50-237

AMENDMENT 82 TO POL CONSISTING OF CHANGES TO TECHNICAL SPECIFICATIONS

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50-237
8/6/84

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RECORDS FACILITY BRANCH

DPR-19

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

PROVISIONAL OPERATING LICENSE

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

a. The application for a facility license dated November 17, 1967, as amended, (Amendment Nos. 7 through 20, dated August 30, 1968, November 21, 1968, February 28, 1969, March 18, 1969, April 16, 1969, May 20, 1969, July 2, 1969, July 22, 1969, August 5, 1969, August 8, 1969, August 18, 1969, August 18, 1969, September 2, 1969, and October 16, 1969, respectively) complies with the requirements of the Atomic Energy Act of 1954, as amended, and the Commission's regulations set forth in Title 10, Chapter 1, CFR;

b. The facility has been constructed in accordance with the applications, as amended, and the provisions of Provisional Construction Permit No. CPPR-18;

c. There are involved features, characteristics and components to which it is desirable to obtain actual operating experience before the issuance of an operating license for the full term requested in the application;

d. There is reasonable assurance (i) that the facility can be operated at steady-state power levels not in excess of 2527 megawatts (thermal) in accordance with this license without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

e. The applicant is technically and financially qualified to engage in the activities authorized by this operating license; in accordance with the rules and regulations of the Commission;

f. The applicant has furnished proof of financial protection to satisfy the requirements of 10 CFR Part 140;

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

Provisional Operating License No. DPR-19 is hereby issued to Commonwealth Edison Company (Commonwealth Edison), as follows:

1. This license applies to the Dresden Nuclear Power Station Unit 2, a single cycle, boiling, light water reactor, and electric generating equipment (the facility). The facility is located at the Dresden Nuclear Power Station in Grundy County, Illinois, and is described in the "Safety Analysis Report," as supplemented and amended (Amendment Nos. 7 through 20).

2. Subject to the conditions and requirements incorporated herein the Commission hereby licenses Commonwealth Edison:

   A. Pursuant to Section 104b of Atomic Energy Act of 1954, as amended (the Act), and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location at the Dresden Nuclear Power Station;

   B. Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear materials, not including plutonium, as reactor fuel, in accordance with the limitations for storage and amounts required for operation as described in the Final Safety Analysis Report, as supplemented and amended as of September 3, 1976;

Am. 24
9/03/76

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

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

Am. 34 1/30/78 2. E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Dresden Nuclear Power Station, Units Nos. 1, 2 and 3.

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations; 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50-54 and 50-59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to the additional conditions specified below:

A. Maximum Power Level

Commonwealth Edison is authorized to operate the facility at steady state power levels not in excess of 2527 megawatts (thermal).

Am. 82 08/06/84

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment 82 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

Commonwealth Edison shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

Commonwealth Edison shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. For the purpose of repairing a crack in the recirculation bypass line in the "A" loop, the licensee may perform the repair program as described in a report entitled "Commonwealth Edison Company Dresden Station 2A Recirculations Pump 4" Equalizing Line Repair Program" transmitted by letter dated September 23, 1974.

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3\. F. Restrictions

Operation in the coastdown mode is permitted to 40% power. Should off-normal feedwater heating be necessary for extended periods during coastdown (i.e. greater than 24 hours) the Licensee shall perform a safety evaluation to determine if the MCPR Operating Limit and calculated peak pressure for the worst case abnormal operating transient remain bounding for the new condition.

Am. 58
3/31/81

G. Equalizer Valve Restriction

Am. 21
5/23/77

The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation.

H. The licensee may proceed with and is required to complete the modification identified in Paragraphs 3.1.1 through 3.1.23 of the NRC's Fire Prtotection Safety Evaluation (SE) dated March 1978 on the facility. All modifications are to be completed by start-up following the 1979 Unit 2 refueling outage. In addition, the licensee shall submit the additional information identified in Table 3.1 of this SE in accordance with the schedule contained therein. In the event these dates for submittal cannot be met, the licensee shall submit a report, explaining the circumstances, together with a revised schedule.

Am. 36
3/22/78

I. Physical Protection

The licensee shall full implement and maintain in effect all provisions of the following Commission approved documents, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). These approved documents consist of information withheld from public disclosure pursuant to 10 CFR 2.790(d).

(1) "Security Plan for the Dresden Nuclear Power Station", dated November 19, 1977 as revised May 19, 1978, May 27, 1978, July 28, 1978 and February 19, 1979.

Am. 56
2/11/81

(2) "Dresden Nuclear Power Station Safeguards Contingency Plan", dated March 1980, as revised June 27, 1980, submitted pursuant to 10 CFR 73.40. The Contingency Plan shall be fully implemented, in accordance with 10 CFR 73.40(b), within 30 days of this approval by the Commission.

Am. 56
2/11/81

I. (3) "Dresden Nuclear Power Station Guard Training and Qualification Plan", submitted by letter dated August 16, 1979, as revised by letter dated August 11, 1980. This Plan shall be full implemented in accordance with 10 CFR 73.55(b)(4), within 60 days of this approval by the Commission. All security personnel shall be qualified within two years of this approval.

J. Systems Integrity

Am. 55
2/06/81

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

1. Provisions establishing preventive maintenance and periodic visual inspecton requirements, and

2. Leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

K. Iodine Monitoring

Am. 55
2/06/81

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

1. Training of personnel;

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

Am. 53
12/30/80

L. The licensee shall, by January 4, 1981, install a recirculation pump trip, or in the alternative, place and maintain the facility in a cold shutdown or refueling mode of operation.

M. Provisions to allow operation with one recirculation loop out of service:

Am. 63
7/09/81
Am. 75
4/07/83

1. The steady-state thermal power level will not exceed 50% of rated

2. The Minimum Critical Power Ratio (MCPR) Safety Limit will be increased 0.03 (TS 1.1.A and 3.3.B.5.C)

M. 3. The MCPR Limiting Condition for Operation (LCO) will be increased 0.03 (TS 3.5.K and Fig. 3.5-2)

Am. 75
4/07/83

4. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits will be reduced to 70% of current values for all fuel types.

(T.S. reference 3.5.I)

5. The APRM Scram and Rod Block Setpoints and the RBM Setpoints shall be reduced by 3.5% to read as follows:

Am. 75
4/07/83
Corrected by
Letter dated
10/5/83

T.S. 2.1.A.1 $S \leq (.58\ WD + 58.5)$
T.S. 2.1.A.1* $S \leq (.58\ WD + 58.5)$ FRP/MFLPD
T.S. 2.1.B $S \leq (.58\ WD + 46.5)$
T.S. 2.1.B* $S \leq (.58\ WD + 46.5)$ FRP/MFLPD
T.S. 3.2.C (Table 3.2.3):
APRM Upscale $\leq (.58\ WD + 46.5)$ FRP/MFLPD
RBM Upscale $\leq (.65\ WD + 41.5)$

Am. 63
7/9/81

6. The suction valve in the idle loop is closed and electrically isolated until the idle loop is being prepared for return to service.

7. APRM flux noise will be measured once per shift and the recirculation pump speed will be reduced if the flux noise averaged over 1/2 hour exceeds 5% peak to peak, as measured on the APRM chart recorder.

8. The core plate delta p noise will be measured once per shift and the recirculation pump speed will be reduced if the noise exceeds 1 psi peak to peak.

* In the event that MFLPD exceeds FRP for General Electric fuel.

N. Spent Fuel Storage Racks*

Am. 74
8/27/82

1. The licensee is authorized to install and use 33 high-density fuel storage racks in the spent fuel storage pool at Dresden Station Unit 2.

2. Fuel stored in the spent fuel pool shall have a U-235 loading less than or equal to 14.8 grams per axial centimeter.

*(See Note next page)

Am. 74 8/27/82 N. 3. No fuel loads heavier than the weight of a single spent fuel assembly and handling tool shall be carried over fuel stored in the spent fuel pool.

DRL Order 6/10/71

4. This license is effective as of the date of issuance and shall expire December 22, 1972, unless extended for good cause shown, or upon the earlier issuance of a superseding operating license

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by

Peter A. Morris, Director

Division of Reactor Licensing

Attachment:

Appendix A - Technical Specifications

Date of Issuance: December 22, 1969

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* Prior to the effective date of the ASLB's Final Initial Decision of August 17, 1982, the licensee had submitted to the Commission an updated FSAR for Dresden Station, Unit 2 to reflect the commitments set forth in the ASLB's Partial Initial Decision dated September 24, 1981, as referenced in the ASLB's Final Initial Decision, Section III.4.

TECHNICAL SPECIFICATION PAGE INDEX 2/14/84
Dresden Station Unit 2 Nuclear Operating License No. DPR-19

.D PAGE	ORIGINAL SHEET	AMENDMENT NO.	DATE	*NEW PAGE
Appendix A				
Cover Sheet	X (original sheet)			i
i		62	06/25/81	ii
ii		21	05/23/76	iii
iii		36	05/01/78	iv
1		75	04/07/83	1.0-1
2		75	04/07/83	1.0-2 & 3
3		21	05/23/76	1.0-4
4		58	03/31/81	1.0-5
5		75	04/07/83	1/2.1-1
6		75	04/07/83	1/2.1-2
6a		75	04/07/83	1/2.1-2
7		75	04/07/83	1/2.1-3 & 4
8		43	04/16/80	1/2.1-5
9		Blank Page	03/31/81	None
10		75	04/07/83	B 1/2.1-6
11		75	04/07/83	B 1/2.1-7
11a		75	04/07/83	B 1/2.1-8
12		58	03/31/81	B 1/2.1-8 & 9
13		75	04/07/83	B 1/2.1-10
14		75	04/07/83	B 1/2.1-10 & 11
15		75	04/07/83	B 1/2.1-12
15a		75	04/07/83	B 1/2.1-13
16		75	04/07/83	B 1/2.1-13
17		21	05/23/76	B 1/2.1-14
18		75	04/07/83	B 1/2.1-15
18a		21	05/23/76	B 1/2.1-16
18b		59	04/30/81	B 1/2.1-17
19		75	04/07/83	1/2.2-1
20		75	04/07/83	B 1/2.2-2 & 3
21		75	04/07/83	B 1/2.2-4
21a		62	06/25/81	3.0-1 & 2
21b		62	06/25/81	B 3.0-3
22		80	02/14/84	3/4.1-1 & 2
22a		80	02/14/84	3/4.1-3 & 4
23		75	04/07/83	3/4.1-5
24		75	04/07/83	3/4.1-6 & 7

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) PAGE	ORIGINAL SHEET	AMENDMENT NO.	DATE	*NEW PAGE
25		67	03/26/82	3/4.1-8
26		67	03/26/82	3/4.1-9
27		Change 16	11/03/71	3/4.1-10
28		80	02/14/84	B 3/4.1-11
28a		80	02/14/84	B 3/4.1-12
29		75	04/07/83	B 3/4.1-13
30		71	05/13/82	B 3/4.1-14
31	X			B 3/4.1-15
32	X			B 3/4.1-16 & 17
33	X			B 3/4.1-18
34		75	04/07/83	B 3/4.1-19
35	X			B 3/4.1-20
36		75	04/07/83	3/4.2-1
36a		75	04/07/83	3/4.2-2 & 3
37	X			3/4.2-3 & 4
38		78	12/12/83	3/4.2-5
39		78	12/12/83	3/4.2-6
40		43	04/24/79	3/4.2-7
41		72	05/19/82	3/4.2-8
42		75	04/07/83	3/4.2-9
42a		75	04/07/83	3/4.2-10
43		72	05/19/82	3/4.2-11
44		78	12/12/83	3/4.2-11
44a		58	03/31/81	3/4.2-12
45		81	02/24/84	3/4.2-13
46		75	04/07/83	B 3/4.2-14
47		75	04/07/83	B 3/4.2-15 & 16
48		71	05/13/82	B 3/4.2-17
49		58	03/31/81	B 3/4.2-18 & 19
49A		Change 16	11/03/71	B 3/4.2-19
50	X			B 3/4.2-20
51	X			B 3/4.2-20 & 21
52	X			B 3/4.2-22
53	X			B 3/4.2-23
54		13	01/15/76	3/4.3-1 & 2
55		67	03/26/82	3/4.3-3
56		77	11/03/83	3/4.3-4, 5, & 6
57		58	03/31/81	3/4.3-7 & 8
57A		58	03/31/81	3/4.3-9
58		79	12/12/83	3/4.3-10 & 11

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) PAGE	ORIGINAL SHEET	AMENDMENT NO.	DATE	*NEW PAGE
59		19	05/04/76	3/4.3-11
60		58	03/31/81	3/4.3-12 & 13
61		No Change Number	08/27/75	3/4.3-14 & 15
61a		75	04/07/83	3/4.3-15 & 16
62		75	04/07/83	3/4.3-17
62a		75	04/07/83	3/4.3-18
62b		75	04/07/83	3/4.3-18
63		75	04/07/83	3/4.3-19
64		75	04/07/83	3/4.3-20
65		75	04/07/83	3/4.3-21 & 22
66		Change 28 and 19	04/19/74	3/4.4-1
67		Change 28 and 19	04/19/74	3/4.4-2
68		Change 28 and 19	04/19/74	3/4.4-3
69		Change 22	10/23/73	3/4.4-4
70		Change 22	10/23/73	3/4.4-5
71		58	03/31/81	B 3/4.3-6 & 7
72	X			B 3/4.3-7
73		Corrected page	04/06/78	3/4.5-1
74		Change 12	06/15/71	3/4.5-2 & 3
75		14	02/25/76	3/4.5-3 & 4
76		Change 34	05/16/75	3/4.5-5
77		73	07/09/82	3/4.5-6 & 7
78		75	04/07/83	3/4.5-8 & 9
79		Change 28 and 19	04/19/74	3/4.5-10
80		Change 28 and 19	04/19/74	3/4.5-11 & 12
81		58	03/31/81	3/4.5-13
81a		Change 29	07/16/74	3/4.5-14
81b		75	04/07/83	3/4.5-15
81b-1		75	04/07/83	3/4.5-15 & 16
81c (Deleted)				None
81c-1		58	03/31/81	3/4.5-17
81c-2		75	04/07/83	3/4.5-18
81c-3		75	04/07/83	3/4.5-19
81c-4		75	04/07/83	3/4.5-20
81c-5		75	04/07/83	3/4.5-21
81d		75	04/07/83	3/4.5-22
81e		75	04/07/83	3/4.5-24
81e-1		75	04/07/83	3/4.5-25
81f		21	05/23/76	3/4.5-23 & 26
81g		38	06/14/78	3/4.5-27

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82		75	04/07/83	B 3/4.5-28
83		Change 12	06/15/71	B 3/4.5-29 & 30
84		21	05/23/76	B 3/4.5-30 & 31
85		20	06/09/76	B 3/4.5-31 & 32
85a		75	04/07/83	B 3/4.5-32 & 33
85b		75	04/07/83	B 3/4.5-34
85b-1		75	04/07/83	B 3/4.5-35
85c		21	05/23/76	B 3/4.5-35
85d		30	05/02/77	B 3/4.5-36
86		30	05/02/77	B 3/4.5-37
86a		75	04/07/83	B 3/4.5-37 & 38
86b		21	05/23/76	B 3/4.5-39
86c		21	05/23/76	B 3/4.5-39 & 40
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87		33	12/02/77	3/4.6-1
88		44	08/13/79	3/4.6-2 & 3
89		75	04/07/83	3/4.6-4
90		75	04/07/83	3/4.6-5 & 6
90a		75	04/07/83	3/4.6-7
91		54	12/31/80	3/4.6-7
91a		26	11/19/76	3/4.6-8 & 9
91a-1		26	11/19/76	3/4.6-10 & 11
91a-2		26	11/19/76	3/4.6-12
91b		76	10/21/83	3/4.6-13 & 14
91c		70	04/28/82	3/4.6-15 & 16
91c-1		70	04/28/82	3/4.6-22 & 23
91d-1		76	10/21/83	3/4.6-18
91d-2		60	05/06/81	3/4.6-19
91e-1		75	04/07/83	3/4.6-20
91e-2		70	04/28/82	3/4.6-21
91f		39	11/13/78	3/4.6-24
92		39	11/13/79	B 3/4.6-25
93		44	08/13/79	B 3/4.6-26
93a		44	08/13/79	B 3/4.6-27
94	X			B 3/4.6-29
94a		44	08/13/79	B 3/4.6-30
95	X			B 3/4.6-28
96	X			B 3/4.6-31
97		75	04/07/83	B 3/4.6-32
98		54	12/31/80	B 3/4.6-33

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99		Change 28 and 19	04/19/74	B 3/4.6-34
99a		70	04/28/82	B 3/4.6-35
99b		76	10/21/83	B 3/4.6-36
100		Left Blank	12/31/80	B 3/4.6-37 & 38
101		Left Blank	12/31/80	None
102		Left Blank	12/31/80	None
103		Left Blank	12/31/80	None
104		Left Blank	12/31/80	None
105		Left Blank	12/31/80	None
106		Left Blank	12/31/80	None
106A		Left Blank	12/31/80	None
107		Left Blank	12/31/80	None
108		20	06/09/76	3/4.7-1 & 2
108A		68	04/01/82	3/4.7-3
109		70	04/28/82	3/4.7-4, 5, & 6
110		70	04/28/82	3/4.7-6, 7, 8, & 9
111		68 (Blank)	----	None
112		68 (Blank)	----	None
113		68 (Blank)	----	None
114		68 (Blank)	----	None
115		68 (Blank)	----	None
116		Change 28 and 19	04/19/74	3/4.7-10
116a		Change 28 and 19	04/19/74	3/4.7-11 & 12
117		Change 28 and 19	04/19/74	3/4.7-13 & 14
117a		Change 28 and 19	04/19/74	3/4.7-15 & 16
117b		37	06/07/78	3/4.7-17 & 18
118		17	11/17/76	3/4.7-19
118a		17	11/17/76	3/4.7-20
118b		17	11/17/76	3/4.7-21 & 22
118c		17	11/17/76	3/4.7-23
119		17	11/17/76	3/4.7-24 & 25
120		40	02/27/79	3/4.7-26
121	X			3/4.7-27 & 28
122		23	08/09/76	3/4.7-29 & 30
123		48	04/17/80	3/4.7-31
124	X			3/4.7-32
125		43	04/24/79	B 3/4.7-33
125a		Change 36	10/06/75	B 3/4.7-34
126		Change 28 and 19	04/19/74	B 3/4.7-35 & 36
126a		Change 28 and 19	04/19/74	B 3/4.7-37
127		37	06/07/78	B 3/4.7-37 & 38

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127a		37	06/07/78	B 3/4.7-39 & 40
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129		Change 36	10/06/75	B 3/4.7-40
129a		Change 36	10/06/75	B 3/4.7-41
130		68	04/01/82	B 3/4.7-42 & 43
131		Change 28 and 19	04/19/74	B 3/4.7-43 & 44
131a		17	11/17/76	B 3/4.7-45
131b		17	11/17/76	B 3/4.7-46
132		17	11/17/76	B 3/4.7-46 & 47
133		Change 15	09/17/71	3/4.8-1
133a		52	12/29/80	3/4.8-2 & 3
134	X			3/4.8-4, 5, & 6
134a	X			3/4.8-7 & 8
134b	X			3/4.8-8
135		62	06/25/81	3/4.8-9 & 10
136	X			3/4.8-10 & 11
137		32	10/18/77	3/4.8-12 & 13
137a		24	09/03/76	3/4.8-14 & 15
138		32	10/18/77	3/4.8-16
139		32	10/18/77	3/4.8-16
140		32	10/18/77	3/4.8-17
141-142b (one page only)		32	10/18/77	None
143	X			B 3/4.8-18
143a	X			B 3/4.8-19
144		32	10/18/77	B 3/4.8-20
145		32	10/18/77	B 3/4.8-21 & 22
145a		32	10/18/77	B 3/4.8-23
146		72	05/19/82	3/4.9-1
147		72	05/19/82	3/4.9-2 & 3
148		72	05/19/82	3/4.9-4 & 5
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149	X			
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APPENDIX A

TO

OPERATING LICENSE DPR-19

TECHNICAL SPECIFICATIONS

AND BASES

FOR

DRESDEN NUCLEAR POWER STATION UNIT 3

GRUNDY COUNTY, ILLINOIS

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

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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. (Deleted)

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.

C. 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 XN-3 correlation. (Reference XN-NF-512).

D. Hot Standby - Hot standby means operation with the reactor critical, system pressure less than 600 psig, and the main steam isolation valves closed.

E. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

F. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm, or trip. Response time is not part of the routine instrument calibration, but will be checked once per cycle.

G. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument response alarm, and/or initiating action.

H. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

I. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

J. 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 represents 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.

K. Fraction of Limiting Power Density (FLPD) - For fuel fabricated by GE, the fraction of limiting power density is the ratio of the Linear Heat Generation Rate (LHGR) existing at a given location to the design LHGR for that bundle type. FLPD does not apply to Exxon Nuclear Company (ENC) fuel.

L. Logic System Function Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

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

N. Mode - The reactor mode is that which is established by the mode-selector-switch.

O. Operable - A system, subsystem, train, component, or device shall be operable when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

P. Operating - Operating means that a system, subsystem, train, component or device is performing its intended functions in its required manner.

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

R. 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 connecting 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.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

S. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

T. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady-state power level of 2527 thermal megawatts.

U. Rated Thermal Power - Rated thermal power means a steady-state power level of 2527 thermal megawatts.

V. 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 thermal power.

1. Startup/Hot Standby Mode - In this mode the reactor protection scram trips, initiated by condensor low vacuum and main steamline isolation valve closure, are by-passed when reactor pressure is less than 600 psig; the low pressure main steamline isolation valve closure trip is bypassed, the reactor protection system is energized with IRM neutron-monitoring system trips and control rod withdrawal interlocks in service.

2. Run Mode - In this mode the reactor protection system is energized with APRM protection and RBM interlocks in service.

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

X. 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 subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

Y. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary system are assured. Exceeding such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission (NRC) 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.

Z. 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 are secured in the isolated position.

AA. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alternations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized.

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.

BB. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

CC. Surveillance Interval - Each surveillance requirement shall be performed within the specified surveillance interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval.

b. A total maximum combined interval time for any 3 consecutive intervals not to exceed 3.25 times the specified surveillance interval.

DD. Fraction of Rated Power (FRP) - The fraction of rated power is the ratio of core thermal power to rated thermal power of 2527 Mwth.

EE. 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.

FF. Maximum Fraction of Limiting Power Density (MFLPD) - The maximum fraction of limiting power density is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

1.1 SAFETY LIMIT

FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to these variables which monitor the fuel thermal behavior.

Objective:

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications:

A. Reactor Pressure greater than 800 psig and Core Flow greater than 10% of Rated.

The existence of a minimum critical power ratio (MCPR) less than 1.06 for GE 8x8R fuel, or less than 1.05 for ENC or GE 8x8 fuel, shall constitute violation of the MCPR fuel cladding integrity safety limit.

2.1 LIMITING SAFETY SYSTEM SETTING

FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications:

A. Neutron Flux Trip Settings

The limiting safety system trip settings shall be as specified below:

1. APRM Flux Scram Trip Setting (Run Mode)

When the reactor mode switch is in the run position, the APRM flux scram setting shall be:

S less than or equal to $[.58W_D + 62]$

with a maximum set point of 120% for core flow equal to 98×10^6 lb/hr. and greater, where:

S - setting in per cent of rated power.

1.1 SAFETY LIMIT (Cont'd.)

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

W_D = per cent of drive flow required to produce a rated core flow of 98 Mlb/hr.

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

Where: S is less than or equal to
$(.58W_D + 62)$ [FRP/MFLPD]

FRP = fraction of rated thermal power (2527 MWt)

MFLPD = maximum fraction of limiting power density for GE fuel

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

This adjustment may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

2. APRM Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

1.1 SAFETY LIMIT (Cont'd.)

B. Core Thermal Power Limit (Reactor Pressure is less than or equal to 800 psig)

When the reactor pressure is less than or equal to 800 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

When the reactor mode switch is in the refuel startup/hot standby position, the APRM scram shall be set at less than or equal to 15% of rated neutron flux.

3. IRM Flux Scram Trip Setting

The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

B. APRM Rod Block Setting

The APRM rod block setting shall be:

S is less than or equal to [$.58W_D + 50$]

The definitions used above for the APRM scram trip apply.

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

S is less than or equal to ($.58W_D + 50$) [FRP/MFLPD]

The definitions used above for the APRM scram trip apply.

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

1.1 SAFETY LIMIT (Cont'd.)

C. Power Transient

1. The neutron flux shall not exceed the scram setting established in Specification 2.1.A for longer than 1.5 seconds as indicated by the process computer.

2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds the scram setting established by Specification 2.1.A and a control rod scram does not occur.

D. Reactor Water Level (Shutdown Condition)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core.
Note: Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

The adjustment may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

C. Reactor low water level scram setting shall be greater than or equal to 144" above the top of the active fuel at normal operating conditions.
Note: Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

D. Reactor low water level ECCS initiation shall be 84"(+4", minus 0") above the top of the active fuel at normal operating conditions.
Note: Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

1.1 SAFETY LIMIT (Cont'd.)

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

E. Turbine stop valve scram shall be less than or equal to 10% valve closure from full open.

F. Generator Load Rejection Scram shall initiate upon actuation of the fast closure solenoid valves which trip the turbine control valves.

G. Main Steamline Isolation Valve Closure Scram shall be less than or equal to 10% valve closure from full open.

H. Main Steamline Pressure initiation of main steamline isolation valve closure shall be greater than or equal to 850 psig.

I. Turbine Control Valve Fast Closure Scram on loss of control oil pressure shall be set at greater than or equal to 900 psig.

1.1 SAFETY LIMIT BASES

FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damages would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than the MCPR fuel cladding integrity safety limit. MCPR greater than the MCPR fuel cladding integrity safety limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity by assuring that the fuel does not experience transition boiling.

The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosions or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforation signals a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling. See reference XN-NF-524.

A. Reactor Pressure greater than 800 psig and Core Flow greater than 10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical

1.1 SAFETY LIMIT BASES (Cont'd.)

power ratio (CPR) which is the ratio of the bundle power which would produce onset of boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. (Figure 2.1-3).

The MCPR Fuel Cladding Integrity Safety Limit assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR=1.00) and the MCPR Fuel Cladding Integrity Safety Limit is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the XN-3 critical power correlation. Refer to XN-NF-524 for the methodology used in determining the MCPR Fuel Cladding Integrity Safety Limit.

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. The assumed reactor conditions used in defining the safety limit introduce conservatism into the limit because boundingly high radial power peaking factors and boundingly flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that during sustained operation at the MCPR Fuel Cladding Integrity Safety Limit there would be no transition boiling in the core. If boiling transition were to occur, however, there is reason to believe that the integrity of the fuel would not necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach; much of the data indicates that LWR fuel can survive for an extended period in an environment of transition boiling.

1.1 SAFETY LIMIT BASES (Cont'd.)

If the reactor pressure should ever exceed the limit of applicability of the XN-3 critical power correlation as defined in XN-NF-512, it would be assumed that the MCPR Fuel Cladding Integrity Safety Limit had been violated. This applicability pressure limit is higher than the pressure safety limit specified in Specification 1.2. For fuel fabricated by General Electric Company, operation is further constrained to a maximum linear heat generation rate (LHGR) of 13.4 kW/ft by Specification 3.5.J. This constraint is established to provide adequate safety margin to 1% plastic strain for abnormal operational transients initiated from high power conditions. Specification 2.1.A.1 provides for equivalent safety margin for transients initiated from lower power conditions by adjusting the APRM flow-biased scram by the ratio of FRP/MFLPD. Specification 3.5.J establishes the maximum value of LHGR which cannot be exceeded during steady power operation for GE fuel types.

For fuel fabricated by Exxon Nuclear Company, (ENC) fuel design criteria have been established to provide protection against fuel centerline melting and cladding strain, ENC has performed fuel design analysis which demonstrate that centerline melting is not predicted to occur during transient overpower conditions throughout the life of the fuel. Protection of the MCPR and MAPLHGR limits and operation within the power distribution assumptions of the fuel design analysis will provide adequate protection against centerline melt and ensures compliance with ENC's clad overstrain criteria for steady state and transient operation. Since ENC's design criteria are more conservative than the 1% plastic strain limitation on GE fuel, the LHGR limitation and APRM scram adjustment for GE fuel established in specifications 3.5.J and 2.1.A.1 respectively are unnecessary for the protection of ENC fuel. The procedural controls of specification 3.1.B will ensure that operation of ENC fuel remains within the power distribution assumptions of the fuel design analysis.

B. Core Thermal Power Limit (Reactor Pressure less than 800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of $28x10^3$ lbs/hr. bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow

1.1 SAFETY LIMIT BASES (Cont'd.)

with a 4.56 psi driving head will be greater than $28x10^3$ lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. At 25% of rated thermal power, the peak powered bundle would have to be operating at 3.84 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

C. Power Transient

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel which is 8 to 9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail. In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a significant increase in surface heat flux occurs.

Control rod scram times are checked as required by Specifications 4.3.C. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analyses show that even if the bypass system fails to operate, the MCPR fuel cladding integrity safety limit is not exceeded. Thus, use of a 1.5 second limit provides additional margin.

The computer provided has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be

1.1 SAFETY LIMIT BASES (Cont'd.)

available for any scram analysis, Specification 1.1.C.2 will be relied on to determine if a safety limit has been violated.

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel* provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

*Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

2.1 LIMITING SAFETY SYSTEM SETTING BASES

FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the units have been analyzed throughout the spectrum of planned operating conditions up to the rated thermal power condition of 2527 MWt. In addition, 2527 MWt is the licensed maximum steady-state power level of the units. This maximum steady-state power level will never knowingly be exceeded. See Reference XN-NF-79-71.

Conservatism is incorporated into the transient analyses which define the MCPR operating limits. Variables which inherently possess little or no uncertainty or whose uncertainty has little or no effect on the outcome of the limiting transient are selected at bounding values. Variables which possess significant uncertainty that may have undesirable effects on thermal margins are addressed statistically. Statistical methods used in the transient analyses are described in XN-NF-81-22. The MCPR operating limits are established such that the occurrence of the limiting transient will not result in the violation of the MCPR Fuel Cladding Integrity Safety Limit in at least 95% of the random statistical combinations of uncertainties. In general, the variables with the greatest statistical significance to the consequences of anticipated operational occurrences are the reactivity feedback associated with the formation and removal of coolant voids and the timing of the control rod scram.

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

Steady-state operation without forced recirculation will not be permitted, except during startup testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

The bases for individual trip settings are discussed in the following paragraphs. For analyses of the thermal consequences of the transients, the MCPR's stated in paragraph 3.5.K as the limiting condition of operation bound those which are conservatively assumed to exist prior to initiation of the transients.

A. Neutron Flux Trip Settings

1. 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. 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 percent 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.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

The scram trip setting must be adjusted to ensure that the LHGR transient peak for G.E. fuel is not increased for any combination of Maximum Fraction of Limiting Power Density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in specification 2.1.A.1 when the MFLPD is greater than the fraction of rated power (FRP).

The adjustment may also be accomplished by increasing the APRM gain by the reciprocal of FRP/MFLPD. This provides the same degree of protection as reducing the trip setting by FRP/MFLPD by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

2. APRM Flux Scram Trip Setting
(Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

3. IRM Flux Scram Trip Setting

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are broken down into 10 ranges, each being one-half of a decade in size.

The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservation was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the MCPR fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

B. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR fuel cladding integrity safety limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worse case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward or APRM gain increased if the maximum fraction of limiting power density for G.E. fuel exceeds the fraction of rated power, thus preserving the APRM rod block safety margin.

C. Reactor Low Water Level Scram - The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained. The scram setpoint is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

D. Reactor Low Low Water Level ECCS Initiation Trip Point - The emergency core 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 well below the clad melting 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 emergency core cooling system component was established based on the reactor low water level scram setpoint. To lower the setpoint of the low water level scram would increase the capacity requirement for each of the ECCS 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 ECCS capacity requirements.

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

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

E. Turbine Stop Valve Scram - The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the MCPR fuel cladding integrity safety limit, even during the worst case transient that assumes the turbine bypass is closed.

F. Generator Load Rejection Scram - The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass; i.e., it prevents MCPR from becoming less than the MCPR fuel cladding integrity safety limit for this transient. For the load rejection without bypass transient from 100% power, the peak heat flux (and therefore LHGR) increases on the order of 15% which provides wide margin to the value corresponding to fuel centerline melting and 1% cladding strain.

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure - The low pressure isolation at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 850 psig would not necessarily constitute an unsafe condition.

H. Main Steam Line Isolation Valve Closure Scram - The low pressure isolation of the main steam lines at 850 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scram set at 10% valve closure, there is no appreciable increase in neutron flux.

I. Turbine Control Valve Fast Closure Scram

The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the generator load rejection scram since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection scram on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the generator load rejection. The scram setpoint of 900 psig is set high enough to provide the necessary anticipatory function and low enough to minimize the number of spurious scrams. Normal operating pressure for this system is 1250 psig. Finally the control valve will not start to close until the fluid pressure is 600 psig. Therefore, the scram occurs well before valve closure begins.

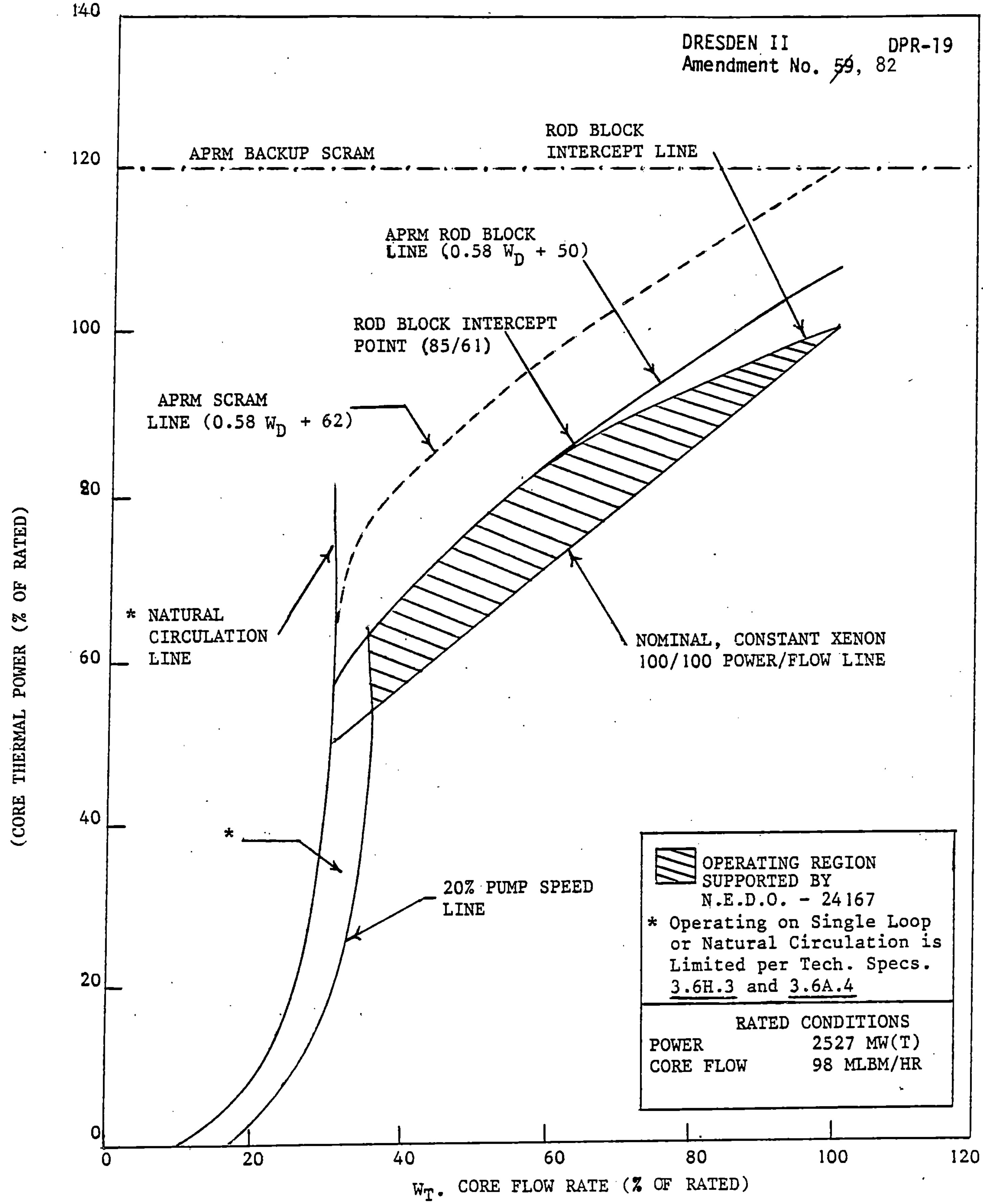


FIGURE 2.1-3
(SCHEMATIC)
APRM FLOW BIAS SCRAM RELATIONSHIP
TO NORMAL OPERATING CONDITIONS

1.2 SAFETY LIMIT

REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1345 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 LIMITING SAFETY SYSTEM SETTING

REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

A. Reactor Coolant High Pressure Scram shall be less than or equal to 1060 psig.

B. Primary System Safety Valve Nominal Settings shall be as follows:

1 valve at 1135 psig*
2 valves at 1240 psig
2 valves at 1250 psig
2 valves at 1260 psig
2 valves at 1260 psig

The allowable setpoint error for each valve shall be plus or minus 1%.

* Target Rock combination safety/relief valve

1.2 SAFETY LIMIT 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 pressure safety limit of 1345 psig as measured by the vessel steam space pressure indicator ensures margin to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel and coolant system piping. The respective design pressures are 1250 psig at 575°F and 1175 psig at 560°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and USASI Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% X 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the reactor vessel. The design pressure for the recirc. suction line piping (1175 psig) was chosen relative to the reactor vessel design pressure. Demonstrating compliance of the peak vessel pressure with the ASME overpressure protection limit (1375 psig) assures compliance of the suction piping with the USASI limit (1410 psig). Evaluation methodology used to assure that this safety limit pressure is not exceeded for any reload is documented in Reference XN-NF-79-71. The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig: this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At that pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the primary system piping and provide a similar margin of protection at the established safety pressure limit.

The normal operating pressure of the reactor coolant system is 1000 psig. For the turbine trip or loss of electrical load transients, the turbine trip scram or generator load rejection

1.2 SAFETY LIMIT BASES (Cont'd.)

scram, together with the turbine bypass system, limit the pressure to approximately 1100 psig (See Note below). In addition, pressure relief valves have been provided to reduce the probability of the safety valves, which discharged to the drywell, operating in the event that the turbine bypass should fail.

Finally, the safety valves are sized to keep the reactor vessel peak pressure below 1375 psig with no credit taken for the relief valves during the postulated full closure of all MSIV's without direct (valve position switch) scram. Credit is taken for the neutron flux scram, however.

The indirect flux scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full scale pressure recorder.

Note: SAR, Section 11.2.2 -
also: "Dresden 3 Second Reload License Submittal," 9-14-73
also: "Dresden Station Special Report No. 29 Supplement B."

2.2 LIMITING SAFETY SYSTEM SETTING BASES

In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the neutron flux scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus exceeding the pressure safety limit. The pressure scram is available as a backup protection to the direct valve position trip scrams and the high flux scram.

If the high flux scram were to fail, a high pressure scram would occur at 1060 psig. Analyses are performed as described in reference XN-NF-79-71 for each reload to assure that the pressure safety limit is not exceeded.

3.0 LIMITING CONDITION FOR OPERATION

A. In the event a Limiting Condition for Operation cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least hot shutdown within 12 hours and in cold shutdown within the following 24 hours unless corrective measures are completed that satisfy the Limiting Conditions for Operation. Exceptions to these requirements are stated in the individual specifications.

B. When a system, subsystem, train, component, or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operations, provided: (1) its corresponding normal or emergency power source is operable; and (2) all of its redundant system(s), subsystem(s), train(s), component(s), and device(s) in the other division are operable, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least

3.0 LIMITING CONDITION FOR OPERATION (Cont'd.)

hot shutdown within 12 hours, and in at least cold shutdown within the following 24 hours.

C. Specifications 3.0.A and 3.0.B are not applicable in refueling or cold shutdown.

3.0 LIMITING CONDITION FOR OPERATION BASES

3.0.A. This specification delineates the action to be taken for circumstances not directly provided for in the Limiting Condition for Operation statements and whose occurrence would violate the intent of the specification.

3.0.B. This specification delineates what additional conditions must be satisfied to permit operation to continue, consistent with the Limiting Condition for Operation statements for power sources, when a normal or emergency power source is not operable. Power sources are defined as AC Auxiliary Electrical Systems as defined in Section 3.9.A.1, 3.9.A.2, and 3.9.A.3. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the Limiting Condition for Operation Statements associated with individual systems, subsystems, trains, components or devices to be consistent with the Limiting Condition for Operation statements of the associated electrical power source. It allows operation to be governed by the time limits of action statements associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual action statements for each system, subsystem, train, component, or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

3.1 LIMITING CONDITIONS FOR OPERATION

REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

A. Reactor Protection System

1. 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. The system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milliseconds.

2. If during operation, the maximum fraction of limiting power density for fuel fabricated by GE exceeds the fraction of rated power when operating above 25% rated thermal power, either:

4.1 SURVEILLANCE REQUIREMENTS

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. Reactor Protection System

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

2. Daily during reactor power operation above 25% rated thermal power, the core power distribution shall be checked for:

3.1 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

a. The APRM scram and rod block settings shall be reduced to the values given by the equations in Specifications 2.1.A.1 and 2.1.B. This may be accomplished by increasing APRM gains as described therein.

b. The power distribution shall be changed such that the maximum fraction of limiting power density no longer exceeds the fraction of rated power.

For fuel fabricated by ENC, operation of the core shall be limited to ensure the power distribution is consistent with that assumed in the Fuel Design Analysis for overpower conditions.

3. Two RPS electric power monitoring channels for each inservice RPS MG set or alternate source shall be OPERABLE at all times.

4.1 SURVEILLANCE REQUIREMENTS (Cont'd.)

a. Maximum fraction of limiting power density for fuel fabricated by GE (MFLPD) and compared with the fraction of rated power (FRP).

b. For compliance with assumptions of the Fuel Design Analysis of overpower conditions for fuel fabricated by ENC.

3. The RPS power monitoring system instrumentation shall be determined OPERABLE:

a. At least once per 6 months by performing a CHANNEL FUNCTIONAL TEST, and

3.1 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

4. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable channel to OPERABLE

4.1 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. At least once per operating cycle by demonstrating the OPERABILITY of overvoltage, undervoltage, and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic, and output circuit breakers, and verifying the following setpoints:

Surveillance Requirements: Reactor Protection Buses

(1) Overvoltage
126.5V $\pm$ 2.5%

Min. 123.3V
Max. 129.6V

(2) Undervoltage
108V $\pm$ 2.5%

Min. 105.3V
Max. 110.7V

(3) Underfrequency
56.0 Hz $\pm$ 1% of 60 Hz

Min. 55.4 Hz
Max. 56.6 Hz

3.1 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

5. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

4.1 SURVEILLANCE REQUIREMENTS (Cont'd.)

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Minimum Number Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable: Refuel (7)	Startup/Hot Standby	Run	Action*
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
3	High Flux	(LT/E) 120/125 of Full Scale	X	X	X(5)	A
3	Inoperative		X	X	X(5)	A
	APRM					
2	High Flux	Specification 2.1.A.1	X	X(9)	X	A or B
2	Inoperative**		X	X(9)	X	A or B
2	Downscale	(GT/E) 5/125 of Full Scale	X(12)	X(12)	X(13)	A or B
2	High Flux (15% Scram)	Specification 2.1.A.2	X	X	X(14)	A
2	High Reactor Pressure	(LT/E) 1060 psig	X(11)	X	X	A
2	High Drywell Pressure	(LT/E) 2 psig	X(8), X(10)	X(8), (10)	X(10)	A
2	Reactor Low Water Level	(GT/E) 1 inch***	X	X	X	A
2	High Water Level in Scram Discharge Tank	(LT/E) 50 Gallons	X(2)	X	X	A
2	Turbine Condenser Low Vacuum	(GT/E) 23 in. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Steam Line High Radiation	(LT/E) 3 X Full Power Background	X(3)	X(3)	X(15)	A or C
4(6)	Main Steam Line Isolation Valve Closure	(LT/E) 10% Valve Closure	X(3)	X(3)	X	A or C
2	Generator Load Rejection	****	X(4)	X(4)	X(4)	A or C
2	Turbine Stop Valve Closure	(LT/E) 10% Valve Closure	X(4)	X(4)	X(4)	A or C
2	Turbine Control - Loss of Control Oil Pressure	(GT/E) 900 psig	X	X	X	A or C

Notes: (LT/E) = Less than or equal to.

(GT/E) = Greater than or equal to.

(Notes continue on next two pages)

<u>NOTES</u>: (For Table 3.1.1)

1. There shall be two operable or tripped trip systems for each function.

2. Permissible to bypass, with control rod block, for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.

3. Permissible to bypass when reactor pressure less than 600 psig.

4. Permissible to bypass when first stage turbine pressure less than that which corresponds to 45% rated steam flow.

5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any one valve without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a. Mode Switch in Shutdown
   b. Manual Scram
   c. High Flux IRM
   d. Scram Discharge Volume High Level

8. Not required to be operable when primary containment integrity is not required.

9. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

10. May be bypassed when necessary during purging for containment inerting or deinerting.

11. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

12. The APRM downscale trip function is automatically bypassed when the reactor mode switch is in the refuel and startup/hot standby positions.

13. The APRM downscale trip function is automatically bypassed when the IRM instrumentation is operable and not high.

(Cont'd. next page)

NOTES: (For Table 3.1.1 Cont'd.)

14. The APRM 15% scram is bypassed in the run mode.

15. Due to addition of hydrogen to the primary coolant, the Main Steam Line Radiation monitor setting will be less than or equal to 3 times full power background without hydrogen addition for all conditions except for greater than 20% power with hydrogen being injected during which the Main Steam Line Radiation trip setting will be less than or equal to 3 times full power background with hydrogen addition. Required changes in Main Steam Line Radiation Monitor trip setting will be made within 24 hrs. except during controlled power descensions at which time the setpoint change will be made prior to going below 20% power. If due to a recirculation pump trip or other unanticipated power reduction event the reactor is below 20% power without the setpoint change, control rod motion will be suspended until the necessary trip setpoint adjustment is made.

* If the first column cannot be met for one of the trip systems, that trip system shall be tripped.

If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

a. Initiate insertion of operable rods and complete insertion of all operable rods within 4 hours.

b. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.

c. Reduce turbine load and close main steam line isolation valves within 5 hours.

d. In the refuel mode, when any control rod is withdrawn, suspend all operations involving core alterations and insert all insertable control rods within one hour.

** An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there are less than 56% of the normal complement of LPRM's to an APRM.

*** 1 inch on the water level instrumentation is greater than or equal to 504" above vessel zero (see Bases 3.2).

**** Trips upon actuation of the fast closure solenoid which trips the turbine control valves.

TABLE 4.1.1

SCRAM INSTRUMENTATION FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

Instrument Channel	Group (3)	Functional Test	Minimum Frequency (4)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
* High Flux	C	Trip Channel and Alarm (5)	Before Each Startup (6)
* Inoperative	C	Trip Channel and Alarm	Before Each Startup (6)
APRM			
High Flux	B	Trip Output Relays (5)	Once Each Week
Inoperative	B	Trip Output Relays	Once Each Week
Downscale	B	Trip Output Relays (5)	Once Each Week
High Flux (15% scram)	B	Trip Output Relays	Before Each Startup
High Reactor Pressure	A	Trip Channel and Alarm	(1)
High Drywell Pressure	A	Trip Channel and Alarm	(1)
Reactor Low Water Level (2)	A	Trip Channel and Alarm	(1)
High Water Level in Scram Discharge	A	Trip Channel and Alarm (7)	Every 3 Months
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	(1)
Main Steam Line High Radiation (2)	B	Trip Channel and Alarm (5)	Once Each Week
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	(1)
Generator Load Rejection	A	Trip Channel and Alarm	(1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	(1)
Turbine Control - Loss of Control Oil Pressure	A	Trip Channel and Alarm	(1)

Notes: (See next page)

<u>NOTES:</u> (For Table 4.1.1)

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months. 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 Dresden Unit 2.

2. An instrument check shall be performed on low reactor water level once per day and on high steam line radiation once per shift.

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

4. Functional tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

5. This instrumentation is exempted from the Instrument Functional Test Definition (1.0.G). This Instrument Function Test will consist of injecting a simulated electrical signal into the measurement channels.

6. If reactor start-ups occur more frequently than once per week, the functional test need not be performed; i.e., the maximum functional test frequency shall be once per week.

7. The Functional Test of the Scram Discharge Volume float switch shall include actuation of the switch using a water column.

TABLE 4.1.2

SCRAM INSTRUMENTATION CALIBRATIONS

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration Test	Minimum Frequency (2)
*High Flux IRM	C	Comparison to APRM after Heat Balance	Every Shutdown (4)
High Flux APRM			
Output Signal	B	Heat Balance	Once Every 7 Days
Flow Bias	B	Standard Pressure and Voltage Source	Refueling Outage
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Water Level	Every 3 Months
Turbine Condenser Low Vacuum	A	Standard Vacuum Source	Every 3 Months
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 Months
Turbine Control - Loss of Control Oil Pressure	A	Pressure Source	Every 3 Months

NOTES: (For Table 4.1.2)

1. A description of the three groups is included in the bases of this Specification.

2. Calibration tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made during each refueling outage.

*4. If reactor startups occur more frequently than once per week, the functional test need not be performed; i.e., the maximum functional test frequency shall be once per week.

3.1 LIMITING CONDITION FOR OPERATION 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 primary system.

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

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still 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 reactor protection system is of the dual channel type. (Ref. Section 7.7.1.2 SAR.) 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.

Specifications are provided to ensure the operability of the RPS Bus Electrical Protector Assemblies (EPA's). Each channel from either overvoltage, undervoltage, or under frequency will trip the associated MG set or alternate power source.

This system meets the requirements of the proposed IEEE Standard for Nuclear Power Plant Protection Systems issued September 13, 1966. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) and Intermediate Range Monitor (IRM) channels, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the

3.1 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

APRM's #1 and #3 operate contacts in a one subchannel and APRM's #2 and #3 operate contacts in the other subchannel. APRM's #4, #5 and #6 are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, generator load rejection, and turbine stop valve closure are discussed in Specification 2.3.

Instrumentation (pressure switches) in the drywell are provided to detect a loss of coolant accident and initiate the emergency core cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 2.2. A scram is provided at the same setting as the emergency core cooling system (ECCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent the reactor from going critical following the accident.

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. A part of this system is an instrument volume U-tube in the piping which accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for the 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 or no 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 50 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.

3.1 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient with bypass closure. (Ref. Section 4.4.3 SAR) The condenser low vacuum scram is a backup to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 23" Hg vacuum, stop valve closure occurs at 20" Hg vacuum, and bypass closure at 7" Hg vacuum.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds 3 times full power background for all condition except for greater than 20% power with hydrogen being injected during which the Main Steam Line trip setting is less than or equal to 3 times full power background with hydrogen addition (See Note 15 of Table 3.1.1). The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors which cause an isolation of the main condenser off-gas line provided the limit specified in Specification 3.8 is exceeded.

The main steam line isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. (Ref. Section 7.7.1.2 SAR.)

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

3.1 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The IRM system provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges. (Ref. Sections 7.4.4.2 and 7.4.4.3 SAR.) A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram functions. (Ref. Section 7.4.3.2 SAR.) Thus, the IRM is required in the "Refuel" and "Start/Hot Standby" modes. In the power range the APRM system provides required protection. (Ref. Section 7.3.5.2 SAR.) Thus, the IRM system is not required in the "Run" mode. The APRM's cover only the power range, the IRM's provide adequate coverage in the start-up and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level, and scram discharge volume high level scrams are required for Startup/Hot Standby and Run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have all scram functions except those listed in Note 8 of Table 3.1.1 operable in the Refuel mode is to assure that shifting to the Refuel mode during reactor power operation does not diminish the need for the reactor protection system.

The turbine condenser low vacuum scram is only required during power operation and must be bypassed to start up the unit. At low power conditions a turbine stop valve closure does not result in a transient which could not be handled safely by other scrams such as the APRM.

The requirement that the IRM's be inserted in the core when the APRM's read 5/125 of full scale assures that there is proper overlap in the neutron monitoring systems and thus, that adequate coverage is provided for all ranges of reactor operation.

4.1 SURVEILLANCE REQUIREMENT 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 for Dresden 3. 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, faulted cables, etc., 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.

Surveillance requirements are provided for the RPS EPA's to demonstrate their operability. The setpoints for overvoltage, undervoltage and under frequency have been chosen based on analysis. (Reference T. Raush letter to H. Denton 02-04-83).

The 13 channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups respecting functional testing. These are:

1. On-Off sensors that provide a scram trip function.

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

3. Devices which only serve a useful function during some restricted mode of operation, such as startup or shutdown, or for which the only practical test is one that can be performed at shutdown.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. Actual history on this class of sensors operating in nuclear power plants shows 4 failures in 472 sensor years, or a failure rate of 0.97 X 10^{-6}/hr. During design, a goal of 0.99999 probability of success (at the 50% confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function

*Reference (6); See next page.

4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

of the sensor failure rate and the test interval. A three-month test interval was planned for group (A) sensors. This is in keeping with good operating practice, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95% confidence level is proposed. With the (1 out of 2) X (2) logic, this requires that each sensor have an availability of 0.993 at the 95% confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history (See Reference 6). To facilitate the implementation of this technique, Figure 4.1.1 is provided to indicate an appropriate trend in test interval. The procedure is as follows:

1. Like sensors are pooled into one group for the purpose of data acquisition.

2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T (M = nT).

3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1.1.

4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.

5. A test interval of one month will be used initially until a trend is established.

Group (B) devices utilize an analog sensor followed by an amplifier and a bi-stable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks"

Reference 6:
Reliability of Engineered Safety Features as a Function of Testing Frequency, I.M. Jacobs, Nuclear Safety, Vol. 9, No. 4, July-Aug. 1968, pp. 310-312.

4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

midscale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. For purposes of analysis, it is assumed that this rare failure will be detected within two hours.

The bi-stable trip circuit which is a part of the Group (B) devices can sustain unsafe failures which are 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 (Proposed IEEE Standard for Nuclear Power Plant Protection Systems, Section 4.9, September 13, 1966). 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 zero flow signal will be sent to half of the APRM's resulting in a half scram and rod block condition. Thus, if the calibration was performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

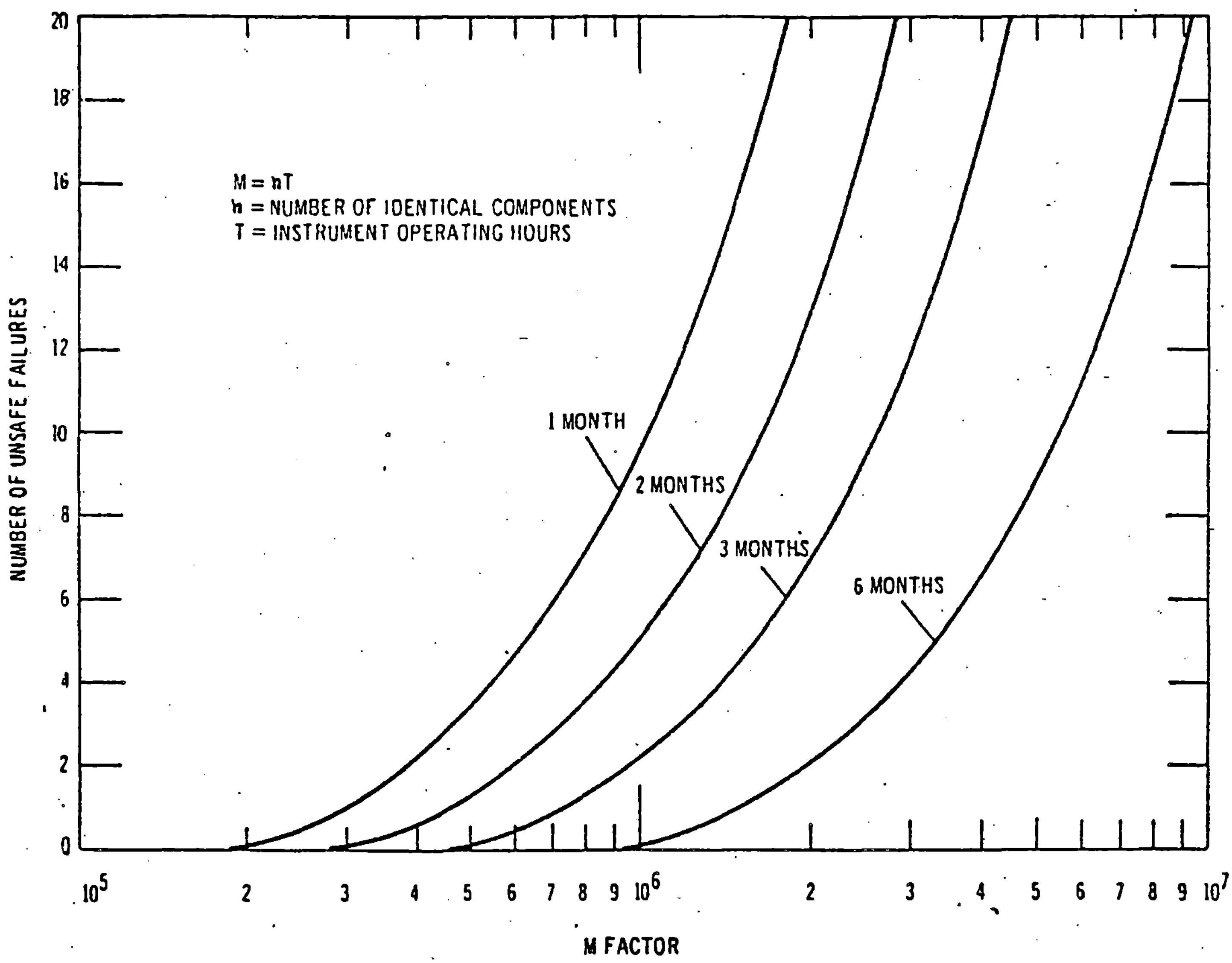


Figure 4.1.1
Graphical Aid in the Selection of an Adequate Interval Between Tests

4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in Commonwealth Edison generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month, a drift of 0.4% would occur and thus provide for adequate margin.

For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.1 and 4.1.2 indicates that six instrument channels have not been included in the latter Table. These are: Mode Switch in Shutdown, Manual Scram, High Water Level in Scram Discharge Volume Float Switches, Main Steam Line Isolation Valve Closure, Generator Load Rejection, and Turbine Stop Valve Closure. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration is not applicable; i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving; e.g., the switches in the scram discharge volume tank. Based on the above, no calibration is required for these six instrument channels.

B. The MFLPD for fuel fabricated by GE shall be checked once per day to determine if the APRM gains or scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations.

4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

Only a small number of control rods are moved daily and thus the peaking factors are not expected to change significantly and thus a daily check of the MFLPD is adequate.

For fuel fabricated by ENC, the power distribution will be checked once per day to ensure consistency with the power distribution assumptions of the fuel design analysis for overpower conditions. During periods of operation beyond these power distribution assumptions, the APRM gains or scram settings may be adjusted to ensure consistency with the fuel design criteria for overpower conditions.

3.2 LIMITING CONDITION FOR OPERATION

PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the plant instrumentation which performs a protective function.

Objective:

To assure the operability of protective instrumentation.

Specification:

A. Primary Containment Isolation Functions

When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table 3.2.1.

B. Core and Containment Cooling Systems - Initiation and Control

The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2.2. This instrumentation must be operable when the system(s) it initiates or controls are required to be operable.

4.2 SURVEILLANCE REQUIREMENT

PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the surveillance requirements of the instrumentation that performs a protective function.

Objective:

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

Specification:

Instrumentation shall be functionally tested and calibrated as indicated in Table 4.2.1.

3.2 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. Control Rod Block Actuation

1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3.

2. The minimum number of operable instrument channels specified in Table 3.2.3 for the Rod Block Monitor may be reduced by one in one of the trip systems for maintenance and/or testing, provided that this condition does not last longer than 24 hours in any 30-day period. In addition, one channel may be bypassed above 30% rated power without a time restriction provided that a limiting control rod pattern does not exist and the remaining RBM channel is operable.

D. Steam Jet-Air Ejector Off Gas System

1. Except as specified in 3.2.D.2. below, both steam-jet air ejector off-gas system radiation monitors shall be operable during reactor power operation. The trip settings for the monitors shall be set

4.2 SURVEILLANCE REQUIREMENT (Cont'd.)

3.2 LIMITING CONDITION FOR OPERATION (Cont'd.)

at a value not to exceed the equivalent of the stack release limit specified in Specification 3.8. The time delay setting for closure of the steam jet-air ejector isolation valves shall not exceed 15 minutes.

2. From and after the date that one of the two steam-jet air ejector off-gas system radiation monitors is made or found to be inoperable, continued reactor power operation is permissible during the next seven days provided the inoperable monitor is tripped in the upscale position.

E. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

1. Except as specified in 3.2.E.2 below, four radiation monitors shall be operable at all times.

2. One of the two radiation monitors in the ventilation duct and one of the two radiation monitors on the refueling floor may be inoperable for 24 hours. If the inoperable monitor is

4.2 SURVEILLANCE REQUIREMENT (Cont'd.)

3.2 LIMITING CONDITION FOR OPERATION (Cont'd.)

not restored to service in this time, the reactor building ventilation system shall be isolated and the standby gas treatment operated until repairs are complete.

3. The radiation monitors shall be set to trip as follows:

   a. ventilation duct -- 11 mr/hr

   b. refueling floor -- 100 mr/hr

4.2 SURVEILLANCE REQUIREMENT (Cont'd.)

TABLE 3.2.1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION FUNCTIONS

Minimum No. of Operable Inst. Channels per Trip System (1)	Instruments	Trip Level Setting	Action (3)
2	Reactor Low Water	(GT/E)144" above top of active fuel*	A
2	Reactor Low Low Water	(GT/E)84" above top of active fuel*	A
2	High drywell pressure	(LT/E)2 psig rated (4), (5)	A
2 (2)	High Flow Main Steam line	(LT/E)120% of rated steam flow	B
2 of 4 in each of 4 sets	High Temperature Main Steam Line Tunnel	(LT/E)200°F	B
2	High Radiation Main Steam Line Tunnel (6)	(LT/E)3 times normal rated power background	B
2	Low Pressure Main Steamline	(GT/E)850 psig	B
	High Flow Isolation Condenser Line		C
1	Steamline Side	(LT/E)20 psi diff. on steamline side	
1	Condensate Return Side	(LT/E)32" water diff. on condensate return side	C
2	High Flow HPCI Steam Line	(LT/E)150" water (7)	D
4	High Temperature HPCI Steam Line Area	(LT/E)200 degrees F	D

Notes: (LT/E) = Less than or equal to.
(GT/E) = Greater than or equal to.
(GT) = Greater than.
(LT) = Less than.

(Notes continue on next page)

NOTES: (For Table 3.2.1)

1. Whenever primary containment integrity is required, there shall be two operable or tripped trip systems for each function, except for low pressure main steamline which only need be available in the RUN position.

2. Per each steamline.

3. Action: If the first column cannot be met for one of the trip systems, that trip system shall be tripped.

   If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

   A. Initiate an orderly shutdown and have reactor in cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have reactor in Hot Standby within 8 hours.

   C. Close isolation valves in isolation condenser system.

   D. Close isolation valves in HPCI subsystems.

4. Need not be operable when primary containment integrity is not required.

5. May be bypassed when necessary during purging for containment inerting and deinerting.

6. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in the primary coolant.

7. Due to addition of hydrogen to the primary coolant, the Main Steam Line Radiation monitor setting will be less than or equal to 3 times full power background without hydrogen addition for all conditions except for greater than 20% power with hydrogen being injected during which the Main Steam Line Radiation trip setting will be less than or equal to 3 times full power background with hydrogen addition. Required changes in the Main Steam Line Radiation Monitor trip setting will be made within 24 hours except during controlled power descensions at which time the setpoint change will be made prior to going below 20% power. If due to a recirculation pump trip or other unanticipated power reduction event the reactor is below 20% power without the setpoint change, control rod motion will be suspended until the necessary trip setpoint adjustment is made.

8. Verification of time delay setting between 3 and 9 seconds shall be performed during each refueling outage.

* Top of active fuel is defined as 360" above vessel zero for all water levels used in the LOCA analyses (see Bases 3.2).

TABLE 3.2.2

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Min. No. of Operable Inst. Channels per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	Reactor Low Low Water Level	84" (plus 4", minus 0") above top of active fuel*	1. In conjunction with low reactor pressure initiates core spray and LPCI. 2. In conjunction with high dry-well pressure 120 sec. time delay, and low pressure core cooling interlock initiates auto blowdown. 3. Initiates HPCI and SBGTS. 4. Initiates starting of diesel generators.
2	High Drywell Pressure (2), (3)	(LT/E) 2 psig	1. Initiates core spray, LPCI, HPCI, and SBGTS. 2. In conjunction with low low water level, 120 sec. time delay, and low pressure core cooling interlock initiates auto blowdown. 3. Initiates starting of diesel generators.
1	Reactor Low Pressure	300 psig $\leq$ p $\leq$ 350 psig	1. Permissive for opening core spray and LPCI admission valves. 2. In conjunction with low low reactor water level initiates core spray and LPCI.
	Containment Spray Interlock		Prevents inadvertent operation of containment spray during accident conditions.
1(4)	2/3 Core Height	(GT/E) 2/3 core height	
2(4)	Containment High Pressure	0.5 psig $\leq$ p $\leq$ 1.5 psig	
1	Timer Auto Blowdown	(LT/E) 120 seconds	In conjunction with low low reactor water level, high dry-well pressure, and low pressure core cooling interlock initiates auto blowdown.
2	Low Pressure Core Cooling Pump Discharge Pressure	50 psig $\leq$ p $\leq$ 100 psig	Defers APR actuation pending confirmation of low pressure core cooling system operation.
2/Bus	Under Voltage on Emergency Buses	(GT/E) 3092 volts (equals 3255 less 5% tolerance)	1. Initiates starting of diesel generators. 2. Permissive for starting ECCS pumps. 3. Removes nonessential loads from buses.
2	Sustained High Reactor Pressure	(LT/E) 1070 psig for 15 seconds	Initiates isolation condenser.
2/Bus	Degraded Voltage on 4 KV Emergency Buses	(GT/E) 3708 volts (equals 3784 volts less 2% tolerance) after (LT/E) 5 minutes (plus 5% tolerance) with a 7-second (+ or - 20%) inherent time delay	Initiates alarm and picks up time delay relay. Diesel generator picks up load if degraded voltage not corrected after time delay.

Notes:

(LT) = Less than
(GT) = Greater than
(LT/E) = Less than or equal to
(GT/E) = Greater than or equal to
APR = Automatic Pressure Relief
(Notes continue on next page)

NOTES: (For Table 3.2.2)

1. For all positions of the Reactor Mode Selector Switch (except for the containment interlock) whenever any ECCS subsystem is required to be operable, there shall be two operable or tripped trip systems. If the first column cannot be met for one of the trip systems, that system shall be tripped. If the first column cannot be met for both trip systems, immediately initiate an orderly shutdown to cold conditions.

2. Need not be operable when primary containment integrity is not required.

3. May be bypassed when necessary during purging for containment inerting or deinerting.

4. If an instrument is inoperable, it shall be placed (or simulated) in the tripped condition so that it will not prevent containment spray.

* Top of active fuel is defined as 360" above vessel zero for all water levels used in the LOCA analyses (see Bases 3.2).

INSTRUMENTATION THAT INITIATES ROD BLOCK

Table 3.2.3

Minimum No. of Operable Inst. Channels Per Trip System (1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias) (7)	(LT/E) $(0.58W_D + 50)$ (FRP/MFLPD) (Note 2)
1	APRM upscale (refuel and Startup/Hot Standby mode)	(LT/E) 12/125 full scale
2	APRM downscale (7)	(GT/E) 3/125 full scale
1	Rod block monitor upscale (flow bias) (7)	(LT/E) (.65W + 45) (Note 2)
1	Rod block monitor downscale (7)	(GT/E) 5/125 full scale
3	IRM downscale (3)	(GT/E) 5/125 full scale
3	IRM upscale	(LT/E) 108/125 full scale
3	IRM detector not fully inserted in the core	
2 (5)	SRM detector not in startup position	(4)
2 (5) (6)	SRM upscale	(LT/E) 10^5 counts/sec.
1	Scram discharge volume water level - high	25 gal.

Notes: (LT/E) = Less than or equal to
(GT/E) = Greater than or equal to
(Notes continue on next page)

NOTES: (For Table 3.2.3)

1. For the Startup/Hot Standby and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function, except the SRM rod blocks, IRM upscale, IRM downscale and IRM detector not fully inserted in the core need not be operable in the "Run" position and APRM downscale, APRM upscale (flow bias), and RBM downscale need not be operable in the Startup/Hot Standby mode. The RBM upscale need not be operable at less than 30% rated thermal power. One channel may be bypassed above 30% rated thermal power provided that a limiting control rod pattern does not exist. For systems with more than one channel per trip system, if the first column cannot be met for both trip systems, the systems shall be tripped. For the Scram Discharge Volume water level high rod block, there is one instrument channel for one trip system.

2. W_D percent of drive flow required to produce a rated core flow of 98 Mlb/m. MFLPD = highest value of FLPD for G.E. fuel.

3. IRM downscale may be bypassed when it is on its lowest range.

4. This function may be bypassed when the count rate is greater than or equal to 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

7. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS INSTRUMENTATION, ROD BLOCKS, AND ISOLATIONS

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
ECCS INSTRUMENTATION			
1. Reactor Low-Low Water Level	(1)	Once/3 Months	Once/Day
2. Drywell High Pressure	(1)	Once/3 Months	None
3. Reactor Low Pressure	(1)	Once/3 Months	None
4. Containment Spray Interlock			
a. 2/3 Core Height	(1)	Once/3 Months	None
b. Containment High Pressure	(1)	Once/3 Months	None
5. Low Pressure Core Cooling Pump Discharge	(1)	Once/3 Months	None
6. Undervoltage Emergency Bus	Refueling Outage	Refueling Outage	Once/3 months
7. Sustained High Reactor Pressure	(1)	Once/3 Months	None
8. Degraded Voltage Emergency Bus	Refueling Outage (10)	Refueling Outage	Monthly
ROD BLOCKS			
1. APRM Downscale	(1) (3)	Once/3 Months	None
2. APRM Flow Variable	(1) (3)	Refueling Outage	None
3. APRM Upscale (Startup/Hot Standby)	(2) (3)	(2) (3)	(2)
4. IRM Upscale	(2) (3)	(2) (3)	(2)
5. IRM Downscale	(2) (3)	(2) (3)	(2)
6. IRM detector not fully inserted in the core	(2)	N/A	None
7. RBM Upscale	(1) (3)	Refueling Outage	None
8. RBM Downscale	(1) (3)	Once/3 Months	None
9. SRM Upscale	(2) (3)	(2) (3)	(2)
10. SRM Detector Not in Startup Position	(2) (3)	(2) (3)	(2)
11. Scram Instr. Vol. Level - High	Once/3 Months (9)	None	None
MAIN STEAM LINE ISOLATION			
1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	None
2. Steam Line High Flow	(1)	Once/3 Months	Once/Day
3. Steam Line Low Pressure	(1)	Once/3 Months	None
4. Steam Line High Radiation	(1) (3)	Once/3 Months (4)	Once/Day
ISOLATION CONDENSER ISOLATION			
1. Steam Line High Flow	(1)	Once/3 Months	None
2. Condensate Line High Flow	(1)	Once/3 Months	None
HPCI ISOLATION			
1. Steam Line High Flow	(1) (11)	Once/3 Months (11)	None
2. Steam Line Area High Temperature	Refueling Outage	Refueling Outage	None
3. Low Reactor Pressure	(1)	Once/3 Months	None
REACTOR BUILDING VENTILATION SYSTEM VIOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION			
1. Ventilation Exhaust Duct Radiation Monitors	(1)	Once/3 Months	Once/Day
2. Refueling Floor Radiation Monitors	(1)	Once/3 Months	Once/Day
STEAM JET-AIR EJECTOR OFF-GAS ISOLATION			
1. Radiation Monitors	(1) (3)	Once/3 Months (4)	Once/Day
CONTAINMENT MONITORING			
1. Pressure			
a. -5 in. Hg to +5 psig Indicator	None	Once/3 Months	Once/Day
b. 0 to 75 psig Indicator	None	Once/3 Months	None
2. Temperature	None	Refueling Outage	Once/Day
3. Drywell-Torus Differential Pressure (5) (6) (0-3 psid)	None	Once/6 Months (two channels operable) Once/Month (one channel operable)	None
4. Torus Water Level (5) (6)	None	Once/6 Months	
a. +/-25" Wide Range Indicator			
b. 18" Sight Glass			
SAFETY/RELIEF VALVE MONITORING			
1. Safety/Relief Valve Position Indicator (Acoustic Monitor) (8)	(7)	None	Once Per 31 Days
2. Safety/Relief Valve Position Indicator (Temperature monitor) (8)	None	Once every 18 months	Once Per 31 Days
3. Safety Valve Position Indicator (Acoustic Monitor) (8)	(7)	None	Once Per 31 Days
4. Safety Valve Position Indicator (Temperature Monitor) (8)	None	Once every 18 months	Once Per 31 Days

Notes: (See next two pages)

NOTES: (For Table 4.2.1))

1. Initially once per month until exposure hours (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. 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 Dresden Unit 2.

2. Function test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. See Note 4.

4. These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed during each refueling outage.

5. A minimum of two channels is required.

6. From and after the date that one of these parameters (. . . either drywell-torus differential pressure or torus water level indication) is reduced to one indication, continued operation is not permissible beyond thirty days, unless such instrumentation is sooner made operable. In the event that all indications of these parameters (. . . either drywell-torus differential pressure or torus water level) is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty four hours.

7. Functional tests will be conducted before startup at the end of each refueling outage or after maintenance is performed on a particular Safety/Relief Valve.

8. If the number of position indicators is reduced to one indication on one or more valves, continued operation is permissible; however, if the reactor is in a shutdown condition for more than seventy-two hours, it may not be started up until all position indication is restored. In the event that all position indication is lost on one or more valves and such indication cannot be restored in thirty days, an orderly shutdown shall be initiated, and the reactor shall be depressurized to less than 90 psig in 24 hours.

(Cont'd. next page)

NOTES: (For Table 4.2.1) (Cont'd.)

9. The Functional Test of the Scram Discharge Volume float switch shall include actuation of the switch using a water column.

10. Functional test shall include verification of the second level undervoltage (degraded voltage) timer bypass and shall verify operation of the degraded voltage 5-minute timer and inherent 7-second timer.

11. Verification of time delay setting between 3 and 9 seconds shall be performed during each refueling outage.

3.2 LIMITING CONDITION FOR OPERATION BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminates operator errors before they result in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are: 1) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and 2) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiates or controls core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low-reactor water level instrumentation is set to trip at greater than 8 inches on the level instrument (top of active fuel

3.2 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

is defined to be 360 inches above vessel zero) and after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above top of active fuel. Retrofit 8 X 8 fuel has an active fuel length 1.34 inches longer than earlier fuel design. However, present trip setpoints were used in the LOCA analyses.

This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR Section 7.7.2). For a trip setting of 504 inches above vessel zero (144 inches above top of active fuel) and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break; the setting is therefore adequate.

The low low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, - 59 inches is 84 inches above the top of active fuel). This trip initiates closure of Group I primary containment isolation valves (Ref. Section 7.7.2.2 SAR), and also activates the ECC subsystems, starts the emergency diesel generator and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spacious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria.

The high drywell pressure instrumentation is a backup to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 Isolation valves. For the breaks discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 Isolation valves include the drywell vent, purge, and sump Isolation valves. High drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions and only the valves in Group 2 are required to close. The low low water level instrumentation initiates protection for the full spectrum of loss of coolant accidents and causes a trip of Group 1 primary system isolation valves.

3.2 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst case accident, main steamline break outside the drywell, this trip setting of 120% of rated steam flow in conjunction with the flow limiters and main steamline valve closure, limit the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. (Ref. Sections 14.2.3.9 and 14.2.3.10 SAR)

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a back-up to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

High radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times full power background for all conditions except for greater than 20% power with hydrogen being injected during which the Main Steamline trip setting is less than or equal to 3 times full power background with hydrogen addition, and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. (Ref. Section 14.2.1.7 SAR.) The performance of the process radiation monitoring system relative to detecting fuel leakage shall be evaluated during the first five years of operation. The conclusions of this evaluation will be reported to the NRC.

Pressure instrumentation is provided which trips when main steamline pressure drops below 850 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "Refuel" and "Startup/Hot Standby" mode this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 850 psig inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500°F; thus, there are no fission products available for release other than those in the reactor water (Ref. Section 11.2.3 SAR).

3.2 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Two sensors on the isolation condenser supply and return lines are provided to detect the failure of isolation condenser line and actuate isolation action. The sensors on the supply and return sides are arranged in a 1 out of 2 logic and, to meet the single failure criteria, all sensors and instrumentation are required to be operable. The trip settings of 20 psig and 32" of water and valve closure time are such as to prevent uncovering the core or exceeding site limits. The sensors will actuate due to high flow in either direction.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI piping. Tripping of this instrumentation results in actuation of HPCI isolation valves, i.e., Group 4 valves. Tripping logic for this function is the same as that for the isolation condenser and thus all sensors are required to be operable to meet the single failure criteria. The trip settings of 200°F and 300% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not go below the MCPR fuel cladding integrity safety limit. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, 8 IRM's, or 4 SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria are met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. This time period is only approximately 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the refuel and startup/hot standby modes, the APRM rod block function is set at 12% of rated power. This control rod block

3.2 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

provides the same type of protection in the Refuel and Startup/Hot Standby mode as the APRM flow biased rod block does in the run mode; i.e., prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worse case single control rod withdrawal error is analyzed for each reload to assure that with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the MCPR fuel cladding integrity safety limit.

Below 30 percent power, the worst case withdrawal of a single control rod without rod block action will not violate the MCPR fuel cladding integrity safety limit. Thus, the RBM rod block function is not required below this power level.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity safety limit.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale trips are set at 5/125 of full scale.

The rod block which occurs when the IRM detectors are not fully inserted in the core for the refuel and startup/hot standby position of the mode switch has been provided to assure that these detectors are in the core during reactor startup. This, therefore, assures that these instruments are in proper position to provide protection during reactor startup. The IRM's primarily provide protection against local reactivity effects in the source and intermediate neutron range.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a back-up to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip

3.2 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

settings given in the specification are adequate to assure the above criteria are met. (Ref. Section 6.2.6.3 SAR.) The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a fifteen minute delay before the air ejector off-gas isolation valve is closed. This delay is accounted for by the 30-minute holdup time of the off-gas before it is released to the stack.

Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct and on the refueling floor. The trip logic is a 1 out of 2 for each set and each set can initiate a trip independent of the other set. Any upscale trip will cause the desired action. Trip settings of 11 mr/hr for the monitors in the ventilation duct are based upon initiating normal ventilation isolation and standby gas treatment system operation to limit the dose rate at the nearest site boundary to less than the dose rate allowed by 10CFR20. Trip settings of 100 mr/hr for the monitors on the refueling floor 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 stack but that all the activity is processed by the standby gas treatment system.

4.2 SURVEILLANCE REQUIREMENT BASES

The instrumentation listed in Table 4.2.1 will be functionally tested and calibrated at regularly scheduled intervals. Although this instrumentation is not generally considered to be as important to plant safety as the Reactor Protection System, the same design reliability goal of 0.99999 is generally applied for all applications of (1 out of 2) X (2) logic. Therefore, on-off sensors are tested once/3 months, and bi-stable trips associated with analog sensors and amplifiers are tested once/week.

Those instruments which, when tripped, result in a rod block have their contacts arranged in a 1 out of n logic, and all are capable of being bypassed. For such a tripping arrangement with bypass capability provided, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (See Note 7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = (2t/r)^{1/2}$$

Where:

i = optimum interval between tests
t = the time the trip contacts are disabled from performing their function while the test is in progress
r = the expected failure rate of the relays

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 minutes to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10^{-6} failures per hour. Using this data and the above operation, the optimum test interval is:

$$i = \left(\frac{2(0.5)}{10^{-6}}\right)^{1/2} = 1 \times 10^{3} \text{ hours}$$

$$= \text{approximately 40 days}$$

For additional margin a test interval of once per month will be used initially.

Note:
(7) UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, Benjamin Epstein, Albert Shiff, July 16, 1968, page 10, Equation (24), Lawrence Radiation Laboratory.

4.2 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The sensors and electronic apparatus have not been included here as these are analog devices with readouts in the control room and the sensors and electronic apparatus can be checked by comparison with other like instruments. The checks which are made on a daily basis are adequate to assure operability of the sensors and electronic apparatus, and the test interval given above provides for optimum testing of the relay circuits.

The above calculated test interval optimizes each individual channel, considering it to be independent of all others. As an example, assume that there are two channels with an individual technician assigned to each. Each technician tests his channel at the optimum frequency, but the two technicians are not allowed to communicate so that one can advise the other that his channel is under test. Under these conditions, it is possible for both channels to be under test simultaneously. Now, assume that the technicians are required to communicate and that two channels are never tested at the same time.

Forbidding simultaneous testing improves the availability of the system over that which would be achieved by testing each channel independently. These one out of n trip systems will be tested one at a time in order to take advantage of this inherent improvement in availability.

Optimizing each channel independently may not truly optimize the system considering the overall rules of system operation. However, true system optimization is a complex problem. The optimums are broad, not sharp, and optimizing the individual channels is generally adequate for the system.

The formula given above minimizes the unavailability of a single channel which must be bypassed during testing. The minimization of the unavailability is illustrated by curve No. 1 of Figure 4.2.2 which assumes that a channel has a failure rate of 0.1×10^{-6}/hour and that 0.5 hours is required to test it. The unavailability is a minimum at a test interval i, of 3.16×10^{3} hours.

If two similar channels are used in a 1 out of 2 configuration, the test interval for minimum unavailability changes as a function of the rules for testing. The simplest case is to test each one independent of the other. In this case, there is assumed to be a finite probability that both may be bypassed at one time. This case is shown by Curve No. 2. Note that the unavailability is lower as expected for a redundant system and the minimum occurs at

4.2 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

the same test interval. Thus, if the two channels are tested independently, the equation above yields the test interval for minimum unavailability.

A more usual case is that the testing is not done independently. If both channels are bypassed and tested at the same time, the result is shown in Curve No. 3. Note that the minimum occurs at about 40,000 hours, much longer than for cases 1 and 2. Also, the minimum is not nearly as low as Case 2 which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed, tested and restored, and then immediately following the second channel be bypassed, tested, and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1 out of n system may be treated the same as a single channel in terms of choosing a test interval; and

2. More than one channel should not be bypassed for testing at any one time.

The radiation monitors in the ventilation duct and on the refueling floor which initiate building isolation and standby gas treatment operation are arranged in two 1 out of 2 logic systems. The bases given above for the rod blocks applies here also and were used to arrive at the functional testing frequency.

