and 95° 38' west longitude. The site consists of 1351 acres of land owned by Nebraska Public Power District. About 205 acres of this property is located in Atchison County, Missouri, opposite the Nebraska portion of the station site. The land area upon which the station is constructed is crossed by the Missouri River on the east and is bounded by privately owned property on the north, south, and west. At the west site boundary, a county road and Burlington Northern Railroad spur pass the site.

The reactor (center line) is located approximately 3600 feet from the nearest property boundary. No part of the present property shall be sold or leased by the applicant which would reduce the minimum distance from the reactor to the nearest site boundary to less than 3600 feet without prior NRC approval.

The protected area is formed by a seven foot chain link fence which surrounds the site buildings.

5.2 Reactor

A. The core shall consist of not more than 548 fuel assemblies in any combination of 7x7 (49 fuel rods) and 8x8 (63 fuel rods) and 8x8R/P8x8R (62 fuel rods).

B. The core shall contain 137 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% theoretical density.

5.3 Reactor Vessel

The reactor vessel shall be as described in Section IV-20 of the SAR. The applicable design shall be as described in this section of the SAR.

5.4 Containment

A. The principal design parameters for the primary containment shall be as given in Table V-2-1 of the SAR. The applicable design shall be as described in Section XII-2.3 of the SAR.

B. The secondary containment shall be as described in Section V-3.0 of the SAR.

C. Penetrations to the primary containment and piping passing through such

G. A Fire Brigade of at least 5 members shall be maintained at all times. This excludes the 3 members of the minimum shift crew necessary for safe shutdowns, and other personnel required for other essential functions during a fire emergency. Three fire Brigade members shall be from the Operations Department and 2 support members may be from other departments inclusive of Security personnel.

Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.

H. In order to perform the function of accident assessment an engineer from the normal plant engineering staff shall be assigned to each shift during reactor operation. If the lack of qualified engineers necessitates, an additional senior reactor operator assigned to each shift may substitute in the performance of the accident assessment function. This requirement is effective until January 1, 1981.

6.1.4 The minimum qualifications, training, replacement training, and retraining of plant personnel at the time of fuel loading or appointment to the active position shall meet the requirements as described in the American National Standards Institute N-18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants". The Assistant to Station Superintendent qualifications shall comply with Section 4.2 of ANSI-N18.1-1971. The Chemistry and Health Physics Supervisor shall meet or exceed the qualifications of Regulatory Guide 1.8, Sept. 1975; personnel qualification equivalency as stated in the Regulatory Guide may be proposed in selected cases. The minimum frequency of the retraining program shall be every two years. The training program shall be under the direction of a designated member of the plant staff.

A. A training program for the fire brigade will be maintained under the direction of the plant training coordinator and shall meet or exceed the requirements of Section 27 of the NFPA Code 1976, except for Fire Brigade training sessions which shall be held at least quarterly.

The training program requirements will be provided by a qualified fire protection engineer.

6.2 Review and Audit

6.2.1 The organization and duties of committees for the review and audit of station operation shall be as outlined below:

A. Station Operations Review Committee

1. Membership:

a. Chairman: Station Superintendent or Assistant to Station Superintendent

b. Engineering Supervisor

c. Operations Supervisor

d. Chemistry and Health Physics Supervisor

e. Maintenance Supervisor

f. Quality Assurance Supervisor - non-voting member.

Alternate members shall be appointed in writing by the Station Superintendent to serve on a temporary basis; however, no more than one alternate shall serve on the Committee at any one time.

2. Meeting Frequency: Monthly, and as required on call of the Chairman.

3. Quorum: Station Superintendent or Assistant to Station Superintendent plus two other members including alternates.

4. Responsibilities:

a. Review all proposed normal, abnormal, maintenance and emergency operating procedures specified in 6.3.1, 6.3.2, 6.3.3, and 6.3.4 and proposed changes thereto: and any other proposed procedures or changes thereto determined by any member to effect nuclear safety.

b. Review all proposed tests and experiments and their results, which involve nuclear hazards not previously reviewed for conformance with technical specifications. Submit tests which may constitute an unreviewed safety question to the NPPD Safety Review and Audit Board for review.

c. Review proposed changes to Technical Specifications, license and the Final Safety Analysis Report.

d. Review proposed changes or modifications to station systems or equipment as discussed in the FSAR or which involves an unreviewed safety question as defined in 10CFR50.59(c). Submit changes to equipment or systems having safety significance to the NPPD Safety Review and Audit Board for review.

e. Review station operation to detect potential unsafe conditions.