Based on experience at Dresden Unit 1 with instruments of similar design, a testing interval of once every three months has been found to be adequate.

The automatic pressure relief instrumentation can be considered to be a 1 out of 2 logic system and the discussion above applies also.

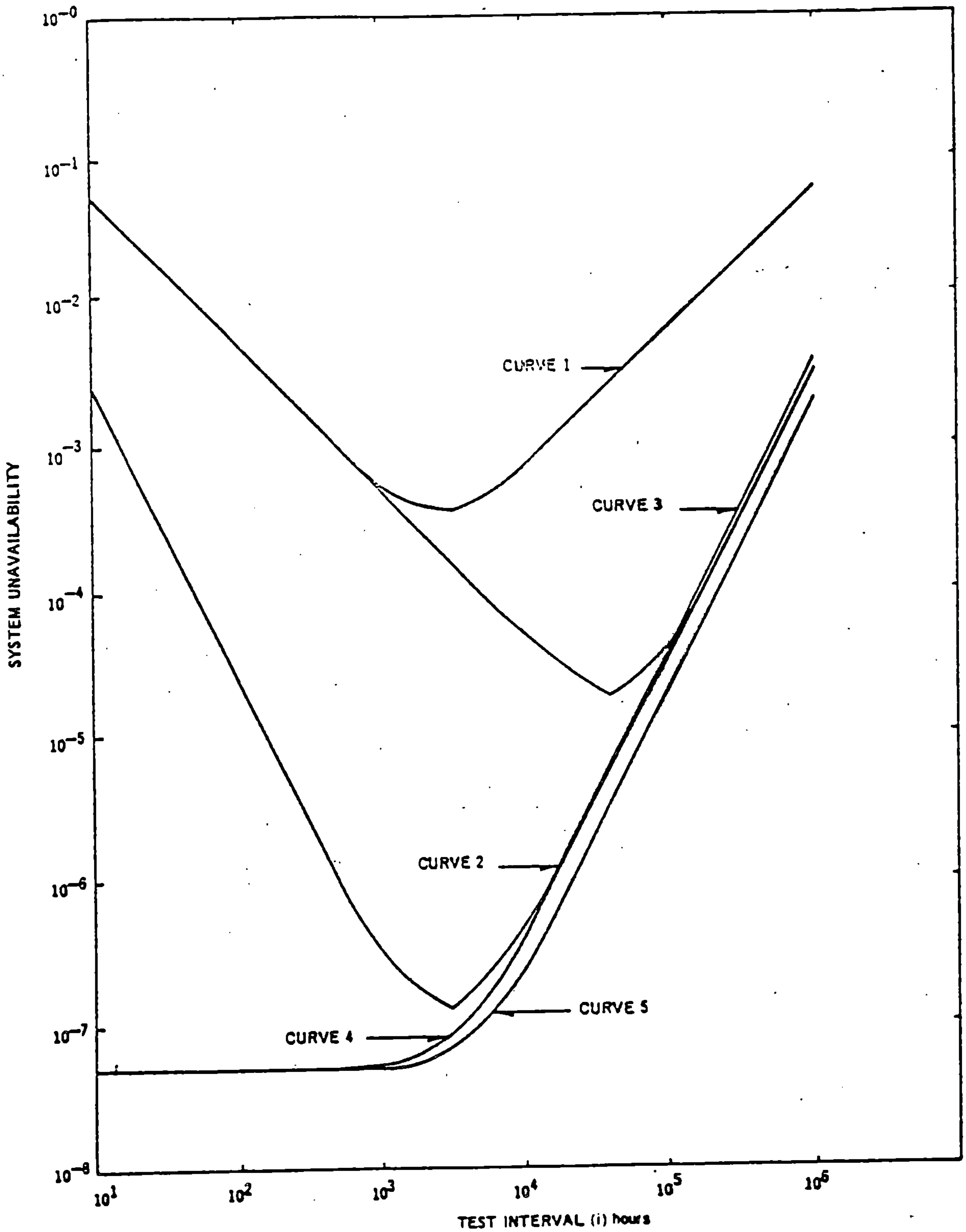


Figure 4.2.2.

TEST INTERVAL VS. SYSTEM UNAVAILABILITY

3.3 LIMITING CONDITION FOR OPERATION

REACTIVITY CONTROL

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

   The core loading shall be limited to that which can be made subcritical in the most reactive condition during the operating cycle with the strongest operable control rod in its full-out position and all other operable rods fully inserted.

4.3 SURVEILLANCE REQUIREMENT

REACTIVITY CONTROL

Applicability:

Applies to the surveillance requirements of the control rod system.

Objective:

To verify the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

   Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.25 percent delta k that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. Reactivity margin - inoperable control rods

   a. Control rod drives 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 valves for inoperable control rods shall be disarmed electrically and the

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

2. Reactivity margin - inoperable control rods

   Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week. This test shall be performed at least once per 24 hours 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.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

control rods shall be in such positions that Specification 3.3.A.1 is met.

c. Control rod drives which are fully inserted and electrically disarmed shall not be considered inoperable.

d. Control rods with scram times greater than those permitted by Specification 3.3.C are inoperable, but if they can be moved with control rod drive pressure, they need not be disarmed electrically if Specification 3.3.A.1 is met for each position of these rods.

e. During reactor power operation, the number of inoperable control rods shall not exceed eight.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

B. Control Rods

1. All control rods shall be coupled to their drive mechanisms when the mode switch is in "Startup" or "Run". With a control rod not coupled to its associated drive mechanism, operation may continue provided:
   a. Below 20% power, the rod shall be declared inoperable, full inserted, and

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

3. The scram discharge volume vent and drain valves shall be verified open at least once per 31 days. These valves may be closed intermittently for testing under administrative control and at least once per 92 days, each valve shall be cycled through at least one complete cycle of full travel. At least once each Refueling Outage, the scram discharge volume vent and drain valves will be demonstrated to:

   a. Close within 30 seconds after receipt of a signal for control rods to scram, and

   b. Open when the scram signal is reset.

B. Control Rods

1. Coupling Integrity

   a. The coupling integrity of each control rod shall be demonstrated by

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

the directional control valves electrically disarmed until recoupling can be attempted at all-rods-in or at power levels above 20 percent power.

b. Above 20% power, recoupling is being attempted

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

withdrawing each control rod to the fully withdrawn position and verifying that the rod does not go to the overtravel position;

i. Prior to reactor criticality after completing alteration of the reactor core,

ii. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

iii. For specifically affected invidiaual control rods following maintenance on or modification to the control rod or rod drive system which could affect the rod drive coupling integrity.

b. Normal operating practice is to observe the

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

in accordance with an established procedure or the rod shall be declared inoperable, fully inserted and the directional control valves electrically disarmed.

2. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

expected response of the nuclear instrumentation to verify that the control rod is following its drive each time that control rod is withdrawn. For control rod drives that have experienced uncoupling and no response is discernable on the nuclear instrumentation, the response should be verified when the reactor is operating at power levels above 20 percent.

2. The control rod drive drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. a. Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would be such that the rod drop accident design limit of 280 cal/gm is not exceeded.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

3. a. To consider the rod worth minimizer operable, the following steps must be performed:

   i. The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct.

   ii. The rod worth minimizer computer on-line diagnostic test shall be successfully completed.

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

   iv. The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

b. Whenever the reactor is in the startup or run mode below 20% rated thermal power, the Rod Worth Minimizer shall be operable. A second operator or qualified technical person may be used as a substitute for an inoperable Rod Worth Minimizer which fails after withdrawal of at least 12 control rods to the fully withdrawn position. The Rod Worth Minimizer may also be bypassed for low power physics testing to demonstrate the shutdown margin requirements of specifications 3.3.A.1 if a nuclear engineer is present and verifies the step-by-step rod movements of the test procedure.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

b. If the rod worth minimizer is inoperable while the reactor is in the startup or run mode below 20% rated thermal power and a second independent operator or engineer is being used, he shall verify that all rod positions are correct prior to commencing withdrawal of each rod group.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

5. During operating with limiting control rod patterns, as determined by the nuclear engineer, 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 the MCPR will remain above the MCPR fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of any single operable control rod.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

4. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels have been 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) and daily thereafter.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization 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	0.900
50	2.00
90	3.50

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	3.800

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

C. Scram Insertion Times

1. After each refueling outage, prior to operation greater than 30 percent of rated thermal power, all control rods shall be subject to scram-time tests from the fully withdrawn position with reactor pressure above 800 psig. If the control rods are tested individually, their hydraulic control units shall be isolated from the control rod drive pumps.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator,
2. Directional control valve electrically disarmed while in a non-fully inserted position.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

2. At 16 week intervals, at least 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% or more of the control rod drives have been tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. Following completion of each set of scram testing as described above, the results will be compared against the average scram speed distribution used in the transient analysis to verify the applicability of the current MCPR Operating Limit. Refer to Specification 3.5.K.

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. Scram insertion greater than maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator and the rod block associated with that inoperable accumulator may be bypassed.

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% delta K. If this limit is exceeded, the reactor will be shutdown until the cause has been determined and corrective actions have been taken if such actions are appropriate. In accordance with Specification 6.6, the NRC shall be notified of this reportable occurrence within 24 hours.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

E. Reactivity Anomalies

During the startup test program and startups 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 equivalent full power month.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

G. Economic Generation Control System

Operation of the unit with the Economic Generation Control system with automatic flow control shall be permissible only in the range of 65-100% of rated core flow, with reactor power above 20%.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

F. (N/A)

G. Automatic Generation Control System

Weekly, the range set into the Economic Generation Control System shall be recorded.

3.3 LIMITING CONDITION FOR OPERATION BASES

A. Reactivity Limitations

1. Reactivity margin--core loading

The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. The generalized form is that the reactivity of the core loading will be limited so the core can be made sub-critical by at least R + 0.25% delta k in the most reactive condition during the operating cycle, with the strongest control rod fully withdrawn and all others fully inserted. The value of R in % delta k is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check; i.e., the initial loading. R must be a positive quantity or zero. A core which contains temporary control or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum and then decreases thereafter. See Figure 3.3.2 of the SAR for such a curve.

The value of R is the difference between the calculated core reactivity at the beginning of the operating cycle and the calculated value of core reactivity any time later in the cycle where it would be greater than at the beginning. For the first fuel cycle, R was calculated to be not greater than 0.10% delta k. A new value of R must be determined for each fuel cycle.

The 0.25% delta k in the expression R + 0.25% delta k is provided as a finite, demonstrable, sub-criticality margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to insert at least R + 0.25% delta k in reactivity. Observation of sub-criticality in this condition assures sub-criticality with not only the strongest rod fully withdrawn but at least a R + 0.25% delta k margin beyond this.

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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 available to the operator. The number of rods permitted to be inoperable could be many more than the eight allowed by the Specification, particularly late in the operation cycle; however, the occurrence of more than eight could be indicative of a generic control rod drive problem and the reactor will be shutdown. Also, 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 BWR's. 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.

* To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

3. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume. This ensures that water accumulation does not occur which would cause an early termination of control rod movement during a full core scram. These specifications provide for the periodic verification that the valves are open and for testing of these valves under reactor scram conditions during each Refueling Outage.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in Reference XN-NF-80-19, Vol. 1, 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 would provide cause for suspecting a rod to be uncoupled and stuck. Restricting recoupling verifications to power levels above 20% provides assurance that a rod drop during a recoupling verification would not result in a rod drop accident.

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 Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. 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 sub-critical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod sequences which are

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

withdrawn could not be worth enough to cause the rod drop accident design limit of 280 cal/gm to be exceeded if they were to drop out of the core in the manner defined for the Rod Drop Accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM or a second qualified station employee. These sequences are developed to limit reactivity worths of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content, 425 cal/gm, at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference XN-NF-80-19, Volume 1.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2 and 14.2.1.4 of the Safety Analysis Report. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident.

Parametric Control Rod Drop Accident analyses have shown that for wide ranges of key reactor parameters (which envelope the operating ranges of these variables), the fuel enthalpy rise during a postulated control rod drop accident remains considerably lower than the 280 cal/gm limit. For each operating cycle, cycle-specific parameters such as maximum control rod worth, Doppler coefficient effective delayed neutron fraction and maximum four-bundle local peaking factor are compared with the results of the parametric analyses to determine the peak fuel rod enthalpy rise. This value is then compared against the Technical Specification limit of 280 cal/gm to demonstrate compliance for each operating cycle. If cycle specific values of the above parameters are outside the range assumed in the parametric analyses, an extension of the analysis or a cycle specific analysis may be required. Conservatism present in the analysis, results of the parametric studies, and a detailed description of the methodology for performing the Control Rod Drop Accident analysis are provided in Reference XN-NF-80-19, Volume 1 (Supplements 1 and 2).

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The Rod Worth Minimizer provides 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. (Ref. Section 7.9 SAR.) It serves as a backup to procedural control of control rod worth. In the event that the Rod Worth Minimizer is out of service, when required, a licensed operator or other qualified technical employee can manually fulfill the control rod pattern conformance functions of the Rod Worth Minimizer. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 20% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is required to be operating during a startup for the withdrawal of a significant number of control rods for any startup after June 1, 1974.

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. This is needed for knowledgeable and efficient reactor startup at low neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement 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 from 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.

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

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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 rods according to a written sequence. 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. Amendments 17/18 and 19/20 present the results of an evaluation of a rod block monitor failure. These amendments show that during reactor operation with certain limiting control rod pattern, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPRS less than the MCPR fuel cladding integrity safety limit. 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 Nuclear 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.

C. Scram Insertion Times

The performance of the control rod insertion system is analyzed to verify the system's ability to bring the reactor subcritical at a rate fast enough to prevent violation of the MCPR Fuel Cladding Integrity Safety Limit and thereby avoid fuel damage. The analyses demonstrate that if the reactor is operated within the limitations set in Specification 3.5.K. the negative reactivity insertion rates associated with the observed scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR safety limit.

In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

analyses, and is also included in the allowable scram insertion times specified in Specification 3.3.C. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Fifty percent of the control rod drives in the reactor are tested every sixteen weeks to verify adequate performance. Observed plant data were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly. If an individual test or group of tests should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken (as required by Specification 3.5.K) unless it can be shown that the number of individual drives falling outside the statistical population defining the nominal performance is less than the allowable number of inoperable control rod drives. If the number of statistically aberrant drives falls within this limitation, operation will be allowed to continue without redetermination of thermal margin requirements provided the identified aberrant drives are fully inserted into the core and deenergized in the manner of an inoperable rod drive.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no reassessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 second is greater than 0.999 for a normal distribution.

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

D. Control Rod Accumulators

The basis for this specification was not described in the SAR and, therefore, is presented in its entirety. Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a k_{eff} less than 1.0 -- other repeating rod sequences with more rods withdrawn resulted in k_{eff} greater than 1.0. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle excess operating 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 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% delta k. Deviations in core reactivity greater than 1% delta k 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.

F. (N/A)

G. Economic Generation Control System

Operation of the facility with the Economic Generation Control System with automatic flow control is limited to the range of

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

65-100% of rated core flow. In this flow range and with reactor power above 20% the reactor can safely tolerate a rate of change of load of 8 MW(e)/sec. (Reference FSAR Amendment 9-Unit 2, 10-Unit 3). Limits within the Economic Generation Control System and Reactor Flow Control System preclude rates of change greater than approximately 4 MWe/sec.

When the Economic Generation Control System is in operation, this fact will be indicated on the main control room console. The results of initial testing will be provided to the NRC at the onset of routine operation with the Economic Generation Control System.

4.3 SURVEILLANCE REQUIREMENT BASES

None

3.4 LIMITING CONDITION FOR OPERATION

STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the standby liquid control system.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. Normal Operation

During periods when fuel is in the reactor the standby liquid control system shall be operable except when the reactor is in the Cold Shutdown Condition and all control rods are fully inserted and Specification 3.3.A is met or as specified in 3.4.B below.

4.4 SURVEILLANCE REQUIREMENT

STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirements for the standby liquid control system.

Objective:

To verify the operability of the standby liquid control system.

Specification:

A. Normal Operation

The operability of the standby liquid control system shall be verified by performance of the following tests:

1. At least once per month -

   Demineralized water shall be recycled to the test tank. Pump minimum flow rate of 39 gpm shall be verified against a system head of 1275 psig.

2. At least once during each operating cycle

   a. Actuate one of the two standby liquid control systems using the normal actuation switch and pump demineralized water into the reactor

3.4 LIMITING CONDITION FOR OPERATION (Cont'd.)

B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A shall be considered fulfilled, and continued operation permitted provided that the component is returned to an operable condition within 7 days.

4.4 SURVEILLANCE REQUIREMENT (Cont'd.)

vessel. Pump minimum flow rate shall be verified against a previous test at the same reactor vessel pressure. The replacement charges will be selected from a batch from which at least one charge has been successfully test fired and which will not exceed five years life when their use is terminated. Both systems shall be tested and inspected, including each explosive actuated valve, in the course of two operating cycles.

b. Test that the setting of the system pressure relief valves is between 1400 and 1490 psig.

B. Surveillance with Inoperable Components

When a component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter.

3.4 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. The liquid poison tank shall contain a boron bearing solution that satisfies the volume-concentration requirements of Figure 3.4.1 and at all times when the standby liquid control system is required to be operable and the solution temperature including that in the pump suction piping shall not be less than the temperature presented in Figure 3.4.2.

D. If specification 3.4.A through C are not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.4 SURVEILLANCE REQUIREMENT (Cont'd.)

C. The availability of the proper boron bearing solution shall be verified by performance of the following tests:

1. At least once per month - Boron concentration shall be determined. In addition, the boron concentration shall be determined any time water or boron are added or if the solution temperature drops below the limits specified by Figure 3.4.2.

2. At least once per day - Solution volume shall be checked.

3. At least once per day - The solution temperature shall be checked.

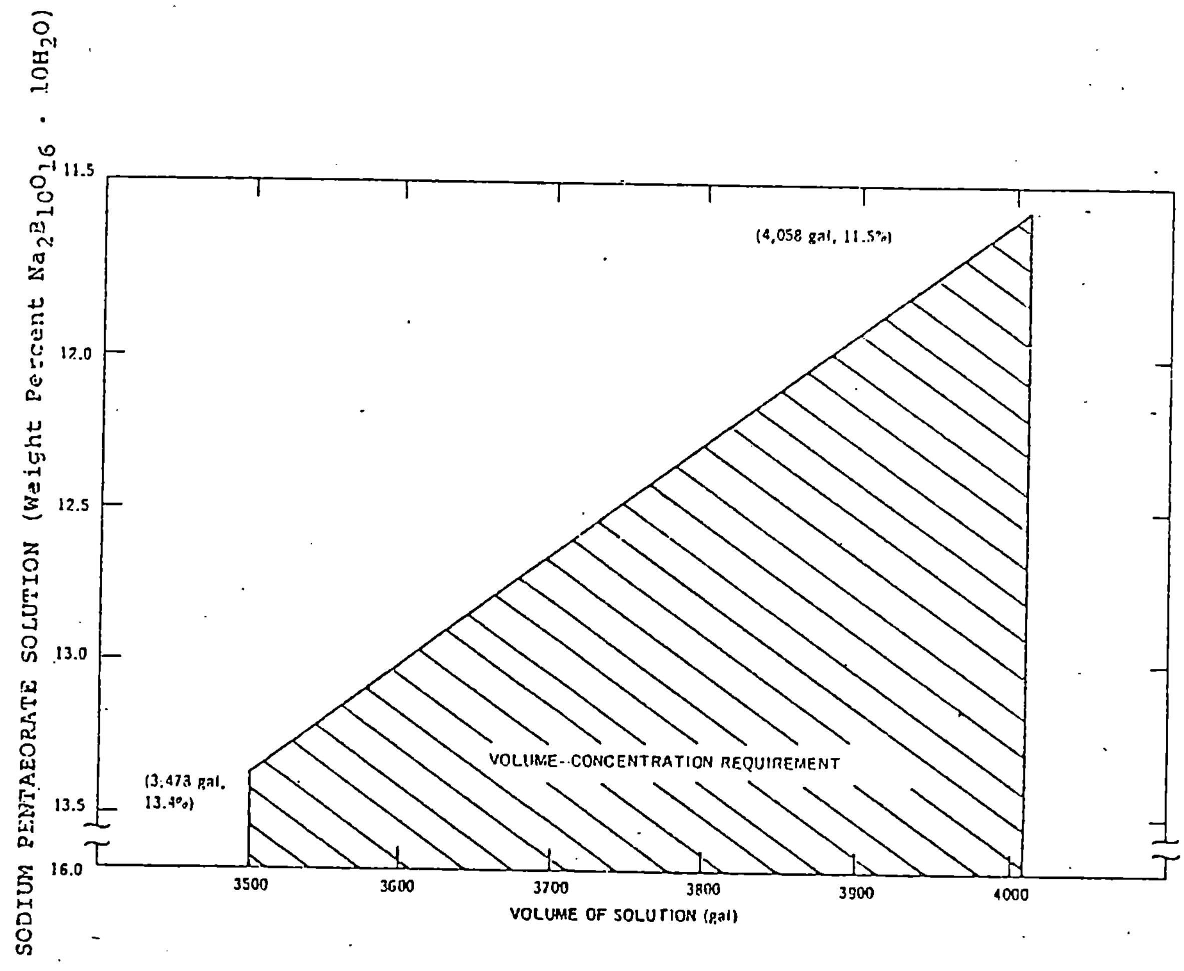


Figure 3.4.1 Standby Liquid Control Solution Requirements

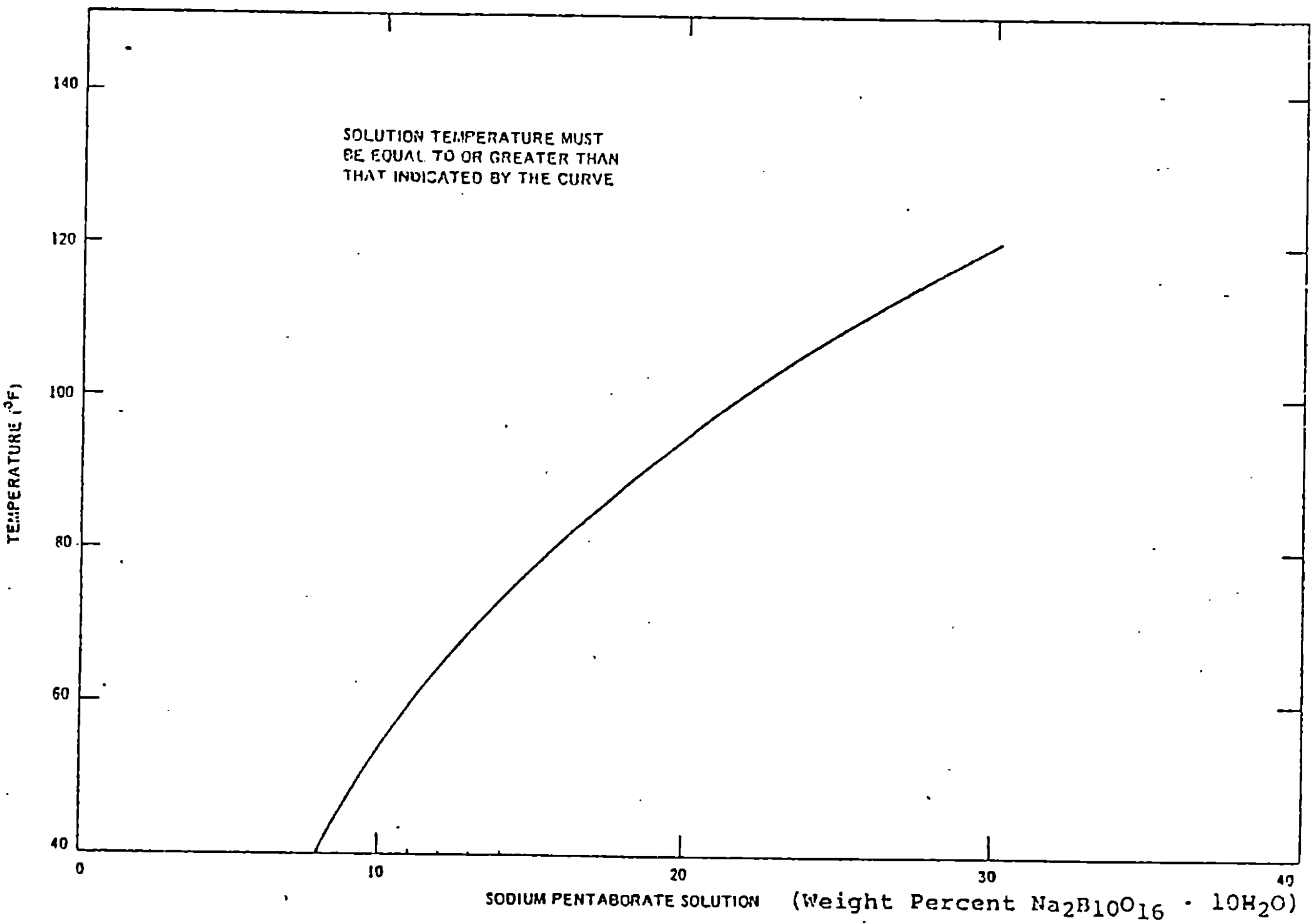


Figure 3.4.2 Sodium Pentaborate Solution Temperature Requirements

3.4 LIMITING CONDITION FOR OPERATION BASES

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of no less than 600 ppm of boron in the reactor core in less than 100 minutes. 600 ppm boron concentration in the reactor core is required to bring the reactor from full power to a 3% delta k or more subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional margin (25%) for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3478 gallons of solution having a 13.4% sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (100 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required pumping rate of 39 gallons per minute, the maximum storage volume of the boron solution is established as 4,059 gallons (158 gallons are contained below the pump suction and, therefore, cannot be inserted).

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Experience with pump operability indicates that monthly testing is adequate to detect if failures have occurred.

Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of less than once during each operating cycle unnecessary. A test of one installed explosive charge is made at least once during each operating cycle to assure that the charges have not deteriorated, the actuation circuit is functioning properly, the valve functions properly, and no flow blockages exist. The replacement charge will be selected from a batch for which there has been a successful test firing. Recommendations of the vendor shall be followed in maintaining a five-year life of the explosive charges. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

3.4 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The relief valves in the standby liquid control system protect the system piping and positive displacement pumps which are nominally designed for 1500 psig protection from over-pressure. The pressure relief valves discharge back to the standby liquid control solution tank.

B. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the remaining system will perform its intended function and that the reliability of the system is good is obtained by demonstrating operation of the pump in the operable circuit at least once daily.

C. The solution saturation temperature of 13% sodium pentaborate, by weight, is 59°F. To guard against boron precipitation, the solution including that in the pump suction piping is kept at least 10°F above the saturation temperature by a tank heater and by heat tracing in the pump suction piping. The 10°F margin is included in Figure 3.3.1. Temperature and liquid level alarms for the system are annunciated in the control room.

Pump operability is checked on a frequency to assure a high reliability of operation of the system should it ever be required.

Once the solution has been made up, boron concentration will not vary unless more boron or more water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

4.4 SURVEILLANCE REQUIREMENT BASES

None

3.5 LIMITING CONDITION FOR OPERATION

CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the emergency cooling subsystems.

Objective:

To assure adequate cooling capability for heat removal in the event of a loss of coolant accident or isolation from the normal reactor heat sink.

Specification:

A. Core Spray and LPCI Subsystems

1. Except as specified in 3.5.A.2, 3.5.A.3, and 3.5.F.3 below, both core spray subsystems shall be operable whenever irradiated fuel is in the reactor vessel.

4.5 SURVEILLANCE REQUIREMENT

CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to periodic testing of the emergency cooling subsystems.

Objective:

To verify the operability of the emergency cooling subsystems.

Specification:

A. Surveillance of the Core Spray and LPCI Subsystems shall be performed as follows:

1. Core Spray Subsystem Testing:

Item	Frequency
a. Simulated Automatic Actuation Test	Each Refueling Outage
b. Flow Rate Test Core spray pumps shall deliver at least 4500 gpm against a system head corresponding	After pump maintenance and every 3 months

3.5 LIMITING CONDITION FOR OPERATION (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, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the other core spray subsystem and the LPCI subsystem and the diesel generators required for operation

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

to a reactor vessel pressure of 90 psig

c. Pump Operability	Once/month
d. Motor Operated Valve	Once/month
e. Core Spray header delta p instrumentation:	
check	Once/day
calibrate	Once/3 months
test	Once/3 months
f. Logic System Functional Test	Each Refueling Outage

2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem and the LPCI subsystem and the diesel generators required for operation of such components if no external source of power were available shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

of such components if no external source of power were available shall be operable.

3. Except as specified in 3.5.A.4, 3.5.A.5 and 3.5.F.3 below, the LPCI subsystem shall be operable whenever irradiated fuel is in the reactor vessel.

4. From and after the date that one of the LPCI pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days the remaining active components of the LPCI and containment cooling subsystem and all active components of both core spray subsystems and the diesel generators required for operation of such components if no external source of power were available shall be operable.

5. From and after the date that the LPCI subsystem is made or found to be inoperable

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3. LPCI Subsystem Testing shall be as specified in 4.5.A.1.a, b, c, d, and f, except that three LPCI pumps shall deliver at least 14,500 gpm against a system head corresponding to a reactor vessel pressure of 20 psig.

4. When it is determined that one of the LPCI Pumps is inoperable, the remaining active components of the LPCI and containment cooling subsystem, both core spray subsystems and the diesel generators required for operation of such components if no external source of power were available shall be demonstrated to be operable immediately and the operable LPCI pumps daily thereafter.

5. When it is determined that the LPCI subsystem is inoperable, both core spray subsystems,

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

for any reason, reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided that during such seven days all active components of both core spray subsystems, the containment cooling subsystem (including 2 LPCI pumps) and the diesel generators required for operation of such components if no external source of power were available shall be operable.

6. Containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a maximum of one drywell spray loop may be inoperable for thirty days when the reactor water temperature is greater than 212°F.

7. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours. Subsequently, the reactor may be placed in Refuel, for post maintenance testing of control rod drives only, provided no work is being performed which has the potential to drain the reactor-vessel.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

the containment cooling subsystem, and the diesel generators required for operation of such components if no external source of power were available shall be demonstrated to be operable immediately and daily thereafter.

6. During each five year period an air test shall be performed on the drywell spray headers and nozzles.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

B. Containment Cooling Subsystem

1. Except as specified in 3.5.B.2, 3.5.B.3, and 3.5.F.3 below, both containment cooling subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that one of the containment cooling service water subsystem pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days all other active components of the containment cooling subsystem are operable.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

B. Surveillance of the Containment Cooling Subsystem shall be performed as follows:

1. Containment Cooling Service Water Subsystem Testing:

Item	Frequency
a. Pump & Valve Operability	Once/3 months
b. Flow Rate Test. Each containment cooling water pump shall deliver at least 3500 gpm against a pressure of 180 psig.	After pump maintenance and every 3 months

2. When it is determined that one containment cooling service water pump is inoperable, the remaining components of that subsystem and the other containment cooling subsystem shall be demonstrated to be operable immediately and daily thereafter.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. From and after the date that one containment cooling subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that all active components of the other containment cooling subsystem, both core spray subsystems and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

4. If the requirements of 3.5.B cannot be met an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

C. HPCI Subsystem

1. Except as specified in 3.5.C.2 below, the HPCI subsystem shall be operable whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When one containment cooling subsystem becomes inoperable, the operable subsystem and the diesel generators required for operation of such components shall be demonstrated to be operable immediately and the operable containment cooling subsystem daily thereafter.

C. Surveillance of HPCI Subsystem shall be performed as follows:

1. HPCI Subsystem Testing shall be as specified in 4.5.A.1.a, b, c, d, and f, except that the HPCI pump shall deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of 1150 psig to 150 psig.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. From and after the date that the HPCI subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the Automatic Pressure Relief Subsystem, the core spray subsystems, LPCI subsystem, and isolation cooling system are operable.

3. If the requirements of 3.5.C cannot be met an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

2. When it is determined that HPCI subsystem is inoperable, the LPCI subsystem, both core spray subsystems, the automatic pressure relief subsystem, and the motor operated isolation valves and shell side make-up system for the isolation condenser system shall be demonstrated to be operable immediately. The motor operated isolation valves and shell side make-up system of the isolation condenser shall be demonstrated to be operable daily thereafter. Daily demonstration of the automatic pressure relief subsystem operability is not required provided that two feedwater pumps are operating at power levels above 300 MWe; and one feedwater pump is operating as normally required with one additional feedwater pump operable at power levels less than 300 MWe.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

D. Automatic Pressure Relief Subsystems

1. Except as specified in 3.5.D.2 and 3 below, the Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig and and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

D. Surveillance of the Automatic Pressure Relief Subsystem shall be performed as follows:

1. During each operating cycle the following shall be performed:

   a. A simulated automatic initiation which opens all pilot valves, and

   b. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure.

   c. A logic system functional test shall be performed each refueling outage.

2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

vessel, reactor operation is permissible only during the succeeding seven days unless repairs are made and provided that during such time the HPCI Subsystem is operable.

3. From and after the date that more than one of five relief valves of the automatic pressure relief subsystem made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 24 hours unless repairs are made and provided that during such time the HPCI Subsystem is operable.

4. If the requirements of 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

E. Isolation Condenser System

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When it is determined that more than one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately.

E. Surveillance of the Isolation Condenser System shall be performed as follows:

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. Whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel, the isolation condenser shall be operable except as specified in 3.5.F.2.

2. From and after the date that the isolation condenser system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such system is sooner made operable, provided that during such seven days all active components of the HPCI subsystem are operable.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

1. Isolation Condenser System Testing:

   a. The shell side water level and temperature shall be checked daily.

   b. Simulated automatic actuation and functional system testing shall be performed during each refueling outage or whenever major repairs are completed on the system.

   c. The system heat removal capability shall be determined once every five years.

   d. Calibrate vent line radiation monitors quarterly.

2. When it is determined that the isolation condenser system is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately and daily thereafter.

3.5 <u>LIMITING CONDITION FOR OPERATION (Cont'd.)</u>

3. If the requirements of 3.5.E cannot be met an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

F. Minimum Core and Containment Cooling System Availability

1. During any period when the unit or shared diesel generator is inoperable, continued reactor operation is permissible only during the succeeding <u>seven</u> days provided that all of the low pressure core cooling and containment cooling subsystems shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

4.5 <u>SURVEILLANCE REQUIREMENT (Cont'd.)</u>

F. Surveillance of Core and Containment Cooling System

1. When it is determined that either the unit or shared diesel generator is inoperable, all low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter.

2. Actions necessary to assure that the plant can be safely shut down and maintained in this condition in case of failure of the Dresden Dam shall be demonstrated to be adequate every third refueling outage. If this Specification has been complied with for Dresden Unit 3, it shall not be required for Dresden Unit 2.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. When irradiated fuel is in the reactor vessel and reactor is in the cold shutdown condition, all low pressure core and containment cooling subsystems may be inoperable provided no work is being done which has the potential for draining the reactor vessel.

4. When irradiated fuel is in the reactor vessel and the reactor is in the refuel condition, the torus may be drained completely and control rod drive maintenance performed provided that the spent fuel pool gates are open, the fuel pool water level is maintained above the low level alarm point, and the minimum total condensate storage reserve is maintained at 230,000 gallons, and provided that not more than one control rod drive housing is open at one time, the control rod drive housing is blanked following removal of the control rod drive, no work is being performed in the reactor vessel while the housing is open and a special flange is

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

available which can be used to blank an open housing in the event of a leak.

5. When irradiated fuel is in the reactor and the vessel head is removed, work that has the potential for draining the vessel may be performed with less than 112,000 ft^3 of water in the suppression pool, provided that: 1) the total volume of water in the suppression pool, dryer separator above the shield blocks, refueling cavity, and the fuel storage pool above the bottom of the fuel pool gate is greater than 112,000 ft^3; 2) the fuel storage pool gate is removed; 3) the low pressure coolant injection and core spray systems are operable; and 4) the automatic mode of the drywell sump pumps is disabled.

H. Maintenance of Filled Discharge Pipe

Whenever core spray, LPCI, or HPCI ECCS are required to be operable, the discharge piping from the pump discharge

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

H. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to, to assure that the discharge piping of the core spray,

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

of these systems to the last check valve shall be filled.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

LPCI, and HPCI are filled:

1. Every month prior to the testing of the LPCI and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow observed.

2. Following any period where the LPCI or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI system is lined up to take suction from the torus, the discharge piping of the HPCI shall be vented from the high point of the system and water flow observed on a monthly basis.

4. The pressure switches which monitor the LPCI and core spray system discharge lines to assure that they are full shall be functionally tested every month and calibrated every three months.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

I. Average Planar LHGR

During steady state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of all the rods in any fuel assembly, as a function of average planar exposure for G.E. fuel and average bundle exposure for Exxon fuel at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5-1. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. LOCAL LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly fabricated

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

I. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure for G.E. fuel and average bundle exposure for Exxon fuel shall be determined daily during reactor operation at greater than or equal to 25% rated thermal power.

J. Linear Heat Generation Rate (LHGR)

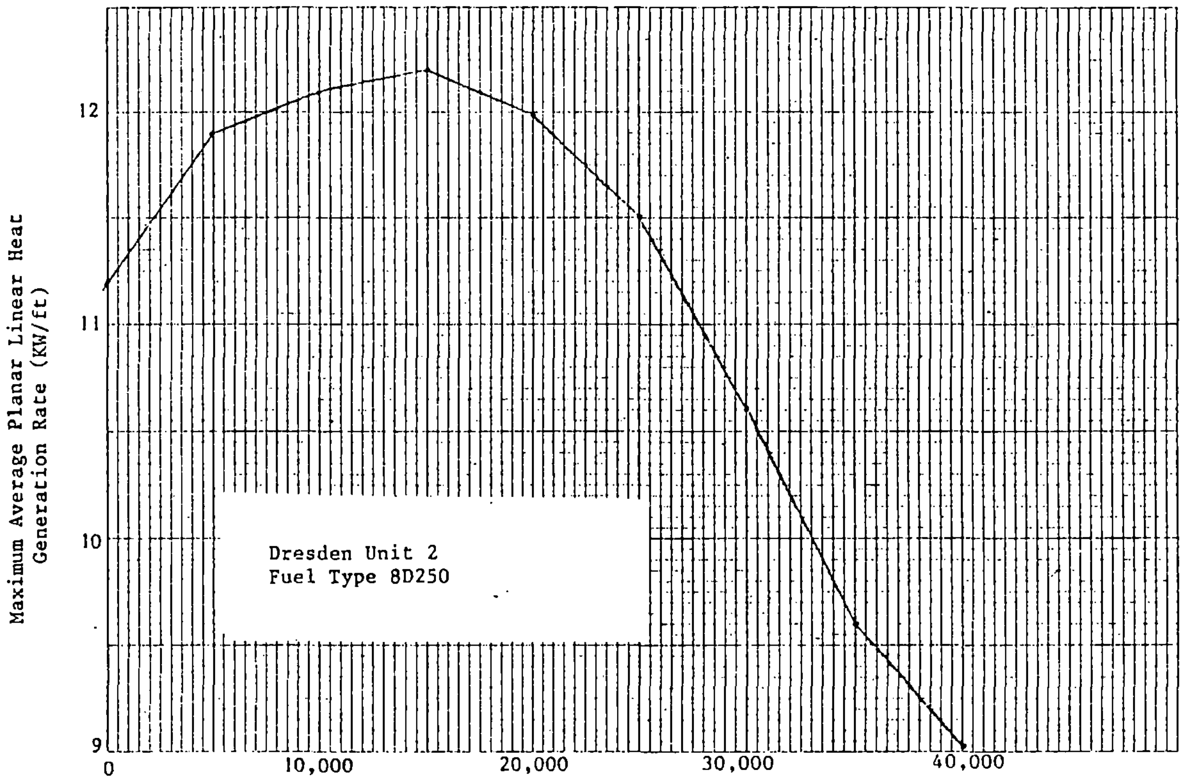
The LHGR shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.

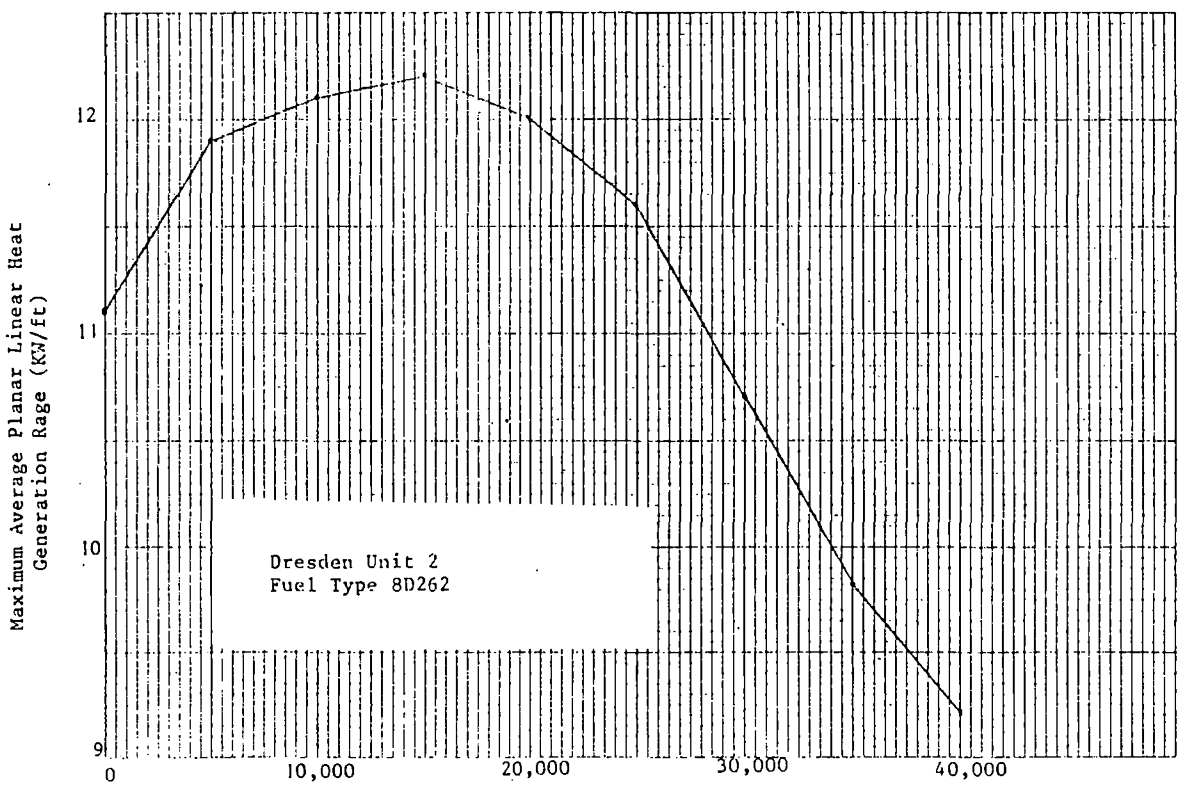
3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

by GE at any axial location shall not exceed the design value of 13.4 kw/ft.

If at any time during operation, it is determined by normal surveillance that the limiting value for LHGR for G.E. fuel is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

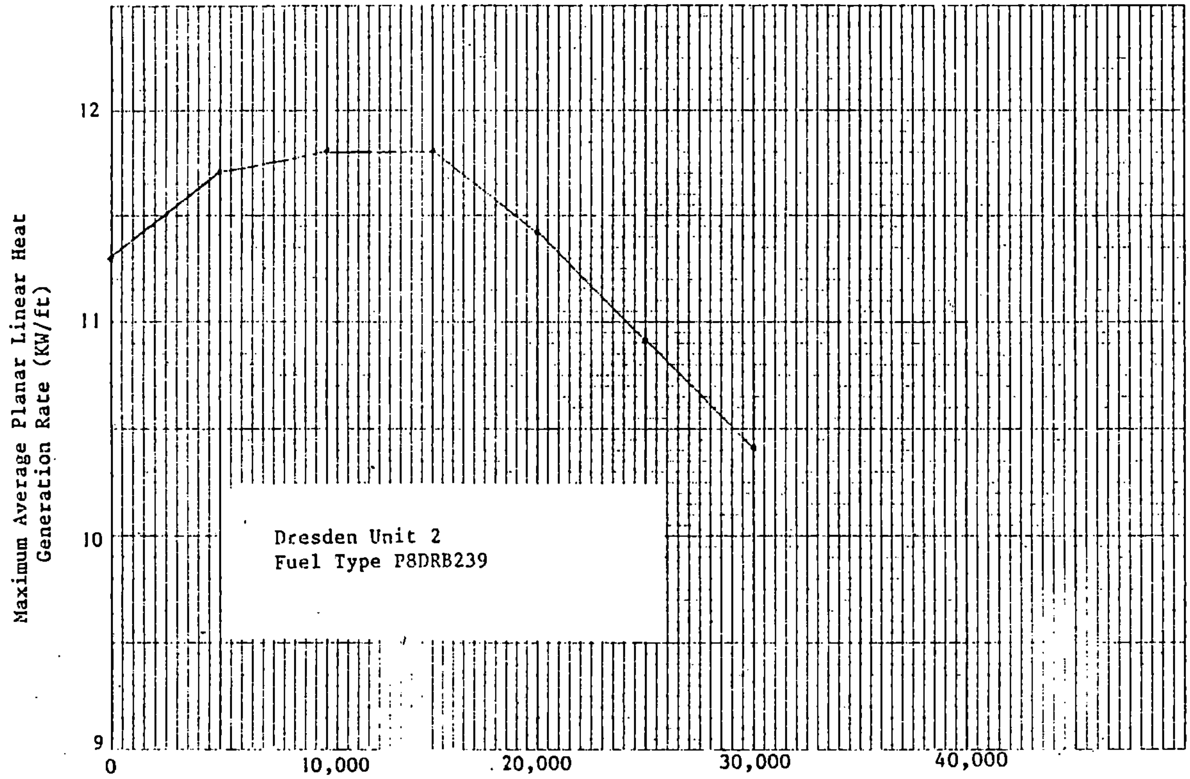
4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

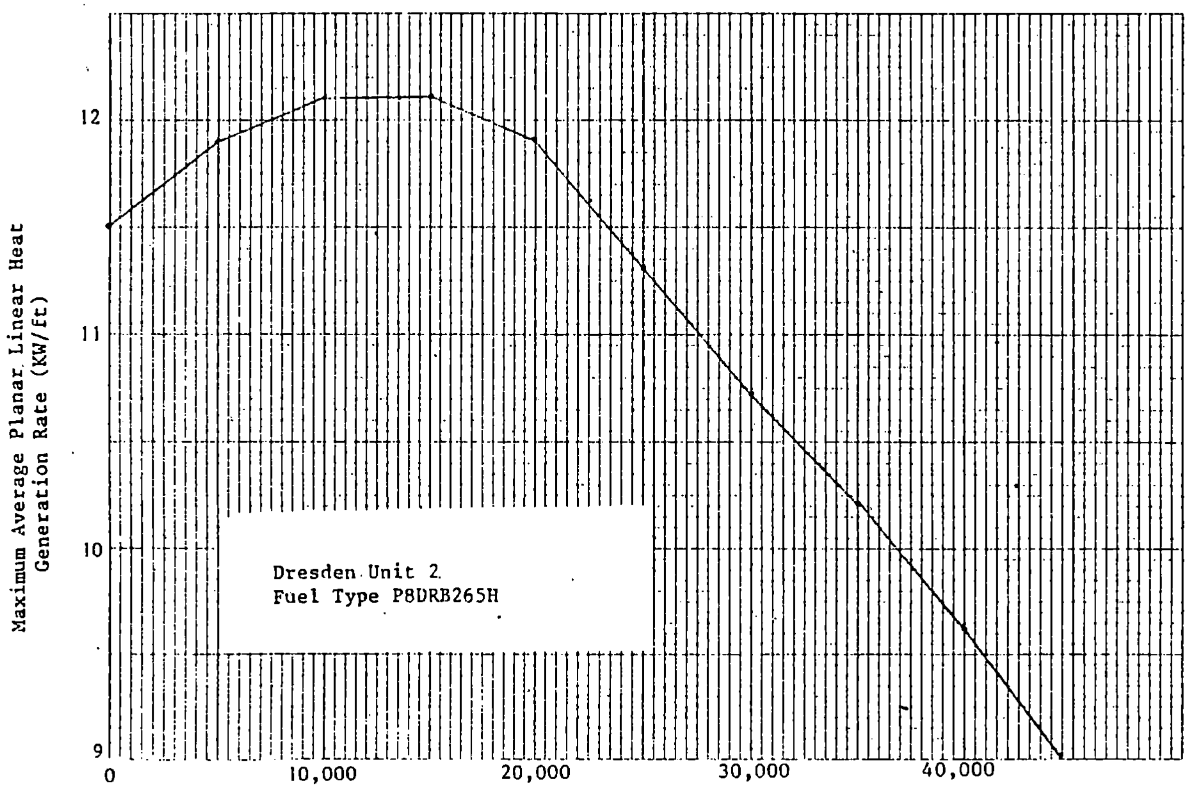




Average Planar Exposure (MWd/t)
MAPLHGR Versus Average Planar Exposure
Figure 3.5-1

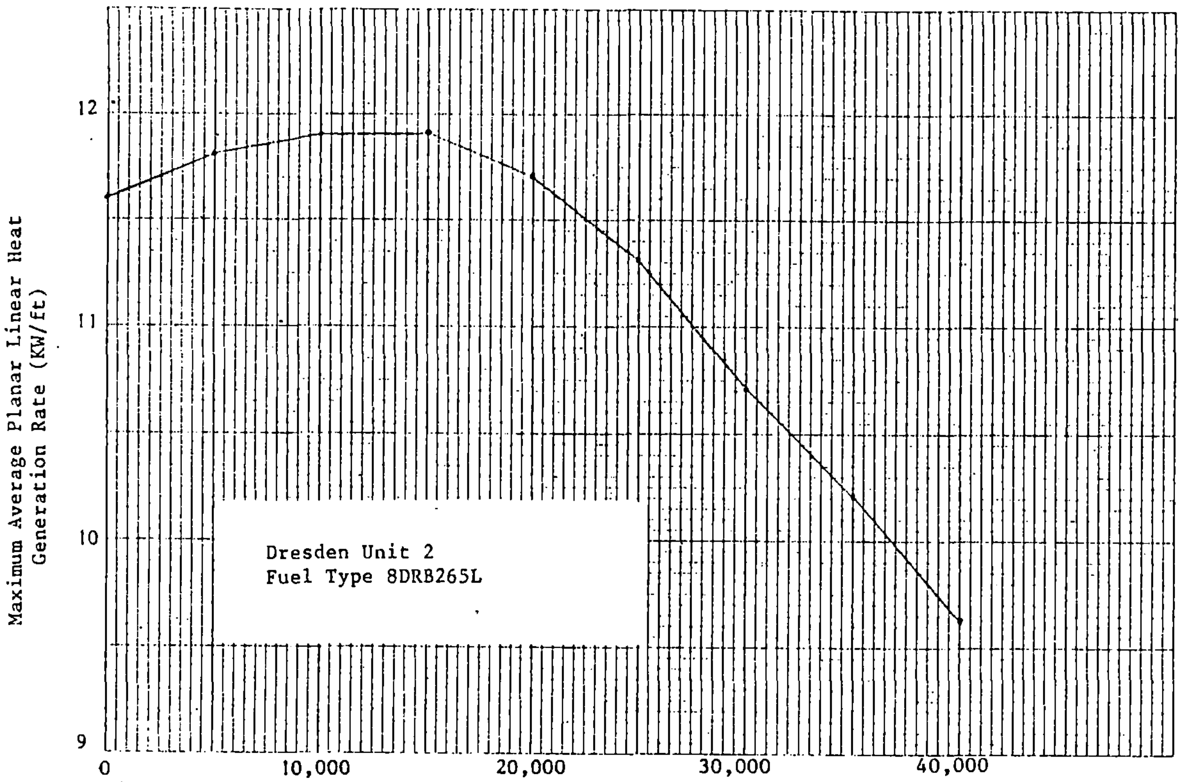
(Sheet 1 of 5)

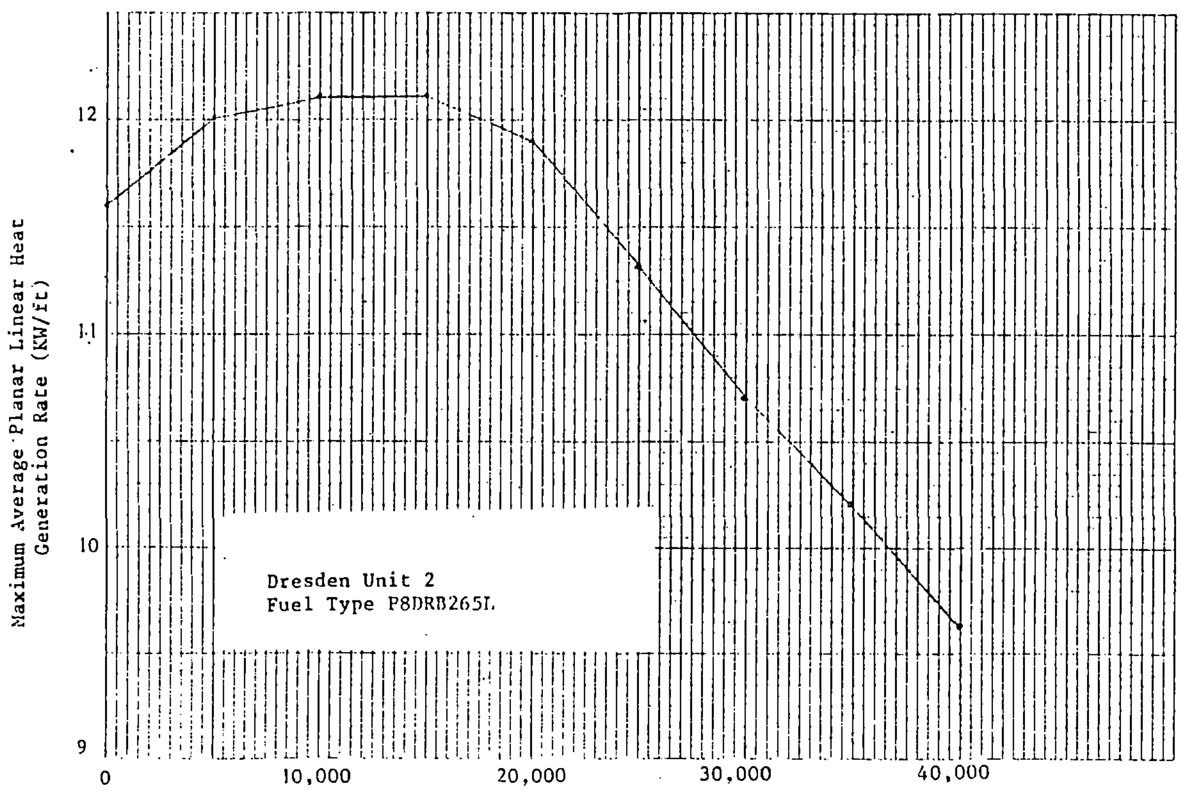




Average Planar Exposure (MWd/t)
MAPLHAR Versus Average Planar (Exposure)
Figure 3.5-1

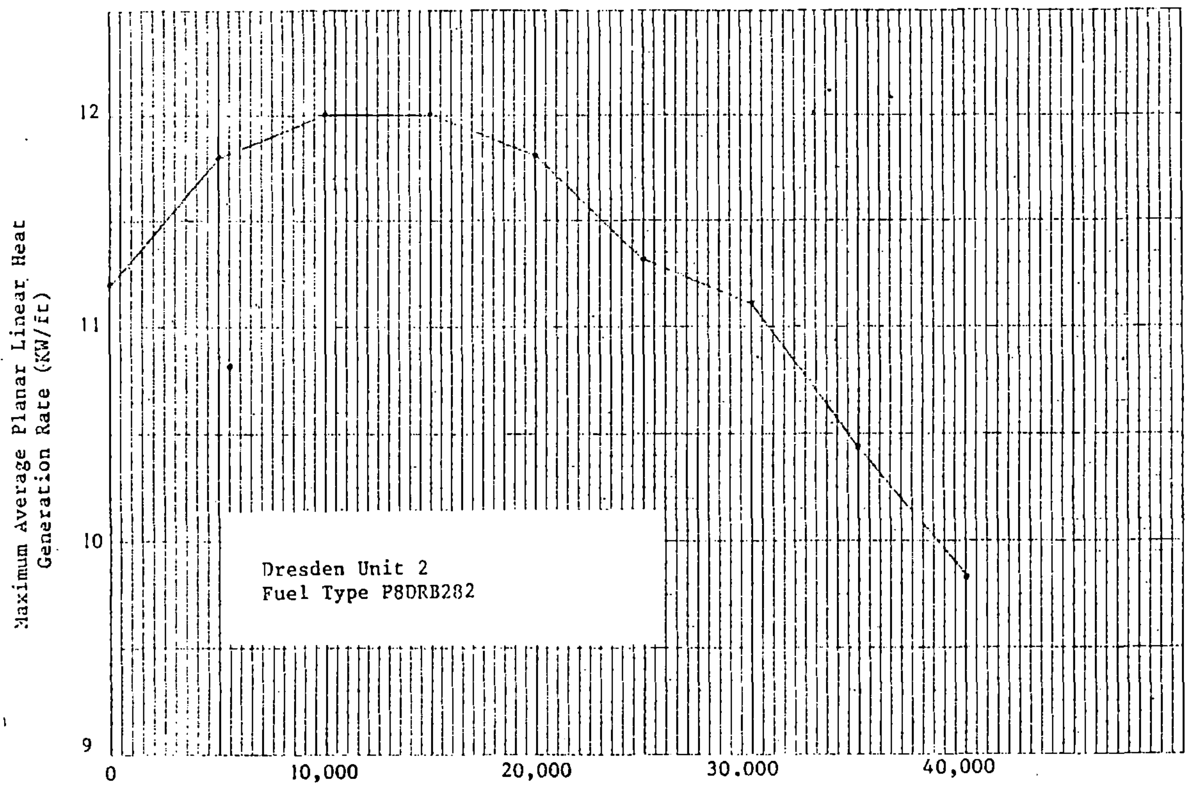
(Sheet 2 of 5)

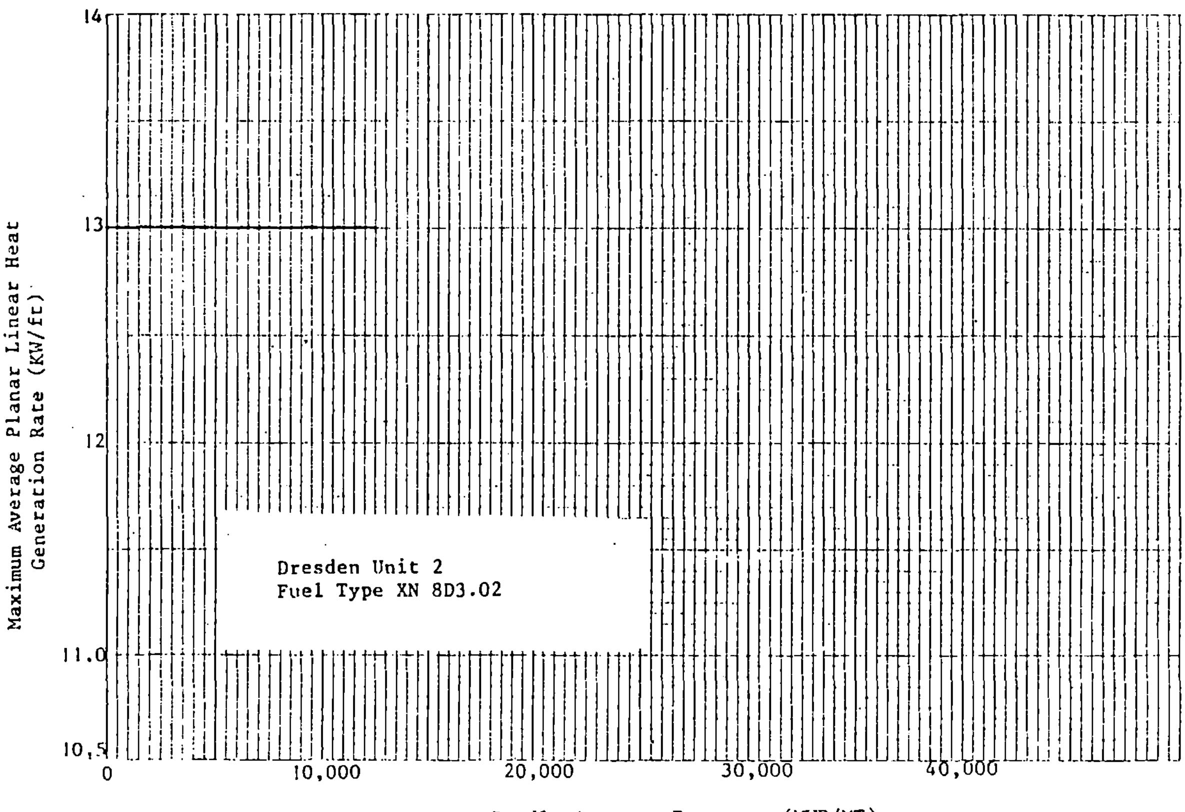




Average Planar Exposure (MWd/t)
MAPLHAR Versus Average Planar Exposure
Figure 3.5-1

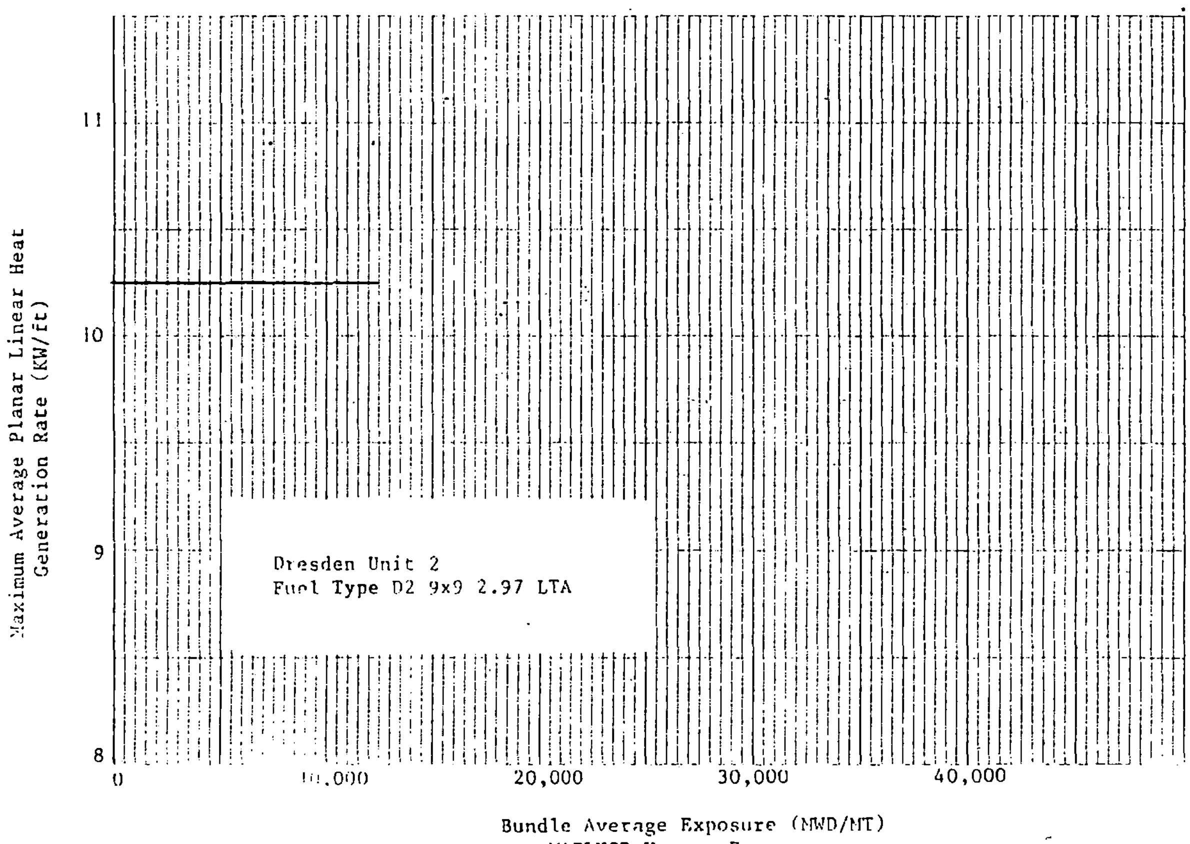
(Sheet 3 of 5)





Bundle Average Exposure (MWD/MT)
MAPHLGR Versus Exposure
Figure 3.5-1

(Sheet 4 of 5)



MAPLHGR Versus Exposure
Figure 3.1-5

(Sheet 5 of 5)

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

During steady state operation at rated core flow, MCPR shall be greater than or equal to; 1.34 for XN-1 8x8 and G.E. 8x8 Fuel types 1.35 for G.E. 8x8R 1.38 for XN-1 9x9 LTA

For core flows other than rated, the MCPR operating limit shall be as follows:

1. Manual Flow Control - the MCPR Operating Limit shall be the value from Figure 3.5-2 Sheet 1 or the above rated core flow value, which ever is greater.
2. Automatic Flow Control - the MCPR Operating Limit shall be the value from Figure 3.5-2 Sheet 1, Sheet 2 or the above rated core flow value, whichever is greatest.

If at any time during steady state power operation, it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at greater than or equal to 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

In the event the average 90% scram insertion time determined by Spec. 3.3.C for all operable control rods exceeds 2.74 seconds, the MCPR limit shall be increased by the amount equal to [0.092T - 0.252] where T equals the average 90% scram insertion time for the most recent half-core or full core surveillance data from Spec. 4.3.C.

L. Condensate Pump Room Flood Protection

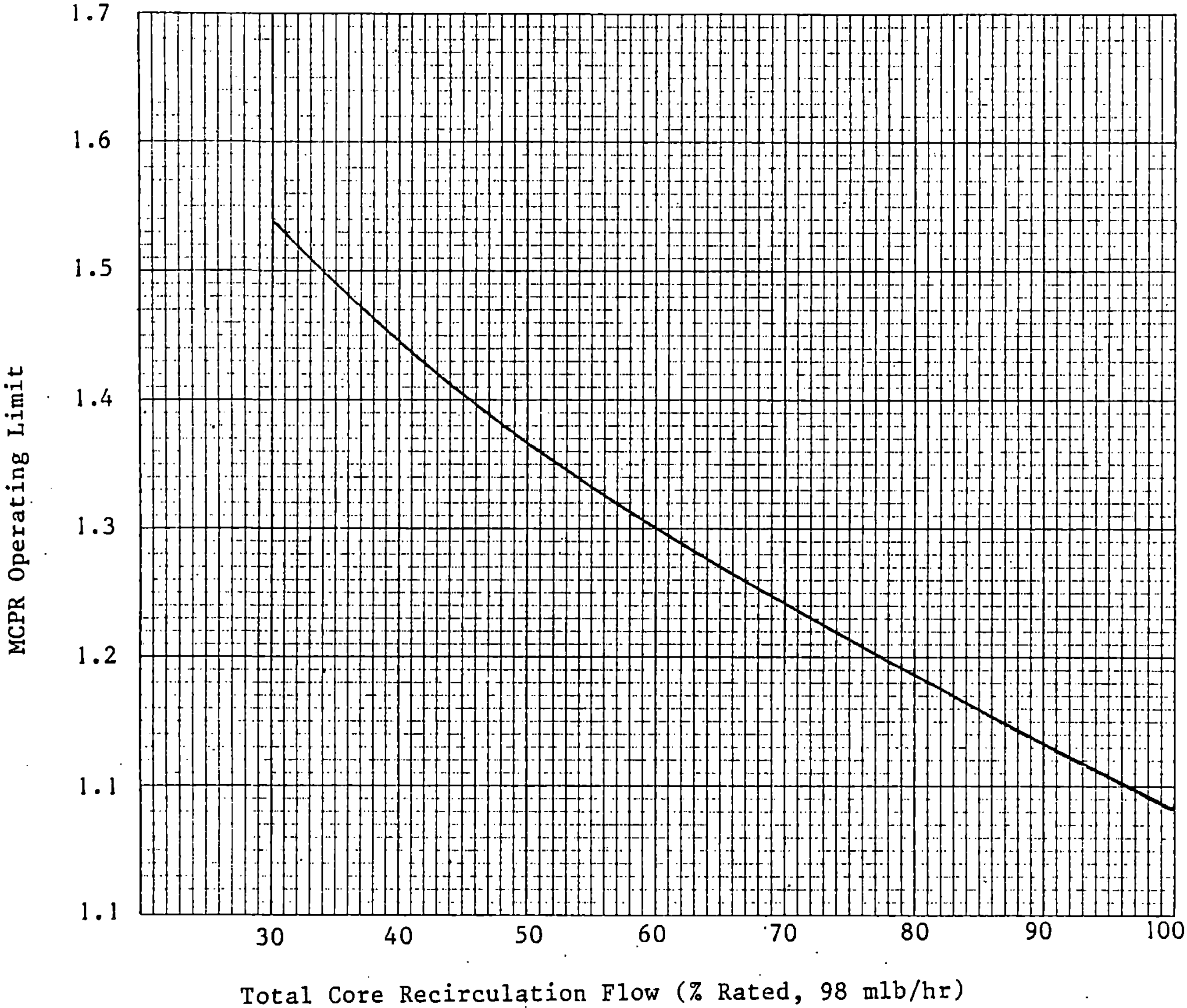
1. The system is installed to prevent or mitigate the consequences of flooding of the condensate pump room shall be operable prior to startup of the reactor.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

L. Condensate Pump Room Flood Protection

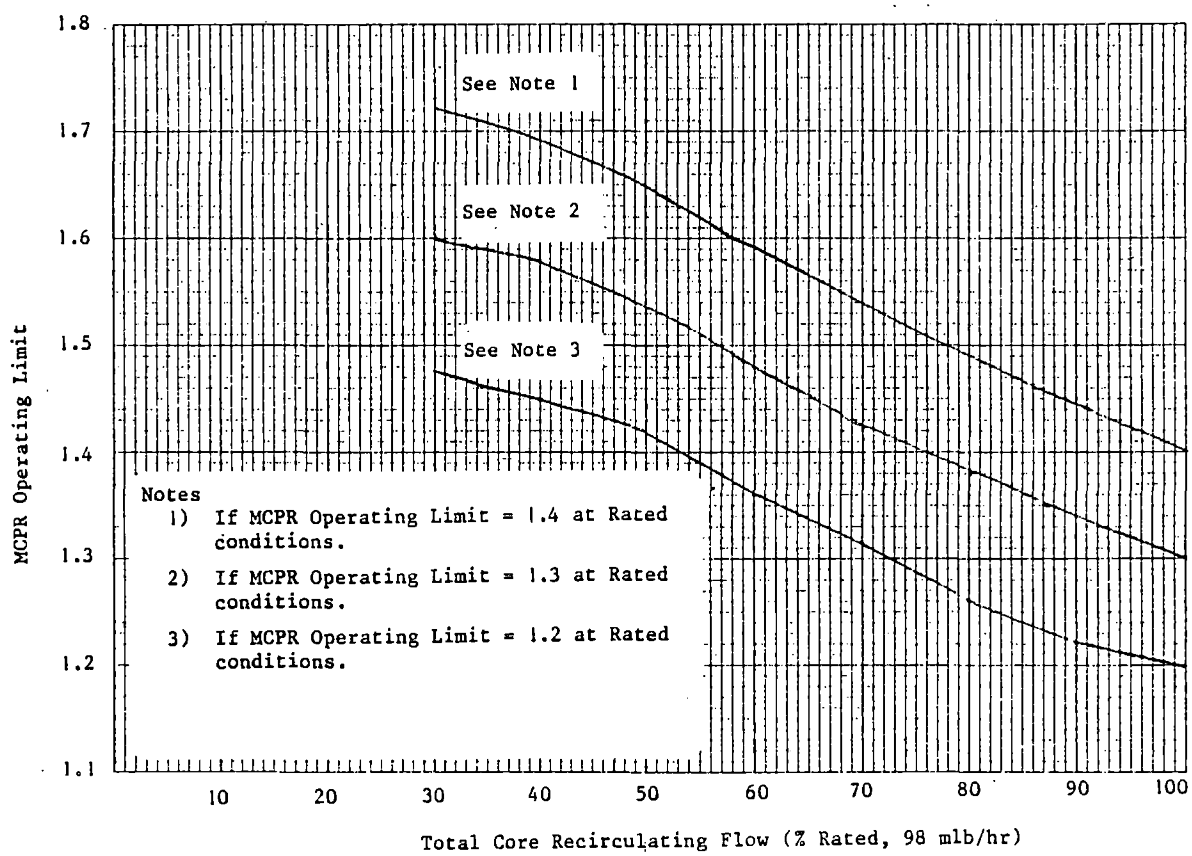
1. The following surveillance requirements shall be observed to assure that the condensate pump room flood protection is operable.

   a. The testable penetrations through the walls of CCSW pump vaults shall be checked during each operating cycle by pressurizing to 15 plus or minus 2 psig and checking for leaks using a soap bubble solution. The criteria for acceptance should be no visible leakage through the soap bubble solution. The bulkhead door shall be checked during each operating cycle by hydrostatically testing the door at 15 plus or minus 2 psig and checking to verify that leakage around the door is less than one gallon per hou



MCPR Limit for Reduced Core Flow

Figure 3.5-2 (Sheet 1 of 2)



MCPR Limit for Automatic Flow Control
Figure 3.5-2 (Sheet 2 of 2)

MCPR Limit For Automatic Flow Control
Figure 3.5-2
(Sheet 2 of 2)

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

b. The CCSW Vault Floor drain shall be checked during each operating cycle by assuring that water can be run through the drain line and actuating the air operated valves by operation of the following sensor:

i. loss of air

ii. high level in the condensate pump room (5'0")

c. The condenser pit five foot trip shall have a trip setting of less than or equal to five feet zero inches. The five foot trip circuit for each channel shall be checked once every three months. The 3 and 1 foot alarms shall have a setting of less than or equal to three feet zero inches and less than or equal to 1 foot 0 inches. A logic system functional test, including all alarms, shall be performed during the refueling outage.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The condenser pit water level switches shall trip the condenser circulating water pumps and alarm in the control room if water level in the condenser pit exceeds a level of 5 feet above the pit floor. If a failure occurs in one of these trip and alarm circuits, the failed circuit shall be immediately placed in a trip condition and reactor operation shall be permissible for the following seven days unless the circuit is sooner made operable.

3. If Specification 3.5.L.1 and 2 cannot be met, reactor startup shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3.5 LIMITING CONDITION FOR OPERATION BASES

A. Core Spray and LPCI Mode of the RHR System - This specification assures that adequate emergency cooling capability is available.

Based on the loss of coolant analyses included in References (1) and (2) in accordance with 10CFR50.46 and Appendix K, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit the calculated peak clad temperature to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

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 (3). Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems 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 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 45 days and this specification is within this period. For multiple failures, a shorter interval is specified and to improve the assurance that the remaining

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(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revisions 1, April 1979.

(2) NEDO-20566, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K.

(3) APED-"Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards" - April 1969, I.M. Jacobs and P.W. Marriott.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

systems will function, a daily test is called for. Although it is recognized that the information given in reference 3 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgement.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the reactor 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. Based on judgements of the reliability of the remaining systems; i.e. the core spray and LPCI, a 7-day repair period was obtained.

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 30-day repair period is justified. If the LPCI subsystem is not available, at least 2 LPCI pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. Containment Cooling Service Water - The containment heat removal portion of the LPCI/containment cooling subsystem is provided to remove heat energy from the containment in the event of a loss of coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and, therefore, is more than ample to provide the required heat removal capability. (Ref. Section 5.2.3.2 SAR).

The containment cooling subsystem consists of two sets of 2 service water pumps, 1 heat exchanger and 2 LPCI pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one containment cooling service water pump does not seriously jeopardize the containment cooling capability as any 2 of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left a 30-day repair period is adequate. Loss

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

of 1 containment cooling subsystem leaves one remaining system to perform the containment cooling function. The operable system is demonstrated to be operable each day when the above condition occurs. Based on the facts that when one containment cooling subsystem becomes inoperable only one system remains which is tested daily. A 7-day repair period was specified.

C. High Pressure Coolant Injection - The high pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of off-site electrical power. For the pipe breaks for which the HPCI is intended to function the core never uncovers and is continuously cooled and thus no clad damage occurs. (Ref. Section 6.2.5.3 SAR). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

D. Automatic Pressure Relief - The relief valves of the automatic pressure relief subsystem are a back-up to the HPCI subsystem. They enable the core spray or LPCI to provide protection against the small pipe break in the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core sprays or LPCI. The core spray and/or LPCI provide sufficient flow of coolant to adequately cool the core.

Loss of 1 of the relief valves affects the pressure relieving capability and therefore a 7 day repair period is specified. Loss of more than 1 relief valve significantly reduces the pressure relief capability and thus a 24-hour repair period is specified.

E. Isolation Cooling System - The turbine main condenser is normally available. The isolation condenser is provided for core decay heat removal following reactor isolation and scram. The isolation condenser has a heat removal capacity sufficient to handle the decay heat production at 300 seconds following a scram. Water will be lost from the reactor vessel through the relief valves in the 300 seconds following isolation and scram. This represents a minor loss relative to the vessel inventory.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1060 psig sustained for 15 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. To be considered operable the shell side of the isolation condenser must contain at least 11,300 gallons of water. Make-up water to the shell side of the isolation condenser is provided by the condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are operable from on-site power. The fire protection system is also available as make-up water. An alternate method of cooling the core upon isolation from the main condenser is by using the relief valves and HPCI subsystem in a feed and bleed manner. Therefore, the high pressure relief function and the HPCI must be available together to cope with an anticipated transient so the LCO for HPCI and relief valves is set upon this function rather than their function as depressurization means for a small pipe break.

F. Emergency Cooling Availability - The purpose of Specification D is to assure a minimum of core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only 2 LPCI pumps would be available. Likewise, if 2 LPCI pumps were out of service and 2 containment service water pumps on the opposite side were also out of service no containment cooling would be available. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

Dresden Units 2 and 3 share certain process systems such as the makeup demineralizers and the radwaste system and also some safety systems such as the standby gas treatment system, batteries, and diesel generators. All of these systems have been sized to perform their intended function considering the simultaneous operation of both units.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

For the safety related shared features of each plant, the Technical Specifications for that unit contain the operability and surveillance requirements for the shared feature; thus, the level of operability for one unit is maintained independently of the status of the other. For example, the shared diesel (2/3 diesel) would be mentioned in the specifications for both Units 2 and 3 and even if Unit 3 were in the Cold Shutdown Condition and needed no diesel power, readiness of the 2/3 diesel would be required for continuing Unit 2 operation.

G. Specification 3.5.F.4 provides that should this occur, no work will be performed which could preclude adequate emergency cooling capability being available. Work is prohibited unless it is in accordance with specified procedures which limit the period that the control rod drive housing is open and assures that the worst possible loss of coolant resulting from the work will not result in uncovering the reactor core. Thus, this specification assures adequate core cooling. Specification 3.9 must be consulted to determine other requirements for the diesel generator.

Specification 3.5.F.5 provides assurance that an adequate supply of coolant water is immediately available to the low pressure core cooling systems and that the core will remain covered in the event of a loss of coolant accident while the reactor is depressurized with the head removed.

H. Maintenance of Filled Discharge Pipe - If the discharge piping of the core spray, LPCI, and HPCI are not filled, a water hammer can develop in this piping when the pump and/or pumps are started.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10CFR50 Appendix K considering the postulated affects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average LHGR of all the rods in a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within a fuel assembly. Since expected

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than plus or minus 20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the 10CFR50, Appendix K limit.

The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in Reference (1) and in reference (2). Power operation with APLHGRs at or below those shown in Fig. 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit.

The maximum average planar LHGRs for G.E. fuel plotted in Fig. 3.5.1 at higher exposures result in a calculated peak clad temperature of less than 2200°F. However, the maximum average planar LHGRs are shown on Fig. 3.5.1 as limits because conformance calculations have not been performed to justify operation at LHGHs in excess of those shown.

J. Local LHGR

This specification assures that the maximum linear heat generation rate in any fuel rod fabricated by G.E. is less than the design linear heat generation rate even if fuel pellet densification is postulated.

For fuel fabricated by ENC, protection of the MCPR and MAPLHGR limits and operation within the power distribution assumptions of the Fuel Design Analysis provides adequate protection against cladding strain limits, hence the LHGR limitation for GE fuel is unnecessary for the protection of ENC fuel.

(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision 1, April, 1979.

(2) XN-NF-82-88 "Dresden Unit 2 LOCA Analysis Using the ENC EXEM/BWR Evaluation Model MAPLHGR Results"

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

The steady-state values for MCPR specified in the Specification were determined using the THERMEX thermal limits methodology described in XN-NF-80-19, Volume 3. The safety limit implicit in the Operating limits is established so that during sustained operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. The Limiting Transient delta CPR implicit in the operating limits was calculated such that the occurrence of the limiting transient from the operating limit will not result in violation of the MCPR safety limit in at least 95% of the random statistical combinations of uncertainties.

Transient events of each type anticipated during operation of a BWR/3 were evaluated to determine which is most restrictive in terms of thermal margin requirements. The generator load rejection/turbine trip without bypass is typically the limiting event. The thermal margin effects of the event are evaluated with the THERMEX Methodology and appropriate MCPR limits consistent with the XN-3 critical power correlation are determined. Several factors influence which transient results in the largest reduction in critical power ratio, such as the cycle-specific fuel loading, exposure and fuel type. The current cycle's reload licensing analyses identifies the limiting transient for that cycle.

As described in Specification 4.3.C.3 and the associated Bases, observed plant data were used to determine the average scram performance used in the transient analyses for determining the MCPR Operating Limit. If the current cycle scram time performance falls outside of the distribution assumed in the analyses, an adjustment of the MCPR limit may be required to maintain margin to the MCPR Safety Limit during transients. Compliance with the assumed distribution and adjustment of the MCPR Operating Limit will be performed in accordance with Technical Specifications 4.3.C.3 and 3.5.K.

For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the MCPR Safety Limit in the event of an uncontrolled recirculation flow increase to the physical limit of pump flow. This protection is provided for manual and automatic flow control by choosing the MCPR operating limit as the value from Figure 3.5-2 Sheet 1 or the rated core flow value, whichever is greater. For Automatic Flow Control, in addition to protecting the MCPR Safety Limit during the flow

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

run-up event, protection is provided against violating the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow. This protection is provided by the reduced flow MCPR limits shown in Figure 3.5-2 Sheet 2 where the curve corresponding to the current rated flow MCPR limit is used (linear interpolation between the MCPR limit lines depicted is permissible). Therefore, for Automatic Flow Control, the MCPR Operating Limit is chosen as the value from Figure 3.5-2 Sheet 1, Sheet 2 or the rated flow value, whichever is greatest. It should be noted that if the rated flow MCPR Limit must be increased due to degradation of control rod scram times during the current cycle, the new value of the rated flow MCPR limit is applied when using Figure 3.5-2 Sheet 2.

L. Flood Protection

Condensate pump room flood protection will assure the availability of the containment cooling service water system (CCSW) during a postulated incident of flooding in the turbine building. The redundant level switches in the condenser pit will preclude any postulated flooding of the turbine building to an elevation above river water level. The level switches provide alarm and circulating water pump trip in the event a water level is detected in the condenser pit.

4.5 SURVEILLANCE REQUIREMENT BASES

(A thru F)

The testing interval for the core and containment cooling systems is based on quantitative reliability analysis, judgement and practicality. The core cooling systems have not been designed to be fully testable during operation. For example the core spray final admission valves do not open until reactor pressure has fallen to 350 psig thus during operation even if high drywell pressure were stimulated the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

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 preventative 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 due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The requirement of 180 psig at 3500 gpm at the containment cooling service water (CCSW) pump discharge provides adequate margin to ensure that the LPCI/CCSW system provides the design

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

bases cooling water flow and maintains 20 psig differential pressure at the containment cooling heat exchanger. This differential pressure precludes reactor coolant from entering the river water side of the containment cooling heat exchangers.

The verification of Main Steam Relief Valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the valve actuation mechanism during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for relief valve opening. This test method may be performed over a wide range of reactor pressure greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

G. Deleted

H. Maintenance of Filled Discharge Pipe

The surveillance requirements to assure that the discharge piping of the core spray, LPCI, and HPCI systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided to monitor the presence of water in the discharge piping. This instrumentation will be calibrated on the same frequency as the safety system instrumentation. This period of periodic testing ensures that during the intervals between the monthly checks the status of the discharge piping is monitored on a continuous basis.

I. Average Planar LHGR

At core thermal power levels less than or equal to 25 per cent, operating plant experience and thermal hydraulic analyses indicate that the resulting average planar LHGR is below the maximum average planar LHGR by a considerable

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

margin; therefore, evaluation of the average planar LHGR below this power level is not necessary. The daily requirement for calculating average planar LHGR above 25 per cent rated thermal power is sufficient since power distribution shifts are slow when there have not been significant power or control rod changes.

J. Local LHGR

The LHGR for G.E. fuel shall be checked daily during reactor operation at greater than or equal to 25 per cent power to determine if fuel burnup or control rod movement has caused changes in power distribution. A limiting LHGR value is precluded by a considerable margin when employing a permissible control rod pattern below 25% rated thermal power.

K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 per cent, the reactor will be operating at 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 and thermal hydraulic analysis indicates 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.

The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

In addition, the k_f correction applied to the LCO provides margin for flow increase from low flows.

L. Flood Protection

The watertight bulkhead door and the penetration seals for pipes and cables penetrating the vault walls have been designed to withstand the maximum flood conditions. To assure that their installation is adequate for maximum flood conditions, a method of testing each seal has been devised.

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

To test a pipe seal, another test seal is installed in the opposite side of the penetration creating a space between the two seals that can be pressurized. Compressed air is then supplied to a fitting on the test seal and the space inside the sleeve is pressurized to approximately 15 psi. The outer face of the permanent seal is then tested for leaks using a soap bubble solution.

On completion of the test, the test seal is removed for use on other pipes and penetrations of the same size.

In order to test the watertight bulkhead doors, a test frame must be installed around each door. At the time of the test, a reinforced steel box with rubber gasketing is clamped to the wall around the door. The fixture is then pressurized to approximately 15 psig to test for leak tightness.

Floor drainage of each vault is accomplished through a carbon steel pipe which penetrates the vault. When open, this pipe will drain the vault floor to a floor drain sump in the condensate pump room.

Equipment drainage from the vault coolers and the CCSW pump bedplates will also be routed to the vault floor drains. The old equipment drain pipes will be permanently capped to preclude the possibility of back-flooding the vault.

As a means of preventing backflow from outside the vaults in the event of a flood, a check valve and an air operated valve are installed in the 2" vault floor drain line 6'0" above the floor of the condensate pump room.

The check valve is a 2" swing check designed for 125 psig service. The air operated valve is a control valve designed for a 50 psi differential pressure. The control valve will be in the normally open position in the energized condition and will close upon any one of the following:

a. Loss of air or power

b. High level (5'0") in the condensate pump room

Closure of the air operated valve on high water level in the condensate pump room is effected by use of a level switch set at a water level of 5'0". Upon actuation, the switch will close the control valve and alarm in the control room.

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The operator will also be aware of problems in the vaults/condensate pump room if the high level alarm on the equipment drain sump is not terminated in a reasonable amount of time. It must be pointed out that these alarms provide information to the operator but that operator action upon the above alarms is not a necessity for reactor safety since the other provisions provide adequate protection.

A system of level switches has been installed in the condenser pit to indicate and control flooding of the condenser area. The following switches are installed:

	Level	Function
a.	1'0" (1 switch)	Alarm, Panel Hi-Water-Condenser Pit
b.	3'0" (1 switch)	Alarm, Panel High-Circ. Water Condenser Pit
c.	5'0" (2 redundant switch pairs)	Alarm and Circ. Water Pump Trip

Level (a) indicates water in the condenser pit from either the hotwell or the circulating water system. Level (b) is above the hotwell capacity and indicates a probable circulating water failure.

Should the switches at level (a) and (b) fail or the operator fail to trip the circulating water pumps on alarm at level (b), the actuation of either level switch pair at level (c) shall trip the circulating water pumps automatically and alarm in the control room. These redundant level switch pairs at level (c) are designed and installed to IEEE-279, "Criteria for Nuclear Power Plant Protection Systems." As the circulating water pumps are tripped, either manually or automatically, at level (c) of 5'0", the maximum water level reached in the condenser pit due to pumping will be at the 491'0" elevation (10' above condenser pit floor elevation 481'0"; 5' plus an additional 5' attributed to pump coastdown).

In order to prevent overheating of the CCSW pump motors, a vault cooler is supplied for each pump. Each vault cooler is designed to maintain the vault at a maximum 105°F temperature during operation of its respective pump. For example, if CCSW pump 2B-1501 starts, its cooler will also start and compensate

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

for the heat supplied to the vault by the 2B pump motor keeping the vault at less than 105°F.

Each of the coolers is supplied with cooling water from its respective pump's discharge line. After the water has been passed through the cooler, it returns to its respective pump's suction line. In this way, the vault coolers are supplied with cooling water totally inside the vault. The cooling water quantity needed for each cooler is approximately 1% to 5% of the design flow of the pumps so that the recirculation of this small amount of heated water will not affect pump or cooler operation.

Operation of the fans and coolers is required during pump operability testing and thus additional surveillance is not required.

Verification that access doors to each vault are closed, following entrance by personnel, is covered by station operating procedures.

3.6 LIMITING CONDITION FOR OPERATION

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 Limitations

1. Except as indicated in 3.6.A.2 below, 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. A step reduction in reactor coolant temperature of 240°F is permissible so long as the limit in Specification 3.6.A.3 below is met.

3. At all times, the shell flange to shell temperature differential shall not exceed 140°F.

4.6 SURVEILLANCE REQUIREMENT

PRIMARY SYSTEM BOUNDARY

Applicability:

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

Objective:

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

Specification:

A. Thermal Limitations

1. During heatups and cooldowns the following temperatures shall be permanently recorded at 15 minute intervals:

   a. reactor vessel shell

   b. reactor vessel shell flange

   c. recirculation loops A & B

2. The temperatures listed in 4.6.A.1 shall be permanently recorded subsequent to a heatup or cooldown at 15 minute intervals until three consecutive readings are within 5 degrees of each other.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4. Core thermal power shall not exceed 25% of rated thermal power without forced recirculation.

B. Pressurization Temperature

1. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Curve C of Figure 3.6.1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Fig. 3.6.1. Figure 3.6.1 is effective through 6 effective full power years. At least six months prior to 6 effective full power years new curves will be submitted.

2. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel shell immediately below the vessel flange is greater than or equal to 100°F.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

B. Pressurization Temperature

1. Reactor Vessel shell temperature and reactor coolant pressure shall be permanently recorded at 15 minute intervals whenever the shell temperature is below 220°F and the reactor vessel is not vented.

2. When the reactor vessel head bolting studs are tightened or loosened the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed 20 microcuries of total iodine per ml of water.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3. Neutron flux monitors and samples shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The monitor and sample program where possible conform to ASTM E 185. The monitors and samples will be removed and tested as outlined in Table 4.6.2 to experimentally verify the calculated values of integrated neutron flux that are used to determine NDTT for Figure 4.6.1.

C. Coolant Chemistry

1. a. A Sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactivity.

   b. Isotopic analysis of a sample of reactor coolant shall be made at least once per month.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in 3.6.C.3:

   Conductivity 2 micro-mho/cm
   Chloride ion 0.1 ppm

3. For reactor startups the maximum value for conductivity shall not exceed 10 micro-mho/cm and the maximum value for chloride ion concentration shall not exceed 0.1 ppm, for the first 24 hours after placing the reactor in the power operating condition.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

2. During startups and at steaming rates below 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for conductivity and chloride content.

3. a. With steaming rates greater than or equal to 100,000 pounds per hour, a reactor coolant sample shall be taken at least every 96 hours and when the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes) and analyzed for conductivity and chloride ion content.

   b. When the continuous conductivity monitor is inoperable, a reactor coolant sample should be taken at least daily and analyzed for conductivity and chloride ion content.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4. Except as specified in 3.6.C.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hour.

   Conductivity 5 micro-mho/cm
   Chloride ion 0.5 ppm

5. If Specification 3.6.C.1, 3.6.C.2, 3.6.C.3 or 3.6.C.4 is not met, an orderly shutdown shall be initiated.

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

D. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system. Sump flow monitoring and recording shall be performed once per 4 hours. Air sampling shall be performed once per day.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. After completion of the investigation, or containment inspection, specified in 4.6.D.2.a or 4.6.D.2.b, if the leakage is determined to be due to a thru wall pipe crack on the reactor coolant pressure boundary, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

E. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320°F, all nine of the safety valves shall

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

2. The following additional leakage limits shall be met until the recirculation piping indications have been resolved.

Whenever the reactor is at operating pressure, the following will apply to unidentified leakage:

a. If a 1 gpm increase over the previous 4 hours occurs or when leakage equals 3 gpm total, an investigation of the cause of the leakage increase will be performed. This investigation should consist of taking drywell air and water samples, and a review of any previous plant evolutions to the extent necessary to determine the source of leakage.

b. If leakage equals 4 gpm, a containment inspection will be conducted to determine the source of leakage.

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outages. The popping point of the safety valves shall be set as follows:

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

2. If Specification 3.6.E.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be less than or equal to 90 psig and less than or equal to 320° F within 24 hours.

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components".

Components of the primary system boundary whose inservice examination reveals the absence of flaw indications or flaw

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

Number of Valves	Set Point (Psig)
1	1135*
2	1240
2	1250
2	1260
2	1260

The allowable set point error for each valve is plus or minus 1%.

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Valve No.	Set Point (psig)
203-3A	1124*
203-3B	1101
203-3C	1101
203-3D	1124
203-3E	1124

* Target Rock combination safety/relief valve

The allowable setpoint error for each valve is plus or minus 1%.

F. Structural Integrity

1. Beginning November 1, 1978, and updated every 40 months thereafter, the component inservice inspection program shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been given by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

indications not in excess of the allowable indication standards of this Code are acceptable for continued service. Plant operation with components which have inservice examination flaw indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

a. Components whose inservice examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

(i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI.

(ii) Prior to the resumption of service, the NRC shall review the analysis and evaluation and

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

b. For components approved for continued service in accordance with paragraph "a" above, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive inservice inspection. An analysis and evaluation shall be submitted to the NRC following each inservice inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each inservice inspection, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

the affected component or require that the component be repaired or replaced.

c. Repair or replacement of components, including reexaminations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

G. Jet Pumps

1. Whenever the Reactor is in the Startup/Hot Standby or Run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

G. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the Startup/Hot Standby or Run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

H. Recirculation Pump Flow Mismatch

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pumps in the idle loop shall not vary by more than 10% from established patterns.

3. The baseline date required to evaluate the conditions in Specifications 4.6.G.1 and 4.6.G.2 will be acquired each operating cycle.

H. Recirculation Pump Flow Mismatch

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

2. If specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.

3. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.

4. Whenever one pump is operable and the remaining pump is in the tripped position, the operable pump shall be at a speed less than 65% before starting the inoperable pump.

I. Snubbers (Shock Suppressors)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

Recirculation pumps speed shall be checked daily for mismatch.

I. Snubbers (Shock) Suppressors)

The following surveillance requirements apply to all

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. During all modes of operation except cold shutdown and refuel, all safety related snubbers listed in Table 3.6.1.a and 3.6.1b shall be operable except as noted in Specification 3.6.I.2 through 3.6.I.4.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

safety related snubbers listed in Tables 3.6.1a and 3.6.1b.

1. Visual Inspection

   An independent visual inspection shall be performed on the safety related hydraulic and mechanical snubbers contained in Tables 3.6.1a and 3.6.1b in accordance with the below schedule.

   a. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connection to the piping and anchor to verify snubber operability.

   b. All mechanical snubbers shall be visually inspected. This

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

inspection shall consist of, but not necessarily be limited to, inspection of the snubber and attachments to the piping and anchor for indications of damage or impaired operability.

No. of Snubbers Found Inoperable During Inspection Interval	Next Required Inspection Interval
0	18 months plus or minus 25%
1	12 months plus or minus 25%
2	6 months plus or minus 25%
3,4	124 days plus or minus 25%
5,6,7	62 days plus or minus 25%
8 or more	31 days plus or minus 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible," based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. From and after the time a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced. Torus Ring Header snubbers may be inoperable in either of the following configurations until January 19, 1984 to facilitate the installation of the Mark I torus attached piping modifications.

Configuration A: Every other existing snubber pair (up to 3 pairs) on the ECCS header, or

Configuration B: One existing snubber from each of the 6 existing snubber pairs on the ECCS header.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

2. Functional Testing

a. Once each refueling cycle, a representative sample of approximately 10% of the hydraulic snubbers contained in Table 3.6.1a shall be functionally tested for operability, including:

*(i) Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

*(ii) Snubber bleed, or release rate, where required, is within the specified range in compression or tension.

*NOTE: Paragraphs (i) and (ii) shall not become effective until competitive marketable test fixtures are available. Until such time, but in no case to exceed 12/31/83, demonstration of snubber bleed, or release, shall be sufficient.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

For each unit and subsequent unit found inoperable, an additional 10% of the hydraulic snubbers shall be tested until no more failures are found or all units have been tested.

b. Once each refueling cycle, a representative sample of approximately 10% of the mechanical snubbers contained in Table 3.6.1b shall be functionally tested for operability. The test shall consist of two parts:

** i. Verification that the force that initiates free movement of the snubber in either tension or compression is less than the specified maximum breakaway friction force.

** (See next page)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

**ii. Verify that the activation (restraining action) is achieved within the specified range of acceleration in both tension and compression.

For each unit and subsequent unit found inoperable, an additional 10% of the mechanical snubbers shall be so tested until no more failures are found or all units have been tested.

c. In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the resampling.

** NOTE: Paragraph (ii) shall not become effective until competitive marketable test fixtures are available. Until then, but in no case to exceed 12/31/83, the functional test will be limited to only paragraph (i).

TABLE 3.6.1.a

HYDRAULIC SNUBBERS

TABLE 3.6.1a

SAFETY RELATED HYDRAULIC SNUBBERS*

DPR-19

Note 1. These snubbers are being replaced with mechanical snubbers as delineated in Section 3.6.1.2 and a revised table will be issued upon completion of the Mark I Torus attached piping modification.

SNUBBER NO.	LOCATION	ELEVATION	AZIMUTH	SNUBBER IN HIGH RADIATION AREA DURING SHUTDOWN	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
2	Torus Ring Header 1501-24" (Note 1)	483'	83°			X
3	Torus Ring Header 1501-24" (Note 1)	483'	74°			X
4	Torus Ring Header 1501-24" (Note 1)	483'	38°			X
5	Torus Ring Header 1501-24" (Note 1)	483'	29°			X
7	Torus Ring Header 1501-24" (Note 1)	483'	331°			X
8	Torus Ring Header 1501-24" (Note 1)	483'	322°			X
9	Torus Ring Header 1501-24" (Note 1)	483'	286°			X
10	Torus Ring Header 1501-24" (Note 1)	483'	277°			X
12	Torus Ring Header 1501-24" (Note 1)	483'	218°			X
13	Torus Ring Header 1501-24" (Note 1)	483'	209°			X
15	Torus Ring Header 1501-24" (Note 1)	483'	151°			X
16	Torus Ring Header 1501-24" (Note 1)	483'	142°			X

*Modifications to this table due to changes in high radiation should be submitted to the NRC as part of the next license amendment request.

TABLE 3.6.1.a

HYDRAULIC SNUBBERS

TABLE 3.6.1a

SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	LOCATION	ELEVATION	AZIMUTH	SNUBBER IN HIGH RADIATION AREA DURING SHUTDOWN	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
	Isolation Condenser Pipeway Room:					
1	Isolation Condenser Line 1303-12"	558'	180°	X		X
2	Isolation Condenser Line 1303-12"	568'	180°	X		X
3	Isolation Condenser Line 1302-14"	580'	195°	X		X

*Modifications to this table due to changes in high radiation should be submitted to the NRC as part of the next license amendment request.

TABLE 3.6.1.b

. MECHANICAL SNUBBERS

TABLE 3.6.1b

SAFETY RELATED MECHANICAL SNUBBERS

SNUBBER NO.	LOCATION	ELEVATION	AZIMUTH	SNUBBER IN HIGH RADIATION AREA DURING SHUTDOWN	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
1	Drywell Recirc. Motor 2B-202	524'	328°	X	X	
2	Drywell Recirc. Motor 2B-202	524'	302°	X	X	
3	Drywell Recirc. Motor 2B-202	524'	315°	X	X	
4	Drywell Recirc. Motor 2A-202	524'	148°	X	X	
5	Drywell Recirc. Motor 2A-202	524'	122°	X	X	
6	Drywell Recirc. Motor 2A-202	524'	135°	X	X	
7	Drywell Recirc. Pump 2B-202	512'	326°	X	X	
8	Drywell Recirc. Pump 2B-202	512'	304°	X	X	
9	Drywell Recirc. Pump 2B-202	517'	315°	X	X	
10	Drywell Recirc. Pump 2A-202	512'	124°	X	X	
11	Drywell Recirc. Pump 2A-202	512'	146°	X	X	
12	Drywell Recirc. Pump 2A-202	507'	135°	X	X	
13-16	Removed					
17	Drywell Recirc Header 201B-22"	533'6"	195°	X	X	
18-20	Removed					
21	Drywell Recirc Header 201A-22"	533'6"	22°	X	X	
22-23	Removed					
24	Drywell Feedwater Line 3204D-12"	530'	108°	X	X	
25-29	Removed					
30	Drywell Core Spray Line 1403-10"	575'	336°	X	X	
31	Drywell Core Spray Line 1404-10"	562'	231°	X	X	
32	Drywell Target Rock Valve 203-3A	542'6"	16°	X	X	
33	Drywell Target Rock Valve 203-3A	542'4"	31°	X	X	
34	Drywell Target Rock Valve 203-3A	540'0"	19°	X	X	
35	Removed					
36	Drywell Recirc. Line 201B-28"	518'	270°	X	X	

Modifications to this table due to changes in high radiation should be submitted to the NRC as part of the next license Amendment request.

TABLE 3.6.1.b

MECHANICAL SNUBBERS

TABLE 3.6.1b (Continued)

SAFETY RELATED MECHANICAL SNUBBERS

SNUBBER NO.	LOCATION	ELEVATION	AZIMUTH	SNUBBER IN HIGH RADIATION AREA DURING SHUTDOWN	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
37	Drywell Recirc. Line 201A-28"	518'	90°	X	X	
38	Drywell Shutdown Cooling Line 1001A-16"	523'	0°	X	X	
39	Drywell Rx Water Cleanup Line 1201-8"	533'	316°	X	X	
40	Drywell Rx Water Cleanup Line 1201-8"	533'	301°	X	X	
41	Drywell Main Steam Line 3001B-20"	534'	28°	X	X	
42	Drywell Main Steam Line 3001A-20"	536'	14°	X	X	
43	Drywell Main Steam Line 3001D-20"	536'	346°	X	X	
44	Drywell Main Steam Line 3001C-20"	536'	312°	X	X	
45	Drywell Main Steam Line 3001B-20"	545'	105°	X	X	
46	Drywell Main Steam Line 3001B-20"	543'	105°	X	X	
47	Drywell Main Steam Line 3001A-20"	542'	73°	X	X	
48	Drywell Main Steam Line 3001A-20"	543'6"	73°	X	X	
49	Drywell Main Steam Line 3001A-20"	539'	20°	X	X	
50	Drywell Main Steam Line 3001C-20"	543'6"	195°	X	X	
51	Drywell Main Steam Line 3001C-20"	542'	195°	X	X	
52	Drywell Main Steam Line 3001D-20"	543'	343°	X	X	
53	Drywell Main Steam Line 3001D-20"	544'	343°	X	X	

Modifications to this table due to changes in high radiation should be submitted to the NRC as part of the next license amendment request.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in cold shutdown or refuel condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the cold shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup. This requirement does not apply to Torus Ring Header snubbers for the period identified in paragraph 3.6.I.2. above.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When a snubber is is deemed inoperable, a review of all pertinent facts shall be conducted to determine the snubber mode of failure and to decide if an engineering evaluation should be performed on the supported system or components. If said evaluation is deemed necessary, it will determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

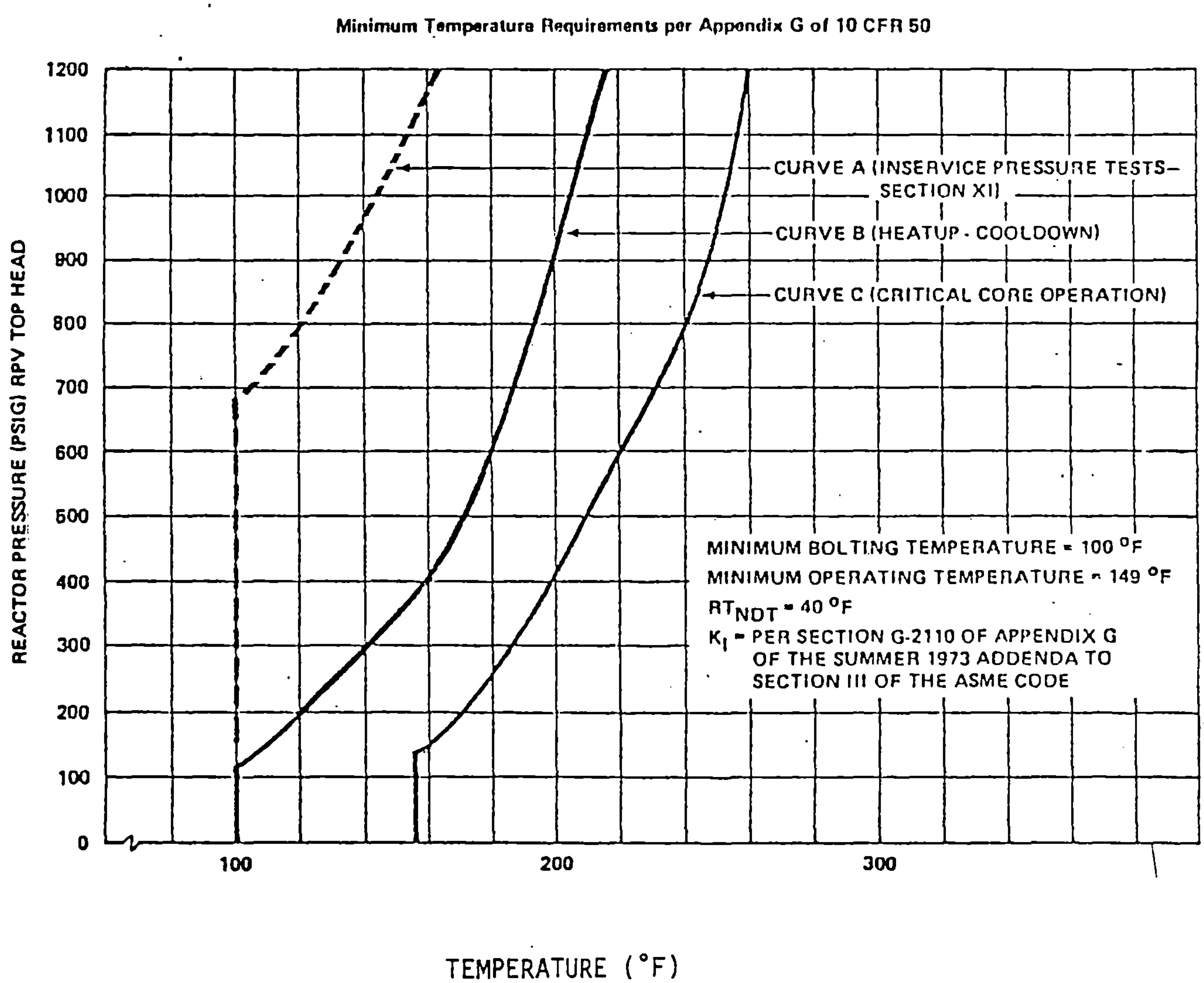
4. If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and, if determined to be a generic deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

5. Snubbers may be added to safety related systems without prior license amendment to Tables 3.6.1a and/or 3.6.1b provided that a revision to Tables 3.6.1a and/or 3.6.1b is included with the next license amendment request.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

5. Snubber service life monitoring shall be followed by existing station record systems, including the central filing system, maintenance files, safety related work packages, and snubber inspection records. The above record retention methods shall be used to prevent the hydraulic snubbers from exceeding a service life of 10 years and the mechanical snubbers from exceeding a service life of 40 years (lifetime of the plant).



TEMPERATURE (°F)

Fig. 3.6.1
MINIMUM TEMPERATURE REQUIREMENTS PER APPENDIX G OF 10 CFR 50

3.6 LIMITING CONDITION FOR OPERATION BASES

A. Thermal Limitations - The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 100°F per hour averaged over a period of one hour. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

Specific analyses were made based on a heating and cooling rate of 100°F/hour applied continuously over a temperature range of 100°F to 550°F. Because of the slow temperature-time response of the massive flanges relative to the adjacent head and shell sections, calculated temperatures obtained were 500°F (shell) and 360°F (flange) (140°F differential). Both axial and radial thermal stresses were considered to act concurrently with full primary loadings. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III stress intensity and fatigue limits even at the flange area where maximum stress occurs.

The flange metal temperature differential of 140°F occurred as a result of sluggish temperature response and the fact that the heating rate continued over a 450°F coolant temperature range.

The uncontrolled cooldown rate of 240°F was based on the maximum expected transient over the lifetime of the reactor vessel. This maximum expected transient is the injection of cold water into the vessel by the high pressure coolant injection subsystem. This transient was considered in the design of the pressure vessel and five such cycles were considered in the design. Detailed stress analyses were conducted to assure that the injection of cold water into the vessel by the HPCI would not exceed ASME stress code limitations.

B. Specification 3.6.A.4 increases margin of safety for thermal-hydraulic stability and startup of recirculation pump.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Pressurization Temperature - The reactor coolant system is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected. These restrictions on inservice hydrostatic testing, on heatup and cooldown, and on critical core operation shown in Figure 3.6.1, were established to be in conformance with Appendix G to 10 CFR 50.

In evaluating the adequacy of ferritic steels Sa302B it is necessary that the following be established:

a) The reference nil-ductility temperature (RT_{NDT}) for all vessel and adjoining materials,

b) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev), and

c) the fluence at the location of a postulated flow.

The initial RT_{NDT} of the main closure flange, the shell and head materials connecting to these flanges, and connecting welds is 10°F. However, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. (Reference Appendix F to the FSAR) The closure flanges and connecting shell materials are not subject to any appreciable neutron radiation exposure, nor are the vertical electroslag seams. The flange area is moderately stressed by tensioning the head bolts. Therefore, as is indicated in curves (a) and (b) of Figure 3.6.1, the minimum temperature of the vessel shell immediately below the vessel flange is established as 100°F below a pressure of 400 psig. (40°F + 60°F, where 40°F is the RT_{NDT} of the electroslag weld and 60°F is a conservatism required by the ASME Code). Above approximately 400 psig pressure, the stresses associated with pressurization are more limiting than the bolting stresses, a fact that is reflected in the non-linear portion of curves (a) and (b). Curve (c), which defines the temperature limitations for critical core operation, was established per Section IV 2.c. of Appendix G of 10CFR50. Each of the curves, (a), (b) and (c) define temperature limitations for unirradiated ferric steels. Provision has been made for the modification of these curves to account for the change in RT_{NDT} as a result of neutron embrittlement.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The withdrawal schedule in Table 4.6.2 is based on the three capsule surveillance program as defined in Section 11.C.3.a of 10 CFR 50 Appendix H. The accelerated capsule (Near Core Top Guide) are not required by Appendix H but will be tested to provide additional information on the vessel material.

This surveillance program conforms to ASTM E 185-73 "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" with one exception. The base metal specimens of the vessel were made with their longitudinal axes parallel to the principal rolling direction of the vessel plate.

C. Coolant Chemistry - A radioactivity concentration limit of 20 Micro-Ci/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Specification 3.8.C.1 or there is a failure or a prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture, outside the drywell, the resultant radiological dose at the site boundary would be about 10 rem to the thyroid. This does was calculated on the basis of a total iodine activity limit of 20 Micro-Ci/ml, meteorology corresponding to Type F conditions with a one meter per second wind speed, and a valve closure time of five seconds. If the valve closed in ten seconds, then the resultant dose would increase to about 25 rem.

The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hours. In addition, the trend of the stack off-gas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam lines.

Materials in the primary system are primarily 304 stainless steel and the Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

the stainless steel. The attached graph, Figure 4.6.2, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolves oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup and hot standby. During these periods with steaming rates less than 100,000 pounds per hour, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Figure 4.6.2 are not exceeded. At steaming rates of at least 100,000 pounds per hour, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt; e.g., Na_2SO_4, which would not have an affect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include, operation of the reactor clean-up system, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the clean-up system to re-establish the purity of the reactor coolant.

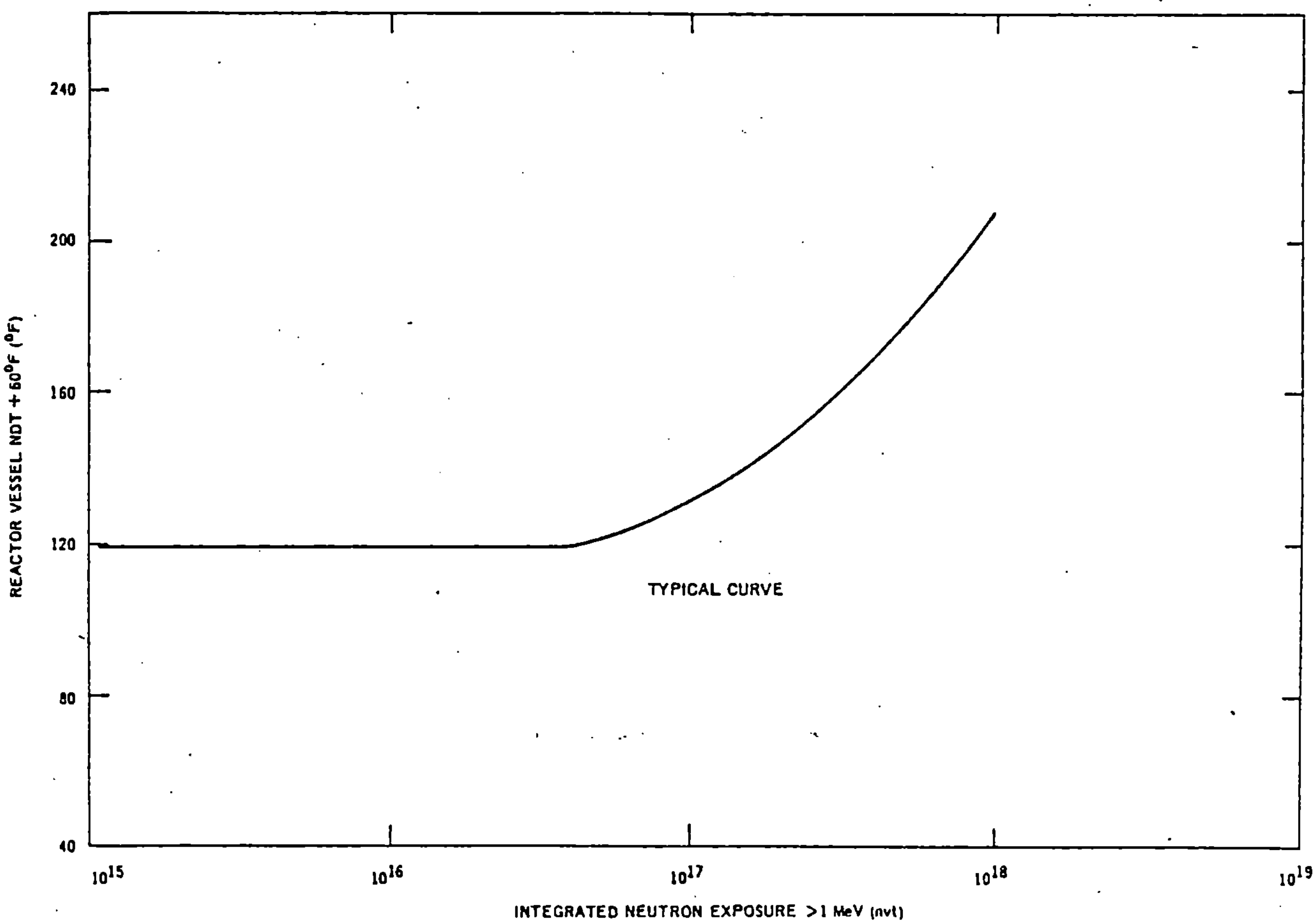


Figure 4.6.1

MINIMUM REACTOR PRESSURIZATION TEMPERATURE

TABLE 4.6.2

NEUTRON FLUX AND SAMPLES WITHDRAWAL
SCHEDULE FOR DRESDEN UNIT 2

Withdrawal Year	Part No.	Location	Comments
1977	6	Near Core Top Guide - 180°	Accelerated Sample
1980	8	Wall - 215°	
2000	7	Wall - 95°	
	9	Wall - 245°	Standby
	10	Wall - 275°	Standby

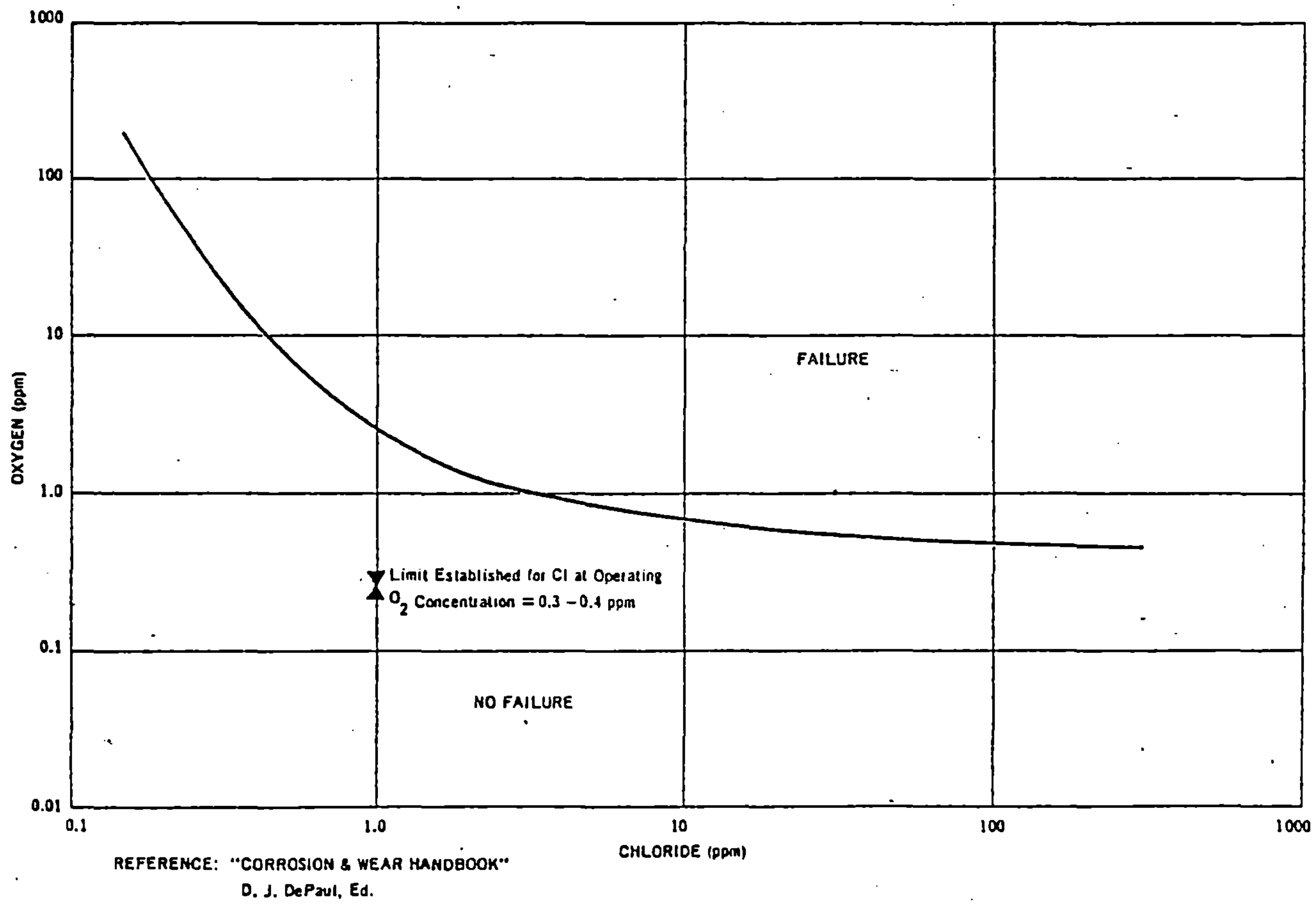


Figure 4.6.2

CHLORIDE STRESS CORROSION TEST RESULTS AT 500°F

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

During start-up periods, which are in the category of less than 100,000 pounds per hour, conductivity may exceed 2 micro-mho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 micro-mho (other than short term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.C.3 may be performed by a gamma scan.

D. Coolant Leakage - Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite a-c power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study 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.D 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.D, 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

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

The additional leakage requirements will be in effect only while the reactor is operated with the recirculation flaws detected during the 1983 Refueling Outage. The additional leakage requirements will provide more conservative detection and corrective action should the current flaws propagate thru wall.

The capacity of the drywell sump is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

The performance of reactor coolant leakage detection system will be evaluated during the first five years of station operation and the conclusions of this evaluation will be reported to the NRC.

It is estimated that the main steam line tunnel leakage detection system is capable of detecting the order of 3000 lb/hr.

The system performance will be evaluated during the first five years of plant operation and the conclusions of the evaluation will be reported to the NRC.

E. Safety and Relief Valves - The frequency and testing requirements for the safety and relief valves are specified in the Inservice Testing Program which is based on Section XI of the ASME Boiler and Pressure Vessel Code. Adherence to these code requirements provides adequate assurance as to the proper operational readiness of these valves. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as plus or minus 1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher than the reactor coolant pressure safety limit of 1375 psig is not exceeded. The safety valves are required to be operable above the design pressure (90 psig) at which the core spray subsystems are not designed to deliver full flow.

F. Structural integrity - A pre-service inspection of the components in the primary coolant pressure boundary will be conducted after site erection to assure the system is free of gross defects and as a reference base for later inspections. Prior to operation, the reactor primary system will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout life.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Inservice Inspections of ASME Code Class 1, 2 and 3 components will be performed in accordance with the applicable version of Section XI of the ASME Boiler and Pressure Vessel Code. Relief from any of the above requirements must be provided in writing by the Commission. The Inservice Inspection program and the written relief do not form a part of these Technical Specifications.

These studies show that it requires thousands of stress cycles at stresses beyond any expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results only a small number of stress cycles, at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

After five years of operation, a program for in-service inspection of piping and components within the primary pressure boundary which are outside the downstream containment isolation valve shall be submitted to the NRC.

G. Jet Pumps - Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser increases the cross sectional flow area for blowdown following the postulated design basis double-ended recirculation line break. Therefore, if a failure occurs, repairs must be made to assure the validity of the calculated consequences.

The following factors form the basis for the surveillance requirements:

A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Section 4.6.G.1 and 2.

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive or broken jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the twenty individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow. Thus the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing pump. Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operatng map point.

A nozzle-riser system failure could also generate the coincident failure of a 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.

H. Recirculation Pump Flow Mismatch

The LPCI loop selection logic has been described in the Dresden Nuclear Power Station Units 2 and 3 FSAR, Amendments 7 and 8. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions, the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

The licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. Below 80% power, the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

In addition, during the start-up of Dresden Unit 2, it was found that a flow mismatch between the two sets of jet pumps caused by a difference in recirculation loops could set up a vibration until a mismatch in speed of 27% occurred. The 10% and 15% speed mismatch restrictions provide additional margin before a pump vibration problem will occur.

ECCS performance during reactor operation with one recirculation loop out of service has not been analyzed. Therefore, sustained reactor operation under such conditions is not permitted.

I. Snubbers (Shock Suppressors)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.I.4 prohibits startup with inoperable snubbers.

When a snubber is found inoperable, a review shall be performed to determine the snubber mode of failure. Results of the review shall be used to determine if an engineering evaluation of the safety-related system or component is necessary. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the support component or system.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

All safety related mechanical snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation and attachments to the piping and anchor for indication of damage or impaired operability.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To further increase the assurance of snubber reliability, functional tests will be performed once each refueling cycle. A representative sample of 10% of the safety-related snubbers will be functionally tested. Observed failures on these samples will require testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as different entities for the above surveillance programs.

Hydraulic snubber testing will include stroking of the snubbers to verify piston movement, lock-up, and bleed. Functional testing of the mechanical snubbers will consist of verification that the force that initiates free movement of the snubber in either tension or compression is less than the maximum breakaway friction force. The remaining portion of the functional test consisting of verification that the activation (restraining action) is achieved within the specified range of acceleration in both tension and compression will not be done. This is due to the lack of competitive marketable test fixtures available for station use. Therefore, until such time as test fixtures become available, only part (i) of the test will be performed; part (ii) will not be done.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

When the cause of rejection of the snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

Monitoring of snubber service life shall consist of the existing station record systems, including the central filing system, maintenance files, safety-related work packages, and snubber inspection records. The record retention programs employed at the station shall allow station personnel to maintain snubber integrity. The service life for hydraulic snubbers is 10 years. The hydraulic snubbers existing locations do not impose undue safety implications on the piping and components because they are not exposed to excesses in environmental conditions. The service life for mechanical snubbers is 40 years, lifetime of the plant. The mechanical snubbers are installed in areas of harsh environmental conditions because of their dependability over hydraulic snubbers in these areas. All snubber installations have been thoroughly engineered providing the necessary safety requirements. Evaluations of all snubber locations and environmental conditions justify the above conservative snubber service lives.

A re-analysis of the ring header design based upon acceleration response spectra derived from the original suction header analysis report demonstrates that for normal operation plus a seismic event, neither the header nor the torus penetration are over-stressed with all snubbers inoperable. The limitation of a maximum of 3 pairs or 1 snubber from each pair inoperable out of 6 pairs is considered conservative. Since the analysis shows that the plant can operate safely indefinitely with no snubbers on the ring header the limitation on operation and startup with inoperable snubbers until January 19, 1984 is justified. This time frame is adequate to allow completion of the Mark I torus attached piping modification.

4.6 SURVEILLANCE REQUIREMENT BASES

None

3.7 LIMITING CONDITION FOR OPERATION

CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, except as permitted by Specification 3.5.F.3, 3.5.F.4, the suppression pool water volume and temperature shall be maintained within the following limits.

    a. Maximum water volume - 115,655 ft^3

    b. Minimum water volume - 112,000 ft^3

4.7 SURVEILLANCE REQUIREMENTS

CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. The surveillances are as follows:

    a. The suppression pool water level and temperature shall be checked once per day.

    b. Whenever there is indication of relief valve operation or

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

c. Maximum water temperature

(1) During normal power operation; maximum 95°F.

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10° F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 150 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F.

Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 150 psig at normal cooldown rates if the pool temperature reaches 120°F.

d. Maximum downcomer submergence is 4.00 ft.

e. Minimum downcomer submergence is 3.67 ft.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

f. If Specifications 3.7.A.1.a or 3.7.A.1.b are not met and suppression pool water volume cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MW(t).

a. Primary containment leakage rates are defined from:

(1) The calculated peak containment internal pressure, P_a, is equal to 48 psig.

(2) The containment vessel reduced test pressure, P_t, is equal to 25 psig.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

2. The primary containment integrity shall be demonstrated by conducting Primary Containment Leak Tests and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and references therein.

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at approximately equal intervals during each 10 year plant in-service inspection interval at either P_a or P_t with the last being done during the 10-year in-service inspection shutdown.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(3) The maximum allowable leakage rate at a pressure of P_a, L_a, is equal to 1.6 percent by weight of the containment air per 24 hours at 48 psig.

(4) The maximum allowable test leakage rate at a pressure of P_t, L_t, is less than or equal to L_a (L_{tm}/L_{am}). If L_{tm}/L_{am} is greater than 0.7, L_t is (specified as equal to) L_a (P_t/P_a) E 1/2.

(5) The total measured leakage rates at pressures of P_a and P_t are L_{am} and L_{tm}, respectively.

b. When primary containment integrity is required, primary containment leakage rates shall be limited to:

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. If any periodic Type A test fails to meet either 75 percent of L_a or 75 percent of L_t the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(1) An overall integrated leakage rate for Type A tests of:

(a) L_{am} less than or equal to 75 percent of L_a.

(b) L_{tm} less than or equal to 75 percent of L_t.

(2) (a) A combined leakage rate of less than or equal to 60 percent of L_a for all testable penetrations and isolation valves subject to Type B and C tests except for main steam isolation valves.

(b) A leakage rate of less than or equal to 3.75 percent of L_a for any one air lock when pressurized to 10 psig.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(c) 11.5 SCF per hour for any main steam isolation valve at a test pressure of 25 psig.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

c. If two consecutive Type A tests fail to meet either 75 percent of L_a or 75 percent of L_t, a Type A test shall be performed at each shutdown for refueling or approximately every 18 months until two consecutive Type A tests meet the above requirements, at which time the normal test schedule may be resumed.

d. The accuracy of each Type A test shall be verified by a supplemental test which:

(1) Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 25 percent of L_a or 25 percent of L_t.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

(2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

(3) Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 9 scfm.

e. Type B and C tests shall be conducted at P_a, at intervals no greater than 24 months except for tests involving:

(1) Main steam line isolation valves which shall be tested at a pressure of 25 psig each operating cycle.

(2) Bolted double-gasketed seals which shall be tested at a pressure of 48 psig

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

whenever the seal is closed after being opened and each operating cycle.

(3) Air locks which shall be tested at 10 psig each operating cycle.

f. Continuous Leak Rate Monitor

(1) When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system make-up requirements.

(2) This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

g. The interior surfaces of the drywell shall be visually inspected each operating cycle for evidence of deterioration.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in Specifications 3.7.A.3.b below, two pressure suppression chamber - reactor building vacuum breakers in each line shall be operable at all times when the primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber - reactor building air operated vacuum breakers shall not exceed 0.5 psid. The vacuum breakers shall open fully when subjected to a force equivalent to or less than 0.5 psid acting on the valve disk.

b. From and after the date that one of the pressure suppression chamber - reactor building vacuum breakers is made

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. The pressure suppression chamber - reactor building vacuum breakers and associated instrumentation, including setpoint, 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

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

or found to be inoperable for any reason, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the procedure does not violate primary containment integrity.

4. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. When primary containment is required, all pressure suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated in Specifications 3.7.A.4.b, c and d., below, pressure suppression chamber - drywell vacuum breakers shall be considered operable if:

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

breaker does not exceed the force specified in Specification 3.7.A.3.a. and each vacuum breaker shall be inspected and verified to meet design requirements.

4. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. Periodic Operability Tests

Once each month each pressure suppression chamber - drywell vacuum breaker shall be exercised. Operability of position switches and position indicators and alarms shall be verified.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(1) The valve is demonstrated to open fully with the applied force at all valve positions not exceeding the equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

(2) The valve can be closed by gravity when released after being opened by manual means, to within the equivalent of 1/16" at all points along the seal surface of the disk.

(3) The position alarm system will annunciate in the control room if the valve opening exceeds the equivalent of 1/16" at all points along the seal surface of the disk.

b. Reactor operation may continue provided that no more than one

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. During each refueling outage:

(1) The pressure

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

quarter of the number of pressure suppression chamber - drywell vacuum breakers are determined to be inoperable provided that they are secured or known to be in the closed position.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open.

(2) Vacuum breakers position indication and alarm systems shall be calibrated and functionally tested.

(3) At least 25% of the vacuum breakers shall be inspected such that all vacuum breakers shall have been inspected following every fourth refueling outage. If deficiencies are found, all vacuum breakers shall be inspected and deficiencies corrected.

(4) A drywell to suppression chamber leak test shall

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

c. Reactor operation may continue for fifteen (15) days provided that at least one position alarm circuit for each operable vacuum breaker is operable and each suppression chamber - drywell vacuum breaker is physically verified to be closed immediately and daily thereafter.

5. Oxygen Concentration

a. The primary containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

demonstrate that with initial differential pressure of not less than 1.0 psi, the differential pressure decay rate does not exceed the rate which would occur through a 1-inch orifice without the addition of air or nitrogen.

5. Oxygen Concentration

The primary containment oxygen concentration shall be measured and recorded on a weekly basis.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

cooling pressure above 90 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. Deinerting may commence 24 hours prior to a shutdown.

6. Containment Atmospheric Dilution and Purge

a. Whenever the reactor is in power operation the normal containment makeup inerting system shall be operable and capable of supplying nitrogen to containment for atmosphere dilution if required by post LOCA conditions. If this specification cannot be met, the system must be restored to an operable condition within 7 days or

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

6. Containment Atmospheric Dilution and Purge

a. Once a month, the valves in the nitrogen makeup system shall be actuated to determine operability.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

the reactor must be taken out of power operation.

b. Whenever either Unit 2 or 3 is in power operation, the containment makeup inerting system nitrogen storage tank level liquid level shall be equal to or greater than 60 inches. If this minimum level cannot be met, the minimum level shall be restored within 7 days or both Unit 2 and Unit 3 shall be taken out of power operation. During such seven day interval the minimum level shall be 20 inches or both Unit 2 and 3 shall be taken out of power operation.

c. Whenever the reactor is in power operation, the primary containment purge system shall be operable. If this specification cannot be met the reactor must be taken out of power operation.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. The level in the liquid N_2 storage tank shall be recorded weekly and after reinerting containment.

c. Once a month , the valves in the purge line to the standby gas treatment system shall be actuated to determine operability.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

d. Whenever the reactor is in power operation, the primary containment oxygen sampling system shall be operable. If this specification cannot be met, the system must be restored to an operable condition within 7 days or the reactor must be taken out of power operation.

e. The maximum containment repressurization pressure using the containment makeup inerting system shall be 26 psig.

7. Drywell Suppression Chamber Differential Pressure

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.00 psid except as specified in (1) and (2) below:

(1) This differential shall be established within the 24

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

d. The containment oxygen analyzing system shall be functionally tested once per week and shall be calibrated once per 6 months.

7. Drywell Suppression Chamber Differential Pressure

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift when the differential pressure is required.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

hour period subsequent to placing the reactor mode switch into the run mode during a startup and may be relaxed 24 hours prior to a rector shutdown when the provisions of 3.7.A.5 (b) apply.

(2) This differential may be decreased to less than 1.00 psid for a maximum of 4 hours during required operability testing of the drywell pressure suppression chamber vacuum breakers, HPCI testing and reactor pressure relief valve testing.

b. If the Specifications of 3.7.A.7.a cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

shall be initiated and the rector shall be in a cold shutdown condition in the following 24 hours.

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment system circuits shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1(a) and (b).

   a. After one of the standby gas treatment system circuits is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment system shall be demonstrated to be operable within 2 hours

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

B. Standby Gas Treatment System

1. At least once per month, initiate from the control room 4000 cfm (plus or minus 10%) flow through both circuits of the standby gas treatment system for at least 10 hours with the circuit heaters operating at rated power.

   a. Within 2 hours from the time that one standby gas treatment system circuit is made or found to be inoperabe for any reason and daily thereafter for the next succeeding seven days, initiate from the control room 4000 cfm (plus or minus 10%) flow through the operable circuit of the standby gas treatment system for at least 10 hours

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

and daily thereafter. Within 36 hours following the 7 days, the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.1.(a) through (d).

b. If both standby gas treatment system circuits are not operable, within 36 hours the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.1.(a) through (d).

2. Performance Requirement (See Note 1, Page 3/4.7-24)

a. Periodic Requirements:

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

with the circuit heaters operating.

2. Performance Requirement Tests (See Note 1)

a. At least once per 720 hours of system operation; or once per operating cycle, but not to exceed 18 months, whichever occurs first; or following painting, fire, or chemical

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(1) The results of the in-place DOP tests at 4000 cfm (plus or minus 10%) on HEPA filters shall show less than or equal to 1% DOP penetration.

(2) The results of in-place halogenated hydrocarbon tests at 4000 cfm (plus or minus 10%) on charcoal banks shall show less than or equal to 1% penetration.

(3) The results of laboratory carbon sample analysis shall show greater than or equal to 90% methyl iodide removal efficiency when tested at 130°C, 95% R. H.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

release in any ventilation zone communicating with the system while the system is operating that could contaminate the HEPA filters or charcoal absorbers; perform the following:

(1) In-place DOP test the HEPA filter banks to verify leak tight integrity

(2) In-place test the charcoal adsorber banks with halogenated hydrocarbon tracer to verify leak tight integrity.

(3) Remove one carbon test canister from the charcoal adsorber. Subject this sample to a laboratory analysis to verify methyl iodide removal efficiency.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

b. The system shall be shown to operate.

3. Post Maintenance Requirements (See Note 1; Page 3/4.7-24)

   a. After any maintenance or testing that could affect the HEPA filter or HEPA filter mounting frame leak tight integrity, the results of the inplace DOP tests at 4000 cfm (plus or minus 10%) on HEPA filters shall show less than or equal to 1% DOP penetration in

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. At least once per operating cycle, but not to exceed 18 months, the following conditions shall be demonstrated:

   (1) Pressure drop across the combined filters of each standby gas treatment system circuit is less than 6 inches of water at 4000 cfm (plus or minus 10%) flow rate.

   (2) Operability of inlet heater at rated power.

   (3) Automatic initiation of each standby gas treatment system circuit.

3. Post Maintenance Testing (See Note 1)

   a. After any maintenance or testing that could affect the leak tight integrity of the HEPA filters, perform in-place DOP tests on the HEPA filters in accordance with Specification 3.7.B.2.a. (1).

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

accordance with Specification 3.7.B.2.a(1).

b. After any maintenance or testing that could affect the charcoal adsorber leak tight integrity, the results of in-place halogenated hydrocarbon tests at 4000 cfm (plus or minus 10%) on charcoal adsorber banks shall show less than or equal to 1% penetration in accordance with Specification 3.7.B.2.a(2).

c. The results of in-place air distribution tests shall show the air distribution is uniform within plus or minus 20% to each HEPA filter when tested initially and after any maintenance or testing that could affect the air distribution within the standby gas treatment system.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. After any maintenance or testing that could affect the leak tight integrity of the charcoal adsorber banks, perform halogenated hydrocarbon tests on the charcoal adsorbers in accordance with Specification 3.7.B.2.a.(2).

c. Perform an air distribution test on the HEPA filter bank initially and after any maintenance or testing that could affect the air distribution within the standby gas treatment system. The test shall be performed at 4000 cfm (plus or minus 10%) flow rate.

4. Standby gas treatment system surveillance shall be performed as indicated below:

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

Note 1: Because the accomplishment of Specifications 3.7.B.2, 3.7.B.3, 4.7.B.2, and 4.7.B.3 will require equipment modifications, their implementation will be delayed until about December 31, 1976. Until that time, the surveillance requirements of Specification 4.7.B.4 shall apply.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

a. At least once per operating cycle it shall be demonstrated that:

(1) Pressure drop across the combined high-efficiency and charcoal filters is less than 5.7 inches of water at 400 cfm and

(2) Inlet heater delta T shall be a minimum of 14°F at 4000 cfm.

b. At least once during each scheduled secondary containment leak rate test, whenever a filter is changed, whenever work is performed that could affect the filter system efficiency and at intervals not to exceed six months between refueling outages, it shall be demonstrated that

(1) The removal efficiency of the particulate filters is not less than 99% for particulate matter larger than 0.3 micron.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. Secondary Containment

1. Secondary containment integrity shall be maintained during all modes of plant operation except when all of the following conditions are met.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

(2) The removal efficiency of the charcoal filters is not less than 99% for iodine.

c. At least once each five years removable charcoal cartridges shall be removed and absorption shall be demonstrated.

d. At least once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.

e. At least once per operating cycle manual operability of the bypass valve for filter cooling shall be demonstrated.

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

3.7 LIMITING CONDITION FOR OPERATION (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.

4.7 SURVEILLANCE REQUIREMENTS (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 a 1/4 inch of water vacuum under calm wind (less than 5 mph) conditions with a filter train flow rate of not more than 4000 cfm.

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 a 1/4 inch of water vacuum under calm wind (less than 5 mph) conditions with a filter train flow rate of not more than 4000 cfm, shall be demonstrated at each refueling outage prior to refueling.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

d. The fuel cask or irradiated fuel is not being moved in the reactor building.

2. The doors of the core spray and LPCI pump compartments shall be closed at all times except during passage in order to consider the core spray and the LPCI subsystems operable.

3. If Specification 3.7.C.1 cannot be met procedures shall be initiated to establish conditions listed in Specification 3.7.C.1.a through d.

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 SURVEILLANCE REQUIREMENTS (Cont'd.)

d. For the 24 hour period commencing on February 4, 1979, at 1:00 p.m., reactor operation is permitted provided that a negative pressure of 0.2 inches of water is maintained in the Unit 2 reactor building areas below the refueling floor.

2. Whenever the LPCI and core spray subsystems are required to be operable, the doors of the core spray and LPCI pump compartments shall be verified to be closed weekly.

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.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. In the event any isolation valve specified in Table 3.7.1 becomes inoperable, reactor power operation may continue provided at least one valve in

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. At least once per operating cycle the instrument line flow check valves shall be tested for proper operation.

c. At least once per quarter:

(1) All normally open power-operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened.

(2) With the reactor power less than 50% of rated, trip main steam isolation valves (one at a time) and verify closure time.

d. At least twice per week the main steamline power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

2. Whenever an isolation valve listed in Table 3.7.1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

4. The temperature of the main steamline air pilot valves shall be less than 170°F except as specified in 3.7.D.5 below.

5. From and after the date that the temperature of any main steamline air pilot valve is found to be greater than 170°F, reactor operation is permissible only during the succeeding seven days unless the temperature of such valve is sooner reduced to less than 170°F, provided the main steamline isolation valves are operable.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

3. The temperature of the main steamline air pilot valves shall be recorded daily.

4. When it is determined that the temperature of any main steamline air pilot valve is greater than 170°F, the main steamline isolation valves shall be demonstrated to be operable immediately and daily thereafter. The demonstration of operability shall be according to Specification 4.7.D.1.d.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

6. When it is determined that it will take longer than seven days to reduce the temperature of any main steamline air pilot valve to less than 170°F, a report detailing the circumstances and the estimated date for returning the air pilot valve temperature to a value less than 170°F shall be submitted to the NRC prior to the end of the seven day period.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

TABLE 3.7.1

PRIMARY CONTAINMENT ISOLATION

	Isolation Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
	1	Main Steam Line Isolation	4	4	3 * T * 5	0	GC
	1	Main Steam Line Drain	1		* 35	C	SC
	1	Main Steam Line Drain		1	* 35	C	SC
Note 1	1	Recirculation Loop Sample Line	1	1	* 5	0	SC
	1	Isolation Condenser Vent to main steam line	1		* 5	0	GC
	1	Isolation Condenser Vent to main steam line		1	* 5	0	GC
	2	Drywell floor drain		2	* 20	0	GC
	2	Drywell Equipment drain		2	* 20	0	GC
	2	Drywell Vent		2	* 10	C	SC
	2	Drywell Vent Relief		1	* 15	C	SC
	2	Drywell Inert and purge #1601-21		1	* 10	C	SC
	2	Drywell N_2 Makeup #1601-59	1		* 15	0	GC
	2	Drywell and Suppression Chamber N_2 Makeup #1601-57		1	* 15	0	GC
	2	Drywell and Suppression Chamber Inert #1601-55		1	* 15	0	GC
	2	Suppression Chamber N_2 Makeup #1601-58		1	* 15	C	GC
	2	Suppression Chamber inert and purge #1601-56		1	* 10	0	GC
	2	Drywell and Suppression chamber vent from reactor building #1601-22		1	* 10	C	SC
	2	Drywell vent to standby gas treatment system		1	* 10	C	SC
	2	Suppression chamber vent		1	* 10	C	SC
	2	Suppression chamber vent relief		1	* 15	C	SC
Note 1	2	Drywell air sampling system		10	* 5	0	GC
	2	Drywell Pneumatic Supply Isolation		2	* 10	0	GC
	2	Torus to Condenser Drain		2	* 10	C	SC
	3	Cleanup demineralizer System	1		* 30	0	GC
	3	Cleanup demineralizer System		2	* 30	0	GC
	3	Shutdown cooling system	2		* 40	C	SC
	3	Shutdown cooling system		1	* 40	C	SC
	3	Shutdown cooling system		1	* 40	C	SC
	3	Reactor head cooling line		1	* 15	C	SC
	4	HPCI Turbine Steam supply	1		* 25	0	GC
	4	HPCI Turbine Steam supply		1	* 25	0	GC
	5	Isolation condenser steam supply	1		* 30	0	GC
	5	Isolation condenser steam supply		1	* 30	0	GC
	5	Isolation condenser condensate return	1		* 30	0	GC
	5	Isolation condenser condensate return		1	* 30	C	SC
		Feedwater Check Valves	2	2	NA	0	Process
		Control Rod Hydraulic Return Check Valves	1	1	NA	0	Process
		Reactor Head Cooling Check Valves	1		NA	C	Process
		Standby Liquid Control Check Valves	1	1	NA	C	Process

Notes: (See Next Page)

Table 3.7.1 (Cont'd.)

Notes for Table 3.7.1

* Less than or equal to

Note 1; Valve can be reopened after isolation for sampling.

Key:
O = Open
C = Closed
SC = Stays Closed
GC = Goes Closed

Note: Isolation groupings are as follows:

GROUP 1: The valves in Group 1 are closed upon any one of the following conditions:

1. Reactor low-low water level
2. Main steam line high radiation
3. Main steam line high flow
4. Main steam line tunnel high temperature
5. Main steam line low pressure

GROUP 2: The actions in Group 2 are initiated by any one of the following conditions:

1. Reactor low water level
2. High drywell pressure

GROUP 3: Reactor low water level alone initiates the following:

1. Cleanup demineralizer system isolation
2. Shutdown cooling system isolation
3. Reactor Head cooling isolation.

GROUP 4: Isolation valves in the high pressure coolant injection system (HPCI) are closed upon any one of the following signals:

1. HPCI steam line high flow
2. High temperature in the vicinity of the HPCI steam line
3. Low reactor pressure

GROUP 5: Isolation valves associated with the isolation condenser are closed upon indication of either high isolation condenser steam or condensate flow.

3.7 LIMITING CONDITION FOR OPERATION BASES

A. Primary Containment - The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to preclude a peak fuel enthalpy of 280 cal/gm. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offers a significant barrier to keep off-site doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. (Ref. Section 5.2.3 FSAR)

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 48 psig which is below the design of 62 psig. Maximum water volume of 115,655 ft^3 results in a

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

downcomer submergence of 4 feet and the minimum volume of 112,000 ft^3 results in a submergence approximately 4 inches less. The majority of the Bodega tests [9] were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

The maximum temperature at the end of blowdown tested during the Humboldt Bay [10] and Bodega Bay tests were 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for emergency core cooling systems operability as explained in basis 3.5.F.

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(9) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(10) Robbins, C. H., "Tests of a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Using a 50°F rise (Section 5.2.3.1 SAR) in the suppression chamber water temperature and a maximum initial temperature of 95°F, a temperature of 145°F is achieved which is well below the 170°F temperature which is used for complete condensation.

For an initial maximum suppression chamber water temperature of 95°F and assuming the normal complement of containment cooling pumps (2 LPCI pumps and 2 containment cooling service water pumps) containment pressure is not required to maintain adequate net positive suction head (NPSH) for the core spray, LPCI and HPCI pumps.

If a loss of coolant accident were to occur when the reactor water temperature is below 330°F, the containment pressure will not exceed the 62 psig design pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperatures above 212°F provides additional margin above that available at 330°F.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber 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% flow pipes each containing two vacuum relief breakers. Operation of either flow pipe will maintain the pressure differential less than 1 psig, the external design pressure of the primary containment. Redundancy of lines justifies reactor operation with one valve out of service for repairs for a period of seven days.

The capacity of the pressure suppression chamber-drywell vacuum breakers is designed to limit the pressure differential between the suppression chamber and drywell to not greater than 0.5 psi during post-accident drywell cooling. They are sized on the basis of the Bodega Bay pressure suppression system test.

Based on these tests, design flow from the suppression chamber to the drywell can be obtained with three (3) of the vacuum breakers closed without exceeding the 0.5 psi differential pressure limit.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Reactor operation is permissible if the bypass area between the primary containment drywell and suppression chamber does not exceed an allowable area. The allowable bypass area is based upon analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is equivalent to all vacuum breakers open the equivalent of 1/16" at all points along the seal surface of the disk (see Dresden Special Report No. 23).

Each drywell-suppression chamber vacuum breaker is fitted with a redundant pair of position switches which provide signals of disk position to panel mounted indicators and annunciate an alarm in the control room if the disk is open more than allowable. The alarm systems meet the intent of IEEE 279 standards. The quality of the alarm system justifies continued reactor operation for 15 days between differential pressure decay rate tests if one alarm system is inoerable for one or more operable vacuum breakers.

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss of coolant accident would lead to the liberation of sufficient hydrogen to a result in a flammable concentration in the containment. Subsequent ignition of the hydrogen if it is present in sufficient quantities to result in excessively rapid recombination, could lead to failure of the containment to maintain a low leakage integrity. The 4% oxygen concentration minimizes the possibility of hydrogen combustion following a loss of coolant accident.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss of coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

In order to ensure that the containment atmosphere remains inerted, i.e. the oxygen-hydrogen mixture remains below the flammable limit, the capability to inject nitrogen into the containment after a LOCA is provided. During an interim period prior to installation of the Containment Atmospheric Dilution (CAD) system the normal inerting nitrogen makeup system will be available for post-LOCA nitrogen injection.

By maintaining a minimum level of 60 inches in the liquid nitrogen storage tank, a minimum of 200,000 cubic feet of nitrogen is assured which corresponds to a seven day supply. During reinerting of containment the supply may temporarily drop below a seven day supply but at no time is the inventory to drop below a minimum of a two day supply (20 inch level). By normally maintaining at least a 7-day supply of nitrogen on site and maintaining a minimum of a two day supply there will be assurance of sufficient time after the occurrence of a LOCA for obtaining additional nitrogen supply.

A system for controlled purging through the Standby Gas Treatment system is necessary to limit repressurization pressure from post LOCA nitrogen addition in a manner which will limit offsite doses. Controlled purging also provides a backup method of controlling hydrogen concentration.

A means to determine post LOCA containment oxygen concentration is necessary to readily enable the reactor operator to take appropriate action to control containment atmosphere. In the interim, prior to installation of the CAD and associated monitoring systems, the containment oxygen analyzing system will be available.

The maximum containment repressurization pressure of 26 psi provides adequate margin to containment design pressure and a delay time prior to purge which results in acceptable purge doses.

Following a LOCA, periodic operation of the drywell and torus sprays will be used to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and O_2 concentration exceeds 4%.

(15) "Dresden Nuclear Generating Plant Units 2 & 3 Short Term Program Plant Unique Torus Support and Attached Piping Analysis", August 1976 NUTECH Report COM-01-040.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed (Reference 15) which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.00 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.67 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

B. Standby Gas Treatment System and
C. Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

Only one of the two standby gas treatment system circuits 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. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is placed in a condition that does not require a standby gas treatment system.

While only a small amount of particulates are released from the primary containment as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal 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% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed).*

*Bases in parentheses will not be applicable until about December 31, 1976, when equipment modifications are completed to allow increased testing.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. One standby gas treatment fan is designed to automatically start upon containment isolation and to maintain the reactor building pressure to approximately a negative 1/4-inch water guage pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter system is designed to start automatically. Each of the two fans has 200% capacity. (Ref. Section 5.3.2 SAR.) If one standby gas treatment system circuit is inoperable, the other circuit will be tested daily. This substantiates the availability of the operable circuit and results in no added risk; thus, reactor operation or refueling operation can continue. If neither circuit is operable the plant is brought to a condition where the system is not required.

While only a small amount of particulates are released from the pressure suppression chamber system as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The inplace test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates. Laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

D. Primary Containment Isolation Valves - Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

4.7 SURVEILLANCE REQUIREMENT BASES

A. Primary Containment

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The primary containment preoperational 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 48 psig which would rapidly reduce to 25 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 25 psig within 10 seconds, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay[12].

The design pressure of the drywell and absorption chamber is 62 psig[12]. The design leak rate is 0.5%/day at a pressure of 62 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were 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 2.0%/day at 48 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID 14844, the maximum total whole body passing cloud dose is about 8 rem and the maximum total thyroid dose is about 185 rem at the site boundary over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population distance of 5 miles are lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, the 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.

(12) Section 5.2 of the FSAR

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The maximum allowable test leak rate is 1.6%/day at a pressure of 48 psig. This value for the test condition was derived from the maximum allowable accident leak rate of about 2.0%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air whereas under test conditions the test medium would be air or nitrogen at ambient conditions. Considering the difference in mixture composition and temperatures, the appropriate correction factor applied was 0.8 and determined from the guide on containment testing(13).

Although the dose calculations suggest that the accident leak rate could be allowed to increase to about 3.2%/day before the guideline thyroid doses value given in 10 CFR 100 would be exceeded, establishing the test limit of 1.6%/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 or the allowable test leak rate 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 frequency is based on the AEC guide for developing leak rate testing and surveillance of reactor containment vessels(14). 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.

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(13) TID 20583, Leakage Characteristics of Steel Containment Vessel and the Analysis of Leakage Rate Determinations.

(14) Technical Safety Guide, "Reactor Containment Leakage Testing and Surveillance Requirements USAEC, Division of Safety Standards, Revised Draft, December 15, 1966.

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The data reduction methods of the applicable ANSI standard will be applied for the integrated leak rate tests as specified in Appendix J of 10 CFR 50.

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 double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 48 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure. 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. The personnel air lock is tested at 10 psig, because the inboard door is not designed to shut in the opposite direction.

The results of the loss-of-coolant accident analyses presented in Amendment No. 18 of the SAR indicates that fission products would not be released directly to the environs because of leakage from the main steam line isolation valves due to holdup in the steam system complex. Although this effect would indicate that an adequate margin exists with regard to the release of fission products, a program will be undertaken to further reduce the potential for such leakage to bypass the standby gas treatment system.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

Surveillance of the reactor building-pressure suppression chamber vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness test). These vacuum breakers are normally in the closed position and open only during tests or a post accident condition. As a result, a testing frequency of 3 months for operability is considered justified for this equipment. Inspections and calibrations are performed during refueling outages, this frequency being based on experience and judgment.

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

Pressure suppression chamber-drywell vacuum breakers monthly operability tests are performed to check capability of the disks to open and close and to verify that the position indication and alarm circuits function properly. The disk opens during post accident conditions and occasionally during transient additions of energy to the torus through relief valves. This infrequent operation of the disks and the quality of equipment justify the frequency of operability tests of this equipment.

Measurement of force to open, calibration of position switches, inspection of equipment and functional testing are performed during each refueling outage. This frequency is based on equipment quality, experience and judgment. Also a stringent differential pressure decay rate test is performed during refueling outages. This test is performed to verify that total leakage paths between the drywell and suppression chamber are not in excess of the equivalent to a 1-inch orifice.

This small leakage path is only a small fraction of the allowable, thus integrity of the containment system is assured prior to startup following each refueling outage (See Dresden Special Report No. 23).

When a suppression chamber-drywell vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights at the remote test panel are designed to function as follows:

Full Closed (Closed to less than or equal to 1/16" open)	2 Green - On
Intermediate Position (greater than 1/16" open to full open)	2 Green - Off

The remote test panel consists of two green lights for each of the twelve valves. The two switches controlling the green lights are adjusted to provide indication and alarm if a disk opening occurs that is equivalent to one-sixteenth of an inch (1/16") at all points around the circumference of the valve disk. The control room alarm circuits for each vacuum breaker are redundant and fail safe. This assures that no single failure will defeat alarming the control room when a valve is open beyond allowable and when power to the switches fails. The alarm is needed to alert the operator that action must be taken to correct a malfunction or that system degradation has

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

occurred and additional testing is required immediately. The frequency of testing the alarms is based on experience and quality of the equipment. During each refueling outage, three drywell-suppression chamber vacuum breakers will be inspected to assure sealing surfaces and components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in 1/10 of the design lifetime is extremely conservative.

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 once a week the oxygen concentration will be determined as added assurance.

Recording N_2 storage tank level weekly and after containment reinerting provides assurance of an adequate onsite supply.

Weekly testing of the oxygen analyzer and monthly actuation of the nitrogen makeup and purge line valves provides assurance of operational readiness.

B. Standby Gas Treatment System and
C. Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain the design negative pressure within the secondary containment provides an adequate test of the reactor building isolation valves and the standby gas treatment system. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system operational capability. (The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Standby gas treatment system in-place testing procedures will be established utilizing applicable sections of ANSI N510-1975 standard as a procedural guideline only. Operation of the standby gas treatment system every month for 10 hours will reduce the moisture buildup on the adsorbent. If painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

analysis should be performed as required for operational use. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52, Revision 1 (June 1976). The charcoal adsorber efficiency test procedures will allow for the removal of one representative sample cartridge and testing in accordance with the guidelines of Table 3 of Regulatory Guide 1.52, Revision 1 (June 1976). The sample will be at least two inches in diameter and a length equal to the thickness of the bed. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system will be replaced. High efficiency particulate filters are installed before and after the charcoal filters to prevent clogging of the carbon adsorbers and to minimize potential release of particulates to the environment. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by in-place testing with DOP as the testing medium. Any HEPA filters found defective will be replaced with filters qualified pursuant to regulatory guide position C.3.d of Regulatory Guide 1.52, Revision 1 (June 1976). Once per operating cycle demonstration of HEPA filter pressure drop, operability of inlet heaters at rated power, air distribution to each HEPA filter, and automatic initiation of each standby gas treatment system circuit is necessary to assure system performance capability).*

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein are adequate to prevent loss of more coolant from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, this isolation valve closure time is sufficient to prevent uncovering the core.

* Bases in parentheses will not be applicable until about December 31, 1976, when equipment modificatons are completed to allow increased testing.

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds. However, for added margin the Technical Specifications require a valve closure time of not greater than 5 seconds.

For reactor coolant system temperature less than 212°F, the containment could not become pressurized due to a loss of coolant accident. The 212°F limit is based on preventing pressurization of the reactor building and rupture of the blowout panels. These valves are highly reliable, have low service requirement and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. (Ref. Section 5.2.2 and Table 5.2.4 SAR.) The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a more reliable system.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The containment is penetrated by a large number of small diameter instrument lines. A program for periodic testing and examination of the floor check valves in these lines is performed similar to that described in Amendment No. 22, Millstone Unit 1, Dkt. 50-245.

3.8 LIMITING CONDITION FOR OPERATION

RADIOACTIVE MATERIALS

Applicability:

Applies to the radioactive effluents from the plant.

Objective:

To assure that radioactive material is not released to the environment in an uncontrolled manner and to assure that any material released is kept as low as practicable and, in any event, is within the limits of 10 CFR 20.

Specification:

A. Airborne Effluents

1. Radioactive gases released from the reactor building ventilation stack and plant chimney shall be continuously monitored. To accomplish this, at least one reactor building ventilation stack monitoring system and plant chimney monitoring system shall be operable at all times. During the period when plateout tests are being performed on the chimney monitoring system and the reactor is operating at a steady state the steam

4.8 SURVEILLANCE REQUIREMENT

RADIOACTIVE MATERIALS

Applicability:

Applies to the periodic monitoring and recording of radioactive effluents.

Objective:

To ascertain that radioactive releases are within allowable values.

Specification:

A. Airborne Effluents

1. The plant chimney and reactor building ventilation stack monitoring systems shall be functionally tested and calibrated every three months.

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

jet air ejector monitors may be used to satisfy the plant chimney monitoring requirements.

The Unit 2/3 plant chimney gas sampling system may be out of service for 48 hours for the purpose of installing the high range noble gas monitor as long as the following conditions are satisfied:

a. Both units are at steady state conditions with the recombiners and charcoal adsorbers in service for the operating unit(s).

b. The chimney release rate must be shown by calculation to be less than the limits of 3.8.A.2.a. and b., assuming the charcoal adsorbers are bypassed on both units.

c. Both offgas monitors on Unit 2 and Unit 3 must be operational and the monitor

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

reading correlated to the chimney release rate based on the conservative assumption of both units' charcoal adsorbers being bypassed.

d. If the provisions of 3.8.A.1.a., b. or c. cannot be met, an orderly load reduction of the unit(s) shall be initiated immediately.

Due to the existence of the Dresden Unit 1 and Unit 2/3 stacks in close vicinity, a set of equations are needed to express the airborne effluents limits. The symbols in the equations stand for the following:

Q_1 = release rate from Unit 1 plant chimney

Q_2 = release rate from the Units 2/3 plant chimney with only Unit 2 or only Unit 3 operating (not both)

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

$Q_{2,3}$ = release rate from the Units 2/3 plant chimney with both units operating

Q_{RS} = release rate from Units 2 and 3 reactor building ventilation stack

2. a. The site release rate for gross activity, except for halogens and particulates with half lives longer than eight days, shall not exceed:

$$\frac{Q_1}{0.56} + \left(\frac{Q_2}{0.7} \text{ or } \frac{Q_{2,3}}{0.9}\right) + \frac{Q_{RS}}{0.09} \text{ less than or equal to } 1.0$$

where Q is measured in Curies/sec.

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

2. a. Station records of gross ventilation stack and plant chimney release rate of gaseous activity shall be maintained on an hourly basis to assure that the specified rates are not exceeded and to yield information governing general integrity of the fuel cladding. Records of isotopic analyses shall also be maintained. Within one month after initial commercial service of the unit, an isotopic analysis will be made of the gaseous activity release rate. From this sample a ratio of long lived to short lived activity will be established. Daily

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

b. In addition to any other requirement of these technical specifications the licensee has volunteered:

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

samples of off-gas will be taken and gross ratio of long lived to short lived activity determined. When the daily samples indicate a change in the ratio of greater than 20% from the ratio established by the previous isotopic analysis, a new isotopic analysis will be performed.

A new isotopic analysis of off-gas will be performed at least quarterly. Gaseous release of tritium shall be calculated on a monthly basis from measured data.

b. Station records of release of iodines shall be maintained on the basis of all stack and plant chimney filter cartridges counted. The filter cartridges shall be counted weekly, when the measured release rate of gross beta-gama activity is less than 10% of the release limit specification 3.8.A.2.a, other-

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

(1) During reactor power operation of Units 2 and/or 3, operating procedures will be implemented to reduce release rates to those consistent with 3.8.E of these specifications prior to the release rate for gross activity, except for halogens and particulates with half lives longer than eight days, exceeding 0.105 ci/sec with Unit 2 or 3 operating alone, or 0.135 ci/sec for both units operating simultaneously.

(2) The release rates specified in 3.8.A.2.b.(1) shall not be exceeded for a time period in excess of that established by the following equations:

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

wise the cartridges shall be counted at least twice a week. Particulate isotopic analysis shall be made and recorded quarterly.

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

a. Dresden 2 or 3 operating
$t = 560/Q_X$
(Ci - hr/sec)

b. Dresden 2 and 3 operating
$t = 720/Q_Y$
(Ci - hr/sec)

Where:

t = cumulative hours of operation permitted at release rate Q_X or Q_Y, or above in the 12 months ending with the month for which the calculation is made

Q_X = release rate above 0.105 ci/sec for Dresden Unit 2 or Unit 3 operating separately

Q_Y = release rate above 0.135 ci/sec when both Dresden Units 2 and 3 are operating simultaneously.

(3) If the limits of 3.8.A.2.b.(1) are exceeded for a period of greater than 48 hours, the

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

licensee shall then notify the Director, Division of Reactor Licensing in writing within 48 hours of its plans for reducing the effluent release rate to a level which is consistent with Section 3.8.E of these specifications.

c. The summation of release rates of halogens and particulates with half lives longer than 8 days released to the environs as part of the airborne effluents shall not exceed:

$$\frac{Q_1}{2.4\times10^{-6}} + \left(\frac{Q_2}{3.5\times10^{-6}} \text{ or } \frac{Q_{2,3}}{4.3\times10^{-6}}\right)^* + \frac{Q_{RS}}{0.12\times10^{-6}} \text{ less than or equal to } 1.0$$

where Q is measured in Curies/sec.

*(Note for equations 3.8.A.2.a. and c.):

Where the term in parentheses the operational status of Units 2 and 3. If either Units 2 or 3 is shutdown, then the first term (Q_2/value) shall be used. If both units are in operation, then the second term ($Q_{2,3}$/value) shall be used.

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. If the limits of 3.8.A.2.a., or 3.8.A.2.c. are exceeded, an orderly load reduction of the unit(s) causing these limits to be exceeded shall be initiated immediately to reduce the releases below the limits of 3.8.A.2.a. or 3.8.A.2.c. The provisions of Specification 3.0.A. are not applicable.

B. Mechanical Vacuum Pump

1. The mechanical vacuum pump shall be capable of being isolated and secured on a signal of high radioactivity, whenever the main steam isolation valves are open.

2. If the limits of 3.8.B. are not met following a routine surveillance check, orderly shutdown shall be initiated.

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

B. Mechanical Vacuum Pump

At least once during each operating cycle verify automatic securing and isolation of the mechanical vacuum pump.

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. Liquid Effluents

1. Radioactive liquid released from the facility shall be continuously monitored. To accomplish this either the radiation monitor or the discharge line on the discharge canal sampler shall be operable.

2. The concentration of gross beta activity (above background) in the condenser cooling water discharge canal shall not exceed the limits stated below unless the discharge is controlled on a radionuclide basis in accordance with Appendix B, Table II, Column 2 of 10CFR20 and note 1 thereto:

Maximum Concentration -

1 X 10^{-7} micro-Ci/ml

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

C. Liquid Effluents

1. The radiation monitor shall be calibrated quarterly and functionally tested monthly. The operability of the sampler shall be verified on a daily basis.

2. Station records shall be maintained of the radioactive concentration and volume of each batch of liquid effluent released and of the condenser cooling water flow at time of discharge.

Isotopic analyses including determination of tritium of representative batches of liquid effluent shall be performed and recorded at least once per quarter. Each batch of effluent released shall be counted for gross alpha and beta activity and the results recorded. At least once per month a gamma scan of representative batches of effluent shall be performed and recorded to determine the gamma energy peaks

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. Two independent samples from a tank shall be taken and analyzed and the valve line-up checked prior to discharge of liquid effluents from that tank.

4. If the limits of 3.8.C. cannot be met, radioactive liquid effluents shall not be released.

D. Radioactive Waste Storage

The maximum amount of radioactivity in liquid storage in the Waste Sample Tanks, the Floor Drain Sample Tanks and the Waste Surge Tank shall not exceed 3.0 curies and the maximum amount of radioactivity in any tank shall not exceed 0.7 curies. If these conditions cannot be met, the stored liquid shall be recycled within 24 hours to the Waste Collector Tanks or the Waste Neutralizer Tanks until the condition is met.

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

of these batches. If energy peaks other than those determined by the previous isotopic analyses are found, a new set of isotopic analyses shall be performed and recorded.

3. The performance and results of independent samples and valve checks shall be logged.

D. Radioactive Waste Storage

A sample from each of the Waste Sample Tanks, Floor Drain Sample Tanks, and Waste Surge Tank shall be taken, analyzed and recorded every 72 hours. If no additions to a tank have occurred since the last sample, the tank need not be sampled until the next addition.

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

E. General

It is expected that releases of radioactive material in effluents will be kept as small fractions of the limits specified in Section 20.106 of 10 CFR Part 20. At the same time the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such small fractions, but still within the limits specified in Section 20.106 of 10 CFR Part 20. It is expected that in using this operational flexibility under unusual operating conditions the licensee will exert his best efforts to keep levels of radioactive material in effluents as low as is reasonably achievable.

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

E. General

1. Operating procedures shall be developed and used, and equipment which has been installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences, shall be maintained and used, to keep levels or radioactive material in effluents released to unrestricted areas as low as is reasonably achievable. The environmental monitoring program given in Table 4.8.1 shall be conducted.

2. A census of animals producing milk for human consumption shall be conducted annually during the grazing season to determine their location and number with respect to the site. The census shall be conducted

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

under the following conditions:

1. Within a 2-mile radius from the plant site, enumeration by a door-to-door or equivalent counting technique.

2. Within a 5-mile radius, enumeration by using referenced information from county agricultural agents or other reliable sources.

If it is learned from this census that animals are present at a location which yields a calculated thyroid dose greater than from previously sampled animals, the new location shall be added to the surveillance program as soon as practicable. The sampling location having the lowest calculated dose may then be dropped from the surveillance program at the end of the grazing season during which the census was conducted. Also, any location from which milk can no longer be obtained may be dropped from the surveillance program after notifying the NRC in writing that

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

F. Miscellaneous Radioactive Materials Sources

Source Leakage Test

Specification

Each sealed source containing radioactive material in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and either decontaminated and repaired or disposed of in accordance with Commission Regulations.

A complete inventory of radioactive materials in the licensee's possession shall be maintained current at all times.

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

milk-producing animals are no longer present at that location.

F. Miscellaneous Radioactive Materials Sources

Each sealed source shall be tested for leakage and/or contamination by the licensee or by other persons specifically authorized by the Commission or an Agreement State. The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

Each category of sealed sources shall be tested at at the frequency described below.

1. Sources in use (excluding startup sources previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive material:

   a. With a half-life greater than 30 days (excluding Hydrogen 3), and

   b. In any form other than gas.

2. Stored sources not in use - Each sealed source shall be tested

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

3. Startup sources - Each sealed startup source shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.6.C.3 if source leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

TABLE 4.8-1

DRESDEN STANDARD RADIOLOGICAL MONITORING PROGRAM

Sample Media	Collection Site	Type of Analysis	Frequency	Non-Routine Reporting Levels (1)
1. Air Monitoring	(a) Onsite and near field	1. Filter - gross beta(2)	1. Weekly	Cs-134 10, CS-137 20 pCi/m^3 0.7 pCi/m^3
	(1) Onsite Station #1 (2) Onsite Station #2 (3) Onsite Station #3 (4) Collins Road (5) Bennitt Farm (6) Pheasant Trail	2. Charcoal - I-131 3. Sampling Train - Test and Maintenance	2. Bi Weekly* 3. Weekly	
	(b) Far Field (1) Clay Products (2) Prairie Park (3) Coal City (4) Goose Lake Village (5) Morris (6) Lisbon (7) Minooka (8) Channahon (9) Joliet (10) Elwood (11) Wilmington	1. Filter Exchange 2. Charcoal Exchange 3. Sampling Train - Test and Maintenance	1. Weekly 2. Bi-Weekly* 3. Weekly	Same as 1(a) when analyses are made
2. TID	Same as 1	Gamma Radiation	Quarterly	
3. Fish	Dresden Pool of Illinois River	Gamma isotopic	Semi-annual	Mn-54 $3x10^4$, Fe-59 $1x10^4$ Co-58 $3x10^4$, Co-60 $1x10^4$ Zn-65 $2x10^4$, Cs-134 $1x10^3$ Cs-137 $2x10^3$ pCi/Kg wet weight
4. Milk	2 dairy farms	I-131	1. Weekly - Grazing Season - May to Oct 2. Monthly - Nov to Apr	I-131 3 pCi/l Cs-134 60 pCi/l Cs-137 70 pCi/l Ba-La-140 300 pCi/l
5. Surface Water	Illinois River at EJ&E R.R. Bridge	Gamma isotopic	Monthly analysis of weekly composites	Footnote**
6. Cooling Water Sample	(a) Inlet (1) Unit 1 (b) Discharge (1) Unit 1 (2) Unit 2/3	Gross Beta	Weekly	
7. Sediment	(a) Dresden Lock & Dam	Gamma Isotopic	Annual	

Notes for Table 4.8-1

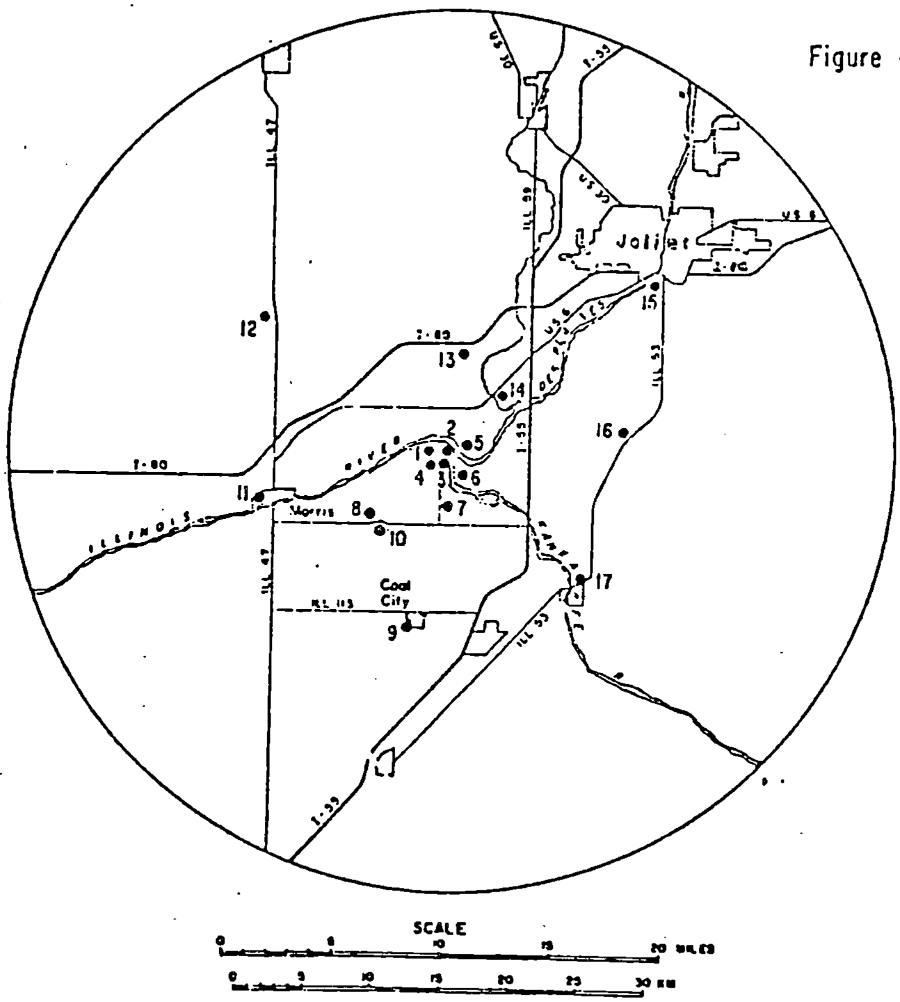
* Bi-weekly shall mean that the frequency is once every other week.

(1) Average concentration over calendar quarter

** H-3 $2x10^4$, Mn-54 $1x10^3$, Fe-59 $1x10^2$, Co-58 $6x10^2$, Co-60 $2x10^2$, Zn-65 $2x10^2$, Zr-Nb-95 $4x10^2$, I-131 3, Cs-134 30, Cs-137 60, Ba-La-140 $1x10^2$ pCi/l.

(2) A gamma isotopic analysis shall be performed whenever the gross beta concentration in a sample exceeds by five times (5x) the average concentration of the preceding calendar quarter for the sample location.

Figure 4.8-1



1 - Onsite Station 1
2 - Onsite Station 2
3 - Onsite Station 3
4 - Collins Road
5 - Bennitt Farm
6 - Pheasant Trail
7 - Clay Products
8 - Prairie Park
9 - Coal City
10 - Goose Lake Village
11 - Morris
12 - Lisbon
13 - Minooka
14 - Channahon
15 - Joliet
16 - Elwood
17 - Wilmington

Figure 4.8-1

LOCATIONS OF FIXED ENVIRONMENTAL RADIOLOGICAL MONITORING STATIONS

3.8 LIMITING CONDITIONS FOR OPERATION BASES

A. Airborne Effluents - Detailed dose calculations for several locations offsite have been made and are described in Appendix A of the SAR. These calculations consider site meteorology, buoyancy characteristics, and isotopic content of the effluent of each unit. Independent dose calculations for several locations offsite have been made by the AEC staff. The method utilized onsite meteorological data developed by the applicant and utilized diffusion assumptions as developed by the applicant with the exception that: (1) the Stumke correction factor for plume rise was not allowed, (2) the height of the bluff north of the site (30 meters) was subtracted from the stack height for calculational purposes, (3) Pasquill diffusion parameters rather than Hanford parameters were used, (4) the staff used a reflection factor of 2 for the calculation of Specification 3.8.A.2.

The method utilized by the staff is described in Section 7-5.2.5 of "Meteorology and Atomic Energy-1968," equation 7.63 being used. The results of these calculations were more conservative than those generated by the applicant and were thus chosen to be used as the basis of establishment of the limits. Based on these calculations, a release rate limit of gross activity, except for halogens and particulates with half-lives greater than eight days, in the amount of (a) 0.56 curies/sec. from the Unit 1 plant chimney or (b) 0.9 curies/sec. from the Units 2/3 plant chimney or (c) 0.09 curies/sec. from the ventilation stack will not result in offsite annual doses in excess of the limits specified in 10 CFR 20. Because a lower bouyancy factor is obtained when either Unit 2 or 3 is shut down, the equation must be changed so that the operating unit discharge is 0.7 curies per sec. These limits are based on a noble gas mixture whose energy with 30 minute holdup is 0.7 Mev. If on analysis this average energy increases, the average annual release limit must be decreased accordingly.

Considering the above, 3.8.A.2 gives equations to be used in summing the airborne effluents from the Unit 1 plant chimney. The Unit 2/3 plant chimney, and the Unit 2/3 ventilation stack that will assure that total off-site doses are not in excess of the limits specified in 10 CFR 20.

The intent of Section 3.8.A.2.b is not to relieve the licensee of its obligation to exert its best efforts to keep levels of radioactive material in effluents as low as practicable. At the action level specified in Section 3.8.A.2.b, the Commission is to be informed of the licensee's plans for continued operation of the facilities.

3.8 LIMITING CONDITIONS FOR OPERATION BASES (Cont'd.)

The equations given in Specification 3.8.A.2.b.2 give the cumulative hours which represent the limits of permissible operation which reduce the permissible activity released compared to continuing operation at the conditions stated in Section 20.106 of 10 CFR Part 20. The time periods under discussion permit short-term releases higher than small fractions of the limits specified in Section 20.106 of 10 CFR Part 20.

In addition, Commonwealth Edison has embarked on a program of selecting, designing, and installing additional equipment to reduce off-gas emissions on Dresden Units 2 and 3. This equipment is expected to reduce substantially the releases of radioactive material in the effluent. Commonwealth Edison will submit a description of its proposed design for this equipment, and a schedule for its installation, prior to June 1, 1971, unless an extension of this time is granted. Upon completion of the installation of this equpment, these Technical Specifications will be revised to include the effect of operation of the emission-reducing equipment.

Detailed calculations of ground level air concentrations of halogens and particulates with half-lives greater than 8 days at several offsite locations have been made as described in Appendix A of the SAR. These calculations consider site meteorology and buoyancy characteristics of the effluent from each unit. Based on these calculations, the release rate limit for these isotopes in the equation in Section 3.8.A.2.b is obtained. Use of this equation assures that releases will not result in off-site doses in excess of those specified in 10 CFR 20.

The assumptions used by the AEC staff for these calculations were: (1) Onsite meteorological data were used for the most critical 22.5 degree sector. (2) No building wake credit was used. (3) To consider possible reconcentration effects a reduction factor of 700 was applied to allow for the milk production and consumption mode of uptake.

Before initial operation of the nearby Midwest Fuel Reprocessing Plant the above limits will be adjusted to reflect the dose contribution of this facility.

B. Mechanical Vacuum Pump - The purpose of isolating the mechanical vacuum pump line is to limit release of activity from the main condenser. During an accident, fission products would be transported from the reactor through the main

3.8 LIMITING CONDITIONS FOR OPERATION BASES (Cont'd.)

steamlines to the main condenser. The fission product radioactivity would be sensed by the main steamline radioactivity monitors which initiate isolation.

C. Liquid Effluents - Liquid effluent release rate will be controlled in terms of the concentration in the discharge canal. In the case of unidentified mixtures such concentration limit is based on assumption that the entire content is made up of the most restrictive isotope in accordance with 10 CFR 20. Such a limit assures that even if a person obtained all of his daily water intake from such a source, the resultant dose would not exceed that specified in 10 CFR 20. Since no such use of the discharge canal is made and considerable natural dilution occurs prior to any location where such dose usage could occur, this assures that off-site doses from this source will be far less than the limits specified in 10 CFR 20.

In addition to the two independent samples of each batch prior to discharge, a radiation monitor on the discharge line and a sampler in the discharge canal give further assurance that discharges are kept at or below the maximum limits at all times.

D. Radioactive Waste Storage - As discussed in the SAR, the radioactive waste tanks that are at or above grade are located such that their postulated catastrophic failure could cause release of their contained radioactivity to the Illinois River. To assure that such a postulated release would not raise radioactivity levels in the river to values greater than ten times 10 CFR 20, a limit on the amount of radioactivity the tanks can contain is established.

The performance of the radiation monitoring system relative to detecting fuel leakage shall be evaluated during the first five years of operation. The conclusions of this evaluation will be reported to the Atomic Energy Commission.

E. General - The environmental radiological monitoring program is designed to:

1. Provide data on measurable levels of radiation and radioactive materials in the environment in order to evaluate the calculational models used to relate the quantities of radioactive material released in effluents to the radiation doses received by individuals via the principal pathways of exposure;

3.8 LIMITING CONDITIONS FOR OPERATION BASES (Cont'd.)

2. Identify changes in the use of nearby unrestricted areas for agricultural purposes to permit modifications in monitoring programs for evaluating doses to individuals from principal pathways of exposure;

3. Maintain off-site air samplers in a readiness posture in case of an unplanned release; and

4. Provide year-round coverage of certain principal pathways.

Attainment of these objectives will be met by dividing the monitoring program into two distinct and independent components.

Standard Monitoring Program

The standard monitoring program is designed to provide year-round coverage of certain principal dose pathways, identify land use changes, and maintain the offsite air samplers in a state of readiness in case of an unplanned release.

The sampling and analytical schedule for environmental samples, and the testing and maintenance schedule for off-site air samplers, is given in Table 4.8-1. Figure 4.8-1 shows the locations of the fixed sampling sites.

Table 4.8-2 indicates acceptable detection capabilities for radioactive materials in environmental samples. These detection capabilities are tabulated in terms of the lower limits of detection (LLDs) at the 95% confidence level. The LLDs shall be determined in the manner described in HASL-300, section D-08, August 1976. The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, contractor omission which is corrected as soon as discovered, malfunction of sampling equipment, or if a person who participates in the program by providing samples goes out of business. If the equipment malfunctions, corrective actions shall be completed as soon as practical. If a person supplying samples goes out of business, a replacement will be found as soon as possible. All deviations from the sampling schedule shall be described in the annual report.

3.8 LIMITING CONDITIONS FOR OPERATION BASES (Cont'd.)

Environmental Dose Pathway Study

The environmental dose pathway study (EDPS) will use the best practicable monitoring techniques for measuring radioactivity in effluents and certain environmental media in order to evaluate the relationship between the effluents and doses to individuals in unrestricted areas.

The EDPS will provide data on measurable levels of radiation and certain radioactive materials in effluent and the environs to evaluate the relationship between the quantitative and qualitative nature of effluent discharges and the doses to individuals.

It is planned that most of the monitoring will be conducted during the warmer months of the year (May through October) so that such principal pathways as milk can be studied. Use will be made of the onsite meteorological data to predict atmospheric dispersion so that comparisons can be made with measured data.

A description of the EDPS program to be performed is described in the report to the NRC entitled "Proposal to Change Environmental Radiological Monitoring Programs" by the Commonwealth Edison Company and Dr. Bernd Kahn, Georgia Institute of Technology, dated October 1976.

F. Miscellaneous Radioactive Materials Sources

The objective of this specification is to assure that leakage from byproduct, source and special nuclear material sources does not exceed allowable limits. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium.

4.8 SURVEILLANCE REQUIREMENT BASES

None

TABLE 4.8-2

PRACTICAL LOWER LIMITS OF DETECTION (LLD)
FOR STANDARD ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM

Sample Media	Analysis	LLD (4.66 σ)	Units
Airborne "Particulate"	Gross Beta **	0.01	pCi/m^3
	Gamma Isotopic	0.01	pCi/m^3
	Sr-89,90	0.01	pCi/m^3
Airborne I-131	Iodine-131	0.10	pCi/m^3
Liquids	Sr-89	10	pCi/l
	Sr-90	2	pCi/l
	I-131	5*	pCi/l
	Cs-134	10	pCi/l
	Cs-137	10 ***	pCi/l
	Tritium	0.2	pCi/ml
	Gross Beta **	5	pCi/l
	Gamma Isotopic	(LT)20	pCi/l/nuclide
Vegetation	Gross Beta **	2	pCi/g wet
	I-131	0.03	pCi/g wet
	Sr-89,90	1	pCi/g wet
	Gamma Isotopic	0.2	pCi/g wet
Soil, Sediment	Gross Beta **	2	pCi/g dry
	Sr-89,90	1	pCi/g dry
	Gamma Isotopic	0.2	pCi/g dry
Animal Tissue	Sr-89,90	0.1	pCi/g wet
	I-131 - Thyroid	0.1	pCi/g wet
	Cs-134,137	0.1	pCi/g wet
	Gross Beta **	1.0	pCi/g wet

(LT) = Less than
* 0.5 pCi/l on milk samples collected during the pasture season.

** Referenced to CS-137

*** 5.0 pCi/l on milk samples.

3.9 LIMITING CONDITION FOR OPERATION

AUXILIARY ELECTRICAL SYSTEMS

Applicability:

Applies to the auxiliary electrical power system.

Objective:

To assure an adequate supply of electrical power during plant operation.

Specification:

A. The reactor shall not be made critical unless all the following requirements are satisfied:

1. One 345 KV line, associated switchgear, and the reserve auxiliary power transformer capable of carrying power to Unit 2.

2. The Dresden 2 diesel generator and the Unit 2/3 diesel generator shall be operable.

3. An additional source of power consisting of one of the following:

   (a) One other 138 KV line, fully operational and

4.9 SURVEILLANCE REQUIREMENT

AUXILIARY ELECTRICAL SYSTEMS

Applicability:

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

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Station Batteries

1. Every week the specific gravity and voltage of the pilot cell and temperature of adjacent cells and overall battery voltage shall be measured.

2. Every three months the measurements shall be made of voltage of each cell to nearest 0.01 volt, specific gravity of each cell, and temperature of every fifth cell.

3. Every refueling outage, the station batteries shall be subjected to a rated load discharge test. Determine specific gravity and voltage of each cell after the discharge.

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

capable of carrying auxiliary power to Unit 2.

(b) One 345 KV line from Unit 3 capable of carrying auxiliary power to an essential electrical bus of Unit 2 through the 4160 volt bus tie.

4. (a) 4160 volt buses 23-1 and 24-1 are energized.

(b) 480 volt buses 28 and 29 are energized.

5. The unit 24/48 volt batteries, the two station 125 volt batteries and the two station 250 volt batteries and a battery charger for each required battery are operable.

B. Except when the reactor is in the Cold Shutdown or Refueling modes with the head off, the availability of electric power shall be as specified in 3.9.A, except as specified in 3.9.B.1, 3.9.B.2, and 3.9.B.3.

1. From and after the date that incoming power is available from only one line, reactor operation is

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

If this specification has been complied with for a particular battery for Dresden Unit 3, it shall not be required for Dresden Unit 2.

B. N/A

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

permissible only during the succeeding seven days unless an additional line is sooner placed in service providing both the Unit 2 and Unit 2/3 emergency diesel generators are operable. From and after the date that incoming power is not available from any line, reactor operation is permissible providing both the Unit 2 and Unit 2/3 emergency diesel generators are operating and all core and containment cooling systems are operable and the NRC is notified within 24 hours of the situation, the precautions to be taken during this situation, and the plans for prompt restoration of incoming power.

2. From and after the date that one of the diesel generators and/or its associated bus is made or found to be inoperable for any reason, reactor operation is permissible according to Specification 3.5/4.5F and 3.9D only during the succeeding seven days unless such diesel generator and/or bus is sooner made

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

operable, provided that during such seven days the operable diesel generator shall be demonstrated to be operable at least once each day and two off-site lines are available.

3. From and after the date that one of the two 125/250 battery systems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such battery system is sooner made operable.

C. Diesel Fuel

There shall be a minimum of 10,000 gallons of diesel fuel supply on site for each diesel.

D. Diesel Generator Operability

Whenever the reactor is in the Cold Shutdown or Refueling modes, a minimum of one diesel generator (either the Dresden 2 diesel generator or the Unit 2/3 diesel generator) shall be operable whenever any work is being done which has

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

C. Diesel Fuel

Once a month the quantity of diesel fuel available shall be logged.

Once a month a sample of diesel fuel shall be checked for quality.

D. Diesel Generator Operability

1. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue until both the diesel engine and the generator are at equilibrium

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

the potential for draining the vessel, secondary containment is required, or a core or containment cooling system is required.

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

conditions of temperature while full load output is maintained.

2. During the monthly generator test the diesel starting air compressor shall be checked for operation and its ability to recharge air receivers.

3. During the monthly generator test the diesel fuel oil transfer pumps shall be operated.

4. Additionally, during each refueling outage, a simulated loss of off-site power in conjunction with an ECCS initiation signal test shall be performed on the 4160 volt emergency bus by:

   (a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

   (b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency buses with permanently connected loads, energizes the auto-connected

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

emergency loads through the load sequencer, and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads.

3.9 LIMITING CONDITION FOR OPERATION BASES

A. The general objective of this Specification is to assure an adequate source of electrical power to operate the auxiliaries during plant operation, to operate facilities to cool and lubricate the plant during shutdown, and to operate the engineered safeguards following an accident. There are three sources of electrical energy available; namely, the 138 KV transmission system, the diesel generators, and the 345 KV transmission system through the 4160 volt bus tie.

The d-c supply is required for control and motive power for switchgear and engineered safety features. The electrical power required provides for the maximum availability of power; i.e., one active off-site source and two back-up sources of off-site power and the maximum amount of on-site sources.

B. Auxiliary power for Unit 2 is supplied from two sources, either the Unit 2 auxiliary transformer or the Unit 2 reserve auxiliary transformer. Both of these transformers are sized to carry 100% of the auxiliary load. If the reserve auxiliary transformer is lost, the unit can continue to run for 7 days since the unit auxiliary transformer is available and both diesel generators are operational. A reduced period is provided since if an accident occurs during this period, the unit would trip and power to the unit auxiliary transformer would be lost and the diesels would be the only source of power.

In the normal mode of operation the 138 KV system is operating and two diesel generators are operational. One diesel generator may be allowed out of service based on the availability of power to the 138 KV switchyard, a source of power available from the 345 KV system through a 4160 volt bus tie and the fact that one diesel carries sufficient engineered safeguards equipment to cover all breaks. Off-site power is quite reliable. In the last 25 years there has only been one instance in which all off-site power was lost at a Commonwealth Edison generating station.

A battery charger is supplied with each of the 125 and 250 volt batteries and in addition a shared battery charger is supplied which can be used for Units 2 or 3. Thus, on loss of the normal battery charger, the shared charger can be used. Since an alternate charging source is available, one battery charger can be allowed out of service for thirty days without loss of this source of power. The 125 volt battery system shall have a minimum of 105 volts at the battery terminals to be considered operable.

3.9 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

C. The diesel fuel supply of 10,000 gallons will supply each diesel generator with a minimum of two days of full load operation or about four days at 1/2 load. Additional diesel fuel can be obtained and delivered to the site within an 8-hour period; thus a 2-day supply provides for adequate margin.

4.9 SURVEILLANCE REQUIREMENT BASES

A. 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.

B. 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 would contribute to excessive corrosion of the system causing decreased reliability. The growth of micro-organisms results in slime formations which are one of the chief causes of jellying in hydro-carbon fuels. Minimizing of such slimes is also essential to assuring high reliability.

C. 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 will be manually started, synchronized to the bus and load picked up. The diesel shall be loaded to at least half load to prevent fouling of the engine. It is expected that the diesel generator will be run for one to two hours. Diesel generator experience at other Commonwealth Edison generating stations indicates that the testing frequency is adequate and provides a high reliability of operation should the system be required. In addition, during the test when the generator is synchronized to the bus, it is also synchronized to the off-site power source and thus not completely independent of this source. To maintain the maximum amount of independence, a thirty-day testing interval is also desirable.

4.9 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

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

Following the monthly test of the diesels, the fuel oil day tank will be approximately 1/2 full based on a two-hour test at full load and 205 gallons per hour at full load. At the end of the monthly load test of the diesel generators, the fuel oil transfer pumps will be operated to refill the day tank and to check the operation of these pumps from the emergency source. The test of the emergency diesel generator during the refueling outage will be more comprehensive in that it will functionally test the system; i.e., it will check diesel starting and closure of diesel breaker and sequencing of loads on the diesel. The diesel 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. The timing sequence will be checked to assure proper loading in the time required. The only load on the diesel is that due to friction and windage and a small amount of bypass flow on each pump. Periodic tests between refueling outages verify the ability of the diesel 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 at a refueling interval are sufficient to maintain adequate reliability.

3.10 LIMITING CONDITIONS FOR OPERATION

REFUELING

Applicability:

Applies to fuel handling and core reactivity limitations.

Objective:

To assure core reactivity is within capability of the control rods and to prevent criticality during refueling.

Specification:

A. Refueling Interlocks

The reactor mode switch shall be locked in the "Refuel" position during core alterations and the refueling interlocks shall be operable except as specified in Specifications 3.10.D and 3.10.E.

B. Core Monitoring

During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:

1. The SRM shall be inserted to the normal

4.10 SURVEILLANCE REQUIREMENTS

REFUELING

Applicability:

Applies to the periodic testing of those interlocks and instruments used during refueling.

Objective:

To verify the operability of instrumentation and interlocks used in refueling.

Specification:

A. Refueling Interlocks

Prior to any fuel handling, with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

B. Core Monitoring

Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, the SRM's will be checked daily for response, except when the conditions of 3.10.B.2.a and 3.10.B.2.b are met.

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors are permissible as long as the detector is connected into the normal SRM circuit.)

2. The SRM or dunking type detector shall have a minimum of 3 cps with all rods fully inserted in the core except when both of the following conditions are fulfilled:

   a) No more than two fuel assemblies are present in the core quadrant associated with the SRM.

   b) While in core, these fuel assemblies are in locations adjacent to the SRM.

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool water level shall be maintained at a level of 33 feet.

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool level shall be recorded daily.

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

D. Control Rod and Control Rod Drive Maintenance

* A maximum of two non-adjacent control rods separated by more than two control cells in any direction, may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance provided the following conditions are satisfied:

1. The reactor mode switch shall be locked in the "re-fuel" position. The re-fueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other re-fueling interlocks shall be operable.

* 2. Specification 3.3.A.1 shall be met or, the control rod directional control valves for a minimum of eight control rods surrounding each drive out of service for maintenance will be disarmed electrically and sufficient margin to criticality demonstrated.

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

D. Control Rod Drive and Control Rod Drive Maintenance

1. This surveillance requirement is the same as given in 4.10.A.

* 2. Sufficient control rods shall be withdrawn prior to performing this maintenance to demonstrate with a margin of 0.25 percent delta k that the core can be made subcritical at any time during the maintenance with the strongest operable control rod fully withdrawn and all other

*Revised with change 17 to DPR-19 dated 3/17/72
Revised with change 9 to DPR-25 dated 3/17/72

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

3. SRM's shall be operable (a) in each core quadrant containing a control rod on which maintenance is being performed, and (b) in a quadrant adjacent to one of the quadrants specified in 3.10.D.3.a above. Requirements for an SRM to be considered operable are given in 3.10.B.

E. Extended Core Maintenance

More than two control rods may be withdrawn from the reactor core provided the following conditions are satisfied:

1. The reactor mode switch shall be locked in the "re-fuel"

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

operable rods fully inserted. Alternately, if a minimum of eight control rods surrounding each control rod out of service for maintenance are to be fully inserted and have their directional control valves electrically disarmed, the 0.25 percent delta k margin will be met with the strongest control rod remaining in service during the maintenance period fully withdrawn.

3. This surveillance requirement is the same as that given in 4.10.B.

E. Extended Core Maintenance

1. This surveillance requirement is the same as that given in 4.10.A.

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other re-fueling interlocks shall be operable.

2. SRM's shall be operable in the core quadrant where fuel or control rods are being moved and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in 3.10.B.

F. Spent Fuel Cask Handling

1. Fuel cask handling above the 545' elevation will be done with the reactor building crane in the RESTRICTED MODE only except as specified in 3.10.F.2.

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

2. This surveillance requirement is the same as that given in 4.10.B.

F. Spent Fuel Cask Handling

1. Prior to fuel cask handling operations, the redundant crane including the rope, hooks, slings, shackles and other operating mechanisms will be inspected.

The rope will be replaced if any of the following conditions exist:

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

2. Fuel cask handling in other than the RESTRICTED MODE will be permitted in emergency or equipment failure situations only to the extent necessary to get the cask to the closest acceptable stable location.

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

a. Twelve (12) randomly distributed broken wires in one lay or four(4) broken wires in one strand of one rope lay.

b. Wear of one-third the original diameter of outside individual wire.

c. Kinking, crushing, or any other damage resulting in distortion of the rope.

d. Evidence of any type of heat damage.

e. Reductions from nominal diameter of more than 1/16 inch for a rope diameter from 7/8" to 1 1/4" inclusive.

2. Before August 30, 1976, prior to operations in in the RESTRICTED MODE

a. the "two-block" limit switches will be tested.

On and after August 30, 1976, prior to operation in the RESTRICTED MODE

a. the controlled area limit switches will be tested:

b. the "two-block" limit switches will be tested;

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

3. Before August 30, 1976, fuel cask handling is permitted, without the mechanically operated power limit switch in the main hoist motor power circuit and without an operable control system for limiting the crane/cask travel to a restricted area, provided an operator, in constant communication with the crane operator and with personnel directing crane operation is stationed at the main breaker supplying power to the overhead crane with no duties other than monitoring crane operation. The operator will be ordered to remove power from the crane in the event that a malfunction either causes the cask to be lifted above a six-inch limit or causes the cask to deviate from the restricted area.

On and after August 30, 1976, operation with a failed controlled area limit switch is permissible for 48 hours providing an operator is on the refueling floor to assure the crane is operated within the restricted zone painted on the floor.

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

c. the "inching hoist" controls will be tested.

3. The empty spent fuel cask will be lifted free of all support by a maximum of 1 foot and left hanging for 5 minutes prior to any series of fuel cask handling operations.

3.10 LIMITING CONDITION FOR OPERATION BASES

A. Refueling Interlocks

During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specifications 3.2 limits the core alterations to assure that the resulting core loading can be controlled with the reactivity control system and interlocks at any time during shutdown or the following operating cycle.

Addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

For a new core the dropping of a fuel assembly into a vacant fuel location adjacent to a withdrawn control rod does not result in an excursion or a critical configuration, thus adequate margin is provided.

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. Requiring a minimum of 3 counts per second whenever criticality is possible provides assurance that neutron flux is being monitored. Criticality is considered to be impossible if there are no more than two assemblies in a quadrant and if these are in locations adjacent to the SRM. In this case only, the SRM or dunking type detector count rate is permitted to be less than 3 counts per second.

3.10 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

C. Fuel Storage Pool Water Level

To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (37'9") well above a level to assure adequate cooling (just above active fuel) and above the level at which the GSEP action is initiated (5' uncontrolled loss of level with level decreasing).

D. During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time. This specification provides assurance that inadvertent criticality does not occur during such maintenance.

The maintenance is performed with the mode switch in the "re-fuel" position to provide the re-fueling interlocks normally available during re-fueling operations as explained in Part A of these Bases. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the re-fueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated with the control rods remaining in service insures that inadvertent criticality cannot occur during this maintenance. The Shutdown margin is verified by demonstrating that the core is shut down even if the strongest control rod remaining in service is fully withdrawn. Disarming the directional control valves does not inhibit control rod scram capability.

The requirement for SRM operability during the maintenance is covered in Part B of these Bases.

E. The intent of this specification is to permit the unloading of a significant portion of the reactor core for such purposes as in-service inspection requirements, examination of the core support plate, etc. This specification provides assurance that inadvertent criticality does not occur during such operation.

This operation is performed with the mode switch in the "re-fuel" position to provide the re-fueling interlocks normally available during re-fueling as explained in Part A of these Bases. In order to withdraw more than one control rod, it is necessary to bypass the re-fueling interlock on each

3.10 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlocks can be bypassed insures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with that control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.

The requirement for SRM operability during these operations is covered in Part B of these Bases.

F. The operation of the redundant crane in the Restricted Mode during fuel cask handling operations assures that the cask remains within the controlled area once it has been removed from its transport vehicle (i.e., once it is above the 545' elevation). Handling of the cask on the Refueling Floor in the Unrestricted Mode is allowed only in the case of equipment failures or emergency conditions when the cask is already suspended. The Unrestricted Mode of operation is allowed only to the extent necessary to get the cask to a suitable stationary position so the required repairs can be made. Operation with a failed controlled area microswitch will be allowed for a 48-hour period providing an Operator is on the floor in addition to the crane operator to assure that the cask handling is limited to the controlled area as marked on the floor. This will allow adequate time to make repairs but still will not restrict cask handling operations unduly.

The Surveillance Requirements specified assure that the redundant crane is adequately inspected in accordance with the accepted ANSI Standard (B.30.2.0) and manufacturer's recommendations to determine that the equipment is in satisfactory condition. The testing of the controlled area limit switches assures that the crane operation will be limited to the designated area in the Restricted Mode of operation. The test of the "two-block" limit switch assures the power to the hoisting motor will be interrupted before an actual "two-blocking" incident can occur. The test of the inching hoist assures that this mode of load control is available when required.

Requiring the lifting and holding of the cask for 5 minutes during the initial lift of each series of cask handling

3.10 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

operations puts a load test on the entire crane lifting mechanism as well as the braking system.

Performing this test when the cask is being lifted initially from the cask car assures that the system is operable prior to lifting the load to an excessive height.

4.10 SURVEILLANCE REQUIREMENT BASES

None

3.11 LIMITING CONDITIONS FOR OPERATION

High Energy Piping Integrity

Applicability:

Applies to operating status of certain piping outside primary containment.

Objective:

To assure the integrity of sections of piping which is postulated to effect safe plant shutdown.

Specification:

1. The high energy piping sections identified in Table 4.11-1 shall be maintained free of visually observable through wall leaks.

   A. If a leak is detected by the surveillance program of 4.11, efforts to identify the source of the leaks shall be started immediately.

   B. If the source of leakage can not be identified within 24 hours of detection or if the leak is found to be from a break in the piping sections identified in Table 4.11-1, the pressure within the section of piping shall be brought to atmospheric pressure within 48 hours.

4.11 SURVEILLANCE REQUIREMENTS

High Energy Piping Integrity

Applicability:

Applies to the periodic examination requirements for certain piping outside primary containment.

Objective:

To determine the condition of the section of piping.

Specification:

The inspections listed in Table 4.11-1 shall be performed as specified.

3.11 LIMITING CONDITIONS FOR OPERATION (Cont'd.

2. When the modifications identified in the Commonwealth Edison letter to the NRC dated September 16, 1975 (G. Abrell to D. Ziemann), have been completed, Technical Specifications 3.11 and 4.11 will no longer be required.

4.11 SURVEILLANCE REQUIREMENTS (Cont'd.)

TABLE 4.11-1

SURVEILLANCE REQUIREMENTS FOR HIGH ENERGY PIPING OUTSIDE CONTAINMENT

Piping	Surveillance Area	Surveillance Technique	Frequency
Main Steam	from primary containment penetration to secondary containment penetration	Visual (1)	30 days
Reactor Feedwater Piping	from primary containment penetration to secondary containment penetration	Visual (1)	30 days
	discharge and "A" (2) Reactor Feed Pump to the 24-inch Diameter Feedwater Header	Visual (1)	30 days
HPCI Steam Piping	from the primary containment penetration to the reactor building penetration	Visual (1)	30 days

Notes for Table 4.11-1

(1) Visual observation of piping insulation and area for evidence of wetness or any physical damage resulting from a leak. Surveillance to be performed using normal access without scaffolding or any other access aids.

(2) "A" Reactor Feed Pump for Unit 2
"C" Reactor Feed Pump for Unit 3

3.11 LIMITING CONDITION FOR OPERATION BASES

High Energy Piping Integrity (Outside Containment)

Intensive analysis and review has shown that there are specific postulated high energy piping system failures which have the potential to inhibit safe cold shutdown of the reactor. This conclusion is based on utilizing the basic NRC high energy line break criteria. To reduce the probability of such failures, certain plant modifications are necessary. Until these modifications are complete, additional surveillance will be performed during plant operation to enhance the detection of piping system defects. The inservice examination and the frequency of inspection will provide a means for timely detection of such piping defects.

4.11 SURVEILLANCE REQUIREMENT BASES

None

3.12 LIMITING CONDITIONS FOR OPERATION

FIRE PROTECTION SYSTEMS

Applicability:

Applies to the fire protection systems whenever the equipment or systems being protected are required to be operable.

Objective:

To ensure that adequate protection against fires is maintained during all modes of facility operation.

Specification:

A. Fire Detection Instrumentation

1. As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.12-1 shall be operable at all times when equipment in that fire detection zone is required to be operable.

2. With the number of operable fire detection instruments less than required by Table 3.12-1;

   a. Perform an inspection of the affected zone, if accessible, within 1 hour. Perform additional inspections at least once per hour, except in inaccessible areas.

4.12 SURVEILLANCE REQUIREMENTS

FIRE PROTECTION SYSTEMS

Applicability:

Applies to the periodic testing requirements of the fire protection systems whenever the fire protection systems are required to be operable.

Objective:

To verify operability of the fire protection systems.

Specification:

A. Fire Detection Instrumentation

1. Each of the fire detection instruments given by Table 3.12-1 shall be demonstrated OPERABLE at least every 6 months by a channel functional test.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

b. Restore the inoperable instrument(s) to operable status within 14 days, or prepare and submit a report to the Commission pursuant to Specification 6.6.B.2 within the next 30 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to operable status.

c. The provisions of Specification 3.0.A. are not applicable.

B. Fire Suppression Water System

1. The Fire Suppression Water System shall be operable at all times with:

a. A flow path capable of taking suction from the 2/3 Intake Canal for Unit 2/3 Fire Pump.

b. A flow path capable of taking suction from the Unit 1 Intake Canal for Unit 1 fire pump.

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

B. Fire Suppression Water System

1. The Fire Suppression Water System shall be demonstrated operable:

a. At least once per 31 days by verifying valve positions.

b. At least once per 12 months by cycling each testable valve in the flow path through one complete cycle.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

c. The Unit 2/3 fire pump (2000 GPM) with its discharge aligned to the fire suppression header (from Unit 2/3 Intake Structure).

d. The Unit 1 fire pump (2000 GPM) with its discharge aligned to the fire suppression header (from Unit 1 Intake Structure).

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

c. At least once per year by performance of a system flush.

d. At least once per operating cycle:

1) By performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence and verifying that each automatic valve in the flow path actuates to its correct position.

2) By verifying that the Unit 2/3 fire pump develops at least 2000 gpm at a system head of 238 feet.

3) By verifying that the Unit 1 fire pump starts and develops at least 2000 gpm at a system head of 238 ft.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

e. Automatic initiation logic for each fire pump.

f. Fire suppression header piping with sectional control valves to:

1) The yard loop.

2) The front valve ahead of the water flow alarm device on each sprinkler or water spray system.

3) The standpipe system.

2. With an inoperable fire pump or associated water supply, restore the inoperable equipment to operable status within 7 days, or prepare and submit a

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

4) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

e. At least once per 3 years by performing flow tests of the system in accordance with the "Test of Water Supplies" Chapter in the NFPA Fire Protection Handbook.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

report to the Commission pursuant to Specification 6.6.B.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

3. With no Fire Suppression Water System operable, within 24 hours;

   a. Establish a backup Fire Suppression Water System.

   b. Notify the Commission pursuant to Specification 6.6.B.1 outlining the actions taken and the plans and schedule for restoring the system to operable status.

4. If the requirements of 3.12.B.3.a cannot be met, an orderly shutdown shall be initiated, and the reactor shall be in cold shutdown condition within 24 hours.

C. Sprinkler Systems

1. The sprinkler systems given in Table 3.12-2 shall be operable at all times when equipment in the area

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

C. Sprinkler Systems

1. At least once per 31 days by verifying that each valve, manual, power-operated, or automatic, in the flow

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

that is sprinkler protected is required to be operable.

2. With a sprinkler system inoperable, establish fire inspections with backup fire supression equipment within 1 hour.

   a. In the Unit 2/3 turbine mezzanine 538' elevation area or Unit 2 hydrogen seal oil area, a continuous fire watch is to be established.

   b. In all other areas given in Table 3.12-2 perform surveillance hourly.

3. Restore the system to operable status within 14 days, or prepare and submit a report to the Commission pursuant to Specification 6.6.B.2 within the next 30 days outlining the cause of inoperability, action taken and the plans for restoring the system to operable status.

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

path is in its correct position.

2. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

3. At least once per operating cycle:

   a. A system functional test shall be performed which includes simulated automatic actuation of the system and verifying that the automatic valves in the flow path actuate to their correct positions.

   b. The sprinkler headers shall be inspected to verify their integrity.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

4. The provisions of Specification 3.0.A are not applicable.

D. CO_2 System

1. The CO_2 Storage Tank shall have a minimum standby level of 50 percent and a minimum pressure of 250 psig.

2. The CO_2 System given in Table 3.12-3 shall be operable.

3. Specifications 3.12.D.1 and 3.12.D.2 above apply when the equipment in the areas given in Table 3.12-3 is required to be operable.

4. With a CO_2 System inoperable, establish fire inspection with backup fire suppression equipment in

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

c. Each nozzle shall be inspected to verify no blockage of the spray pattern.

4. At least every other operating cycle, a flow test will be performed to verify that each open head spray nozzle is unobstructed.

D. CO_2 System

1. At least once per 7 days the CO_2 Storage Tank level and pressure will be verified.

2. At least once per 31 days by verifying that each valve, manual, power-operated, or automatic, in the flow path is in the correct position.

3. At least once per operating cycle, the system valves and associated dampers will be verified to actuate automatically and manually. A brief flow test shall be made to verify flow from each nozzle.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

unprotected areas within 1 hour, and perform inspection at least hourly.

5. Restore the system to operable status within 14 days, or prepare and submit a report to the Commission pursuant to Specification 6.6.B.2 within the next 30 days outlining the cause of inoperability, action taken and the plans and schedule for restoring the system to operable status.

6. The provisions of Specification 3.0.A. are not applicable.

E. Fire Hose Stations

1. The Fire Hose Stations given in Table 3.12-4 shall be operable at all times when the equipment in the area is required to be operable.

2. With a hose station inoperable route an additional equivalent capacity hose to the unprotected area from an operable hose station within 1 hour.

3. When a hose station becomes inoperable, restore to operable

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

E. Fire Hose Stations

1. At least once per 31 days, a visual inspection of each fire hose station shall be made to assure all equipment is available at the station.

2. At least once per operating cycle, the hose will be removed for inspection and repacked. Degraded gaskets in the couplings will be replaced.

3. At least once per 3 years, each hose station valve will be

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

status within 14 days or report to the Commission pursuant to specification 6.6.B.2 within the next 30 days outlining the cause of inoperability and plans for restoring the hose station to operability.

4. The provisions of Specification 3.0.A are not applicable.

F. Penetration Fire Barriers

1. All penetration fire barriers (including fire doors and fire dampers) protecting safety related areas shall be intact, except as stated in specification 3.12.F.2 below.

2. With one or more of the required penetration fire barriers not intact, establish a continuous fire watch on at least one side of the affected penetration within 1 hour when

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

partially opened to verify valve operability and no blockage.

4. At least once per 3 years a hydrostatic test will be conducted on each hose at a pressure at least 50 psig above line pressure at that station.

F. Penetration Fire Barriers

1. Each of the required penetration fire barriers shall be verified to be intact by a visual inspection:

   a. At least once per 18 months, and

   b. Prior to declaring a penetration fire barrier intact following repairs or maintenance.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

the area on either side of the affected penetration contains equipment required to be operable.

3. The provisions of Specification 3.0.A are not applicable.

4. Restore the non-functional fire barrier penetrations to operable status within 7 days or prepare and submit a report to the Commission pursuant to Specification 6.6.B.2. within the next 30 days outlining the cause of inoperability, action taken and the plans and schedule for restoring the penetration fire barriers to operable status.

G. See 3.12.B.

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

G. Fire Pump Diesel Engine

1. The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) The fuel storage day tank contains at least 150 gallons of fuel, and

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

2) The diesel starts from ambient conditions and operates for at least 30 minutes.

3) The fuel transfer pump starts and transfers fuel from the storage tank to the day tank.

b. At least once per 92 days a sample of diesel fuel shall be checked for viscosity, water and sediment. The procedure used shall be consistent with existing station procedures used to check diesel fuel in the main storage tanks.

c. At least once per 18 months, by:

1) Subjecting the diesel to an inspection in accordance with procedure prepared in conjunction with its manufacturer's recommendations for the class of service, and

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

2) Verifying the diesel starts from ambient conditions on the autostart signal and operates for greater than or equal to 30 minutes while loaded with the fire pump.

2. The fire pump diesel engine batteries shall be demonstrated operable:

a. At least once per 7 days by verifying that:

1) The electroylyte level of each battery is above the plates, and

2) The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

H. Halon System

1. The following Halon system shall be OPERABLE with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure.

   a. Auxiliary Electrical Equipment Room

2. With one or more of the above required Halon systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or, prepare and submit a report to the

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

1) The batteries and battery racks show no visual indication of physical damage or abnormal deterioration, and

2) The battery-to-battery and terminal connections are clean, light, free of corrosion and coated with anti-corrosion material.

H. Halon System

1. At least once per 31 days, verify that each valve in the flow path is in the correct position.

2. At least once per 6 months, the Halon storage tank weight and pressure will be verified.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

Commission pursuant to Specification 6.6.B within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

3. At least once per operating cycle, the system, including associated ventilation dampers, will be verified to actuate manually and automatically. A flow test shall be made through headers and nozzles to assure no blockage.

3.12 LIMITING CONDITIONS FOR OPERATION BASES

Fire Protection Systems

Operability of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment, and is an integral element in the overall facility fire protection program. In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire inspections in the affected areas is required to provide detection capability until the inoperable instrumentation is returned to service.

The operability of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, sprinklers, CO_2 systems, Halon system, and fire hose stations, and is adequate to minimize potential damage to safety related equipment, and is a major element in the facility fire protection program.

In the event that portions of the fire suppression system are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the affected equipment can be restored to service.

A fire suppression water system shall consist of a water source, pumps, and distribution piping with associated valves. Such valves shall include sectional control valves, and the first valve ahead of the water flow alarm device on each sprinkler or hose standpipe riser.

In the event that the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a 24-hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving

3.12 LIMITING CONDITIONS FOR OPERATION BASES (Cont'd.)

several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the barriers are not functional, a fire watch is required to be maintained in the vicinity of the affected barrier until the barrier is restored to functional status.

4.12 SURVEILLANCE REQUIREMENT BASES

None

TABLE 3.12-1

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE
1. Main Control Room 34 Smoke Detectors	24
2. Unit 2/3 Aux. Elect. Equip. Room 12 Smoke Detectors	8
3. Unit 2/3 Computer Room 8 Smoke Detectors	6
4. Unit 2 Battery Room 10 Smoke Detectors	7
5. Unit 2 Turb Bldg. 480v MCC 17 Smoke Detectors	12
6. Unit 2 Turb Bldg. 4KV Switchgear 10 Smoke Detectors	7
7. Unit 2/3 Diesel Generator Area 4 Heat Detectors	3
8. Unit 2 Diesel Generator Area 3 Heat Detectors	2
9. Unit 2 Rx Bldg. 480v MCC (570') 3 Smoke Detectors	2
10. Unit 2 Rx Bldg. 480v MCC (517') 7 Smoke Detectors	5
11. Unit 2 Rx Bldg. 4KV Switchgear 4 Smoke Detectors	3
12. Unit 2 Standby Liquid Control Area 1 Smoke Detector	1

TABLE 3.12-2

SPRINKLER SYSTEMS

1. Unit 2 Emergency Diesel Oil Day Tank
2. Unit 2/3 Turbine Mezzanine 538' Elevation
3. Unit 2/3 Emergency Diesel Oil Day Tank
4. Unit 2 Reactor Feed Pump Area
5. Unit 2 Hydrogen Seal Oil Area
6. Unit 2 Condensate Feed Pump Area
7. Unit 2 HPCI Area
8. Unit 2 Turbine Building East Mezzanine Area
9. Unit 2 EHC Area
10. 2/3 Fire Pump Area
11. Unit 2 Trackway
12. Unit 2 Instrument Air Compressor Area

TABLE 3.12-3

CO_2 SYSTEMS

1. Unit 2 Emergency Diesel Generator
2. Unit 2/3 Emergency Diesel Generator

TABLE 3.12-4
FIRE HOSE STATIONS

NO.	LOCATION
F21	Cribhouse - 517-ft., by Bus 20
F54	Reactor Building - 589-ft., North East Wall at Elevator
F55	Reactor Building - 589-ft., South of Standby Liquid Tank
F56	Reactor Building - 589-ft., South East of Isolation Condenser
F57	Reactor Building - 589-ft., South West Stairway
F58	Reactor Building - 570-ft., North Wall at Elevator
F59	Reactor Building - 570-ft., Across from Cleanup Demineralizer P.C. Tank
F60	Reactor Building - 570-ft., C.R.D. Repair Room
F61	Reactor Building - 570-ft., West Wall Near RBCCW Tank
F61A	Reactor Building - 570-ft., by South Stairs
F62	Reactor Building - 545-ft., North Wall Near Elevator
F63	Reactor Building - 545-ft., South Wall at RBCCW Heat Exchanger
F64	Reactor Building - 545-ft., South West Stairway
F65	Reactor Building - 545-ft., North of Bus 23
F66	Reactor Building - 517-ft., at Elevator

TABLE 3.12-4 (Continued, page 2)

FIRE HOSE STATIONS

NO.	LOCATION
F67	Reactor Building - 517-ft., South East Wall
F68	Reactor Building - 517-ft., South East Stairway
F73	Reactor Building - 476-ft., 2B LPCI Pump
F74	Reactor Building - 476-ft., 2C Core Spray Pump
F80A	Turbine Building - 549-ft., North Wall Unit 2 Battery Room
F80B	Turbine Building - 549-ft., Outside of Passenger Elevator
F82	Turbine Building - 538-ft., Stator Cooling Pump
F82A	Turbine Building - 538-ft., East of Trackway Equipment Hatch
F82B	Turbine Building - 534-ft., Across from Switchgear 23 and 24
F84	Turbine Building - 534-ft., East of Standby Gas System
F85	Turbine Building - 517-ft., Near U-2 Transformer Valve
F86	Turbine Building - 517-ft., at U-2 Emergency Turbine Diesel
F87	Turbine Building - 517-ft., Across from 2C RFP West Walk
F88	Turbine Building - 495-ft., at C.R.D. Pumps
F89	Turbine Building - 469-ft., Across from 2D Condensate Pump

5.0 DESIGN FEATURES

5.1 Site

Dresden Unit 2 is located at the Dresden Nuclear Power Station which consists of a tract of land of approximately 953 acres located in the northeast quarter of the Morris 15-minute quadrangle (as designated by the United States Geological Survey), Goose Lake Township, Grudy County, Illinois. The tract is situated in portions of Sections 25, 26, 27, 34, 35, and 36 of Township 34 North, Range 8 East of the Third Principal Meridian.

5.2 Reactor

A. The core shall consist of not more than 724 fuel assemblies

B. The reactor core shall contain 177 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% of theoretical density.

5.3 Reactor Vessel

The reactor vessel shall be as described in Table 4.1.1 of the SAR. The applicable design codes shall be as described in Table 4.1.1 of the SAR.

5.4 Containment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the SAR.

B. The secondary containment shall be as described in Section 5.3.2 of the SAR and the applicable codes shall be as described in Section 12.1.1.3 of the SAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.2 of the SAR and the applicable codes shall be as described in Section 12.1.1.3 of the SAR.

5.5 Fuel Storage

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

5.0 DESIGN FEATURES (Cont'd.)

5.6 Seismic Design

The reactor building and all contained engineered safeguards are designed for the maximum credible earthquake ground motion with an acceleration of 20 per cent of gravity. Dynamic analysis was used to determine the earthquake acceleration, applicable to the various elevations in the reactor building.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization, Review, Investigation and Audit

A. The Station Superintendent shall have overall full-time responsibility for safe operation of the facility. During periods when the Station Superintendent is unavailable, he shall designate this responsibility to an established alternate who satisfies the ANSI N18.1 experience requirements for plant manager.

B. The corporate management which relates to the operation of this station is shown in Figure 6.1.1.

C. The normal functional organization for operation of the station shall be as shown in Figure 6.1.2. The shift manning for the station shall be as shown in Table 6.1.1.

D. Qualifications of the station management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971 with the exception of the Radiological Chemical Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September, 1975 and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents. The individual filling the position of Administrative Assistant shall meet the minimum acceptable level for "Technical Manager" as described in 4.2.4 of ANSI N18.1 - 1971.

A fire brigade of at least 5 members shall be maintained on-site at all times. This excludes the shift crew necessary for safe shutdown of the plant and any personnel required for essential functions during a fire emergency.

E. Restraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971.

A training program for the fire brigade shall be maintained under the direction of the Operating Engineer and shall meet or exceed the requirements of Section 27 of the NFPA Code - 1975, except for fire brigade training sessions which shall be held at least quarterly.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

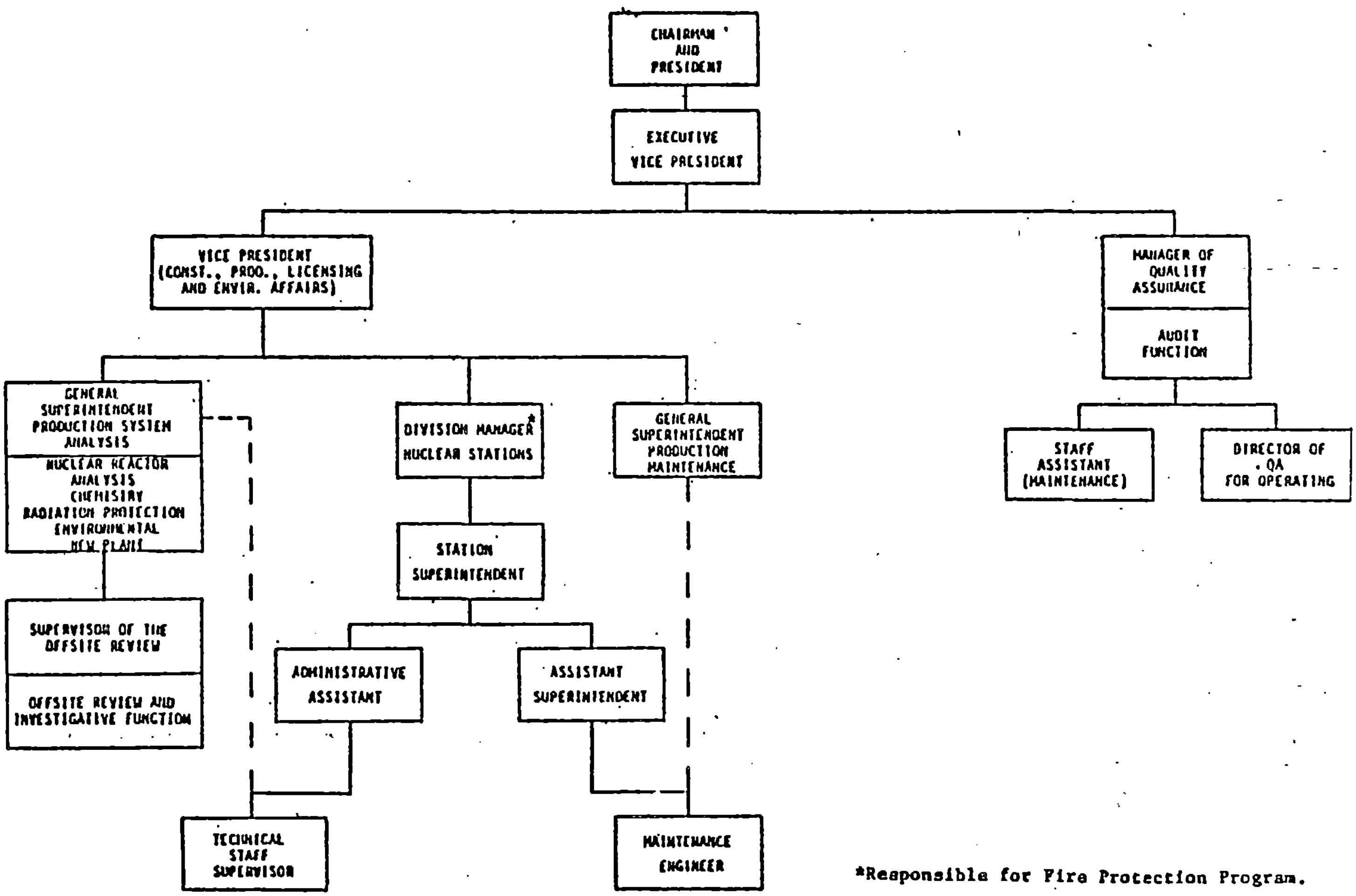
F. Retraining shall be conducted at intervals not exceeding two years.

G. The Review and Investigative Function and the Audit Function of activities affecting quality during facility operations shall be constituted and have the responsibilities and authorities outlined below:

1. The Supervisor of the Offsite Review and Investigative Function shall be appointed by the Vice President of Construction, Production, Licensing, and Environmental Affairs. The Audit Function shall be the responsibility of the Manager of Quality Assurance and shall be independent of operations.

   a. Offsite Review and Investigative Function

      The Supervisor of the Offsite Review and Investigative Function shall: (i) provide directions for the review and investigative function and appoint a senior participant to provide appropriate direction, (ii) select each participant for this function, (iii) select a complement of more than one participant who collectively possess background and qualifications in the subject matter under review to provide comprehensive interdisciplinary review coverage under this function, (iv) independently review and approve the findings and recommendations developed by personnel performing the review and investigative function, (v) approve and report in a timely manner all findings of noncompliance with NRC requirements and provide recommendations to the Station Superintendent, Division Manager Nuclear Stations, Manager of Quality Assurance, and the Vice President of Construction, Production, Licensing and Environmental Affairs. During periods when the Supervisor of the Offsite Review and Investigative Function is unavailable, he shall designate this responsibility to an established alternate who satisfies the formal training and experience requirements for the supervisor of the Offsite Review and Investigative Function.



*Responsible for Fire Protection Program.

FIGURE 6.1-1

CORPORATE ORGANIZATION

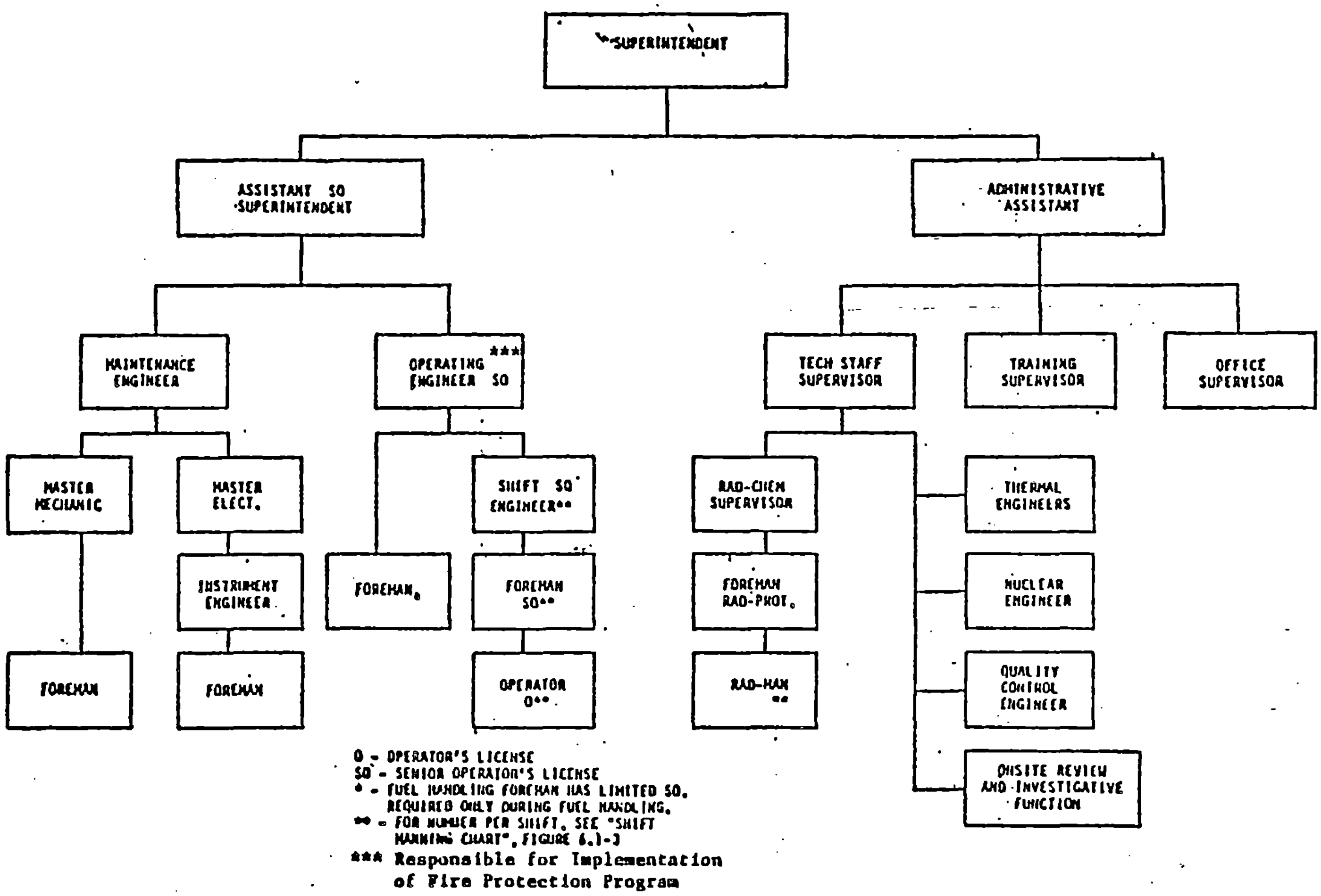

*SUPERINTENDENT
ASSISTANT SO SUPERINTENDENT
ADMINISTRATIVE ASSISTANT
MAINTENANCE ENGINEER
OPERATING ENGINEER SO ***
TECH STAFF SUPERVISOR
TRAINING SUPERVISOR
OFFICE SUPERVISOR
MASTER MECHANIC
MASTER ELECT.
SHIFT SO ENGINEER**
RAD-CHEM SUPERVISOR
THERMAL ENGINEERS
INSTRUMENT ENGINEER
FOREMAN
FOREMAN SO**
FOREMAN RAD-PROT.
NUCLEAR ENGINEER
FOREMAN
FOREMAN
OPERATOR O**
RAD-MAN **
QUALITY CONTROL ENGINEER
ONSITE REVIEW AND INVESTIGATIVE FUNCTION
O - OPERATOR'S LICENSE
SO - SENIOR OPERATOR'S LICENSE
* - FUEL HANDLING FOREMAN HAS LIMITED SO, REQUIRED ONLY DURING FUEL HANDLING.
** - FOR NUMBER PER SHIFT, SEE "SHIFT MANNING CHART", FIGURE 6.1-3
*** Responsible for Implementation of Fire Protection Program

FIGURE 6.1-2

TABLE 6.1.1

MINIMUM SHIFT MANNING CHART***

CONDITION OF ONE UNIT	CONDITION OF		NO. OF MEN IN EACH POSITION				
	SECOND UNIT	THIRD UNIT	LSO*	STA	LO*	NON-LIC.	RAD MEN
COLD SHUTDOWN	Cold Shutdown	Cold Shutdown	1*	0	3	5	1
	Cold Shutdown	Refuel	1**	0	3	5	1
	Cold Shutdown	Above Cold Shutdown	1	1	3	5	1
	Refuel	Refuel	2**	0	3	5	1
	Refuel	Above Cold Shutdown	2	1	3	5	1
	Above Cold Shutdown	Above Cold Shutdown	2	1	3	5	1
REFUEL	Refuel	Refuel	2	0	4	5	1
	Refuel	Above Cold Shutdown	2	1	4	5	1
	Above Cold Shutdown	Above Cold Shutdown	2	1	4	5	1
ABOVE COLD SHUTDOWN	Above Cold Shutdown	Above Cold Shutdown	3	1	4	5	1

STA - Shift Technical Advisor

LSO - Licensed Senior Operator

LO - Licensed Operator

NON-LIC. - Equipment Operators and Equipment Attendants

RAD MEN - Radiation Protection Men

* - Shall not operate units on which they are not licensed.

** - Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE OPERATIONS.

*** - Shift crew 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 on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

The responsibilities of the personnel performing this function are stated below. The Offsite Review and Investigative Function shall review:

(1) The safety evaluations for 1) changes to procedures, equipment or systems as described in the safety analysis report and 2) tests or experiments completed under the provision of 10 CFR Section 50.59 to verify that such actions did not constitute unreviewed safety questions. Proposed changes to the Quality Assurance Program description shall be reviewed and approved by the Manager of Quality Assurance.

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

(3) Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59 10 CFR.

(4) Proposed changes in Technical Specifications or NRC operating licenses.

(5) Noncompliance with NRC requirements, or of internal procedures or instructions having nuclear safety significance.

(6) Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function.

(7) Reportable Occurrences requiring 24 hour notification to the Commission.

(8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems or components.

(9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such changes.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Superintendent, Division Manager - Nuclear Stations and Manager of Quality Assurance.

b. Audit Function

The Audit Function shall be the responsibility of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance for Operating and to the Staff Assistant to the Manager of Quality Assurance for maintenance quality assurance activities.

Either shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within a period of two years or less as designated below.

(1) Audit of the Conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per year.

(2) Audit of the adherence to procedures, training and qualification of the station staff at least once per year.

(3) Audit of the results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or methods of operation that affect nuclear safety at least once per six months.

(4) Audit of the performance of activities required by the Quality Assurance Program to meet the Criteria of Appendix "B", 10 CFR 50.

(5) Audit of the Facility Emergency Plan and implementing procedures.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(6) Audit of the Facility Security Plan and implementing procedures.

(7) Audit onsite and offsite reviews.

(8) Audit of Facility Fire Protection Program and implementing procedures at least once per 24 months.

(9) Report all findings of noncompliance with NRC requirements and recommendations and results of each audit to the Station Superintendent, the Division Manager-Nuclear Stations, Manager of Quality Assurance, the General Superintendent of Production Systems Analysis, and to the Vice President of Construction, Production, Licensing and Environmental Affairs.

c. Authority

The Manager of Quality Assurance reports to the Executive Vice-President and the Supervisor of the Offsite Review and Investigative Function reports to the General Superintendent of Production Systems Analysis. Either the Manager of Quality Assurance or the Supervisor of the Offsite Review and Investigative Function has the authority to order unit shutdown or request any other action which he deems necessary to avoid unsafe plant conditions.

d. Records

(1) Reviews, audits and recommendations shall be documented and distributed as covered in 6.1.G.1.a and 6.1.G.1.b.

(2) Copies of documentation, reports, and correspondence shall be kept on file at the station.

e. Procedures

Written administrative procedures shall be prepared and maintained for the off-site reviews and investigative functions described in Specifications 6.1.G.1.a. These procedures shall cover the following:

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(1) Content and method of submission of presentations to the Supervisor of the Offsite Review and Investigative Function.

(2) Use of committees and consultants.

(3) Review and approval.

(4) Detailed listing of items to be reviewed.

(5) Method of (a) appointing personnel, (b) performing reviews, investigations, (c) reporting findings and recommendations of reviews and investigations, (d) approving reports, and (e) distributing reports.

(6) Determining satisfactory completion of action required based on approved findings and recommendations reported by personnel performing the review and investigative function.

f. Personnel

(1) The persons, including consultants, performing the review and investigative function, in addition to the Supervisor of the Offsite Review and Investigative Function, shall have expertise in one or more of the following disciplines as appropriate for the subject or subjects being reviewed and investigated.

(a) nuclear power plant technology
(b) reactor operations
(c) utility operations
(d) power plant design
(e) reactor engineering
(f) radiological safety
(g) reactor safety analysis
(h) instrumentation and control
(i) metallurgy
(j) any other appropriate disciplines required by unique characteristics of the facility.

(2) Individuals performing the Review and Investigative Function shall possess a minimum formal training and experience as listed below for each discipline.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(a) Nuclear Power Plant Technology

Engineering graduate or equivalent with 5 years experience in the nuclear power field design and/or operation.

(b) Reactor Operations

Engineering graduate or equivalent with 5 years experience in nuclear power plant operations.

(c) Utility Operations

Engineering graduate or equivalent with at least 5 years of experience in utility operation and/or engineering.

(d) Power Plant Design

Engineering graduate or equivalent with at least 5 years of experience in power plant design and/or operation.

(e) Reactor Engineering

Engineering graduate or equivalent. In addition, at least 5 years of experience in nuclear plant engineering, operation, and/or graduate work in nuclear engineering or equivalent in reactor physics is required.

(f) Radiological Safety

Engineering graduate or equivalent with at least 5 years of experience in radiation control and safety.

(g) Safety Analysis

Engineering graduate or equivalent with at least 5 years of experience in nuclear engineering.

(h) Instrumentation and Control

Engineering graduate or equivalent with at least 5 years of experience in instrumentation and control design and/or operation.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(i) Metallurgy

Engineering graduate or equivalent with at least 5 years of experience in the metallurgical field.

(3) The Supervisor of the Offsite Review and Investigative Function shall have experience and training which satisfy ANSI N18.1 - 1971 requirements for plant managers.

2. The Onsite Review and Investigative Function shall be supervised by the Station Superintendent.

a. Onsite Review and Investigative Function

The Station Superintendent shall: (i) provide direction for the Review and Investigative Function and appoint the Technical Staff Supervisor, or other comparably qualified individual as a senior participant to provide appropriate direction; (ii) approve participants for this function; (iii) assure that a complement of more than one participant who collectively possess background and qualifications in the subject matter under review are selected to provide comprehensive inter-disciplinary review coverage under this function; (iv) independently review and approve the findings and recommendations developed by personnel performing the Review and Investigative Function; (v) report all findings of noncompliance with NRC requirements, and provide recommendations to the Division Manager-Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function; and (vi) submit to the Offsite Review and Investigative Function for concurrence in a timely manner, those items described in Specification 6.1.G.1.a which have been approved by the Onsite Review and Investigative Function.

The responsibilities of the personnel performing this function are stated below:

(1) Review of: 1) procedures required by Specification 6.2 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Plant Superintendent to affect nuclear safety.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(2) Review of all proposed tests and experiments that affect nuclear safety.

(3) Review of all proposed changes to the Technical Specifications.

(4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

(5) Investigate all noncompliance with NRC requirements and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Division Manager-Nuclear Stations and to the Supervisor of the Offsite Review and Investigative Function.

(6) Review of facility operations to detect potential safety hazards.

(7) Performance of special reviews and investigations and reports thereon as requested by the Supervisor of the Offsite Review and Investigative Function.

(8) Review the Station Security Plan and shall submit recommended changes to the Division Manager-Nuclear Stations.

(9) Review the Emergency Plan and station implementing procedures and shall submit recommended changes to the Division Manager-Nuclear Stations.

(10) Review of reportable occurrences and actions taken to prevent recurrence.

b. Authority

The Technical Staff Supervisor is responsible to the Station Superintendent and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

when in his opinion such action is necessary. The Station Superintendent shall follow such recommendations or select a course of action that is more conservative regarding safe operation of the facility. All such disagreements shall be reported immediately to the Division Manager-Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function.

c. Records

(1) Reports, reviews, investigations, and recommendations shall be documented with copies to the Division Manager-Nuclear Stations, the Supervisor of the Offsite Review and Investigative Function, the Station Superintendent and the Manager of Quality Assurance.

(2) Copies of all records and documentation shall be kept on file at the station.

d. Procedures

Written administrative procedures shall be prepared and maintained for conduct of the Onsite Review and Investigative Function. These procedures shall include the following:

(1) Content and method of submission and presentation to the Station Superintendent, Division Manager-Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function.

(2) Use of committees.

(3) Review and approval.

(4) Detailed listing of items to be reviewed.

(5) Procedures for administration of the quality control activities.

(6) Assignment of responsibilities.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

e. Personnel

(1) The personnel performing the Onsite Review and Investigative Function, in addition to the Station Superintendent, shall consist of persons having expertise in:

(a) nuclear power plant technology
(b) reactor operations
(c) reactor engineering
(d) radiological safety and chemistry
(e) instrumentation and control
(f) mechanical and electric systems.

(2) Personnel performing the Onsite Review and Investigative Function shall meet minimum acceptable levels as described in ANSI N18.1-1971, Sections 4.2 and 4.4.

H. 1. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified off-site licensee personnel or an outside fire protection firm.

2. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

6.2 Plant Operating Procedures

A. Detailed written procedures including applicable checkoff lists covering items listed below shall be prepared, approved, and adhered to:

1. Normal startup, operation, and shutdown of the reactor and other systems and components involving nuclear safety of the facility.

2. Refueling operations.

3. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components including responses to alarms, suspected primary system leaks, and abnormal reactivity changes.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

4. Emergency conditions involving potential or actual release of radioactivity - "Generating Stations Emergency Plan" and station emergency and abnormal procedures.

5. Instrumentation operation which could have an effect on the safety of the facility.

6. Preventive and corrective maintenance operations which could have an effect on the safety of the facility.

7. Surveillance and testing requirements.

8. Tests and experiments.

9. Procedure to ensure safe shutdown of the plant.

10. Station Security Plan and implementing procedures.

11. Fire Protection Program implementation.

B. Radiation control procedures shall be maintained, made available to all station personnel and adhered to. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.

C. 1. Procedures for items identified in Specification 6.2.A and any changes to such procedures shall be reviewed and approved by the Operating Engineer and the Technical Staff Supervisor in the areas of operation, fuel handling, or instrument maintenance, and by Maintenance Engineer and Technical Staff Supervisor in the areas of plant maintenance and plant inspection. Procedures for items identified in Specification 6.2.B and any changes to such procedures shall be reviewed and approved by the Technical Staff Supervisor and the Radiological Chemical Supervisor. At least one person approving each of the above procedures shall hold a valid senior operator's license. In addition, these procedures and changes thereto must have authorization by the Station Superintendent before being implemented.

2. Work and instruction type procedures which implement approved maintenance or modification procedures shall be approved and authorized by the Maintenance Engineer where the written authority has been provided by the Station

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

Superintendent. The "Maintenance/Modification Procedure" utilized for safety related work shall be so approved only if procedures referenced in the "Maintenance/Modification Procedure" have been approved as required by 6.2.A. Procedures which do not fall within the requirements of 6.2.A or 6.2.B may be approved by the Department Heads.

D. Temporary changes to procedures 6.2.A and 6.2.B above may be made provided:

1. The intent of the original procedure is not altered.

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

3. The change is documented, reviewed by the Onsite Review and Investigative Function and approved by the Station Superintendent within 14 days of implementation.

E. Drills of the emergency procedures described in Specification 6.2.A.4 shall be conducted quarterly. These drills will be planned so that during the course of the year, communication links are tested and outside agencies are contacted.

6.3 Action to be Taken in the Event of a Reportable Occurrence in Plant Operation

Any reportable occurrence shall be promptly reported to the Division Manager-Nuclear Stations or his designated alternate. The incident shall be promptly reviewed pursuant to Specification 6.1.G.2.a(5) and a separate report for each reportable occurrence shall be prepared in accordance with the requirements of Specification 6.6.B.

6.4 Action to be Taken in the Event a Safety Limit is Exceeded

If a safety limit is exceeded, the reactor shall be shut down immediately and reactor operation shall not be resumed until authorized by the NRC. The conditions of shutdown shall be promptly reported to the Division Manager-Nuclear Stations or his designated alternate. The incident shall be reviewed pursuant to Specification 6.1.G.1.a and 6.1.G.2.a and a separate report for each occurrence shall be prepared in accordance with Specification 6.6.B.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

6.5 Plant Operating Records

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least five years.

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

2. Records of principal maintenance activities, including inspection and repair, regarding principal items of equipment pertaining to nuclear safety.

3. Records and reports of reportable and safety limit occurrences.

4. Records and periodic checks, inspection and/or calibrations performed to verify the Surveillance Requirements (See Section 4 of these Specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded.

5. Records of changes made to the equipment or reviews of tests and experiments to comply with 10 CFR 50.59.

6. Records of radioactive shipments.

7. Records of physic tests and other tests pertaining to nuclear safety.

8. Records of changes to operating procedures.

9. Shift Engineers Logs.

10. By-product material inventory records and source leak test results.

B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life of the plant.

1. Substitution or replacement of principal items of equipment pertaining to nuclear safety.

2. Changes made to the plant as it is described in the Safety Analysis Report.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

3. Records of new and spent fuel inventory and assembly histories.

4. (Deleted)

5. Updated, corrected, and as-built drawings of the plant.

6. Records of plant radiation and contamination surveys.

7. Records of off-site environmental monitoring surveys.

8. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant in accordance with 10 CFR 20.

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

10. Records of transient or operational cycling for those components that have been designed to operate safely for a limited number of transient or operational cycles.

11. Records of individual staff members indicating qualifications, experience, training and retraining.

12. Inservice inspections of the reactor coolant system.

13. Minutes of meetings and results of reviews performed by the off-site and on-site review functions.

14. Records for Environmental Qualification which are covered under the provisions of paragraph 6.7.

6.6 REPORTING REQUIREMENTS

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

A. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

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

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

2. A tabulation shall be submitted 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, (See Note); 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 totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

Note:

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

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experiences shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, US Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the appropriate Regional Office, to arrive no later than the 15th of each month following the calendar month covered by the report.

B. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. In general, the importance of an occurrence with respect to safety significance determines the immediacy of reporting required. In some cases, however, the significance of an event may not be obvious at the time of its occurrence. In such cases, the NRC shall be informed promptly of an increased significance in the licensee's assessment of the event. In addition, supplemental reports may be required to fully describe final resolution of occurrence. 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. Prompt Notification With Written Followup

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 Director of the appropriate Regional Office, or his designate no later than the first working day following the event, with a written followup report within two weeks. The written followup 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.

a. 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

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

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 B.1.e., B.1.f., or B.2.a. below.

b. 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 2.b. below.

c. 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.

d. Reactivity anomalies, involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation, greater than or equal to 1% delta k/k; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if sub-critical, an unplanned reactivity insertion of more than 0.5% delta k/k or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

of the functional requirements of systems required to cope with accidents analyzed in the SAR.

Note: For items B.1.e. and B.1.f. reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items B.2.b. and B.2.c. below.

g. Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

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

2. Thirty Day Written Reports

The reportable occurrences discussed below have lesser immediate importance than those described under B.1. above. Such events shall be the subject of written reports to the Director of 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.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

a. 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.

b. 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 B.2.a. and B.2.b. need not be reported except where test results themselves reveal a degraded mode above.

c. 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.

d. Abnormal degradation of systems other than those specified in item B.1.e. 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.

C. Unique Reporting Requirements

1. Radioactive Effluent Release Report

A report shall be submitted to the Commission within 60 days after January 1 and July 1 of each year specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents during the previous 6 months. The format and content of the report shall be in accordance with Regulatory Guide 1.21 (Revision 1) dated June 1974.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

2. Environmental Radioactivity Data

a. Standard Radiological Monitoring Program

(1) Non-Routine Report

(a) If a confirmed measured radionuclide concentration in an environmental sampling medium averaged over any calendar quarter sampling period exceeds the reporting level given in Table 4.8-1 and if the radioactivity is attributable to plant operation, a written report shall be submitted to the Director of the NRC Regional Office, with a copy to the Director, Office of Nuclear Reactor Regulation, within 30 days from the end of the quarter. When more than one of the radionuclides in Table 4.8-1 are detected in the medium, the reporting level shall have been exceeded if

$$\frac{\Sigma C_i}{RL_i} \text{ equal to or greater than } 1$$

where C is the concentration of the i^{th} radionuclide in the medium and RL is the reporting level of radionuclide i.

(b) If radionuclides other than those in Table 4.8-1 are detected and are due to plant effluents, a reporting level is exceeded if the potential annual dose to an individual is equal to or greater than the design objective doses of 10 CFR 50, Appendix I.

(c) This report shall include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous effect.

(2) Annual Operating Report

An annual report containing the data taken in the standard radiological monitoring program (Table 4.8-1) shall be submitted by March 31 of the next year. The content of the report shall include:

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(a) Results of environmental sampling summarized on a quarterly basis following the format of Regulatory Guide 4.8 Table 1 (December 1975); (individual sample results will be retained at the station);

(b) An assessment of the monitoring results and radiation dose via the principal pathways of exposure resulting from plant emissions of radioactivity; and

(c) Results of the census to determine the locations of animals producing milk for human consumption.

b. Environmental Dose Pathways Studies (EDPS)

The EDPS schedule for Dresden Station is May 1978 through April 1979 with the project report submitted by December 31, 1979.

3. Special Reports

Special reports shall be submitted as indicated in Table 6.6.1.

TABLE 6.6.1

SPECIAL REPORTS

AREA	SPECIFICATION REFERENCE	SUBMITTAL DATE
a. Response time of safety related instruments (2)	1.0.E (Dres. 1)	Annual Report
b. Main steam isolation valve and feedwater power operated isolation valves closure times (2)	3.7.B.1.c (Dres. 1)	Annual Report
c. Primary Coolant leakage to Drywell (4)	4.6.D Bases	5 years (1)
d. In-Service Inspection Evaluation (4)	Table 4.6.1	5 years (1)
e. Evaluation of ECCS operation (4)	3.3.F Bases	Upon completion of initial testing
f. Failed Fuel Detection (4)	3.2 Bases	5 years (1)
g. Main Steam Line Leakage to Steam Tunnel (4)	4.6.D Bases	5 years (1)
h. In-service Inspection Development (4)	4.6.1 Bases	5 years (1)
i. In-Service Inspection of Sensitized Stainless Steel Components (3)	4.6.F	4 years (1)
j. Secondary Containment Leak Rate Test (4)	3.7.C.1	within 90 days after completion of each test
k. High off-gas discharge rate (2)	3.8.A.4 (Dresden 1)	within 24 hours of occurrence
l. Radioactive Source Leak Testing (5)	3.8.F	Annual Report

NOTES:

1. The report shall be submitted within the period of time listed based on the commercial service date as the starting point.
2. Dresden 1 only
3. Dresden 2 only
4. Dresden 2 and 3 only.
5. The report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

6.7 ENVIRONMENTAL QUALIFICATION

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-19 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

50-249

AMENDMENT 75 TO FOL CONSISTING OF CHANGES TO TECHNICAL SPECIFICATION

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RECORDS FACILITY BRANCH

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

FACILITY OPERATING LICENSE

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

a. Commonwealth Edison Company (the applicant) has submitted to the Commission all technical information required by Provisional Construction Permit No., CPPR-22, the Atomic Energy Act of 1954, as amended (the Act), and the rules and regulations of the Commission to complete the application for a construction permit and facility license dated February 10, 1966, as supplemented by application for a facility license dated November 17, 1967 and amended by Amendment Nos. 8 through 24, dated August 30, 1968, November 21, 1968, February 28, 1969, March 18, 1969, April 16, 1969, May 20, 1969, July 2, 1969, July 22, 1969, August 5, 1969, August 8, 1969, August 10, 1969, August 18, 1969, September 2, 1969, October 16, 1969, May 7, 1970, August 11, 1970 and September 4, 1970, respectively, (the application); and supplemented by the applicant's letter dated December 17, 1970, and telegram dated December 18, 1970;

b. The Dresden Nuclear Power Station Unit 3 (the facility) has been substantially completed in conformity with Provisional Construction Permit No. CPPR-22, the application, the provisions of the Act and the rules and regulations of the Commission;

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

DLR Authorization Dated 6-28-71

d. There is reasonable assurance (i) that the facility can be operated at power levels not in excess of 2527 megawatt (thermal) in accordance with this license without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

e. The applicant is technically and financially qualified to engage in the activities authorized by this operating license; in accordance with the rules and regulations of the Commission;

f. The applicant has furnished proof of financial protection to satisfy the requirements of 10 CFR Part 140;

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

Facility Operating License No. DPR-25 is hereby issued to Commonwealth Edison Company (Commonwealth Edison), as follows:

1. This license applies to the Dresden Nuclear Power Station Unit 3, a single cycle, boiling, light water reactor, and electric generating equipment (the facility). The facility is located at the Dresden Nuclear Power Station in Grundy County, Illinois, and is described in the "Safety Analysis Report," as supplemented and amended (Amendment Nos. 8 through 24).

2. Subject to the conditions and requirements incorporated herein the Commission hereby licenses Commonwealth Edison:

   A. Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location at the Dresden Nuclear Power Station;

   B. Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear materials, not including plutonium, as reactor fuel, in accordance with the limitations for storage and amounts required for operation as described in the Final Safety Analysis Report, as supplemented and amended as of September 3, 1976;

Am. 21
9/03/76

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

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

Am. 31 1/30/78

2\. E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Dresden Nuclear Power Station, Units Nos. 1, 2 and 3.

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

Am. 1 3-2-71 DRL Authorized 6-28-71

A. Maximum Power Level

Commonwealth Edison is authorized to operate the facility at steady state power levels not in excess of 2527 megawatts (thermal), except that Commonwealth Edison shall not operate the facility at power levels in excess of five (5) megawatts (thermal), until satisfactory completion of modifications and final testing of the station output transformer, the auto-depressurization interlock, and the feedwater system, as described in Commonwealth Edison's telegrams dated February 26, 1971, have been verified in writing by the Commission.

Am. 75 08/06/84

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment 75 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

Commonwealth Edison shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

Commonwealth Edison shall keep facility operating records in accordance with the requirements of the Technical Specifications.

3. E. Restrictions

Operation in the coastdown mode is permitted to 40% power. Should off-normal feedwater heating be necessary for extended periods during coastdown (i.e. greater than 24 hours) the Licensee shall perform a safety evaluation to determine if the MCPR Operating Limit and calculated peak pressure for the worst case abnormal operating transient remain bounding for the new condition.

Am. 42
4/16/80

F. Equalizer Valve Restriction

Am. 5
8/29/75

The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation.

G. The licensee may proceed with and is required to complete the modification identified in Paragraphs 3.1.1 through 3.1.23 of the NRC's Fire Protection Safety Evaluation (SE) dated March 1978 on the facility. All modifications are to be completed by start-up following the 1979 Unit 3 refueling outage. In addition, the licensee shall submit the additional information identified in Table 3.1 of the SE in accordance with the schedule contained therein. In the event these dates for submittal cannot be met, the licensee shall submit a report, explaining the circumstances, together with a revised schedule.

Am. 33
3/22/78

H. Physical Protection

The licensee shall full implement and maintain in effect all provisions of the following Commission approved documents, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). These approved documents consist of information withheld from public disclosure pursuant to 10 CFR 2.790(d).

(1) "Security Plan for the Dresden Nuclear Power Station", dated November 19, 1977, as revised May 19, 1978, May 27, 1978, July 28, 1978 and February 19, 1979.

Am. 49
2/11/81

(2) "Dresden Nuclear Power Station Safeguards Contingency Plan", dated March 1980, as revised June 27, 1980, submitted pursuant to 10 CFR 73.40. The Contingency Plan shall be fully implemented, in accordance with 10 CFR 73.40(b), within 30 days of this approval by the Commission.

Am. 49
2-11-81

H. (3) "Dresden Nuclear Power Station Guard Training and Qualification Plan", submitted by letter dated August 16, 1979, as revised by letter dated August 11, 1980. This Plan shall be full implemented in accordance with 10 CFR 73.55(b)(4), within 60 days of this approval by the Commission. All security personnel shall be qualified within two years of this approval.

I. Systems Integrity

Am. 48
2/06/81

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

1. Provisions establishing preventive maintenance and periodic visual inspecton requirements, and

2. Leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

Am. 49
(see 3H)

J. Deleted.

K. Iodine Monitoring

Am. 48
2/06/81

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

1. Training of personnel;

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

L. Provisions to allow operation with one recirculation loop out of service:

Am. 54
7/9/81

1. The steady-state thermal power level will not exceed 50% of rated

Am. 63
4/29/82

2. The Minimum Critical Power Ratio (MCPR) Safety Limit will be increased 0.03 (TS 1.1.A and 3.3.5C)

3. The MCPR Limiting Condition for Operation (LCO) and Figure 3.5.2 will be increased 0.03 (TS 3.5.K)

Am. 54
7/9/81

4. The Minimum Average Planar Linear Heat Generation Rate (MAPLHGR) limits will be reduced by 0.7 for all fuel types.
(T.S. reference 3.5.I)

L. 5. The APRM Scram and Rod Block Setpoints and the RBM Setpoints shall be reduced by 3.5% to read as follows:

Am. 63
4/29/82

T.S. 2.1.A.1 S $\leq$ (.58 WD + 58.5)
T.S. 2.1.A.1* S $\leq$ (.58 WD + 58.5) FRP/MFLPD
T.S. 2.1.B S $\leq$ (.58 WD + 46.5)
T.S. 2.1.B* S $\leq$ (.58 WD + 46.5) FRP/MFLPD
T.S. 3.2.C (Table 3.2.3):
APRM Upscale $\leq$ (.58 WD + 46.5) FRP/MFLPD
RBM Upscale $\leq$ (.58 WD + 41.5)

* In the event that MFLPD exceeds FRP.

6. The suction valve in the idle loop is closed and electrically isolated until the idle loop is being prepared for return to service.

Am. 54
7/9/81

7. APRM flux noise will be measured once per shift and the recirculation pump speed will be reduced if the flux noise averages over 1/2 hour exceeds 5% peak to peak, as measured on the APRM chart recorder.

8. The core plate delta p noise will be measured once per shift and the recirculation pump speed will be reduced if the noise exceeds 1 psi peak to peak.

M. <u>Spent Fuel Storage Racks</u>

Am. 66
8/27/82

1. The licensee is authorized to install and use 33 high-density fuel storage racks in the spent fuel storage pool at Dresden Station Unit 3.

2. Fuel stored in the spent fuel pool shall have a U-235 loading less than or equal to 14.8 grams per axial centimeter.

Am. 56
10/29/81

3. No fuel loads heavier than the weight of a single spent fuel assembly and handling tool shall be carried over fuel stored in the spent fuel pool.

4. The licensee shall update the Dresden Unit 3 FSAR within 8 months from the effective date of this amendment to reflect the following commitments:

   a. A corrosion surveillance program for the racks to insure that any loss of neutron absorber material and/or swelling of the storage tubes is detected.

4. b. In situ neutron attenuation tests to verify that tubes and racks contain a sufficient number of Boral plates such that K-effective will not be greater than 0.95 when the spent fuel is in place.

c. If one boral plate is detected missing, the associated tube will be blocked to prohibit insertion of a fuel assembly. If more than one missing boral plate is detected per pool,

Am. 56
10/29/81

the licensee will remove the storage rack or racks containing any additional missing boral plates from the pool. Such storage racks will not be replaced in the pool until a specific criticality analysis covering the proposed corrective action has been submitted to and approved by the NRC.

d. Before any storage rack is place in the Dresden pools, the licensee will check each storage location with a plug gauge to confirm that the minimum dimension between the lead-in clips at the top of each storage location is at least 5.758 inches. If necessary, the licensee will grind down the storage clips to ensure this dimension is achieved.

(Renumbered)
(Per Am. 2)
(1-26-73)

4. This license is effective as of the date of issuance, and shall expire at Midnight October 14, 2006.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by

Peter A. Morris, Director

Division of Reactor Licensing

Enclosures:

Appendix A - Technical Specifications

Date of Issuances, January 12, 1971

TECHNICAL SPECIFICATION PAGE INDEX 3/09/84

Dresden Station Unit 3 Nuclear Operating License No. DPR-25

D PAGE	ORIGINAL SHEET	AMENDMENT NO.	DATE	*NEW PAGE
Appendix A Coversheet	X (original sheet)		01/12/71	i
i		53	06/25/81	ii
ii		23	05/23/76	iii
iii		33	05/01/78	iv
1		63	04/29/82	1.0-1
2		63	04/29/82	1.0-2
3		Change 22	08/29/75	1.0-3 & 4
4		42	04/16/80	1.0-5
5		63	04/29/82	1.0-6
6		63	04/29/82	1/2.1-2
6A		63	04/29/82	1/2.1-2 & 3
7		63	04/29/82	1/2.1-3 & 4
8		42	04/16/80	1/2.1-6
9		42	04/16/80	None
10		63	04/29/82	B 1/2.1-7
11		63	04/29/82	B 1/2.1-8
11A		63	04/29/82	B 1/2.1-9
12		42	04/16/80	B 1/2.1-10
13		63	04/29/82	B 1/2.1-11
14		63	04/29/82	B 1/2.1-12
15		63	04/29/82	B 1/2.1-13
15A		63	04/29/82	B 1/2.1-14
16		63	04/29/82	B 1/2.1-14
17		Change 22	08/09/75	B 1/2.1-15
18		63	04/29/82	B 1/2.1-16
18A		22	08/29/75	B 1/2.1-17
18B		52	04/30/81	B 1/2.1-18
19		71	01/04/84	1/2.2-1
20		74	03/09/84	B 1/2.2-2 & 3
21		63	04/29/82	B 1/2.2-4
21a		53	06/25/81	3.0-1 & 2
21b		53	06/25/81	B 3.0-3
22		72	02/14/84	3/4.1-1
22A		72	02/14/84	3/4.1-2
23		59	03/26/82	3/4.1-3
24		65	07/09/82	3/4.1-4
24a		59	03/26/82	3/4.1-5
25		59	03/26/82	3/4.1-6
26		59	03/26/82	3/4.1-7
27		59	03/26/82	3/4.1-8

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Dresden Station Unit 3 Nuclear Operating License No. DPR-25

.D PAGE	ORIGINAL SHEET	AMENDMENT NO.	DATE	*NEW PAGE
28		72	02/14/84	B 3/4.1-9
28a		72	02/14/84	B 3/4.1-10
29		59	03/26/82	B 3/4.1-11
30	X			B 3/4.1-12
31		72	02/14/84	B 3/4.1-14
32	X			B 3/4.1-15
33	X			B 3/4.1-16
34		63	04/29/82	B 3/4.1-17
35	X			3/4.2-1
36		63	04/29/82	3/4.2-2
36A		63	04/29/82	3/4.2-3
37	X			3/4.2-4
38		69	12/12/83	3/4.2-5
39		69	12/12/83	3/4.2-6
40		42	04/16/80	3/4.2-7
41		64	05/19/82	3/4.2-7
42		63	04/29/82	3/4.2-9
42a		63	04/29/82	3/4.2-10
43		64	05/19/82	3/4.2-11
44		69	12/12/83	3/4.2-12
4a		48	02/06/81	3/4.2-12
45		73	02/24/84	3/4.2-13
46		63	04/29/82	B 3/4.2-14
47		42	04/16/80	B 3/4.2-15
48		42	04/16/80	B 3/4.2-16 & 17
49		42	04/16/80	B 3/4.2-18
49A		Change 8	11/03/71	B 3/4.2-19
50	X			B 3/4.2-20
51	X			B 3/4.2-21
52	X			B 3/4.2-21
53	X			B 3/4.2-23
54		11	01/15/76	3/4.3-1
55		59	03/26/82	3/4.3-2
56		68	11/03/83	3/4.3-4
57		42	04/16/80	3/4.3-5, 6, 7
57A		42	04/16/80	3/4.3-8 & 9
58		70	12/12/83	3/4.3-10
59		17	05/04/76	3/4.3-11
60		12	02/25/76	3/4.3-12 & 13
61		Revised	09/30/74	B 3/4.3-14
61a		63	04/29/82	B 3/4.3-15 & 16
2		59	03/26/82	B 3/4.3-17

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TECHNICAL SPECIFICATION PAGE INDEX 3/09/84

Dresden Station Unit 3 Nuclear Operating License No. DPR-25

OLD PAGE	ORIGINAL SHEET	AMENDMENT NO.	DATE	*NEW PAGE
62a		63	04/29/82	B 3/4.3-17
62b		63	04/29/82	B 3/4.3-18
63		63	04/29/82	B 3/4.3-19
64		63	04/29/82	B 3/4.3-20
65		63	04/29/82	B 3/4.3-21
66		Change 28 and 19	04/19/74	3/4.4-1
67		Change 28 and 19	04/19/74	3/4.4-2
68		Change 28 and 19	04/19/74	3/4.4-3
69	X			3/4.4-4
70	X			3/4.4-5
71		42	04/16/80	B 3/4.4-6
72	X			B 3/4.4-7
73		Corrected page	04/06/78	3/4.5-1
74		Change 4	06/15/71	3/4.5-2 & 3
75		Change 4	06/15/71	3/4.5-4
76		Change 23	05/16/75	3/4.5-5
77		65	07/09/82	3/4.5-6 & 7
78		63	04/29/82	3/4.5-8 & 9
79		Change 28 and 19	04/19/74	3/4.5-10
80		Change 28 and 19	04/19/74	3/4.5-11 & 12
81		18	06/09/76	3/4.5-13
81A		Change 21	07/16/74	3/4.5-14
81B		63	04/29/82	3/4.5-15
81B-1		63	04/29/82	3/4.5-16
81C-1		74	03/09/84	3/4.5-17
81C-2		63	04/29/82	3/4.5-18
81C-3		42	04/16/80	3/4.5-19
81C-4		63	04/29/82	3/4.5-20
81C-5		42	04/16/80	3/4.5-21
81D		74	03/09/84	3/4.5-22
81E		74	03/09/84	3/4.5-24
81E-1		74	03/09/84	3/4.5-23 & 24
81F		Change 22	08/29/75	3/4.5-23
81G		36	06/14/78	B 3/4.5-26
82		63	04/29/82	B 3/4.5-27
83		Change 4	06/15/71	B 3/4.5-28 & 29
84		Change 22	08/29/75	B 3/4.5-30
85		18	06/09/76	B 3/4.5-31
85A		63	04/29/82	B 3/4.5-31 & 32
85B		63	04/29/82	B 3/4.5-33
85B-1		63	04/29/82	B 3/4.5-34

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OLD PAGE	ORIGINAL SHEET	AMENDMENT NO.	DATE	*NEW PAGE
85C		Change 22	08/29/75	B 3/4.5-34
85D		28	05/02/77	B 3/4.5-35
86		28	05/02/77	B 3/4.5-36
86A		74	03/09/84	B 3/4.5-37
86B		Change 22	08/29/75	B 3/4.5-38
86C		Change 22	08/29/75	B 3/4.5-39
86D		Change 22	08/29/75	B 3/4.5-40
87		34	05/03/78	3/4.6-1
88		39	08/13/79	3/4.6-2
89		50	03/27/81	3/4.6-3, 4, & 5
90		71	01/04/84	3/4.6-6
91		47	12/31/80	3/4.6-7 & 8
91a		25	11/19/76	3/4.6-9
91a-1		25	11/19/76	3/4.6-10 & 11
91a-2		25	11/19/76	3/4.6-12
91b		67	10/21/83	3/4.6-13
91c		62	04/28/82	3/4.6-14
91c-1		62	04/28/82	3/4.6-22
91d-1		67	10/21/83	3/4.6-18
91d-2		62	04/28/82	3/4.6-19
91e-1		67	10/21/83	3/4.6-20
91e-2		67	10/21/83	3/4.6-21
91f		37	11/13/78	3/4.6-24
92		37	11/13/78	B 3/4.6-25
93		39	08/13/79	B 3/4.6-26
93a		39	08/13/79	B 3/4.6-27
94	X			B 3/4.6-29
94a		39	08/13/79	B 3/4.6-30
95	X			B 3/4.6-28
96	X			B 3/4.6-31
97	X			B 3/4.6-32
98		47	12/31/80	B 3/4.6-33
99		Change 28 and 19	04/19/74	B 3/4.6-34
99a		62	04/28/82	B 3/4.6-35
99b		67	10/21/83	B 3/4.6-37
100		Left Blank	12/31/80	None
101		Left Blank	12/31/80	None
102		Left Blank	12/31/80	None
103		Left Blank	12/31/80	None

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OLD PAGE	ORIGINAL SHEET	AMENDMENT NO.	DATE	*NEW PAGE
104		Left Blank	12/31/80	None
104A		Left Blank	12/31/80	None
104B		Left Blank	12/31/80	None
105		Left Blank	12/31/80	None
106		Left Blank	12/31/80	None
107		Left Blank	12/31/80	None
108		18	06/09/76	3/4.7-1
108A		60	04/01/82	3/4.7-3
109		63	04/29/82	3/4.7-4, 5, & 7
110		63	04/29/82	3/4.7-8 & 9
111		Left Blank	04/01/82	None
112		Left Blank	04/01/82	None
113		Left Blank	04/01/82	None
114		Left Blank	04/01/82	None
115		Left Blank	04/01/82	None
116		Change 28 and 19	04/19/74	3/4.7-10
116a		Change 28 and 19	04/19/74	3/4.7-11, 12, 13
117		Change 28 and 19	04/19/74	3/4.7-14
117a		Change 28 and 19	04/19/74	3/4.7-15 & 16
117b		35	06/07/78	3/4.7-17 & 18
118		15	11/17/76	3/4.7-19
118a		15	11/17/76	3/4.7-20 & 21
118b		15	11/17/76	3/4.7-22
118c		15	11/17/76	3/4.7-23
119		15	11/17/76	3/4.7-24
120	X			3/4.7-26
121	X			3/4.7-27 & 28
122		20	08/09/76	3/4.7-29 & 30
123		43	04/17/80	3/4.7-31
124	X			3/4.7-32
125		42	04/16/80	B 3/4.7-33
125a		Change 25	10/06/75	B 3/4.7-34
126		Change 28 and 19	04/19/74	B 3/4.7-35
126a		Change 28 and 19	04/19/74	B 3/4.7-36 & 37
127		35	06/07/78	B 3/4.7-38
127a		35	06/07/78	B 3/4.7-39
128		15	11/17/76	B 3/4.7-40
129		Change 25	10/06/75	B 3/4.7-40
129a		Change 25	10/06/75	B 3/4.7-41 & 42
130		60	04/01/82	B 3/4.7-43

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131		Change 28 and 19	04/19/74	B 3/4.7-43 & 44
131a		15	11/17/76	B 3/4.7-45
131b		15	11/17/76	B 3/4.7-46
132		15	11/17/76	B 3/4.7-46
133		Change 7	10/07/71	3/4.8-1 & 2
133a		46	12/29/80	3/4.8-3 & 4
134	X			3/4.8-5
134a	X			3/4.8-6 & 7
134b	X			3/4.8-8
135		53	06/25/81	3/4.8-9 & 10
136	X			3/4.8-11
137		30	10/18/77	3/4.8-12 & 13
137a		21	09/03/76	3/4.8-14 & 15
138		30	10/18/77	3/4.8-15 & 16
139		30	10/18/77	3/4.8-16
140		30	10/18/77	3/4.8-17
141-142B (one page only)		30	10/18/77	None
143	X			B 3/4.8-18
143a	X			B 3/4.8-19
[illegible]4		30	10/18/77	B 3/4.8-20
145		30	10/18/77	B 3/4.8-21 & 22
145a		30	10/18/77	B 3/4.8-23
146		64	05/19/82	3/4.9-1
147		64	05/19/82	3/4.9-2 & 3
148		64	05/19/82	3/4.9-4
148a		64	05/19/82	3/4.9-4, 5, & 6
149	X			B 3/4.9-7
150		64	05/19/82	B 3/4.9-8
151		44	09/08/80	3/4.10-1
152		44	09/08/80	3/4.10-2 & 3
153		Change 17 (DPR-19) and 9	03/17/72	3/4.10-4
154		19	06/03/76	3/4.10-5 & 6
154a		19	06/03/76	3/4.10-7
155		44	09/08/80	B 3/4.10-8
155a		19	06/03/76	B 3/4.10-9
156		19	06/03/76	B 3/4.10-10
156a		14	05/12/76	3/4.11-1 & 2
156b		14	05/12/76	3/4.11-3
156c		14	05/12/76	3/4.11-4
156d		61	04/12/82	3/4.12-1
[illegible]56e		61	04/12/82	3/4.12-2, 3, & 4

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OLD PAGE	ORIGINAL SHEET	AMENDMENT NO.	DATE	*NEW PAGE
156f		61	04/12/82	3/4.12-5
156g		61	04/12/82	3/4.12-6 & 7
156h		61	04/12/82	3/4.12-8
156i		61	04/12/82	3/4.12-9, 10 & 11
156j		61	04/12/82	3/4.12-12
156k		61	04/12/82	3/4.12-13 & 14
156l		61	04/12/82	B 3/4.12-15 & 16
156m		61	04/12/82	B 3/4.12-17
156n		61	04/12/82	B 3/4.12-18
156o		61	04/12/82	B 3/4.12-19
156p		61	04/12/82	B 3/4.12-23
156q		61	04/12/82	B 3/4.12-24
157		74	03/09/84	5-1 & 2
158		55	08/21/81	6-1 & 2
159		33	05/01/78	6-3
160		33	05/01/78	6-4
161		55	08/21/81	6-5
162		26	02/04/77	6-6
163		33	05/01/78	6-7 & 8
164		26	02/04/77	6-9
165		26	02/04/77	6-10 & 11
166		26	02/04/77	6-12 & 13
167		26	02/04/77	6-14
167a		33	05/01/78	6-14
168		33	05/01/78	6-15
169		32	03/01/78	6-16
175		26	02/04/77	6-17
176		Per NRC Order	10/24/80	6-18
177		32	03/01/78	6-18 & 19
177b		32	03/01/78	6-20
177c		Change 26	12/04/75	6-20
177d		Change 26	12/04/75	6-21
177e		Change 26	12/04/75	6-22
177f		30	10/18/77	6-23
177g		30	10/18/77	6-25
178		21	09/03/76	6-26
179		21	09/03/76	6-26
180		Per NRC Order	10/24/80	6-27

* Approximate Page

APPENDIX A

TO

OPERATING LICENSE DPR-25

TECHNICAL SPECIFICATIONS

AND BASES

FOR

DRESDEN NUCLEAR POWER STATION UNIT 3

GRUNDY COUNTY, ILLINOIS

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

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List of Tables

List of Figures

1.0 DEFINITIONS

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

A. (Deleted)

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.

C. 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 XN-3 correlation. (Reference XN-NF-512).

D. Hot Standby - Hot standby means operation with the reactor critical, system pressure less than 600 psig, and the main steam isolation valves closed.

E. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

F. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm, or trip. Response time is not part of the routine instrument calibration, but will be checked once per cycle.

G. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument response alarm, and/or initiating action.

H. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

I. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

J. 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 represents 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.

K. Fraction of Limiting Power Density (FLPD) - For fuel fabricated by GE, the fraction of limiting power density is the ratio of the Linear Heat Generation Rate (LHGR) existing at a given location to the design LHGR for that bundle type. FLPD does not apply to Exxon Nuclear Company (ENC) fuel.

L. Logic System Function Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

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

N. Mode - The reactor mode is that which is established by the mode-selector-switch.

O. Operable - A system, subsystem, train, component, or device shall be operable when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

P. Operating - Operating means that a system, subsystem, train, component or device is performing its intended functions in its required manner.

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

R. 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 connecting 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.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

S. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

T. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady-state power level of 2527 thermal megawatts.

U. Rated Thermal Power - Rated thermal power means a steady-state power level of 2527 thermal megawatts.

V. 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 thermal power.

1. Startup/Hot Standby Mode - In this mode the reactor protection scram trips, initiated by condensor low vacuum and main steamline isolation valve closure, are by-passed when reactor pressure is less than 600 psig; the low pressure main steamline isolation valve closure trip is bypassed, the reactor protection system is energized with IRM neutron-monitoring system trips and control rod withdrawal interlocks in service.

2. Run Mode - In this mode the reactor protection system is energized with APRM protection and RBM interlocks in service.

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

X. 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 subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

Y. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary system are assured. Exceeding such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission (NRC) 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.

Z. 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 are secured in the isolated position.

AA. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alternations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized.

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.

BB. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

CC. Surveillance Interval - Each surveillance requirement shall be performed within the specified surveillance interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval.

b. A total maximum combined interval time for any 3 consecutive intervals not to exceed 3.25 times the specified surveillance interval.

DD. Fraction of Rated Power (FRP) - The fraction of rated power is the ratio of core thermal power to rated thermal power of 2527 Mwth.

EE. 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.

FF. Maximum Fraction of Limiting Power Density (MFLPD) - The maximum fraction of limiting power density is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

1.1 SAFETY LIMIT

FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to these variables which monitor the fuel thermal behavior.

Objective:

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications:

A. Reactor Pressure greater than 800 psig and Core Flow greater than 10% of Rated.

The existence of a minimum critical power ratio (MCPR) less than 1.06 for GE 8x8R fuel, or less than 1.05 for ENC or GE 8x8 fuel, shall constitute violation of the MCPR fuel cladding integrity safety limit.

2.1 LIMITING SAFETY SYSTEM SETTING

FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications:

A. Neutron Flux Trip Settings

The limiting safety system trip settings shall be as specified below:

1. APRM Flux Scram Trip Setting (Run Mode)

When the reactor mode switch is in the run position, the APRM flux scram setting shall be:

S less than or equal to $[.58W_D + 62]$

with a maximum set point of 120% for core flow equal to 98×10^6 lb/hr. and greater, where:

S - setting in per cent of rated power.

1.1 SAFETY LIMIT (Cont'd.)

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

W_D = per cent of drive flow required to produce a rated core flow of 98 Mlb/hr.

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

Where: S is less than or equal to
$(.58W_D + 62)$ [FRP/MFLPD]

FRP = fraction of rated thermal power (2527 MWt)

MFLPD = maximum fraction of limiting power density for GE fuel

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

This adjustment may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

2. APRM Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

1.1 SAFETY LIMIT (Cont'd.)

B. Core Thermal Power Limit (Reactor Pressure is less than or equal to 800 psig)

When the reactor pressure is less than or equal to 800 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

When the reactor mode switch is in the refuel startup/hot standby position, the APRM scram shall be set at less than or equal to 15% of rated neutron flux.

3. IRM Flux Scram Trip Setting

The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

B. APRM Rod Block Setting

The APRM rod block setting shall be:

S is less than or equal to [$.58W_D + 50$]

The definitions used above for the APRM scram trip apply.

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

S is less than or equal to ($.58W_D + 50$) [FRP/MFLPD]

The definitions used above for the APRM scram trip apply.

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

1.1 SAFETY LIMIT (Cont'd.)

C. Power Transient

1. The neutron flux shall not exceed the scram setting established in Specification 2.1.A for longer than 1.5 seconds as indicated by the process computer.

2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds the scram setting established by Specification 2.1.A and a control rod scram does not occur.

D. Reactor Water Level (Shutdown Condition)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core.

Note: Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

The adjustment may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

C. Reactor low water level scram setting shall be greater than or equal to 144" above the top of the active fuel at normal operating conditions.

Note: Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

D. Reactor low water level ECCS initiation shall be 84" (plus 4", minus 0") above the top of the active fuel at normal operating conditions.

Note: Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

1.1 SAFETY LIMIT (Cont'd.)

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

E. Turbine stop valve scram shall be less than or equal to 10% valve closure from full open.

F. Generator Load Rejection Scram shall initiate upon actuation of the fast closure solenoid valves which trip the turbine control valves.

G. Main Steamline Isolation Valve Closure Scram shall be less than or equal to 10% valve closure from full open.

H. Main Steamline Pressure initiation of main steamline isolation valve closure shall be greater than or equal to 850 psig.

I. Turbine Control Valve Fast Closure Scram on loss of control oil pressure shall be set at greater than or equal to 900 psig.

1.1 SAFETY LIMIT BASES

FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damages would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than the MCPR fuel cladding integrity safety limit. MCPR greater than the MCPR fuel cladding integrity safety limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity by assuring that the fuel does not experience transition boiling.

The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosions or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforation signals a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling. See reference XN-NF-524.

A. Reactor Pressure greater than 800 psig and Core Flow greater than 10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical

1.1 SAFETY LIMIT BASES (Cont'd.)

power ratio (CPR) which is the ratio of the bundle power which would produce onset of boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. (Figure 2.1-3).

The MCPR Fuel Cladding Integrity Safety Limit assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR=1.00) and the MCPR Fuel Cladding Integrity Safety Limit is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the XN-3 critical power correlation. Refer to XN-NF-524 for the methodology used in determining the MCPR Fuel Cladding Integrity Safety Limit.

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. The assumed reactor conditions used in defining the safety limit introduce conservatism into the limit because boundingly high radial power peaking factors and boundingly flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that during sustained operation at the MCPR Fuel Cladding Integrity Safety Limit there would be no transition boiling in the core. If boiling transition were to occur, however, there is reason to believe that the integrity of the fuel would not necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach; much of the data indicates that LWR fuel can survive for an extended period in an environment of transition boiling.

1.1 SAFETY LIMIT BASES (Cont'd.)

If the reactor pressure should ever exceed the limit of applicability of the XN-3 critical power correlation as defined in XN-NF-512, it would be assumed that the MCPR Fuel Cladding Integrity Safety Limit had been violated. This applicability pressure limit is higher than the pressure safety limit specified in Specification 1.2. For fuel fabricated by General Electric Company, operation is further constrained to a maximum linear heat generation rate (LHGR) of 13.4 kW/ft by Specification 3.5.J. This constraint is established to provide adequate safety margin to 1% plastic strain for abnormal operational transients initiated from high power conditions. Specification 2.1.A.1 provides for equivalent safety margin for transients initiated from lower power conditions by adjusting the APRM flow-biased scram by the ratio of FRP/MFLPD. Specification 3.5.J establishes the maximum value of LHGR which cannot be exceeded during steady power operation for GE fuel types.

For fuel fabricated by Exxon Nuclear Company, (ENC) fuel design criteria have been established to provide protection against fuel centerline melting and cladding strain, ENC has performed fuel design analysis which demonstrate that centerline melting is not predicted to occur during transient overpower conditions throughout the life of the fuel. Protection of the MCPR and MAPLHGR limits and operation within the power distribution assumptions of the fuel design analysis will provide adequate protection against centerline melt and ensures compliance with ENC's clad overstrain criteria for steady state and transient operation. Since ENC's design criteria are more conservative than the 1% plastic strain limitation on GE fuel, the LHGR limitation and APRM scram adjustment for GE fuel established in specifications 3.5.J and 2.1.A.1 respectively are unnecessary for the protection of ENC fuel. The procedural controls of specification 3.1.B will ensure that operation of ENC fuel remains within the power distribution assumptions of the fuel design analysis.

B. Core Thermal Power Limit (Reactor Pressure less than 800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of $28x10^3$ lbs/hr. bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow

1.1 SAFETY LIMIT BASES (Cont'd.)

with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. At 25% of rated thermal power, the peak powered bundle would have to be operating at 3.84 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

C. Power Transient

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel which is 8 to 9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail. In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a significant increase in surface heat flux occurs.

Control rod scram times are checked as required by Specifications 4.3.C. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analyses show that even if the bypass system fails to operate, the MCPR fuel cladding integrity safety limit is not exceeded. Thus, use of a 1.5 second limit provides additional margin.

The computer provided has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be

1.1 SAFETY LIMIT BASES (Cont'd.)

available for any scram analysis, Specification 1.1.C.2 will be relied on to determine if a safety limit has been violated.

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel* provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

*Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

2.1 LIMITING SAFETY SYSTEM SETTING BASES

FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the units have been analyzed throughout the spectrum of planned operating conditions up to the rated thermal power condition of 2527 MWt. In addition, 2527 MWt is the licensed maximum steady-state power level of the units. This maximum steady-state power level will never knowingly be exceeded. See Reference XN-NF-79-71.

Conservatism is incorporated into the transient analyses which define the MCPR operating limits. Variables which inherently possess little or no uncertainty or whose uncertainty has little or no effect on the outcome of the limiting transient are selected at bounding values. Variables which possess significant uncertainty that may have undesirable effects on thermal margins are addressed statistically. Statistical methods used in the transient analyses are described in XN-NF-81-22. The MCPR operating limits are established such that the occurrence of the limiting transient will not result in the violation of the MCPR Fuel Cladding Integrity Safety Limit in at least 95% of the random statistical combinations of uncertainties. In general, the variables with the greatest statistical significance to the consequences of anticipated operational occurrences are the reactivity feedback associated with the formation and removal of coolant voids and the timing of the control rod scram.

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

Steady-state operation without forced recirculation will not be permitted, except during startup testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

The bases for individual trip settings are discussed in the following paragraphs. For analyses of the thermal consequences of the transients, the MCPR's stated in paragraph 3.5.K as the limiting condition of operation bound those which are conservatively assumed to exist prior to initiation of the transients.

A. Neutron Flux Trip Settings

1. 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. 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 percent 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.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

The scram trip setting must be adjusted to ensure that the LHGR transient peak for G.E. fuel is not increased for any combination of Maximum Fraction of Limiting Power Density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in specification 2.1.A.1 when the MFLPD is greater than the fraction of rated power (FRP).

The adjustment may also be accomplished by increasing the APRM gain by the reciprocal of FRP/MFLPD. This provides the same degree of protection as reducing the trip setting by FRP/MFLPD by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

2. APRM Flux Scram Trip Setting
(Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

3. IRM Flux Scram Trip Setting

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are broken down into 10 ranges, each being one-half of a decade in size.

The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservation was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the MCPR fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

B. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR fuel cladding integrity safety limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worse case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward or APRM gain increased if the maximum fraction of limiting power density for G.E. fuel exceeds the fraction of rated power, thus preserving the APRM rod block safety margin.

C. Reactor Low Water Level Scram - The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained. The scram setpoint is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

D. Reactor Low Low Water Level ECCS Initiation Trip Point - The emergency core 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 well below the clad melting 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 emergency core cooling system component was established based on the reactor low water level scram setpoint. To lower the setpoint of the low water level scram would increase the capacity requirement for each of the ECCS 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 ECCS capacity requirements.

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

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

E. Turbine Stop Valve Scram - The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the MCPR fuel cladding integrity safety limit, even during the worst case transient that assumes the turbine bypass is closed.

F. Generator Load Rejection Scram - The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass; i.e., it prevents MCPR from becoming less than the MCPR fuel cladding integrity safety limit for this transient. For the load rejection without bypass transient from 100% power, the peak heat flux (and therefore LHGR) increases on the order of 15% which provides wide margin to the value corresponding to fuel centerline melting and 1% cladding strain.

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure - The low pressure isolation at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 850 psig would not necessarily constitute an unsafe condition.

H. Main Steam Line Isolation Valve Closure Scram - The low pressure isolation of the main steam lines at 850 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scram set at 10% valve closure, there is no appreciable increase in neutron flux.

I. Turbine Control Valve Fast Closure Scram

The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the generator load rejection scram since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection scram on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the generator load rejection. The scram setpoint of 900 psig is set high enough to provide the necessary anticipatory function and low enough to minimize the number of spurious scrams. Normal operating pressure for this system is 1250 psig. Finally the control valve will not start to close until the fluid pressure is 600 psig. Therefore, the scram occurs well before valve closure begins.

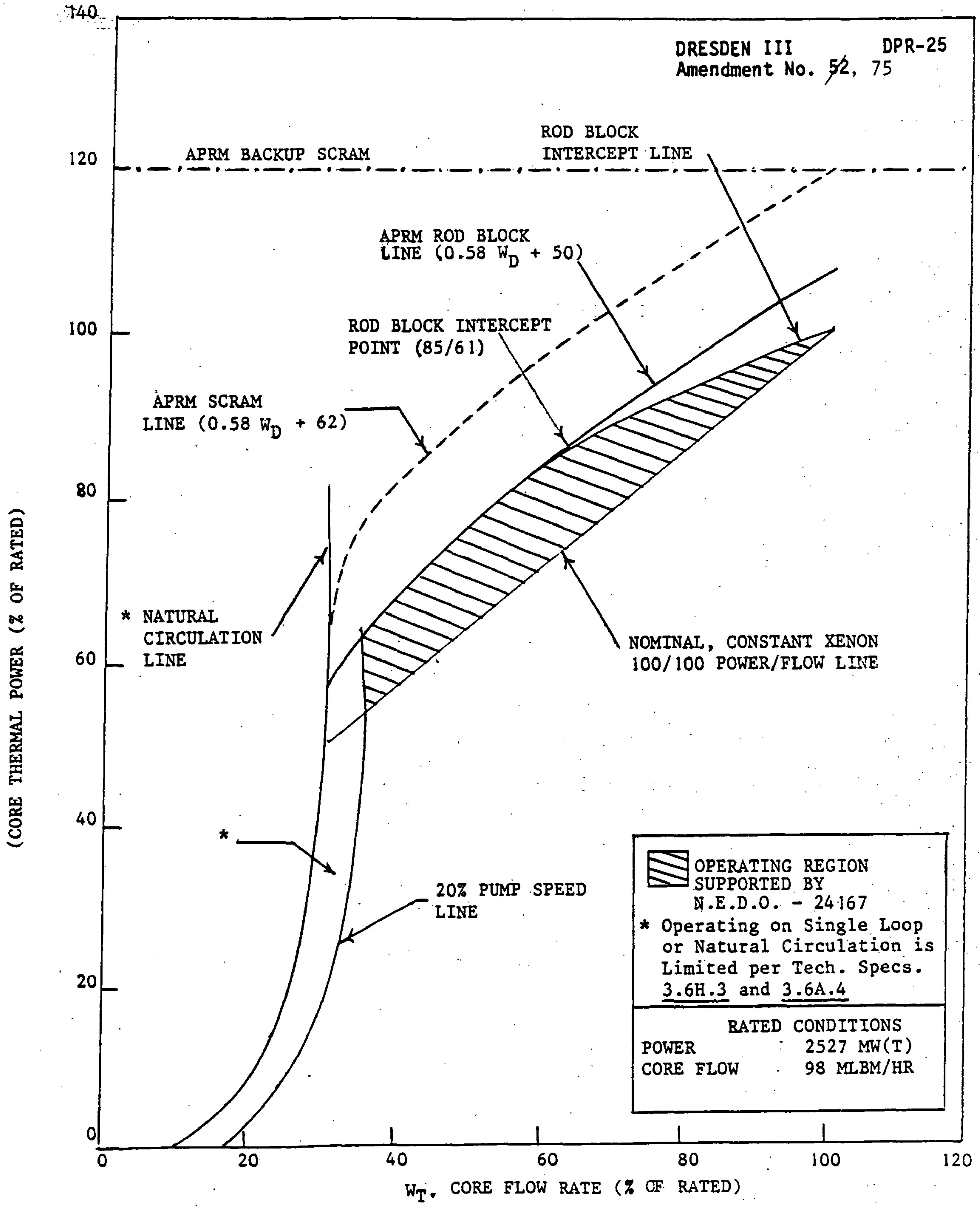


FIGURE 2.1-3
(SCHEMATIC)
APRM FLOW BIAS SCRAM RELATIONSHIP
TO NORMAL OPERATING CONDITIONS

1.2 SAFETY LIMIT

REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1345 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 LIMITING SAFETY SYSTEM SETTING

REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

A. Reactor Coolant High Pressure Scram shall be less than or equal to 1060 psig.

B. Primary System Safety Valve Nominal Settings shall be as follows:

   1 valve at 1135 psig*
   2 valves at 1240 psig
   2 valves at 1250 psig
   2 valves at 1260 psig
   2 valves at 1260 psig

   The allowable setpoint error for each valve shall be plus or minus 1%.

* Target Rock combination safety/relief valve

1.2 SAFETY LIMIT 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 pressure safety limit of 1345 psig as measured by the vessel steam space pressure indicator insures margin to 1375 psig at the lowest elevation of the reactor vessel. The 1375 psig value is derived from the design pressures of the reactor pressure vessel and coolant system piping . The respective design pressures are 1250 psig at 575°F and 1175 psig at 560°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% X 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the reactor vessel. The design pressure for the recirc. suction line piping (1175 psig) was chosen relative to the reactor vessel design pressure. Demonstrating compliance of the peak vessel pressure with the ASME overpressure protection limit (1375 psig) assures compliance of the suction piping with the USASI limit (1410 psig). Evaluation methodology used to assure that this safety limit pressure is not exceeded for any reload is documented in Reference XN-NF-79-71. The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig: this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At that pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the primary system piping and provide a similar margin of protection at the established safety pressure limit.

The normal operating pressure of the reactor coolant system is 1000 psig. For the turbine trip or loss of electrical load transients, the turbine trip scram or generator load rejection

1.2 SAFETY LIMIT BASES (Cont'd.)

scram, together with the turbine bypass system, limit the pressure to approximately 1100 psig (See Note below). In addition, pressure relief valves have been provided to reduce the probability of the safety valves, which discharged to the drywell, operating in the event that the turbine bypass should fail.

Finally, the safety valves are sized to keep the reactor vessel peak pressure below 1375 psig with no credit taken for the relief valves during the postulated full closure of all MSIV's without direct (valve position switch) scram. Credit is taken for the neutron flux scram, however.

The indirect flux scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full scale pressure recorder.

Note: SAR, Section 11.2.2 -
also: "Dresden 3 Second Reload License Submittal," 9-14-73
also: "Dresden Station Special Report No. 29 Supplement B."

2.2 LIMITING SAFETY SYSTEM SETTING BASES

In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the neutron flux scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus exceeding the pressure safety limit. The pressure scram is available as a backup protection to the direct valve position trip scrams and the high flux scram.

If the high flux scram were to fail, a high pressure scram would occur at 1060 psig. Analyses are performed as described in reference XN-NF-79-71 for each reload to assure that the pressure safety limit is not exceeded.

3.0 LIMITING CONDITION FOR OPERATION

A. In the event a Limiting Condition for Operation cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least hot shutdown within 12 hours and in cold shutdown within the following 24 hours unless corrective measures are completed that satisfy the Limiting Conditions for Operation. Exceptions to these requirements are stated in the individual specifications.

B. When a system, subsystem, train, component, or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operations, provided: (1) its corresponding normal or emergency power source is operable; and (2) all of its redundant system(s), subsystem(s), train(s), component(s), and device(s) in the other division are operable, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least

3.0 LIMITING CONDITION FOR OPERATION (Cont'd.)

hot shutdown within 12 hours, and in at least cold shutdown within the following 24 hours.

C. Specifications 3.0.A and 3.0.B are not applicable in refueling or cold shutdown.

3.0 LIMITING CONDITION FOR OPERATION BASES

3.0.A. This specification delineates the action to be taken for circumstances not directly provided for in the Limiting Condition for Operation statements and whose occurrence would violate the intent of the specification.

3.0.B. This specification delineates what additional conditions must be satisfied to permit operation to continue, consistent with the Limiting Condition for Operation statements for power sources, when a normal or emergency power source is not operable. Power sources are defined as AC Auxiliary Electrical Systems as defined in Section 3.9.A.1, 3.9.A.2, and 3.9.A.3. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the Limiting Condition for Operation Statements associated with individual systems, subsystems, trains, components or devices to be consistent with the Limiting Condition for Operation statements of the associated electrical power source. It allows operation to be governed by the time limits of action statements associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual action statements for each system, subsystem, train, component, or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

3.1 LIMITING CONDITIONS FOR OPERATION

REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

A. Reactor Protection System

1. 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. The system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milliseconds.

2. If during operation, the maximum fraction of limiting power density for fuel fabricated by GE exceeds the fraction of rated power when operating above 25% rated thermal power, either:

4.1 SURVEILLANCE REQUIREMENTS

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. Reactor Protection System

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

2. Daily during reactor power operation above 25% rated thermal power, the core power distribution shall be checked for:

3.1 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

a. The APRM scram and rod block settings shall be reduced to the values given by the equations in Specifications 2.1.A.1 and 2.1.B. This may be accomplished by increasing APRM gains as described therein.

b. The power distribution shall be changed such that the maximum fraction of limiting power density no longer exceeds the fraction of rated power.

For fuel fabricated by ENC, operation of the core shall be limited to ensure the power distribution is consistent with that assumed in the Fuel Design Analysis for overpower conditions.

3. Two RPS electric power monitoring channels for each inservice RPS MG set or alternate source shall be OPERABLE at all times.

4.1 SURVEILLANCE REQUIREMENTS (Cont'd.)

a. Maximum fraction of limiting power density for fuel fabricated by GE (MFLPD) and compared with the fraction of rated power (FRP).

b. For compliance with assumptions of the Fuel Design Analysis of overpower conditions for fuel fabricated by ENC.

3. The RPS power monitoring system instrumentation shall be determined OPERABLE:

a. At least once per 6 months by performing a CHANNEL FUNCTIONAL TEST, and

3.1 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

4. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable channel to OPERABLE

4.1 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. At least once per operating cycle by demonstrating the OPERABILITY of overvoltage, undervoltage, and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic, and output circuit breakers, and verifying the following setpoints:

Surveillance Requirements: Reactor Protection Buses

(1) Overvoltage
126.5V ± 2.5%

Min. 123.3V
Max. 129.6V

(2) Undervoltage
108V ± 2.5%

Min. 105.3V
Max. 110.7V

(3) Underfrequency
56.0 Hz ± 1% of 60 Hz

Min. 55.4 Hz
Max. 56.6 Hz

3.1 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

5. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

4.1 SURVEILLANCE REQUIREMENTS (Cont'd.)

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Minimum Number Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable: Refuel (7)	Startup/Hot Standby	Run	Action*
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
3	High Flux	(LT/E) 120/125 of Full Scale	X	X	X(5)	A
3	Inoperative		X	X	X(5)	A
	APRM					
2	High Flux	Specification 2.1.A.1	X	X(9)	X	A or B
2	Inoperative**		X	X(9)	X	A or B
2	Downscale	(GT/E) 5/125 of Full Scale	X(12)	X(12)	X(13)	A or B
2	High Flux (15% Scram)	Specification 2.1.A.2	X	X	X(14)	A
2	High Reactor Pressure	(LT/E) 1060 psig	X(11)	X	X	A
2	High Drywell Pressure	(LT/E) 2 psig	X(8), X(10)	X(8), (10)	X(10)	A
2	Reactor Low Water Level	(GT/E) 1 inch***	X	X	X	A
2 (Per Bank)	High Water Level in Scram Discharge Volume (Float and dp Switch)	(LT/E) 37.25 inches above bottom of the Instrument Volume	X(2)	X	X	A or D
2	Turbine Condenser Low Vacuum	(GT/E) 23 in. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Steam Line High Radiation	(LT/E) 3 X Normal Full Power Background	X(3)	X(3)	X	A or C
4(6)	Main Steam Line Isolation Valve Closure	(LT/E) 10% Valve Closure	X(3)	X(3)	X	A or C
2	Generator Load Rejection	****	X(4)	X(4)	X(4)	A or C
2	Turbine Stop Valve Closure	(LT/E) 10% Valve Closure	X(4)	X(4)	X(4)	A or C
2	Turbine Control - Loss of Control Oil Pressure	(GT/E) 900 psig	X	X	X	A or C

Notes: (LT/E) = Less than or equal to.
(GT/E) = Greater than or equal to.

(Notes continue on next two pages)

NOTES: (For Table 3.1.1)

1. There shall be two operable or tripped trip systems for each function.

2. Permissible to bypass, with control rod block, for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.

3. Permissible to bypass when reactor pressure less than 600 psig.

4. Permissible to bypass when first stage turbine pressure less than that which corresponds to 45% rated steam flow.

5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any one valve without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a. Mode Switch in Shutdown
   b. Manual Scram
   c. High Flux IRM
   d. Scram Discharge Volume High Level

8. Not required to be operable when primary containment integrity is not required.

9. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

10. May be bypassed when necessary during purging for containment inerting or deinerting.

11. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

12. The APRM downscale trip function is automatically bypassed when the reactor mode switch is in the refuel and startup/hot standby positions.

13. The APRM downscale trip function is automatically bypassed when the IRM instrumentation is operable and not high.

(Cont'd. next page)

NOTES: (For Table 3.1.1 Cont'd.)

14. The APRM 15% scram is bypassed in the run mode.

* If the first column cannot be met for one of the trip systems, that trip system shall be tripped.

If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

a. Initiate insertion of operable rods and complete insertion of all operable rods within 4 hours.

b. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.

c. Reduce turbine load and close main steam line isolation valves within 5 hours.

d. In the refuel mode, when any control rod is withdrawn, suspend all operations involving core alterations and insert all insertable control rods within one hour.

** An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there are less than 56% of the normal complement of LPRM's to an APRM.

*** 1 inch on the water level instrumentation is greater than or equal to 504" above vessel zero (see Bases 3.2).

**** Trips upon actuation of the fast closure solenoid which trips the turbine control valves.

TABLE 4.1.1

SCRAM INSTRUMENTATION FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

Instrument Channel	Group (3)	Functional Test	Minimum Frequency (4)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
* High Flux	C	Trip Channel and Alarm (5)	Before Each Startup (6)
* Inoperative	C	Trip Channel and Alarm	Before Each Startup (6)
APRM			
High Flux	B	Trip Output Relays (5)	Once Each Week
Inoperative	B	Trip Output Relays	Once Each Week
Downscale	B	Trip Output Relays (5)	Once Each Week
High Flux (15% scram)	B	Trip Output Relays	Before Each Startup
High Reactor Pressure	A	Trip Channel and Alarm	(1)
High Drywell Pressure	A	Trip Channel and Alarm	(1)
Reactor Low Water Level (2)	A	Trip Channel and Alarm	(1)
High Water Level in Scram Discharge Volumes (Float and dp Switch)	A	Trip Channel and Alarm (7)	Every 3 Months
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	(1)
Main Steam Line High Radiation (2)	B	Trip Channel and Alarm (5)	Once Each Week
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	(1)
Generator Load Rejection	A	Trip Channel and Alarm	(1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	(1)
Turbine Control - Loss of Control Oil Pressure	A	Trip Channel and Alarm	(1)

Notes: (See next page)

NOTES: (For Table 4.1.1)

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months. 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 Dresden Unit 3.

2. An instrument check shall be performed on low reactor water level once per day and on high steam line radiation once per shift.

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

4. Functional tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

5. This instrumentation is exempted from the Instrument Functional Test Definition (1.0.G). This Instrument Function Test will consist of injecting a simulated electrical signal into the measurement channels.

6. If reactor start-ups occur more frequently than once per week, the functional test need not be performed; i.e., the maximum functional test frequency shall be once per week.

7. The Functional Test of the Scram Discharge Volume float switch shall include actuation of the switch using a water column.

TABLE 4.1.2

SCRAM INSTRUMENTATION CALIBRATIONS

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration Test	Minimum Frequency (2)
*High Flux IRM	C	Comparison to APRM after Heat Balance	Every Shutdown (4)
High Flux APRM			
Output Signal	B	Heat Balance	Once Every 7 Days
Flow Bias	B	Standard Pressure and Voltage Source	Refueling Outage
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Water Level	Every 3 Months
Turbine Condenser Low Vacuum	A	Standard Vacuum Source	Every 3 Months
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 Months
Turbine Control - Loss of Control Oil Pressure	A	Pressure Source	Every 3 Months
High Water Level in Scram Discharge Volume (dp only)	A	Water Level	Once per Refueling Outage

NOTES: (For Table 4.1.2)

1. A description of the three groups is included in the bases of this Specification.

2. Calibration tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made during each refueling outage.

*4. If reactor startups occur more frequently than once per week, the functional test need not be performed; i.e., the maximum functional test frequency shall be once per week.

3.1 LIMITING CONDITION FOR OPERATION 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 primary system.

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

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still 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 reactor protection system is of the dual channel type. (Ref. Section 7.7.1.2 SAR.) 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.

Specifications are provided to ensure the operability of the RPS Bus Electrical Protector Assemblies (EPA's). Each channel from either overvoltage, undervoltage, or under frequency will trip the associated MG set or alternate power source.

This system meets the requirements of the proposed IEEE Standard for Nuclear Power Plant Protection Systems issued September 13, 1966. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) and Intermediate Range Monitor (IRM) channels, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the

3.1 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

APRM's #1 and #3 operate contacts in a one subchannel and APRM's #2 and #3 operate contacts in the other subchannel. APRM's #4, #5 and #6 are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, generator load rejection, and turbine stop valve closure are discussed in Specification 2.3.

Instrumentation (pressure switches) in the drywell are provided to detect a loss of coolant accident and initiate the emergency core cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 2.2. A scram is provided at the same setting as the emergency core cooling system (ECCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent the reactor from going critical following the accident.

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. A part of this system is an individual instrument volume for each of the east and west CRD accumulators. These two volumes and their piping can hold in excess of 90 gallons of water and is the low point in the piping. No credit was taken for these volumes in the design of the discharge piping relative to the amount of water which must be accommodated during a scram. During normal operations, the discharge volumes are empty; however, should either volume fill with water, the water discharged to the piping from the reactor may not be accommodated which could result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been installed in both volumes which will alarm and scram the reactor when the volume remaining in either instrument volume is approximately 41 gallons. For diversity of level sensing methods that will ensure and provide a scram, both differential pressure switches and float switches have been incorporated into the design and logic of the system. The setpoint for the scram signal has been chosen on the basis of providing sufficient volume remaining to accommodate a scram even with 5 gpm leakage per drive into the SDV. As indicated above,

3.1 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or the 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 properly.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient with bypass closure. The condenser low vacuum scram is a backup to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 23" Hg vacuum, stop valve closure occurs at 20" Hg vacuum, and bypass closure at 7" Hg vacuum.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector offgas monitors which cause an isolation of the main condenser offgas line provided the limit specified in Specification 3.8 is exceeded.

The main steam line isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. (Ref. Section 7.7.1.2 SAR.)

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

3.1 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The IRM system provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges. (Ref. Sections 7.4.4.2 and 7.4.4.3 SAR.) A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram functions. (Ref. Section 7.4.3.2 SAR.) Thus, the IRM is required in the "Refuel" and "Start/Hot Standby" modes. In the power range the APRM system provides required protection. (Ref. Section 7.3.5.2 SAR.) Thus, the IRM system is not required in the "Run" mode. The APRM's cover only the power range, the IRM's provide adequate coverage in the start-up and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level, and scram discharge volume high level scrams are required for Startup/Hot Standby and Run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have all scram functions except those listed in Note 8 of Table 3.1.1 operable in the Refuel mode is to assure that shifting to the Refuel mode during reactor power operation does not diminish the need for the reactor protection system.

The turbine condenser low vacuum scram is only required during power operation and must be bypassed to start up the unit. At low power conditions a turbine stop valve closure does not result in a transient which could not be handled safely by other scrams such as the APRM.

The requirement that the IRM's be inserted in the core when the APRM's read 5/125 of full scale assures that there is proper overlap in the neutron monitoring systems and thus, that adequate coverage is provided for all ranges of reactor operation.

4.1 SURVEILLANCE REQUIREMENT 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 for Dresden 3. 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, faulted cables, etc., 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.

Surveillance requirements are provided for the RPS EPA's to demonstrate their operability. The setpoints for overvoltage, undervoltage and under frequency have been chosen based on analysis. (Reference T. Raush letter to H. Denton 02-04-83).

The 13 channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups respecting functional testing. These are:

1. On-Off sensors that provide a scram trip function.

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

3. Devices which only serve a useful function during some restricted mode of operation, such as startup or shutdown, or for which the only practical test is one that can be performed at shutdown.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. Actual history on this class of sensors operating in nuclear power plants shows 4 failures in 472 sensor years, or a failure rate of 0.97×10^{-6}/hr. During design, a goal of 0.99999 probability of success (at the 50% confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function

*Reference (6); See next page.

4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

of the sensor failure rate and the test interval. A three-month test interval was planned for group (A) sensors. This is in keeping with good operating practice, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95% confidence level is proposed. With the (1 out of 2) X (2) logic, this requires that each sensor have an availability of 0.993 at the 95% confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history (See Reference 6). To facilitate the implementation of this technique, Figure 4.1.1 is provided to indicate an appropriate trend in test interval. The procedure is as follows:

1. Like sensors are pooled into one group for the purpose of data acquisition.

2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T (M = nT).

3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1.1.

4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.

5. A test interval of one month will be used initially until a trend is established.

Group (B) devices utilize an analog sensor followed by an amplifier and a bi-stable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks"

Reference 6:
Reliability of Engineered Safety Features as a Function of Testing Frequency, I.M. Jacobs, Nuclear Safety, Vol. 9, No. 4, July-Aug. 1968, pp. 310-312.

4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

midscale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. For purposes of analysis, it is assumed that this rare failure will be detected within two hours.

The bi-stable trip circuit which is a part of the Group (B) devices can sustain unsafe failures which are 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 (Proposed IEEE Standard for Nuclear Power Plant Protection Systems, Section 4.9, September 13, 1966). 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 zero flow signal will be sent to half of the APRM's resulting in a half scram and rod block condition. Thus, if the calibration was performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

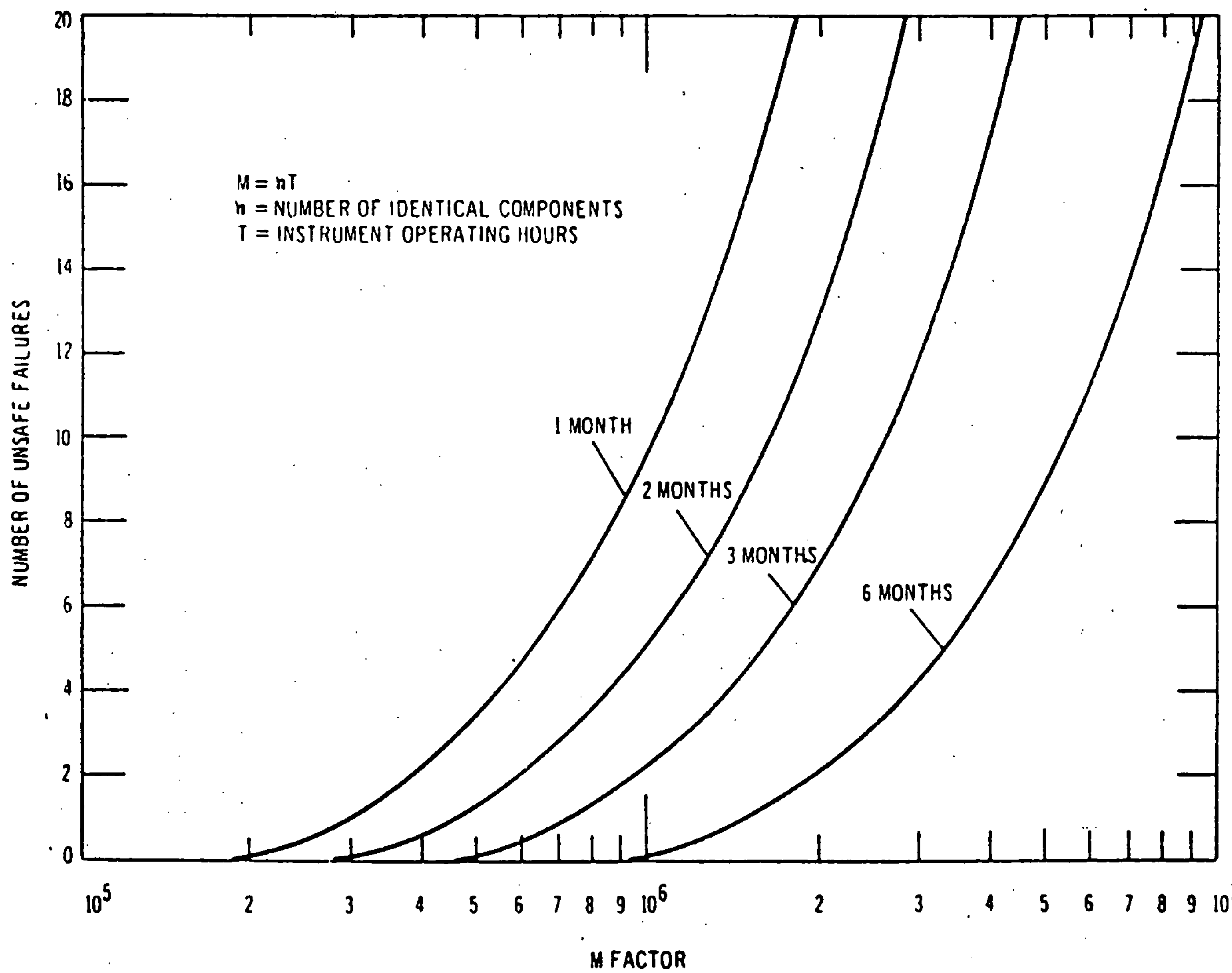


Figure 4.1.1
Graphical Aid in the Selection of an Adequate Interval Between Tests

4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in Commonwealth Edison generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.19/month; i.e., in the period of a month, a drift of .19 would occur and thus provide for adequate margin.

For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.1 and 4.1.2 indicates that six instrument channels have not been included in the latter Table. These are: Mode Switch in Shutdown, Manual Scram, High Water Level in Scram Discharge Volume Float Switches, Main Steam Line Isolation Valve Closure, Generator Load Rejection, and Turbine Stop Valve Closure. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration is not applicable; i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving; e.g., the switches in the scram discharge volume tank. Based on the above, no calibration is required for these six instrument channels.

B. The MFLPD for fuel fabricated by GE shall be checked once per day to determine if the APRM gains or scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations.

4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

Only a small number of control rods are moved daily and thus the peaking factors are not expected to change significantly and thus a daily check of the MFLPD is adequate.

For fuel fabricated by ENC, the power distribution will be checked once per day to ensure consistency with the power distribution assumptions of the fuel design analysis for overpower conditions. During periods of operation beyond these power distribution assumptions, the APRM gains or scram settings may be adjusted to ensure consistency with the fuel design criteria for overpower conditions.

3.2 LIMITING CONDITION FOR OPERATION

PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the plant instrumentation which performs a protective function.

Objective:

To assure the operability of protective instrumentation.

Specification:

A. Primary Containment Isolation Functions

When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table 3.2.1.

B. Core and Containment Cooling Systems - Initiation and Control

The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2.2. This instrumentation must be operable when the system(s) it initiates or controls are required to be operable.

4.2 SURVEILLANCE REQUIREMENT

PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the surveillance requirements of the instrumentation that performs a protective function.

Objective:

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

Specification:

Instrumentation shall be functionally tested and calibrated as indicated in Table 4.2.1.

3.2 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. Control Rod Block Actuation

1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3.

2. The minimum number of operable instrument channels specified in Table 3.2.3 for the Rod Block Monitor may be reduced by one in one of the trip systems for maintenance and/or testing, provided that this condition does not last longer than 24 hours in any 30-day period. In addition, one channel may be bypassed above 30% rated power without a time restriction provided that a limiting control rod pattern does not exist and the remaining RBM channel is operable.

D. Steam Jet-Air Ejector Off Gas System

1. Except as specified in 3.2.D.2. below, both steam-jet air ejector off-gas system radiation monitors shall be operable during reactor power operation. The trip settings for the monitors shall be set

4.2 SURVEILLANCE REQUIREMENT (Cont'd.)

3.2 LIMITING CONDITION FOR OPERATION (Cont'd.)

at a value not to exceed the equivalent of the stack release limit specified in Specification 3.8. The time delay setting for closure of the steam jet-air ejector isolation valves shall not exceed 15 minutes.

2. From and after the date that one of the two steam-jet air ejector off-gas system radiation monitors is made or found to be inoperable, continued reactor power operation is permissible during the next seven days provided the inoperable monitor is tripped in the upscale position.

E. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

1. Except as specified in 3.2.E.2 below, four radiation monitors shall be operable at all times.

2. One of the two radiation monitors in the ventilation duct and one of the two radiation monitors on the refueling floor may be inoperable for 24 hours. If the inoperable monitor is

4.2 SURVEILLANCE REQUIREMENT (Cont'd.)

3.2 LIMITING CONDITION FOR OPERATION (Cont'd.)

not restored to service in this time, the reactor building ventilation system shall be isolated and the standby gas treatment operated until repairs are complete.

3. The radiation monitors shall be set to trip as follows:

   a. ventilation duct -- 11 mr/hr

   b. refueling floor -- 100 mr/hr

4.2 SURVEILLANCE REQUIREMENT (Cont'd.)

TABLE 3.2.1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION FUNCTIONS

Minimum No. of Operable Inst. Channels per Trip System (1)	Instruments	Trip Level Setting	Action (3)
2	Reactor Low Water	(GT/E)144" above top of active fuel*	A
2	Reactor Low Low Water	(GT/E)84" above top of active fuel*	A
2	High drywell pressure	(LT/E)2 psig rated (4), (5)	A
2 (2)	High Flow Main Steam line	(LT/E)120% of rated steam flow	B
2 of 4 in each of 4 sets	High Temperature Main Steam Line Tunnel	(LT/E)200°F	B
2	High Radiation Main Steam Line Tunnel (6)	(LT/E)3 times normal rated power background	B
2	Low Pressure Main Steamline	(GT/E)850 psig	B
	High Flow Isolation Condenser Line		
1	Steamline Side	(LT/E)20 psi diff. on steamline side	C
1	Condensate Return Side	(LT/E)32" water diff. on condensate return side	C
2	High Flow HPCI Steam Line	(LT/E)150" water (7)	D
4	High Temperature HPCI Steam Line Area	(LT/E)200 degrees F	D

Notes: (LT/E) = Less than or equal to.
(GT/E) = Greater than or equal to.
(GT) = Greater than.
(LT) = Less than.

(Notes continue on next page)

NOTES: (For Table 3.2.1)

1. Whenever primary containment integrity is required, there shall be two operable or tripped trip systems for each function, except for low pressure main steamline which only need be available in the RUN position.

2. Per each steamline.

3. Action: If the first column cannot be met for one of the trip systems, that trip system shall be tripped.

   If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

   A. Initiate an orderly shutdown and have reactor in cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have reactor in Hot Standby within 8 hours.

   C. Close isolation valves in isolation condenser system.

   D. Close isolation valves in HPCI subsystems.

4. Need not be operable when primary containment integrity is not required.

5. May be bypassed when necessary during purging for containment inerting and deinerting.

6. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in the primary coolant.

7. Verification of time delay setting between 3 and 9 seconds shall be performed during each refueling outage.

* Top of active fuel is defined as 360" above vessel zero for all water levels used in the LOCA analyses (see Bases 3.2).

TABLE 3.2.2

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Min. No. of Operable Inst. Channels per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	Reactor Low Low Water Level	84" (plus 4", minus 0") above top of active fuel *	1. In conjunction with low reactor pressure initiates core spray and LPCI. 2. In conjunction with high dry-well pressure 120 sec. time delay, and low pressure core cooling interlock initiates auto blowdown. 3. Initiates HPCI and SBGTS. 4. Initiates starting of diesel generators.
2	High Drywell Pressure (2), (3)	(LT/E) 2 psig	1. Initiates core spray, LPCI, HPCI, and SBGTS. 2. In conjunction with low low water level, 120 sec. time delay, and low pressure core cooling interlock initiates auto blowdown. 3. Initiates starting of diesel generators.
1	Reactor Low Pressure	300 psig $\leq$p $\leq$350 psig	1. Permissive for opening core spray and LPCI admission valves. 2. In conjunction with low low reactor water level initiates core spray and LPCI.
	Containment Spray Interlock		
1(4)	2/3 Core Height	(GT/E) 2/3 core height	Prevents inadvertent operation of containment spray during accident conditions.
2(4)	Containment High Pressure	0.5 psig$\leq$p $\leq$1.5 psig	
1	Timer Auto Blowdown	(LT/E) 120 seconds	In conjunction with low low reactor water level, high dry-well pressure, and low pressure core cooling interlock initiates auto blowdown.
2	Low Pressure Core Cooling Pump Discharge Pressure	50 psig$\leq$p $\leq$100 psig	Defers APR actuation pending confirmation of low pressure core cooling system operation.
2/Bus	Under Voltage on Emergency Buses	(GT/E) 3092 volts (equals 3255 less 5% tolerance)	1. Initiates starting of diesel generators. 2. Permissive for starting ECCS pumps. 3. Removes nonessential loads from buses.
2	Sustained High Reactor Pressure	(LT/E) 1070 psig for 15 seconds	Initiates isolation condenser.
2/Bus	Degraded Voltage on 4 KV Emergency Buses	(GT/E) 3708 volts (equals 3784 volts less 2% tolerance) after (LT/E) 5 minutes (plus 5% tolerance) with a 7-second (+ or - 20%) inherent time delay	Initiates alarm and picks up time delay relay. Diesel generator picks up load if degraded voltage not corrected after time delay.

Notes:

(LT) = Less than
(GT) = Greater than
(LT/E) = Less than or equal to
(GT/E) = Greater than or equal to
APR = Automatic Pressure Relief

(Notes continue on next page)

NOTES: (For Table 3.2.2)

1. For all positions of the Reactor Mode Selector Switch (except for the containment interlock) whenever any ECCS subsystem is required to be operable, there shall be two operable or tripped trip systems. If the first column cannot be met for one of the trip systems, that system shall be tripped. If the first column cannot be met for both trip systems, immediately initiate an orderly shutdown to cold conditions.

2. Need not be operable when primary containment integrity is not required.

3. May be bypassed when necessary during purging for containment inerting or deinerting.

4. If an instrument is inoperable, it shall be placed (or simulated) in the tripped condition so that it will not prevent containment spray.

* Top of active fuel is defined as 360" above vessel zero for all water levels used in the LOCA analyses (see Bases 3.2).

Table 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum No. of Operable Inst. Channels Per Trip System (1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias) (7)	(LT/E) $(0.58W_D + 50)$ (FRP/MFLPD) (Note 2)
1	APRM upscale (refuel and Startup/Hot Standby mode)	(LT/E) 12/125 full scale
2	APRM downscale (7)	(GT/E) 3/125 full scale
1	Rod block monitor upscale (flow bias) (7)	(LT/E) (.65W + 45) (Note 2)
1	Rod block monitor downscale (7)	(GT/E) 5/125 full scale
3	IRM downscale (3)	(GT/E) 5/125 full scale
3	IRM upscale	(LT/E) 108/125 full scale
3	IRM detector not fully inserted in the core	
2 (5)	SRM detector not in startup position	(4)
2 (5) (6)	SRM upscale	(LT/E) 10^5 counts/second
1	Scram discharge volume water level - high	(LT/E) 20 inches above the bottom of the instrument volume

Notes: (LT/E) = Less than or equal to
(GT/E) = Greater than or equal to
(Notes continue on next page)

NOTES: (For Table 3.2.3)

1. For the Startup/Hot Standby and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function, except the SRM rod blocks, IRM upscale, IRM downscale and IRM detector not fully inserted in the core need not be operable in the "Run" position and APRM downscale, APRM upscale (flow bias), and RBM downscale need not be operable in the Startup/Hot Standby mode. The RBM upscale need not be operable at less than 30% rated thermal power. One channel may be bypassed above 30% rated thermal power provided that a limiting control rod pattern does not exist. For systems with more than one channel per trip system, if the first column cannot be met for both trip systems, the systems shall be tripped. For the Scram Discharge Volume water level high rod block, there is one instrument per bank.

2. W_D percent of drive flow required to produce a rated core flow of 98 Mlb/m. MFLPD = highest value of FLPD for G.E. fuel.

3. IRM downscale may be bypassed when it is on its lowest range.

4. This function may be bypassed when the count rate is greater than or equal to 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

7. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
ECCS INSTRUMENTATION			
1. Reactor Low-Low Water Level	(1)	Once/3 Months	Once/Day
2. Drywell High Pressure	(1)	Once/3 Months	None
3. Reactor Low Pressure	(1)	Once/3 Months	None
4. Containment Spray Interlock			
a. 2/3 Core Height	(1)	Once/3 Months	None
b. Containment High Pressure	(1)	Once/3 Months	None
5. Low Pressure Core Cooling Pump Discharge	(1)	Once/3 Months	None
6. Undervoltage Emergency Bus	Refueling Outage	Refueling Outage	Once/3 months
7. Sustained High Reactor Pressure	(1)	Once/3 Months	None
8. Degraded Voltage Emergency Bus	Refueling Outage (10)	Refueling Outage	Monthly
ROD BLOCKS			
1. APRM Downscale	(1) (3)	Once/3 Months	None
2. APRM Flow Variable	(1) (3)	Refueling Outage	None
3. APRM Upscale (Startup/Hot Standby)	(2) (3)	(2) (3)	(2)
4. IRM Upscale	(2) (3)	(2) (3)	(2)
5. IRM Downscale	(2) (3)	(2) (3)	(2)
6. IRM detector not fully inserted in the core	(2)	N/A	None
7. RBM Upscale	(1) (3)	Refueling Outage	None
8. RBM Downscale	(1) (3)	Once/3 Months	None
9. SRM Upscale	(2) (3)	(2) (3)	(2)
10. SRM Detector Not in Startup Position	(2) (3)	(2) (3)	(2)
11. Scram Instr. Vol. Level - High	Once/3 Months (9)	None	None
MAIN STEAM LINE ISOLATION			
1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	None
2. Steam Line High Flow	(1)	Once/3 Months	Once/Day
3. Steam Line Low Pressure	(1)	Once/3 Months	None
4. Steam Line High Radiation	(1) (3)	Once/3 Months (4)	Once/Day
ISOLATION CONDENSER ISOLATION			
1. Steam Line High Flow	(1)	Once/3 Months	None
2. Condensate Line High Flow	(1)	Once/3 Months	None
HPCI ISOLATION			
1. Steam Line High Flow	(1) (11)	Once/3 Months (11)	None
2. Steam Line Area High Temperature	Refueling Outage	Refueling Outage	None
3. Low Reactor Pressure	(1)	Once/3 Months	None
REACTOR BUILDING VENTILATION SYSTEM VIOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION			
1. Ventilation Exhaust Duct Radiation Monitors	(1)	Once/3 Months	Once/Day
2. Refueling Floor Radiation Monitors	(1)	Once/3 Months	Once/Day
STEAM JET-AIR EJECTOR OFF-GAS ISOLATION			
1. Radiation Monitors	(1) (3)	Once/3 Months (4)	Once/Day
CONTAINMENT MONITORING			
1. Pressure			
a. -5 in. Hg to +5 psig Indicator	None	Once/3 Months	Once/Day
b. 0 to 75 psig Indicator	None	Once/3 Months	None
2. Temperature	None	Refueling Outage	Once/Day
3. Drywell-Torus Differential Pressure (5) (6) (0-3 psid)	None	Once/6 Months (two channels operable) Once/Month (one channel operable)	None
4. Torus Water Level (5) (6)	None	Once/6 Months	
a. +/-25" Wide Range Indicator			
b. 18" Sight Glass			
SAFETY/RELIEF VALVE MONITORING			
1. Safety/Relief Valve Position Indicator (Acoustic Monitor) (8)	(7)	None	Once Per 31 Days
2. Safety/Relief Valve Position Indicator (Temperature monitor) (8)	None	Once every 18 months	Once Per 31 Days
3. Safety Valve Position Indicator (Acoustic Monitor) (8)	(7)	None	Once Per 31 Days
4. Safety Valve Position Indicator (Temperature Monitor) (8)	None	Once every 18 months	Once Per 31 Days

Notes: (See next two pages)

NOTES: (For Table 4.2.1))

1. Initially once per month until exposure hours (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. 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 Dresden Unit 2.

2. Function test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. See Note 4.

4. These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed during each refueling outage.

5. A minimum of two channels is required.

6. From and after the date that one of these parameters (. . . either drywell-torus differential pressure or torus water level indication) is reduced to one indication, continued operation is not permissible beyond thirty days, unless such instrumentation is sooner made operable. In the event that all indications of these parameters (. . . either drywell-torus differential pressure or torus water level) is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty four hours.

7. Functional tests will be conducted before startup at the end of each refueling outage or after maintenance is performed on a particular Safety/Relief Valve.

8. If the number of position indicators is reduced to one indication on one or more valves, continued operation is permissible; however, if the reactor is in a shutdown condition for more than seventy-two hours, it may not be started up until all position indication is restored. In the event that all position indication is lost on one or more valves and such indication cannot be restored in thirty days, an orderly shutdown shall be initiated, and the reactor shall be depressurized to less than 90 psig in 24 hours.

(Cont'd. next page)

NOTES: (For Table 4.2.1) (Cont'd.)

9. The Functional Test of the Scram Discharge Volume float switch shall include actuation of the switch using a water column.

10. Functional test shall include verification of the second level undervoltage (degraded voltage) timer bypass and shall verify operation of the degraded voltage 5-minute timer and inherent 7-second timer.

11. Verification of time delay setting between 3 and 9 seconds shall be performed during each refueling outage.

3.2 LIMITING CONDITION FOR OPERATION BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminates operator errors before they result in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are: 1) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and 2) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiates or controls core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low-reactor water level instrumentation is set to trip at greater than 8 inches on the level instrument (top of active fuel

3.2 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

is defined to be 360 inches above vessel zero) and after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above top of active fuel. Retrofit 8 X 8 fuel has an active fuel length 1.34 inches longer than earlier fuel design. However, present trip setpoints were used in the LOCA analyses.

This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR Section 7.7.2). For a trip setting of 504 inches above vessel zero (144 inches above top of active fuel) and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break; the setting is therefore adequate.

The low low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, - 59 inches is 84 inches above the top of active fuel). This trip initiates closure of Group I primary containment isolation valves (Ref. Section 7.7.2.2 SAR), and also activates the ECC subsystems, starts the emergency diesel generator and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spacious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria.

The high drywell pressure instrumentation is a backup to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 Isolation valves. For the breaks discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 Isolation valves include the drywell vent, purge, and sump Isolation valves. High drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions and only the valves in Group 2 are required to close. The low low water level instrumentation initiates protection for the full spectrum of loss of coolant accidents and causes a trip of Group 1 primary system isolation valves.

3.2 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst case accident, main steamline break outside the drywell, this trip setting of 120% of rated steam flow in conjunction with the flow limiters and main steamline valve closure, limit the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. (Ref. Sections 14.2.3.9 and 14.2.3.10 SAR)

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a back-up to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

High radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times normal background, and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. (Ref. Section 14.2.1.7 SAR). The performance of the process radiation monitoring system relative to detecting fuel leakage shall be evaluated during the first five years of operation. The conclusions of this evaluation will be reported to the Nuclear Regulatory Commission.

Pressure instrumentation is provided which trips when main steamline pressure drops below 850 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "Refuel" and "Startup/Hot Standby" mode this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 850 psig inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500°F; thus, there are no fission products available for release other than those in the reactor water (Ref. Section 11.2.3 SAR).

3.2 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Two sensors on the isolation condenser supply and return lines are provided to detect the failure of isolation condenser line and actuate isolation action. The sensors on the supply and return sides are arranged in a 1 out of 2 logic and, to meet the single failure criteria, all sensors and instrumentation are required to be operable. The trip settings of 20 psig and 32" of water and valve closure time are such as to prevent uncovering the core or exceeding site limits. The sensors will actuate due to high flow in either direction.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI piping. Tripping of this instrumentation results in actuation of HPCI isolation valves, i.e., Group 4 valves. Tripping logic for this function is the same as that for the isolation condenser and thus all sensors are required to be operable to meet the single failure criteria. The trip settings of 200°F and 300% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not go below the MCPR fuel cladding integrity safety limit. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, 8 IRM's, or 4 SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria are met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. This time period is only approximately 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the refuel and startup/hot standby modes, the APRM rod block function is set at 12% of rated power. This control rod block

3.2 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

provides the same type of protection in the Refuel and Startup/Hot Standby mode as the APRM flow biased rod block does in the run mode; i.e., prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worse case single control rod withdrawal error is analyzed for each reload to assure that with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the MCPR fuel cladding integrity safety limit.

Below 30 percent power, the worst case withdrawal of a single control rod without rod block action will not violate the MCPR fuel cladding integrity safety limit. Thus, the RBM rod block function is not required below this power level.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity safety limit.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale trips are set at 5/125 of full scale.

The rod block which occurs when the IRM detectors are not fully inserted in the core for the refuel and startup/hot standby position of the mode switch has been provided to assure that these detectors are in the core during reactor startup. This, therefore, assures that these instruments are in proper position to provide protection during reactor startup. The IRM's primarily provide protection against local reactivity effects in the source and intermediate neutron range.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a back-up to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip

3.2 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

settings given in the specification are adequate to assure the above criteria are met. (Ref. Section 6.2.6.3 SAR.) The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a fifteen minute delay before the air ejector off-gas isolation valve is closed. This delay is accounted for by the 30-minute holdup time of the off-gas before it is released to the stack.

Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct and on the refueling floor. The trip logic is a 1 out of 2 for each set and each set can initiate a trip independent of the other set. Any upscale trip will cause the desired action. Trip settings of 11 mr/hr for the monitors in the ventilation duct are based upon initiating normal ventilation isolation and standby gas treatment system operation to limit the dose rate at the nearest site boundary to less than the dose rate allowed by 10CFR20. Trip settings of 100 mr/hr for the monitors on the refueling floor 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 stack but that all the activity is processed by the standby gas treatment system.

4.2 SURVEILLANCE REQUIREMENT BASES

The instrumentation listed in Table 4.2.1 will be functionally tested and calibrated at regularly scheduled intervals. Although this instrumentation is not generally considered to be as important to plant safety as the Reactor Protection System, the same design reliability goal of 0.99999 is generally applied for all applications of (1 out of 2) X (2) logic. Therefore, on-off sensors are tested once/3 months, and bi-stable trips associated with analog sensors and amplifiers are tested once/week.

Those instruments which, when tripped, result in a rod block have their contacts arranged in a 1 out of n logic, and all are capable of being bypassed. For such a tripping arrangement with bypass capability provided, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (See Note 7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = (2t/r)^{1/2}$$

Where:

i = optimum interval between tests
t = the time the trip contacts are disabled from performing their function while the test is in progress
r = the expected failure rate of the relays

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 minutes to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10E-6 failures per hour. Using this data and the above operation, the optimum test interval is:

$$i = \left(\frac{2(0.5)}{10^6}\right)^{1/2} = 1 \times 10^3 \text{ hours}$$

$$= \text{approximately 40 days}$$

For additional margin a test interval of once per month will be used initially.

Note:
(7) UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, Benjamin Epstein, Albert Shiff, July 16, 1968, page 10, Equation (24), Lawrence Radiation Laboratory.

4.2 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The sensors and electronic apparatus have not been included here as these are analog devices with readouts in the control room and the sensors and electronic apparatus can be checked by comparison with other like instruments. The checks which are made on a daily basis are adequate to assure operability of the sensors and electronic apparatus, and the test interval given above provides for optimum testing of the relay circuits.

The above calculated test interval optimizes each individual channel, considering it to be independent of all others. As an example, assume that there are two channels with an individual technician assigned to each. Each technician tests his channel at the optimum frequency, but the two technicians are not allowed to communicate so that one can advise the other that his channel is under test. Under these conditions, it is possible for both channels to be under test simultaneously. Now, assume that the technicians are required to communicate and that two channels are never tested at the same time.

Forbidding simultaneous testing improves the availability of the system over that which would be achieved by testing each channel independently. These one out of n trip systems will be tested one at a time in order to take advantage of this inherent improvement in availability.

Optimizing each channel independently may not truly optimize the system considering the overall rules of system operation. However, true system optimization is a complex problem. The optimums are broad, not sharp, and optimizing the individual channels is generally adequate for the system.

The formula given above minimizes the unavailability of a single channel which must be bypassed during testing. The minimization of the unavailability is illustrated by curve No. 1 of Figure 4.2.2 which assumes that a channel has a failure rate of 0.1×10^{-6}/hour and that 0.5 hours is required to test it. The unavailability is a minimum at a test interval i, of 3.16×10^{3} hours.

If two similar channels are used in a 1 out of 2 configuration, the test interval for minimum unavailability changes as a function of the rules for testing. The simplest case is to test each one independent of the other. In this case, there is assumed to be a finite probability that both may be bypassed at one time. This case is shown by Curve No. 2. Note that the unavailability is lower as expected for a redundant system and the minimum occurs at

4.2 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

the same test interval. Thus, if the two channels are tested independently, the equation above yields the test interval for minimum unavailability.

A more usual case is that the testing is not done independently. If both channels are bypassed and tested at the same time, the result is shown in Curve No. 3. Note that the minimum occurs at about 40,000 hours, much longer than for cases 1 and 2. Also, the minimum is not nearly as low as Case 2 which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed, tested and restored, and then immediately following the second channel be bypassed, tested, and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1 out of n system may be treated the same as a single channel in terms of choosing a test interval; and

2. More than one channel should not be bypassed for testing at any one time.

The radiation monitors in the ventilation duct and on the refueling floor which initiate building isolation and standby gas treatment operation are arranged in two 1 out of 2 logic systems. The bases given above for the rod blocks applies here also and were used to arrive at the functional testing frequency.

Based on experience at Dresden Unit 1 with instruments of similar design, a testing interval of once every three months has been found to be adequate.

The automatic pressure relief instrumentation can be considered to be a 1 out of 2 logic system and the discussion above applies also.

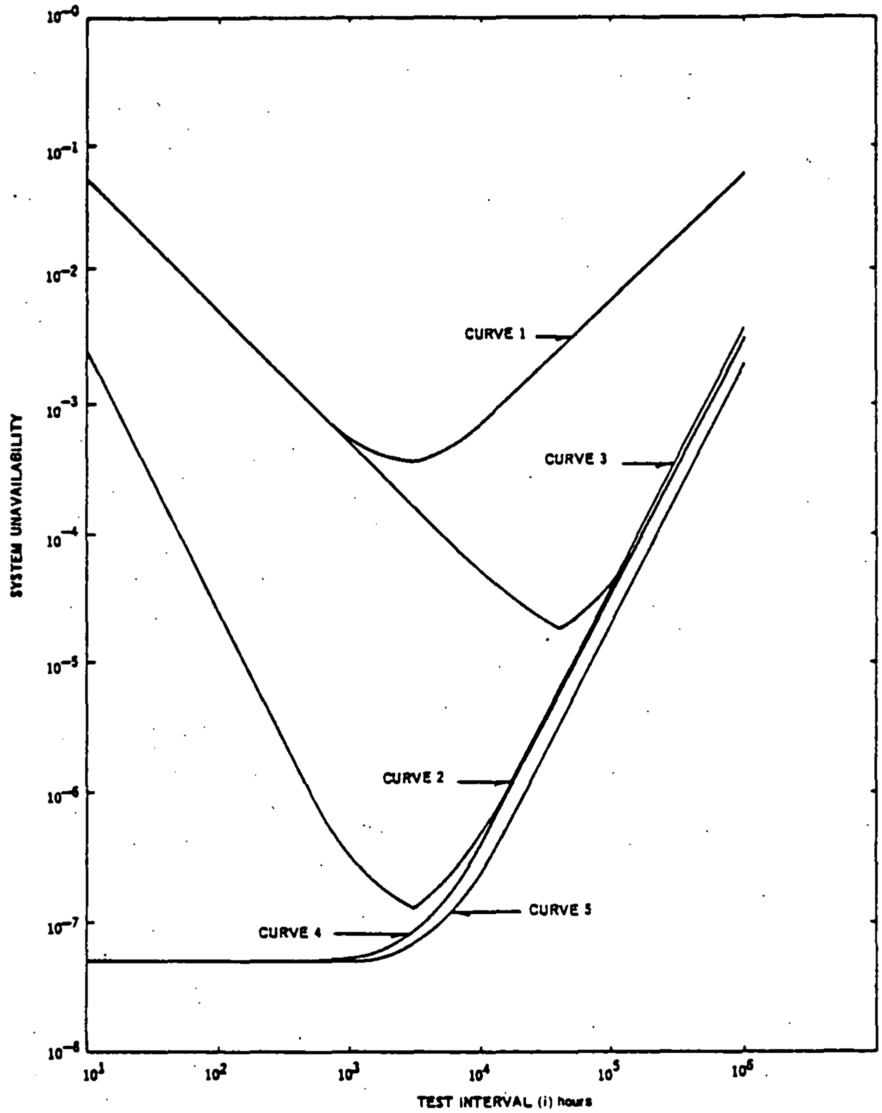


Figure 4.2.2.

TEST INTERVAL VS. SYSTEM UNAVAILABILITY

3.3 LIMITING CONDITION FOR OPERATION

REACTIVITY CONTROL

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

   The core loading shall be limited to that which can be made subcritical in the most reactive condition during the operating cycle with the strongest operable control rod in its full-out position and all other operable rods fully inserted.

4.3 SURVEILLANCE REQUIREMENT

REACTIVITY CONTROL

Applicability:

Applies to the surveillance requirements of the control rod system.

Objective:

To verify the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

   Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.25 percent delta k that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. Reactivity margin - inoperable control rods

   a. Control rod drives 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 valves for inoperable control rods shall be disarmed electrically and the

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

2. Reactivity margin - inoperable control rods

   Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week. This test shall be performed at least once per 24 hours 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.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

control rods shall be in such positions that Specification 3.3.A.1 is met.

c. Control rod drives which are fully inserted and electrically disarmed shall not be considered inoperable.

d. Control rods with scram times greater than those permitted by Specification 3.3.C are inoperable, but if they can be moved with control rod drive pressure, they need not be disarmed electrically if Specification 3.3.A.1 is met for each position of these rods.

e. During reactor power operation, the number of inoperable control rods shall not exceed eight.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

B. Control Rods

1. All control rods shall be coupled to their drive mechanisms when the mode switch is in "Startup" or "Run". With a control rod not coupled to its associated drive mechanism, operation may continue provided:
   a. Below 20% power, the rod shall be declared inoperable, full inserted, and

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

3. The scram discharge volume vent and drain valves shall be verified open at least once per 31 days. These valves may be closed intermittently for testing under administrative control and at least once per 92 days, each valve shall be cycled through at least one complete cycle of full travel. At least once each Refueling Outage, the scram discharge volume vent and drain valves will be demonstrated to:
   a. Close within 30 seconds after receipt of a signal for control rods to scram, and
   b. Open when the scram signal is reset.

B. Control Rods

1. Coupling Integrity
   a. The coupling integrity of each control rod shall be demonstrated by

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

the directional control valves electrically disarmed until recoupling can be attempted at all-rods-in or at power levels above 20 percent power.

b. Above 20% power, recoupling is being attempted

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

withdrawing each control rod to the fully withdrawn position and verifying that the rod does not go to the overtravel position;

i. Prior to reactor criticality after completing alteration of the reactor core,

ii. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

iii. For specifically affected invidiaual control rods following maintenance on or modification to the control rod or rod drive system which could affect the rod drive coupling integrity.

b. Normal operating practice is to observe the

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

in accordance with an established procedure or the rod shall be declared inoperable, fully inserted and the directional control valves electrically disarmed.

2. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

expected response of the nuclear instrumentation to verify that the control rod is following its drive each time that control rod is withdrawn. For control rod drives that have experienced uncoupling and no response is discernable on the nuclear instrumentation, the response should be verified when the reactor is operating at power levels above 20 percent.

2. The control rod drive drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. a. Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would be such that the rod drop accident design limit of 280 cal/gm is not exceeded.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

3. a. To consider the rod worth minimizer operable, the following steps must be performed:

   i. The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct.

   ii. The rod worth minimizer computer on-line diagnostic test shall be successfully completed.

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

   iv. The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

b. Whenever the reactor is in the startup or run mode below 20% rated thermal power, the Rod Worth Minimizer shall be operable. A second operator or qualified technical person may be used as a substitute for an inoperable Rod Worth Minimizer which fails after withdrawal of at least 12 control rods to the fully withdrawn position. The Rod Worth Minimizer may also be bypassed for low power physics testing to demonstrate the shutdown margin requirements of specifications 3.3.A.1 if a nuclear engineer is present and verifies the step-by-step rod movements of the test procedure.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

b. If the rod worth minimizer is inoperable while the reactor is in the startup or run mode below 20% rated thermal power and a second independent operator or engineer is being used, he shall verify that all rod positions are correct prior to commencing withdrawal of each rod group.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

5. During operating with limiting control rod patterns, as determined by the nuclear engineer, 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 the MCPR will remain above the MCPR fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of any single operable control rod.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

4. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels have been 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) and daily thereafter.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization 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	0.900
50	2.00
90	3.50

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	3.800

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

C. Scram Insertion Times

1. After each refueling outage, prior to operation greater than 30 percent of rated thermal power, all control rods shall be subject to scram-time tests from the fully withdrawn position with reactor pressure above 800 psig. If the control rods are tested individually, their hydraulic control units shall be isolated from the control rod drive pumps.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator,
2. Directional control valve electrically disarmed while in a non-fully inserted position.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

2. At 16 week intervals, at least 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% or more of the control rod drives have been tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. Following completion of each set of scram testing as described above, the results will be compared against the average scram speed distribution used in the transient analysis to verify the applicability of the current MCPR Operating Limit. Refer to Specification 3.5.K.

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. Scram insertion greater than maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator and the rod block associated with that inoperable accumulator may be bypassed.

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% delta K. If this limit is exceeded, the reactor will be shutdown until the cause has been determined and corrective actions have been taken if such actions are appropriate. In accordance with Specification 6.6, the NRC shall be notified of this reportable occurrence within 24 hours.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

E. Reactivity Anomalies

During the startup test program and startups 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 equivalent full power month.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

G. Economic Generation Control System

Operation of the unit with the Economic Generation Control system with automatic flow control shall be permissible only in the range of 65-100% of rated core flow, with reactor power above 20%.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

F. (N/A)

G. Automatic Generation Control System

Weekly, the range set into the Economic Generation Control System shall be recorded.

3.3 LIMITING CONDITION FOR OPERATION BASES

A. Reactivity Limitations

1. Reactivity margin--core loading

The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. The generalized form is that the reactivity of the core loading will be limited so the core can be made sub-critical by at least R + 0.25% delta k in the most reactive condition during the operating cycle, with the strongest control rod fully withdrawn and all others fully inserted. The value of R in % delta k is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check; i.e., the initial loading. R must be a positive quantity or zero. A core which contains temporary control or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum and then decreases thereafter. See Figure 3.3.2 of the SAR for such a curve.

The value of R is the difference between the calculated core reactivity at the beginning of the operating cycle and the calculated value of core reactivity any time later in the cycle where it would be greater than at the beginning. The value of R shall include the potential shutdown margin loss assuming full B_4C setting in all inverted poison tubes present in the core. A new value of R must be determined for each fuel cycle.

The 0.25% delta k in the expression R + 0.25% delta k is provided as a finite, demonstrable, sub-criticality margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to insert at least R + 0.25% delta k in reactivity. Observation of sub-criticality in this condition assures sub-criticality with not only the strongest rod fully withdrawn but at least a R + 0.25% delta k margin beyond this.

NOTE: This change issued by letter dated 08/27/75 which noted that 0.02% delta k should be included in the value.

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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 available to the operator. The number of rods permitted to be inoperable could be many more than the eight allowed by the Specification, particularly late in the operation cycle; however, the occurrence of more than eight could be indicative of a generic control rod drive problem and the reactor will be shutdown. Also, 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 BWR's. 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.

* To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

3. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume. This ensures that water accumulation does not occur which would cause an early termination of control rod movement during a full core scram. These specifications provide for the periodic verification that the valves are open and for testing of these valves under reactor scram conditions during each Refueling Outage.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in Reference XN-NF-80-19, Vol. 1, 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 would provide cause for suspecting a rod to be uncoupled and stuck. Restricting recoupling verifications to power levels above 20% provides assurance that a rod drop during a recoupling verification would not result in a rod drop accident.

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 Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. 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 sub-critical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod sequences which are

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

withdrawn could not be worth enough to cause the rod drop accident design limit of 280 cal/gm to be exceeded if they were to drop out of the core in the manner defined for the Rod Drop Accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM or a second qualified station employee. These sequences are developed to limit reactivity worths of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content, 425 cal/gm, at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference XN-NF-80-19, Volume 1.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2 and 14.2.1.4 of the Safety Analysis Report. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident.

Parametric Control Rod Drop Accident analyses have shown that for wide ranges of key reactor parameters (which envelope the operating ranges of these variables), the fuel enthalpy rise during a postulated control rod drop accident remains considerably lower than the 280 cal/gm limit. For each operating cycle, cycle-specific parameters such as maximum control rod worth, Doppler coefficient effective delayed neutron fraction and maximum four-bundle local peaking factor are compared with the results of the parametric analyses to determine the peak fuel rod enthalpy rise. This value is then compared against the Technical Specification limit of 280 cal/gm to demonstrate compliance for each operating cycle. If cycle specific values of the above parameters are outside the range assumed in the parametric analyses, an extension of the analysis or a cycle specific analysis may be required. Conservatism present in the analysis, results of the parametric studies, and a detailed description of the methodology for performing the Control Rod Drop Accident analysis are provided in reference XN-NF-80-19, Volume 1 (Supplements 1 and 2).

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The Rod Worth Minimizer provides 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. (Ref. Section 7.9 SAR.) It serves as a backup to procedural control of control rod worth. In the event that the Rod Worth Minimizer is out of service, when required, a licensed operator or other qualified technical employee can manually fulfill the control rod pattern conformance functions of the Rod Worth Minimizer. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 20% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is required to be operating during a startup for the withdrawal of a significant number of control rods for any startup after June 1, 1974.

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. This is needed for knowledgeable and efficient reactor startup at low neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement 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 from 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.

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

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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 rods according to a written sequence. 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. Amendments 17/18 and 19/20 present the results of an evaluation of a rod block monitor failure. These amendments show that during reactor operation with certain limiting control rod pattern, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPRS less than the MCPR fuel cladding integrity safety limit. 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 Nuclear 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.

C. Scram Insertion Times

The performance of the control rod insertion system is analyzed to verify the system's ability to bring the reactor subcritical at a rate fast enough to prevent violation of the MCPR Fuel Cladding Integrity Safety Limit and thereby avoid fuel damage. The analyses demonstrate that if the reactor is operated within the limitations set in Specification 3.5.K. the negative reactivity insertion rates associated with the observed scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR safety limit.

In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

analyses, and is also included in the allowable scram insertion times specified in Specification 3.3.C. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Fifty percent of the control rod drives in the reactor are tested every sixteen weeks to verify adequate performance. Observed plant data were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly. If an individual test or group of tests should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken (as required by Specification 3.5.K) unless it can be shown that the number of individual drives falling outside the statistical population defining the nominal performance is less than the allowable number of inoperable control rod drives. If the number of statistically aberrant drives falls within this limitation, operation will be allowed to continue without redetermination of thermal margin requirements provided the identified aberrant drives are fully inserted into the core and deenergized in the manner of an inoperable rod drive.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no reassessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 second is greater than 0.999 for a normal distribution.

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

D. Control Rod Accumulators

The basis for this specification was not described in the SAR and, therefore, is presented in its entirety. Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a k_{eff} less than 1.0 -- other repeating rod sequences with more rods withdrawn resulted in k_{eff} greater than 1.0. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle excess operating 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 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% delta k. Deviations in core reactivity greater than 1% delta k 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.

F. (N/A)

G. Economic Generation Control System

Operation of the facility with the Economic Generation Control System with automatic flow control is limited to the range of

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

65-100% of rated core flow. In this flow range and with reactor power above 20% the reactor can safely tolerate a rate of change of load of 8 MW(e)/sec. (Reference FSAR Amendment 9-Unit 2, 10-Unit 3). Limits within the Economic Generation Control System and Reactor Flow Control System preclude rates of change greater than approximately 4 MWe/sec.

When the Economic Generation Control System is in operation, this fact will be indicated on the main control room console. The results of initial testing will be provided to the NRC at the onset of routine operation with the Economic Generation Control System.

4.3 SURVEILLANCE REQUIREMENT BASES

None

3.4 LIMITING CONDITION FOR OPERATION

STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the standby liquid control system.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. Normal Operation

During periods when fuel is in the reactor the standby liquid control system shall be operable except when the reactor is in the Cold Shutdown Condition and all control rods are fully inserted and Specification 3.3.A is met or as specified in 3.4.B below.

4.4 SURVEILLANCE REQUIREMENT

STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirements for the standby liquid control system.

Objective:

To verify the operability of the standby liquid control system.

Specification:

A. Normal Operation

The operability of the standby liquid control system shall be verified by performance of the following tests:

1. At least once per month -

   Demineralized water shall be recycled to the test tank. Pump minimum flow rate of 39 gpm shall be verified against a system head of 1275 psig.

2. At least once during each operating cycle

   a. Actuate one of the two standby liquid control systems using the normal actuation switch and pump demineralized water into the reactor

3.4 LIMITING CONDITION FOR OPERATION (Cont'd.)

B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A shall be considered fulfilled, and continued operation permitted provided that the component is returned to an operable condition within 7 days.

4.4 SURVEILLANCE REQUIREMENT (Cont'd.)

vessel. Pump minimum flow rate shall be verified against a previous test at the same reactor vessel pressure. The replacement charges will be selected from a batch from which at least one charge has been successfully test fired and which will not exceed five years life when their use is terminated. Both systems shall be tested and inspected, including each explosive actuated valve, in the course of two operating cycles.

b. Test that the setting of the system pressure relief valves is between 1400 and 1490 psig.

B. Surveillance with Inoperable Components

When a component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter.

3.4 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. The liquid poison tank shall contain a boron bearing solution that satisfies the volume-concentration requirements of Figure 3.4.1 and at all times when the standby liquid control system is required to be operable and the solution temperature including that in the pump suction piping shall not be less than the temperature presented in Figure 3.4.2.

D. If specification 3.4.A through C are not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.4 SURVEILLANCE REQUIREMENT (Cont'd.)

C. The availability of the proper boron bearing solution shall be verified by performance of the following tests:

1. At least once per month - Boron concentration shall be determined. In addition, the boron concentration shall be determined any time water or boron are added or if the solution temperature drops below the limits specified by Figure 3.4.2.

2. At least once per day - Solution volume shall be checked.

3. At least once per day - The solution temperature shall be checked.

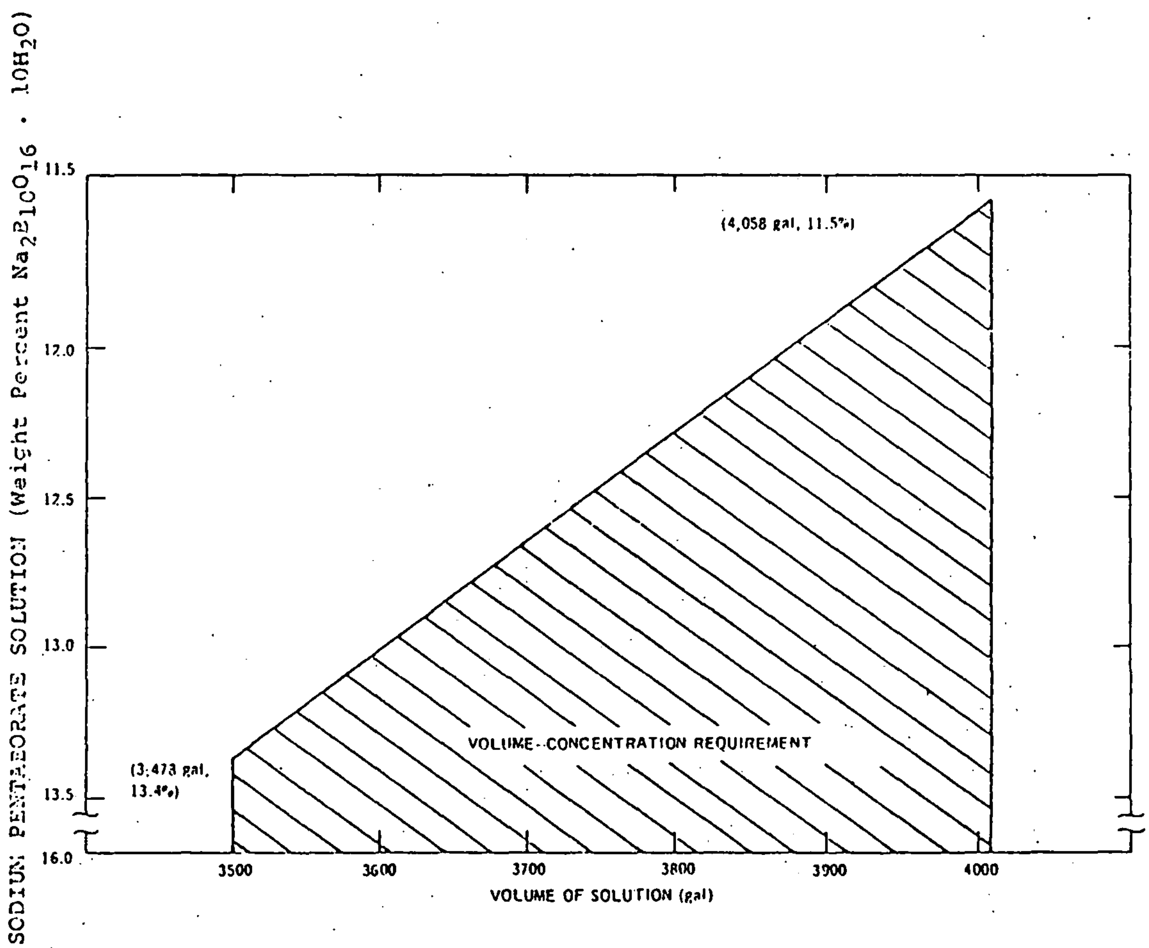


Figure 3.4.1 Standby Liquid Control Solution Requirements

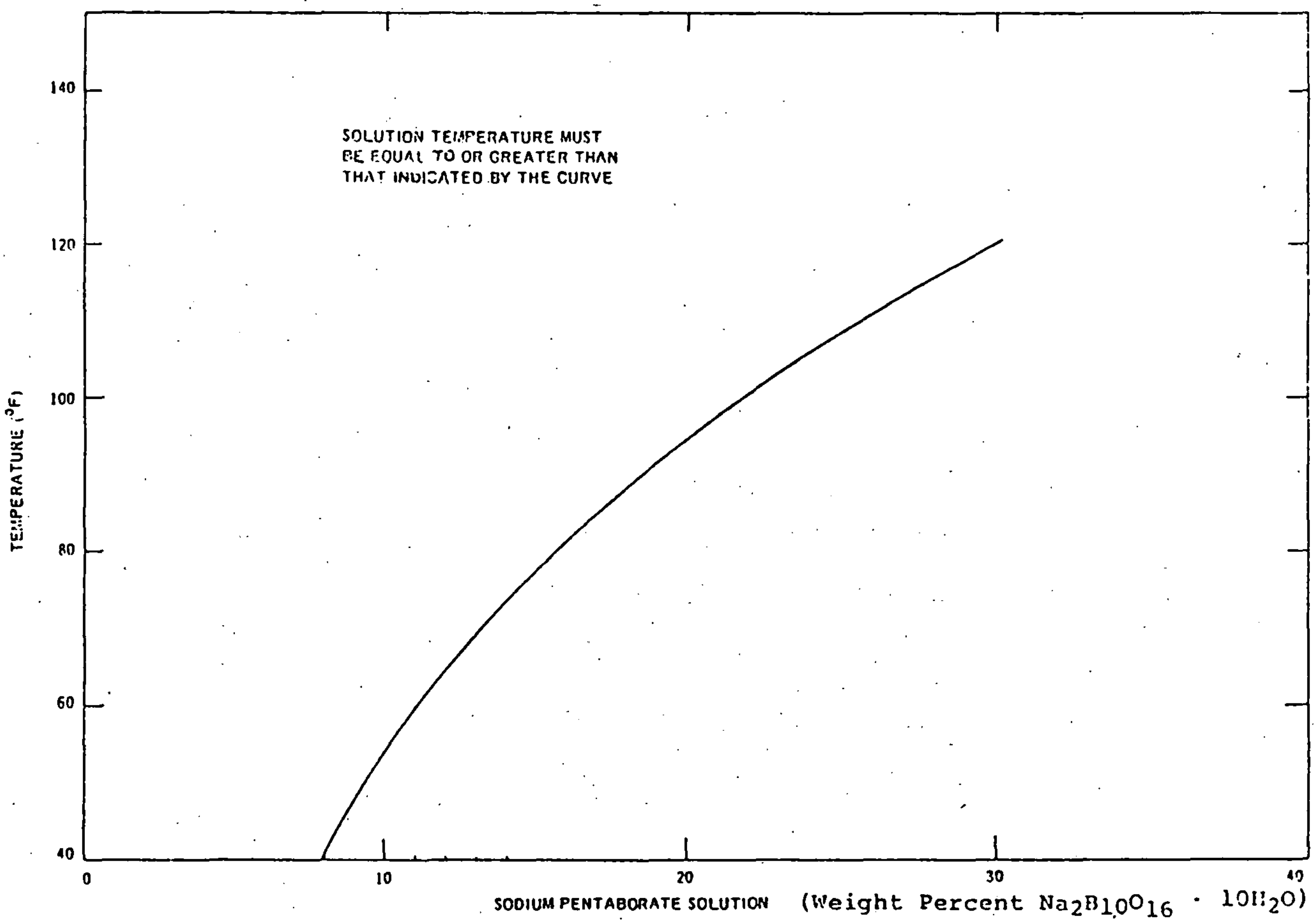


Figure 3.4.2 Sodium Pentaborate Solution Temperature Requirements

3.4 LIMITING CONDITION FOR OPERATION BASES

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, neutron-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of no less than 600 ppm of boron in the reactor core in less than 100 minutes. 600 ppm boron concentration in the reactor core is required to bring the reactor from full power to a 3% delta k or more subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional margin (25%) for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3478 gallons of solution having a 13.4% sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (100 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required pumping rate of 39 gallons per minute, the maximum storage volume of the boron solution is established as 4,059 gallons (158 gallons are contained below the pump suction and, therefore, cannot be inserted).

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Experience with pump operability indicates that monthly testing is adequate to detect if failures have occurred.

Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of less than once during each operating cycle unnecessary. A test of one installed explosive charge is made at least once during each operating cycle to assure that the charges have not deteriorated, the actuation circuit is functioning properly, the valve functions properly, and no flow blockages exist. The replacement charge will be selected from a batch for which there has been a successful test firing. Recommendations of the vendor shall be followed in maintaining a five-year life of the explosive charges. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

3.4 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The relief valves in the standby liquid control system protect the system piping and positive displacement pumps which are nominally designed for 1500 psig protection from over-pressure. The pressure relief valves discharge back to the standby liquid control solution tank.

B. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the remaining system will perform its intended function and that the reliability of the system is good is obtained by demonstrating operation of the pump in the operable circuit at least once daily.

C. The solution saturation temperature of 13% sodium pentaborate, by weight, is 59°F. To guard against boron precipitation, the solution including that in the pump suction piping is kept at least 10°F above the saturation temperature by a tank heater and by heat tracing in the pump suction piping. The 10°F margin is included in Figure 3.3.1. Temperature and liquid level alarms for the system are annunciated in the control room.

Pump operability is checked on a frequency to assure a high reliability of operation of the system should it ever be required.

Once the solution has been made up, boron concentration will not vary unless more boron or more water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

4.4 SURVEILLANCE REQUIREMENT BASES

None

3.5 LIMITING CONDITION FOR OPERATION

CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the emergency cooling subsystems.

Objective:

To assure adequate cooling capability for heat removal in the event of a loss of coolant accident or isolation from the normal reactor heat sink.

Specification:

A. Core Spray and LPCI Subsystems

1. Except as specified in 3.5.A.2, 3.5.A.3, and 3.5.F.3 below, both core spray subsystems shall be operable whenever irradiated fuel is in the reactor vessel.

4.5 SURVEILLANCE REQUIREMENT

CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to periodic testing of the emergency cooling subsystems.

Objective:

To verify the operability of the emergency cooling subsystems.

Specification:

A. Surveillance of the Core Spray and LPCI Subsystems shall be performed as follows:

1. Core Spray Subsystem Testing:

Item	Frequency
a. Simulated Automatic Actuation Test	Each Refueling Outage
b. Flow Rate Test Core spray pumps shall deliver at least 4500 gpm against a system head corresponding	After pump maintenance and every 3 months

3.5 LIMITING CONDITION FOR OPERATION (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, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the other core spray subsystem and the LPCI subsystem and the diesel generators

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

to a reactor vessel pressure of 90 psig	
c. Pump Operability	Once/month
d. Motor Operated Valve	Once/month
e. Core Spray header delta p instrumentation:	
check	Once/day
calibrate	Once/3 months
test	Once/3 months
f. Logic System Functional Test	Each Refueling Outage

2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem and the LPCI subsystem and the diesel generators required for operation of such components if no external source of power were available shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

required for operation of such components if no external source of power were available shall be operable.

3. Except as specified in 3.5.A.4, 3.5.A.5 and 3.5.F.3 below, the LPCI subsystem shall be operable whenever irradiated fuel is in the reactor vessel.

4. From and after the date that one of the LPCI pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days the remaining active components of the LPCI and containment cooling subsystem and all active components of both core spray subsystems and the diesel generators required for operation of such components if no external source of power were available shall be operable.

5. From and after the date that the LPCI subsystem is made or found to be inoperable

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3. LPCI Subsystem Testing shall be as specified in 4.5.A.1.a, b, c, d, and f, except that three LPCI pumps shall deliver at least 14,500 gpm against a system head corresponding to a reactor vessel pressure of 20 psig.

4. When it is determined that one of the LPCI Pumps is inoperable, the remaining active components of the LPCI and containment cooling subsystem, both core spray subsystems and the diesel generators required for operation of such components if no external source of power were available shall be demonstrated to be operable immediately and the operable LPCI pumps daily thereafter.

5. When it is determined that the LPCI subsystem is inoperable, both core spray subsystems,

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

for any reason, reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided that during such seven days all active components of both core spray subsystems, the containment cooling subsystem (including 2 LPCI pumps) and the diesel generators required for operation of such components if no external source of power were available shall be operable.

6. Containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a maximum of one drywell spray loop may be inoperable for thirty days when the reactor water temperature is greater than 212°F.

7. If the requirements of 3.5.A cannot be met, either 3.5.G shall be compiled with or an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

the containment cooling subsystem, and the diesel generators required for operation of such components if no external source of power were available shall be demonstrated to be operable immediately and daily thereafter.

6. During each five year period an air test shall be performed on the drywell spray headers and nozzles.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

B. Containment Cooling Subsystem

1. Except as specified in 3.5.B.2, 3.5.B.3, and 3.5.F.3 below, both containment cooling subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that one of the containment cooling service water subsystem pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days all other active components of the containment cooling subsystem are operable.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

B. Surveillance of the Containment Cooling Subsystem shall be performed as follows:

1. Containment Cooling Service Water Subsystem Testing:

Item	Frequency
a. Pump & Valve Operability	Once/3 months
b. Flow Rate Test. Each containment cooling water pump shall deliver at least 3500 gpm against a pressure of 180 psig.	After pump maintenance and every 3 months

2. When it is determined that one containment cooling service water pump is inoperable, the remaining components of that subsystem and the other containment cooling subsystem shall be demonstrated to be operable immediately and daily thereafter.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. From and after the date that one containment cooling subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that all active components of the other containment cooling subsystem, both core spray subsystems and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

4. If the requirements of 3.5.B cannot be met an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

C. HPCI Subsystem

1. Except as specified in 3.5.C.2 below, the HPCI subsystem shall be operable whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When one containment cooling subsystem becomes inoperable, the operable subsystem and the diesel generators required for operation of such components shall be demonstrated to be operable immediately and the operable containment cooling subsystem daily thereafter.

C. Surveillance of HPCI Subsystem shall be performed as follows:

1. HPCI Subsystem Testing shall be as specified in 4.5.A.1.a, b, c, d, and f, except that the HPCI pump shall deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of 1150 psig to 150 psig.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. From and after the date that the HPCI subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the Automatic Pressure Relief Subsystem, the core spray subsystems, LPCI subsystem, and isolation cooling system are operable.

3. If the requirements of 3.5.C cannot be met an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

2. When it is determined that HPCI subsystem is inoperable, the LPCI subsystem, both core spray subsystems, the automatic pressure relief subsystem, and the motor operated isolation valves and shell side make-up system for the isolation condenser system shall be demonstrated to be operable immediately. The motor operated isolation valves and shell side make-up system of the isolation condenser shall be demonstrated to be operable daily thereafter. Daily demonstration of the automatic pressure relief subsystem operability is not required provided that two feedwater pumps are operating at power levels above 300 MWe; and one feedwater pump is operating as normally required with one additional feedwater pump operable at power levels less than 300 MWe.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

D. Automatic Pressure Relief Subsystems

1. Except as specified in 3.5.D.2 and 3 below, the Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig and and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

D. Surveillance of the Automatic Pressure Relief Subsystem shall be performed as follows:

1. During each operating cycle the following shall be performed:

   a. A simulated automatic initiation which opens all pilot valves, and

   b. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure.

   c. A logic system functional test shall be performed each refueling outage.

2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

vessel, reactor operation is permissible only during the succeeding seven days unless repairs are made and provided that during such time the HPCI Subsystem is operable.

3. From and after the date that more than one of five relief valves of the automatic pressure relief subsystem made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 24 hours unless repairs are made and provided that during such time the HPCI Subsystem is operable.

4. If the requirements of 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

E. Isolation Condenser System

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When it is determined that more than one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately.

E. Surveillance of the Isolation Condenser System shall be performed as follows:

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. Whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel, the isolation condenser shall be operable except as specified in 3.5.F.2.

2. From and after the date that the isolation condenser system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such system is sooner made operable, provided that during such seven days all active components of the HPCI subsystem are operable.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

1. Isolation Condenser System Testing:

   a. The shell side water level and temperature shall be checked daily.

   b. Simulated automatic actuation and functional system testing shall be performed during each refueling outage or whenever major repairs are completed on the system.

   c. The system heat removal capability shall be determined once every five years.

   d. Calibrate vent line radiation monitors quarterly.

2. When it is determined that the isolation condenser system is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately and daily thereafter.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. If the requirements of 3.5.E cannot be met an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

F. Minimum Core and Containment Cooling System Availability

1. During any period when the unit or shared diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days provided that all of the low pressure core cooling and containment cooling subsystems shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

F. Surveillance of Core and Containment Cooling System

1. When it is determined that either the unit or shared diesel generator is inoperable, all low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter.

2. Actions necessary to assure that the plant can be safely shut down and maintained in this condition in case of failure of the Dresden Dam shall be demonstrated to be adequate every third refueling outage. If this Specification has been complied with for Dresden Unit 2, it shall not be required for Dresden Unit 3.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When irradiated fuel is in the reactor vessel and reactor is in the cold shutdown condition, all low pressure core and containment cooling subsystems may be inoperable provided no work is being done which has the potential for draining the reactor vessel.

4. When irradiated fuel is in the reactor vessel and the reactor is in the refuel condition, the torus may be drained completely and control rod drive maintenance performed provided that the spent fuel pool gates are open, the fuel pool water level is maintained above the low level alarm point, and the minimum total condensate storage reserve is maintained at 230,000 gallons, and provided that not more than one control rod drive housing is open at one time, the control rod drive housing is blanked following removal of the control rod drive, no work is being performed in the reactor vessel while the housing is open and a special flange is

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

available which can be used to blank an open housing in the event of a leak.

5. When irradiated fuel is in the reactor and the vessel head is removed, work that has the potential for draining the vessel may be performed with less than 112,000 ft^3 of water in the suppression pool, provided that: 1) the total volume of water in the suppression pool, dryer separator above the shield blocks, refueling cavity, and the fuel storage pool above the bottom of the fuel pool gate is greater than 112,000 ft^3; 2) the fuel storage pool gate is removed; 3) the low pressure coolant injection and core spray systems are operable; and 4) the automatic mode of the drywell sump pumps is disabled.

H. Maintenance of Filled Discharge Pipe

Whenever core spray, LPCI, or HPCI ECCS are required to be operable, the discharge piping from the pump discharge

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

H. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to, to assure that the discharge piping of the core spray,

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

of these systems to the last check valve shall be filled.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

LPCI, and HPCI are filled:

1. Every month prior to the testing of the LPCI and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow observed.

2. Following any period where the LPCI or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI system is lined up to take suction from the torus, the discharge piping of the HPCI shall be vented from the high point of the system and water flow observed on a monthly basis.

4. The pressure switches which monitor the LPCI and core spray system discharge lines to assure that they are full shall be functionally tested every month and calibrated every three months.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

I. Average Planar LHGR

During steady state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of all the rods in any fuel assembly, as a function of average planar exposure for G.E. fuel and average bundle exposure for Exxon fuel at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5-1. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. LOCAL LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly fabricated

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

I. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure for G.E. fuel and average bundle exposure for Exxon fuel shall be determined daily during reactor operation at greater than or equal to 25% rated thermal power.

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

by GE at any axial location shall not exceed the design value of 13.4 kw/ft.

If at any time during operation, it is determined by normal surveillance that the limiting value for LHGR for G.E. fuel is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

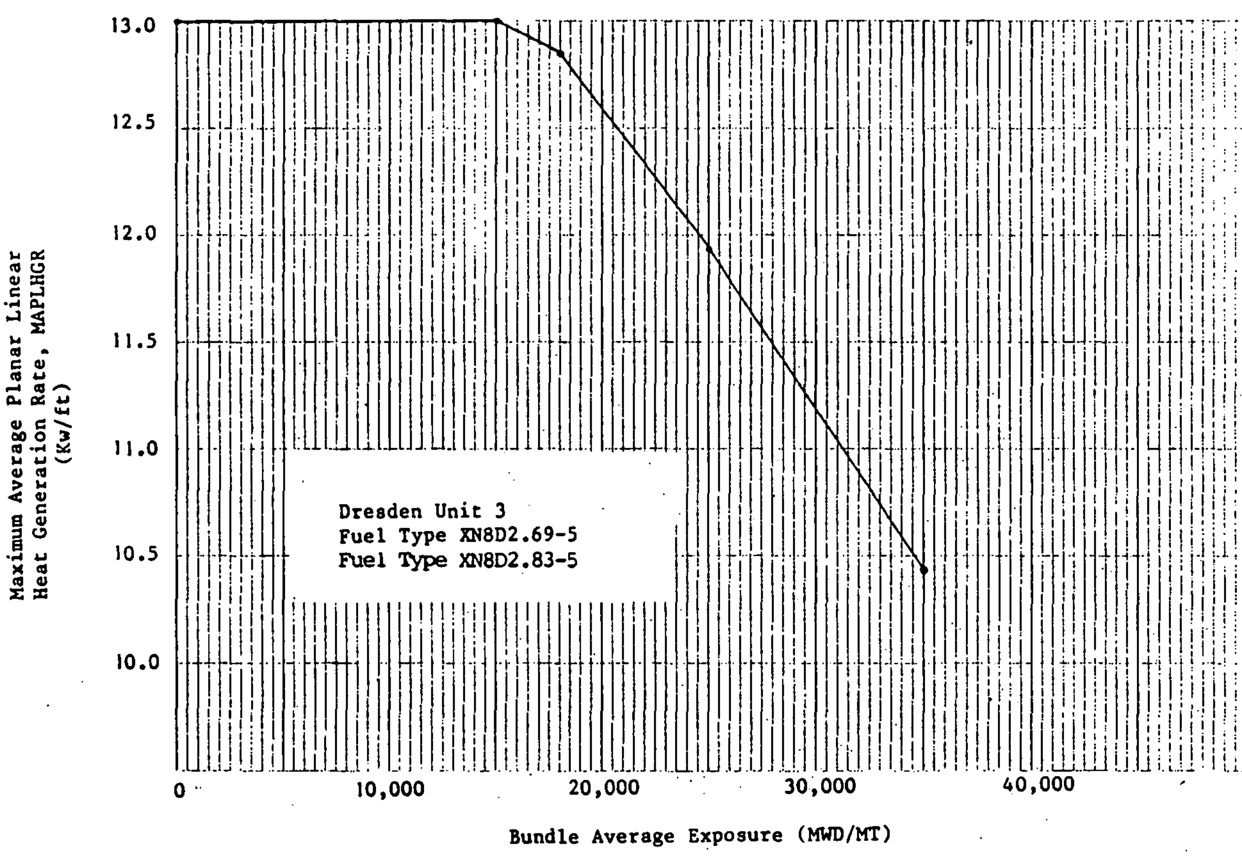


Figure 3.5-1
(Sheet 1 of 5)

Maximum Average Planar
Linear Heat Generation Rate (MAPLHGR)
vs. Bundle Average Exposure

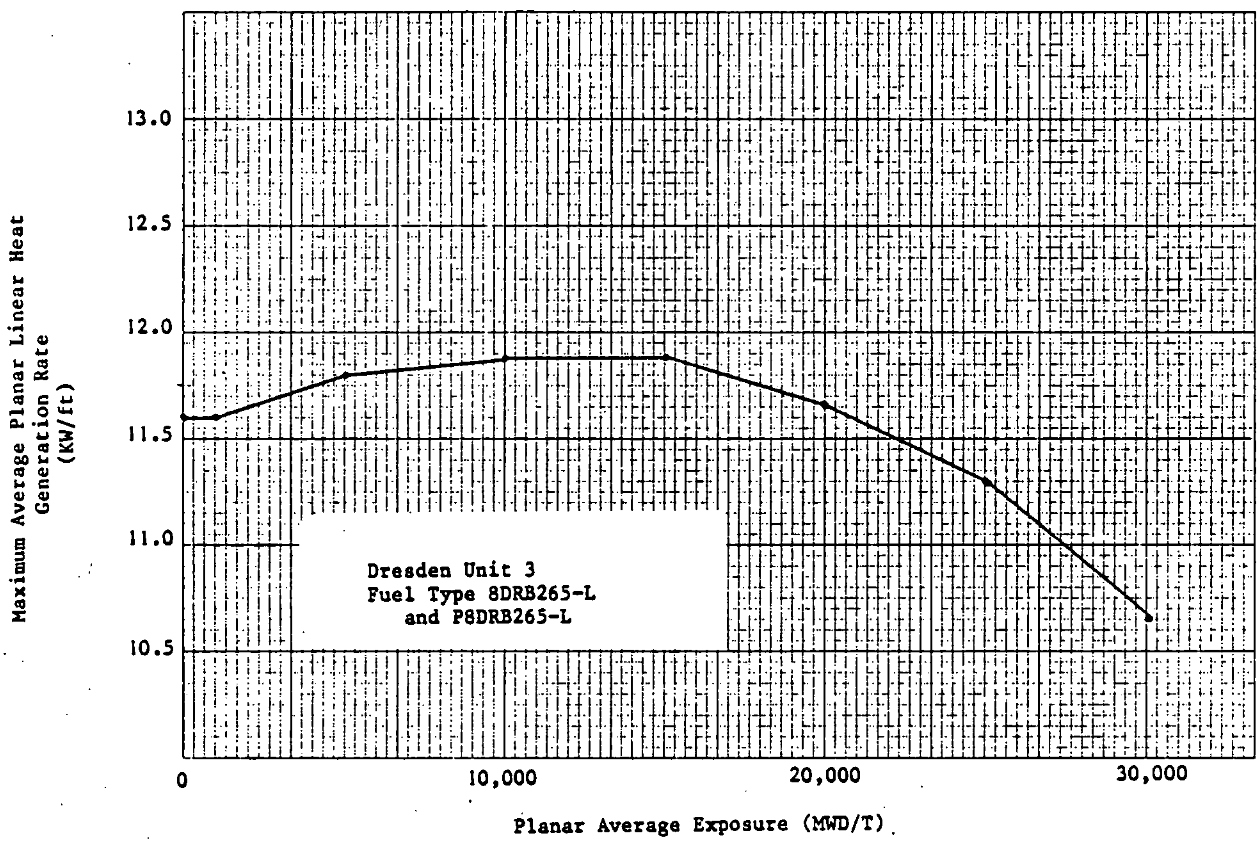
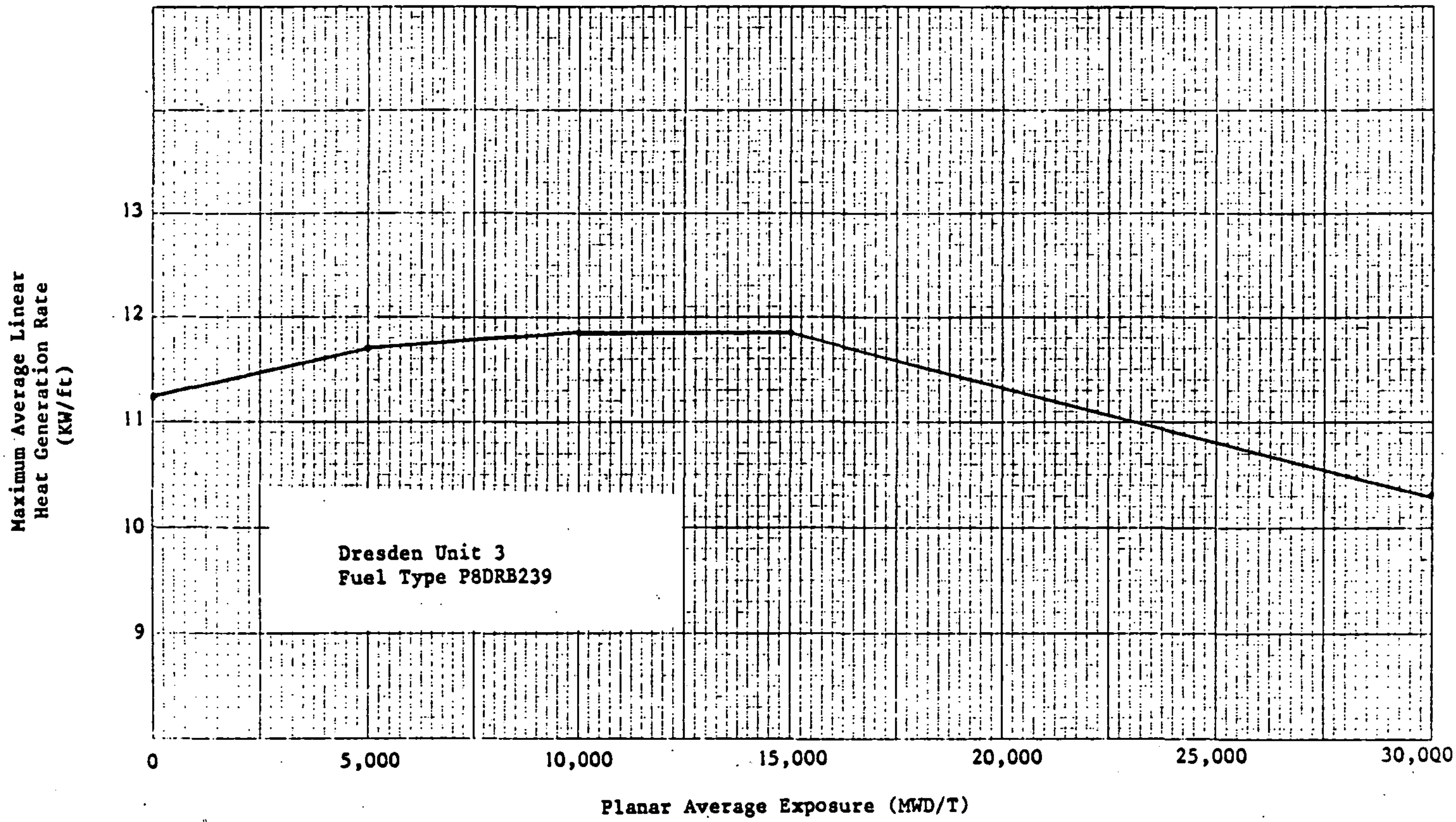


Figure 3.5-1
(Sheet 2 of 5)

Figure 3.5-1
(Sheet 2 of 5)

Maximum Average Planar
Linear Heat Generation Rate (MAPLHGR)
Vs. Planar Average Exposure



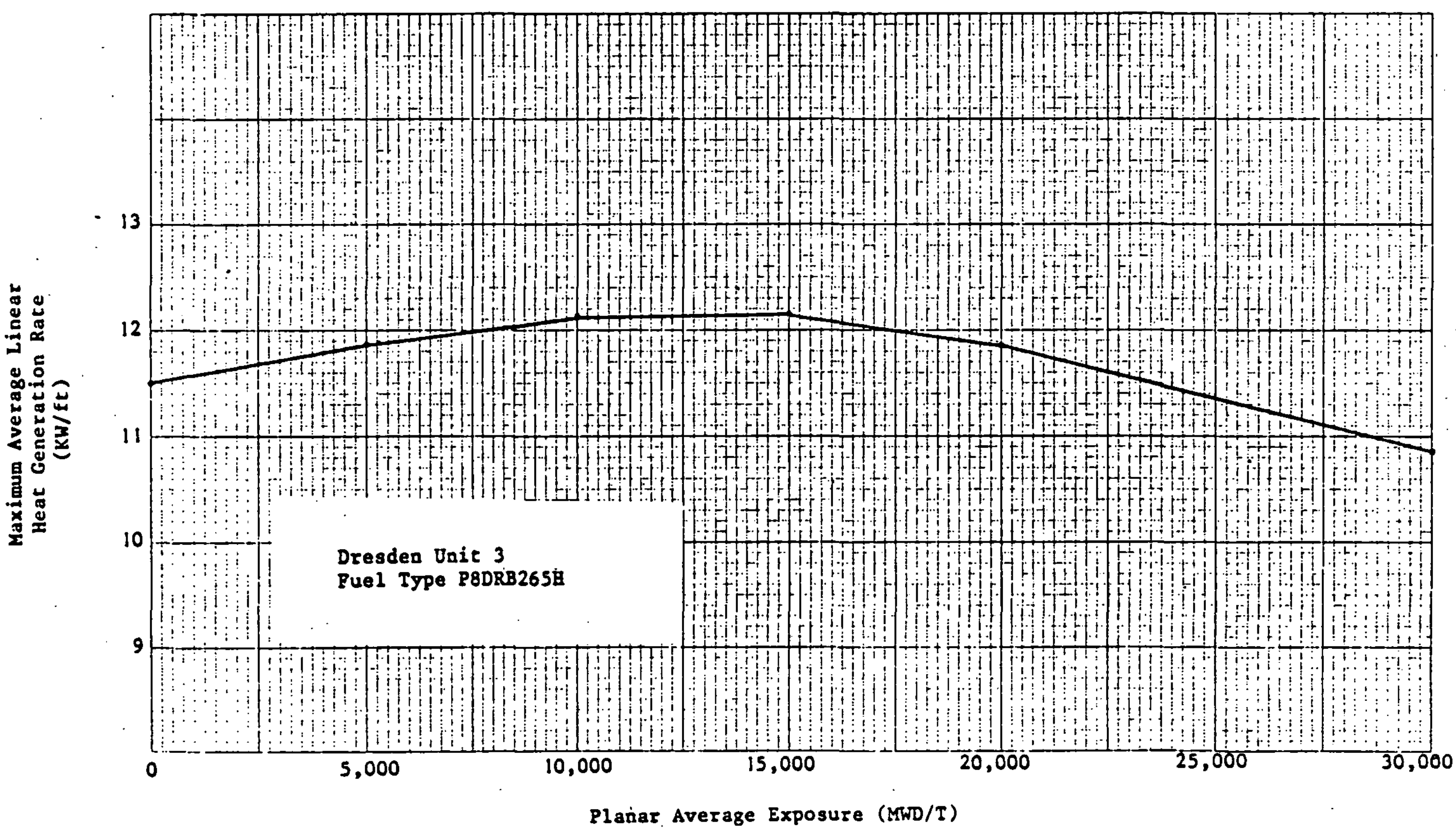


Figure 3.5-1

(Sheet 3 of 5)

Maximum Average Planar Linear Heat Generation Rate
(MAPLHGR) vs. Planar Average Exposure

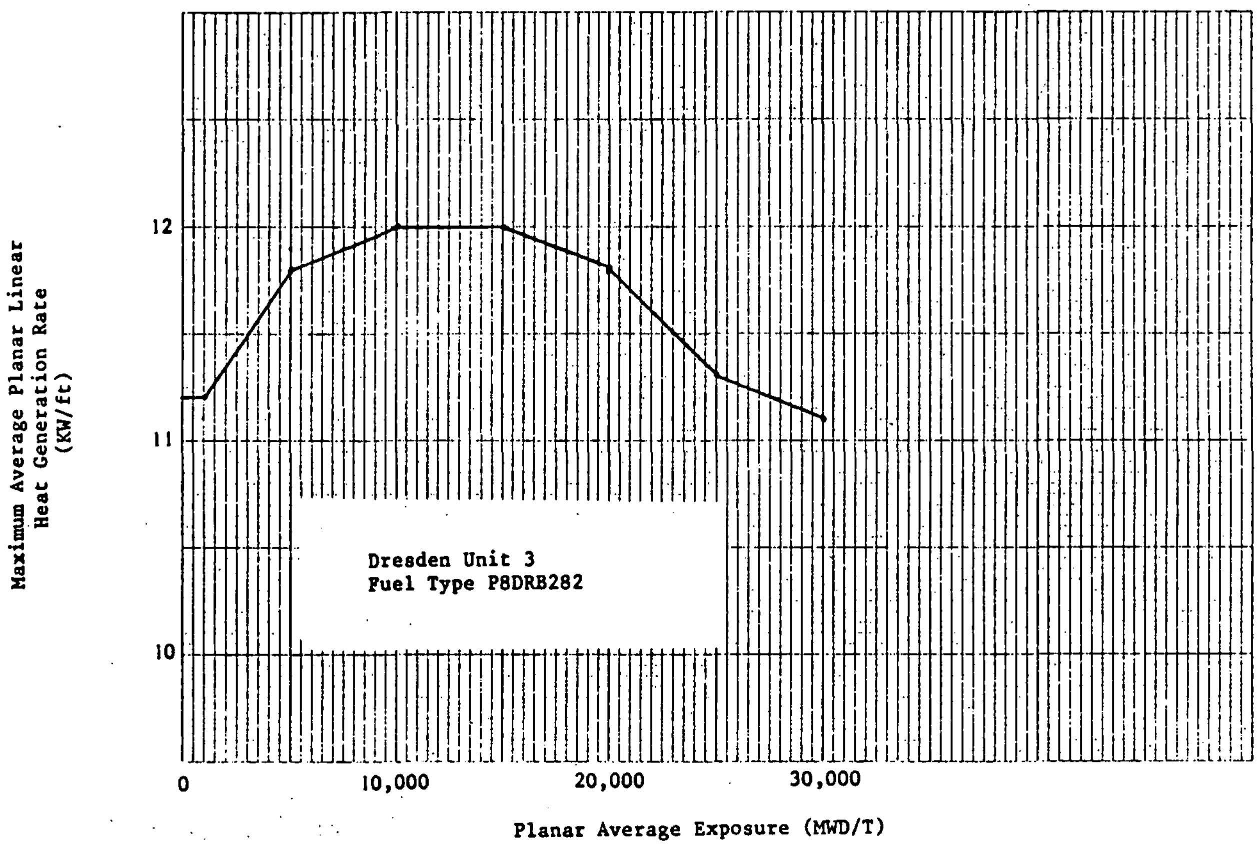
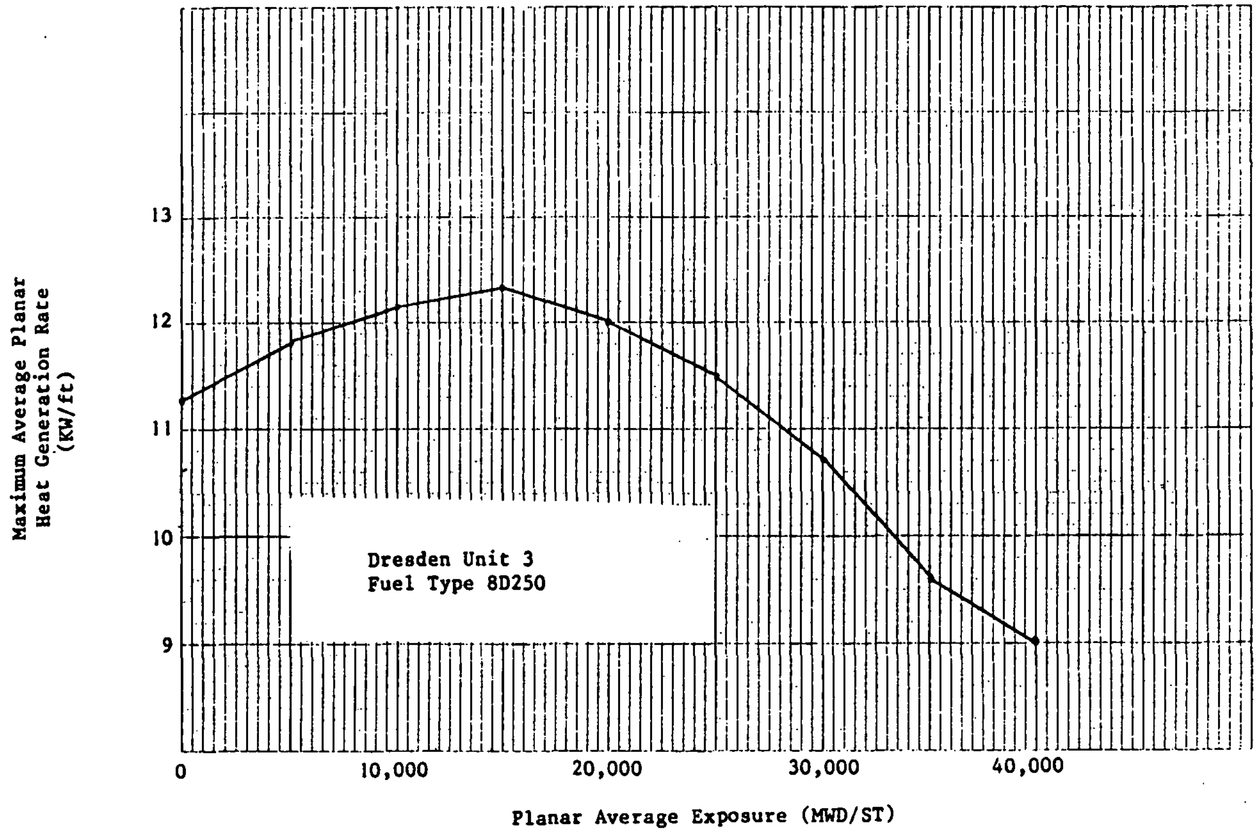


Figure 3.5-1
(Sheet 4 of 5)

Figure 3.5-1
(Sheet 4 of 5)

Maximum Average Planar
Linear Heat Generation Rate (MAPLHGR)
vs. Planar Average Exposure



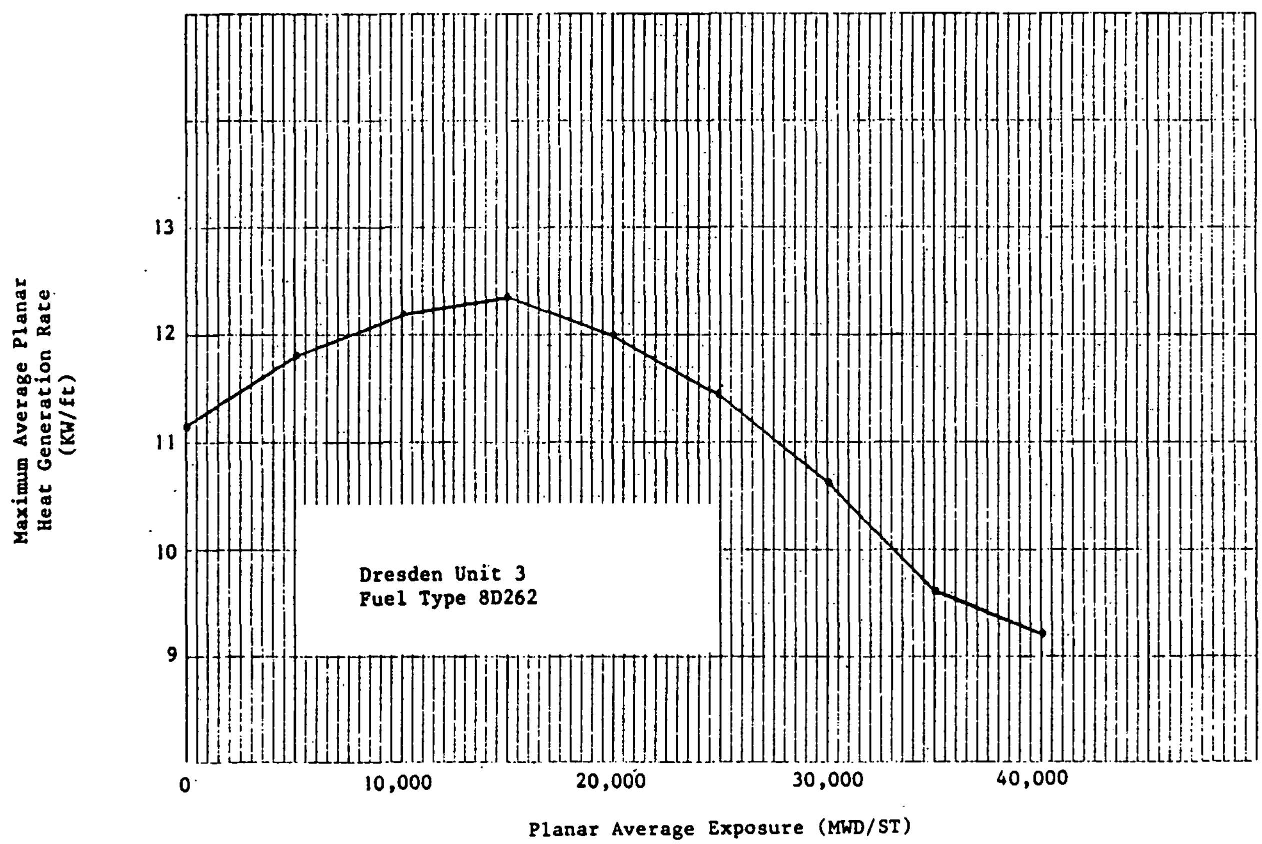


Figure 3.5-1

(Sheet 5 of 5)

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

During steady state operation at rated core flow, MCPR shall be greater than or equal to

1.34 for GE 8 x 8R fuel

1.33 for ENC and GE 8 x 8 fuel

For core flows other than rated, the MCPR Operating Limit shall be as follows:

1. Manual Flow Control - the MCPR Operating Limit shall be the value from Figure 3.5-2 sheet 1 or the above rated flow value, whichever is greater.
2. Automatic Flow Control - the MCPR Operating Limit shall be the value from Figure 3.5-2 sheet 1, sheet 2, or the above rated flow value, whichever is greatest.

If at any time during steady state power operation, it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at greater than or equal to 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

In the event the average 90% scram insertion time determined by Spec. 3.3.C for all operable control rods exceeds 2.58 seconds, the MCPR limit shall be increased by the amount equal to [0.0544T - 0.14] where T equals the average 90% scram insertion time for the most recent half-core or full core surveillance data from Spec. 4.3.C.

L. Condensate Pump Room Flood Protection

1. The system is installed to prevent or mitigate the consequences of flooding of the condensate pump room shall be operable prior to startup of the reactor.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

L. Condensate Pump Room Flood Protection

1. The following surveillance requirements shall be observed to assure that the condensate pump room flood protection is operable.

   a. The testable penetrations through the walls of CCSW pump vaults shall be checked during each operating cycle by pressurizing to 15 plus or minus 2 psig and checking for leaks using a soap bubble

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

solution. The criteria for acceptance should be no visible leakage through the soap bubble solution. The bulkhead door shall be checked during each operating cycle by hydrostatically testing the door at plus or minus 2 psig and checking to verify that leakage around the door is less than one gallon per hour.

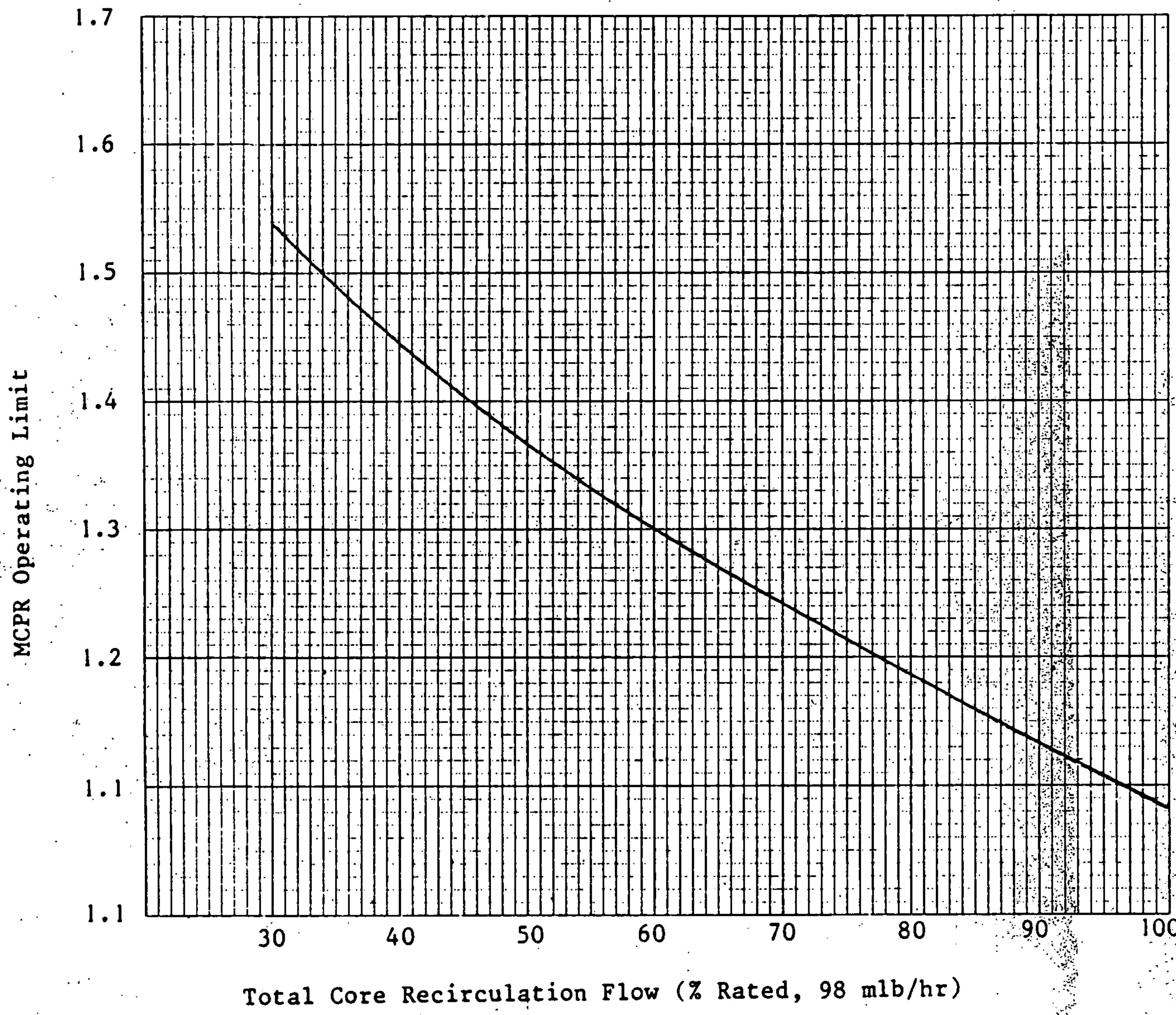


Figure 3.5-2 (Sheet 1 of 2)
MCPR Limit For Reduced Core Flow

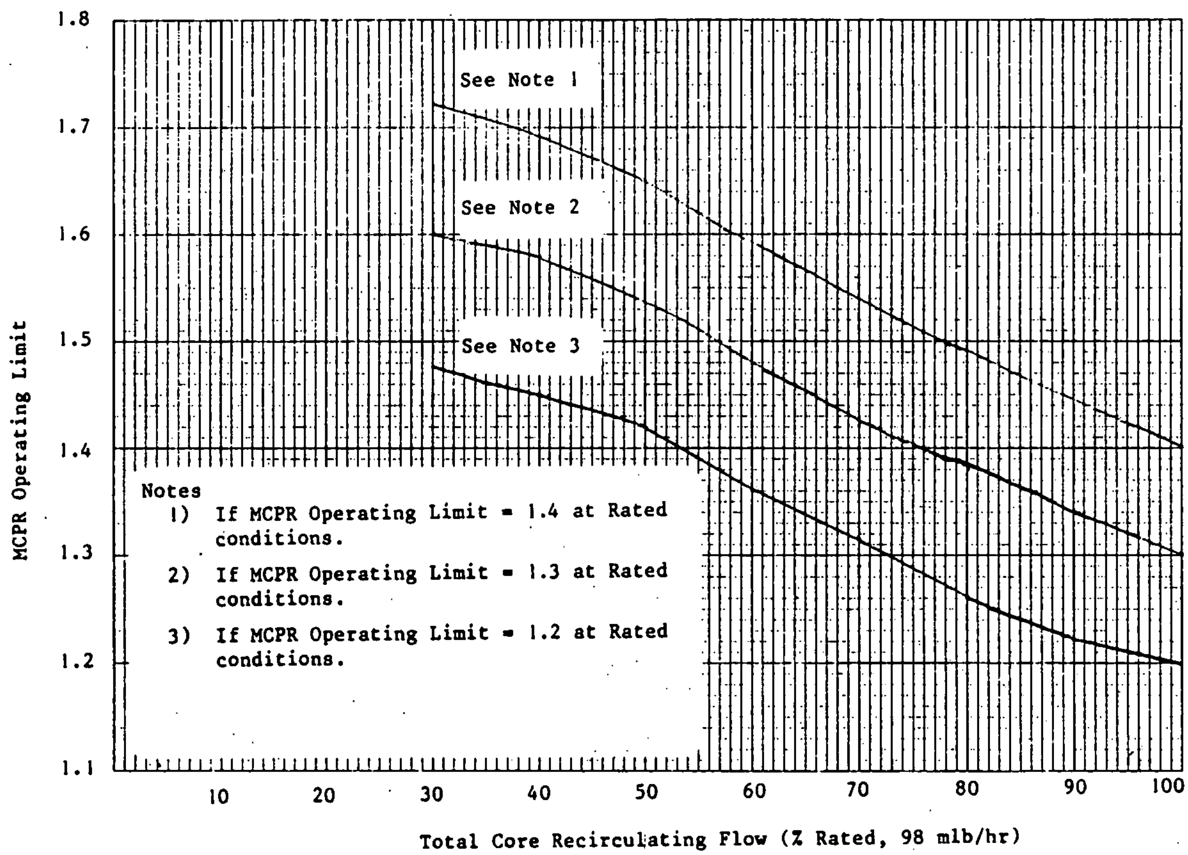


Figure 3.5-2 (Sheet 2 of 2)
MCPR Limit For Automatic Flow Control

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

b. The CCSW Vault Floor drain shall be checked during each operating cycle by assuring that water can be run through the drain line and actuating the air operated valves by operation of the following sensor:

   i. loss of air

   ii. high level in the condensate pump room (5'0")

c. The condenser pit five foot trip shall have a trip setting of less than or equal to five feet zero inches. The five foot trip circuit for each channel shall be checked once every three months. The 3 and 1 foot alarms shall have a setting of less than or equal to three feet zero inches and less than or equal to 1 foot 0 inches. A logic system functional test, including all alarms, shall be performed during the refueling outage.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The condenser pit water level switches shall trip the condenser circulating water pumps and alarm in the control room if water level in the condenser pit exceeds a level of 5 feet above the pit floor. If a failure occurs in one of these trip and alarm circuits, the failed circuit shall be immediately placed in a trip condition and reactor operation shall be permissible for the following seven days unless the circuit is sooner made operable.

3. If Specification 3.5.L.1 and 2 cannot be met, reactor startup shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3.5 LIMITING CONDITION FOR OPERATION BASES

A. Core Spray and LPCI Mode of the RHR System - This specification assures that adequate emergency cooling capability is available.

Based on the loss of coolant analyses included in References (1) and (2) in accordance with 10CFR50.46 and Appendix K, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit the calculated peak clad temperature to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

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 (3). Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems 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 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 45 days and this specification is within this period. For multiple failures, a shorter interval is specified and to improve the assurance that the remaining

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(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revisions 1, April 1979.

(2) NEDO-20566, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K.

(3) APED-"Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards" - April 1969, I.M. Jacobs and P.W. Marriott.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

systems will function, a daily test is called for. Although it is recognized that the information given in reference 3 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgement.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the reactor 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. Based on judgements of the reliability of the remaining systems; i.e. the core spray and LPCI, a 7-day repair period was obtained.

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 30-day repair period is justified. If the LPCI subsystem is not available, at least 2 LPCI pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. Containment Cooling Service Water - The containment heat removal portion of the LPCI/containment cooling subsystem is provided to remove heat energy from the containment in the event of a loss of coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and, therefore, is more than ample to provide the required heat removal capability. (Ref. Section 5.2.3.2 SAR).

The containment cooling subsystem consists of two sets of 2 service water pumps, 1 heat exchanger and 2 LPCI pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one containment cooling service water pump does not seriously jeopardize the containment cooling capability as any 2 of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left a 30-day repair period is adequate. Loss

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

of 1 containment cooling subsystem leaves one remaining system to perform the containment cooling function. The operable system is demonstrated to be operable each day when the above condition occurs. Based on the facts that when one containment cooling subsystem becomes inoperable only one system remains which is tested daily. A 7-day repair period was specified.

C. High Pressure Coolant Injection - The high pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of off-site electrical power. For the pipe breaks for which the HPCI is intended to function the core never uncovers and is continuously cooled and thus no clad damage occurs. (Ref. Section 6.2.5.3 SAR). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

D. Automatic Pressure Relief - The relief valves of the automatic pressure relief subsystem are a back-up to the HPCI subsystem. They enable the core spray or LPCI to provide protection against the small pipe break in the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core sprays or LPCI. The core spray and/or LPCI provide sufficient flow of coolant to adequately cool the core.

Loss of 1 of the relief valves affects the pressure relieving capability and therefore a 7 day repair period is specified. Loss of more than 1 relief valve significantly reduces the pressure relief capability and thus a 24-hour repair period is specified.

E. Isolation Cooling System - The turbine main condenser is normally available. The isolation condenser is provided for core decay heat removal following reactor isolation and scram. The isolation condenser has a heat removal capacity sufficient to handle the decay heat production at 300 seconds following a scram. Water will be lost from the reactor vessel through the relief valves in the 300 seconds following isolation and scram. This represents a minor loss relative to the vessel inventory.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1060 psig sustained for 15 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. To be considered operable the shell side of the isolation condenser must contain at least 11,300 gallons of water. Make-up water to the shell side of the isolation condenser is provided by the condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are operable from on-site power. The fire protection system is also available as make-up water. An alternate method of cooling the core upon isolation from the main condenser is by using the relief valves and HPCI subsystem in a feed and bleed manner. Therefore, the high pressure relief function and the HPCI must be available together to cope with an anticipated transient so the LCO for HPCI and relief valves is set upon this function rather than their function as depressurization means for a small pipe break.

F. Emergency Cooling Availability - The purpose of Specification D is to assure a minimum of core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only 2 LPCI pumps would be available. Likewise, if 2 LPCI pumps were out of service and 2 containment service water pumps on the opposite side were also out of service no containment cooling would be available. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

Dresden Units 2 and 3 share certain process systems such as the makeup demineralizers and the radwaste system and also some safety systems such as the standby gas treatment system, batteries, and diesel generators. All of these systems have been sized to perform their intended function considering the simultaneous operation of both units.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

For the safety related shared features of each plant, the Technical Specifications for that unit contain the operability and surveillance requirements for the shared feature; thus, the level of operability for one unit is maintained independently of the status of the other. For example, the shared diesel (2/3 diesel) would be mentioned in the specifications for both Units 2 and 3 and even if Unit 3 were in the Cold Shutdown Condition and needed no diesel power, readiness of the 2/3 diesel would be required for continuing Unit 2 operation.

G. Specification 3.5.F.4 provides that should this occur, no work will be performed which could preclude adequate emergency cooling capability being available. Work is prohibited unless it is in accordance with specified procedures which limit the period that the control rod drive housing is open and assures that the worst possible loss of coolant resulting from the work will not result in uncovering the reactor core. Thus, this specification assures adequate core cooling. Specification 3.9 must be consulted to determine other requirements for the diesel generator.

Specification 3.5.F.5 provides assurance that an adequate supply of coolant water is immediately available to the low pressure core cooling systems and that the core will remain covered in the event of a loss of coolant accident while the reactor is depressurized with the head removed.

H. Maintenance of Filled Discharge Pipe - If the discharge piping of the core spray, LPCI, and HPCI are not filled, a water hammer can develop in this piping when the pump and/or pumps are started.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10CFR50 Appendix K considering the postulated affects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average LHGR of all the rods in a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within a fuel assembly. Since expected

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than plus or minus 20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the 10CFR50, Appendix K limit.

The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in Reference (1) and in reference (2). Power operation with APLHGRs at or below those shown in Fig. 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit.

The maximum average planar LHGRs for G.E. fuel plotted in Fig. 3.5.1 at higher exposures result in a calculated peak clad temperature of less than 2200°F. However, the maximum average planar LHGRs are shown on Fig. 3.5.1 as limits because conformance calculations have not been performed to justify operation at LHGHs in excess of those shown.

J. Local LHGR

This specification assures that the maximum linear heat generation rate in any fuel rod fabricated by G.E. is less than the design linear heat generation rate even if fuel pellet densification is postulated.

For fuel fabricated by ENC, protection of the MCPR and MAPLHGR limits and operation within the power distribution assumptions of the Fuel Design Analysis provides adequate protection against cladding strain limits, hence the LHGR limitation for GE fuel is unnecessary for the protection of ENC fuel.

(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision 1, April, 1979.

(2) XN-NF-81-75 "Dresden Unit 3 LOCA Model Using the ENC EXEM Evaluation Model MAPLHGR Results"

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

The steady-state values for MCPR specified in the Specification were determined using the THERMEX thermal limits methodology described in XN-NF-80-19, Volume 3. The safety limit implicit in the Operating limits is established so that during sustained operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. The Limiting Transient delta CPR implicit in the operating limits was calculated such that the occurrence of the limiting transient from the operating limit will not result in violation of the MCPR safety limit in at least 95% of the random statistical combinations of uncertainties.