6.2 (cont'd)

f. Investigate all reported instances of violations of Technical Specifications, including reporting evaluation and recommendations to prevent recurrence, to the Division Manager of Power Operations and to the Chairman of the NPPD Safety Review and Audit Board.

g. Perform special reviews and investigations and render reports thereon as requested by the Chairman of the Safety Review and Audit Board.

h. Review all events which are required by regulations or Technical Specifications to be reported to the NRC in writing within 24 hours.

i. Review drills on emergency procedures (including plant evacuation) and adequacy of communication with off site groups.

j. Review all procedures required by these Technical Specifications, including procedures of the Emergency Plan and the Security Plan with a frequency commensurate with their safety significance but at an interval of not more than two years.

5. Authority

a. The Station Operations Review Committe shall be advisory.

b. The Station Operations Review Committee shall recommend to the Station Superintendent approval or disapproval of proposals under items 4, a through e and j above. In case of disagreement between the recommendations of the Station Operations Review Committee and the Station Superintendent, the course determined by the Station Superintendent to be the more conservative will be followed. A written summary of the disagreement will be sent to the Division Manager of Power Operations and to the NPPD Safety Review and Audit Board.

c. The Station Operations Review Committee shall report to the Chairman of the NPPD Safety Review and Audit Board on all reviews and investigations conducted under items 4.f, 4.g, 4.h, and 4.i.

d. The Station Operations Review Committee shall make tentative determinations regarding whether or not proposals considered by the Committee involve unreviewed safety questions. This determination shall be subject to review and approval by the NPPD Safety Review and Audit Board.

6. Records:

Minutes shall be kept for all meetings of the Station Operations Review Committee and shall include identification of all documentary material reviewed; copies of the minutes shall be forwarded to the Chairman of the NPPD Safety Review and Audit Board and the Division Manager of Power Operations within one month.

7. Procedures:

Written administrative procedures for Committee operation shall be prepared and maintained describing the method for submission and content of presentations to the committee, provisions for use of subcommittees, review and approval by members of written Committee evaluations and recommendations, dissemination of minutes, and such other matters as may be appropriate.

B. NPPD Safety Review and Audit Board.

The board must: verify that operation of the plant is consistent with company policy and rules, approve operating procedures and operating license provisions; review safety related plant changes, proposed tests and procedures; verify that unusual events are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events; and detect trends which may not be apparent to a day-to-day observer.

Audits of selected aspects of plant operation shall be performed with a frequency commensurate with their safety significance and in such a manner as to assure that an audit of all nuclear safety related activities is completed within a period of two years. Periodic review of the audit programs should be performed by the Board at least twice a year to assure that such audits are being accomplished in accordance with requirements of Technical Specifications. The audits shall be performed in accordance with appropriate written instructions or procedures and should include verification of compliance with internal rules, procedures (for example, normal, off-normal, emergency, operating, maintenance, surveillance, test and radiation control procedures and the emergency and security plans), regulations involving nuclear safety and operating license provisions; training, qualification and performance of operating staff; and corrective actions following abnormal occurrences or unusual events. A representative portion of procedures and records of the activities performed during the audit period shall be audited and, in addition, observations of performance of operating and maintenance activities shall be included. Written reports of such audits shall be reviewed at a scheduled meeting of the Board and by appropriate members of management including those having responsibility in the area audited. Follow-up action, including reaudit of deficient areas, shall be taken when indicated.

In addition to the above, the Safety Review and Audit Board will audit the facility fire protection and its implementing procedures at least once every 24 months.

1. Membership

   a. Senior Division Manager of Power Operations (chairman)

   b. Division Manager of Licensing and Quality Assurance (alternate Chairman)

   c. Division Manager of Power Projects

   d. Division Manager of Power Supply

   e. Division Manager of Environmental Affairs

   f. Consultants (as required)

   The Board members shall collectively have the capability required to review problems in the following areas: nuclear power plant operations, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, mechanical and electrical engineering, and other appropriate fields associated with the unique characteristics of the nuclear power plant involved. When the nature of a particular problem dictates, special consultants will be utilized.

   Alternate members shall be appointed in writing by the Board Chairman to serve on a temporary basis; however, no more than two alternates shall serve on the Board at any one time.

2. Meeting frequency: Semiannually, and as required on call of the Chairman.

3. Quorum: Chairman or Vice Chairman, plus three members including alternates. No more than a minority of the quorum shall be from groups holding line responsibility for the operation of the plant.