Transient events of each type anticipated during operation of a BWR/3 were evaluated to determine which is most restrictive in terms of thermal margin requirements. The generator load rejection/turbine trip without bypass is typically the limiting event. The thermal margin effects of the event are evaluated with the THERMEX Methodology and appropriate MCPR limits consistent with the XN-3 critical power correlation are determined. Several factors influence which transient results in the largest reduction in critical power ratio, such as the cycle-specific fuel loading, exposure and fuel type. The current cycle's reload licensing analyses identifies the limiting transient for that cycle.

As described in Specification 4.3.C.3 and the associated Bases, observed plant data were used to determine the average scram performance used in the transient analyses for determining the MCPR Operating Limit. If the current cycle scram time performance falls outside of the distribution assumed in the analyses, an adjustment of the MCPR limit may be required to maintain margin to the MCPR Safety Limit during transients. Compliance with the assumed distribution and adjustment of the MCPR Operating Limit will be performed as directed by the nuclear fuel vendor in accordance with station procedures.

For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the MCPR Safety Limit in the event of an uncontrolled recirculation flow increase to the physical limit of pump flow. This protection is provided for manual and automatic flow control by choosing the MCPR operating limit as the value from Figure 3.5-2 Sheet 1 or the rated core flow value, whichever is greater. For Automatic Flow Control, in addition to protecting the MCPR Safety Limit during the flow

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

run-up event, protection is provided against violating the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow. This protection is provided by the reduced flow MCPR limits shown in Figure 3.5-2 Sheet 2 where the curve corresponding to the current rated flow MCPR limit is used (linear interpolation between the MCPR limit lines depicted is permissible). Therefore, for Automatic Flow Control, the MCPR Operating Limit is chosen as the value from Figure 3.5-2 Sheet 1, Sheet 2 or the rated flow value, whichever is greatest. It should be noted that if the rated flow MCPR Limit must be increased due to degradation of control rod scram times during the current cycle, the new value of the rated flow MCPR limit is applied when using Figure 3.5-2 Sheet 2.

L. Flood Protection

Condensate pump room flood protection will assure the availability of the containment cooling service water system (CCSW) during a postulated incident of flooding in the turbine building. The redundant level switches in the condenser pit will preclude any postulated flooding of the turbine building to an elevation above river water level. The level switches provide alarm and circulating water pump trip in the event a water level is detected in the condenser pit.

4.5 SURVEILLANCE REQUIREMENT BASES

(A thru F)

The testing interval for the core and containment cooling systems is based on quantitative reliability analysis, judgement and practicality. The core cooling systems have not been designed to be fully testable during operation. For example the core spray final admission valves do not open until reactor pressure has fallen to 350 psig thus during operation even if high drywell pressure were stimulated the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

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 preventative 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 due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The requirement of 180 psig at 3500 gpm at the containment cooling service water (CCSW) pump discharge provides adequate margin to ensure that the LPCI/CCSW system provides the design

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

bases cooling water flow and maintains 20 psig differential pressure at the containment cooling heat exchanger. This differential pressure precludes reactor coolant from entering the river water side of the containment cooling heat exchangers.

The verification of Main Steam Relief Valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the valve actuation mechanism during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for relief valve opening. This test method may be performed over a wide range of reactor pressure greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

G. Deleted

H. Maintenance of Filled Discharge Pipe

The surveillance requirements to assure that the discharge piping of the core spray, LPCI, and HPCI systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided to monitor the presence of water in the discharge piping. This instrumentation will be calibrated on the same frequency as the safety system instrumentation. This period of periodic testing ensures that during the intervals between the monthly checks the status of the discharge piping is monitored on a continuous basis.

I. Average Planar LHGR

At core thermal power levels less than or equal to 25 per cent, operating plant experience and thermal hydraulic analyses indicate that the resulting average planar LHGR is below the maximum average planar LHGR by a considerable

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

margin; therefore, evaluation of the average planar LHGR below this power level is not necessary. The daily requirement for calculating average planar LHGR above 25 per cent rated thermal power is sufficient since power distribution shifts are slow when there have not been significant power or control rod changes.

J. Local LHGR

The LHGR for G.E. fuel shall be checked daily during reactor operation at greater than or equal to 25 per cent power to determine if fuel burnup or control rod movement has caused changes in power distribution. A limiting LHGR value is precluded by a considerable margin when employing a permissible control rod pattern below 25% rated thermal power.

K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 per cent, the reactor will be operating at 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 and thermal hydraulic analysis indicates 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.

The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

L. Flood Protection

The watertight bulkhead door and the penetration seals for pipes and cables penetrating the vault walls have been designed to withstand the maximum flood conditions. To assure that their installation is adequate for maximum flood conditions, a method of testing each seal has been devised.

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

To test a pipe seal, another test seal is installed in the opposite side of the penetration creating a space between the two seals that can be pressurized. Compressed air is then supplied to a fitting on the test seal and the space inside the sleeve is pressurized to approximately 15 psi. The outer face of the permanent seal is then tested for leaks using a soap bubble solution.

On completion of the test, the test seal is removed for use on other pipes and penetrations of the same size.

In order to test the watertight bulkhead doors, a test frame must be installed around each door. At the time of the test, a reinforced steel box with rubber gasketing is clamped to the wall around the door. The fixture is then pressurized to approximately 15 psig to test for leak tightness.

Floor drainage of each vault is accomplished through a carbon steel pipe which penetrates the vault. When open, this pipe will drain the vault floor to a floor drain sump in the condensate pump room.

Equipment drainage from the vault coolers and the CCSW pump bedplates will also be routed to the vault floor drains. The old equipment drain pipes will be permanently capped to preclude the possibility of back-flooding the vault.

As a means of preventing backflow from outside the vaults in the event of a flood, a check valve and an air operated valve are installed in the 2" vault floor drain line 6'0" above the floor of the condensate pump room.

The check valve is a 2" swing check designed for 125 psig service. The air operated valve is a control valve designed for a 50 psi differential pressure. The control valve will be in the normally open position in the energized condition and will close upon any one of the following:

a. Loss of air or power

b. High level (5'0") in the condensate pump room

Closure of the air operated valve on high water level in the condensate pump room is effected by use of a level switch set at a water level of 5'0". Upon actuation, the switch will close the control valve and alarm in the control room.

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The operator will also be aware of problems in the vaults/condensate pump room if the high level alarm on the equipment drain sump is not terminated in a reasonable amount of time. It must be pointed out that these alarms provide information to the operator but that operator action upon the above alarms is not a necessity for reactor safety since the other provisions provide adequate protection.

A system of level switches has been installed in the condenser pit to indicate and control flooding of the condenser area. The following switches are installed:

	Level	Function
a.	1'0" (1 switch)	Alarm, Panel Hi-Water-Condenser Pit
b.	3'0" (1 switch)	Alarm, Panel High-Circ. Water Condenser Pit
c.	5'0" (2 redundant switch pairs)	Alarm and Circ. Water Pump Trip

Level (a) indicates water in the condenser pit from either the hotwell or the circulating water system. Level (b) is above the hotwell capacity and indicates a probable circulating water failure.

Should the switches at level (a) and (b) fail or the operator fail to trip the circulating water pumps on alarm at level (b), the actuation of either level switch pair at level (c) shall trip the circulating water pumps automatically and alarm in the control room. These redundant level switch pairs at level (c) are designed and installed to IEEE-279, "Criteria for Nuclear Power Plant Protection Systems." As the circulating water pumps are tripped, either manually or automatically, at level (c) of 5'0", the maximum water level reached in the condenser pit due to pumping will be at the 491'0" elevation (10' above condenser pit floor elevation 481'0"; 5' plus an additional 5' attributed to pump coastdown).

In order to prevent overheating of the CCSW pump motors, a vault cooler is supplied for each pump. Each vault cooler is designed to maintain the vault at a maximum 105°F temperature during operation of its respective pump. For example, if CCSW pump 2B-1501 starts, its cooler will also start and compensate

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

for the heat supplied to the vault by the 2B pump motor keeping the vault at less than 105°F.

Each of the coolers is supplied with cooling water from its respective pump's discharge line. After the water has been passed through the cooler, it returns to its respective pump's suction line. In this way, the vault coolers are supplied with cooling water totally inside the vault. The cooling water quantity needed for each cooler is approximately 1% to 5% of the design flow of the pumps so that the recirculation of this small amount of heated water will not affect pump or cooler operation.

Operation of the fans and coolers is required during pump operability testing and thus additional surveillance is not required.

Verification that access doors to each vault are closed, following entrance by personnel, is covered by station operating procedures.

3.6 LIMITING CONDITION FOR OPERATION

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 Limitations

1. Except as indicated in 3.6.A.2 below, 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. A step reduction in reactor coolant temperature of 240°F is permissible so long as the limit in Specification 3.6.A.3 below is met.

3. At all times, the shell flange to shell temperature differential shall not exceed 140°F.

4.6 SURVEILLANCE REQUIREMENT

PRIMARY SYSTEM BOUNDARY

Applicability:

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

Objective:

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

Specification:

A. Thermal Limitations

1. During heatups and cooldowns the following temperatures shall be permanently recorded at 15 minute intervals:

   a. reactor vessel shell

   b. reactor vessel shell flange

   c. recirculation loops A & B

2. The temperatures listed in 4.6.A.1 shall be permanently recorded subsequent to a heatup or cooldown at 15 minute intervals until three consecutive readings are within 5 degrees of each other.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4. Core thermal power shall not exceed 25% of rated thermal power without forced recirculation.

B. Pressurization Temperature

1. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Curve C of Figure 3.6.1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Fig. 3.6.1. Figure 3.6.1 is effective through 6 effective full power years. At least six months prior to 6 effective full power years new curves will be submitted.

2. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel shell immediately below the vessel flange is greater than or equal to 100°F.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

B. Pressurization Temperature

1. Reactor Vessel shell temperature and reactor coolant pressure shall be permanently recorded at 15 minute intervals whenever the shell temperature is below 220°F and the reactor vessel is not vented.

2. When the reactor vessel head bolting studs are tightened or loosened the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3. Neutron flux monitors and samples shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The monitor and sample program where possible conform to ASTM E 185. The monitors and samples will be removed and tested as outlined in Table 4.6.2 to experimentally verify the calculated values of integrated neutron flux that are used to determine NDTT for Figure 4.6.1.

C. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed 20 microcuries of total iodine per ml of water.

C. Coolant Chemistry

1. a. A Sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactivity.

b. Isotopic analysis of a sample of reactor coolant shall be made at least once per month.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in 3.6.C.3:

   Conductivity 2 micro-mho/cm
   Chloride ion 0.1 ppm

3. For reactor startups the maximum value for conductivity shall not exceed 10 micro-mho/cm and the maximum value for chloride ion concentration shall not exceed 0.1 ppm, for the first 24 hours after placing the reactor in the power operating condition.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

2. During startups and at steaming rates below 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for conductivity and chloride content.

3. a. With steaming rates greater than or equal to 100,000 pounds per hour, a reactor coolant sample shall be taken at least every 96 hours and when the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes) and analyzed for conductivity and chloride ion content.

   b. When the continuous conductivity monitor is inoperable, a reactor coolant sample should be taken at least daily and analyzed for conductivity and chloride ion content.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4. Except as specified in 3.6.C.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hour.

   Conductivity 5 micro-mho/cm
   Chloride ion 0.5 ppm

5. If Specification 3.6.C.1, 3.6.C.2, 3.6.C.3 or 3.6.C.4 is not met, an orderly shutdown shall be initiated.

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

D. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system. Sump flow monitoring and recording shall be performed once per shift. Air sampling shall be performed once per day.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The primary containment sump sampling system and an air sampling system shall be operable during power operation. If either a sump water sample or a containment air sample cannot be obtained for any reason, reactor operation is permissible only during the succeeding seven days unless the system is made operable during this period.

E. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320°F, all nine of the safety valves shall be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

2. The primary containment sump sampling and air sampling system operability will be observed as part of 4.6.D.2.

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outages. The popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (Psig)
1	1135*
2	1240
2	1250
2	1260
2	1260

The allowable set point error for each valve is plus or minus 1%.

* (See next page)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. If Specification 3.6.E.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be less than or equal to 90 psig and less than or equal to 320° F within 24 hours.

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components".

Components of the primary system boundary whose inservice examination reveals the absence of flaw indications or flaw indications not in excess of the allowable indication standards of this Code are acceptable for continued service. Plant operation with components which have inservice examination flaw

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Valve No.	Set Point (psig)
203-3A	1124*
203-3B	1101
203-3C	1101
203-3D	1124
203-3E	1124

* Target Rock combination safety/relief valve

The allowable setpoint error for each valve is plus or minus 1%.

F. Structural Integrity

1. Beginning November 1, 1978, and updated every 40 months thereafter, the component inservice inspection program shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been given by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

a. Components whose inservice examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

(i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI.

(ii) Prior to the resumption of service, the NRC shall review the analysis and evaluation and

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

b. For components approved for continued service in accordance with paragraph "a" above, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive inservice inspection. An analysis and evaluation shall be submitted to the NRC following each inservice inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each inservice inspection, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

the affected component or require that the component be repaired or replaced.

c. Repair or replacement of components, including reexaminations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

G. Jet Pumps

1. Whenever the Reactor is in the Startup/Hot Standby or Run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

G. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the Startup/Hot Standby or Run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

H. Recirculation Pump Flow Mismatch

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of any jet pumps in the idle loop shall not vary by more than 10% from established patterns.

3. The baseline date required to evaluate the conditions in Specifications 4.6.G.1 and 4.6.G.2 will be acquired each operating cycle.

H. Recirculation Pump Flow Mismatch

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

2. If specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.

3. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.

4. Whenever one pump is operable and the remaining pump is in the tripped position, the operable pump shall be at a speed less than 65% before starting the inoperable pump.

I. Snubbers (Shock Suppressors)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

Recirculation pumps speed shall be checked daily for mismatch.

I. Snubbers (Shock) Suppressors)

The following surveillance requirements apply to all

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. During all modes of operation except cold shutdown and refuel, all safety related snubbers listed in Table 3.6.1.a and 3.6.1b shall be operable except as noted in Specification 3.6.I.2 through 3.6.I.4.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

safety related snubbers listed in Tables 3.6.1a and 3.6.1b.

1. Visual Inspection

   An independent visual inspection shall be performed on the safety related hydraulic and mechanical snubbers contained in Tables 3.6.1a and 3.6.1b in accordance with the below schedule.

   a. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connection to the piping and anchor to verify snubber operability.

   b. All mechanical snubbers shall be visually inspected. This

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

inspection shall consist of, but not necessarily be limited to, inspection of the snubber and attachments to the piping and anchor for indications of damage or impaired operability.

No. of Snubbers Found Inoperable During Inspection Interval	Next Required Inspection Interval
0	18 months plus or minus 25%
1	12 months plus or minus 25%
2	6 months plus or minus 25%
3,4	124 days plus or minus 25%
5,6,7	62 days plus or minus 25%
8 or more	31 days plus or minus 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible," based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. From and after the time a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced. Torus Ring Header snubbers may be inoperable in either of the following configurations until January 19, 1984 to facilitate the installation of the Mark I torus attached piping modifications.

Configuration A:
Every other existing snubber pair (up to 3 pairs) on the ECCS header, or

Configuration B:
One existing snubber from each of the 6 existing snubber pairs on the ECCS header.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

2. Functional Testing

a. Once each refueling cycle, a representative sample of approximately 10% of the hydraulic snubbers contained in Table 3.6.1a shall be functionally tested for operability, including:

*(i) Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

*(ii) Snubber bleed, or release rate, where required, is within the specified range in compression or tension.

*NOTE: Paragraphs (i) and (ii) shall not become effective until competitive marketable test fixtures are available. Until such time, but in no case to exceed 12/31/83, demonstration of snubber bleed, or release, shall be sufficient.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

For each unit and subsequent unit found inoperable, an additional 10% of the hydraulic snubbers shall be tested until no more failures are found or all units have been tested.

b. Once each refueling cycle, a representative sample of approximately 10% of the mechanical snubbers contained in Table 3.6.1b shall be functionally tested for operability. The test shall consist of two parts:

** i. Verification that the force that initiates free movement of the snubber in either tension or compression is less than the specified maximum breakaway friction force.

** (See next page)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

**ii. Verify that the activation (restraining action) is achieved within the specified range of acceleration in both tension and compression.

For each unit and subsequent unit found inoperable, an additional 10% of the mechanical snubbers shall be so tested until no more failures are found or all units have been tested.

c. In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the resampling.

** NOTE: Paragraph (ii) shall not become effective until competitive marketable test fixtures are available. Until then, but in no case to exceed 12/31/83, the functional test will be limited to only paragraph (i).

TABLE 3.6.1.a

HYDRAULIC SNUBBERS

SNUBBER NO.	LOCATION	ELEVATION	AZIMUTH	SNUBBER IN HIGH RADIATION AREA DURING SHUTDOWN	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
2	Torus Ring Header 1501-24" (Note 1)	483'	83°			X
3	Torus Ring Header 1501-24" (Note 1)	483'	74°			X
4	Torus Ring Header 1501-24" (Note 1)	483'	38°			X
5	Torus Ring Header 1501-24" (Note 1)	483'	29°			X
7	Torus Ring Header 1501-24" (Note 1)	483'	331°			X
8	Torus Ring Header 1501-24" (Note 1)	483'	286°			X
9	Torus Ring Header 1501-24" (Note 1)	483'	286°			X
10	Torus Ring Header 1501-24" (Note 1)	483'	227°			X
12	Torus Ring Header 1501-24" (Note 1)	483'	209°			X
13	Torus Ring Header 1501-24" (Note 1)	483'	209°			X
15	Torus Ring Header 1501-24" (Note 1)	483'	151°			X
	Isolation Condenser Pipeway Room:					
1	Isolation Condenser Line 1303-12"	558'	180°	X		X
2	Isolation Condenser Line 1303-12"	568'	180°	X		X
3	Isolation Condenser Line 1303-14"	580'	195°	X		X
	Drywell:					
23	Drywell Cleanup Line 1201-8"	537'6"	84°	X	X	

\# Modifications to this table, due to changes in high radiation, should be submitted to the NRC as part of next license amendment request.

Note 1. These snubbers are being replaced with mechanical snubbers as delineated in Section 3.6.1.2 and a revised table will be issued upon completion of the Mark I Torus attached piping modification.

TABLE 3.6.1.a

HYDRAULIC SNUBBERS

SNUBBER NO.	LOCATION	ELEVATION	AZIMUTH	SNUBBER IN HIGH RADIATION AREA DURING SHUTDOWN	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
	Isolation Condenser Pipeway Room:					
1	Isolation Condenser Line 1303-12"	558'	180°	X		X
2	Isolation Condenser Line 1303-12"	568'	180°	X		X
3	Isolation Condenser Line 1302-14"	580'	195°	X		X

*Modifications to this table due to changes in high radiation should be submitted to the NRC as part of the next license amendment request.

TABLE 3.6.1.b

MECHANICAL SNUBBERS

SNUBBER NO.	LOCATION	ELEVATION	AZIMUTH	SNUBBER IN HIGH RADIATION AREA DURING SHUTDOWN	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
1	Drywell Recirc. Motor 3B-202	524'	328°	X	X	
2	Drywell Recirc. Motor 3B-202	524'	302°	X	X	
3	Drywell Recirc. Motor 3B-202	524'	315°	X	X	
4	Drywell Recirc. Motor 3A-202	524'	148°	X	X	
5	Drywell Recirc. Motor 3A-202	524'	122°	X	X	
6	Drywell Recirc. Motor 3A-202	524'	135°	X	X	
7	Drywell Recirc. Pump 3B-202	512'	326°	X	X	
8	Drywell Recirc. Pump 3B-202	512'	304°	X	X	
9	Drywell Recirc. Pump 3B-202	507'	315°	X	X	
10	Drywell Recirc. Pump 3A-202	512'	124°	X	X	
11	Drywell Recirc. Pump 3A-202	512'	146°	X	X	
12	Drywell Recirc. Pump 3A-202	507'	135°	X	X	
15	Drywell LPCI Line 1506-16"	513'	256°	X	X	
16	Drywell LPCI Line 1519-16"	513	95°	X	X	
21	Drywell Recirc. Header 201A-22"	533'6"	22°	X	X	
22	Drywell HPCI Line 2305-10"	550'	121°	X	X	
25	Drywell Cleanup Line 1201-8"	537'6"	78°	X	X	
27	Drywell Cleanup Line 1201-8"	538'6"	60°	X	X	
29	Drywell Core Spray Line 1404-10"	573'	231°	X	X	
30	Drywell Core Spray Line 1403-10"	561'	336°	X	X	
31	Drywell HPCI Line 2305-10"	563'	140°	X	X	

* Modifications to this table, due to changes in high radiation, should be submitted to the NRC as part of next license amendment request.

TABLE 3.6.1.b

MECHANICAL SNUBBERS

SNUBBER NO.	LOCATION	ELEVATION	AZIMUTH	SNUBBER IN HIGH RADIATION AREA DURING SHUTDOWN	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
32	Drywell Target Rock Valve 203-3A	542'6"	14°	X	X	
33	Drywell Target Rock Valve 203-3A	542'2"	31°	X	X	
34	Drywell Target Rock Valve 203-3A	540'	19°	X	X	
35	Drywell Target Rock Valve 203-3A	540'6"	34°	X	X	
36	Drywell Recirc. Line 3-201B-22"	532'6"	183°	X	X	
37	Drywell Feedwater Line 3-3204D-12"	537'	110°	X	X	
38	Drywell Feedwater Line 3-3204E-12"	538'6"	260°	X	X	
41	Drywell Main Steam Line 3-3001B-20"	534'9"	28°	X	X	
42	Drywell Main Steam Line 3-3001A-20"	534'8"	14°	X	X	
43	Drywell Main Steam Line 3-3001C-20"	534'8"	332°	X	X	
44	Drywell Main Steam Line 3-3001B-20"	542'8"	112°	X	X	
45	Drywell Main Steam Line 3-3001B-20"	543'6"	100°	X	X	
46	Drywell Main Steam Line 3-3001A-20"	543'6"	75°	X	X	
47	Drywell Main Steam Line 3-3001A-20"	544'1"	75°	X	X	
48	Drywell Main Steam Line 3-3001D-20"	542'8"	285°	X	X	
49	Drywell Main Steam Line 3-3001-D-20"	543'6"	285°	X	X	
50	Drywell Main Steam Line 3-3001C-20"	543'6"	255°	X	X	
51	Drywell Main Steam Line 3-3001C-20"	543'6"	255°	X	X	
	Torus					
16	Torus Ring Header 1501-24"	483'	142°			X

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in cold shutdown or refuel condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the cold shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup. This requirement does not apply to Torus Ring Header snubbers for the period identified in paragraph 3.6.I.2. above.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When a snubber is is deemed inoperable, a review of all pertinent facts shall be conducted to determine the snubber mode of failure and to decide if an engineering evaluation should be performed on the supported system or components. If said evaluation is deemed necessary, it will determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

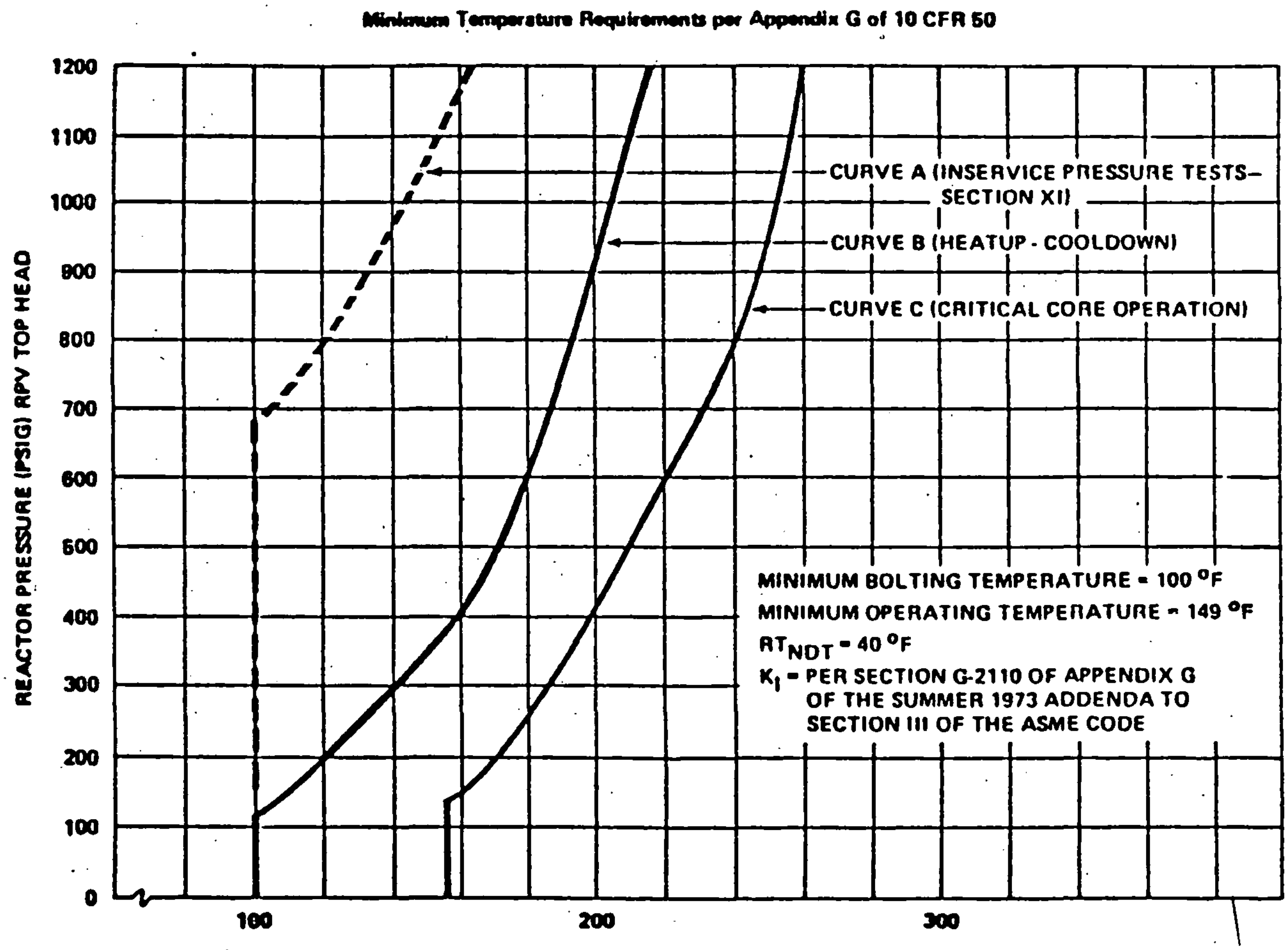
4. If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and, if determined to be a generic deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

5. Snubbers may be added to safety related systems without prior license amendment to Tables 3.6.1a and/or 3.6.1b provided that a revision to Tables 3.6.1a and/or 3.6.1b is included with the next license amendment request.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

5. Snubber service life monitoring shall be followed by existing station record systems, including the central filing system, maintenance files, safety related work packages, and snubber inspection records. The above record retention methods shall be used to prevent the hydraulic snubbers from exceeding a service life of 10 years and the mechanical snubbers from exceeding a service life of 40 years (lifetime of the plant).



TEMPERATURE (°F)

Fig. 3.6.1
MINIMUM TEMPERATURE REQUIREMENTS PER APPENDIX G OF 10 CFR 50

3.6 LIMITING CONDITION FOR OPERATION BASES

A. Thermal Limitations - The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 100°F per hour averaged over a period of one hour. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

Specific analyses were made based on a heating and cooling rate of 100°F/hour applied continuously over a temperature range of 100°F to 550°F. Because of the slow temperature-time response of the massive flanges relative to the adjacent head and shell sections, calculated temperatures obtained were 500°F (shell) and 360°F (flange) (140°F differential). Both axial and radial thermal stresses were considered to act concurrently with full primary loadings. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III stress intensity and fatigue limits even at the flange area where maximum stress occurs.

The flange metal temperature differential of 140°F occurred as a result of sluggish temperature response and the fact that the heating rate continued over a 450°F coolant temperature range.

The uncontrolled cooldown rate of 240°F was based on the maximum expected transient over the lifetime of the reactor vessel. This maximum expected transient is the injection of cold water into the vessel by the high pressure coolant injection subsystem. This transient was considered in the design of the pressure vessel and five such cycles were considered in the design. Detailed stress analyses were conducted to assure that the injection of cold water into the vessel by the HPCI would not exceed ASME stress code limitations.

B. Specification 3.6.A.4 increases margin of safety for thermal-hydraulic stability and startup of recirculation pump.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Pressurization Temperature - The reactor coolant system is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected. These restrictions on inservice hydrostatic testing, on heatup and cooldown, and on critical core operation shown in Figure 3.6.1, were established to be in conformance with Appendix G to 10 CFR 50.

In evaluating the adequacy of ferritic steels Sa302B it is necessary that the following be established:

a) The reference nil-ductility temperature (RT_{NDT}) for all vessel and adjoining materials,

b) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev), and

c) the fluence at the location of a postulated flow.

The initial RT_{NDT} of the main closure flange, the shell and head materials connecting to these flanges, and connecting welds is 10°F. However, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. (Reference Appendix F to the FSAR) The closure flanges and connecting shell materials are not subject to any appreciable neutron radiation exposure, nor are the vertical electroslag seams. The flange area is moderately stressed by tensioning the head bolts. Therefore, as is indicated in curves (a) and (b) of Figure 3.6.1, the minimum temperature of the vessel shell immediately below the vessel flange is established as 100°F below a pressure of 400 psig. (40°F + 60°F, where 40°F is the RT_{NDT} of the electroslag weld and 60°F is a conservatism required by the ASME Code). Above approximately 400 psig pressure, the stresses associated with pressurization are more limiting than the bolting stresses, a fact that is reflected in the non-linear portion of curves (a) and (b). Curve (c), which defines the temperature limitations for critical core operation, was established per Section IV 2.c. of Appendix G of 10CFR50. Each of the curves, (a), (b) and (c) define temperature limitations for unirradiated ferric steels. Provision has been made for the modification of these curves to account for the change in RT_{NDT} as a result of neutron embrittlement.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The withdrawal schedule in Table 4.6.2 is based on the three capsule surveillance program as defined in Section 11.C.3.a of 10 CFR 50 Appendix H. The accelerated capsule (Near Core Top Guide) are not required by Appendix H but will be tested to provide additional information on the vessel material.

This surveillance program conforms to ASTM E 185-73 "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" with one exception. The base metal specimens of the vessel were made with their longitudinal axes parallel to the principal rolling direction of the vessel plate.

C. Coolant Chemistry - A radioactivity concentration limit of 20 Micro-Ci/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Specification 3.8.C.1 or there is a failure or a prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture, outside the drywell, the resultant radiological dose at the site boundary would be about 10 rem to the thyroid. This does was calculated on the basis of a total iodine activity limit of 20 Micro-Ci/ml, meteorology corresponding to Type F conditions with a one meter per second wind speed, and a valve closure time of five seconds. If the valve closed in ten seconds, then the resultant dose would increase to about 25 rem.

The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hours. In addition, the trend of the stack off-gas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam lines.

Materials in the primary system are primarily 304 stainless steel and the Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

the stainless steel. The attached graph, Figure 4.6.2, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolves oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup and hot standby. During these periods with steaming rates less than 100,000 pounds per hour, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Figure 4.6.2 are not exceeded. At steaming rates of at least 100,000 pounds per hour, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt; e.g., Na_2SO_4, which would not have an affect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include, operation of the reactor clean-up system, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the clean-up system to re-establish the purity of the reactor coolant.

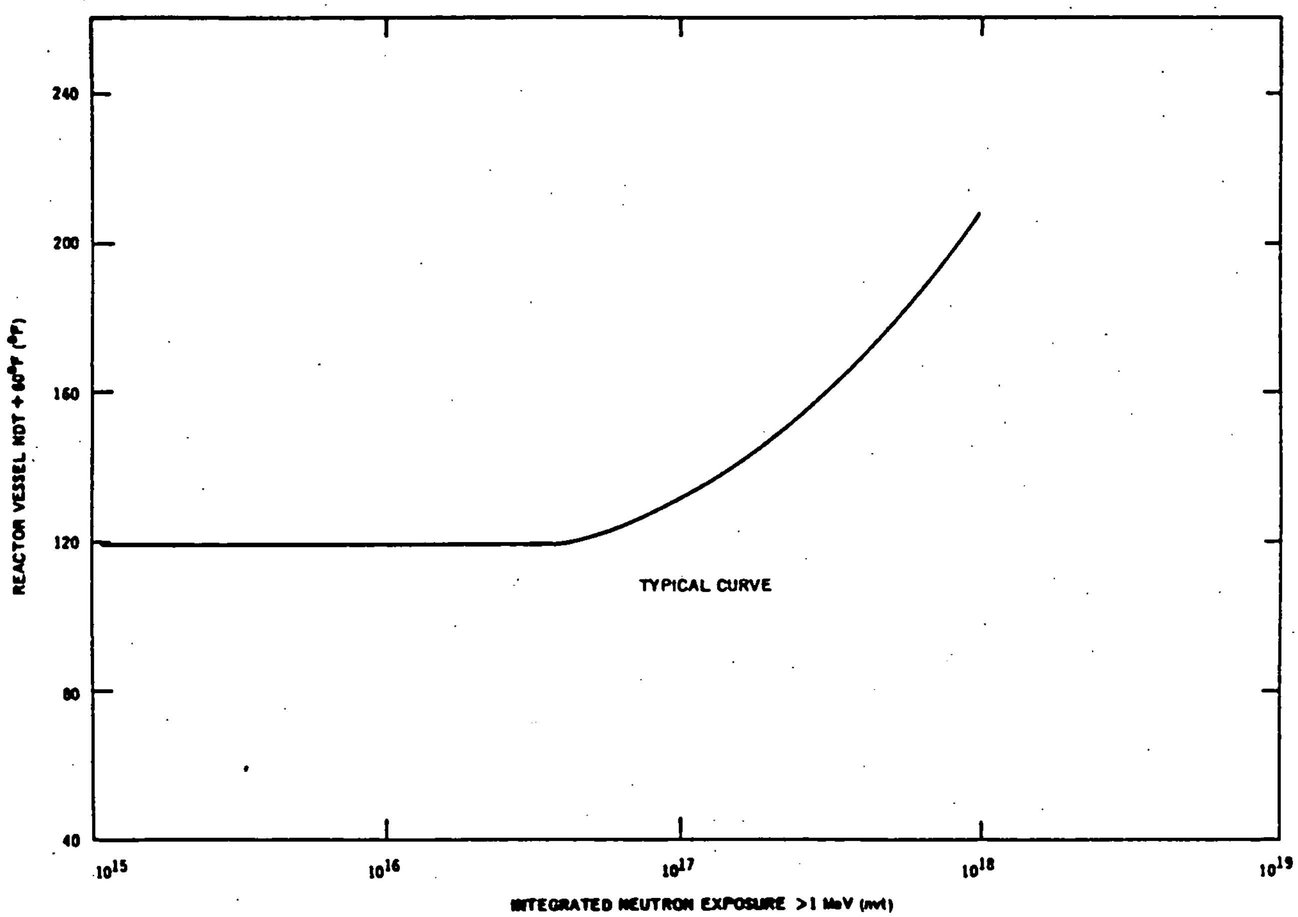


Figure 4.6.1

MINIMUM REACTOR PRESSURIZATION TEMPERATURE

TABLE 4.6.2

NEUTRON FLUX AND SAMPLES WITHDRAWAL
SCHEDULE FOR DRESDEN UNIT 3

Withdrawal Year	Part No.	Location	Comments
1978	16	Near Core Top Guide - 180°	Accelerated
1981	18	Wall - 215°	
2001	19	Wall - 245°	
	15	Wall - 65°	Standby
	20	Wall - 275°	Standby

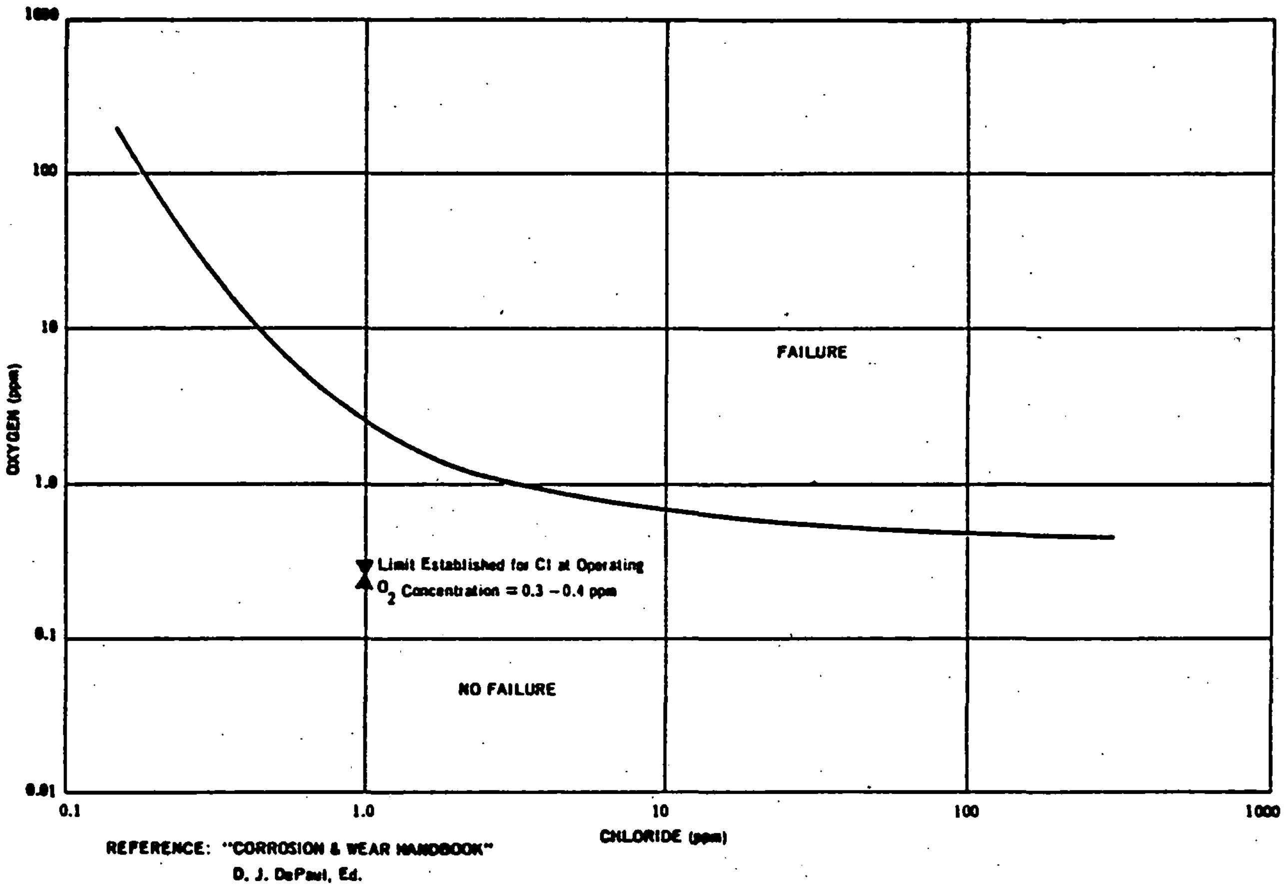


Figure 4.6.2

CHLORIDE STRESS CORROSION TEST RESULTS AT 500°F

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

During start-up periods, which are in the category of less than 100,000 pounds per hour, conductivity may exceed 2 micro-mho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 micro-mho (other than short term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.C.3 may be performed by a gamma scan.

D. Coolant Leakage - Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite a-c power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study 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.D 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.D, 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

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

The capacity of the drywell sump is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

The performance of reactor coolant leakage detection system will be evaluated during the first five years of station operation and the conclusions of this evaluation will be reported to the NRC.

It is estimated that the main steam line tunnel leakage detection system is capable of detecting the order of 3000 lb/hr.

The system performance will be evaluated during the first five years of plant operation and the conclusions of the evaluation will be reported to the NRC.

E. Safety and Relief Valves - The frequency and testing requirements for the safety and relief valves are specified in the Inservice Testing Program which is based on Section XI of the ASME Boiler and Pressure Vessel Code. Adherence to these code requirements provides adequate assurance as to the proper operational readiness of these valves. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as plus or minus 1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher than the reactor coolant pressure safety limit of 1375 psig is not exceeded. The safety valves are required to be operable above the design pressure (90 psig) at which the core spray subsystems are not designed to deliver full flow.

F. Structural integrity - A pre-service inspection of the components in the primary coolant pressure boundary will be conducted after site erection to assure the system is free of gross defects and as a reference base for later inspections. Prior to operation, the reactor primary system will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout life.

Inservice Inspections of ASME Code Class 1, 2 and 3 components will be performed in accordance with the applicable version of Section XI of the ASME Boiler and Pressure Vessel Code. Relief from any of the above requirements must be provided in writing by the Commission. The Inservice Inspection program and the

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

written relief do not form a part of these Technical Specifications.

These studies show that it requires thousands of stress cycles at stresses beyond any expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results only a small number of stress cycles, at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

After five years of operation, a program for in-service inspection of piping and components within the primary pressure boundary which are outside the downstream containment isolation valve shall be submitted to the NRC.

G. Jet Pumps - Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser increases the cross sectional flow area for blowdown following the postulated design basis double-ended recirculation line break. Therefore, if a failure occurs, repairs must be made to assure the validity of the calculated consequences.

The following factors form the basis for the surveillance requirements:

A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Section 4.6.G.1 and 2.

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive or broken jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the twenty individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow. Thus the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing pump. Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operatng map point.

A nozzle-riser system failure could also generate the coincident failure of a 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.

H. Recirculation Pump Flow Mismatch

The LPCI loop selection logic has been described in the Dresden Nuclear Power Station Units 2 and 3 FSAR, Amendments 7 and 8. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions, the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

The licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. Below 80% power, the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

In addition, during the start-up of Dresden Unit 2, it was found that a flow mismatch between the two sets of jet pumps caused by a difference in recirculation loops could set up a vibration until a mismatch in speed of 27% occurred. The 10% and 15% speed mismatch restrictions provide additional margin before a pump vibration problem will occur.

ECCS performance during reactor operation with one recirculation loop out of service has not been analyzed. Therefore, sustained reactor operation under such conditions is not permitted.

I. Snubbers (Shock Suppressors)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.I.4 prohibits startup with inoperable snubbers.

When a snubber is found inoperable, a review shall be performed to determine the snubber mode of failure. Results of the review shall be used to determine if an engineering evaluation of the safety-related system or component is necessary. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the support component or system.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

All safety related mechanical snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation and attachments to the piping and anchor for indication of damage or impaired operability.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To further increase the assurance of snubber reliability, functional tests will be performed once each refueling cycle. A representative sample of 10% of the safety-related snubbers will be functionally tested. Observed failures on these samples will require testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as different entities for the above surveillance programs.

Hydraulic snubber testing will include stroking of the snubbers to verify piston movement, lock-up, and bleed. Functional testing of the mechanical snubbers will consist of verification that the force that initiates free movement of the snubber in either tension or compression is less than the maximum breakaway friction force. The remaining portion of the functional test consisting of verification that the activation (restraining action) is achieved within the specified range of acceleration in both tension and compression will not be done. This is due to the lack of competitive marketable test fixtures available for station use. Therefore, until such time as test fixtures become available, only part (i) of the test will be performed; part (ii) will not be done.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

When the cause of rejection of the snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

Monitoring of snubber service life shall consist of the existing station record systems, including the central filing system, maintenance files, safety-related work packages, and snubber inspection records. The record retention programs employed at the station shall allow station personnel to maintain snubber integrity. The service life for hydraulic snubbers is 10 years. The hydraulic snubbers existing locations do not impose undue safety implications on the piping and components because they are not exposed to excesses in environmental conditions. The service life for mechanical snubbers is 40 years, lifetime of the plant. The mechanical snubbers are installed in areas of harsh environmental conditions because of their dependability over hydraulic snubbers in these areas. All snubber installations have been thoroughly engineered providing the necessary safety requirements. Evaluations of all snubber locations and environmental conditions justify the above conservative snubber service lives.

A re-analysis of the ring header design based upon acceleration response spectra derived from the original suction header analysis report demonstrates that for normal operation plus seismic, neither the header nor the torus penetration are over-stressed with all snubbers inoperable. The limitation of a maximum of 3 pairs or 1 snubber from each pair inoperable out of 6 pairs is considered conservative. Since the analysis shows that the plant can operate safely indefinitely with no snubbers on the ring header the limitation on operation and startup with inoperable snubbers until January 19, 1984 is justified. This time frame is adequate to allow completion of the Mark I torus attached piping modification.

4.6 SURVEILLANCE REQUIREMENT BASES

None

3.7 LIMITING CONDITION FOR OPERATION

CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, except as permitted by Specification 3.5.F.3, 3.5.F.4, or 3.5.F.5, the suppression pool water volume and temperature shall be maintained within the following limits.

    a. Maximum water volume - 115,655 ft^3

    b. Minimum water volume - 112,000 ft^3

4.7 SURVEILLANCE REQUIREMENTS

CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. The surveillances are as follows:

    a. The suppression pool water level and temperature shall be checked once per day.

    b. Whenever there is indication of relief valve operation or

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

c. Maximum water temperature

(1) During normal power operation; maximum 95°F.

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10° F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 150 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F.

Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 150 psig at normal cooldown rates if the pool temperature reaches 120°F.

d. Maximum downcomer submergence is 4.00 ft.

e. Minimum downcomer submergence is 3.67 ft.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

f. If Specifications 3.7.A.1.a or 3.7.A.1.b are not met and suppression pool water volume cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MW(t).

a. Primary containment leakage rates are defined from:

(1) The calculated peak containment internal pressure, P_a, is equal to 48 psig.

(2) The containment vessel reduced test pressure, P_t, is equal to 25 psig.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

2. The primary containment integrity shall be demonstrated by conducting Primary Containment Leak Tests and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and references therein.

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at approximately equal intervals during each 10 year plant in-service inspection interval at either P_a or P_t with the last being done during the 10-year in-service inspection shutdown.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(3) The maximum allowable leakage rate at a pressure of P_a, L_a, is equal to 1.6 percent by weight of the containment air per 24 hours at 48 psig.

(4) The maximum allowable test leakage rate at a pressure of P_t, L_t, is less than or equal to L_a (L_{tm}/L_{am}). If L_{tm}/L_{am} is greater than 0.7, L_t is (specified as equal to) L_a (P_t/P_a) E 1/2.

(5) The total measured leakage rates at pressures of P_a and P_t are L_{am} and L_{tm}, respectively.

b. When primary containment integrity is required, primary containment leakage rates shall be limited to:

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. If any periodic Type A test fails to meet either 75 percent of L_a or 75 percent of L_t the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(1) An overall integrated leakage rate for Type A tests of:

(a) L_{am} less than or equal to 75 percent of L_a.

(b) L_{tm} less than or equal to 75 percent of L_t.

(2) (a) A combined leakage rate of less than or equal to 60 percent of L_a for all testable penetrations and isolation valves subject to Type B and C tests except for main steam isolation valves.

(b) A leakage rate of less than or equal to 3.75 percent of L_a for any one air lock when pressurized to 10 psig.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(c) 11.5 SCF per hour for any main steam isolation valve at a test pressure of 25 psig.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

c. If two consecutive Type A tests fail to meet either 75 percent of L_a or 75 percent of L_t, a Type A test shall be performed at each shutdown for refueling or approximately every 18 months until two consecutive Type A tests meet the above requirements, at which time the normal test schedule may be resumed.

d. The accuracy of each Type A test shall be verified by a supplemental test which:

(1) Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 25 percent of L_a or 25 percent of L_t.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

(2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

(3) Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 9 scfm.

e. Type B and C tests shall be conducted at P_a, at intervals no greater than 24 months except for tests involving:

(1) Main steam line isolation valves which shall be tested at a pressure of 25 psig each operating cycle.

(2) Bolted double-gasketed seals which shall be tested at a pressure of 48 psig

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

whenever the seal is closed after being opened and each operating cycle.

(3) Air locks which shall be tested at 10 psig each operating cycle.

f. Continuous Leak Rate Monitor

(1) When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system make-up requirements.

(2) This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

g. The interior surfaces of the drywell shall be visually inspected each operating cycle for evidence of deterioration.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in Specifications 3.7.A.3.b below, two pressure suppression chamber - reactor building vacuum breakers in each line shall be operable at all times when the primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber - reactor building air operated vacuum breakers shall not exceed 0.5 psid. The vacuum breakers shall open fully when subjected to a force equivalent to or less than 0.5 psid acting on the valve disk.

b. From and after the date that one of the pressure suppression chamber - reactor building vacuum breakers is made

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. The pressure suppression chamber - reactor building vacuum breakers and associated instrumentation, including setpoint, 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

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

or found to be inoperable for any reason, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the procedure does not violate primary containment integrity.

4. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. When primary containment is required, all pressure suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated in Specifications 3.7.A.4.b, c and d., below, pressure suppression chamber - drywell vacuum breakers shall be considered operable if:

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

breaker does not exceed the force specified in Specification 3.7.A.3.a. and each vacuum breaker shall be inspected and verified to meet design requirements.

4. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. Periodic Operability Tests

      Once each month each pressure suppression chamber - drywell vacuum breaker shall be exercised. Operability of position switches and position indicators and alarms shall be verified.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(1) The valve is demonstrated to open fully with the applied force at all valve positions not exceeding the equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

(2) The valve can be closed by gravity when released after being opened by manual means, to within the equivalent of 1/16" at all points along the seal surface of the disk.

(3) The position alarm system will annunciate in the control room if the valve opening exceeds the equivalent of 1/16" at all points along the seal surface of the disk.

b. Reactor operation may continue provided that no more than one

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. During each refueling outage:

(1) The pressure

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

quarter of the number of pressure suppression chamber - drywell vacuum breakers are determined to be inoperable provided that they are secured or known to be in the closed position.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open.

(2) Vacuum breakers position indication and alarm systems shall be calibrated and functionally tested.

(3) At least 25% of the vacuum breakers shall be inspected such that all vacuum breakers shall have been inspected following every fourth refueling outage. If deficiencies are found, all vacuum breakers shall be inspected and deficiencies corrected.

(4) A drywell to suppression chamber leak test shall

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

c. Reactor operation may continue for fifteen (15) days provided that at least one position alarm circuit for each operable vacuum breaker is operable and each suppression chamber - drywell vacuum breaker is physically verified to be closed immediately and daily thereafter.

5. Oxygen Concentration

a. The primary containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

demonstrate that with initial differential pressure of not less than 1.0 psi, the differential pressure decay rate does not exceed the rate which would occur through a 1-inch orifice without the addition of air or nitrogen.

5. Oxygen Concentration

The primary containment oxygen concentration shall be measured and recorded on a weekly basis.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

cooling pressure above 90 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. Deinerting may commence 24 hours prior to a shutdown.

6. Containment Atmospheric Dilution and Purge

a. Whenever the reactor is in power operation the normal containment makeup inerting system shall be operable and capable of supplying nitrogen to containment for atmosphere dilution if required by post LOCA conditions. If this specification cannot be met, the system must be restored to an operable condition within 7 days or

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

6. Containment Atmospheric Dilution and Purge

a. Once a month, the valves in the nitrogen makeup system shall be actuated to determine operability.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

the reactor must be taken out of power operation.

b. Whenever either Unit 2 or 3 is in power operation, the containment makeup inerting system nitrogen storage tank level liquid level shall be equal to or greater than 60 inches. If this minimum level cannot be met, the minimum level shall be restored within 7 days or both Unit 2 and Unit 3 shall be taken out of power operation. During such seven day interval the minimum level shall be 20 inches or both Unit 2 and 3 shall be taken out of power operation.

c. Whenever the reactor is in power operation, the primary containment purge system shall be operable. If this specification cannot be met the reactor must be taken out of power operation.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. The level in the liquid N_2 storage tank shall be recorded weekly and after reinerting containment.

c. Once a month , the valves in the purge line to the standby gas treatment system shall be actuated to determine operability.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

d. Whenever the reactor is in power operation, the primary containment oxygen sampling system shall be operable. If this specification cannot be met, the system must be restored to an operable condition within 7 days or the reactor must be taken out of power operation.

e. The maximum containment repressurization pressure using the containment makeup inerting system shall be 26 psig.

7. Drywell Suppression Chamber Differential Pressure

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.00 psid except as specified in (1) and (2) below:

(1) This differential shall be established within the 24

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

d. The containment oxygen analyzing system shall be functionally tested once per week and shall be calibrated once per 6 months.

7. Drywell Suppression Chamber Differential Pressure

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift when the differential pressure is required.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

hour period subsequent to placing the reactor mode switch into the run mode during a startup and may be relaxed 24 hours prior to a rector shutdown when the provisions of 3.7.A.5 (b) apply.

(2) This differential may be decreased to less than 1.00 psid for a maximum of 4 hours during required operability testing of the drywell pressure suppression chamber vacuum breakers, HPCI testing and reactor pressure relief valve testing.

b. If the Specifications of 3.7.A.7.a cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

shall be initiated and the rector shall be in a cold shutdown condition in the following 24 hours.

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment system circuits shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1(a) and (b).

   a. After one of the standby gas treatment system circuits is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment system shall be demonstrated to be operable within 2 hours

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

B. Standby Gas Treatment System

1. At least once per month, initiate from the control room 4000 cfm (plus or minus 10%) flow through both circuits of the standby gas treatment system for at least 10 hours with the circuit heaters operating at rated power.

   a. Within 2 hours from the time that one standby gas treatment system circuit is made or found to be inoperabe for any reason and daily thereafter for the next succeeding seven days, initiate from the control room 4000 cfm (plus or minus 10%) flow through the operable circuit of the standby gas treatment system for at least 10 hours

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

and daily thereafter. Within 36 hours following the 7 days, the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.1.(a) through (d).

b. If both standby gas treatment system circuits are not operable, within 36 hours the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.1.(a) through (d).

2. Performance Requirement (See Note 1, Page 3/4.7-24)

a. Periodic Requirements:

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

with the circuit heaters operating.

2. Performance Requirement Tests (See Note 1)

a. At least once per 720 hours of system operation; or once per operating cycle, but not to exceed 18 months, whichever occurs first; or following painting, fire, or chemical

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(1) The results of the in-place DOP tests at 4000 cfm (plus or minus 10%) on HEPA filters shall show less than or equal to 1% DOP penetration.

(2) The results of in-place halogenated hydrocarbon tests at 4000 cfm (plus or minus 10%) on charcoal banks shall show less than or equal to 1% penetration.

(3) The results of laboratory carbon sample analysis shall show greater than or equal to 90% methyl iodide removal efficiency when tested at 130°C, 95% R. H.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

release in any ventilation zone communicating with the system while the system is operating that could contaminate the HEPA filters or charcoal absorbers; perform the following:

(1) In-place DOP test the HEPA filter banks to verify leak tight integrity

(2) In-place test the charcoal adsorber banks with halogenated hydrocarbon tracer to verify leak tight integrity.

(3) Remove one carbon test canister from the charcoal adsorber. Subject this sample to a laboratory analysis to verify methyl iodide removal efficiency.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

b. The system shall be shown to operate.

3. Post Maintenance Requirements (See Note 1; Page 3/4.7-24)

a. After any maintenance or testing that could affect the HEPA filter or HEPA filter mounting frame leak tight integrity, the results of the inplace DOP tests at 4000 cfm (plus or minus 10%) on HEPA filters shall show less than or equal to 1% DOP penetration in

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. At least once per operating cycle, but not to exceed 18 months, the following conditions shall be demonstrated:

(1) Pressure drop across the combined filters of each standby gas treatment system circuit is less than 6 inches of water at 4000 cfm (plus or minus 10%) flow rate.

(2) Operability of inlet heater at rated power.

(3) Automatic initiation of each standby gas treatment system circuit.

3. Post Maintenance Testing (See Note 1)

a. After any maintenance or testing that could affect the leak tight integrity of the HEPA filters, perform in-place DOP tests on the HEPA filters in accordance with Specification 3.7.B.2.a. (1).

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

accordance with Specification 3.7.B.2.a(1).

b. After any maintenance or testing that could affect the charcoal adsorber leak tight integrity, the results of in-place halogenated hydrocarbon tests at 4000 cfm (plus or minus 10%) on charcoal adsorber banks shall show less than or equal to 1% penetration in accordance with Specification 3.7.B.2.a(2).

c. The results of in-place air distribution tests shall show the air distribution is uniform within plus or minus 20% to each HEPA filter when tested initially and after any maintenance or testing that could affect the air distribution within the standby gas treatment system.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. After any maintenance or testing that could affect the leak tight integrity of the charcoal adsorber banks, perform halogenated hydrocarbon tests on the charcoal adsorbers in accordance with Specification 3.7.B.2.a.(2).

c. Perform an air distribution test on the HEPA filter bank initially and after any maintenance or testing that could affect the air distribution within the standby gas treatment system. The test shall be performed at 4000 cfm (plus or minus 10%) flow rate.

4. Standby gas treatment system surveillance shall be performed as indicated below:

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

Note 1: Because the accomplishment of Specifications 3.7.B.2, 3.7.B.3, 4.7.B.2, and 4.7.B.3 will require equipment modifications, their implementation will be delayed until about December 31, 1976. Until that time, the surveillance requirements of Specification 4.7.B.4 shall apply.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

a. At least once per operating cycle it shall be demonstrated that:

(1) Pressure drop across the combined high-efficiency and charcoal filters is less than 5.7 inches of water at 400 cfm and

(2) Inlet heater delta T shall be a minimum of 14°F at 4000 cfm.

b. At least once during each scheduled secondary containment leak rate test, whenever a filter is changed, whenever work is performed that could affect the filter system efficiency and at intervals not to exceed six months between refueling outages, it shall be demonstrated that

(1) The removal efficiency of the particulate filters is not less than 99% for particulate matter larger than 0.3 micron.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. Secondary Containment

1. Secondary containment integrity shall be maintained during all modes of plant operation except when all of the following conditions are met.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

(2) The removal efficiency of the charcoal filters is not less than 99% for iodine.

c. At least once each five years removable charcoal cartridges shall be removed and absorption shall be demonstrated.

d. At least once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.

e. At least once per operating cycle manual operability of the bypass valve for filter cooling shall be demonstrated.

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

3.7 LIMITING CONDITION FOR OPERATION (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.

4.7 SURVEILLANCE REQUIREMENTS (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 a 1/4 inch of water vacuum under calm wind (less than 5 mph) conditions with a filter train flow rate of not more than 4000 cfm.

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 a 1/4 inch of water vacuum under calm wind (less than 5 mph) conditions with a filter train flow rate of not more than 4000 cfm, shall be demonstrated at each refueling outage prior to refueling.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

d. The fuel cask or irradiated fuel is not being moved in the reactor building.

2. The doors of the core spray and LPCI pump compartments shall be closed at all times except during passage in order to consider the core spray and the LPCI subsystems operable.

3. If Specification 3.7.C.1 cannot be met procedures shall be initiated to establish conditions listed in Specification 3.7.C.1.a through d.

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 SURVEILLANCE REQUIREMENTS (Cont'd.)

2. Whenever the LPCI and core spray subsystems are required to be operable, the doors of the core spray and LPCI pump compartments shall be verified to be closed weekly.

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.

b. At least once per operating cycle the instrument line flow check valves shall be tested for proper operation.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. In the event any isolation valve specified in Table 3.7.1 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

c. At least once per quarter:

(1) All normally open power-operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened.

(2) With the reactor power less than 50% of rated, trip main steam isolation valves (one at a time) and verify closure time.

d. At least twice per week the main steamline power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

2. Whenever an isolation valve listed in Table 3.7.1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

4. The temperature of the main steamline air pilot valves shall be less than 170°F except as specified in 3.7.D.5 below.

5. From and after the date that the temperature of any main steamline air pilot valve is found to be greater than 170°F, reactor operation is permissible only during the succeeding seven days unless the temperature of such valve is sooner reduced to less than 170°F, provided the main steamline isolation valves are operable.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

3. The temperature of the main steamline air pilot valves shall be recorded daily.

4. When it is determined that the temperature of any main steamline air pilot valve is greater than 170°F, the main steamline isolation valves shall be demonstrated to be operable immediately and daily thereafter. The demonstration of operability shall be according to Specification 4.7.D.1.d.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

6. When it is determined that it will take longer than seven days to reduce the temperature of any main steamline air pilot valve to less than 170°F, a report detailing the circumstances and the estimated date for returning the air pilot valve temperature to a value less than 170°F shall be submitted to the NRC prior to the end of the seven day period.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

TABLE 3.7.1

PRIMARY CONTAINMENT ISOLATION

Isolation Group	Valve Identification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
1	Main Steam Line Isolation	4	4	3 * T * 5	O	GC
1	Main Steam Line Drain	1		* 35	C	SC
1	Main Steam Line Drain		1	* 35	C	SC
Note 1 1	Recirculation Loop Sample Line	1	1	* 5	O	SC
1	Isolation Condenser Vent to main steam line	1		* 5	O	GC
1	Isolation Condenser Vent to main steam line		1	* 5	O	GC
2	Drywell floor drain		2	* 20	O	GC
2	Drywell Equipment drain		2	* 20	O	GC
2	Drywell Vent		2	* 10	C	SC
2	Drywell Vent Relief		1	* 15	C	SC
2	Drywell Inert and purge #1601-21		1	* 10	C	SC
2	Drywell N_2 Makeup #1601-59	1		* 15	O	GC
2	Drywell and Suppression Chamber N_2 Makeup #1601-57		1	* 15	O	GC
2	Drywell and Suppression Chamber Inert #1601-55		1	* 15	O	GC
2	Suppression Chamber N_2 Makeup #1601-58		1	* 15	C	SC
2	Suppression Chamber inert and purge #1601-56		1	* 10	O	GC
2	Drywell and Suppression chamber vent from reactor building #1601-22		1	* 10	C	SC
2	Drywell vent to standby gas treatment system		1	* 10	C	SC
2	Suppression chamber vent		1	* 10	C	SC
2	Suppression chamber vent relief		1	* 15	C	SC
Note 1 2	Drywell air sampling system		10	* 5	O	GC
2	Drywell Pneumatic Supply Isolation		2	* 10	O	GC
2	Torus to Condenser Drain		2	* 10	C	SC
3	Cleanup demineralizer System	1		* 30	O	GC
3	Cleanup demineralizer System		2	* 30	O	GC
3	Shutdown cooling system	2		* 40	C	SC
3	Shutdown cooling system		1	* 40	C	SC
3	Shutdown cooling system		1	* 40	C	SC
3	Reactor head cooling line		1	* 15	C	SC
4	HPCI Turbine Steam supply	1		* 25	O	GC
4	HPCI Turbine Steam supply		1	* 25	O	GC
5	Isolation condenser steam supply	1		* 30	O	GC
5	Isolation condenser steam supply		1	* 30	O	GC
5	Isolation condenser condensate return	1		* 30	O	GC
5	Isolation condenser condensate return		1	* 30	C	SC
	Feedwater Check Valves	2	2	NA	O	Process
	Control Rod Hydraulic Return Check Valves	1	1	NA	O	Process
	Reactor Head Cooling Check Valves	1		NA	C	Process
	Standby Liquid Control Check Valves	1	1	NA	C	Process

Notes: (See next page)

Notes for Table 3.7.1

* Less than or equal to

Note 1; Valve can be reopened after isolation for sampling.

Key: O = Open
C = Closed
SC = Stays Closed
GC = Goes Closed

Note: Isolation groupings are as follows:

GROUP 1: The valves in Group 1 are closed upon any one of the following conditions:

1. Reactor low-low water level
2. Main steam line high radiation
3. Main steam line high flow
4. Main steam line tunnel high temperature
5. Main steam line low pressure

GROUP 2: The actions in Group 2 are initiated by any one of the following conditions:

1. Reactor low water level
2. High drywell pressure

GROUP 3: Reactor low water level alone initiates the following:

1. Cleanup demineralizer system isolation
2. Shutdown cooling system isolation
3. Reactor Head cooling isolation

GROUP 4: Isolation valves in the high pressure coolant injection system (HPCI) are closed upon any one of the following signals:

1. HPCI steam line high flow
2. High temperature in the vicinity of the HPCI steam line
3. Low reactor pressure

GROUP 5: Isolation valves associated with the isolation condenser are closed upon indication of either high isolation condenser steam or condensate flow.

3.7 LIMITING CONDITION FOR OPERATION BASES

A. Primary Containment - The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to preclude a peak fuel enthalpy of 280 cal/gm. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offers a significant barrier to keep off-site doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. (Ref. Section 5.2.3 FSAR)

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 48 psig which is below the design of 62 psig. Maximum water volume of 115,655 ft^3 results in a

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

downcomer submergence of 4 feet and the minimum volume of 112,000 ft^3 results in a submergence approximately 4 inches less. The majority of the Bodega tests [9] were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

The maximum temperature at the end of blowdown tested during the Humboldt Bay [10] and Bodega Bay tests were 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for emergency core cooling systems operability as explained in basis 3.5.F.

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(9) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(10) Robbins, C. H., "Tests of a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Using a 50°F rise (Section 5.2.3.1 SAR) in the suppression chamber water temperature and a maximum initial temperature of 95°F, a temperature of 145°F is achieved which is well below the 170°F temperature which is used for complete condensation.

For an initial maximum suppression chamber water temperature of 95°F and assuming the normal complement of containment cooling pumps (2 LPCI pumps and 2 containment cooling service water pumps) containment pressure is not required to maintain adequate net positive suction head (NPSH) for the core spray, LPCI and HPCI pumps.

If a loss of coolant accident were to occur when the reactor water temperature is below 330°F, the containment pressure will not exceed the 62 psig design pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperatures above 212°F provides additional margin above that available at 330°F.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber 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% flow pipes each containing two vacuum relief breakers. Operation of either flow pipe will maintain the pressure differential less than 1 psig, the external design pressure of the primary containment. Redundancy of lines justifies reactor operation with one valve out of service for repairs for a period of seven days.

The capacity of the pressure suppression chamber-drywell vacuum breakers is designed to limit the pressure differential between the suppression chamber and drywell to not greater than 0.5 psi during post-accident drywell cooling. They are sized on the basis of the Bodega Bay pressure suppression system test.

Based on these tests, design flow from the suppression chamber to the drywell can be obtained with three (3) of the vacuum breakers closed without exceeding the 0.5 psi differential pressure limit.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Reactor operation is permissible if the bypass area between the primary containment drywell and suppression chamber does not exceed an allowable area. The allowable bypass area is based upon analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is equivalent to all vacuum breakers open the equivalent of 1/16" at all points along the seal surface of the disk (see Dresden Special Report No. 23).

Each drywell-suppression chamber vacuum breaker is fitted with a redundant pair of position switches which provide signals of disk position to panel mounted indicators and annunciate an alarm in the control room if the disk is open more than allowable. The alarm systems meet the intent of IEEE 279 standards. The quality of the alarm system justifies continued reactor operation for 15 days between differential pressure decay rate tests if one alarm system is inoerable for one or more operable vacuum breakers.

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss of coolant accident would lead to the liberation of sufficient hydrogen to a result in a flammable concentration in the containment. Subsequent ignition of the hydrogen if it is present in sufficient quantities to result in excessively rapid recombination, could lead to failure of the containment to maintain a low leakage integrity. The 4% oxygen concentration minimizes the possibility of hydrogen combustion following a loss of coolant accident.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss of coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

In order to ensure that the containment atmosphere remains inerted, i.e. the oxygen-hydrogen mixture remains below the flammable limit, the capability to inject nitrogen into the containment after a LOCA is provided. During an interim period prior to installation of the Containment Atmospheric Dilution (CAD) system the normal inerting nitrogen makeup system will be available for post-LOCA nitrogen injection.

By maintaining a minimum level of 60 inches in the liquid nitrogen storage tank, a minimum of 200,000 cubic feet of nitrogen is assured which corresponds to a seven day supply. During reinerting of containment the supply may temporarily drop below a seven day supply but at no time is the inventory to drop below a minimum of a two day supply (20 inch level). By normally maintaining at least a 7-day supply of nitrogen on site and maintaining a minimum of a two day supply there will be assurance of sufficient time after the occurrence of a LOCA for obtaining additional nitrogen supply.

A system for controlled purging through the Standby Gas Treatment system is necessary to limit repressurization pressure from post LOCA nitrogen addition in a manner which will limit offsite doses. Controlled purging also provides a backup method of controlling hydrogen concentration.

A means to determine post LOCA containment oxygen concentration is necessary to readily enable the reactor operator to take appropriate action to control containment atmosphere. In the interim, prior to installation of the CAD and associated monitoring systems, the containment oxygen analyzing system will be available.

The maximum containment repressurization pressure of 26 psi provides adequate margin to containment design pressure and a delay time prior to purge which results in acceptable purge doses.

Following a LOCA, periodic operation of the drywell and torus sprays will be used to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and O_2 concentration exceeds 4%.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed (Reference 15) which

(15) "Dresden Nuclear Generating Plant Units 2 & 3 Short Term Program Plant Unique Torus Support and Attached Piping Analysis", August 1976 NUTECH Report COM-01-040.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.00 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.67 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

B. Standby Gas Treatment System and
C. Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

Only one of the two standby gas treatment system circuits 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. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is placed in a condition that does not require a standby gas treatment system.

While only a small amount of particulates are released from the primary containment as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal 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% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed).*

*Bases in parentheses will not be applicable until about December 31, 1976, when equipment modifications are completed to allow increased testing.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. One standby gas treatment fan is designed to automatically start upon containment isolation and to maintain the reactor building pressure to approximately a negative 1/4-inch water guage pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter system is designed to start automatically. Each of the two fans has 200% capacity. (Ref. Section 5.3.2 SAR.) If one standby gas treatment system circuit is inoperable, the other circuit will be tested daily. This substantiates the availability of the operable circuit and results in no added risk; thus, reactor operation or refueling operation can continue. If neither circuit is operable the plant is brought to a condition where the system is not required.

While only a small amount of particulates are released from the pressure suppression chamber system as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The inplace test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates. Laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

D. Primary Containment Isolation Valves - Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

4.7 SURVEILLANCE REQUIREMENT BASES

A. Primary Containment

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppres- sion pool temperature to be continually monitored and fre- quently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The primary containment preoperational 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 48 psig which would rapidly reduce to 25 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 25 psig within 10 seconds, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay(12).

The design pressure of the drywell and absorption chamber is 62 psig(12). The design leak rate is 0.5%/day at a pressure of 62 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were 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 2.0%/day at 48 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID 14844, the maximum total whole body passing cloud dose is about 8 rem and the maximum total thyroid dose is about 185 rem at the site boundary over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population distance of 5 miles are lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, the 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.

(12) Section 5.2 of the FSAR

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The maximum allowable test leak rate is 1.6%/day at a pressure of 48 psig. This value for the test condition was derived from the maximum allowable accident leak rate of about 2.0%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air whereas under test conditions the test medium would be air or nitrogen at ambient conditions. Considering the difference in mixture composition and temperatures, the appropriate correction factor applied was 0.8 and determined from the guide on containment testing(13).

Although the dose calculations suggest that the accident leak rate could be allowed to increase to about 3.2%/day before the guideline thyroid doses value given in 10 CFR 100 would be exceeded, establishing the test limit of 1.6%/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 or the allowable test leak rate 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 frequency is based on the AEC guide for developing leak rate testing and surveillance of reactor containment vessels(14). 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.

(13) TID 20583, Leakage Characteristics of Steel Containment Vessel and the Analysis of Leakage Rate Determinations.

(14) Technical Safety Guide, "Reactor Containment Leakage Testing and Surveillance Requirements USAEC, Division of Safety Standards, Revised Draft, December 15, 1966.

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The data reduction methods of the applicable ANSI standard will be applied for the integrated leak rate tests as specified in Appendix J of 10 CFR 50.

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 double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 48 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure. 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. The personnel air lock is tested at 10 psig, because the inboard door is not designed to shut in the opposite direction.

The results of the loss-of-coolant accident analyses presented in Amendment No. 18 of the SAR indicates that fission products would not be released directly to the environs because of leakage from the main steam line isolation valves due to holdup in the steam system complex. Although this effect would indicate that an adequate margin exists with regard to the release of fission products, a program will be undertaken to further reduce the potential for such leakage to bypass the standby gas treatment system.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

Surveillance of the reactor building-pressure suppression chamber vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness test). These vacuum breakers are normally in the closed position and open only during tests or a post accident condition. As a result, a testing frequency of 3 months for operability is considered justified for this equipment. Inspections and calibrations are performed during refueling outages, this frequency being based on experience and judgment.

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

Pressure suppression chamber-drywell vacuum breakers monthly operability tests are performed to check capability of the disks to open and close and to verify that the position indication and alarm circuits function properly. The disk opens during post accident conditions and occasionally during transient additions of energy to the torus through relief valves. This infrequent operation of the disks and the quality of equipment justify the frequency of operability tests of this equipment.

Measurement of force to open, calibration of position switches, inspection of equipment and functional testing are performed during each refueling outage. This frequency is based on equipment quality, experience and judgment. Also a stringent differential pressure decay rate test is performed during refueling outages. This test is performed to verify that total leakage paths between the drywell and suppression chamber are not in excess of the equivalent to a 1-inch orifice.

This small leakage path is only a small fraction of the allowable, thus integrity of the containment system is assured prior to startup following each refueling outage (See Dresden Special Report No. 23).

When a suppression chamber-drywell vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights at the remote test panel are designed to function as follows:

Full Closed (Closed to less than or equal to 1/16" open)	2 Green - On
Intermediate Position (greater than 1/16" open to full open)	2 Green - Off

The remote test panel consists of two green lights for each of the twelve valves. The two switches controlling the green lights are adjusted to provide indication and alarm if a disk opening occurs that is equivalent to one-sixteenth of an inch (1/16") at all points around the circumference of the valve disk. The control room alarm circuits for each vacuum breaker are redundant and fail safe. This assures that no single failure will defeat alarming the control room when a valve is open beyond allowable and when power to the switches fails. The alarm is needed to alert the operator that action must be taken to correct a malfunction or that system degradation has

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

occurred and additional testing is required immediately. The frequency of testing the alarms is based on experience and quality of the equipment. During each refueling outage, three drywell-suppression chamber vacuum breakers will be inspected to assure sealing surfaces and components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in 1/10 of the design lifetime is extremely conservative.

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 once a week the oxygen concentration will be determined as added assurance.

Recording N_2 storage tank level weekly and after containment reinerting provides assurance of an adequate onsite supply.

Weekly testing of the oxygen analyzer and monthly actuation of the nitrogen makeup and purge line valves provides assurance of operational readiness.

B. Standby Gas Treatment System and
C. Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain the design negative pressure within the secondary containment provides an adequate test of the reactor building isolation valves and the standby gas treatment system. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system operational capability. (The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Standby gas treatment system in-place testing procedures will be established utilizing applicable sections of ANSI N510-1975 standard as a procedural guideline only. Operation of the standby gas treatment system every month for 10 hours will reduce the moisture buildup on the adsorbent. If painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

analysis should be performed as required for operational use. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52, Revision 1 (June 1976). The charcoal adsorber efficiency test procedures will allow for the removal of one representative sample cartridge and testing in accordance with the guidelines of Table 3 of Regulatory Guide 1.52, Revision 1 (June 1976). The sample will be at least two inches in diameter and a length equal to the thickness of the bed. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system will be replaced. High efficiency particulate filters are installed before and after the charcoal filters to prevent clogging of the carbon adsorbers and to minimize potential release of particulates to the environment. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by in-place testing with DOP as the testing medium. Any HEPA filters found defective will be replaced with filters qualified pursuant to regulatory guide position C.3.d of Regulatory Guide 1.52, Revision 1 (June 1976). Once per operating cycle demonstration of HEPA filter pressure drop, operability of inlet heaters at rated power, air distribution to each HEPA filter, and automatic initiation of each standby gas treatment system circuit is necessary to assure system performance capability).*

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein are adequate to prevent loss of more coolant from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, this isolation valve closure time is sufficient to prevent uncovering the core.

* Bases in parentheses will not be applicable until about December 31, 1976, when equipment modificatons are completed to allow increased testing.

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds. However, for added margin the Technical Specifications require a valve closure time of not greater than 5 seconds.

For reactor coolant system temperature less than 212°F, the containment could not become pressurized due to a loss of coolant accident. The 212°F limit is based on preventing pressurization of the reactor building and rupture of the blowout panels. These valves are highly reliable, have low service requirement and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. (Ref. Section 5.2.2 and Table 5.2.4 SAR.) The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a more reliable system.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The containment is penetrated by a large number of small diameter instrument lines. A program for periodic testing and examination of the floor check valves in these lines is performed similar to that described in Amendment No. 22, Millstone Unit 1, Dkt. 50-245.

3.8 LIMITING CONDITION FOR OPERATION

RADIOACTIVE MATERIALS

Applicability:

Applies to the radioactive effluents from the plant.

Objective:

To assure that radioactive material is not released to the environment in an uncontrolled manner and to assure that any material released is kept as low as practicable and, in any event, is within the limits of 10 CFR 20.

Specification:

A. Airborne Effluents

1. Radioactive gases released from the reactor building ventilation stack and plant chimney shall be continuously monitored. To accomplish this, at least one reactor building ventilation stack monitoring system and plant chimney monitoring system shall be operable at all times. During the period when plateout tests are being performed on the chimney monitoring system and the reactor is operating at a steady state the steam

4.8 SURVEILLANCE REQUIREMENT

RADIOACTIVE MATERIALS

Applicability:

Applies to the periodic monitoring and recording of radioactive effluents.

Objective:

To ascertain that radioactive releases are within allowable values.

Specification:

A. Airborne Effluents

1. The plant chimney and reactor building ventilation stack monitoring systems shall be functionally tested and calibrated every three months.

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

jet air ejector monitors may be used to satisfy the plant chimney monitoring requirements.

The Unit 2/3 plant chimney gas sampling system may be out of service for 48 hours for the purpose of installing the high range noble gas monitor as long as the following conditions are satisfied:

a. Both units are at steady state conditions with the recombiners and charcoal adsorbers in service for the operating unit(s).

b. The chimney release rate must be shown by calculation to be less than the limits of 3.8.A.2.a. and b., assuming the charcoal adsorbers are bypassed on both units.

c. Both offgas monitors on Unit 2 and Unit 3 must be operational and the monitor

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

reading correlated to the chimney release rate based on the conservative assumption of both units' charcoal adsorbers being bypassed.

d. If the provisions of 3.8.A.1.a., b. or c. cannot be met, an orderly load reduction of the unit(s) shall be initiated immediately.

Due to the existence of the Dresden Unit 1 and Unit 2/3 stacks in close vicinity, a set of equations are needed to express the airborne effluents limits. The symbols in the equations stand for the following:

Q_1 = release rate from Unit 1 plant chimney

Q_2 = release rate from the Units 2/3 plant chimney with only Unit 2 or only Unit 3 operating (not both)

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

$Q_{2,3}$ = release rate from the Units 2/3 plant chimney with both units operating

Q_{RS} = release rate from Units 2 and 3 reactor building ventilation stack

2. a. The site release rate for gross activity, except for halogens and particulates with half lives longer than eight days, shall not exceed:

$$\frac{Q_1}{0.56} + \left(\frac{Q_2}{0.7} \text{ or } \frac{Q_{2,3}}{0.9}\right) + \frac{Q_{RS}}{0.09} \text{ less than or equal to } 1.0$$

where Q is measured in Curies/sec.

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

2. a. Station records of gross ventilation stack and plant chimney release rate of gaseous activity shall be maintained on an hourly basis to assure that the specified rates are not exceeded and to yield information governing general integrity of the fuel cladding. Records of isotopic analyses shall also be maintained. Within one month after initial commercial service of the unit, an isotopic analysis will be made of the gaseous activity release rate. From this sample a ratio of long lived to short lived activity will be established. Daily

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

b. In addition to any other requirement of these technical specifications the licensee has volunteered:

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

samples of off-gas will be taken and gross ratio of long lived to short lived activity determined. When the daily samples indicate a change in the ratio of greater than 20% from the ratio established by the previous isotopic analysis, a new isotopic analysis will be performed.

A new isotopic analysis of off-gas will be performed at least quarterly. Gaseous release of tritium shall be calculated on a monthly basis from measured data.

b. Station records of release of iodines shall be maintained on the basis of all stack and plant chimney filter cartridges counted. The filter cartridges shall be counted weekly, when the measured release rate of gross beta-gama activity is less than 10% of the release limit specification 3.8.A.2.a, other-

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

(1) During reactor power operation of Units 2 and/or 3, operating procedures will be implemented to reduce release rates to those consistent with 3.8.E of these specifications prior to the release rate for gross activity, except for halogens and particulates with half lives longer than eight days, exceeding 0.105 ci/sec with Unit 2 or 3 operating alone, or 0.135 ci/sec for both units operating simultaneously.

(2) The release rates specified in 3.8.A.2.b.(1) shall not be exceeded for a time period in excess of that established by the following equations:

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

wise the cartridges shall be counted at least twice a week. Particulate isotopic analysis shall be made and recorded quarterly.

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

a. Dresden 2 or 3 operating $t = 560/Q_X$ (Ci - hr/sec)

b. Dresden 2 and 3 operating $t = 720/Q_Y$ (Ci - hr/sec)

Where:

T = cumulative hours of operation permitted at release rate Q_x or Q_y, or above in the 12 months ending with the month for which the calculation is made

Q_X = release rate above 0.105 ci/sec for Dresden Unit 2 or Unit 3 operating separately

Q_y = release rate above 0.135 ci/sec when both Dresden Units 2 and 3 are operating simultaneously.

(3) If the limits of 3.8.A.2.b.(1) are exceeded for a period of greater than 48 hours, the

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

licensee shall then notify the Director, Division of Reactor Licensing in writing within 48 hours of its plans for reducing the effluent release rate to a level which is consistent with Section 3.8.E of these specifications.

c. The summation of release rates of halogens and particulates with half lives longer than 8 days released to the environs as part of the airborne effluents shall not exceed:

$$\frac{Q_1}{2.4\times10^{-6}} + \left(\frac{Q_2}{3.5\times10^{-6}} \text{ or } \frac{Q_{2,3}}{4.3\times10^{-6}}\right)^{*} + \frac{Q_{RS}}{0.12\times10^{-6}} \text{ less than or equal to } 1.0$$

where Q is measured in Curies/sec.

*(Note for equations 3.8.A.2.a. and c.):

Where the term in parentheses the operational status of Units 2 and 3. If either Units 2 or 3 is shutdown, then the first term (Q_2/value) shall be used. If both units are in operation, then the second term ($Q_{2,3}$/value) shall be used.

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. If the limits of 3.8.A.2.a., or 3.8.A.2.c. are exceeded, an orderly load reduction of the unit(s) causing these limits to be exceeded shall be initiated immediately to reduce the releases below the limits of 3.8.A.2.a. or 3.8.A.2.c. The provisions of Specification 3.0.A. are not applicable.

B. Mechanical Vacuum Pump

1. The mechanical vacuum pump shall be capable of being isolated and secured on a signal of high radioactivity, whenever the main steam isolation valves are open.

2. If the limits of 3.8.B. are not met following a routine surveillance check, orderly shutdown shall be initiated.

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

B. Mechanical Vacuum Pump

At least once during each operating cycle verify automatic securing and isolation of the mechanical vacuum pump.

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. Liquid Effluents

1. Radioactive liquid released from the facility shall be continuously monitored. To accomplish this either the radiation monitor or the discharge line on the discharge canal sampler shall be operable.

2. The concentration of gross beta activity (above background) in the condenser cooling water discharge canal shall not exceed the limits stated below unless the discharge is controlled on a radionuclide basis in accordance with Appendix B, Table II, Column 2 of 10CFR20 and note 1 thereto:

Maximum Concentration -

1×10^{-7} Micro-Ci/ml

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

C. Liquid Effluents

1. The radiation monitor shall be calibrated quarterly and functionally tested monthly. The operability of the sampler shall be verified on a daily basis.

2. Station records shall be maintained of the radioactive concentration and volume of each batch of liquid effluent released and of the condenser cooling water flow at time of discharge.

Isotopic analyses including determination of tritium of representative batches of liquid effluent shall be performed and recorded at least once per quarter. Each batch of effluent released shall be counted for gross alpha and beta activity and the results recorded. At least once per month a gamma scan of representative batches of effluent shall be performed and recorded to determine the gamma energy peaks

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. Two independent samples from a tank shall be taken and analyzed and the valve line-up checked prior to discharge of liquid effluents from that tank.

4. If the limits of 3.8.C. cannot be met, radioactive liquid effluents shall not be released.

D. Radioactive Waste Storage

The maximum amount of radioactivity in liquid storage in the Waste Sample Tanks, the Floor Drain Sample Tanks and the Waste Surge Tank shall not exceed 3.0 curies and the maximum amount of radioactivity in any tank shall not exceed 0.7 curies. If these conditions cannot be met, the stored liquid shall be recycled within 24 hours to the Waste Collector Tanks or the Waste Neutralizer Tanks until the condition is met.

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

of these batches. If energy peaks other than those determined by the previous isotopic analyses are found, a new set of isotopic analyses shall be performed and recorded.

3. The performance and results of independent samples and valve checks shall be logged.

D. Radioactive Waste Storage

A sample from each of the Waste Sample Tanks, Floor Drain Sample Tanks, and Waste Surge Tank shall be taken, analyzed and recorded every 72 hours. If no additions to a tank have occurred since the last sample, the tank need not be sampled until the next addition.

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

E. General

It is expected that releases of radioactive material in effluents will be kept as small fractions of the limits specified in Section 20.106 of 10 CFR Part 20. At the same time the licenseeis permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such small fractions, but still within the limits specified in Section 20.106 of 10 CFR Part 20. It is expected that in using this operational flexibility under unusual operating conditions the licensee will exert his best efforts to keep levels of radioactive material in effluents as low as is reasonably achievable.

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

E. General

1. Operating procedures shall be developed and used, and equipment which has been installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences, shall be maintained and used, to keep levels or radioactive material in effluents released to unrestricted areas as low as is reasonably achievable. The environmental monitoring program given in Table 4.8.1 shall be conducted.

2. A census of animals producing milk for human consumption shall be conducted annually during the grazing season to determine their location and number with respect to the site. The census shall be conducted

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

under the following conditions:

1. Within a 2-mile radius from the plant site, enumeration by a door-to-door or equivalent counting technique.

2. Within a 5-mile radius, enumeration by using referenced information from county agricultural agents or other reliable sources.

If it is learned from this census that animals are present at a location which yields a calculated thyroid dose greater than from previously sampled animals, the new location shall be added to the surveillance program as soon as practicable. The sampling location having the lowest calculated dose may then be dropped from the surveillance program at the end of the grazing season during which the census was conducted. Also, any location from which milk can no longer be obtained may be dropped from the surveillance program after notifying the NRC in writing that

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

F. Miscellaneous Radioactive Materials Sources

Source Leakage Test

Specification

Each sealed source containing radioactive material in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and either decontaminated and repaired or disposed of in accordance with Commission Regulations.

A complete inventory of radioactive materials in the licensee's possession shall be maintained current at all times.

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

milk-producing animals are no longer present at that location.

F. Miscellaneous Radioactive Materials Sources

Each sealed source shall be tested for leakage and/or contamination by the licensee or by other persons specifically authorized by the Commission or an Agreement State. The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

Each category of sealed sources shall be tested at at the frequency described below.

1. Sources in use (excluding startup sources previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive material:

   a. With a half-life greater than 30 days (excluding Hydrogen 3), and

   b. In any form other than gas.

2. Stored sources not in use - Each sealed source shall be tested

3.8 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.8 SURVEILLANCE REQUIREMENT (Cont'd.)

prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

3. Startup sources - Each sealed startup source shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.6.C.3 if source leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

TABLE 4.8-1

DRESDEN STANDARD RADIOLOGICAL MONITORING PROGRAM

Sample Media	Collection Site	Type of Analysis	Frequency	Non-Routine Reporting Levels (1)
1. Air Monitoring	(a) Onsite and near field	1. Filter - gross beta(2)	1. Weekly	Cs-134 10, CS-137 20 pCi/m^3
	(1) Onsite Station #1 (2) Onsite Station #2	2. Charcoal - I-131	2. Bi Weekly*	0.7 pCi/m^3
	(3) Onsite Station #3 (4) Collins Road (5) Bennitt Farm (6) Pheasant Trail	3. Sampling Train - Test and Maintenance	3. Weekly	
	(b) Far Field (1) Clay Products (2) Prairie Park	1. Filter Exchange	1. Weekly	Same as 1(a) when analyses
	(3) Coal City (4) Goose Lake Village	2. Charcoal Exchange	2. Bi-Weekly*	are made
	(5) Morris (6) Lisbon (7) Minooka (8) Channahon (9) Joliet (10) Elwood (11) Wilmington	3. Sampling Train - Test and Maintenance	3. Weekly	
2. TID	Same as 1	Gamma Radiation	Quarterly	
3. Fish	Dresden Pool of Illinois River	Gamma isotopic	Semi-annual	Mn-54 $3x10^4$, Fe-59 $1x10^4$ Co-58 $3x10^4$, Co-60 $1x10^4$ Zn-65 $2x10^4$, Cs-134 $1x10^3$ Cs-137 $2x10^3$ pCi/Kg wet weight
4. Milk	2 dairy farms	I-131	1. Weekly - Grazing Season - May to Oct	I-131 3 pCi/l Cs-134 60 pCi/l Cs-137 70 pCi/l
			2. Monthly - Nov to Apr	Ba-La-140 300 pCi/l
5. Surface Water	Illinois River at EJ&E R.R. Bridge	Gamma isotopic	Monthly analysis of weekly composites	Footnote**
6. Cooling Water Sample	(a) Inlet (1) Unit 1	Gross Beta	Weekly	
	(b) Discharge (1) Unit 1 (2) Unit 2/3			
7. Sediment	(a) Dresden Lock & Dam	Gamma Isotopic	Annual	

Notes for Table 4.8-1

* Bi-weekly shall mean that the frequency is once every other week.

(1) Average concentration over calendar quarter

** H-3 $2x10^4$, Mn-54 $1x10^3$, Fe-59 $1x10^2$, Co-58 $6x10^2$, Co-60 $2x10^2$, Zn-65 $2x10^2$, Zr-Nb-95 $4x10^2$, I-131 3, Cs-134 30, Cs-137 60, Ba-La-140 $1x10^2$ pCi/l.

(2) A gamma isotopic analysis shall be performed whenever the gross beta concentration in a sample exceeds by five times (5x) the average concentration of the preceding calendar quarter for the sample location.

3.8 LIMITING CONDITIONS FOR OPERATION BASES

A. Airborne Effluents - Detailed dose calculations for several locations offsite have been made and are described in Appendix A of the SAR. These calculations consider site meteorology, buoyancy characteristics, and isotopic content of the effluent of each unit. Independent dose calculations for several locations offsite have been made by the AEC staff. The method utilized onsite meteorological data developed by the applicant and utilized diffusion assumptions as developed by the applicant with the exception that: (1) the Stumke correction factor for plume rise was not allowed, (2) the height of the bluff north of the site (30 meters) was subtracted from the stack height for calculational purposes, (3) Pasquill diffusion parameters rather than Hanford parameters were used, (4) the staff used a reflection factor of 2 for the calculation of Specification 3.8.A.2.

The method utilized by the staff is described in Section 7-5.2.5 of "Meteorology and Atomic Energy-1968," equation 7.63 being used. The results of these calculations were more conservative than those generated by the applicant and were thus chosen to be used as the basis of establishment of the limits. Based on these calculations, a release rate limit of gross activity, except for halogens and particulates with half-lives greater than eight days, in the amount of (a) 0.56 curies/sec. from the Unit 1 plant chimney or (b) 0.9 curies/sec. from the Units 2/3 plant chimney or (c) 0.09 curies/sec. from the ventilation stack will not result in offsite annual doses in excess of the limits specified in 10 CFR 20. Because a lower bouyancy factor is obtained when either Unit 2 or 3 is shut down, the equation must be changed so that the operating unit discharge is 0.7 curies per sec. These limits are based on a noble gas mixture whose energy with 30 minute holdup is 0.7 Mev. If on analysis this average energy increases, the average annual release limit must be decreased accordingly.

Considering the above, 3.8.A.2 gives equations to be used in summing the airborne effluents from the Unit 1 plant chimney. The Unit 2/3 plant chimney, and the Unit 2/3 ventilation stack that will assure that total off-site doses are not in excess of the limits specified in 10 CFR 20.

The intent of Section 3.8.A.2.b is not to relieve the licensee of its obligation to exert its best efforts to keep levels of radioactive material in effluents as low as practicable. At the action level specified in Section 3.8.A.2.b, the Commission is to be informed of the licensee's plans for continued operation of the facilities.

3.8 LIMITING CONDITIONS FOR OPERATION BASES (Cont'd.)

The equations given in Specification 3.8.A.2.b.2 give the cumulative hours which represent the limits of permissible operation which reduce the permissible activity released compared to continuing operation at the conditions stated in Section 20.106 of 10 CFR Part 20. The time periods under discussion permit short-term releases higher than small fractions of the limits specified in Section 20.106 of 10 CFR Part 20.

In addition, Commonwealth Edison has embarked on a program of selecting, designing, and installing additional equipment to reduce off-gas emissions on Dresden Units 2 and 3. This equipment is expected to reduce substantially the releases of radioactive material in the effluent. Commonwealth Edison will submit a description of its proposed design for this equipment, and a schedule for its installation, prior to June 1, 1971, unless an extension of this time is granted. Upon completion of the installation of this equpment, these Technical Specifications will be revised to include the effect of operation of the emission-reducing equipment.

Detailed calculations of ground level air concentrations of halogens and particulates with half-lives greater than 8 days at several offsite locations have been made as described in Appendix A of the SAR. These calculations consider site meteorology and buoyancy characteristics of the effluent from each unit. Based on these calculations, the release rate limit for these isotopes in the equation in Section 3.8.A.2.b is obtained. Use of this equation assures that releases will not result in off-site doses in excess of those specified in 10 CFR 20.

The assumptions used by the AEC staff for these calculations were: (1) Onsite meteorological data were used for the most critical 22.5 degree sector. (2) No building wake credit was used. (3) To consider possible reconcentration effects a reduction factor of 700 was applied to allow for the milk production and consumption mode of uptake.

Before initial operation of the nearby Midwest Fuel Reprocessing Plant the above limits will be adjusted to reflect the dose contribution of this facility.

B. Mechanical Vacuum Pump - The purpose of isolating the mechanical vacuum pump line is to limit release of activity from the main condenser. During an accident, fission products would be transported from the reactor through the main

3.8 LIMITING CONDITIONS FOR OPERATION BASES (Cont'd.)

steamlines to the main condenser. The fission product radioactivity would be sensed by the main steamline radioactivity monitors which initiate isolation.

C. Liquid Effluents - Liquid effluent release rate will be controlled in terms of the concentration in the discharge canal. In the case of unidentified mixtures such concentration limit is based on assumption that the entire content is made up of the most restrictive isotope in accordance with 10 CFR 20. Such a limit assures that even if a person obtained all of his daily water intake from such a source, the resultant dose would not exceed that specified in 10 CFR 20. Since no such use of the discharge canal is made and considerable natural dilution occurs prior to any location where such dose usage could occur, this assures that off-site doses from this source will be far less than the limits specified in 10 CFR 20.

In addition to the two independent samples of each batch prior to discharge, a radiation monitor on the discharge line and a sampler in the discharge canal give further assurance that discharges are kept at or below the maximum limits at all times.

D. Radioactive Waste Storage - As discussed in the SAR, the radioactive waste tanks that are at or above grade are located such that their postulated catastrophic failure could cause release of their contained radioactivity to the Illinois River. To assure that such a postulated release would not raise radioactivity levels in the river to values greater than ten times 10 CFR 20, a limit on the amount of radioactivity the tanks can contain is established.

The performance of the radiation monitoring system relative to detecting fuel leakage shall be evaluated during the first five years of operation. The conclusions of this evaluation will be reported to the Atomic Energy Commission.

E. General - The environmental radiological monitoring program is designed to:

1. Provide data on measurable levels of radiation and radioactive materials in the environment in order to evaluate the calculational models used to relate the quantities of radioactive material released in effluents to the radiation doses received by individuals via the principal pathways of exposure;

3.8 LIMITING CONDITIONS FOR OPERATION BASES (Cont'd.)

2. Identify changes in the use of nearby unrestricted areas for agricultural purposes to permit modifications in monitoring programs for evaluating doses to individuals from principal pathways of exposure;

3. Maintain off-site air samplers in a readiness posture in case of an unplanned release; and

4. Provide year-round coverage of certain principal pathways.

Attainment of these objectives will be met by dividing the monitoring program into two distinct and independent components.

Standard Monitoring Program

The standard monitoring program is designed to provide year-round coverage of certain principal dose pathways, identify land use changes, and maintain the offsite air samplers in a state of readiness in case of an unplanned release.

The sampling and analytical schedule for environmental samples, and the testing and maintenance schedule for off-site air samplers, is given in Table 4.8-1. Figure 4.8-1 shows the locations of the fixed sampling sites.

Table 4.8-2 indicates acceptable detection capabilities for radioactive materials in environmental samples. These detection capabilities are tabulated in terms of the lower limits of detection (LLDs) at the 95% confidence level. The LLDs shall be determined in the manner described in HASL-300, section D-08, August 1976. The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, contractor omission which is corrected as soon as discovered, malfunction of sampling equipment, or if a person who participates in the program by providing samples goes out of business. If the equipment malfunctions, corrective actions shall be completed as soon as practical. If a person supplying samples goes out of business, a replacement will be found as soon as possible. All deviations from the sampling schedule shall be described in the annual report.

3.8 LIMITING CONDITIONS FOR OPERATION BASES (Cont'd.)

Environmental Dose Pathway Study

The environmental dose pathway study (EDPS) will use the best practicable monitoring techniques for measuring radioactivity in effluents and certain environmental media in order to evaluate the relationship between the effluents and doses to individuals in unrestricted areas.

The EDPS will provide data on measurable levels of radiation and certain radioactive materials in effluent and the environs to evaluate the relationship between the quantitative and qualitative nature of effluent discharges and the doses to individuals.

It is planned that most of the monitoring will be conducted during the warmer months of the year (May through October) so that such principal pathways as milk can be studied. Use will be made of the onsite meteorological data to predict atmospheric dispersion so that comparisons can be made with measured data.

A description of the EDPS program to be performed is described in the report to the NRC entitled "Proposal to Change Environmental Radiological Monitoring Programs" by the Commonwealth Edison Company and Dr. Bernd Kahn, Georgia Institute of Technology, dated October 1976.

F. Miscellaneous Radioactive Materials Sources

The objective of this specification is to assure that leakage from byproduct, source and special nuclear material sources does not exceed allowable limits. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium.

4.8 SURVEILLANCE REQUIREMENT BASES

None

TABLE 4.8-2

PRACTICAL LOWER LIMITS OF DETECTION (LLD)
FOR STANDARD ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM

Sample Media	Analysis	LLD (4.66σ)	Units
Airborne "Particulate"	Gross Beta **	0.01	pCi/m^3
	Gamma Isotopic	0.01	pCi/m^3
	Sr-89,90	0.01	pCi/m^3
Airborne I-131	Iodine-131	0.10	pCi/m^3
Liquids	Sr-89	10	pCi/l
	Sr-90	2	pCi/l
	I-131	5*	pCi/l
	Cs-134	10	pCi/l
	Cs-137	10 ***	pCi/l
	Tritium	0.2	pCi/ml
	Gross Beta **	5	pCi/l
	Gamma Isotopic	(LT)20	pCi/l/nuclide
Vegetation	Gross Beta **	2	pCi/g wet
	I-131	0.03	pCi/g wet
	Sr-89,90	1	pCi/g wet
	Gamma Isotopic	0.2	pCi/g wet
Soil, Sediment	Gross Beta **	2	pCi/g dry
	Sr-89,90	1	pCi/g dry
	Gamma Isotopic	0.2	pCi/g dry
Animal Tissue	Sr-89,90	0.1	pCi/g wet
	I-131 - Thyroid	0.1	pCi/g wet
	Cs-134,137	0.1	pCi/g wet
	Gross Beta **	1.0	pCi/g wet

(LT) = Less than
* 0.5 pCi/l on milk samples collected during the pasture season.

** Referenced to CS-137

*** 5.0 pCi/l on milk samples.

3.9 LIMITING CONDITION FOR OPERATION

AUXILIARY ELECTRICAL SYSTEMS

Applicability:

Applies to the auxiliary electrical power system.

Objective:

To assure an adequate supply of electrical power during plant operation.

Specification:

A. The reactor shall not be made critical unless all the following requirements are satisfied:

1. One 345 KV line, associated switchgear, and the reserve auxiliary power transformer capable of carrying power to Unit 3.

2. The Dresden 3 diesel generator and the Unit 2/3 diesel generator shall be operable.

3. An additional source of power consisting of one of the following:

   (a) One other 345 KV line, fully operational and

4.9 SURVEILLANCE REQUIREMENT

AUXILIARY ELECTRICAL SYSTEMS

Applicability:

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

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Station Batteries

1. Every week the specific gravity and voltage of the pilot cell and temperature of adjacent cells and overall battery voltage shall be measured.

2. Every three months the measurements shall be made of voltage of each cell to nearest 0.01 volt, specific gravity of each cell, and temperature of every fifth cell.

3. Every refueling outage, the station batteries shall be subjected to a rated load discharge test. Determine specific gravity and voltage of each cell after the discharge.

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

capable of carrying auxiliary power to Unit 3.

(b) One 138 KV line from Unit 2 capable of carrying auxiliary power to an essential electrical bus of Unit 3 through the 4160 volt bus tie.

4. (a) 4160 volt buses 33-1 and 34-1 are energized.

(b) 480 volt buses 38 and 39 are energized.

5. The unit 24/48 volt batteries, the two station 125 volt batteries and the two station 250 volt batteries and a battery charger for each required battery are operable.

B. Except when the reactor is in the Cold Shutdown or Refueling modes with the head off, the availability of electric power shall be as specified in 3.9.A, except as specified in 3.9.B.1, 3.9.B.2, and 3.9.B.3.

1. From and after the date that incoming power is available from only one line, reactor operation is

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

B. N/A

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

permissible only during the succeeding seven days unless an additional line is sooner placed in service providing both the Unit 3 and Unit 2/3 emergency diesel generators are operable. From and after the date that incoming power is not available from any line, reactor operation is permissible providing both the Unit 3 and Unit 2/3 emergency diesel generators are operating and all core and containment cooling systems are operable and the NRC is notified within 24 hours of the situation, the precautions to be taken during this situation, and the plans for prompt restoration of incoming power.

2. From and after the date that one of the diesel generators and/or its associated bus is made or found to be inoperable for any reason, reactor operation is permissible according to Specification 3.5/4.5F and 3.9D only during the succeeding seven days unless such diesel generator and/or bus is sooner made

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

operable, provided that during such seven days the operable diesel generator shall be demonstrated to be operable at least once each day and two off-site lines are available.

3. From and after the date that one of the two 125/250 battery systems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such battery system is sooner made operable.

C. Diesel Fuel

There shall be a minimum of 10,000 gallons of diesel fuel supply on site for each diesel.

D. Diesel Generator Operability

Whenever the reactor is in the Cold Shutdown or Refueling modes, a minimum of one diesel generator (either the Dresden 3 diesel generator or the Unit 2/3 diesel generator) shall be operable whenever any work is being done which has

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

C. Diesel Fuel

Once a month the quantity of diesel fuel available shall be logged.

Once a month a sample of diesel fuel shall be checked for quality.

D. Diesel Generator Operability

1. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue until both the diesel engine and the generator are at equilibrium

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

the potential for draining the vessel, secondary containment is required, or a core or containment cooling system is required.

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

conditions of temperature while full load output is maintained.

2. During the monthly generator test the diesel starting air compressor shall be checked for operation and its ability to recharge air receivers.

3. During the monthly generator test the diesel fuel oil transfer pumps shall be operated.

4. Additionally, during each refueling outage, a simulated loss of off-site power in conjunction with an ECCS initiation signal test shall be performed on the 4160 volt emergency bus by:

   (a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

   (b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency buses with permanently connected loads, energizes the auto-connected

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

emergency loads through the load sequencer, and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads.

3.9 LIMITING CONDITION FOR OPERATION BASES

A. The general objective of this Specification is to assure an adequate source of electrical power to operate the auxiliaries during plant operation, to operate facilities to cool and lubricate the plant during shutdown, and to operate the engineered safeguards following an accident. There are three sources of electrical energy available; namely, the 345 KV transmission system, the diesel generators, and the 138 KV transmission system through the 4160 volt bus tie.

The d-c supply is required for control and motive power for switchgear and engineered safety features. The electrical power required provides for the maximum availability of power; i.e., one active off-site source and two back-up sources of off-site power and the maximum amount of on-site sources.

B. Auxiliary power for Unit 3 is supplied from two sources, either the Unit 3 auxiliary transformer or the Unit 3 reserve auxiliary transformer. Both of these transformers are sized to carry 100% of the auxiliary load. If the reserve auxiliary transformer is lost, the unit can continue to run for 7 days since the unit auxiliary transformer is available and both diesel generators are operational. A reduced period is provided since if an accident occurs during this period, the unit would trip and power to the unit auxiliary transformer would be lost and the diesels would be the only source of power.

In the normal mode of operation the 345 KV system is operating and two diesel generators are operational. One diesel generator may be allowed out of service based on the availability of power to the 345 KV switchyard, a source of power available from the 138 KV system through a 4160 volt bus tie and the fact that one diesel carries sufficient engineered safeguards equipment to cover all breaks. Off-site power is quite reliable. In the last 25 years there has only been one instance in which all off-site power was lost at a Commonwealth Edison generating station.

A battery charger is supplied with each of the 125 and 250 volt batteries and in addition a shared battery charger is supplied which can be used for Units 2 or 3. Thus, on loss of the normal battery charger, the shared charger can be used. Since an alternate charging source is available, one battery charger can be allowed out of service for thirty days without loss of this source of power. The 125 volt battery system shall have a minimum of 105 volts at the battery terminals to be considered operable.

3.9 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

C. The diesel fuel supply of 10,000 gallons will supply each diesel generator with a minimum of two days of full load operation or about four days at 1/2 load. Additional diesel fuel can be obtained and delivered to the site within an 8-hour period; thus a 2-day supply provides for adequate margin.

4.9 SURVEILLANCE REQUIREMENT BASES

A. 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.

B. 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 would contribute to excessive corrosion of the system causing decreased reliability. The growth of micro-organisms results in slime formations which are one of the chief causes of jellying in hydro-carbon fuels. Minimizing of such slimes is also essential to assuring high reliability.

C. 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 will be manually started, synchronized to the bus and load picked up. The diesel shall be loaded to at least half load to prevent fouling of the engine. It is expected that the diesel generator will be run for one to two hours. Diesel generator experience at other Commonwealth Edison generating stations indicates that the testing frequency is adequate and provides a high reliability of operation should the system be required. In addition, during the test when the generator is synchronized to the bus, it is also synchronized to the off-site power source and thus not completely independent of this source. To maintain the maximum amount of independence, a thirty-day testing interval is also desirable.

4.9 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

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

Following the monthly test of the diesels, the fuel oil day tank will be approximately 1/2 full based on a two-hour test at full load and 205 gallons per hour at full load. At the end of the monthly load test of the diesel generators, the fuel oil transfer pumps will be operated to refill the day tank and to check the operation of these pumps from the emergency source. The test of the emergency diesel generator during the refueling outage will be more comprehensive in that it will functionally test the system; i.e., it will check diesel starting and closure of diesel breaker and sequencing of loads on the diesel. The diesel 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. The timing sequence will be checked to assure proper loading in the time required. The only load on the diesel is that due to friction and windage and a small amount of bypass flow on each pump. Periodic tests between refueling outages verify the ability of the diesel 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 at a refueling interval are sufficient to maintain adequate reliability.

3.10 LIMITING CONDITIONS FOR OPERATION

REFUELING

Applicability:

Applies to fuel handling and core reactivity limitations.

Objective:

To assure core reactivity is within capability of the control rods and to prevent criticality during refueling.

Specification:

A. Refueling Interlocks

The reactor mode switch shall be locked in the "Refuel" position during core alterations and the refueling interlocks shall be operable except as specified in Specifications 3.10.D and 3.10.E.

B. Core Monitoring

During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:

1. The SRM shall be inserted to the normal

4.10 SURVEILLANCE REQUIREMENTS

REFUELING

Applicability:

Applies to the periodic testing of those interlocks and instruments used during refueling.

Objective:

To verify the operability of instrumentation and interlocks used in refueling.

Specification:

A. Refueling Interlocks

Prior to any fuel handling, with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

B. Core Monitoring

Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, the SRM's will be checked daily for response, except when the conditions of 3.10.B.2.a and 3.10.B.2.b are met.

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors are permissible as long as the detector is connected into the normal SRM circuit.)

2. The SRM or dunking type detector shall have a minimum of 3 cps with all rods fully inserted in the core except when both of the following conditions are fulfilled:

   a) No more than two fuel assemblies are present in the core quadrant associated with the SRM.

   b) While in core, these fuel assemblies are in locations adjacent to the SRM.

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool water level shall be maintained at a level of 33 feet.

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool level shall be recorded daily.

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

D. Control Rod and Control Rod Drive Maintenance

* A maximum of two non-adjacent control rods separated by more than two control cells in any direction, may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance provided the following conditions are satisfied:

1. The reactor mode switch shall be locked in the "re-fuel" position. The re-fueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other re-fueling interlocks shall be operable.

* 2. Specification 3.3.A.1 shall be met or, the control rod directional control valves for a minimum of eight control rods surrounding each drive out of service for maintenance will be disarmed electrically and sufficient margin to criticality demonstrated.

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

D. Control Rod Drive and Control Rod Drive Maintenance

1. This surveillance requirement is the same as given in 4.10.A.

* 2. Sufficient control rods shall be withdrawn prior to performing this maintenance to demonstrate with a margin of 0.25 percent delta k that the core can be made subcritical at any time during the maintenance with the strongest operable control rod fully withdrawn and all other

*Revised with change 17 to DPR-19 dated 3/17/72
Revised with change 9 to DPR-25 dated 3/17/72

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

3. SRM's shall be operable (a) in each core quadrant containing a control rod on which maintenance is being performed, and (b) in a quadrant adjacent to one of the quadrants specified in 3.10.D.3.a above. Requirements for an SRM to be considered operable are given in 3.10.B.

E. Extended Core Maintenance

More than two control rods may be withdrawn from the reactor core provided the following conditions are satisfied:

1. The reactor mode switch shall be locked in the "re-fuel"

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

operable rods fully inserted. Alternately, if a minimum of eight control rods surrounding each control rod out of service for maintenance are to be fully inserted and have their directional control valves electrically disarmed, the 0.25 percent delta k margin will be met with the strongest control rod remaining in service during the maintenance period fully withdrawn.

3. This surveillance requirement is the same as that given in 4.10.B.

E. Extended Core Maintenance

1. This surveillance requirement is the same as that given in 4.10.A.

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other re-fueling interlocks shall be operable.

2. SRM's shall be operable in the core quadrant where fuel or control rods are being moved and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in 3.10.B.

F. Spent Fuel Cask Handling

1. Fuel cask handling above the 545' elevation will be done with the reactor building crane in the RESTRICTED MODE only except as specified in 3.10.F.2.

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

2. This surveillance requirement is the same as that given in 4.10.B.

F. Spent Fuel Cask Handling

1. Prior to fuel cask handling operations, the redundant crane including the rope, hooks, slings, shackles and other operating mechanisms will be inspected.

The rope will be replaced if any of the following conditions exist:

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

2. Fuel cask handling in other than the RESTRICTED MODE will be permitted in emergency or equipment failure situations only to the extent necessary to get the cask to the closest acceptable stable location.

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

a. Twelve (12) randomly distributed broken wires in one lay or four(4) broken wires in one strand of one rope lay.

b. Wear of one-third the original diameter of outside individual wire.

c. Kinking, crushing, or any other damage resulting in distortion of the rope.

d. Evidence of any type of heat damage.

e. Reductions from nominal diameter of more than 1/16 inch for a rope diameter from 7/8" to 1 1/4" inclusive.

2. Before August 30, 1976, prior to operations in in the RESTRICTED MODE

a. the "two-block" limit switches will be tested.

On and after August 30, 1976, prior to operation in the RESTRICTED MODE

a. the controlled area limit switches will be tested:

b. the "two-block" limit switches will be tested;

3.10 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

3. Before August 30, 1976, fuel cask handling is permitted, without the mechanically operated power limit switch in the main hoist motor power circuit and without an operable control system for limiting the crane/cask travel to a restricted area, provided an operator, in constant communication with the crane operator and with personnel directing crane operation is stationed at the main breaker supplying power to the overhead crane with no duties other than monitoring crane operation. The operator will be ordered to remove power from the crane in the event that a malfunction either causes the cask to be lifted above a six-inch limit or causes the cask to deviate from the restricted area.

   On and after August 30, 1976, operation with a failed controlled area limit switch is permissible for 48 hours providing an operator is on the refueling floor to assure the crane is operated within the restricted zone painted on the floor.

4.10 SURVEILLANCE REQUIREMENTS (Cont'd.)

c. the "inching hoist" controls will be tested.

3. The empty spent fuel cask will be lifted free of all support by a maximum of 1 foot and left hanging for 5 minutes prior to any series of fuel cask handling operations.

3.10 LIMITING CONDITION FOR OPERATION BASES

A. Refueling Interlocks

During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specifications 3.2 limits the core alterations to assure that the resulting core loading can be controlled with the reactivity control system and interlocks at any time during shutdown or the following operating cycle.

Addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

For a new core the dropping of a fuel assembly into a vacant fuel location adjacent to a withdrawn control rod does not result in an excursion or a critical configuration, thus adequate margin is provided.

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. Requiring a minimum of 3 counts per second whenever criticality is possible provides assurance that neutron flux is being monitored. Criticality is considered to be impossible if there are no more than two assemblies in a quadrant and if these are in locations adjacent to the SRM. In this case only, the SRM or dunking type detector count rate is permitted to be less than 3 counts per second.

3.10 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

C. Fuel Storage Pool Water Level

To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (37'9") well above a level to assure adequate cooling (just above active fuel) and above the level at which the GSEP action is initiated (5' uncontrolled loss of level with level decreasing).

D. During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time. This specification provides assurance that inadvertent criticality does not occur during such maintenance.

The maintenance is performed with the mode switch in the "re-fuel" position to provide the re-fueling interlocks normally available during re-fueling operations as explained in Part A of these Bases. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the re-fueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated with the control rods remaining in service insures that inadvertent criticality cannot occur during this maintenance. The Shutdown margin is verified by demonstrating that the core is shut down even if the strongest control rod remaining in service is fully withdrawn. Disarming the directional control valves does not inhibit control rod scram capability.

The requirement for SRM operability during the maintenance is covered in Part B of these Bases.

E. The intent of this specification is to permit the unloading of a significant portion of the reactor core for such purposes as in-service inspection requirements, examination of the core support plate, etc. This specification provides assurance that inadvertent criticality does not occur during such operation.

This operation is performed with the mode switch in the "re-fuel" position to provide the re-fueling interlocks normally available during re-fueling as explained in Part A of these Bases. In order to withdraw more than one control rod, it is necessary to bypass the re-fueling interlock on each

3.10 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlocks can be bypassed insures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with that control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.

The requirement for SRM operability during these operations is covered in Part B of these Bases.

F. The operation of the redundant crane in the Restricted Mode during fuel cask handling operations assures that the cask remains within the controlled area once it has been removed from its transport vehicle (i.e., once it is above the 545' elevation). Handling of the cask on the Refueling Floor in the Unrestricted Mode is allowed only in the case of equipment failures or emergency conditions when the cask is already suspended. The Unrestricted Mode of operation is allowed only to the extent necessary to get the cask to a suitable stationary position so the required repairs can be made. Operation with a failed controlled area microswitch will be allowed for a 48-hour period providing an Operator is on the floor in addition to the crane operator to assure that the cask handling is limited to the controlled area as marked on the floor. This will allow adequate time to make repairs but still will not restrict cask handling operations unduly.

The Surveillance Requirements specified assure that the redundant crane is adequately inspected in accordance with the accepted ANSI Standard (B.30.2.0) and manufacturer's recommendations to determine that the equipment is in satisfactory condition. The testing of the controlled area limit switches assures that the crane operation will be limited to the designated area in the Restricted Mode of operation. The test of the "two-block" limit switch assures the power to the hoisting motor will be interrupted before an actual "two-blocking" incident can occur. The test of the inching hoist assures that this mode of load control is available when required.

Requiring the lifting and holding of the cask for 5 minutes during the initial lift of each series of cask handling

3.10 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

operations puts a load test on the entire crane lifting mechanism as well as the braking system.

Performing this test when the cask is being lifted initially from the cask car assures that the system is operable prior to lifting the load to an excessive height.

4.10 SURVEILLANCE REQUIREMENT BASES

None

3.11 LIMITING CONDITIONS FOR OPERATION

High Energy Piping Integrity

Applicability:

Applies to operating status of certain piping outside primary containment.

Objective:

To assure the integrity of sections of piping which is postulated to effect safe plant shutdown.

Specification:

1. The high energy piping sections identified in Table 4.11-1 shall be maintained free of visually observable through wall leaks.

   A. If a leak is detected by the surveillance program of 4.11, efforts to identify the source of the leaks shall be started immediately.

   B. If the source of leakage can not be identified within 24 hours of detection or if the leak is found to be from a break in the piping sections identified in Table 4.11-1, the pressure within the section of piping shall be brought to atmospheric pressure within 48 hours.

4.11 SURVEILLANCE REQUIREMENTS

High Energy Piping Integrity

Applicability:

Applies to the periodic examination requirements for certain piping outside primary containment.

Objective:

To determine the condition of the section of piping.

Specification:

The inspections listed in Table 4.11-1 shall be performed as specified.

3.11 LIMITING CONDITIONS FOR OPERATION (Cont'd.

2. When the modifications identified in the Commonwealth Edison letter to the NRC dated September 16, 1975 (G. Abrell to D. Ziemann), have been completed, Technical Specifications 3.11 and 4.11 will no longer be required.

4.11 SURVEILLANCE REQUIREMENTS (Cont'd.)

TABLE 4.11-1

SURVEILLANCE REQUIREMENTS FOR HIGH ENERGY PIPING OUTSIDE CONTAINMENT

Piping	Surveillance Area	Surveillance Technique	Frequency
Main Steam	from primary containment penetration to secondary containment penetration	Visual (1)	30 days
Reactor Feedwater Piping	from primary containment penetration to secondary containment penetration	Visual (1)	30 days
	discharge and "A" (2) Reactor Feed Pump to the 24-inch Diameter Feedwater Header	Visual (1)	30 days
HPCI Steam Piping	from the primary containment penetration to the reactor building penetration	Visual (1)	30 days

Notes for Table 4.11-1

(1) Visual observation of piping insulation and area for evidence of wetness or any physical damage resulting from a leak. Surveillance to be performed using normal access without scaffolding or any other access aids.

(2) "A" Reactor Feed Pump for Unit 2
"C" Reactor Feed Pump for Unit 3

3.11 LIMITING CONDITION FOR OPERATION BASES

High Energy Piping Integrity (Outside Containment)

Intensive analysis and review has shown that there are specific postulated high energy piping system failures which have the potential to inhibit safe cold shutdown of the reactor. This conclusion is based on utilizing the basic NRC high energy line break criteria. To reduce the probability of such failures, certain plant modifications are necessary. Until these modifications are complete, additional surveillance will be performed during plant operation to enhance the detection of piping system defects. The inservice examination and the frequency of inspection will provide a means for timely detection of such piping defects.

4.11 SURVEILLANCE REQUIREMENT BASES

None

3.12 LIMITING CONDITIONS FOR OPERATION

FIRE PROTECTION SYSTEMS

Applicability:

Applies to the fire protection systems whenever the equipment or systems being protected are required to be operable.

Objective:

To ensure that adequate protection against fires is maintained during all modes of facility operation.

Specification:

A. Fire Detection Instrumentation

1. As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.12-1 shall be operable at all times when equipment in that fire detection zone is required to be operable.

2. With the number of operable fire detection instruments less than required by Table 3.12-1;

   a. Perform an inspection of the affected zone, if accessible, within 1 hour. Perform additional inspections at least once per hour, except in inaccessible areas.

4.12 SURVEILLANCE REQUIREMENTS

FIRE PROTECTION SYSTEMS

Applicability:

Applies to the periodic testing requirements of the fire protection systems whenever the fire protection systems are required to be operable.

Objective:

To verify operability of the fire protection systems.

Specification:

A. Fire Detection Instrumentation

1. Each of the fire detection instruments given by Table 3.12-1 shall be demonstrated OPERABLE at least every 6 months by a channel functional test.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

b. Restore the inoperable instrument(s) to operable status within 14 days, or prepare and submit a report to the Commission pursuant to Specification 6.6.B.2 within the next 30 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to operable status.

c. The provisions of Specification 3.0.A. are not applicable.

B. Fire Suppression Water System

1. The Fire Suppression Water System shall be operable at all times with:

a. A flow path capable of taking suction from the 2/3 Intake Canal for Unit 2/3 Fire Pump.

b. A flow path capable of taking suction from the Unit 1 Intake Canal for Unit 1 fire pump.

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

B. Fire Suppression Water System

1. The Fire Suppression Water System shall be demonstrated operable:

a. At least once per 31 days by verifying valve positions.

b. At least once per 12 months by cycling each testable valve in the flow path through one complete cycle.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

c. The Unit 2/3 fire pump (2000 GPM) with its discharge aligned to the fire suppression header (from Unit 2/3 Intake Structure).

d. The Unit 1 fire pump (2000 GPM) with its discharge aligned to the fire suppression header (from Unit 1 Intake Structure).

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

c. At least once per year by performance of a system flush.

d. At least once per operating cycle:

1) By performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence and verifying that each automatic valve in the flow path actuates to its correct position.

2) By verifying that the Unit 2/3 fire pump develops at least 2000 gpm at a system head of 238 feet.

3) By verifying that the Unit 1 fire pump starts and develops at least 2000 gpm at a system head of 238 ft.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

e. Automatic initiation logic for each fire pump.

f. Fire suppression header piping with sectional control valves to:

1) The yard loop.

2) The front valve ahead of the water flow alarm device on each sprinkler or water spray system.

3) The standpipe system.

2. With an inoperable fire pump or associated water supply, restore the inoperable equipment to operable status within 7 days, or prepare and submit a

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

4) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

e. At least once per 3 years by performing flow tests of the system in accordance with the "Test of Water Supplies" Chapter in the NFPA Fire Protection Handbook.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

report to the Commission pursuant to Specification 6.6.B.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

3. With no Fire Suppression Water System operable, within 24 hours;

   a. Establish a backup Fire Suppression Water System.

   b. Notify the Commission pursuant to Specification 6.6.B.1 outlining the actions taken and the plans and schedule for restoring the system to operable status.

4. If the requirements of 3.12.B.3.a cannot be met, an orderly shutdown shall be initiated, and the reactor shall be in cold shutdown condition within 24 hours.

C. Sprinkler Systems

1. The sprinkler systems given in Table 3.12-2 shall be operable at all times when equipment in the area

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

C. Sprinkler Systems

1. At least once per 31 days by verifying that each valve, manual, power-operated, or automatic, in the flow

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

that is sprinkler protected is required to be operable.

2. With a sprinkler system inoperable, establish fire inspections with backup fire supression equipment within 1 hour.

   a. In the Unit 2/3 turbine mezzanine 538' elevation area or Unit 3 hydrogen seal oil area, a continuous fire watch is to be established.

   b. In all other areas given in Table 3.12-2 perform surveillance hourly.

3. Restore the system to operable status within 14 days, or prepare and submit a report to the Commission pursuant to Specification 6.6.B.2 within the next 30 days outlining the cause of inoperability, action taken and the plans for restoring the system to operable status.

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

path is in its correct position.

2. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

3. At least once per operating cycle:

   a. A system functional test shall be performed which includes simulated automatic actuation of the system and verifying that the automatic valves in the flow path actuate to their correct positions.

   b. The sprinkler headers shall be inspected to verify their integrity.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

4. The provisions of Specification 3.0.A are not applicable.

D. CO_2 System

1. The CO_2 Storage Tank shall have a minimum standby level of 50 percent and a minimum pressure of 250 psig.

2. The CO_2 System given in Table 3.12-3 shall be operable.

3. Specifications 3.12.D.1 and 3.12.D.2 above apply when the equipment in the areas given in Table 3.12-3 is required to be operable.

4. With a CO_2 System inoperable, establish fire inspection with backup fire suppression equipment in

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

c. Each nozzle shall be inspected to verify no blockage.

4. At least every other operating cycle, a flow test will be performed to verify that each open head spray nozzle is unobstructed.

D. CO_2 System

1. At least once per 7 days the CO_2 Storage Tank level and pressure will be verified.

2. At least once per 31 days by verifying that each valve, manual, power-operated, or automatic, in the flow path is in the correct position.

3. At least once per operating cycle, the system valves and associated dampers will be verified to actuate automatically and manually. A brief flow test shall be made to verify flow from each nozzle.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

unprotected areas within 1 hour, and perform inspection at least hourly.

5. Restore the system to operable status within 14 days, or prepare and submit a report to the Commission pursuant to Specification 6.6.B.2 within the next 30 days outlining the cause of inoperability, action taken and the plans and schedule for restoring the system to operable status.

6. The provisions of Specification 3.0.A. are not applicable.

E. Fire Hose Stations

1. The Fire Hose Stations given in Table 3.12-4 shall be operable at all times when the equipment in the area is required to be operable.

2. With a hose station inoperable route an additional equivalent capacity hose to the unprotected area from an operable hose station within 1 hour.

3. When a hose station becomes inoperable, restore to operable

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

E. Fire Hose Stations

1. At least once per 31 days, a visual inspection of each fire hose station shall be made to assure all equipment is available at the station.

2. At least once per operating cycle, the hose will be removed for inspection and repacked. Degraded gaskets in the couplings will be replaced.

3. At least once per 3 years, each hose station valve will be

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

status within 14 days or report to the Commission pursuant to specification 6.6.B.2 within the next 30 days outlining the cause of inoperability and plans for restoring the hose station to operability.

4. The provisions of Specification 3.0.A are not applicable.

F. Penetration Fire Barriers

1. All penetration fire barriers (including fire doors and fire dampers) protecting safety related areas shall be intact, except as stated in specification 3.12.F.2 below.

2. With one or more of the required penetration fire barriers not intact, establish a continuous fire watch on at least one side of the affected penetration within 1 hour when

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

partially opened to verify valve operability and no blockage.

4. At least once per 3 years a hydrostatic test will be conducted on each hose at a pressure at least 50 psig above line pressure at that station.

F. Penetration Fire Barriers

1. Each of the required penetration fire barriers shall be verified to be intact by a visual inspection:

   a. At least once per 18 months, and

   b. Prior to declaring a penetration fire barrier intact following repairs or maintenance.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

the area on either side of the affected penetration contains equipment required to be operable.

3. The provisions of Specification 3.0.A are not applicable.

4. Restore the non-functional fire barrier penetrations to operable status within 7 days or prepare and submit a report to the Commission pursuant to Specification 6.6.B.2. within the next 30 days outlining the cause of inoperability, action taken and the plans and schedule for restoring the penetration fire barriers to operable status.

G. See 3.12.B.

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

G. Fire Pump Diesel Engine

1. The fire pump diesel engine shall be demonstrated OPERABLE:

   a. At least once per 31 days by verifying:

      1) The fuel storage day tank contains at least 150 gallons of fuel, and

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

2) The diesel starts from ambient conditions and operates for at least 30 minutes.

3) The fuel transfer pump starts and transfers fuel from the storage tank to the day tank.

b. At least once per 92 days a sample of diesel fuel shall be checked for viscosity, water and sediment. The procedure used shall be consistent with existing station procedures used to check diesel fuel in the main storage tanks.

c. At least once per 18 months, by:

1) Subjecting the diesel to an inspection in accordance with procedure prepared in conjunction with its manufacturer's recommendations for the class of service, and

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

2) Verifying the diesel starts from ambient conditions on the autostart signal and operates for greater than or equal to 30 minutes while loaded with the fire pump.

2. The fire pump diesel engine batteries shall be demonstrated operable:

a. At least once per 7 days by verifying that:

1) The electroylyte level of each battery is above the plates, and

2) The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

H. Halon System

1. The following Halon system shall be OPERABLE with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure.

   a. Auxiliary Electrical Equipment Room

2. With one or more of the above required Halon systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or, prepare and submit a report to the

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

1) The batteries and battery racks show no visual indication of physical damage or abnormal deterioration, and

2) The battery-to-battery and terminal connections are clean, light, free of corrosion and coated with anti-corrosion material.

H. Halon System

1. At least once per 31 days, verify that each valve in the flow path is in the correct position.

2. At least once per 6 months, the Halon storage tank weight and pressure will be verified.

3.12 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

Commission pursuant to Specification 6.6.B within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

4.12 SURVEILLANCE REQUIREMENTS (Cont'd.)

3. At least once per operating cycle, the system, including associated ventilation dampers, will be verified to actuate manually and automatically. A flow test shall be made through headers and nozzles to assure no blockage.

3.12 LIMITING CONDITIONS FOR OPERATION BASES

Fire Protection Systems

Operability of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment, and is an integral element in the overall facility fire protection program. In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire inspections in the affected areas is required to provide detection capability until the inoperable instrumentation is returned to service.

The operability of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, sprinklers, CO_2 systems, Halon system, and fire hose stations, and is adequate to minimize potential damage to safety related equipment, and is a major element in the facility fire protection program.

In the event that portions of the fire suppression system are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the affected equipment can be restored to service.

A fire suppression water system shall consist of a water source, pumps, and distribution piping with associated valves. Such valves shall include sectional control valves, and the first valve ahead of the water flow alarm device on each sprinkler or hose standpipe riser.

In the event that the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a 24-hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving

3.12 LIMITING CONDITIONS FOR OPERATION BASES (Cont'd.)

several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the barriers are not functional, a fire watch is required to be maintained in the vicinity of the affected barrier until the barrier is restored to functional status.

4.12 SURVEILLANCE REQUIREMENT BASES

None

TABLE 3.12-1

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE
1. Main Control Room 34 Smoke Detectors	24
2. Unit 2/3 Aux. Elect. Equip. Room 12 Smoke Detectors	8
3. Unit 2/3 Computer Room 8 Smoke Detectors	6
4. Unit 2/3 Diesel Generator Area 4 Heat Detectors	3
5. Unit 3 Diesel Generator Area 3 Heat Detectors	2
6. Unit 3 Turb Bldg. 4KV Switchgear 8 Smoke Detectors	6
7. Unit 3 Battery Room 4 Smoke Detectors	3
8. Unit 3 Rx Bldg. 480V MCC (570') 3 Smoke Detectors	2
9. Unit 3 Rx Bldg. 4KV Switchgear 4 Smoke Detectors	3
10. Unit 3 Rx Bldg. 480V MCC (517') 8 Smoke Detectors	6
11. Unit 3 Turb. Bldg. 480V MCC 11 Smoke Detectors	8
12. Unit 3 Cable Tunnel 40 Smoke Detectors	28
13. Unit 3 Standby Liquid Control Area 1 Smoke Detector	1

TABLE 3.12-2

SPRINKLER SYSTEMS

1. Unit 3 Cable Tunnel
2. Unit 3 Emergency Diesel Oil Day Tank
3. Unit 2/3 Turbine Mezzanine 538' Elevation
4. Unit 2/3 Emergency Diesel Oil Day Tank
5. Hydrogen Seal Oil Area
6. Unit 3 Reactor Feed Pump Area
7. Unit 3 Condensate Feed Pump Area
8. Unit 3 HPCI Area
9. Unit 2/3 Fire Pump Area
10. Unit 3 EHC Area
11. Unit 3 Trackway

TABLE 3.12-3

CO_2 SYSTEMS

1. Unit 3 Emergency Diesel Generator
2. Unit 2/3 Emergency Diesel Generator

TABLE 3.12-4

FIRE HOSE STATIONS

NO.	LOCATION
F22	2/3 Intake
F67A	Reactor Building - 517-ft., Interlock to 2/3 Emergency Diesel Generator
F104	Reactor Building - 589-ft., South East Wall
F105	Reactor Building - 589-ft., South West Wall at Isolation Condenser
F106	Reactor Building - 589-ft., North West at Elevator
F107	Reactor Building - 589-ft., North of Standby Liquid Tank
F108	Reactor Building - 570-ft., Across from Cleanup Demineralizer Precoat Tank
F110	Reactor Building - 570-ft., South West Wall Near RBCCW Tank
F111	Reactor Building - 570-ft., South East Wall at Equipment Hatch
F112	Reactor Building - 545-ft., South East Wall at Equipment Hatch
F113	Reactor Building - 545-ft., South West Wall at Bus 34-1
F114	Reactor Building - 545-ft., North West Wall at Elevator
F116	Reactor Building - 517-ft., South East Wall at Stairway
F117	Reactor Building - 517-ft., South West Wall at Stairway

TABLE 3.12-4 (Continued, page 2)

FIRE HOSE STATIONS

NO.	LOCATION
F118	Reactor Building - 517-ft., West Accumulator Area
F119	Reactor Building - 476-ft., South East Corner 3A LPCI Pump
F122	Reactor Building - 476-ft., South West Corner 3D-LPCI Pump
F129	Turbine Building - 538-ft., Fire Water Regulating Valves
F130	Turbine Building - 538-ft., South West Wall at DC Switch Group Room
F130A	Turbine Building - 538-ft., West Wall of Unit 3 Trackway Equipment Hatch
F131	Turbine Building - 518-ft., West Turbine Trackway
F132	Turbine Building - 517-ft., at 3 "C" R.F.P.
F133	Turbine Building - 517-ft., at 3 "A" R.F.P.
F134	Turbine Building - 517-ft., at CO_2 Tank
F136	Turbine Building - 538-ft., at Freight Elevator
F137	Turbine Building - 538-ft., Behind MCC 39-2
F139	Turbine Building - 495-ft., at 3 "A" CRD Pump
F140	Turbine Building - 469-ft., at 3 "A" Condensate Booster Pump

5.0 DESIGN FEATURES

5.1 Site

Dresden Unit 3 is located at the Dresden Nuclear Power Station which consists of a tract of land of approximately 953 acres located in the northeast quarter of the Morris 15-minute quadrangle (as designated by the United States Geological Survey), Goose Lake Township, Grudy County, Illinois. The tract is situated in portions of Sections 25, 26, 27, 34, 35, and 36 of Township 34 North, Range 8 East of the Third Principal Meridian.

5.2 Reactor

A. The core shall consist of not more than 724 fuel assemblies

B. The reactor core shall contain 177 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% of theoretical density, or Hafnium metal.

5.3 Reactor Vessel

The reactor vessel shall be as described in Table 4.1.1 of the SAR. The applicable design codes shall be as described in Table 4.1.1 of the SAR.

5.4 Containment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the SAR.

B. The secondary containment shall be as described in Section 5.3.2 of the SAR and the applicable codes shall be as described in Section 12.1.1.3 of the SAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.2 of the SAR and the applicable codes shall be as described in Section 12.1.1.3 of the SAR.

5.5 Fuel Storage

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

5.0 DESIGN FEATURES (Cont'd.)

5.6 Seismic Design

The reactor building and all contained engineered safeguards are designed for the maximum credible earthquake ground motion with an acceleration of 20 per cent of gravity. Dynamic analysis was used to determine the earthquake acceleration, applicable to the various elevations in the reactor building.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization, Review, Investigation and Audit

A. The Station Superintendent shall have overall full-time responsibility for safe operation of the facility. During periods when the Station Superintendent is unavailable, he shall designate this responsibility to an established alternate who satisfies the ANSI N18.1 experience requirements for plant manager.

B. The corporate management which relates to the operation of this station is shown in Figure 6.1.1.

C. The normal functional organization for operation of the station shall be as shown in Figure 6.1.2. The shift manning for the station shall be as shown in Table 6.1.1.

D. Qualifications of the station management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971 with the exception of the Radiological Chemical Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September, 1975 and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents. The individual filling the position of Administrative Assistant shall meet the minimum acceptable level for "Technical Manager" as described in 4.2.4 of ANSI N18.1 - 1971.

A fire brigade of at least 5 members shall be maintained on-site at all times. This excludes the shift crew necessary for safe shutdown of the plant and any personnel required for essential functions during a fire emergency.

E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971.

A training program for the fire brigade shall be maintained under the direction of the Operating Engineer and shall meet or exceed the requirements of Section 27 of the NFPA Code - 1975, except for fire brigade training sessions which shall be held at least quarterly.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

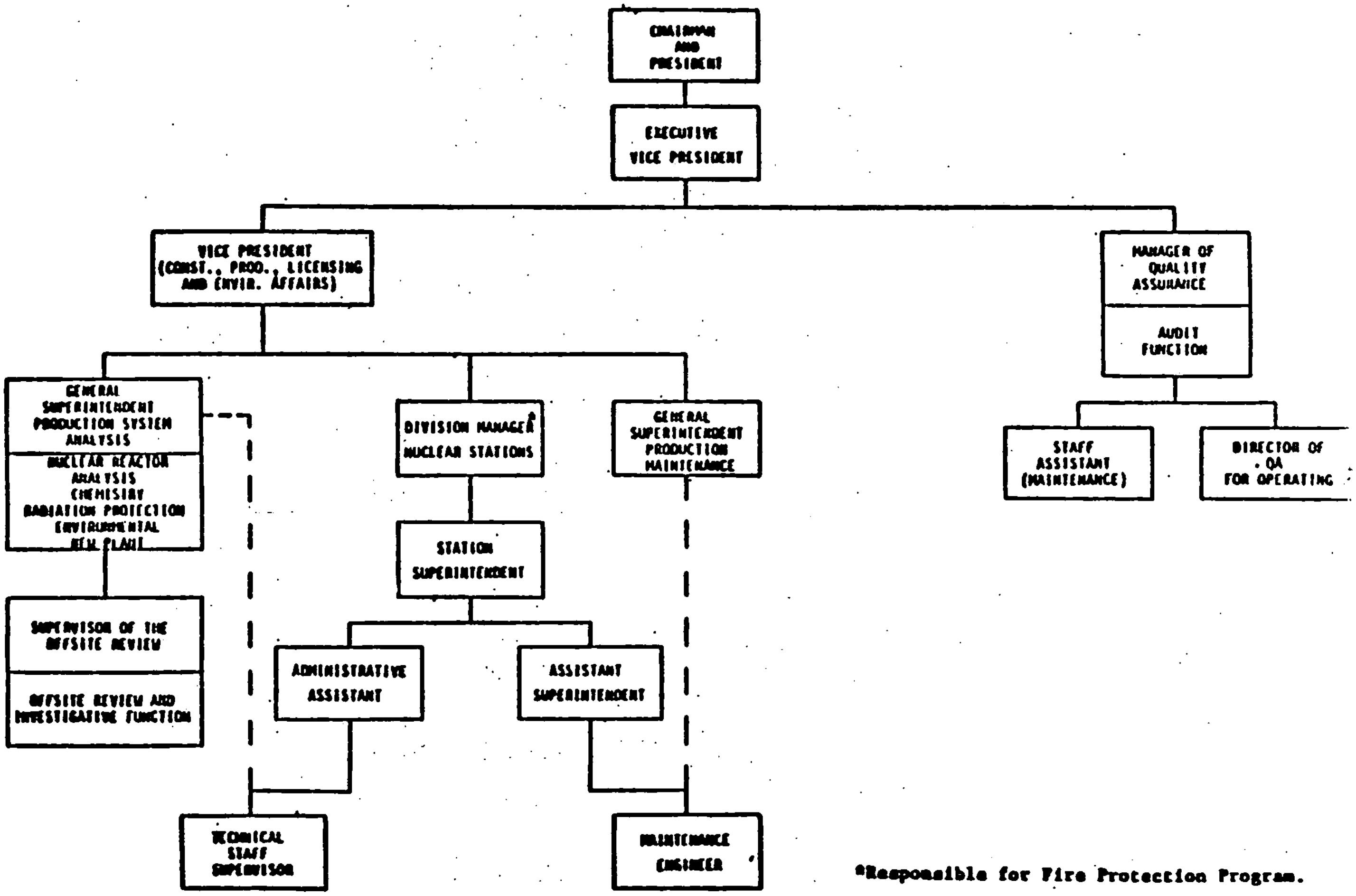
F. Retraining shall be conducted at intervals not exceeding two years.

G. The Review and Investigative Function and the Audit Function of activities affecting quality during facility operations shall be constituted and have the responsibilities and authorities outlined below:

1. The Supervisor of the Offsite Review and Investigative Function shall be appointed by the Vice President of Construction, Production, Licensing, and Environmental Affairs. The Audit Function shall be the responsibility of the Manager of Quality Assurance and shall be independent of operations.

a. Offsite Review and Investigative Function

The Supervisor of the Offsite Review and Investigative Function shall: (i) provide directions for the review and investigative function and appoint a senior participant to provide appropriate direction, (ii) select each participant for this function, (iii) select a complement of more than one participant who collectively possess background and qualifications in the subject matter under review to provide comprehensive interdisciplinary review coverage under this function, (iv) independently review and approve the findings and recommendations developed by personnel performing the review and investigative function, (v) approve and report in a timely manner all findings of noncompliance with NRC requirements and provide recommendations to the Station Superintendent, Division Manager Nuclear Stations, Manager of Quality Assurance, and the Vice President of Construction, Production, Licensing and Environmental Affairs. During periods when the Supervisor of the Offsite Review and Investigative Function is unavailable, he shall designate this responsibility to an established alternate who satisfies the formal training and experience requirements for the supervisor of the Offsite Review and Investigative Function.



*Responsible for Fire Protection Program.

FIGURE 6.1-1

CORPORATE ORGANIZATION

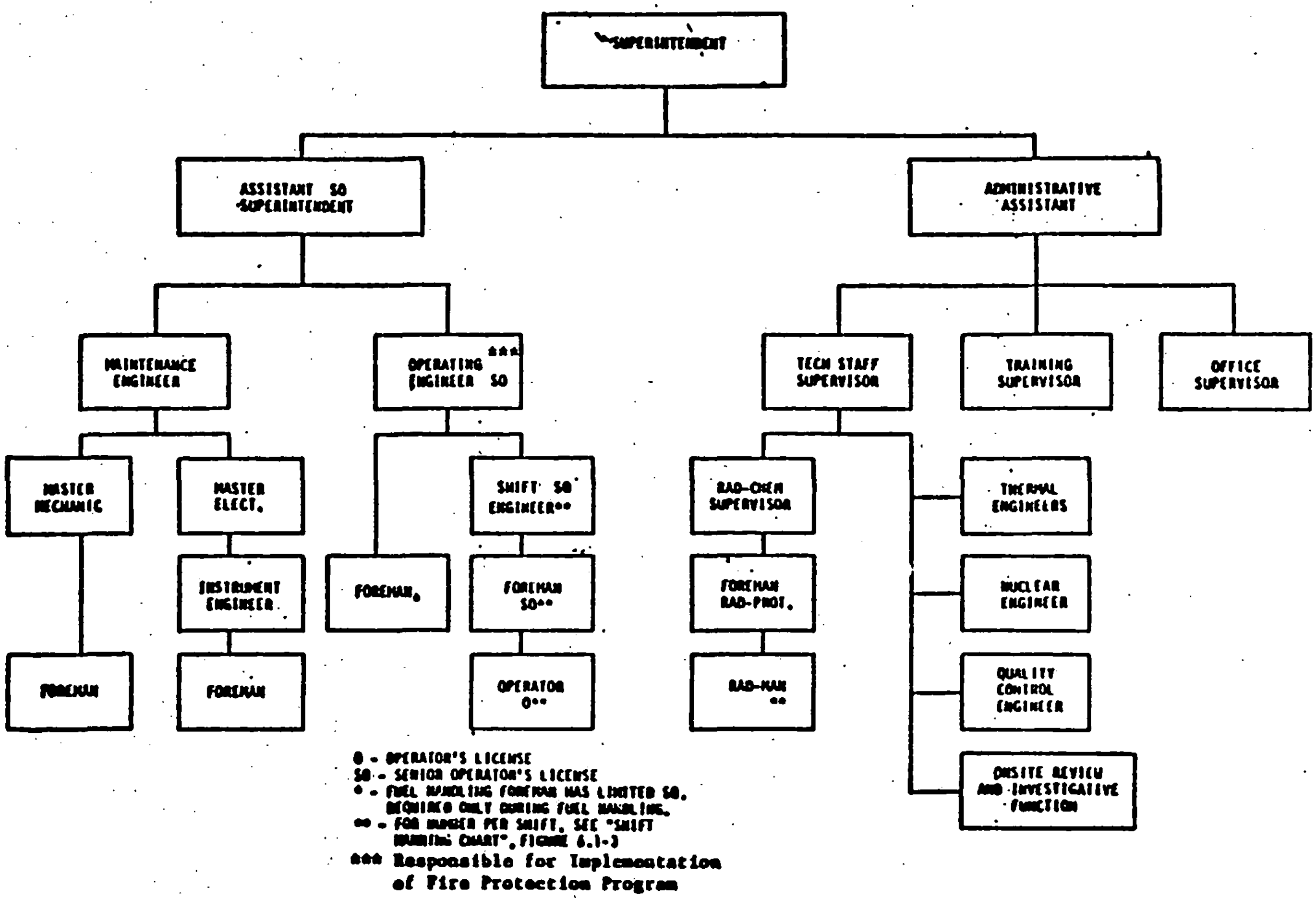

SUPERINTENDENT
ASSISTANT SO SUPERINTENDENT
ADMINISTRATIVE ASSISTANT
MAINTENANCE ENGINEER
OPERATING ENGINEER SO ***
TECH STAFF SUPERVISOR
TRAINING SUPERVISOR
OFFICE SUPERVISOR
MASTER MECHANIC
MASTER ELECT.
SHIFT SO ENGINEER**
RAD-CHEM SUPERVISOR
THERMAL ENGINEERS
INSTRUMENT ENGINEER
FOREMAN*
FOREMAN SO**
FOREMAN RAD-PROT.
NUCLEAR ENGINEER
FOREMAN
FOREMAN
OPERATOR O**
RAD-MAN **
QUALITY CONTROL ENGINEER
ONSITE REVIEW AND INVESTIGATIVE FUNCTION
O - OPERATOR'S LICENSE
SO - SENIOR OPERATOR'S LICENSE
* - FUEL HANDLING FOREMAN HAS LIMITED SO, REQUIRED ONLY DURING FUEL HANDLING.
** - FOR NUMBER PER SHIFT, SEE "SHIFT MANNING CHART", FIGURE 6.1-3
*** Responsible for Implementation of Fire Protection Program

FIGURE 6.1-2

TABLE 6.1.1

MINIMUM SHIFT MANNING CHART***

CONDITION OF ONE UNIT	CONDITION OF SECOND UNIT	CONDITION OF THIRD UNIT	NO. OF MEN IN EACH POSITION LSO*	STA	LO*	NON-LIC.	RAD MEN
COLD SHUTDOWN	Cold Shutdown	Cold Shutdown	1*	0	3	5	1
	Cold Shutdown	Refuel	1**	0	3	5	1
	Cold Shutdown	Above Cold Shutdown	1	1	3	5	1
	Refuel	Refuel	2**	0	3	5	1
	Refuel	Above Cold Shutdown	2	1	3	5	1
	Above Cold Shutdown	Above Cold Shutdown	2	1	3	5	1
REFUEL	Refuel	Refuel	2	0	4	5	1
	Refuel	Above Cold Shutdown	2	1	4	5	1
	Above Cold Shutdown	Above Cold Shutdown	2	1	4	5	1
ABOVE COLD SHUTDOWN	Above Cold Shutdown	Above Cold Shutdown	3	1	4	5	1

STA - Shift Technical Advisor

LSO - Licensed Senior Operator

LO - Licensed Operator

NON-LIC. - Equipment Operators and Equipment Attendants

RAD MEN - Radiation Protection Men

* - Shall not operate units on which they are not licensed.

** - Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE OPERATIONS.

*** - Shift crew 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 on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

The responsibilities of the personnel performing this function are stated below. The Offsite Review and Investigative Function shall review:

(1) The safety evaluations for 1) changes to procedures, equipment or systems as described in the safety analysis report and 2) tests or experiments completed under the provision of 10 CFR Section 50.59 to verify that such actions did not constitute unreviewed safety questions. Proposed changes to the Quality Assurance Program description shall be reviewed and approved by the Manager of Quality Assurance.

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

(3) Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59 10 CFR.

(4) Proposed changes in Technical Specifications or NRC operating licenses.

(5) Noncompliance with NRC requirements, or of internal procedures or instructions having nuclear safety significance.

(6) Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function.

(7) Reportable Occurrences requiring 24 hour notification to the Commission.

(8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems or components.

(9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such changes.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Superintendent, Division Manager - Nuclear Stations and Manager of Quality Assurance.

b. Audit Function

The Audit Function shall be the responsibility of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance for Operating and to the Staff Assistant to the Manager of Quality Assurance for maintenance quality assurance activities.

Either shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within a period of two years or less as designated below.

(1) Audit of the Conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per year.

(2) Audit of the adherence to procedures, training and qualification of the station staff at least once per year.

(3) Audit of the results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or methods of operation that affect nuclear safety at least once per six months.

(4) Audit of the performance of activities required by the Quality Assurance Program to meet the Criteria of Appendix "B", 10 CFR 50.

(5) Audit of the Facility Emergency Plan and implementing procedures.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(6) Audit of the Facility Security Plan and implementing procedures.

(7) Audit onsite and offsite reviews.

(8) Audit of Facility Fire Protection Program and implementing procedures at least once per 24 months.

(9) Report all findings of noncompliance with NRC requirements and recommendations and results of each audit to the Station Superintendent, the Division Manager-Nuclear Stations, Manager of Quality Assurance, the General Superintendent of Production Systems Analysis, and to the Vice President of Construction, Production, Licensing and Environmental Affairs.

c. Authority

The Manager of Quality Assurance reports to the Executive Vice-President and the Supervisor of the Offsite Review and Investigative Function reports to the General Superintendent of Production Systems Analysis. Either the Manager of Quality Assurance or the Supervisor of the Offsite Review and Investigative Function has the authority to order unit shutdown or request any other action which he deems necessary to avoid unsafe plant conditions.

d. Records

(1) Reviews, audits and recommendations shall be documented and distributed as covered in 6.1.G.1.a and 6.1.G.1.b.

(2) Copies of documentation, reports, and correspondence shall be kept on file at the station.

e. Procedures

Written administrative procedures shall be prepared and maintained for the off-site reviews and investigative functions described in Specifications 6.1.G.1.a. These procedures shall cover the following:

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(1) Content and method of submission of presentations to the Supervisor of the Offsite Review and Investigative Function.

(2) Use of committees and consultants.

(3) Review and approval.

(4) Detailed listing of items to be reviewed.

(5) Method of (a) appointing personnel, (b) performing reviews, investigations, (c) reporting findings and recommendations of reviews and investigations, (d) approving reports, and (e) distributing reports.

(6) Determining satisfactory completion of action required based on approved findings and recommendations reported by personnel performing the review and investigative function.

f. Personnel

(1) The persons, including consultants, performing the review and investigative function, in addition to the Supervisor of the Offsite Review and Investigative Function, shall have expertise in one or more of the following disciplines as appropriate for the subject or subjects being reviewed and investigated.

(a) nuclear power plant technology
(b) reactor operations
(c) utility operations
(d) power plant design
(e) reactor engineering
(f) radiological safety
(g) reactor safety analysis
(h) instrumentation and control
(i) metallurgy
(j) any other appropriate disciplines required by unique characteristics of the facility.

(2) Individuals performing the Review and Investigative Function shall possess a minimum formal training and experience as listed below for each discipline.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(a) Nuclear Power Plant Technology

Engineering graduate or equivalent with 5 years experience in the nuclear power field design and/or operation.

(b) Reactor Operations

Engineering graduate or equivalent with 5 years experience in nuclear power plant operations.

(c) Utility Operations

Engineering graduate or equivalent with at least 5 years of experience in utility operation and/or engineering.

(d) Power Plant Design

Engineering graduate or equivalent with at least 5 years of experience in power plant design and/or operation.

(e) Reactor Engineering

Engineering graduate or equivalent. In addition, at least 5 years of experience in nuclear plant engineering, operation, and/or graduate work in nuclear engineering or equivalent in reactor physics is required.

(f) Radiological Safety

Engineering graduate or equivalent with at least 5 years of experience in radiation control and safety.

(g) Safety Analysis

Engineering graduate or equivalent with at least 5 years of experience in nuclear engineering.

(h) Instrumentation and Control

Engineering graduate or equivalent with at least 5 years of experience in instrumentation and control design and/or operation.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(i) Metallurgy

Engineering graduate or equivalent with at least 5 years of experience in the metallurgical field.

(3) The Supervisor of the Offsite Review and Investigative Function shall have experience and training which satisfy ANSI N18.1 - 1971 requirements for plant managers.

2. The Onsite Review and Investigative Function shall be supervised by the Station Superintendent.

a. Onsite Review and Investigative Function

The Station Superintendent shall: (i) provide direction for the Review and Investigative Function and appoint the Technical Staff Supervisor, or other comparably qualified individual as a senior participant to provide appropriate direction; (ii) approve participants for this function; (iii) assure that a complement of more than one participant who collectively possess background and qualifications in the subject matter under review are selected to provide comprehensive inter-disciplinary review coverage under this function; (iv) independently review and approve the findings and recommendations developed by personnel performing the Review and Investigative Function; (v) report all findings of noncompliance with NRC requirements, and provide recommendations to the Division Manager-Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function; and (vi) submit to the Offsite Review and Investigative Function for concurrence in a timely manner, those items described in Specification 6.1.G.1.a which have been approved by the Onsite Review and Investigative Function.

The responsibilities of the personnel performing this function are stated below:

(1) Review of: 1) procedures required by Specification 6.2 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Plant Superintendent to affect nuclear safety.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(2) Review of all proposed tests and experiments that affect nuclear safety.

(3) Review of all proposed changes to the Technical Specifications.

(4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

(5) Investigate all noncompliance with NRC requirements and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Division Manager-Nuclear Stations and to the Supervisor of the Offsite Review and Investigative Function.

(6) Review of facility operations to detect potential safety hazards.

(7) Performance of special reviews and investigations and reports thereon as requested by the Supervisor of the Offsite Review and Investigative Function.

(8) Review the Station Security Plan and shall submit recommended changes to the Division Manager-Nuclear Stations.

(9) Review the Emergency Plan and station implementing procedures and shall submit recommended changes to the Division Manager-Nuclear Stations.

(10) Review of reportable occurrences and actions taken to prevent recurrence.

b. Authority

The Technical Staff Supervisor is responsible to the Station Superintendent and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

when in his opinion such action is necessary. The Station Superintendent shall follow such recommendations or select a course of action that is more conservative regarding safe operation of the facility. All such disagreements shall be reported immediately to the Division Manager-Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function.

c. Records

(1) Reports, reviews, investigations, and recommendations shall be documented with copies to the Division Manager-Nuclear Stations, the Supervisor of the Offsite Review and Investigative Function, the Station Superintendent and the Manager of Quality Assurance.

(2) Copies of all records and documentation shall be kept on file at the station.

d. Procedures

Written administrative procedures shall be prepared and maintained for conduct of the Onsite Review and Investigative Function. These procedures shall include the following:

(1) Content and method of submission and presentation to the Station Superintendent, Division Manager-Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function.

(2) Use of committees.

(3) Review and approval.

(4) Detailed listing of items to be reviewed.

(5) Procedures for administration of the quality control activities.

(6) Assignment of responsibilities.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

e. Personnel

(1) The personnel performing the Onsite Review and Investigative Function, in addition to the Station Superintendent, shall consist of persons having expertise in:

(a) nuclear power plant technology
(b) reactor operations
(c) reactor engineering
(d) radiological safety and chemistry
(e) instrumentation and control
(f) mechanical and electric systems.

(2) Personnel performing the Onsite Review and Investigative Function shall meet minimum acceptable levels as described in ANSI N18.1-1971, Sections 4.2 and 4.4.

H. 1. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified off-site licensee personnel or an outside fire protection firm.

2. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

6.2 Plant Operating Procedures

A. Detailed written procedures including applicable checkoff lists covering items listed below shall be prepared, approved, and adhered to:

1. Normal startup, operation, and shutdown of the reactor and other systems and components involving nuclear safety of the facility.

2. Refueling operations.

3. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components including responses to alarms, suspected primary system leaks, and abnormal reactivity changes.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

4. Emergency conditions involving potential or actual release of radioactivity - "Generating Stations Emergency Plan" and station emergency and abnormal procedures.

5. Instrumentation operation which could have an effect on the safety of the facility.

6. Preventive and corrective maintenance operations which could have an effect on the safety of the facility.

7. Surveillance and testing requirements.

8. Tests and experiments.

9. Procedure to ensure safe shutdown of the plant.

10. Station Security Plan and implementing procedures.

11. Fire Protection Program implementation.

B. Radiation control procedures shall be maintained, made available to all station personnel and adhered to. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.

C. 1. Procedures for items identified in Specification 6.2.A and any changes to such procedures shall be reviewed and approved by the Operating Engineer and the Technical Staff Supervisor in the areas of operation, fuel handling, or instrument maintenance, and by Maintenance Engineer and Technical Staff Supervisor in the areas of plant maintenance and plant inspection. Procedures for items identified in Specification 6.2.B and any changes to such procedures shall be reviewed and approved by the Technical Staff Supervisor and the Radiological Chemical Supervisor. At least one person approving each of the above procedures shall hold a valid senior operator's license. In addition, these procedures and changes thereto must have authorization by the Station Superintendent before being implemented.

2. Work and instruction type procedures which implement approved maintenance or modification procedures shall be approved and authorized by the Maintenance Engineer where the written authority has been provided by the Station

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

Superintendent. The "Maintenance/Modification Procedure" utilized for safety related work shall be so approved only if procedures referenced in the "Maintenance/Modification Procedure" have been approved as required by 6.2.A. Procedures which do not fall within the requirements of 6.2.A or 6.2.B may be approved by the Department Heads.

D. Temporary changes to procedures 6.2.A and 6.2.B above may be made provided:

1. The intent of the original procedure is not altered.

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

3. The change is documented, reviewed by the Onsite Review and Investigative Function and approved by the Station Superintendent within 14 days of implementation.

E. Drills of the emergency procedures described in Specification 6.2.A.4 shall be conducted quarterly. These drills will be planned so that during the course of the year, communication links are tested and outside agencies are contacted.

6.3 Action to be Taken in the Event of a Reportable Occurrence in Operation

Any reportable occurrence shall be promptly reported to the Division Manager-Nuclear Stations or his designated alternate. The incident shall be promptly reviewed pursuant to Specification 6.1.G.2.a(5) and a separate report for each reportable occurrence shall be prepared in accordance with the requirements of Specification 6.6.B.

6.4 Action to be Taken in the Event a Safety Limit is Exceeded

If a safety limit is exceeded, the reactor shall be shut down immediately and reactor operation shall not be resumed until authorized by the NRC. The conditions of shutdown shall be promptly reported to the Division Manager-Nuclear Stations or his designated alternate. The incident shall be reviewed pursuant to Specification 6.1.G.1.a and 6.1.G.2.a and a separate report for each occurrence shall be prepared in accordance with Specification 6.6.B.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

6.5 Plant Operating Records

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least five years.

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

2. Records of principal maintenance activities, including inspection and repair, regarding principal items of equipment pertaining to nuclear safety.

3. Records and reports of reportable and safety limit occurrences.

4. Records and periodic checks, inspection and/or calibrations performed to verify the Surveillance Requirements (See Section 4 of these Specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded.

5. Records of changes made to the equipment or reviews of tests and experiments to comply with 10 CFR 50.59.

6. Records of radioactive shipments.

7. Records of physic tests and other tests pertaining to nuclear safety.

8. Records of changes to operating procedures.

9. Shift Engineers Logs.

10. By-product material inventory records and source leak test results.

B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life of the plant.

1. Substitution or replacement of principal items of equipment pertaining to nuclear safety.

2. Changes made to the plant as it is described in the Safety Analysis Report.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

3. Records of new and spent fuel inventory and assembly histories.

4. (Deleted)

5. Updated, corrected, and as-built drawings of the plant.

6. Records of plant radiation and contamination surveys.

7. Records of off-site environmental monitoring surveys.

8. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant in accordance with 10 CFR 20.

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

10. Records of transient or operational cycling for those components that have been designed to operate safely for a limited number of transient or operational cycles.

11. Records of individual staff members indicating qualifications, experience, training and retraining.

12. Inservice inspections of the reactor coolant system.

13. Minutes of meetings and results of reviews performed by the off-site and on-site review functions.

14. Records for Environmental Qualification which are covered under the provisions of paragraph 6.7.

6.6 REPORTING REQUIREMENTS

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

A. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

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

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

2. A tabulation shall be submitted 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, (See note); 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 totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

Note: This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experiences shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, US Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the appropriate Regional Office, to arrive no later than the 15th of each month following the calendar month covered by the report.

B. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. In general, the importance of an occurrence with respect to safety significance determines the immediacy of reporting required. In some cases, however, the significance of an event may not be obvious at the time of its occurrence. In such cases, the NRC shall be informed promptly of an increased significance in the licensee's assessment of the event. In addition, supplemental reports may be required to fully describe final resolution of occurrence. 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. Prompt Notification With Written Followup

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 Director of the appropriate Regional Office, or his designate no later than the first working day following the event, with a written followup report within two weeks. The written followup 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.

a. 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

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

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 B.1.e., B.1.f., or B.2.a. below.

b. 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 2.b. below.

c. 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.

d. Reactivity anomalies, involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation, greater than or equal to 1% delta k/k; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if sub-critical, an unplanned reactivity insertion of more than 0.5% delta k/k or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

of the functional requirements of systems required to cope with accidents analyzed in the SAR.

Note: For items B.1.e. and B.1.f. reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items B.2.b. and B.2.c. below.

g. Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

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

2. Thirty Day Written Reports

The reportable occurrences discussed below have lesser immediate importance than those described under B.1. above. Such events shall be the subject of written reports to the Director of 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.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

a. 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.

b. 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 B.2.a. and B.2.b. need not be reported except where test results themselves reveal a degraded mode above.

c. 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.

d. Abnormal degradation of systems other than those specified in item B.1.e. 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.

C. Unique Reporting Requirements

1. Radioactive Effluent Release Report

A report shall be submitted to the Commission within 60 days after January 1 and July 1 of each year specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents during the previous 6 months. The format and content of the report shall be in accordance with Regulatory Guide 1.21 (Revision 1) dated June 1974.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

2. Environmental Radioactivity Data

a. Standard Radiological Monitoring Program

(1) Non-Routine Report

(a) If a confirmed measured radionuclide concentration in an environmental sampling medium averaged over any calendar quarter sampling period exceeds the reporting level given in Table 4.8-1 and if the radioactivity is attributable to plant operation, a written report shall be submitted to the Director of the NRC Regional Office, with a copy to the Director, Office of Nuclear Reactor Regulation, within 30 days from the end of the quarter. When more than one of the radionuclides in Table 4.8-1 are detected in the medium, the reporting level shall have been exceeded if

$$\frac{\Sigma C_i}{RL_i} \text{ equal to or greater than } 1$$

where C is the concentration of the i^{th} radionuclide in the medium and RL is the reporting level of radionuclide i.

(b) If radionuclides other than those in Table 4.8-1 are detected and are due to plant effluents, a reporting level is exceeded if the potential annual dose to an individual is equal to or greater than the design objective doses of 10 CFR 50, Appendix I.

(c) This report shall include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous effect.

(2) Annual Operating Report

An annual report containing the data taken in the standard radiological monitoring program (Table 4.8-1) shall be submitted by March 31 of the next year. The content of the report shall include:

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(a) Results of environmental sampling summarized on a quarterly basis following the format of Regulatory Guide 4.8 Table 1 (December 1975); (individual sample results will be retained at the station);

(b) An assessment of the monitoring results and radiation dose via the principal pathways of exposure resulting from plant emissions of radioactivity; and

(c) Results of the census to determine the locations of animals producing milk for human consumption.

b. Environmental Dose Pathways Studies (EDPS)

The EDPS schedule for Dresden Station is May 1978 through April 1979 with the project report submitted by December 31, 1979.

3. Special Reports

Special reports shall be submitted as indicated in Table 6.6.1.

TABLE 6.6.1

SPECIAL REPORTS

	AREA	SPECIFICATION REFERENCE	SUBMITTAL DATE
a.	Response time of safety related instruments (2)	1.0.E (Dres. 1)	Annual Report
b.	Main steam isolation valve and feedwater power operated isolation valves closure times (2)	3.7.B.1.c (Dres. 1)	Annual Report
c.	Primary Coolant leakage to Drywell (4)	4.6.D Bases	5 years (1)
d.	In-Service Inspection Evaluation (4)	Table 4.6.1	5 years (1)
e.	Evaluation of ECCS operation (4)	3.3.F Bases	Upon completion of initial testing
f.	Failed Fuel Detection (4)	3.2 Bases	5 years (1)
g.	Main Steam Line Leakage to Steam Tunnel (4)	4.6.D Bases	5 years (1)
h.	In-service Inspection Development (4)	4.6.1 Bases	5 years (1)
i.	In-Service Inspection of Sensitized Stainless Steel Components (3)	4.6.F	4 years (1)
j.	Secondary Containment Leak Rate Test (4)	3.7.C.1	within 90 days after completion of each test
k.	High off-gas discharge rate (2)	3.8.A.4 (Dresden 1)	within 24 hours of occurrence
l.	Radioactive Source Leak Testing (5)	3.8.F	Annual Report

NOTES:

1. The report shall be submitted within the period of time listed based on the commercial service date as the starting point.
2. Dresden 1 only
3. Dresden 2 only
4. Dresden 2 and 3 only.
5. The report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

6.7 ENVIRONMENTAL QUALIFICATION

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-25 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.