4. Responsibilities: The following subjects shall be reported to and reviewed by the NPPD Safety Review and Audit Board.

   a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

   b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

6.4 Actions to be Taken in the Event of Occurrences Specified in Section 6.7.2.A.

6.4.1 Occurrences, as specified in Section 6.7.2.A., shall be promptly reported to the Station Superintendent, Division Manager of Power Operations and the Chairman of the NPPD Safety Review and Audit Board and shall be promptly reviewed by the Station Operations Review Committee. This committee shall prepare a separate report. This report shall include an evaluation of the cause of the occurrence, a record of the corrective action taken, and recommendations for appropriate action to prevent or reduce the probability of a repetition of the occurrence. Copies of all such reports shall be submitted to the Power Operations Department and the NPPD Safety Review and Audit Board Chairman for review and approval of any recommendations.

6.4.2 All occurrences as specified in Section 6.7.2.A. shall be reported to the General Manager on a periodic basis.

6.5 Action to be Taken if a Safety Limit is Exceeded

6.5.1 If a safety limit is exceeded, reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC. An immediate report shall be made to the Division Manager of Power Operations, the General Manager and to the chairman of the NPPD Safety Review and Audit Board. A complete analysis of the circumstances leading up to and resulting from the situation together with recommendations to prevent a recurrence shall be prepared by the Station Operations Review Committee. This report shall be submitted to the Division Manager of Power Operations and the NPPD Safety Review and Audit Board. Appropriate analyses or reports will be submitted to the NRC. Notification of such occurrences will be made to the NRC by the Station Superintendent within 24 hours as specified in Specification 6.7.

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6.7 Station Reporting Requirements

6.7.1 Routine Reports

A. In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the individual(s) designated in the current revision of Reg. Guide 10.1 unless otherwise noted.

B. Start up Report

1. A summary report of plant startup and power escalation testing shall be submitted following:

   a. Receipt of an operating license.

   b. Amendment to the license involving a planned increase in power level.

   c. Installation of fuel that has a different design or has been manufactured by a different fuel supplier.

   d. 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 FSAR and shall 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.

2. 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) 9 months following initial criticality, whichever is earliest. If all three events are not completed, supplementary reports shall be submitted every three months.

C. Annual Reports

Routine reports covering the subjects noted in 6.7.1.C.1 6.7.1.C.2, and 6.7.1.C.3 for the previous calendar year shall be submitted prior to March 1 of each year.

1. A tabulation on an annual basis of 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, 1/ e.g., reactor operations 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 totaling 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.

2. A summary description of facility changes, tests or experiments in accordance with the requirements of 10CFR50.59(b).

3. Pursuant to 3.8.A, a report of radioactive source leak testing. This report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

D. Monthly Operating Report

Routine reports of operating statistics, shutdown experience, and a narrative summary of operating experience relating to safe operation of the facility, shall be submitted on a monthly basis to the individual designated in the current revision of Reg. Guide 10.1 no later than the tenth of each month following the calendar month covered by the report.

6.7.2. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

1/ This tabulation supplements the requirements of §20.407 of 10CFR Part 20.

A. Prompt Notification With Written Follow-up. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the appropriate Regional Office, no later than the first working day following the event, with a written follow-up report within two weeks. The written follow-up report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

1. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items 6.7.2.A.5, 6.7.2.A.6 or 6.7.2.B.1 below.

2. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item 6.7.2.B.2 below.

3. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

Note: ...is item is intended to provide ... or reporting of potentially generic problems.

B. Thirty Day Written Reports. The reportable occurrences discussed below shall be the subject of written reports to the appropriate Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

1. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

2. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 6.7.2.B.1 and 6.7.2.B.2 need not be reported except where test results themselves reveal a degraded mode as described above.

3. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

4. Abnormal degradation of systems other than those specified in item 6.7.2.A.3 above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 80 to Facility Operating License No. DPR-46 issued to Nebraska Public Power District (the licensee), which revised the Technical Specifications for operation of the Cooper Nuclear Station located in Nemaha County, Nebraska. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications to permit changes to the maximum critical power ratio values and rod block monitor upscale trip level setting, eliminate historical and extraneous material, correct typographical and grammatical errors, reflect organizational title changes, eliminate temporary restrictions no longer applicable, include equipment required by Technical Specifications, clarify definitions and bases sections and update references.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated April 30, 1982, (2) Amendment No. 80 to License No. DPR-46, and (3) the Commission's letter to the licensee dated June 24, 1982. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Auburn Public Library, 118 15th Street, Auburn, Nebraska 68305. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 24th day of June 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing