PACKAGE 3.1

REACTIVITY CONTROL SYSTEMS

CROSS - REFERENCE

CURRENT TECHNICAL SPECIFICATIONS

TO

IMPROVED TECHNICAL SPECIFICATIONS

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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 3.1				
3.1.A.1.a (1)		LCO	3.4.4	
3.1.A.1.a (1)		LCO	3.4.18	
3.1.A.1.a (2)		LCO	3.4.4	
New		SR	3.4.18.1	
New		SR	3.4.18.2	
New		SR	3.4.4.1	
3.1.A.1.b		LCO	3.4.5	
New		LCO	3.4.5	
New		SR	3.4.5.1	
New		SR	3.4.5.2	
New		SR	3.4.5.3	
3.1.A.1.c		LCO	3.4.6	
3.1.A.1.c		LCO	3.4.7	
3.1.A.1.c		(Partial)	Relocated - Bases	
New		LCO	3.4.7	
New		SR	3.4.6.1	
New		SR	3.4.6.2	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
New		SR	3.4.6.3	
New		SR	3.4.7.1	
New		SR	3.4.7.2	
New		SR	3.4.7.3	
3.1.A.1.d(1)		LCO	3.4.8	
3.1.A.1.d(1)		(Partial)	Relocated - Bases	
3.1.A.1.d(2)		LCO	3.4.8	
3.1.A.1.d(2)		LCO	3.4.13	
New		LCO	3.4.8	
New		SR	3.4.8.1	
New		SR	3.4.8.2	
3.1.A.2.a (1)		LCO	3.4.9	
3.1.A.2.a (1)		(Partial)	Relocated - Bases	
New		LCO	3.4.9	
3.1.A.2.a (2)		LCO	3.4.9	
3.1.A.2.a (3)		LCO	3.4.9	
New		LCO	3.4.9	
New		SR	3.4.9.1	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
3.1.A.2.b (1)		LCO	3.4.10	
New		LCO	3.4.10	
3.1.A.2.b (2)			Deleted	
3.1.A.2.c (1)		LCO	3.4.11	
3.1.A.2.c (2)		LCO	3.4.12	
3.1.A.2.c (2)		(Partial)	Relocated - Bases	
New		LOC	3.4.12	
New		SR	3.4.12.1	
New		SR	3.4.12.2	
New		SR	3.4.12.3	
3.1.A.2.c (3)		LCO	3.4.13	
3.1.A.2.c (3)		(Partial)	Relocated - Bases	
New		LCO	3.4.13	
New		SR	3.4.13.1	
New		SR	3.4.13.2	
New		SR	3.4.13.3	
New		SR	3.4.13.4	
3.1.A.3			Relocated - TRM	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
New		LCO	3.4.2	
New		SR	3.4.2.1	
3.1.B.1.a		LCO	3.4.3	
3.1.B.1.b		LCO	3.4.3	
New		LCO	3.4.3	
New		SR	3.4.3.1	
3.1.B.2			Relocated - PTLR	
3.1.B.3			Relocated - PTLR	
3.1.C.1		LCO	3.4.16	
New		LCO	3.4.16	
New		SR	3.4.16.1	
New		SR	3.4.16.2	
New		SR	3.4.16.3	
New		SR	3.4.16.4	
3.1.C.2.a		LCO	3.4.14	
3.1.C.2.b		LCO	3.4.14	
3.1.C.2.b		(Partial)	Relocated - Bases	
3.1.C.2.c		LCO	3.4.14	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
3.1.C.2.d		LCO	3.4.14	
3.1.C.2.e		LCO	3.4.14	
3.1.C.2.e		SR	3.4.14.2	
3.1.C.3		LCO	3.4.15	
New		LCO	3.4.15	
3.1.D.1		LCO	3.4.17	
3.1.D.2		LCO	3.4.17	
New		LCO	3.4.17	
3.1.D.3			Deleted	
3.1.E			Deleted	
3.1.F.1		LCO	3.1.3	
3.1.F.1		(Partial)	Relocated - COLR	
3.1.F.2		LCO	3.1.3	
3.1.F.2		(Partial)	Relocated - COLR	
3.1.F.3.a		LCO	3.1.3	
3.1.F.3.b			Relocated - Bases	
3.1.F.3.c			Deleted	
New		LCO	3.1.3	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
New		SR	3.1.3.1	
New		SR	3.1.3.2	
New		SR	3.1.3.3	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 3.10				
3.10.A.1		LCO	3.1.1	
3.10.A.1			Relocated - COLR	
3.10.A.2		LCO	3.1.1	
3.10.A.3		LCO	3.1.1	
New		SR	3.1.1.1	
New		LCO	3.1.2	
New		SR	3.1.2.1	
3.10.B.1		LCO	3.2.1	
3.10.B.1		LCO	3.2.2	
3.10.B.1		(Partial)	Relocated - COLR	
3.10.B.2		SR	3.2.1.1	
3.10.B.2		SR	3.2.1.2	
3.10.B.2		SR	3.2.2.1	
3.10.B.2		SR	3.2.3.2	
3.10.B.3.a		LCO	3.2.1	
3.10.B.3.a		LCO	3.2.2	
3.10.B.3.a		(Partial)	Relocated - COLR	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
New		LCO	3.2.1	
New		LCO	3.2.2	
3.10.B.3.b		LCO	3.2.1	
3.10.B.3.b		(Partial)	Relocated - COLR	
New		LCO	3.2.1	
3.10.B.3.c		LCO	3.2.2	
3.10.B.3.d		SR	3.2.1.2	
3.10.B.3.d		(Partial)	Relocated - COLR	
3.10.B.4		LCO	3.2.3	
New		LCO	3.2.3	
3.10.B.5		LCO	3.2.3	
3.10.B.6		LCO	3.2.3	
New		LCO	3.2.3	
3.10.B.7		LCO	3.2.3	
3.10.B.8		LCO	3.2.3	
3.10.B.9			Relocated - TRM	
New		SR	3.2.3.1	
3.10.C.1		LCO	3.2.4	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
New		LCO	3.2.4	
New		SR	3.2.4.1	
3.10.C.2			Deleted	
3.10.C.3			Deleted	
3.10.C.4		SR	3.2.4.2	
3.10.C.4		LCO	3.3.1 D	
3.10.C.4		(Partial)	Relocated - Bases	
3.10.D.1		LCO	3.1.5	
New		LCO	3.1.5	
New		SR	3.1.5.1	
3.10.D.2		LCO	3.1.6	
New		LCO	3.1.6	
New		SR	3.1.6.1	
New		SR	3.1.6.2	
New		SR	3.1.6.3	
3.10.D.3		LCO	3.1.5	
3.10.D.3		LCO	3.1.6	
3.10.D.3		LCO	3.1.8	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
New		LCO	3.1.8	
New		SR	3.1.8.1	
New		SR	3.1.8.2	
New		SR	3.1.8.3	
New		SR	3.1.8.4	
3.10.E.1		LCO	3.1.4	
3.10.E.2			Deleted	
3.10.F.1		LCO	3.1.7	
3.10.F.1		(Partial)	Relocated - Bases	
3.10.F.2		LCO	3.1.7	
3.10.F.3		LCO	3.1.7	
3.10.F.4		LCO	3.1.7	
3.10.F.5		LCO	3.1.4	
New		LCO	3.1.7	
3.10.G.1			Relocated - Bases	
3.10.G.2		LCO	3.1.4	
3.10.G.3		LCO	3.1.4	
3.10.G.4		LCO	3.1.4	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
3.10.G.5		LCO	3.1.4	
3.10.G.5		(Partial)	Relocated - Bases	
3.10.G.6		LCO	3.1.4	
New		LCO	3.1.4	
3.10.H		SR	3.1.4.3	
3.10.I.1			Relocated - TRM	
3.10.I.2			Relocated - TRM	
3.10.I.3			Relocated - TRM	
3.10.J		LCO	3.4.1	
3.10.J		(Partial)	Relocated - COLR	
3.10.J		SR	3.4.1.1	
3.10.J		SR	3.4.1.2	
3.10.J		SR	3.4.1.3	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 4.9				
4.9		LCO	3.1.2	
4.9		SR	3.1.2.2	

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Figure 5.6-8		FIGURE	4.3.1-8	
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Figure 5.6-11		FIGURE	4.3.1-11	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section Table				
Table 1-1		TABLE	Table 1.1-1	
Table 1-1	Note *	LCO	3.9.1	
New		LCO	3.9.1	
Table 1-1	Note *	(Partial)	Relocated - COLR	
Table 1-1	Note **		Deleted	
Table 3.5-1	9	TABLE	3.3.5-1	Note c
Table 3.5-1	1	TABLE	3.3.2-1	1c
Table 3.5-1	2a	TABLE	3.3.2-1	2c
Table 3.5-1	2b	TABLE	3.3.2-1	4b
Table 3.5-1	3	TABLE	3.3.2-1	1d
Table 3.5-1	4	TABLE	3.3.2-1	1e
Table 3.5-1	4	TABLE	3.3.2-1	Note b
Table 3.5-1	5	TABLE	3.3.2-1	4c
Table 3.5-1	6	TABLE	3.3.2-1	4d
Table 3.5-1	7	SR	3.6.8.1	
Table 3.5-1	8		Relocated - TRM	
Table 3.5-1	9	TABLE	3.3.5-1	3

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-1	10	SR	3.3.4.2	
Table 3.5-2A	1	TABLE	3.3.1-1	1
Table 3.5-2A	2a	TABLE	3.3.1-1	2a
Table 3.5-2A	2b	TABLE	3.3.1-1	2b
Table 3.5-2A	3	TABLE	3.3.1-1	3a
Table 3.5-2A	4	TABLE	3.3.1-1	3b
Table 3.5-2A	5	TABLE	3.3.1-1	4
Table 3.5-2A	6	TABLE	3.3.1-1	5
Table 3.5-2A	7	TABLE	3.3.1-1	6
Table 3.5-2A	8	TABLE	3.3.1-1	7
Table 3.5-2A	9	TABLE	3.3.1-1	8a
Table 3.5-2A	10	TABLE	3.3.1-1	8b
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Table 3.5-2A	12	TABLE	3.3.1-1	10
Table 3.5-2A	13	TABLE	3.3.1-1	14
Table 3.5-2A	14	TABLE	3.3.1-1	13
Table 3.5-2A	15	TABLE	3.3.1-1	12
Table 3.5-2A	16a	TABLE	3.3.1-1	11a
Table 3.5-2A	16b	TABLE	3.3.1-1	11b

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-2A	17	TABLE	3.3.1-1	15
Table 3.5-2A	18	TABLE	3.3.1-1	19
Table 3.5-2A	19	TABLE	3.3.1-1	17
Table 3.5-2A	20	TABLE	3.3.1-1	17
Table 3.5-2A	New Func	TABLE	3.3.1-1	16
Table 3.5-2A	New Func	TABLE	3.3.1-1	18
Table 3.5-2A	Act 1	LCO	3.3.1 B	
Table 3.5-2A	Action 1	LCO	3.3.1 M	
Table 3.5-2A	Action 2	LCO	3.3.1 D	
Table 3.5-2A	Action 2	LCO	3.3.1 E	
Table 3.5-2A	Act 2	SR	3.2.4.2	
Table 3.5-2A	Act 2c	SR	3.2.4.2	
Table 3.5-2A	Act 3	LCO	3.3.1 F	
Table 3.5-2A	New Action	LCO	3.3.1 G	
Table 3.5-2A	Action 4	LCO	3.3.1 H	
Table 3.5-2A	New Action	LCO	3.3.1 I	
Table 3.5-2A	Action 5	LCO	3.3.1 J	
Table 3.5-2A	Action 6	LCO	3.3.1 E	
Table 3.5-2A	Action 6	LCO	3.3.1 K	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-2A	Action 6	LCO	3.3.1 N	
Table 3.5-2A	Action 7	LCO	3.3.1 O	
Table 3.5-2A	Act 8	LCO	3.3.1 C	
Table 3.5-2A	Action 9a	LCO	3.3.1 S	
Table 3.5-2A	Action 9a	LCO	3.3.1.P	
Table 3.5-2A	Action 9b	LCO	3.3.1 P	
Table 3.5-2A	Action 10	LCO	3.3.1 C	
Table 3.5-2A	Act 10	LCO	3.3.1 P	
Table 3.5-2A	Action11	LCO	3.3.1 L	
Table 3.5-2A	New Action	LCO	3.3.1 Q	
Table 3.5-2A	New Action	LCO	3.3.1 R	
Table 3.5-2A	New Action	LCO	3.3.1 S	
Table 3.5-2A	Note a	TABLE	3.3.1-1	Note a
Table 3.5-2A	Note b	TABLE	3.3.1-1	Note b
Table 3.5-2A	Note c	TABLE	3.3.1-1	Note d
Table 3.5-2A	Note d	TABLE	3.3.1-1	Note i
Table 3.5-2A	New Note	TABLE	3.3.1-1	Note e
Table 3.5-2A	New Note	TABLE	3.3.1-1	Note f
Table 3.5-2A	New Note	TABLE	3.3.1-1	Note g

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Table 3.5-2A	New Note	TABLE	3.3.1-1	Note h
Table 3.5-2A	New Note	TABLE	3.3.1-1	Note j
Table 3.5-2B	1a	TABLE	3.3.2-1	1a
Table 3.5-2B	1b	TABLE	3.3.2-1	1c
Table 3.5-2B	1c	TABLE	3.3.2-1	1e
Table 3.5-2B	1d	TABLE	3.3.2-1	1d
Table 3.5-2B	1e	TABLE	3.3.2-1	1b
Table 3.5-2B	2a	TABLE	3.3.2-1	2a
Table 3.5-2B	2b	TABLE	3.3.2-1	2c
Table 3.5-2B	2c	TABLE	3.3.2-1	2b
Table 3.5-2B	3a	TABLE	3.3.2-1	3c
Table 3.5-2B	3b	TABLE	3.3.2-1	3a
Table 3.5-2B	3c	TABLE	3.3.2-1	3b
Table 3.5-2B	4a	TABLE	3.3.5-1	5
Table 3.5-2B	4b	TABLE	3.3.5-1	1
Table 3.5-2B	4c	TABLE	3.3.5-1	6
Table 3.5-2B	4d	TABLE	3.3.5-1	4
Table 3.5-2B	4e	TABLE	3.3.5-1	3
Table 3.5-2B	4f	TABLE	3.3.5-1	2

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Table 3.5-2B	5b	TABLE	3.3.2-1	4b
Table 3.5-2B	5c	TABLE	3.3.2-1	4d
Table 3.5-2B	5d	TABLE	Not used	
Table 3.5-2B	5e	TABLE	3.3.2-1	4a
Table 3.5-2B	6a	TABLE	3.3.2-1	5b
Table 3.5-2B	6b	TABLE	3.3.2-1	5c
Table 3.5-2B	6c		Relocated - TRM	
Table 3.5-2B	6d	TABLE	3.3.2-1	5a
Table 3.5-2B	7a		Relocated - TRM	
Table 3.5-2B	7b	TABLE	3.3.2-1	6b
Table 3.5-2B	7c	TABLE	3.3.2-1	6d
Table 3.5-2B	7c	TABLE	3.3.2-1	Note f
Table 3.5-2B	7d	TABLE	3.3.2-1	6e
Table 3.5-2B	7d*	TABLE	3.3.2-1	Note g
Table 3.5-2B	7e	TABLE	3.3.2-1	6c
Table 3.5-2B	7f	TABLE	3.3.2-1	6a
Table 3.5-2B	8a	LCO	3.3.4.a	

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Table 3.5-2B	8b	LCO	3.3.4.b	
Table 3.5-2B	9		Deleted - LAR	
Table 3.5-2B	Act 20	LCO	3.3.2 C	
Table 3.5-2B	Act 21	LCO	3.3.2 D	
Table 3.5-2B	Act 21	LCO	3.3.2 E	
Table 3.5-2B	Act 22	LCO	3.3.5 A	
Table 3.5-2B	Act 23	LCO	3.3.2 B	
Table 3.5-2B	Act 24	LCO	3.3.2 D	
Table 3.5-2B	Act 24	LCO	3.3.2 G	
Table 3.5-2B	Act 25	LCO	3.3.2 F	
Table 3.5-2B	Act 26	LCO	3.3.2 I	
Table 3.5-2B	Act 27	LCO	3.7.2	
Table 3.5-2B	Act 28	LCO	3.3.2 F	
Table 3.5-2B	Act 29	LCO	3.3.2 D	
Table 3.5-2B	Act 29	LCO	3.3.2 H	
Table 3.5-2B	Act 30	LCO	3.3.2 I	
Table 3.5-2B	Act 31	LCO	3.3.4 A	
Table 3.5-2B	Act 32		Deleted	
Table 3.5-2B	Act 33	LCO	3.3.4 B	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-2B	Act 34		Deleted - LAR	
Table 3.5-2B	New Action	LCO	3.3.4 C	
Table 3.5-2B	New Action	LCO	3.3.4 D	
Table 3.5-2B	Act 35		Deleted - LAR	
Table 3.5-2B	Act 36		Deleted - LAR	
Table 3.5-2B	Note a	TABLE	3.3.2-1	Note a
Table 3.5-2B	Note b	TABLE	3.3.5-1	Note a, b
Table 3.5-2B	Note c	TABLE	3.3.2-1	Note c
Table 3.5-2B	Note c	LCO	3.7.2	
Table 3.5-2B	Note d	TABLE	3.3.2-1	Note c,d
Table 3.5-2B	New Note	TABLE	3.3.2-1	Note e
Table 3.15-1	1	TABLE	3.3.3-1	1
Table 3.15-1	2	TABLE	3.3.3-1	2
Table 3.15-1	3	TABLE	3.3.3-1	3
Table 3.15-1	4	TABLE	3.3.3-1	4
Table 3.15-1	5	TABLE	3.3.3-1	5
Table 3.15-1	6	TABLE	3.3.3-1	6
Table 3.15-1	7	TABLE	3.3.3-1	7
Table 3.15-1	8	TABLE	3.3.3-1	8

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
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Table 3.15-1	11	TABLE	3.3.3-1	11
Table 3.15-1	12	TABLE	3.3.3-1	12
Table 3.15-1	13	TABLE	3.3.3-1	13
Table 3.15-1	14	TABLE	3.3.3-1	14
Table 3.15-1	15	TABLE	3.3.3-1	15
Table 3.15-1	16	TABLE	3.3.3-1	16
Table 3.15-1	Action a	LCO	3.3.3	
Table 3.15-1	Action a1	LCO	3.3.3 A	
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Current Technical Specification Cross-Reference

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Table 4.1-1B	5a	(partial)	Relocated - IST	

Current Technical Specification Cross-Reference

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Current Technical Specification Cross-Reference

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Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
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Table 4.1-1C	7		Deleted - Boric Acid LAR	
Table 4.1-1C	8	SR	3.3.3.1	
Table 4.1-1C	8	SR	3.3.3.2	
Table 4.1-1C	9		Deleted - Boric Acid LAR	
Table 4.1-1C	10	SR	3.6.8.1	
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Table 4.1-1C	11	SR	3.3.4.1	
Table 4.1-1C	12		Deleted - Boric Acid LAR	
Table 4.1-1C	13		Relocated - TRM	
Table 4.1-1C	14		CTS Deleted	
Table 4.1-1C	15		Relocated - TRM	
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Current Technical Specification Cross-Reference

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Table 4.1-1C	21	SR	3.3.3.1	
Table 4.1-1C	21	SR	3.3.3.2	
Table 4.1-1C	21	SR	3.3.3.3	
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Table 4.1-1C	23		CTS Deleted	
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Table 4.1-1C	24	SR	3.3.6.5	
Table 4.1-1C	24	SR	3.3.6.2	
Table 4.1-1C	25	SR	3.4.12.4	
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Table 4.1-1C	25	SR	3.4.13.5	
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Table 4.1-1C	26		Relocated - TRM	
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Table 4.1-1C	28		Relocated - TRM	
Table 4.1-1C	29	SR	3.3.3.1	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
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Table 4.1-1C	30		Relocated - TRM	
Table 4.1-1C	31		Relocated - TRM	
Table 4.1-1C	Note 30	SR	3.1.7.1	
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Table 4.1-1C	Note 32		Relocated - TRM	
Table 4.1-1C	Note 33		Deleted - Boric Acid LAR	
Table 4.1-1C	Note 34		Deleted	
Table 4.1-1C	Note 35		Deleted	
Table 4.1-1C	Note 36		Deleted	
Table 4.1-1C	Note 37		Deleted	
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Table 4.1-2A	1	(Partial)	Relocated - TRM	
Table 4.1-2A	2	SR	3.1.4.2	
Table 4.1-2A	3	SR	3.4.10.1	
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Table 4.1-2A	7	SR	3.4.11.2	
Table 4.1-2A	8		CTS Deleted	
Table 4.1-2A	9	SR	3.4.14.1	
Table 4.1-2A	10		CTS Deleted	
Table 4.1-2A	11		Relocated - TRM	
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Table 4.1-2B	5		Relocated - TRM	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
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Table 4.1-2B	7		Deleted in CTS	
Table 4.1-2B	8		Relocated - TRM	
Table 4.1-2B	8	SR	3.9.1.1	
Table 4.1-2B	9	SR	3.5.4.2	
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Table 4.1-2B	11	SR	3.6.6.3	
Table 4.1-2B	12	SR	3.5.1.4	
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Table 4.1-2B	Note 5		Deleted	

Current Technical Specification Cross-Reference

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Table 4.12-2		G	5.5.8	
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PACAKGE 3.1

REACTIVITY CONTROL SYSTEMS

CROSS - REFERENCE

IMPROVED TECHNICAL SPECIFICATIONS

TO

CURRENT TECHNICAL SPECIFICATIONS

Section Cross - Reference

Section 3.1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item number	Section Type	CTS Section	CTS Table Item number
ITS Section 3.1				
3.1.1		LCO	3.10.A.1	
3.1.1		LCO	3.10.A.2	
3.1.1		LCO	3.10.A.3	
3.1.1.1		SR	New	
3.1.1.1		SR	New	
3.1.2		LCO	4.9	
3.1.2		LCO	New	
3.1.2.1		SR	New	
3.1.2.2		SR	4.9	
3.1.3		LCO	3.1.F.1	
3.1.3		LCO	3.1.F.2	
3.1.3		LCO	3.1.F.3.a	
3.1.3		LCO	New	
3.1.3.1		SR	New	
3.1.3.2		SR	New	
3.1.3.2		SR	New	
3.1.3.3		SR	New	

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item number	Section Type	CTS Section	CTS Table Item number
3.1.3.3		SR	New	
3.1.4		LCO	3.10.E.1	
3.1.4		LCO	3.10.F.5	
3.1.4		LCO	3.10.G.2	
3.1.4		LCO	3.10.G.3	
3.1.4		LCO	3.10.G.4	
3.1.4		LCO	3.10.G.5	
3.1.4		LCO	3.10.G.6	
3.1.4		LCO	New	
3.1.4.1		SR	Table 4.1-1C	2
3.1.4.2		SR	Table 4.1-2A	2
3.1.4.3		SR	Table 4.1-2A	1
3.1.4.3		SR	3.10.H	
3.1.5		LCO	3.10.D.1	
3.1.5		LCO	3.10.D.3	
3.1.5		LCO	New	
3.1.5.1		SR	New	
3.1.6		LCO	3.10.D.2	
3.1.6		LCO	3.10.D.3	

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item number	Section Type	CTS Section	CTS Table Item number
3.1.6		LCO	New	
3.1.6.1		SR	New	
3.1.6.2		SR	New	
3.1.6.3		SR	New	
3.1.7		LCO	3.10.F.1	
3.1.7		LCO	3.10.F.2	
3.1.7		LCO	3.10.F.3	
3.1.7		LCO	3.10.F.4	
3.1.7		LCO	New	
3.1.7.1		SR	Table 4.1-1C	2
3.1.7.1		SR	Table 4.1-1C	Note 30
3.1.8		LCO	3.10.D.3	
3.1.8		LCO	New	
3.1.8.1		SR	New	
3.1.8.2		SR	New	
3.1.8.3		SR	New	
3.1.8.4		SR	New	

ITS PACKAGE CONTENTS

Package:

3.2

1. Part A Introduction
2. Part B Proposed PI ITS and Bases
3. Part C Markup of PI CTS
4. Part D DOC to PI CTS
5. Part E Markup of ISTS and Bases
6. Part F JD from ISTS
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8. Cross-Reference CTS to ITS
9. Cross-Reference ITS to CTS

PACKAGE 3.2

POWER DISTRIBUTION LIMITS

PART A

INTRODUCTION

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

LICENSE AMENDMENT REQUEST DATED December 11, 2000
Conversion to Improved Standard Technical Specifications

3.2
PART A

Introduction to the Discussion of the proposed Changes to the Current Technical Specifications, Justification of Differences from the Improved Standard Technical Specifications, and the supporting No Significant Hazards Determination

Pursuant to 10 CFR Part 50, Sections 50.59 and 50.90, the holders of Operating Licenses DPR-42 and DPR-60 hereby propose changes to the Facility Operating Licenses and Appendix A, Technical Specifications, as follows and as presented in the accompanying Parts B through G of this Package.

BACKGROUND

Over the past several years the nuclear industry and the Nuclear Regulatory Commission (NRC) have jointly developed Improved Standard Technical Specifications (ISTS). The NRC has encouraged licensees to implement these improved technical specifications as a means for improving plant safety through the more operator-oriented technical specifications, improved and expanded bases, reduced action statement induced plant transients, and more efficient use of NRC and industry resources.

This License Amendment Request (LAR) is submitted to conform the Prairie Island Nuclear Generating Plant (PINGP) Current Technical Specifications (CTS) to NUREG-1431, Improved Standard Technical Specifications, Westinghouse plants, Revision 1 issued April 1995 (ISTS). The resulting new Technical Specifications (TS) for Prairie Island (PI) are the PI Improved Technical Specifications (ITS) which incorporates the PI plant specific information.

NUREG-1431 is based on a hypothetical four loop Westinghouse plant. Since PI is similar in design and vintage to the R.E. Ginna Nuclear Power Plant which has already completed conversion to improved technical specifications, this amendment request relies on the Ginna ITS.

This LAR is also supported by Parts B through G. Part B contains a "clean" copy of the proposed PI ITS and Bases. Part C contains a mark-up of the PI CTS. Part D is the Description of Changes (DOC) to the PI CTS. Part E is a mark-up of the ISTS and Bases which shows the deviations from the standard incorporated to meet PI plant specific requirements. Part F gives the Justification for Deviations (JFD) from the ISTS and Part G provides the No Significant Hazards Determinations (NSHD) for changes to the PI CTS. To facilitate review of this LAR, cross-reference numbers from changes and deviations to the corresponding DOC, JFD and NSHD are provided. The methodology for mark-up and cross-references are described in the next section.

MARK-UP METHODOLOGY

The TS conversion package includes mark-ups of the CTS, the ISTS and the ISTS Bases in accordance with this guidance. Mark-up may be electronic or by hand as indicated.

Current Technical Specifications

The mark-up of the CTS is provided to show where current requirements are placed in the ITS, to show the major changes resulting from the conversion process, and to allow reviewers to evaluate significant differences between the CTS and ITS.

This ITS conversion LAR has been prepared in 14 packages following the Chapter/Section outline of the ITS as follows: 1.0, 2.0, 3.0, 3.1 . . . 3.9, 4.0 and 5.0. Accordingly, each package contains all the elements of Parts A through G as described above. The CTS Bases are not included in the CTS mark-up packages since the Bases have been rewritten in their entirety.

The current Specifications addressed by the associated ITS Chapter/Section are cross-referenced in the left margin to the new ITS location by Specification number and type (G-General, SL-Safety Limit, LCO-Limiting Condition for Operation or SR-Surveillance Requirements). Those portions of each CTS page which are not addressed in the associated ITS Chapter/Section are shadowed (electronic) or clouded and crossed out (by hand) and in the right margin is the comment, “Addressed Elsewhere”.

The CTS are marked-up to incorporate the substance of NUREG-1431 Revision 1. It is not the intent to mark every nuance required to make the format change from CTS to ITS.

In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.

Some apparent changes result from the different conventions and philosophies used in the ITS. Generally these apparent changes will not be marked-up in the CTS if there is no resulting change in plant operating requirements.

Changes are identified by a change number in the right margin which map the changed specification requirement to Part D, Discussion of Changes, and Part G, No Significant Hazards Determination (NSHD) and indicate the NSHD category. The change number form is R3.4-02 where the first two numbers, 3.4 in this example, refer to ITS Chapter/Section number 3.4, and the second number, 02 in this example, is a sequentially assigned number for changes within that Chapter/Section, starting with 01. The prefix letter(s) indicates the classification of the change impact. For CTS changes this is also the NSHD category.

The change impact categories defined below conveniently group the type of changes for consideration of the effect of the change on the current plant license in Part D and are also useful for efficient discussion in Part G the "No Significant Hazards Determination" (NSHD) section. If the same change is made in Part E, then the change impact category will also show up in the change number in Part F. These categories are:

A - Administrative changes, editorial in nature that do not involve technical issues. These include reformatting, renaming (terminology changes), renumbering, and rewording of requirements.

L - Less restrictive requirements included in the PI ITS in order to conform to the guidance of NUREG-1431. Generally these are technical changes to existing TS which may include items such as extending Completion Times or reducing Surveillance Frequencies (extended time interval between surveillances). The less restrictive requirements necessitate individual justification. Each is provided with its specific NSHD.

LR - Less restrictive Removal of details and information from otherwise retained specifications which are removed from the CTS and placed in the Bases, Technical Requirements Manual (TRM), Updated Safety Analysis Report (USAR) or other licensee controlled documents. These changes include details of system design and function, procedural details or methods of conducting surveillances, or alarm or indication-only instrumentation.

M - More restrictive requirements included in the PI ITS in order to provide a complete set of Specifications conforming to the guidance of NUREG-1431. Changes in this category may be completely new requirements or they may be technical changes made to current requirements in the CTS.

R - Relocation of Current Specifications to other controlled documents or deletion of current Specifications which duplicate existing regulatory requirements.

Current requirements in the LCOs or SRs that do not meet the 10 CFR 50.36 selection criteria and may be relocated to the Bases, USAR, Core Operating Limits Report (COLR), Operational Quality Assurance Plan (OQAP), plant procedures or other licensee controlled documents. Relocating requirements to these licensee controlled documents does not eliminate the requirement, but rather, places them under more appropriate regulatory controls, such as 10CFR 50.54 (a)(3) and 10 CFR 50.59, to manage their implementation and future changes. Maintenance of these requirements in the TS commands resources which are not commensurate with their importance to safety and distract resources from more important requirements. Relocation of these items will enable more efficient maintenance of requirements under existing regulations and reduce the need to request TS changes for issues which do not affect public safety.

Deletion of Specifications which duplicate regulations eliminates the need to change Technical Specifications when changes in regulations occur. By law, licensees shall meet applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions; therefore, restatement in the Technical Specifications is unnecessary.

The methodology for marking-up these changes is as follows:

As discussed above, administrative changes may not be marked-up in detail. Portions of the specifications which are no longer included are identified by use of the electronic strike-out feature (or crossed out by hand). Information being added is inserted into the specification in the appropriate location and is identified by use of shading features (or handwritten/insert pages).

Improved Standard Technical Specifications (NUREG-1431, Rev. 1)

The ISTS mark-up is to identify changes from the ISTS required to create a plant specific ITS by incorporating plant specific values in bracketed fields and identifying other changes with cross-reference to the Part F Justification For Differences.

All deviations from the ISTS are cross-referenced to the Part F justification for differences by a change number in the right margin. The change number form is CL3.4-05 where the prefix letter(s), CL in this example, indicate the classification of the reason for the difference, the first two numbers, 3.4 in this example, refer to the ITS Chapter/Section number 3.4, and the second number, 05 in this example, is a sequentially assigned number for deviations within that Chapter/Section, starting with a number which is larger than the last number from the Part C CTS mark-up. In some instances where a change has been made to the CTS and ISTS, the Part D change number is given since the justification for difference is the same as the discussion of change. The following categories are used as prefixes to indicate the general reason for each difference:

CL - Current Licensing basis. Issues that have been previously licensed for PI and have been retained in the ITS. This includes Specifications dictated by plant design features or the design basis. Since no plant modifications have been or will be made to accommodate conversion to ITS, the plant design basis features shall be incorporated into the PI ITS.

PA - Plant, Administrative. Plant specific wording preference or minor editorial improvements made to facilitate operator understanding.

TA - Traveler, Approved. Deviations made to incorporate an industry traveler which has been approved by the NRC.

TP - Traveler, Proposed. Deviation made to incorporate a proposed industry traveler which as of the time of submittal has not been approved by the NRC.

X - Other, Deviation from the ISTS for any other reason than those given above.

Material which is deleted from the ISTS is identified by use of the WordPerfect strike-out feature (or crossed out by hand). Information being added to the ISTS to generate the PI ITS due to any of the deviations discussed above is identified by use of WordPerfect red-line features (or handwritten/insert pages).

Bracketed Information

Many parameters, conditions, notes, surveillances, and portions of sections are bracketed in the ISTS recognizing that plant specific values are likely to vary from the "generic" values provided in the standard.

If the bracketed value applies to PI, then the "generic" information is retained without any special indication and the brackets are marked using the WordPerfect strike-out feature. In some instances, bracketed material is not discussed. If bracketed material is discussed, a change number is provided which includes the appropriate prefix as described above. When bracketed "generic" material is not incorporated, the bracketed material and brackets are marked with the WordPerfect strike-out feature (or crossed out by hand), the plant specific information is substituted for the bracketed information and a change number is provided which includes the appropriate prefix. Information added is indicated by the WordPerfect red-line (shading) feature (or handwritten/insert pages).

Optional Sections

Due to differing Westinghouse plant designs and methodologies, some ISTS section numbers include a letter suffix indicating that only one of these sections is applicable to any specific plant. The appropriate section is indicated in the Table of Contents, the suffix letter is deleted, and justification, if required, is included in the appropriate Chapter/Section package.

Bases, Improved Standard Technical Specifications (NUREG-1431, Rev. 1)

The ISTS Bases have been marked-up to support the plant specific PI ITS and allow reviewers to identify changes from NUREG-1431. To the extent possible, the words of NUREG-1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading with respect to Prairie Island, they have been revised. In addition, descriptions have been added to cover plant specific portions of the specifications. Change numbers have been provided for the ISTS Bases with the same format as the ISTS Specification mark-up. In some instances, the same change number is used to describe the change.

Material which is deleted from the ISTS Bases is identified by use of the strike-out feature of WordPerfect (or crossed out by hand). Information being added to the ISTS Bases to generate the PI ITS is identified by use of the red-line (shading) feature of WordPerfect (or handwritten/insert pages).

Bracketed Material

Many parameters and portions of Bases are bracketed in the ISTS recognizing that plant specific values and discussions are likely to vary from the "generic" information provided in the standard.

If the bracketed information applies to PI, then the "generic" information is retained without any special indication and the brackets are marked using the WordPerfect strike-out feature. No change number or justification is provided for use of bracketed material, unless special circumstances warrant discussion.

When bracketed "generic" Bases material is not incorporated, the bracketed material and brackets are marked with the WordPerfect strike-out feature (or crossed out by hand) and the plant specific information substituted for the bracketed information is indicated by the WordPerfect red-line (shading) feature (or handwritten/insert pages). A change number with the same format as those used for the ISTS Specification mark-up is provided.

ACRONYMS

Many acronyms are used throughout this submittal. The intent of the final ITS (Part B) is that in general acronyms be written in full prior to the first use. Commonly used acronyms may not be written in full. Other parts of this package may not always write in full each acronym prior to first use; therefore, a list of acronyms is attached to assist in the review of this package.

Attachment to Part A

LIST OF ACRONYMS

AB	Auxiliary Building
ABSVS	Auxiliary Building Special Ventilation System
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ALARA	As Low As Reasonably Achievable
ALT	Actuation Logic Test
ASA	Applicable Safety Analyses
ASME	American Society of Mechanical Engineers
AOO	Anticipated Operational Occurrences
AOT	Allowed Outage Time
BAST	Boric Acid Storage Tank
BIT	Boron Injection Tank
BOC	Beginning of Cycle
CC	Component Cooling
COT	CHANNEL OPERATIONAL TEST
CAOC	Constant Axial Offset Control
CET	Core Exit Thermocouple
CL	Cooling Water
CLB	Current Licensing Basis
COLR	Core Operating Limits Reports
CRDM	Control Rod Drive Mechanism
CRSVS	Control Room Special Ventilation System
CS	Containment Spray
CST	Condensate Storage Tanks
CTS	Current Technical Specification(s)
DBA	Design Basis Accident
DDCL	Diesel Driven Cooling Water
DG	Diesel Generator
DNB	Departure from Nucleate Boiling
DNBR	Departure from nucleate boiling ratio
ECCS	Emergency Core Cooling System

EDG	Emergency Diesel Generators
EFPD	Effective Full Power Days
EOC	End of Cycle
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
FWLB	Feedwater Line Break
GDC	General Design Criteria
GITS	Ginna Improved Technical Specifications
HELB	High Energy Line Break
HZP	Hot Zero Power
IPE	Individual Plant Evaluation
ISTS	Improved Standard Technical Specifications
ITC	Isothermal Temperature Coefficient
ITS	Improved Technical Specifications
LA	License Amendment
LAR	License Amendment Request
LBLOCA	Large Break LOCA
LCO	Limiting Conditions for Operation
LHR	Linear Heat Rate
LOCA	Loss of Coolant Accident
LTOP	Low Temperature Overpressure Protection
MFIV	Main Feedwater Isolation Valve
MFRV	Main Feedwater Regulation Valve
MFW	Main Feedwater
MOSCA	MODE or Other Specified Condition of Applicability
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valves
MSLB	Main Steam Line Break
MSLI	Main Steam Line Isolation
MSSV	Main Steam Safety Valves
MTC	Moderator Temperature Coefficient
NIS	Nuclear Instrumentation System
NMC	Nuclear Management Company
NPSH	Net Positive Suction Head

NRCV	Non-Return Check Valve
NUREG-1431	The ISTS for Westinghouse plants
OPPS	OverPressure Protection System
PCT	Peak Cladding Temperature
PI	Prairie Island
PITS	Prairie Island Technical Specifications
PIV	Pressure Isolation Valve
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSV	Pressurizer Safety Valve
PTLR	Pressure and Temperature Limits Report
QTPR	Quadrant Power Tilt Ratio
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RPI	Rod Position Indication
RPS	Reactor Protection System
RTB	Reactor Trip Breaker
RTBB	Reactor Trip Bypass Breaker
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SBVS	Shield Building Ventilation System
SCWS	Safeguards Chilled Water System
SDM	Shut Down Margin
SFDP	Safety Function Determination Program
SFP	Spent Fuel Pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SL	Safety Limit

SLB	Steam Line Break
SR	Surveillance Requirements
SSC	Structures, Systems and Components
TADOT	Trip Actuating Device Operational Test
TDAFW	Turbine Driven Auxiliary Feedwater
TRM	Technical Requirements Manual
TS	Technical Specifications
TSSC	Technical Specification Selection Criteria
TSTF	Term used for a NUREG change (traveler)
VCT	Volume Control Tank
VFTP	Ventilation Filter Test Program
UHS	Ultimate Heat Sink
USAR	Updated Safety Analysis Report
WCAP	Westinghouse technical report

PACKAGE 3.2

POWER DISTRIBUTION LIMITS

PART B

PROPOSED PRAIRIE ISLAND IMPROVED TECHNICAL SPECIFICATIONS AND BASES

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PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----------NOTE----------- Required Action A.3 shall be completed whenever this Condition is entered. ------------------------------ $F_Q^C(Z)$ not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes after each $F_Q^C(Z)$ determination
	<u>AND</u>	
	A.2 Reduce Power Range Neutron Flux -High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	<u>AND</u>	
	A.3 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. -----------NOTE----------- Required Action B.3 shall be completed whenever this Condition is entered. ------------------------------ $F_Q^W(Z)$ not within limits.	B.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours after each $F_Q^W(Z)$ determination
	<u>AND</u>	
	B.2 Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours after each $F_Q^W(Z)$ determination
	<u>AND</u>	
	B.3 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action B.1
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify $F^C_Q(Z)$ is within limit.	Prior to exceeding 75% RTP after each refueling <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F^C_Q(Z)$ was last verified <u>AND</u> 31 effective full power days (EFPD) thereafter

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	---------------------------NOTE--------------------------- If measurements indicate that the maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits; or b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate that the maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$ has not increased. --- Verify $F_Q^W(Z)$ is within limit.	Once within 12 hours after achieving equilibrium conditions after each refueling after THERMAL POWER exceeds 75% RTP <u>AND</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE----------- Required Actions A.2 and A.4 must be completed whenever Condition A is entered. ------------------------------	A.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
$F^N_{\Delta H}$ not within limit.	AND	
	A.2 Perform SR 3.2.2.1.	24 hours
	AND	
	A.3 Reduce Power Range Neutron Flux-High trip setpoints to ≤ 55% RTP.	72 hours
	AND	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 -------------NOTE------------- THERMAL POWER does not have to be reduced to comply with this Required Action. --------------------------------- Perform SR 3.2.2.1.	Prior to exceeding 50% RTP <u>AND</u> Prior to exceeding 75% RTP <u>AND</u> 24 hours after reaching ≥95% RTP
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Prior to exceeding 75% RTP after each refueling AND 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD:

a. Shall be maintained within the target band about the target flux difference. The target band is specified in the COLR.

b. May deviate outside the target band with THERMAL POWER < 90% RTP but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.

c. May deviate outside the target band with THERMAL POWER < 50% RTP.

---------------------------------------NOTES----------------------------------

1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.

2. With THERMAL POWER ≥ 50% RTP, penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

3. With THERMAL POWER < 50% RTP and >15% RTP, penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within acceptable operation limits.

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ 90% RTP. AND AFD not within the target band.	A.1 Restore AFD to within target band.	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 90% RTP.	15 minutes

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -----------NOTE----------- Required Action C.1 must be completed whenever Condition C is entered. ------------------------------ THERMAL POWER < 90% and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. <u>OR</u> THERMAL POWER < 90% and ≥ 50% RTP with AFD not within the acceptable operation limits.	C.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel.	7 days
SR 3.2.3.2	Determine and update target flux difference.	Once within 31 EFPD after each refueling <u>AND</u> 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	A.1 Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination
	<u>AND</u>	
	A.2 Determine QPTR.	Once per 12 hours
	<u>AND</u>	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Actions A.1 <u>AND</u> Once per 7 days thereafter.
	<u>AND</u>	
	A.4 Re-evaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.5 -------------NOTES------------ 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed when Required Action A.5 is performed. --------------------------------- Normalize excore detectors to restore QPTR to within limits. <u>AND</u>	 Prior to increasing THERMAL POWER above the limit of Required Action A.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.6 -------------NOTE------------- Perform Required Action A.6 only after Required Action A.5 is completed. ---------------------------------	
	Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	-------------------------NOTES----------------------------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 85% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. --- Verify QPTR is within limit by calculation.	7 days
SR 3.2.4.2	-------------------------NOTE---------------------------- Not required to be performed until 12 hours after input from one Power Range Neutron Flux channel is inoperable with THERMAL POWER > 85% RTP. --- Verify QPTR is within limit using the movable incore detectors or thermocouples.	12 hours

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents an equilibrium condition, it does not include the variations in the values of $F_Q(Z)$ which are present during non-equilibrium situations such as load following or power ascension.

BASES

BACKGROUND (continued)	To account for these possible variations, the equilibrium value of $F_Q(Z)$ is adjusted as $F_Q^W(Z)$ by an elevation dependent factor that accounts for the calculated worst case transient conditions. Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that could violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During transient conditions arising from events of moderate frequency (Condition II events), there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition (Ref. 1);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Large Break LOCA (LBLOCA) analysis is the analysis that determines the LCO limit for $F_Q(Z)$. The $F_Q(Z)$ assumed in the Safety Analysis for other postulated accidents is either equal to or greater than that assumed in the LBLOCA analysis. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \le \frac{CFQ}{P}\ K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \le \frac{CFQ}{0.5}\ K(Z) \qquad \text{for } P \le 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and is based on the Small Break LOCA analysis, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

For Constant Axial Offset Control operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

BASES

LCO (continued)

An $F^C_Q(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results a measured value $(F^M_Q(Z))$ of $F_Q(Z)$ is obtained. Then,

$$F^C_Q(Z) = F^M_Q(Z)*(1.0815)$$

where 1.0815 is a factor that accounts for fuel manufacturing tolerances (1.03) multiplied by a factor associated with the flux map measurement uncertainty (1.05) (Ref. 3).

$F^C_Q(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

The expression for $F^W_Q(Z)$ is:

$$F^W_Q(Z) = F^C_Q(Z)\ V(Z)$$

where V(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. V(Z) is included in the COLR. The $F^W_Q(Z)$ is calculated at equilibrium conditions.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO precludes core power distributions that could violate the assumptions in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F^C_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

BASES

LCO (continued)

Violating the LCO limits for $F_Q(Z)$ could result in unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

<u>A.1</u>

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by factors accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

BASES

ACTIONS (continued)

A.2

A reduction of the Power Range Neutron Flux-High trip setpoints by $\geq$ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux-High trip setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Power Range Neutron Flux-High trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux-High trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux-High trip setpoints.

A.3

Verification that $F_Q^C(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels, and future operations, are consistent with safety analyses assumptions.

Condition A is modified by a Note that requires Required Action A.3 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action

BASES

ACTIONS

A.3 (continued)

A.3. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, maintains an acceptable absolute power density such that even if a transient occurred, core peaking factors are not exceeded.

B.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action B.1.

BASES

ACTIONS (continued)

B.3

Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action B.1, ensures that core conditions during operation at higher power levels, and future operation, are consistent with safety analyses assumptions.

Condition B is modified by a Note that requires Required Action B.3 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when Condition B is exited prior to performing Required Action B.3. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

C.1

If Required Actions A.1 through A.3 or B.1 through B.3 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during the first power ascension after a refueling. It states

BASES

SURVEILLANCE REQUIREMENTS (continued)

that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F^C_Q(Z)$ and $F^W_Q(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F^C_Q(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of $F^C_Q(Z)$ before exceeding 75% RTP. This ensures that some determination of $F_Q(Z)$ is made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F^C_Q(Z)$ and $F^W_Q(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F^C_Q(Z)$ and $F^W_Q(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured.

SR 3.2.1.1

Verification that $F^C_Q(Z)$ is within its specified limits involves increasing $F^M_Q(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F^C_Q(Z)$. Specifically, $F^M_Q(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F^C_Q(Z) = F^M_Q(Z)*(1.0815)$ (Ref. 3). $F^C_Q(Z)$ is then compared to its specified limits.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

The limit with which $F^C_Q(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F^C_Q(Z)$ limit is met during the power ascension following a refueling, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F^C_Q(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F^C_Q(Z)$ values are being reduced sufficiently with the power increase to stay within the LCO limits).

The Frequency of 31 effective full power days (EFPD) is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated during the nuclear design process by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called V(Z).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

Multiplying the measured total peaking factor, $F_Q^C(Z)$, by V(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power above 50% RTP and directly with the function K(Z) provided in the COLR.

The V(Z) curve is provided in the COLR for discrete core elevations. Flux map data are taken for 61 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 10% inclusive; and

b. Upper core region, from 90 to 100% inclusive.

The top and bottom 10% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated, an evaluation of the expression below is required to account for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

$$\text{maximum over z} \left[\frac{F_Q^C(Z)}{K(Z)}\right]$$

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by an appropriate factor specified in the COLR, or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

Performing the Surveillance once within 12 hours after achieving equilibrium conditions in MODE 1 during the power ascension following a refueling ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^W(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^W(Z)$ values are being reduced sufficiently with the power increase to stay within the LCO limits).

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

BASES (continued)

REFERENCES

1. USAR Section 14.
2. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2.
3. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 effective full power days (EFPD). However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

BASES

BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling ratio (DNBR) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency (referred to as Condition II events). The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to a value greater than the criterion listed in Reference 1. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Controlling $F^N_{\Delta H}$ precludes core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition during Condition II transients (Ref. 1);

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES (continued)

For transients that may be DNB limited, the THERMAL POWER, Reactor Coolant System flow, temperature, pressure and $F^N_{\Delta H}$ are the core parameters of most importance. Except for Static Rod Cluster Control Assembly (RCCA) Misalignment and Dropped Rod events, the limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any Condition II transients. The analyses for Static RCCA Misalignment and Dropped Rod events ensure the DNB design basis is met by assuming a calculated $F^N_{\Delta H}$ plus uncertainties (Ref. 1). The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion listed in Reference 1. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all Condition II transients, except Static RCCA Misalignment and Dropped Rod events, that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 1).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank

BASES

APPLICABLE SAFETY ANALYSES (continued)

Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition I events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses as described in the Applicable Safety Analyses section above.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase by a factor specified in the COLR for every 1% RTP reduction in THERMAL POWER.

BASES (continued)

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1 and A.3

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1 and reduce the Power Range Neutron Flux -High trip setpoint to ≤ 55% RTP in accordance with Required Action A.3. Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1 provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.3 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

BASES

ACTIONS (continued)

A.2

Once the power level has been reduced to < 50% RTP per Required Action A.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by Action A.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.4

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

Condition A is modified by a Note that requires that Required Actions A.2 and A.4 must be completed whenever Condition A is entered.

BASES

ACTIONS (continued)

B.1

When Required Actions A.1 through A.4 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

BASES (continued)

REFERENCES

1. USAR Section 14.
2. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating scheme used to control the axial power distribution, Constant Axial Offset Control (CAOC), involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

The target flux difference is determined at equilibrium xenon conditions in conjunction with verifying $F_Q^W(Z)$ in accordance with SR 3.2.1.2. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 190 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the Fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

The AFD is logged manually or monitored on an automatic basis using the unit process computer that has an AFD monitor alarm. The frequency of monitoring the AFD by the unit computer is once per

BASES

BACKGROUND (continued)

minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFDs for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is ≥ 90% RTP. During operation at THERMAL POWER levels < 90% RTP but > 15% RTP, the computer sends an alarm message when the cumulative penalty deviation time is > 1 hour in the previous 24 hours.

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC and Transient Power Distribution methodologies (Refs. 1 and 2) entail:

a. Establishing an envelope of allowed power shapes and power densities;

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes;

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

The Transient Power Distribution methodology (Ref. 2) determines a function, (V(Z)), that when applied to equilibrium $F_Q^C(Z)$ values will bound $F_Q^C(Z)$ values that could be measured at non-equilibrium conditions. This remains valid provided that the AFD is maintained within the target flux band around a target flux difference that was determined in conjunction with determining the equilibrium $F_Q^W(Z)$.

The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition II, III, and IV events. This ensures that fuel cladding integrity is maintained for these postulated accidents. The most important Condition IV event is the loss of coolant accident. The most significant Condition III event is the loss of RCS flow accident. The most significant Condition II events are uncontrolled bank withdrawal at power and Rod Cluster Control Assembly (RCCA) misalignment.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator, through either the manual operation of the control banks, or automatic motion of control banks responding to temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration, or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors. Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as % ? flux or % ?I.

The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup and target flux difference.

With THERMAL POWER ≥ 90% RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition II, III, or IV event occurs while the AFD is outside its limits.

The LCO is modified by four Notes. Note 1 states the conditions necessary for declaring the AFD outside of the target band. With one channel removed from service (e.g., for calibration, testing or repairs), if two of the remaining channels indicate outside the target band, then the AFD shall be considered outside the target band.

BASES

LCO (continued)

Notes 2 and 3 describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP (i.e., Part b of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside of the target band but within the acceptable operation limits provided in the COLR (Note 2). This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time when THERMAL POWER ≥ 50% RTP. The cumulative penalty time is the sum of penalty times from LCO Notes 2 and 3.

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part c of this LCO), deviations of the AFD outside of the target band are less significant. Note 3 allows the accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band and reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

For surveillance of the power range channels performed according to SR 3.3.1.6, Note 4 allows deviation outside the target band for 16 hours and no penalty deviation time accumulated. Some

BASES

LCO (continued)

deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days.

APPLICABILITY

AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 3).

Between 15% RTP and 90% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events.

Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER ≥ 90% RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

BASES

ACTIONS (continued)

B.1

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP without allowing the plant to remain in an unanalyzed condition for an extended period of time.

C.1

With THERMAL POWER < 90% RTP but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

BASES

ACTIONS

C.1 (continued)

Condition C is modified by a Note that requires that Required Action C.1 must be completed whenever this Condition is entered.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD as indicated by the NIS excore channels is within the target band. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Furthermore, any deviations of the AFD from the target band that is not alarmed should be readily noticed.

The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.2

This Surveillance requires that the target flux difference be determined and updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup.

The target flux difference is determined by averaging the indicated AFD from all OPERABLE excore channels.

To ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during non-equilibrium state conditions, the Transient Power Distribution methodology, i.e. V(Z), (Ref. 2) requires SR 3.2.1.2 to be performed in conjunction with this SR.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.2 (continued)

Following a refueling outage, SR 3.2.1.2, and thus SR 3.2.3.2, are not required to be performed until equilibrium conditions are achieved. Since it may be desirable to provide the operators with some guidance for AFD control during the power ascension, a target flux difference may be posted based on engineering judgement or analytical prediction.

REFERENCES

1. XN-NF-77-57, supplement 1(A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981.

2. Transient Power Distribution, NSPNAD-93003-A.

3. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During transient conditions arising from events of moderate frequency (Condition II events), there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure the assumptions used in the safety analysis remain valid by preventing an undetected change in the gross radial power distribution.

In MODE 1, the QPTR must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the assumptions in the safety analysis are possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to

BASES

APPLICABILITY (continued)

require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in the QPTR would require power reductions within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

BASES

ACTIONS (continued)

A.3

The peaking factors $F_{\Delta H}^{N}$ and $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F_{\Delta H}^{N}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. A Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F_{\Delta H}^{N}$ and $F_Q(Z)$ for changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F_{\Delta H}^{N}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern

BASES

ACTIONS

A.4 (continued)

exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the QPTR is not restored to within limits until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

BASES

ACTIONS (continued)

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limits and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable,

BASES

ACTIONS

B.1 (continued)

based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 85% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.

For those causes of a core power tilt that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one Power Range Neutron Flux channel is inoperable and the THERMAL POWER is > 85% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that the QPTR remains within its limits.

For purposes of monitoring changes in radial core power distribution when one power range channel is inoperable, at least 2 moveable incore detectors or 4 thermocouples per quadrant may be used to calculate an incore core power tilt. This incore core power tilt may be used, instead of the excore detectors, to confirm that the QPTR is within the limits by comparing it to previous flux maps.

REFERENCES

1. USAR Section 14.

2. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

PACKAGE 3.2

POWER DISTRIBUTION LIMITS

PART C

MARKUP OF PRAIRIE ISLAND CURRENT TECHNICAL SPECIFICATIONS

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PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

Improved Technical Specifications Conversion Submittal

~~3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS~~

~~Applicability~~ A3.2-01

~~Applies to the limits on core fission power distribution and to the limits on control rod operations.~~

~~Objective~~

~~To assure 1) core subcriticality after reactor trip, 2) acceptable core power distributions during POWER OPERATION, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.~~

Specification

Addressed Elsewhere

A. Shutdown Margin

1. The SHUTDOWN MARGIN shall be maintained within the limits specified in the Core Operating Limits Report when in HOT SHUTDOWN, INTERMEDIATE SHUTDOWN and COLD SHUTDOWN.

2. With the SHUTDOWN MARGIN less than the applicable limit specified in 3.10.A.1 above, within 15 minutes initiate boration to restore SHUTDOWN MARGIN to within the applicable limit.

3.2~~B.~~ Power Distribution Limits

1. In MODE 1 ~~At all times, except during low power PHYSICS TESTING, measured~~ A3.2-02

LCO3.2.1
LCO3.2.2

hot channel factors, A3.2-03
$F_Q(Z)$ as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$ F^N_Q

and $F^N_{\Delta H}$, ~~as defined below and in the bases,~~ shall meet the LR3.2-04

~~following limits:~~ specified in the COLR.

~~$F^N_Q \times 1.03 \times 1.05^{*} \leq (F_Q^{RTP} / P) \times K(Z)$~~ LR3.2-04

~~$F^N_{\Delta H} \times 1.04^{**} \leq F_{\Delta H}^{RTP} \times [1 + PFDH(1-P)]$~~

~~where the following definitions apply:~~

~~F_Q^{RTP} is the F_Q limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.~~

~~- $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.~~ LR3.2-04

~~- PFDH is the Power Factor Multiplier for $F^{N}_{\Delta H}$ specified in the CORE OPERATING LIMITS REPORT.~~

~~- K(Z) is a normalized function that limits $F_Q(z)$ axially as specified in the CORE OPERATING LIMITS REPORT.~~

A3.2-05

~~* For Unit 1, Cycle 19, when the number of available moveable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 5% measurement uncertainty shall be increased to [5% + (3-T/9)(3%)] where T is the number of available thimbles.~~

~~** For Unit 1, Cycle 19, when the number of available moveable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 4% measurement uncertainty shall be increased to [4% + (3-T/9)(2%)] where T is the number of available thimbles.~~

~~3.10.B.1. – Z is the core height location.~~ LR3.2-04

~~– P is the fraction of RATED THERMAL POWER at which the core is operating. In the F^N_Q limit determination when P $\leq$0.50, set P = 0.50.~~

~~– F^N_Q or $F^N_{\Delta H}$ is defined as the measured F_Q or $F_{\Delta H}$ respectively, with the smallest margin or greatest excess of limit.~~

~~– 1.03 is the engineering hot channel factor, F^E_Q, applied to the measured F^N_Q to account for manufacturing tolerance.~~

~~– 1.05* is applied to the measured F^N_Q to account for measurement uncertainty.~~

~~– 1.04** is applied to the measured $F^N_{\Delta H}$ to account for measurement uncertainty.~~

~~2.~~ Hot channel factors, $F_Q(Z)$ F^N_Q (as approximated by $F_Q(Z)$ and F_Q (Z)) and $F^N_{\Delta H}$, shall be A3.2-03

SR3.2.1.1
SR3.2.1.2
SR3.2.2.1
SR3.2.3.2

measured and the target flux difference determined, at equilibrium conditions according to the following conditions,

~~whichever occurs first:~~ A3.2-06

(a) At least once per 31 effective full-power days in conjunction with the target flux difference determination, ~~or~~ and

(b) Once within 12 hours Upon reaching equilibrium conditions after exceeding the reactor power at which target flux difference was last determined, by 10% or more of RATED THERMAL POWER~~.~~ (Does not apply to SR 3.2.2.1 and SR 3.2.3.2) and M3.2-08 L3.2-09

(c) Once after each refueling prior to THERMAL POWER exceeding 75% RTP (Does not apply to SR 3.2.1.2 and SR 3.2.3.2) M3.2-11

~~F^N_Q (equil) shall meet the following limit for the middle axial 80% of the core:~~ LR3.2-04

~~F^N_Q (equil) x V(Z) x 1.03 x 1.05* $\leq$ (F_Q^{RTP} / P) x K(Z)~~

~~where V(Z) is specified in the CORE OPERATING LIMITS REPORT and other terms are defined in 3.10.B.1 above.~~

A3.2-03

~~3.~~ (a) If either FQ(Z) or $F^N_{\Delta H}$ ~~measured hot channel factor~~ exceeds its

LCO3.2.1 Cond A LCO 3.2.2 Cond A

limit specified in the COLR ~~3.10.B.1~~, for FQ(Z) reduce reactor LR3.2-04

power within 15 minutes after each FQ(Z) determination, and the high neutron flux trip set-point within 72 hours after each FQ(Z) determination M3.2-12

by 1% for each percent that FQ(Z) exceeds the limits specified in the COLR ~~the measured F^N_Q~~ LR3.2-04

and then perform SR 3.2.1.1 and SR 3.2.1.2 (FQ(Z) out of limits) prior to increasing THERMAL POWER above the limit determined above in this paragraph (TS 3.10.B.3(a)). Note, surveillances shall be completed whenever this condition is entered. A3.2-16

or if ~~by the factor specified in the CORE OPERATING LIMITS REPORT for each percent that the measured~~ $F^N_{\Delta H}$ exceeds the ~~3.10.B.1~~ limit specified in the COLR LR3.2-04

reduce THERMAL POWER to < 50% RTP within 4 hours and reduce power range neutron flux-high trip setpoint to ≤ 55% RTP within 72 hours~~.~~ Then follow 3.10.B.3(c). M3.2-13

LCO3.2.1 Cond C LCO3.2.2 Cond B

New Condition: If these conditions are not met, be in MODE 2 within 6 hours. L3.2-17

(b) If FQ(Z) ~~the measured F^N_Q (equil~~) exceeds the A3.2-03

LCO3.2.1 Cond B

~~3.10.B.2~~ limits in the COLR ~~but not the 3.10.B.1 limit,~~ take one of the following actions: LR3.2-04

~~1. Within 48 hours place the reactor in an equilibrium configuration for which Specification 3.10.B.2 is satisfied, or~~ M3.2-14

2. Reduce reactor power within 4 hours after each FQ (Z) determination, and the high neutron flux trip setpoint within 72 hours after each FQ(Z) determination by 1% for each percent M3.2-12

that FQ(Z) ~~the measured F^N_Q (equil) x 1.03 x 1.05* x V(Z)~~ exceeds the limit. LR3.2-04

Perform SR 3.2.1.1 and SR 3.2.1.2 prior to increasing power above the limit determined above in this paragraph (TS 3.10.B.3(b)2). Note, surveillances shall be completed whenever this condition is entered. A3.2-16

~~* For Unit 1, Cycle 19, when the number of available moveable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 5% measurement uncertainty shall be increased to [5% + (3-T/9)(3%)] where T is the number of available thimbles.~~ A3.2-05

~~** For Unit 1, Cycle 19, when the number of available moveable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 4% measurement uncertainty shall be increased to [4% + (3-T/9)(2%)] where T is the number of available thimbles.~~

L3.2-18

~~3.10.B.3.~~(c) ~~If subsequent in-core mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a HOT SHUTDOWN condition with return to power authorized up to 50% of RATED THERMAL POWER for the purpose of PHYSICS TESTING.~~ Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above 50% of RATED THERMAL POWER. THERMAL POWER may then be increased provided ~~F^N_Q or~~ $F^N_{\Delta H}$ is demonstrated through in-core mapping (SR 3.2.2.1) to be within its limits.

LCO3.2.2 Action A

(d) If two successive measurements indicate an increase in FQ(Z) ~~the peak rod power $F^N_{\Delta H}$~~ with exposure, either of the following actions shall be taken:

SR3.2.1.2 Note

M3.2-19

A3.2-03

1. FQ(Z)~~F^N_Q (equil)~~ shall be multiplied by

~~1.02~~ an appropriate factor specified in the COLR ~~x V(Z) x 1.03 x 1.05** for comparison to the limit specified in 3.10.B.2~~

LR3.2-04

and reverify FQ(Z) is within limits, or

A3.2-22

2. FQ(Z)~~F^N_Q (equil)~~ shall be measured at least once per seven effective full power days until two successive maps indicate that FQ(Z) ~~the peak pin power, $F^N_{\Delta H}$,~~ is not increasing.

A3.2-03

M3.2-19

4. In MODE 1 with power > 15% RTP~~Except during PHYSICS TESTS, and except~~ as provided by specifications 5 through 8 below, the indicated axial flux difference

LCO3.2.3 LCO a

A3.2-23

~~for at least three operable excore channels~~ shall be maintained within the target band about the target flux difference. The target band is specified in the CORE OPERATING LIMITS REPORT.

L3.2-24

LCO3.2.3 Note 1

The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.

A3.2-26

5. Above or equal to 90 percent of RATED THERMAL POWER:

LCO3.2.3 Cond A Cond B

A3.2-27

If the indicated axial flux difference of two OPERABLE excore channels deviates from the target band, within 15 minutes ~~either~~ eliminate such deviation, or within another 15 minutes reduce THERMAL POWER to less than 90 percent of RATED THERMAL POWER.

L3.2-28

6. Above or equal to ~~Between~~ 50 and below 90 percent of RATED THERMAL POWER: A3.2-27

LCO3.2.3 LCO b

a. The indicated axial flux difference may deviate from the target band for a maximum of one* hour (cumulative) in any 24- hour period provided that the difference between the indicated axial flux difference about the target flux difference does not exceed the envelope specified in the CORE OPERATING LIMITS REPORT.

LCO3.2.3 Cond C

b. If 6.a is violated for two OPERABLE excore channels then the THERMAL POWER shall be reduced to less than 50% of RATED THERMAL POWER within 30 minutes M3.2-31
NEW NOTE: Once this condition is entered, power must be reduced as required by this Required Action.

~~and the high neutron flux setpoint reduced to less than 55% of RATED THERMAL POWER.~~ L3.2-32

LCO3.2.3 Note 4

* May be extended to 16 hours without penalty deviation time during incore/excore calibration per ITS SR 3.3.1.6. A3.2-33

~~** For Unit 1, Cycle 19, when the number of available moveable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 5% measurement uncertainty shall be increased to [5% + (3-T/9)(3%)] where T is the number of available thimbles.~~ A3.2-05

~~3.10.B.6~~.c. A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference ~~of at least three OPERABLE excore channels~~ being within the target band. [LCO3.2.3 LCO a] [L3.2-24]

~~7.~~ <u>Less than 50 percent of RATED THERMAL POWER:</u>

[LCO3.2.3 LCO c] a. The indicated axial flux difference may deviate from the target band.

[LCO3.2.3 LCO b] b. A power increase to a level greater than 50 percent of RATED THERMAL POWER is contingent upon the indicated axial flux difference ~~of at least three OPERABLE excore channels~~ not being outside the target band for more than one hour (cumulative) out of the preceding 24 hour period. [L3.2-24]

~~8.~~ <u>In applying 6a and 7b above, penalty deviations outside the target band shall be accumulated on a time basis of:</u>

[LCO3.2.3 Note 2] a. One minute penalty deviation for each one minute of power operation outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

[LCO3.2.3 Note 3] b. One-half minute penalty deviation for each one minute of power operation outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

[LR3.2-34]

~~9. If alarms associated with monitoring the indicated axial flux difference deviations from the target band are not operable, the indicated axial flux difference value for each OPERABLE excore channel shall be logged at least once per hour for the first 24 hours and half-hourly thereafter until the alarms are returned to an OPERABLE status. For the purpose of applying this specification, logged values of indicated axial flux difference must be assumed to apply during the previous interval between loggings.~~

[SR3.2.3.1] NEW SR, Verify AFD is within limits. [M3.2-36]

3.2.4~~C.~~ <u>QUADRANT POWER TILT RATIO</u>

[A3.2-37]

[LCO3.2.4] 1. During MODE 1 with Thermal Power > 50% RTP ~~Except for PHYSICS TESTS,~~ if the QUADRANT POWER TILT RATIO exceeds 1.02

~~but is less than 1.07,~~ perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1 24 hours after achieving equilibrium. [M3.2-38]
~~the rod position indication shall be monitored and logged once each shift to verify rod position within each bank assignment and,~~ within two hours, one of the following steps shall be taken:

[LCO3.2.4] a. Correct the QUADRANT POWER TILT RATIO to less than 1.02.

[LCO3.2.4 Cond A] b. Restrict core power level so as not to exceed RATED THERMAL POWER less 3~~2~~% for every 0.01 that the QUADRANT POWER TILT RATIO exceeds 1.0. [M3.2-39]

LCO3.2.4 Cond A | AND (Inserts TS 3.10-5 A and B) | M3.2-41

LCO3.2.4 Cond B | If Required Action A and associated Completion Time are not met, reduce Thermal Power to ≤ 50% RTP within 4 hours. | M3.2-41

SR3.2.4.1 | New SR 3.2.4.1, Verify QPTR is within limit by calculation. | M3.2-43

~~3.10.C.2.~~ ~~If the QUADRANT POWER TILT RATIO exceeds 1.02 but is less than 1.07 for a sustained period of more than 24 hours, or if such a tilt recurs intermittently, the reactor shall be brought to the HOT SHUTDOWN condition. Subsequent operation below 50% of rating, for testing, shall be permitted.~~ L3.2-44

~~3. Except for PHYSICS TESTS if the QUADRANT POWER TILT RATIO exceeds 1.07, the reactor shall be brought to the HOT SHUTDOWN condition. Subsequent operation below 50% of rating, for testing, shall be permitted.~~ L3.2-44

SR3.2.4.2 ~~4.~~ If the core is operating above 85% power with one excore nuclear channel inoperable, then the core quadrant power balance shall be determined each shift ~~daily and after a 10% power change~~ M3.2-46

using either ~~2~~ movable detectors or ~~4~~ core thermocouples ~~per quadrant, per Specification 3.11~~. LR3.2-47

Addressed Elsewhere

D. Rod Insertion Limits

1. The shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT when the reactor is critical or approaching criticality.

2. When the reactor is critical or approaching criticality, the control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT.

3. Insertion limits do not apply during PHYSICS TESTS or during periodic exercise of individual rods. The shutdown margin specified in the Core Operating Limits Report must be maintained except for low power PHYSICS TESTING. For this test the reactor may be critical with all but one high worth full-length control rod inserted for a period not to exceed 2 hours per year provided a rod drop test is run on the high worth full-length rod prior to this particular low power PHYSICS TEST.

Insert TS 3.10-5 A

M3.2-41

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	AND	
	A.2 Determine QPTR.	Once per 12 hours
	AND	
	A.3 Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from THERMAL POWER reduction per Required Actions A.1 AND Once per 7 days thereafter
	AND	
	A.4 Re-evaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND	

A. (continued)	A.5 -------NOTES--------- 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed when Required Action A.5 is performed. -------------------- Normalize excore detectors to restore QPTR to within limits.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>	
	A.6 -------NOTE--------- Perform Required Action A.6 only after Required Action A.5 is completed. -------------------- Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1

~~3.10.I. Monitor Inoperability Requirements~~

Addressed Elsewhere

~~1. If the rod bank insertion limit monitor is inoperable, or if the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift, after a load change greater than 10 percent of RATED THERMAL POWER, and after 30 inches or more of rod motion.~~

~~2. If both the rod position deviation monitor and one or both of the quadrant power tilt monitors are inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93% of RATED THERMAL POWER in addition to the increased surveillance requirements.~~ LR3.2-48

~~3. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours after a load change greater than 10% of RATED THERMAL POWER~~ LR3.2-48

Addressed Elsewhere

J. DNB Parameters

The following DNB related parameters limits shall be maintained during POWER OPERATION:

a. Reactor Coolant System Tavg ≤564°F*
b. Pressurizer Pressure ≥2220 psia*
c. Reactor Coolant Flow ≥the value specified in the CORE OPERATING LIMITS REPORT

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours. Compliance with c. is demonstrated by verifying that the parameter is within its limit after each refueling cycle.

*Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER.

Action Statements

Addressed Elsewhere

ACTION 1: With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours.

ACTION 2: With the number of OPERABLE channels less than the Total Number of Channels HOT STANDBY and/or POWER OPERATION may proceed provided the following conditions are satisfied:

SR3.2.4.2

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.1, and

c. If THERMAL POWER is above 85% of RATED THERMAL POWER, then determine the core quadrant power balance in accordance with the requirements of Specification 3.10.C.4.

SR3.2.4.2

d. One additional channel may be taken out of service for low power PHYSICS TESTS.

ACTION 3: With the number of channels OPERABLE one less than the Total Number of Channels and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below the P-10 (Power Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-10 Setpoint.

ACTION 4: With the number of OPERABLE channels one less than the Total Number of Channels, suspend all operations involving positive reactivity changes.

ACTION 5: With the number of OPERABLE channels one less than the Total Number of Channels, suspend all operations involving positive reactivity changes and restore the inoperable channel to OPERABLE status within 48 hours or within the next hour open the reactor trip breakers.

~~TABLE TS.3.5-2A~~
~~Page 4 of 6~~
~~Rev 111 8/10/94~~

~~3.11 CORE SURVEILLANCE INSTRUMENTATION~~

R3.2-49

~~Applicability~~

~~Applies to the OPERABILITY of the moveable detector instrumentation system and the core thermocouple instrumentation system.~~

~~Objective~~

~~To specify OPERABILITY requirements for the moveable detector and core thermocouple systems.~~

~~Specification~~

~~A. The moveable detector system shall be OPERABLE following each refueling so that the power distribution can be confirmed. If the moveable detector system is degraded to the extent that less than 75% of the detector thimbles are available, the measurement error allowance due to incomplete mapping shall be substantiated by the licensee.~~

~~B. A minimum of 2 moveable detector thimbles per quadrant*, and sufficient detectors, drives, and readout equipment to map these thimbles, shall be operable during recalibration of the excore axial offset detection system per Specification 4.1. If this OPERABILITY for recalibration of excore nuclear instruments when required by Specification 4.1 cannot be achieved, power shall be limited to 90% of RATED THERMAL POWER until recalibration is completed in accor- dance with this specification.~~

~~C. A minimum of 4 thermocouples or 2 moveable detectors per quadrant shall be operable for readout if the reactor is operated above 85% of RATED THERMAL POWER with one excore nuclear power channel inoperable (see Specification 3.10.C.4).~~

~~D. The provisions of specification 3.0.C are not applicable.~~

~~* For Unit 1, Cycle 19, when the number of available moveable detector thimbles is greater than or equal to 50% and less than 75% of the total, there should be a minimum of two thimbles available per quadrant, where quadrant includes both horizontal-vertical quadrants and diagonally- bounded quadrants (eight individual quadrants in total).~~

A3.2-05

PACKAGE 3.2

POWER DISTRIBUTION LIMITS

PART D

DISCUSSION OF CHANGES
(DOC)

to

PRAIRIE ISLAND
CURRENT TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

PART D

PACKAGE 3.2

POWER DISTRIBUTION LIMITS

DISCUSSION OF CHANGES TO CURRENT TECHNICAL SPECIFICATIONS

The proposed changes to PI Operating License Appendix A, TS are discussed below and the specific wording changes are shown in parts B, C and E.

For ease of review, all package part and discussions are organized according to the proposed PI ITS Table of Contents.

NSHD Category	Change Number 3.2-	Discussion of Change
A	01	CTS 3.10. The beginning of each CTS section contains general statements of Applicability and Objectives for that TS section. This Applicability states the systems to which the specifications apply which is a different meaning than the Applicability in NUREG-1431. Since the ITS clearly states within each specification the system to which it applies, administratively these statements have been incorporated. Likewise, the CTS Objectives statement provides an overall purpose for the specifications within the section. These objectives are administratively incorporated in general through the statement of the ITS specification LCO and the supporting Bases. Since these general CTS statements do not establish any regulatory requirements and are incorporated in a broad sense in the ITS, these are considered administrative changes.

NSHD Category	Change Number 3.2-	Discussion of Change
A	02	CTS 3.10.B.1. CTS use prose descriptions of when the specification is applicable. In accordance with the guidance of NUREG-1431, a more precise applicability is defined as MODE 1. CTS require the hot channel factor limits to be met at all times, except for low power Physics Testing. In conformance with the guidance of NUREG-1431, the ITS requires hot channel factor limits to be met in MODE 1. Most low power Physics Tests are performed at power levels less than 5% RTP which is outside of MODE 1. Therefore this change is considered an administrative change.
A	03	CTS 3.10.B.1, 3.10.B.2, 3.10.B.3(a), 3.10.B.3(b), 3.10.B.3(d)1 and 3.10.B.3(d)2. CTS symbols for hot channel factors are different than those used in NUREG-1431. For consistency with the NUREG, the PI ITS has adopted the NUREG-1431 symbols. The entities to which the symbols refer have not changed; thus, this is an administrative change.
LR	04	CTS 3.10.B.1, 3.10.B.2, 3.10.B.3(a), 3.10.B.3(b), 3.10.B.3(b)2 and 3.10.B.3(d)1. The hot channel factor specific equations for determining compliance with the limits have been relocated to the COLR in conformance with the guidance of NUREG-1431. These equations are not required in the TS since they are part of the NRC approved methodologies used to determine the limits in the COLR as required by Administrative Controls in ITS 5.5. Since the COLR limits can be changed without prior NRC approval, relocation of these equations to the COLR is a less restrictive change.

NSHD Category	Change Number 3.2-	Discussion of Change
A	05	CTS 3.10.B.1, 3.10.B.3(d)1 and 3.11. Unit 1, Cycle 19 ended on April 16, 1999 and these CTS requirements which only applied to that specific operating cycle expired. Thus these requirements are not included in the ITS.
A	06	CTS 3.10.B.2. The phrase "whichever occurs first" has been deleted since both SRs will be required to be performed per ITS regardless of which condition occurs first. However, this change does not require more or less performances of these SRs; thus this is an administrative change.
	07	Not used.
M	08	CTS 3.10.B.2(b). This change will require $F_Q(Z)$ to be determined within 12 hours of reaching equilibrium conditions. Since CTS do not specify a time interval for performing these SRs, this change is more restrictive. This change is consistent with the guidance of NUREG-1431 and is included to make the ITS complete. This change will not cause a safety problem since this is an activity which is currently performed at the plant and conducting this SR within a specific time frame more frequently will not pose a safety concern.

NSHD Category	Change Number 3.2-	Discussion of Change
L	09	CTS 3.10.B.2(b). The CTS requirement to determine the target flux difference and $F_{\Delta H}$ following power changes of 10% or more are not included in the ITS since these parameters do not change significantly as power level is changed. Since this change may require less evaluations of core performance, this change is less restrictive. This change is consistent with the guidance of NUREG-1431.
	10	Not used.
M	11	New Requirement. An additional performance of the SRs to determine F^{C}_{Q} and $F_{\Delta H}$ is required to be consistent with the guidance of NUREG-1431. Since this change requires an additional performance of the SR after each startup, this change is more restrictive. This change will not cause a safety problem since this is an activity which is currently performed at the plant and conducting this as an SR following each refueling will not pose a safety concern.
M	12	CTS 3.10.B.3(a) and 3.10.B.3(b)2. This change requires power reduction within a specified time limit and resetting the high neutron flux trip setpoint within 72 hours following entry into the condition. CTS do not specify time limits for taking these remedial actions when these conditions are entered. These changes are more restrictive since these actions must be taken within a specific time. These changes are consistent with the guidance of NUREG-1431 and are included to make the ITS complete. These changes do not introduce a safety concern since the actions are the same; only a new time limit for their completion has been added.

NSHD Category	Change Number 3.2-	Discussion of Change
M	13	CTS 3.10.B.3(a). CTS require power to be reduced by a percentage specified in the COLR when these conditions are entered. ITS will require the power level to be reduced to < 50% and the neutron high flux trip setpoint to ≤ 55% when this condition is entered. Since a set power level is specified which is lower than the power levels which CTS requires, this is a more restrictive change. This change is consistent with the requirements of NUREG-1431. This change does not introduce any safety concerns since it will require taking the plant to a lower, more conservative, power level.
M	14	CTS 3.10.B.3(b)1. The CTS option of placing the reactor in an equilibrium configuration is not included in the ITS. Since this change may reduce plant operating flexibility, this is a more restrictive change. This change is consistent with the guidance of NUREG-1431. This change does not introduce any safety concerns since it only eliminates an option for remaining at power and may require the plant to take more conservative actions to shut down sooner.
	15	Not used.

NSHD Category	Change Number 3.2-	Discussion of Change
A	16	New requirement. New action statements are included to be consistent with the guidance of NUREG-1431 which require performance of SRs to determine the hot channel factors. Under CTS, these SRs would be required to be performed, even though they are not explicitly required, in order to determine that TS requirements are met prior to increasing power level. Therefore this change is considered administrative since it effectively does not make any changes to plant operations and does not require any new SRs.
L	17	New requirement. A new action statement is included which requires the plant to be in MODE 2 if the limitations and action statements for hot channel factors are not met. CTS in 3.10.B.3.c requires the unit to go to MODE 3 for this situation. Since this new action statement allows the plant to remain at a higher power level, this is a less restrictive requirement. This change is acceptable because, once the plant is in MODE 2, the core power level is low enough that the hot channel power factors are not of concern. This change is consistent with the guidance of NUREG-1431.

NSHD Category	Change Number 3.2-	Discussion of Change
L	18	CTS 3.10.B.3(c). CTS requirements to bring the plant to hot shutdown within 24 hours if the hot channel limits are not met are not included. For F_Q^W outside its limits, ITS will allow the plant to continue to reduce power 1% for each 1% that F_Q^W exceeds its limit. For $F_{\Delta H}$ outside its limits, ITS allow the plant to continue to operate at 50% RTP. Since this change may allow the plant to continue operation it is a less restrictive change. This change is acceptable since the power reductions (1% power reduction for each 1% deviation of F_Q^W or 50% RTP for $F_{\Delta H}$) are conservative enough to keep the plant in a safe configuration. This change is consistent with the guidance of NUREG-1431.
M	19	CTS 3.10.B.3(d). CTS requirements based on increasing $F_{\Delta H}$ have been replaced with a more appropriate requirement based on an increasing F_Q^C which is consistent with the guidance of NUREG-1431. The intent of the ITS SR and the notes for its application are essentially the same as the CTS. A review of predicted $F_{\Delta H}$ and F_Q^C for recent cycles indicates that the ITS SR will result in the additional actions being required more often than CTS. Since this change results in additional actions, this is a more restrictive change. Increasing the number of times F_Q^W must be multiplied by an appropriate factor or measuring more frequently, will not result in any new safety or operational concerns.
	20	Not used.
	21	Not used.

NSHD Category	Change Number 3.2-	Discussion of Change
A	22	CTS 3.10.B.3(d)1. Within the Note for SR 3.2.1.2, is a requirement to re-verify that F_Q^W is within limits. This is an administrative change since CTS would also require re-verification under these conditions.
A	23	CTS 3.10.B.4. CTS use prose descriptions of when the specification is applicable. In accordance with the guidance of NUREG-1431, a more precise applicability is defined as MODE 1 with power > 15% RTP. CTS require the AFD to meet limits except during Physics Testing. However, specification limits are only provided for power levels down to 15% in CTS 3.10.B.8.b. Thus this change is also consistent with CTS requirements and this is an administrative change.
L	24	CTS 3.10.B.4, 3.10.B.6.c and 3.10.B.7. The CTS requirement that "at least three operable excore channels" shall be within the target band for AFD is not included in the PI ITS. The ITS states, "AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band". Thus conservative guidance is provided in ITS which assures that the plant is maintained in a safe operating condition and the statement regarding three operable channels is unnecessary in the LCO. A requirement that three channels are in the band may cause confusion due to an undefined condition as follows. If one channel is inoperable and another channel indicates AFD is outside the band, then the plant would be without TS guidance since two channels would be inside the band. Since this requirement is not included in the ITS this is a less restrictive change.

NSHD Category	Change Number 3.2-	Discussion of Change
	25	Not used.
A	26	CTS 3.10.B.4. A statement is included in the ITS which considers AFD to be outside the target band when two or more OPERABLE excore channels indicate it is outside the target band for all power levels. This single statement replaces CTS 3.10.B.5 and 3.10.B.6.b statements which apply to different power levels. Since the intent of these requirements is the same, this is an administrative change.
A	27	CTS 3.10.B.5 and 3.10.B.6. CTS do not accurately define the ranges of applicability for the action statements in that the conditions of being equal to 90% RTP or 50% RTP are not included in the specification. The condition of being "equal to" is included in the ITS with being above the stated power level since the actions for the higher power level are more restrictive. This change is considered administrative since it does not make any real changes in plant operation.
L	28	CTS 3.10.B.5. This change will allow an additional 15 minutes, beyond that allowed in CTS, to correct the AFD when operating at or above 90% RTP. Since this change allows additional time to take corrective action, this change is less restrictive. This change is acceptable since AFD changes slowly and an accident during this additional 15 minutes is very unlikely. This change is consistent with the guidance of NUREG-1431.
	29	Not used.

NSHD Category	Change Number 3.2-	Discussion of Change
	30	Not used.
M	31	CTS 3.10.B.6(b). In accordance with the guidance of NUREG-1431, a time limit of 30 minutes has been included for the completion time when AFD is not corrected for power levels at or above 50% RTP to 90% RTP and a note is included which requires completion of this Required Action once it is entered. Since CTS does not have any time limit and does not specifically require completion of the Required Action, these are more restrictive changes which are included to make the ITS complete. These changes are acceptable since reducing the power within a set time limit does not introduce safety concerns.
L	32	CTS 3.10.B.6(b). The CTS requirement to reduce the high neutron flux setpoint has not been included in the ITS. This requirement is not necessary in the TS to maintain the plant in a safe condition since most of the safety benefit is achieved by lowering the plant power level in accordance with ITS requirements. Also the AFD inputs into the over power ΔT and overtemperature ΔT trip functions provide protection against power excursions. This change is consistent with the guidance of NUREG-1431. Since this change deletes CTS requirements, this is a less restrictive change.
A	33	CTS 3.10.B.6(a). Clarification has been added to this calibration exception Note which makes it consistent with the guidance of NUREG-1431. These wording changes do not change the intent of the Note; thus, this is an administrative change.

NSHD Category	Change Number 3.2-	Discussion of Change
LR	34	CTS 3.10.B.9. The CTS requirements related to AFD monitor alarms are relocated to the TRM since the alarms themselves do not directly relate to the LCO limits. This change is consistent with the guidance of NUREG-1431 as modified by approved TSTF-110. As part of the TRM which by reference is part of the USAR, these requirements will continue to be under the regulatory controls through 10CFR50.59. Since the TRM (USAR) can be changed without prior NRC approval, this change is less restrictive.
	35	Not used.
M	36	A new SR, SR 3.2.3.1, has been included to verify AFD is within limits. This new SR is included to make the ITS complete and conform to the guidance of NUREG-1431. This is an activity that the plant currently performs; therefore, this new SR does not introduce safety concerns. Since this SR is now a TS requirement this is a more restrictive change.
A	37	CTS 3.10.C.1. CTS use prose descriptions of when the specification is applicable which are not very precise. In conformance with the guidance of NUREG-1431, the ITS provides a more precise Applicability which requires QPTR limits to be met in MODE 1 with the power > 50%. By letter dated December 21, 1998 from Tae Kim to Roger O. Anderson, the NRC concluded that CTS 3.10.C is not applicable at power levels less than 50%. Thus this is an administrative change.

NSHD Category	Change Number 3.2-	Discussion of Change
M	38	CTS 3.10.C.1. The CTS requirement to maintain QPTR below 1.07 and perform monitoring and logging has been replaced with the requirement to perform three SRs within 24 hours after achieving equilibrium. The 1.07 limit with concomitant monitoring and logging was chosen as a conservative limit which will assure the reactor is maintained in a safe configuration at all times. However, as a more conservative measure, the ITS requires that SRs are performed to confirm that the reactor is maintained in a safe configuration. Thus this change is more restrictive and is included to conform to the guidance of NUREG-1431 and provide a complete ITS. Performance of these SRs will not introduce safety concerns since they do not significantly impact plant operations.
M	39	CTS 3.10.C.b. The CTS requires that core power level be reduced by 2% for every 1% that the QPTR exceeds 1.0. In conformance with NUREG-1431 and new PI analyses, the ITS will require the power level to be reduced by 3% for every 1% that the QPTR exceeds 1.0. Since the power reduction is more, this is a more restrictive change. This is a more conservative operating restriction which assures that the reactor is maintained in a safe configuration and thus does not introduce safety concerns.
	40	Not used.

NSHD Category	Change Number 3.2-	Discussion of Change
M	41	New Action Statements are included in addition to those in CTS 3.10.C.1 to address remedial actions for QPTR outside of limits. These actions are quite different than those in CTS 3.10.C.2 and 3.10.C.3 and replace them in their entirety. These new Action Statements are consistent with the guidance of NUREG-1431 and are included for completeness. These new Action Statements assure that the reactor is maintained in a safe configuration and thus do not introduce safety concerns. Since this change involves new TS requirements it is a more restrictive change.
	42	Not used.
M	43	A new SR, 3.2.4.1, is included which requires verification that QPTR is within limit by calculation. Since this is a new TS requirement, this change is more restrictive. However, this change does not introduce safety concerns because this test is currently performed by the plant. This new SR is included to make the ITS complete.
L	44	CTS 3.10.C.2 and 3.10.C.3. The CTS Action Statements addressing remedial actions when QPTR limits are not met are not included since they are quite different from those in NUREG-1431. This change is acceptable because these Action Statements have been replaced by new Action Statements which also maintains the reactor in a safe configuration. Since this change deletes CTS requirements, this is a less restrictive change. This change conforms to the guidance of NUREG-1431.

NSHD Category	Change Number 3.2-	Discussion of Change
	45	Not used.
M	46	CTS 3.10.C.4. The surveillance frequency for this SR has been increased to require performance each shift rather than daily or after each 10% power change. Since power changes of 10% occur infrequently while in this condition, the requirement to perform this SR each shift is considered more restrictive. This change is acceptable because performance of this SR more frequently does not introduce safety concerns. This change is consistent with the guidance of NUREG-1431.
LR	47	CTS 3.10.C.4. The number of each type of instrument per quadrant for this SR has been relocated to the Bases. These specification details are unnecessary in the SR since they can be adequately controlled in the Bases. This change is consistent with the guidance of NUREG-1431. Since ITS Bases (under the Bases Control Program in Section 5.5 of the ITS) is licensee controlled, relocation of CTS requirements to the Bases is a less restrictive change.
LR	48	CTS 3.10.I.2 and 3.10.I.3. The CTS requirements related to rod position deviation monitors and quadrant power tilt monitors are relocated to the TRM since these monitors themselves do not directly relate to the LCO limits. This change is consistent with the guidance of NUREG-1431 as modified by approved TSTF-110. As part of the TRM which is part of the USAR, these requirements will continue to be under the regulatory controls through 10CFR50.59. Since the TRM (USAR) can be changed without prior NRC approval, this change is less restrictive.

NSHD Category	Change Number 3.2-	Discussion of Change
R	49	CTS 3.11. CTS on Core Surveillance Instrumentation, 3.11, has been relocated to the TRM which is by reference a part of the USAR. This specification has been relocated because it does not meet the NRC Policy Statement TS Selection Criteria for inclusion in the TS.
		The moveable detector and core thermocouple instrumentation systems are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident. Thus these systems do not satisfy Criterion 1.
		The moveable detector and core thermocouple instrumentation systems are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus these systems do not satisfy Criterion 2.
		The moveable detector and core thermocouple instrumentation systems are not a structure, system or component that is part of the primary success path which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents achallenge to the integrity of a fission product barrier. Thus, these systems do not satisfy Criterion 3.
		As discussed in Section 4.0 (Appendix A, page A-12) of WCAP-11618, the moveable detector system has not been identified as a significant risk contributor. NMC has reviewed this evaluation, considers it applicable to the PI plant and concurs with this assessment. The moveable detector system is not modeled in the PI site-specific PRA since it is a

NSHD Category	Change Number 3.2-	Discussion of Change
R	49	(continued)

non-significant risk contributor. Likewise, the core thermocouple system is not modeled in the PI site-specific PRA since it is a non-significant risk contributor. Thus, these systems do not satisfy Criterion 4.

For the reasons given above, the moveable detector and core thermocouple instrumentation systems do not satisfy the screening criteria for inclusion in the TS and have been relocated to the TRM which by reference is part of the USAR. Changes to the TRM will be controlled under the provisions of 10CFR50.59.

PACKAGE 3.2

POWER DISTRIBUTION LIMITS

PART E

MARKUP OF NUREG-1431
IMPROVED STANDARD TECHNICAL SPECIFICATIONS AND BASES

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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

3.2 POWER DISTRIBUTION LIMITS

PA3.2-61

3.2.1~~B~~ Heat Flux Hot Channel Factor ($F_Q(Z)$) ~~(F_Q Methodology)~~

LCO 3.2.1~~B~~ $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. -----NOTE----- Required Action A.3 shall be completed whenever this Condition is entered. -----------	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes after each $F_Q^C(Z)$ determination	TA3.2-62 TA3.2-63
	AND		
$F_Q^C(Z)$ not within limit.	A.2 Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 ~~8~~ hours after each $F_Q^C(Z)$ determination	TA3.2-64 TA3.2-63
	AND		
	A.3 ~~Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.~~	~~72 hours~~	CL3.2-66
	~~AND~~		
	~~A.4~~ Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1	TA3.2-62

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. --------NOTE-------- Required Action B.3 shall be completed whenever this Condition is entered. ---------------------- $F_Q^W(Z)$ not within limits.	B.1	Reduce THERMAL POWER ~~AFD limits~~ ≥ 1% RTP for each 1% $F_Q^W(Z)$ exceeds limit.	4 ~~2~~ hours after each $F_Q^W(Z)$ determination
	AND		
	B.2	Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours after each $F_Q^W(Z)$ determination
	AND		
	B.3	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action B.1
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

TA3.2-62

TA3.2-63

SURVEILLANCE REQUIREMENTS

---NOTE---
During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q^C(Z)$ is within limit.	~~Once after each refueling~~ ~~P~~Prior to ~~THERMAL POWER~~ exceeding 75% RTP after each refueling AND Once within ~~[~~12~~]~~ hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified AND 31 effective full power days (EFPD) thereafter

PA3.2-67

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-------------------NOTE------------------- If ~~$F_Q^W(Z)$ is within limits and~~ measurements indicate that the $$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by ~~a factor of [1.02]~~ an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits; or b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate that the $$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$ has not increased. -- Verify $F_Q^W(Z)$ is within limit.	TA3.2-62 PA3.2-67 Once within 12 hours after achieving equilibrium conditions after each refueling after ~~prior to~~ THERMAL POWER exceeds~~ing~~ 75% RTP <u>AND</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	Once within ~~[~~12~~]~~ hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. ---------NOTE--------- Required Actions A.2 and A.4~~3~~ must be completed whenever Condition A is entered. ----------------------	~~A.1.1 Restore $F^N_{\Delta H}$ to within limit.~~	~~4 hours~~	PA3.2-68
	~~OR~~		
	A.1~~.2.1~~ Reduce THERMAL POWER to < 50% RTP.	4 hours	
$F^N_{\Delta H}$ not within limit.	AND ~~AND~~		
	A.3~~1.2.2~~ Reduce Power Range Neutron Flux – High trip setpoints to ≤ 55% RTP.	72~~8~~ hours	TA3.2-64
	AND		
	A.2 Perform SR 3.2.2.1.	24 hours	
	AND		
		(continued)	

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4~~3~~	--------NOTE--------- THERMAL POWER does not have to be reduced to comply with this Required Action. ---------------------	
		Perform SR 3.2.2.1.	Prior to ~~THERMAL POWER~~ exceeding 50% RTP AND Prior to ~~THERMAL POWER~~ exceeding 75% RTP AND 24 hours after ~~THERMAL POWER~~ reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

PA3.2-69

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	~~Once after each refueling~~ Pprior to ~~THERMAL POWER~~ exceeding 75% RTP after each refueling AND 31 EFPD thereafter

PA3.2-67

3.2 POWER DISTRIBUTION LIMITS

3.2.3~~A~~ AXIAL FLUX DIFFERENCE (AFD) ~~(Constant Axial Offset Control (CAOC) Methodology)~~

PA3.2-71

LCO 3.2.3 The AFD:

a. Shall be maintained within the target band about the target flux difference. The target band is specified in the COLR.

~~NOTE~~

~~The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.~~

TA3.2-72

b. May deviate outside the target band with THERMAL POWER < 90% RTP but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.

~~NOTE~~

~~Penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.~~

TA3.2-72

c. May deviate outside the target band with THERMAL POWER < 50% RTP.

NOTES

1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.

TA3.2-72

2. With THERMAL POWER ≥ 50% RTP, penalty deviation time shall be accumulated on the basis of a 1 minute

penalty deviation for each 1 minute of power operation with AFD outside the target band.

TA3.2-72

3. With THERMAL POWER < 50% RTP and > 15% RTP, pPenalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within acceptable operation limits.

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP.

~~NOTE~~

~~A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within acceptable operation limits.~~

TA3.2-72

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ 90% RTP. AND AFD not within the target band.	A.1 Restore AFD to within target band.	15 minutes

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1	Reduce THERMAL POWER to < 90% RTP.	15 minutes
C. ---------NOTE--------- Required Action C.1 must be completed whenever Condition C is entered. --------------------- THERMAL POWER < 90% and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. OR THERMAL POWER < 90% and ≥ 50% RTP with AFD not within the acceptable operation limits.	C.1	Reduce THERMAL POWER to < 50% RTP.	30 minutes

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~D. ----------NOTE---------~~ ~~Required Action D.1 must be completed whenever Condition D is entered.~~ ----------------------------- ~~Required Action and associated Completion Time for Condition C not met.~~	~~D.1 Reduce THERMAL POWER to < 15% RTP.~~	~~9 hours~~ CL3.2-74

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel.	7 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
~~SR 3.2.3.2 -------------------NOTE-------------------~~ ~~Assume logged values of AFD exist during the preceding time interval.~~ ~~---~~ ~~Verify AFD is within limits and log AFD for each OPERABLE excore channel.~~	~~-----NOTE-----~~ ~~Only required to be performed if AFD monitor alarm is inoperable~~ ~~---------------~~ ~~Once within 15 minutes and every 15 minutes thereafter when THERMAL POWER ≥ 90% RTP~~ ~~AND~~ ~~Once within 1 hour and every 1 hour thereafter when THERMAL POWER < 90% RTP~~

TA3.2-76

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.2.3.2~~3~~	Determine and U~~U~~pdate target flux difference. ~~of each OPERABLE excore channel by:~~ ~~a. Determining the target flux difference in accordance with SR 3.2.3.4, or~~ ~~b. Using linear interpolation between the most recently measured value, and either the predicted value for the end of cycle or 0% AFD.~~	Once within 31 EFPD after each refueling AND 31 EFPD thereafter	TA3.2-77 PA3.2-78
~~SR 3.2.3.4~~	~~NOTE~~ ~~The initial target flux difference after each refueling may be determined from design predictions.~~ ~~Determine, by measurement, the target flux difference of each OPERABLE excore channel.~~	~~Once within 31 EFPD after each refueling~~ ~~AND~~ ~~92 EFPD thereafter~~	PA3.2-78

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. QPTR not within limit.	A.1	Reduce THERMAL POWER $\geq$ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination TA3.2-63
	AND		
	A.2	Determine QPTR~~Perform SR 3.2.4.1 and reduce THERMAL POWER $\geq$ 3% from RTP for each 1% of QPTR > 1.00~~.	Once per 12 hours TA3.2-80
	AND		
	A.3	Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Actions A.1 TA3.2-75 TA3.2-63 AND Once per 7 days thereafter
	AND		
	A.4	Re-evaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.5	--------NOTES--------- 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed when Required Action A.5 is performed. -------------------- Normalize~~Calibrate~~ excore detectors to restore~~show zero~~ QPTR to within limits.	 Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		
	A.6	--------NOTE--------- Perform Required Action A.6 only after Required Action A.5 is completed. -------------------- Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions at ~~reaching~~ RTP not to exceed ~~OR~~

TA3.2-63

TA3.2-63

TA3.2-75

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
		~~Within~~ 48 hours after increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	-------------------NOTES------------------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤< 8~~7~~5% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance~~ if adequate Power Range Neutron Flux channel inputs are not OPERABLE~~. --- Verify QPTR is within limit by calculation.	TA3.2-63 CL3.2-82 TA3.2-80 7 days ~~AND~~ ~~Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable~~ TA3.2-76
SR 3.2.4.2	-------------------NOTE------------------- Not~~Only~~ required to be performed until 12 hours after~~if~~ input from one~~ or more~~ Power Range Neutron Flux channel is~~ are~~ inoperable with THERMAL POWER >≥ 8~~7~~5% RTP. --- Verify QPTR is within limit using the movable incore detectors or thermocouples.	TA3.2-80 TA3.2-63 CL3.2-82 CL3.2-83 ~~Once within~~ 12 hours ~~AND~~ ~~12 hours thereafter~~

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1~~B~~ Heat Flux Hot Channel Factor ($F_Q(Z)$) ~~(F_Q Methodology)~~

PA3.2-60

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT ~~TILT~~ POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6~~7~~, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis. TA3.2-62 TA3.2-84

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near equilibrium ~~steady state~~ conditions. TA3.2-62

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents an equilibrium ~~steady state~~ condition, it does TA3.2-62

(continued)

BASES

BACKGROUND (continued)

not include the variations in the values of $F_Q(Z)$ which ~~that~~ are present during non-equilibrium situations~~,~~ such as load following or power ascension. TA3.2-62

To account for these possible variations, the equilibrium ~~steady state~~ value of $F_Q(Z)$ is adjusted as $F_Q^W(Z)$ by an elevation dependent factor that accounts for the calculated worst case transient conditions. TA3.2-62

Core monitoring and control under non-equilibrium ~~nonsteady state~~ conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion. TA3.2-62

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During transient conditions arising from events of moderate frequency (Condition II events) ~~a loss of forced reactor coolant flow accident~~, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition (Ref. 1); CL3.2-85

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1~~2~~); and

(continued)

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2~~-3~~).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

~~$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents.~~ The Large Break LOCA (LBLOCA) analysis is the analysis that determines the LCO limit for $F_Q(Z)$. The $F_Q(Z)$ assumed in the Safety Analysis for other postulated accidents is either equal to or greater than that assumed in the LBLOCA analysis. Therefore, this LCO provides conservative limits for other postulated accidents. CL3.2-86

$F_Q(Z)$ satisfies Criterion 2 of ~~the NRC Policy Statement~~10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \le \frac{CFQ}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \le \frac{CFQ}{0.5} K(Z) \qquad \text{for } P \le 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR.

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and is based on the Small Break LOCA analysis, and CL3.2-87

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

~~For this facility, the actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of [2.32], and K(Z) is a function that looks like the one provided in Figure B 3.2.1B-1.~~ PA3.2-88 CL3.2-101

For Constant ~~Relaxed~~ Axial Offset Control operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$. TA3.2-62

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results a ~~we obtain the~~ measured value ($F_Q^M(Z)$) of $F_Q(Z)$ is obtained. Then, PA3.2-89

$$F_Q^C(Z) = F_Q^M(Z) * (\cancel{[}1.0815\cancel{]})$$

where ~~[~~1.0815~~]~~ is a factor that accounts for fuel manufacturing tolerances ~~and~~ (1.03) multiplied by a CL3.2-91

(continued)

---

factor associated with the flux map measurement uncertainty (1.05) (Ref. 3). CL3.2-91

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

LCO (continued)

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z)\ VW(Z)$$ CL3.2-92

where VW(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. VW(Z) is included in the COLR. The $F_Q^C(Z)$ is calculated at equilibrium conditions. TA3.2-62

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO precludes core power distributions that could violate the assumptions ~~requires operation within the bounds assumed~~ in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ ~~$F_Q(Z)$~~ cannot be maintained within the LCO limits, reduction of the core power is required. PA3.2-93 TA3.2-62

Violating the LCO limits for $F_Q(Z)$ could result in ~~produces~~ unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits. PA3.2-93

(continued)

BASES

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by ~~a~~ factors accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit. TA3.2-63

(continued)

BASES

ACTIONS (continued)

A.2

A reduction of the Power Range Neutron Flux—High trip setpoints by ≥ 1% for each 1% by which $F^C_Q(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 ~~8~~ hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux-High trip setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F^C_Q(Z)$ and would require Power Range Neutron Flux-High trip setpoint reductions within 72 hours of the $F^C_Q(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux-High trip setpoints. Decreases in $F^C_Q(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux-High trip setpoints.

TA3.2-64

TA3.2-63

A.3

~~Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which~~ $\cancel{F^C_Q(Z)}$ ~~exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.~~

CL3.2-66

(continued)

~~A.4~~

Verification that $F_Q^C(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels, and future operations, are consistent with safety analyses assumptions. TA3.2-62

Condition A is modified by a Note that requires Required Action A.3 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.3. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER. TA3.2-62

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the THERMAL POWER~~AFD~~ by ≥ 1% RTP for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4~~2~~ hours, maintains an acceptable absolute power density ~~restricts the axial flux distribution~~ such that even if a transient occurred, core peaking factors are not exceeded. TA3.2-62

(continued)

BASES

B.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action B.1. TA3.2-62

B.3

Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action B.1, ensures that core conditions during operation at higher power levels, and future operation, are consistent with safety analyses assumptions. TA3.2-62

Condition B is modified by a Note that requires Required Action B.3 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when Condition B is exited prior to performing Required Action B.3. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER. TA3.2-62

(continued)

BASES

ACTIONS (continued)

C.1

If Required Actions A.1 through A.~~B-4~~ or B.1 through B.3 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. CL3.2-66 TA3.2-62

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F^C_Q(Z)$ and $F^W_Q(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F^C_Q(Z)$~~-and $F^W_Q(Z)$~~ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of $F^C_Q(Z)$ ~~these parameters~~ before exceeding 75% RTP. This ensures that some determination of $F_Q(Z)$ ~~$F^C_Q(Z)$ and $F^W_Q(Z)$ are~~ is made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F^C_Q(Z)$ and $F^W_Q(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In PA3.2-67

(continued)

BASES

the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured. TA3.2-62

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)$ ~~*~~ (~~[~~1.0815~~]~~) (Ref. ~~3~~4). $F_Q^C(Z)$ is then compared to its specified limits. CL3.2-91

The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$ limit is met during the power ascension following a refueling ~~when RTP is achieved~~, because peaking factors generally decrease as power level is increased. PA3.2-96

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required ~~[~~12~~]~~ hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with the power increase to stay within the LCO limits).

(continued)

BASES

The Frequency of 31 effective full power days (EFPD) is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated during the nuclear design process by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called VW(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by VW(Z) gives the CL3.2-92

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power above 50% RTP and directly with the function K(Z) provided in the COLR. TA3.2-62

The VW(Z) curve is provided in the COLR for discrete core elevations. Flux map data are ~~typically~~ taken for 61~~30 to 75~~ core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height: CL3.2-92 PA3.2-97

(continued)

BASES

a. Lower core region, from 0 to 10 ~~15~~% inclusive; and

b. Upper core region, from 90~~85~~ to 100% inclusive. CL3.2-98

The top and bottom 10~~15~~% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated ~~and found to be within its limit~~, an evaluation of the expression below is required to account for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation. TA3.2-65

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

$$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right],$$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by ~~a factor of [1.02],~~ an appropriate factor specified in the COLR, or to evaluate $F_Q(Z)$ more frequently, each EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection. PA3.2-67

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

Performing the Surveillance once within 12 hours after achieving equilibrium conditions in MODE 1 during the power ascension following a refueling ~~prior to exceeding 75% RTP~~ ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased. PA3.2-67

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^W(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^W(Z)$ values are being reduced sufficiently with the power increase to stay within the LCO limits). ~~$F_Q(Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, [12] hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.~~ PA3.2-67

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. USAR Section 14 ~~10 CFR 50.46, 1974~~.

2. ~~Regulatory Guide 1.77, Rev. 0, May 1974.~~

BASES

~~3.~~ AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2 ~~10 CFR 50, Appendix A, GDC 26~~. CL3.2-99

34. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

~~Delete Figure~~ ~~DELETED FIGURE~~

CL3.2-101

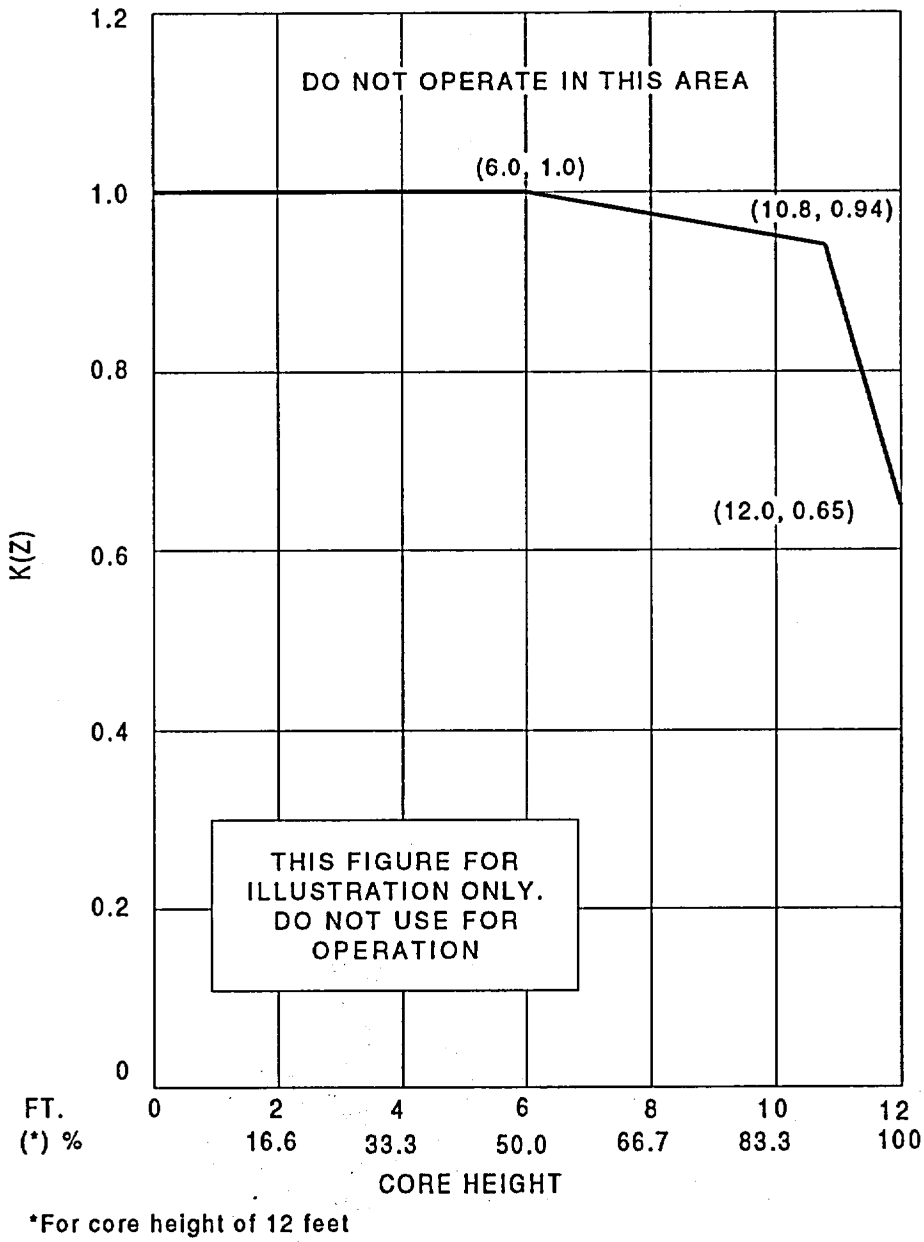

Figure B 3.2.1B-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

PA3.2-60

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion~~ and typically decreases with fuel burnup~~. PA3.2-102

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 effective full power days (EFPD). However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

(continued)

BASES

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling ratio (DNBR) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency (referred to as Condition II events). The departure from nucleate boiling (DNB) design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to a value greater than the criterion listed in Reference 1 ~~[1.3] using the [W3] CHF correlation~~. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements. CL3.2-85 PA3.2-104

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Controlling ~~Limits on~~ $F^N_{\Delta H}$ precludes core power distributions that exceed the following fuel design limits: PA3.2-107

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition during Condition II transients (Ref. 1); CL3.2-85

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

(continued)

BASES

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm ~~[~~(Ref. 1)~~]~~; and

d. ~~Fuel design limits required by GDC 26 (Ref. 2) for the condition when~~ The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2). CL3.2-99

For transients that may be DNB limited, the THERMAL POWER, Reactor Coolant System flow, temperature, pressure and $F^N_{\Delta H}$ are the core parameters of most importance. Except for Static Rod Cluster Control Assembly (RCCA) Misalignment and Dropped Rod events, t~~T~~he limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any Condition II transients ~~arising from events of moderate frequency~~. The analyses for Static RCCA Misalignment and Dropped Rod events ensure the DNB design basis is met by assuming a calculated $F^N_{\Delta H}$ plus uncertainties (Ref. 1). The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion listed in Reference 1 ~~of [1.3] using the [W3] CHF correlation~~. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

PA3.2-108
CL3.2-106
CL3.2-85
CL3.2-106
PA3.2-104

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all Condition II transients, except Static RCCA Misalignment and Dropped Rod events, that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation. CL3.2-85 CL3.2-106

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature [(Ref. [13)]. PA3.2-109

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6~~7~~, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)." TA3.2-84

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition I~~I~~ events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of ~~the NRC Policy Statement~~10 CFR 50.36(c)(2)(ii).

(continued)

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise~~. This channel has the least heat removal capability~~ and thus the highest probability for a DNB. PA3.2-111

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses as described in the Applicable Safety Analyses section above. PA3.2-122

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase by a factor specified in the COLR ~~0.3%~~ for every 1% RTP reduction in THERMAL POWER. CL3.2-112

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

(continued)

BASES

ACTIONS ~~A.1.1~~

~~With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.~~ PA3.2-68

~~However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding~~

ACTIONS A.1.1 (continued)

~~75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.~~

(continued)

<u>A.1~~.2.1~~ and A.~~1.2.2~~3</u>

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1~~.2.1~~ and reduce the Power Range Neutron Flux-High trip setpoint to ≤ 55% RTP in accordance with Required Action A.3~~1.2.2~~. Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1~~.2.1 is consistent with those allowed for in Required Action A.1.1 and~~ provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. ~~The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.~~ PA3.2-68

The allowed Completion Time of 72~~8~~ hours to reset the trip setpoints per Required Action A.3~~1.2.2~~ recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System. TA3.2-64

<u>A.2</u>

Once the power level has been reduced to < 50% RTP per Required Action A.1~~.2.1~~, an incore flux map (SR 3.2.2.1) must be obtained and the measured PA3.2-68

(continued)

BASES

value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by ~~either Action A.1.1 or~~ Action A.1~~.2.1~~. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which PA3.2-68

ACTIONS

A.2 (continued)

is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.~~4~~3 PA3.2-68

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

Condition A is modified by a Note that requires that Required Actions A.2 and A.4 must be completed whenever Condition A is entered. PA3.2-113

(continued)

BASES

B.1

When Required Actions A.1~~.1~~ through A.4~~3~~ cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems. PA3.2-68

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

BASES

REFERENCES

1. USAR Section 14 ~~Regulatory Guide 1.77, Rev. [0], May 1974~~. PA3.2-109

2. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2 ~~10 CFR 50, Appendix A, GDC 26~~. CL3.2-99

~~3. 10 CFR 50.46.~~

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3~~A~~ AXIAL FLUX DIFFERENCE (AFD) ~~(Constant Axial Offset Control (CAOC) Methodology)~~

PA3.2-60
PA3.2-71

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating scheme used to control the axial power distribution, Constant Axial Offset Control (CAOC), involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

The target flux difference is determined at equilibrium xenon conditions in conjunction with verifying $F_Q^W(Z)$ in accordance with SR 3.2.1.2. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 190~~210~~ steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the Fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by PA3.2-114 PA3.2-115

(continued)

BASES (continued)

multiplying the RTP value by the appropriate fractional THERMAL POWER level.

The AFD is logged manually or monitored on an automatic basis using the unit process computer that has an AFD monitor alarm. The frequency of monitoring the AFD by the unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFDs for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is $\geq$ 90% RTP. During operation at THERMAL POWER levels $<$ 90% RTP but $>$ 15% RTP, the computer sends an alarm message when the cumulative penalty deviation time is $>$ 1 hour in the previous 24 hours. TA3.2-76

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

(continued)

BASES (continued)

The CAOC and Transient Power Distribution methodologiesy~~y~~ (Refs. 1~~,~~ and 2~~, and 3~~) entails~~s~~: CL3.2-92

a. Establishing an envelope of allowed power shapes and power densities;

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes;

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

The Transient Power Distribution methodology (Ref. 2) determines a function, (V(Z)), that when applied to equilibrium $F_Q^W(Z)$ values will bound $F_Q^W(Z)$ values that could be measured at non-equilibrium conditions. This remains valid provided that the AFD is maintained within the target flux band around a target flux difference that was determined in conjunction with determining the equilibrium $F_Q^W(Z)$. PA3.2-114

The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition II~~2~~, III~~3~~, and IV4 events. This ensures that fuel cladding integrity is maintained for CL3.2-85

(continued)

BASES (continued)

these postulated accidents. The most important Condition IV~~4~~ event is the loss of coolant accident. The most significant Condition III~~3~~ event is the loss of RCS flow accident. The most significant Condition II~~2~~ events are uncontrolled bank withdrawal at power and Rod Cluster Control Assembly (RCCA) misalignment ~~and boration or dilution accidents. Condition 2 accidents, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower ΔT and Overtemperature ΔT trip setpoints~~. CL3.2-116 CL3.2-117

The limits on the AFD satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator, through either the manual operation of the control banks, or automatic motion of control banks responding to temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration, or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors ~~(Ref. 4)~~. Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire TA3.2-72

(continued)

BASES (continued)

cycle or more than one band, each to be followed for a specific range of cycle burnup and target flux difference.

With THERMAL POWER ≥ 90% RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition II, III, or IV event occurs while the AFD is outside its limits.

~~Part A of this~~ The LCO is modified by ~~a~~ four Notes. Note 1 ~~that~~ states the conditions necessary for declaring the AFD outside of the target band. With one channel removed from service (e.g., for calibration, testing or repairs), if two of the remaining channels indicate outside the target band, then the AFD shall be considered outside the target band. ~~The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup.~~ TA3.2-72 PA3.2-81 TA3.2-72

~~With THERMAL POWER ≥ 90% RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated.~~ TA3.2-72

~~Parts B and C of this LCO are modified by~~ Notes 2 and 3 ~~that~~ describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at TA3.2-72

(continued)

BASES

LCO (continued)

reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP (i.e., Part D~~B~~ of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside of the target band but within the acceptable operation limits provided in the COLR (Note 2). This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time when ~~within the power range of Part B of this LCO (i.e.,~~ THERMAL POWER ≥~~>~~ 50% RTP ~~but < 90% RTP)~~. The cumulative penalty time is the sum of penalty times from ~~Parts B and C of this~~ LCO Notes 2 and 3. TA3.2-72

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part C~~C~~ of this LCO), deviations of the AFD outside of the target band are less significant. Note 3 allows t~~T~~he accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band and reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time. TA3.2-72

~~The frequency of monitoring the AFD by the unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time.~~ TA3.2-76

(continued)

BASES

~~Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.~~ TA3.2-72

~~Figure B 3.2.3A-1 shows a typical target band and typical AFD acceptable operation limits.~~ CL3.2-120

For surveillance of the power range channels performed according to SR 3.3.1.6, Note 4 allows deviation outside the target band for 16 hours and no penalty deviation time accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days. TA3.2-72

APPLICABILITY

AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 3~~1~~).

Between 15% RTP and 90% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

APPLICABILITY (continued)

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events.

~~For surveillance of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is permitted for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days.~~ TA3.2-72

(continued)

BASES

Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER ≥ 90% RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

B.1

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP without allowing the plant to remain in an unanalyzed condition for an extended period of time.

C.1

With THERMAL POWER < 90% RTP but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour

ACTIONS

C.1 (continued)

if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the

(continued)

BASES

resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

Condition C is modified by a Note that requires that Required Action~~s~~ C.1 ~~and C.2~~ must be completed whenever this Condition is entered.

~~D.1~~

~~If Required Action C.1 is not completed within its required Completion Time of 30 minutes, the axial xenon distribution starts to become significantly skewed with the THERMAL POWER ≥ 50% RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% RTP, is no longer valid.~~ CL3.2-74

~~Reducing the power level to < 15% RTP within the Completion Time of 9 hours and complying with LCO penalty deviation~~

(continued)

BASES

~~time requirements for subsequent increases in THERMAL POWER ensure that acceptable xenon conditions are restored.~~

~~This Required Action must also be implemented either if the cumulative penalty deviation time is > 1 hour during the~~

ACTIONS

~~D.1 (continued)~~

~~previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.~~

~~Condition D is modified by a Note that requires Action D.1 be completed whenever this Condition is entered.~~

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

~~The AFD is monitored on an automatic basis using the unit process computer that has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFDs for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is > 90% RTP. During operation at THERMAL POWER levels < 90% RTP but > 15% RTP, the computer sends an alarm message when the cumulative penalty deviation time is > 1 hour in the previous 24 hours.~~ TA3.2-76

This Surveillance verifies that the AFD as indicated by the NIS excore channels is within the target band ~~and consistent with the status of the AFD monitor alarm~~. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Furthermore, any deviations of the AFD from the target band that is not alarmed should be readily noticed.

(continued)

BASES

~~SR 3.2.3.2~~

~~With the AFD monitor alarm inoperable, the AFD is monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at ≥ 90% RTP, the AFD is monitored at a Surveillance Frequency of 15 minutes to ensure that the AFD is within its limits at high THERMAL POWER levels. At power levels < 90% RTP, but > 15% RTP, the Surveillance Frequency is reduced to 1 hour because the AFD may deviate from the target band for up to 1 hour using the methodology of Parts B and C of this LCO to calculate the cumulative penalty deviation time before corrective action is required.~~ TA3.2-76

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1~~2~~ (continued)

~~SR 3.2.3.2 is modified by a Note that states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time.~~ The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band. TA3.2-76

SR 3.2.3.2~~3~~

This Surveillance requires that the target flux difference be determined and ~~is~~ updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup ~~by performing SR 3.2.3.4~~. PA3.2-78

The target flux difference is determined by averaging the indicated AFD from all OPERABLE excore channels. CL3.2-118

(continued)

BASES

To ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during non-equilibrium state conditions, the Transient Power Distribution methodology, i.e. V(Z), (Ref. 2) requires SR 3.2.1.2 to be performed in conjunction with this SR. PA3.2-114

Following a refueling outage, SR 3.2.1.2, and thus SR 3.2.3.2, are not required to be performed until equilibrium conditions are achieved. Since it may be desirable to provide the operators with some guidance for AFD control during the power ascension, a target flux difference may be posted based on engineering judgement or analytical prediction. PA3.2-128

~~Alternatively, linear interpolation between the most recent measurement of the target flux differences and a predicted end of cycle value provides a reasonable update because the AFD changes due to burnup tend toward 0% AFD. When the predicted end of cycle AFD from the cycle nuclear design is different from 0%, it may be a better value for the interpolation.~~ CL3.2-119

~~SR 3.2.3.4~~

~~Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.~~ PA3.2-78

~~A Frequency of 31 EFPD after each refueling and 92 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference for each excore channel to the value measured at steady state conditions. This is~~

(continued)

BASES

~~the basis for the CAOC. Remeasurement at this Surveillance interval also establishes the AFD target flux difference~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.2.3.4 (continued)~~

~~values that account for changes in incore excore calibrations that may have occurred in the interim. A Note modifies this SR to allow the predicted end of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.~~

REFERENCES

1. XN-NF-77-57, supplement 1(A), Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981 ~~WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974~~. CL3.2-92

2. Transient Power Distribution, NSPNAD-93003 ~~A T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC), Attachment: "Operation and Safety Analysis Aspects of an Improved Load Follow Package," January 31, 1980~~. CL3.2-92

3. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974 ~~C. Eicheldinger to D. B. Vassallo (Chief of Light Water Reactors Branch, NRC), Letter NS-CE-687, July 16, 1975~~.

~~4. FSAR, Chapter [15].~~

~~DELETE FIGURE~~ ~~DELETE FIGURE~~

CL3.2-120

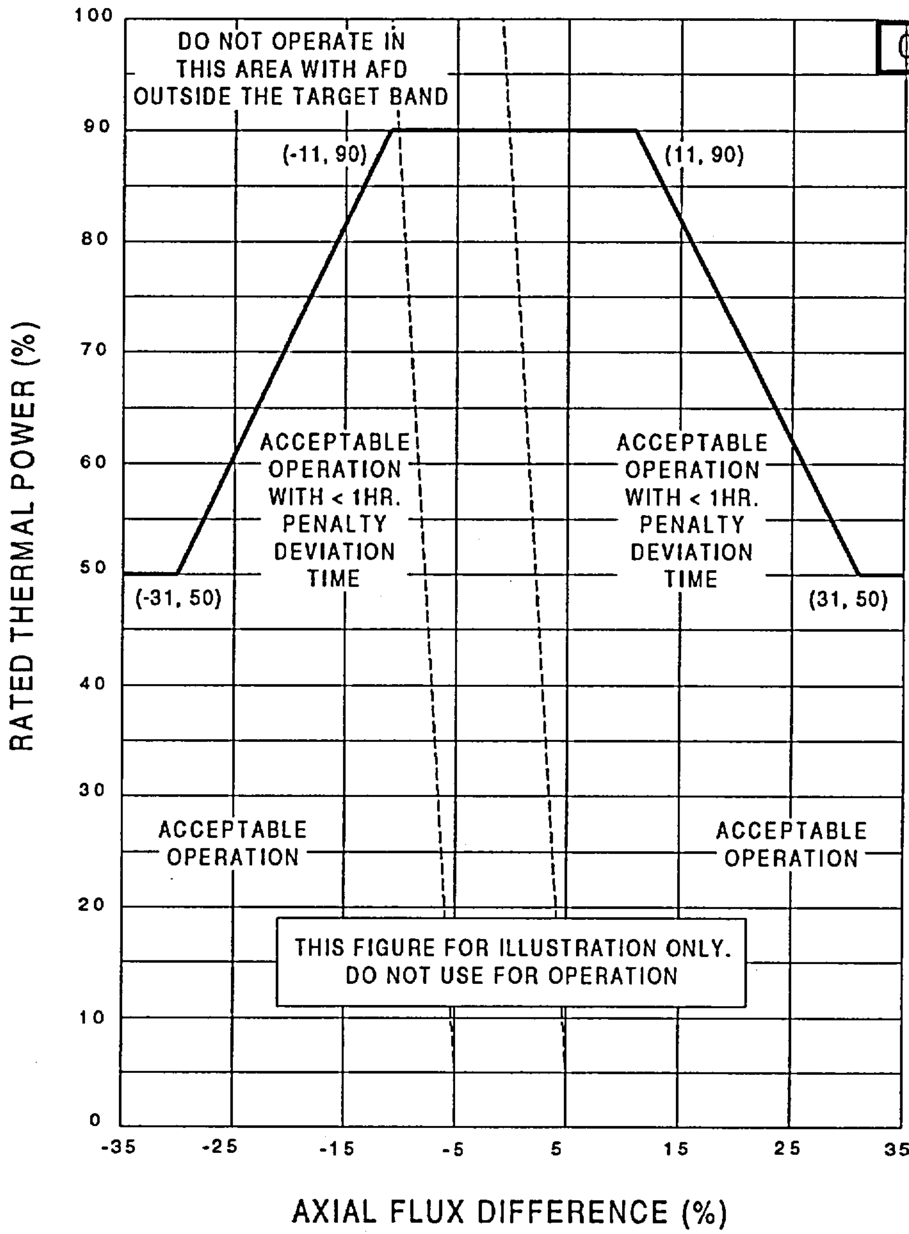

Figure B 3.2.3A-1 (Page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
and Target Band Limits as a Function
of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

PA3.2-60

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6~~7~~, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses. TA3.2-84

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During transient conditions arising from events of moderate frequency (Condition II events) ~~a loss of forced reactor coolant flow accident~~, there must CL3.2-85

(continued)

BASES

be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1~~2~~); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2~~3~~).

APPLICABLE SAFETY ANALYSES (continued)

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure the assumptions used in the safety analysis remain valid ~~that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values~~ by preventing an undetected change in the gross radial power distribution. PA3.2-123

In MODE 1, the QPTR~~, $F^N_{\Delta H}$ and $F_Q(Z)$ limits~~ must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses. PA3.2-124

The QPTR satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the assumptions in PA3.2-124

(continued)

BASES

the safety analysis are ~~margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is~~ possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient

ACTIONS

A.1 (continued)

time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in the QPTR would require power reductions within 2 hours of QPTR determination, if necessary to comply with the

TA3.2-63

(continued)

BASES

decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. ~~If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly.~~ A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow. TA3.2-63

A.3

TA3.2-75

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$, as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$, are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. A Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances

TA3.2-63

TA3.2-63

PA3.2-125

(continued)

BASES

provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ for ~~with~~ changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded

ACTIONS

A.4 (continued)

and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

(continued)

BASES

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limits ~~recalibrated to show a zero QPTR~~ prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR. TA3.2-63

Required Action A.5 is modified by two~~a~~ Notes. Note 1~~ that~~ states that the QPTR is not restored to within limits~~zeroed out~~ until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are~~This Note is~~ intended to prevent any ambiguity about the required sequence of actions. TA3.2-63

A.6

Once the flux tilt is restored to within limits ~~zeroed out~~ (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution~~ at RTP~~ is consistent with the safety analysis TA3.2-63

(continued)

BASES

assumptions, Required Action A.6 requires verification that $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within their specified limits within 24 hours of achieving equilibrium conditions at ~~reaching~~ RTP. As an added precaution, if the

TA3.2-75
PA3.2-129
TA3.2-63

ACTIONS

A.6 (continued)

core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1 ~~of the time when the ascent to power was begun~~. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits ~~calibrated to show zero tilt~~ (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limits ~~calibrated to show zero tilt~~ and the core returned to power. TA3.2-63

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not

(continued)

BASES

apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ ~~<~~ 8~~7~~5% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1 ~~if more than one input from Power Range Neutron Flux channels are inoperable~~.

TA3.2-63

PA3.2-82

TA3.2-80

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room ~~when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection~~.

TA3.2-76

~~When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because f~~For those causes of a core power tilt ~~QPT~~ that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

(continued)

BASES

SR 3.2.4.2

CL3.2-82

TA3.2-80

TA3.2-63

This Surveillance is modified by a Note, which states that it is not required until 12 hours after ~~only when~~ the input from one ~~or more~~ Power Range Neutron Flux channel~~s~~ is ~~are~~ inoperable and the THERMAL POWER is > ~~≥~~ 8~~7~~5% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that the QPTR ~~any tilt~~ remains within its limits.

PA3.2-126

For purposes of monitoring changes in radial core power distribution ~~the QPTR~~ when one power range channel is inoperable, at least 2 ~~the~~ moveable incore detectors or 4 thermocouples per quadrant may be used to calculate an incore core power tilt. This incore core power tilt may be used, instead of the excore detectors, to confirm that the QPTR is within the limits by comparing it to previous flux maps. ~~are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 for three and four loop cores.~~

CL3.2-83

CL3.2-127

(continued)

BASES

~~The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full~~ CL3.2-127

~~SURVEILLANCE REQUIREMENTS~~ ~~SR 3.2.4.2 (continued)~~

~~core flux map, to generate an incore QPTR. Therefore, QPTR can be used to confirm that QPTR is within limits.~~

~~With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.~~

REFERENCES

1. USAR Section 14 ~~10 CFR 50.46~~.

2. ~~Regulatory Guide 1.77, Rev [0], May 1974~~.

~~3.~~ AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2 ~~10 CFR 50, Appendix A, GDC 26~~. CL3.2-99

PACKAGE 3.2

POWER DISTRIBUTION LIMITS

PART F

JUSTIFICATION FOR DIFFERENCES
(JFD)

from

NUREG-1431
IMPROVED STANDARD TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

PART F

PACKAGE 3.2

POWER DISTRIBUTION LIMITS

JUSTIFICATION FOR DIFFERENCES FROM IMPROVED STANDARD TECHNICAL SPECIFICATIONS (NUREG-1431) AND BASES

See Part E for specific proposed wording and location of referenced deviations.

Difference Category	Difference Number 3.2-	Justification for Differences
PA	60	During the development of ITS, certain wording preferences, English conventions, reformatting, renumbering, providing additional descriptive information as related to PI, or editorial rewording consistent with plant specific nomenclature, system names, design, or current licensing bases were adopted. As a result of these changes, the TS should be more readily readable by, and therefore understandable to plant operators and other users. During this process, no technical changes were made to the TS unless they were identified and justified.
PA	61	The F_Q methodology which closest meets CTS is NUREG-1431 3.2.1B. The name of the methodology and "B" have been deleted since they are not needed as part of the ITS.

Difference Category	Difference Number 3.2-	Justification for Differences
TA	62	This change incorporates TSTF-290, Rev. 0. This traveller introduces a third methodology which is the most appropriate for PI. The traveler changes relating to reducing the OPΔT setpoints were not included since PI CTS do not require these setpoints to be reduced. See change CL3.2-66. SR 3.2.1.2 Note Paragraph a. was modified to only require an appropriate factor in the COLR. The NRC approved methodology will determine the upper limit on the factor by which F_Q^W must be increased and this factor will be included in the COLR. Reference to the Westinghouse methodology is not included since PI specific methodology will be developed and be referenced in ITS Section 5.6. TSTF-290 introduces a new Condition B with Required Actions and Completion Times similar to those in Condition A. The Completion Times for Condition B have been corrected to include the changes from TSTF-241 which did not recognize the existence of the new Condition B and therefore did not include the appropriate changes. Also Insert Note B was corrected to reference Condition B rather than Condition A.
TA	63	This change incorporates TSTF-241, Rev. 4. The traveler changes relating to reducing the OPΔT setpoints were not included since PI CTS do not require these setpoints to be reduced. See change CL3.2-66.

Difference Category	Difference Number 3.2-	Justification for Differences
TA	64	This change incorporates TSTF-95, Rev. 0.
TA	65	This change incorporates TSTF-97, Rev. 0.
CL	66	CTS do not require reduction in overpower ΔT trip setpoints when power is reduced for this condition. This is acceptable because most of the safety benefit is achieved by reducing the power level 1% for each 1% F_Q exceeds its limit. Further protection is provided by reducing the neutron flux high trip setpoint by the same amount.
PA	67	The wording for the SR Frequency and the Bases has been clarified, simplified and made more accurate for applicability to PI. Specifically with respect to SR 3.2.1.2, F_Q^W (Z) is undefined until equilibrium conditions have been established. Performing this SR prior to reaching 75% power would serve no purpose since the power escalation is controlled by the Physics Testing Program results. Therefore, the requirement to perform this SR prior to reaching 75% power is not included in the ITS. The wording in part a. of the Note, as modified by TSTF-290, is confusing. Since all appropriate limits are contained in the COLR and the note will require reference to the COLR, this note has been simplified by just stating that the factor is specified in the COLR.

Difference Category	Difference Number 3.2-	Justification for Differences
PA	68	Required Action A.1.1 is not included since restoration of compliance with the LCO is always an option and is not normally included per the Writer's Guide. Associated numbering changes and Bases changes have also been made. This change is consistent with proposed TSTF-240.
PA	69	The term "THERMAL POWER" is not included in any of the Completion Times since the meaning is clearer without this term.
	70	Not used.
PA	71	NUREG-1431 provides two specifications for AFD depending on the plant specific methodology for control. The method used at PI is closest to the CAOC method; thus, specification 3.2.3A has been included. The methodology name is not necessary in the title and has been deleted along with the "A" in the specification number.
TA	72	This change incorporates TSTF-164, Rev. 1.
	73	Not used.

Difference Category	Difference Number 3.2-	Justification for Differences
CL	74	This Condition is unnecessary and is not in CTS; thus, Condition D is not included. AFD is allowed to be outside the target band with THERMAL POWER less than 50% power since penalty time is accumulated which prevents resumption of power operation above 50% power. Therefore the plant is maintained in a safe condition and further power reduction is not necessary. Thus Condition D is not included. Since all of D is omitted, approved TSTF-112 is not included in the PI ITS.
TA	75	Incorporates TSTF-314, Rev. 0.
TA	76	This change incorporates TSTF-110, Revision 2. The paragraph relocated from the Bases SR to Background was modified to include the plant option of manually logging AFD. Also, the discussion does not include THERMAL POWER = 90%, so, > 90% was changed to ≥ 90%.
TA	77	This change incorporates TSTF-24, Rev. 1.
PA	78	NUREG-1431 SR 3.2.3.3 and SR 3.2.3.4 have been combined into one SR (SR 3.2.3.2) which requires the target AFD to be determined and updated. This change was made to be consistent with current plant practices and to make the ITS clearer for operator use.

Difference Category	Difference Number 3.2-	Justification for Differences
	79	Not used.
TA	80	This change incorporates TSTF-109, Rev. 0.
	81	Not used.
CL	82	CTS use 85% power level as the limit for requiring QPTR to be determined using incore instrumentation. Thus, 85% power is used as the point in the ITS for deciding which SR to perform.
CL	83	CTS specify use of incore thermocouple as one of the available instrumentation systems for verifying QPTR. Thus this option is retained in the ITS.
TA	84	Incorporates TSTF-136, Rev. 0.

Difference Category	Difference Number 3.2-	Justification for Differences
CL	85	Throughout the Bases for ITS Section 3.2, the term "Condition II" transients is defined and used to be more accurate. In ISTS 3.2.3, these definitions refer to condition 2, 3, etc. Since the PI USAR uses Roman numerals, these have been redefined as Condition II, III, etc. In ITS Bases for 3.2.1, 3.2.2 and 3.2.4 Applicable Safety Analyses fuel design limit criteria apply to Condition II events; therefore, Condition II was defined and these statements were generalized to apply to all Condition II events. (Condition II events include the loss of forced reactor coolant flow accident.)
CL	86	For PI the LBLOCA analysis sets the $F_Q(z)$ limit. Other transients may use an $F_Q(z)$ value that is equal to or greater than the LBLOCA value. These changes make this clearer and are consistent with the safety analyses for PI.
CL	87	The K(Z) function is based on SB LOCA. This change makes the Bases more accurate and consistent with the PI safety analyses.
PA	88	This paragraph is redundant and is not included in the PI ITS. Previous paragraphs in this Bases state that CFQ and K(Z) values are contained in the COLR.
PA	89	This statement was revised for operator clarity and to improve the flow of the context.

Difference Category	Difference Number 3.2-	Justification for Differences
	90	Not used.
CL	91	The value of 1.0815 is the result of multiplying 1.03 and 1.05. This change retains the specific information in the Bases for 1.0815 and is consistent with our CTS.
CL	92	PI uses an NMC generated penalty factor called V(Z) in accordance with the methodology in NSPNAD-93003-A "Transient Power Distribution". Likewise for AFD limits, NMC uses Exxon's XN-NF-77-57. Appropriate changes have been made in the Bases to reflect these methodologies.
PA	93	The phrasing has been revised to make the Bases clearer for operator use and to be consistent with other Bases sections.
	94	Not used.
	95	Not used.
PA	96	Performing the SR below 75% RTP does not ensure that F_Q will be within its limits at RTP. However it does ensure that the limits will be met during the power ascension and thus, this sentence was changed to be more accurate. The Bases was also revised to be consistent with the SR.

Difference Category	Difference Number 3.2-	Justification for Differences
PA	97	The flux map data core elevations was revised to the specific PI value.
CL	98	PI CTS require the V(Z) penalty factor to be applied in the middle 80% of the core height; thus, these statements have been revised.
CL	99	PI is not designed to and has not committed to 10CFR50 Appendix A. At the time PI was designed and licensed, the AEC GDC were under consideration. PI committed to these draft GDC with caveats as discussed in FSAR (now USAR). Since AEC GDC does not talk about fuel design limits, this paragraph is reworded to be consistent with the fuel design limit criteria in Bases 3.2.1 and 3.2.4.
	100	Not used.
CL	101	The normalized function K(Z) is defined in the COLR for CTS and will continue to be in the COLR for the ITS; thus, a figure defining K(Z) is not included.
PA	102	This phrase was not included since $F_{\Delta H}$ may increase during the cycle.
	103	Not used.

Difference Category	Difference Number 3.2-	Justification for Differences
PA	104	The numerical value of the DNB criteria and correlation were replaced with reference to USAR Section 14. These criteria and correlation could change depending on the type of fuel used and the fuel vendor; thus, referencing the USAR would avoid revising the Bases.
	105	Not used.
CL	106	The Static RCCA Misalignment and Dropped Rod transients are Condition II events but the analyses of these events do not assume that the reactor starts at the TS $F_{\Delta H}$ limit. Therefore, these statements were revised to make the Bases technically correct.
PA	107	The word "limits" could be misinterpreted to be referring to the limits in the LCO. This could be incorrect because the Static RCCA Misalignment and Dropped Rod transients do not start at the limits on $F_{\Delta H}$. Therefore the word "controlling" was used instead.
PA	108	Since reactor power, RCS temperature, and pressure are just as important in the DNB calculations as flow and $F_{\Delta H}$, these were added.

Difference Category	Difference Number 3.2-	Justification for Differences
PA	109	The safety analyses for the ejected rod and LOCA accidents are discussed in the USAR including the acceptance criteria. Since TS implements the requirements in the USAR safety analyses, the references were changed to reference the USAR Section 14.
	110	Not used.
PA	111	The phrase "This channel has the least heat removal capability" is not technically accurate. The channel with the highest enthalpy rise may have the highest amount of nucleate boiling, and thus the highest heat transfer coefficient. In any case the phrase is confusing and not necessary; thus, it was deleted.
CL	112	The numerical value for the increase in $F_{\Delta H}$ limit per % decrease in reactor power is in the COLR. The TS bases should not include these numerical limits; thus, this statement was revised.
PA	113	The Required Actions Note was relocated to be consistent with the format in other ITS Bases and support the correct Required Actions.

Difference Category	Difference Number 3.2-	Justification for Differences
PA	114	Implementation of NMC's Transient Power Distribution methodology requires that the $F_Q^W(z)$ be determined, that is, SR 3.2.1.2 performed, whenever the target flux difference is determined. Information was added to explain the relationship between the AFD specifications and the V(Z) penalties.
PA	115	The PI specific value is provided for Control Bank D normal operating position.
CL	116	For PI the limiting DNB events are rod withdrawal at power and RCCA misalignment; thus, this paragraph was revised.
CL	117	Previously it was stated that Condition II events are assumed to begin from within the AFD limits. The statement that Condition II events are used to confirm the adequacy of the ΔT trip setpoints is misleading. The setpoints are set and confirmed each cycle based on steady state conditions and not Condition II events. These setpoints are then assumed in the Condition II event analyses that demonstrate that the acceptance criteria have been met. Since none of this discussion is related to AFD, the sentence was deleted.
CL	118	Details were added on how the AFD is determined, per our current licensing basis.

Difference Category	Difference Number 3.2-	Justification for Differences
CL	119	NMC's Transient Power Distribution methodology does not allow linear interpolation between the most recent measurements and the predicted end of cycle value; therefore this information is not included.
CL	120	The figure is in the COLR in the CTS and will also be in the COLR to support the PI ITS. Therefore this sentence and ISTS figure are not included in the ITS.
	121	Not used.
PA	122	Reference is made to the previous discussion in the Applicable Safety Analyses to provide clarification.
PA	123	The QPTR does not ensure that the peaking factors will remain below their TS limits as implied in the original wording. The QPTR does ensure that the assumptions in the safety analyses remain valid.
PA	124	This Bases addresses QPTR, not Peaking Factors; thus, this sentence was revised.
PA	125	Minor editorial change to make the meaning clearer.

Difference Category	Difference Number 3.2-	Justification for Differences
PA	126	The QPTR is by definition based on normalized excore readings and not incore tilts. SR 3.2.4.2 allows the use of changes in incore tilts for verification that the QPTR is within the limits. These changes are intended to ensure that QPTR is not confused with incore tilts.
CL	127	Current licensing basis does not require the use of "symmetric" thimbles; thus, this information is not included in the ITS.
PA	128	Until equilibrium conditions are achieved, SR 3.2.1.2 can not be performed. Thus, SR 3.2.3.2 can not be performed. However, by updating a target flux difference, the operators are provided with some guidance for ΔI control. This paragraph is provided to give the operators some background on updating the target flux difference.
PA	129	Required Action A.6 requires performance of three surveillances of which two, SR 3.2.1.1 and SR 3.2.1.2, approximate $F_Q^c(z)$ and $F_Q^w(z)$ respectively. Thus clarification is provided on what is meant by verification of $F_Q(z)$.

PACKAGE 3.2

POWER DISTRIBUTION LIMITS

PART G

NO SIGNIFICANT HAZARDS DETERMINATION
(NSHD)

and

ENVIRONMENTAL ASSESSMENT

for

CHANGES TO PRAIRIE ISLAND
CURRENT TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

PART G

PACKAGE 3.2

POWER DISTRIBUTION LIMITS

NO SIGNIFICANT HAZARDS DETERMINATION AND ENVIRONMENTAL ASSESSMENT

NO SIGNIFICANT HAZARDS DETERMINATION

The proposed changes to the Operating License have been evaluated to determine whether they constitute a significant hazards consideration as required by 10CFR Part 50, Section 50.91 using the standards provided in Section 50.92.

For ease of review, the changes are evaluated in groupings according to the type of change involved. A single generic evaluation may suffice for some of the changes while others may require specific evaluation in which case the appropriate reference change numbers are provided.

A - Administrative (GENERIC NSHD)
(A3.2-01, A3.2-02, A3.2-03, A3.2-05, A3.2-06, A3.2-16, A3.2-22, A3.2-23, A3.2-26, A3.2-27, A3.2-33, A3.2-37)

Most administrative changes have not been marked-up in the Current Technical Specifications, and may not be specifically referenced to a discussion of change. This No Significant Hazards Determination (NSHD) may be referenced in a discussion of change by the prefix "A" if the change is not obviously an administrative change and requires an explanation.

These proposed changes are editorial in nature. They involve reformatting, renaming, renumbering, or rewording of existing Technical Specifications to provide consistency with NUREG-1431 or conformance with the Writer's Guide, or change of current plant terminology to conform to NUREG-1431. Some administrative changes involve

<u>A - Administrative</u> (continued)

relocation of requirements within the Technical Specifications without affecting their technical content. Clarifications within the new Prairie Island Improved Technical Specifications which do not impose new requirements on plant operation are also considered administrative.

1. <u>The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   The proposed conversion of Prairie Island Current Technical Specifications to conform to NUREG-1431 involves reformatting, rewording, changes in terminology and relocating requirements. These changes are simply editorial, or do not involve technical changes and thus they do not impact any initiators of previously analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. <u>The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.</u>

   These proposed administrative changes do not involve physical modification of the plant, no new or different type of equipment will be installed or removed associated with these administrative changes, nor will there be changes in parameters governing normal plant operation. The proposed administrative changes do not impose new or different requirements on plant operation. Therefore, these administrative changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. <u>The proposed amendment will not involve a significant reduction in the margin of safety.</u>

   These proposed administrative changes do not impact any safety analysis assumptions. Therefore, these changes do not involve a reduction in the plant margin of safety.

M - More restrictive (GENERIC NSHD)
(M3.2-08, M3.2-11, M3.2-12, M3.2-13, M3.2-14, M3.2-19, M3.2-31, M3.2-36, M3.2-38, M3.2-39, M3.2-41, M3.2-43, M3.2-46)

This proposed Technical Specifications revision involves modifying the Current Technical Specifications to impose more stringent requirements upon plant operations to achieve consistency with the guidance of NUREG-1431, correct discrepancies or remove ambiguities from the specifications. These more restrictive Technical Specifications have been evaluated against the plant design, safety analyses, and other Technical Specifications requirements to ensure the plant will continue to operate safely with these more stringent specifications.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes provide more stringent requirements for operation of the plant. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event.

   These more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed changes do not involve a physical alteration of the plant; that is, no new or different type of equipment will be installed, nor do they change the methods governing normal plant operation.

   These more stringent requirements do impose different operating restrictions. However, these operating restrictions are consistent with the boundaries established by the assumptions made in the plant safety analyses and licensing bases. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

<u>**M - More restrictive**</u> (continued)

3. <u>The proposed amendment will not involve a significant reduction in the margin of safety.</u>

   The imposition of more stringent requirements on plant operation either has no impact on the plant margin of safety or increases the margin of safety. Each change in this category is by definition providing additional restrictions to enhance plant safety by:

   a) increasing the analytical or safety limit;
   b) increasing the scope of the specifications to include additional plant equipment;
   c) adding requirements to current specifications;
   d) increasing the applicability of the specification;
   e) providing additional actions;
   f) decreasing restoration times;
   g) imposing new surveillances; or
   h) decreasing surveillance intervals.

   These changes maintain requirements within the plant safety analyses and licensing bases. Therefore, these changes do not involve a significant reduction in a margin of safety.

R - Relocation (GENERIC NSHD)
(R3.2-49)

This License Amendment Request (LAR) proposes to relocate requirements contained in the Current Technical Specifications out of the Technical Specifications into licensee controlled programs. These requirements are relocated because they 1) do not meet the Technical Specifications selection criteria defined in 10 CFR 50.36; or 2) are mandated by current Nuclear Regulatory Commission (NRC) regulations and are therefore unnecessary in the Technical Specifications.

In the NRC Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (dated 7/16/93), the NRC stated:

> . . . since 1969, there has been a trend towards including in Technical Specifications not only those requirements derived from the analyses and evaluations included in the safety analysis report but also essentially all other Commission requirements governing the operation of nuclear power reactors... This has contributed to the volume of Technical Specifications and to the several-fold increase, since 1969, in the number of license amendment applications to effect changes to the Technical Specifications. It has diverted both staff and licensee attention from the more important requirements in these documents to the extent that it has resulted in an adverse but unquantifiable impact on safety.

Thus, relocation of unnecessary requirements from the Current Technical Specifications should result in an overall improvement in plant safety through more focused attention to the requirements that are most important to plant safety.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   These proposed changes relocate requirements for structures, systems, components or variables which did not meet the criteria for inclusion in the improved Technical Specifications or duplicate regulatory requirements. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events.

R - Relocation (continued)

These relocated operability requirements will continue to be maintained pursuant to 10 CFR 50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), or the Administrative Controls section of these proposed improved Technical Specifications.

Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   These proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed changes do not impose any different requirements and adequate control of existing requirements will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   These proposed changes will not reduce the margin of safety because they do not impact any safety analysis assumptions. In addition, the relocated requirements for the affected structure, system, component or variables are the same as the current Technical Specifications. Since future changes to these requirements will be evaluated per the requirements of 10 CFR 50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), or the Administrative Control section of the Improved Technical Specifications, proper controls are in place to maintain the plant margin of safety. Therefore, these changes do not involve a significant reduction in the margin of safety.

LR - Less restrictive, Relocated details **(GENERIC NSHD)**
(LR3.2-04, LR3.2-34, LR3.2-47, LR3.2-48)

Some information in the Prairie Island Current Technical Specifications that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and relocated to the proposed Bases, Updated Safety Analysis Report or licensee controlled procedures. The relocation of this descriptive information to the Bases of the Improved Technical Specifications, Updated Safety Analysis Report or licensee controlled procedures is acceptable because these documents will be controlled by the Improved Technical Specifications required programs, procedures or 10CFR50.59. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes relocate detailed, descriptive requirements from the Technical Specifications to the Bases, Updated Safety Analysis Report or licensee controlled procedures. These documents containing the relocated requirements will be maintained under the provisions of 10CFR50.59, a program or procedure based on 10CFR50.59 evaluation of changes, or NRC approved methodologies. Since these documents to which the Technical Specifications requirements have been relocated are evaluated under 10CFR50.59 or its guidance, or in accordance with NRC approved methodologies, no increase in the probability or consequences of an accident previously evaluated will be allowed without prior NRC approval. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   These proposed changes do not necessitate physical alteration of the plant; that is, no new or different type of equipment will be installed, or change parameters governing normal plant operation. The proposed changes will not impose any different requirements and adequate control of the information will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

LR - Less restrictive, Relocated details (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

The proposed changes will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to the Bases, Updated Safety Analysis Report or licensee controlled procedures are the same as the existing Technical Specifications. Since future changes to these requirements will be evaluated under 10CFR50.59 or its guidance, or in accordance with NRC approved methodologies, no reduction in a margin of safety will be allowed without prior NRC approval. Therefore, these changes do not involve a significant reduction in a margin of safety.

L - Less restrictive, Specific

Each CTS change which is designated as Less (L prefix) restrictive on plant operations is provided with a specific NSHD.

Specific NSHD for Change L3.2-09

CTS requires F_Q^C, F_Q^W, $F_{\Delta H}^N$, and AFD to be determined when power reaches equilibrium conditions after exceeding by 10% or more the reactor power at which AFD was last determined. The ITS does not require $F_{\Delta H}^N$ and AFD to be determined following power level changes. This change is acceptable since these parameters do not change significantly when the power level is changed. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This proposed change would remove the requirement to determine two core parameters, $F_{\Delta H}^N$ and AFD following power level changes. These parameters will continue to be monitored as required by the TS before the power level reaches 75% RTP and every 31 EFPD thereafter. Prior to 75% RTP these parameters will be verified to be within their limits. Since these parameters do not change significantly with power level, the 31 EFPD verification will assure that the reactor is maintained in a safe condition. The associated hot channel factors, F_Q^C and F_Q^W, are verified when power has changed by 10% and will give an indication if there are significant changes in reactor power distribution characteristics. Also, the AFD for each excore channel is required to be checked every 7 days. Thus, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated because these parameters do not change significantly with power changes, they continue to be monitored, and other required monitoring will indicate adverse trends before they impact safe operation of the reactor.

Specific NSHD for Change L3.2-09 (continued)

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change makes the PI ITS consistent with the guidance of NUREG-1431 and does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The proposed change makes the PI ITS consistent with the guidance of NUREG-1431 and does not involve a significant reduction in margin of safety. During normal plant operations it is not necessary to measure the hot channel factors providing the reactor is operated in accordance with the following provisions: 1) Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position; 2) Control banks are sequenced with overlapping banks as required by the COLR; and 3) control bank insertion limits specified in the COLR are met. Also, axial power distribution procedures, which are given in terms of flux difference control and control bank insertion limits must be followed. Thus, with the plant operated within these limits, the $F^N_{\Delta H}$ and AFD will not change significantly when the power level is changed. Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.2-17

CTS do not provide Action Statements for the conditions when the remedial actions to correct hot channel factors are unsuccessful or the completion times are not met. Currently, the plant would enter CTS 3.0.C (ITS 3.0.3) which could eventually require the plant to go to cold shutdown (MODE 5). A new Action Statement is included in ITS which requires the plant to be in MODE 2 within 6 hours for these conditions. This change is acceptable since hot channel factors are not of concern at low power levels associated with entry into MODE 2. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change provides a new action statement requiring the plant to go to MODE 2 when the hot channel factors are not within limits and the required actions are not met. With the plant in MODE 2, the reactor has significant margin to DNB and design basis events are not of concern due to the low level of energy being transferred to the coolant. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident.

Specific NSHD for Change L3.2-17 (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The proposed change effectively removes the requirement to bring the plant to MODE 5 (cold shutdown) when the hot channel factors are not within limits and remedial actions to bring the hot channel factor within its limits are not met. Once the reactor is brought to MODE 2 with the power level less than 5% RTP, the energy being transferred to the coolant is low enough that limits on core power distribution are not required. Furthermore, design basis events which are sensitive to the hot channel factors have significant margin to DNB when the reactor is in MODE 2. Thus, this change which allows the reactor to remain in MODE 2 does not involve a significant reduction in margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.2-18

CTS action statements require the reactor to be taken to hot shutdown (MODE 3) when the hot channel factor limits are not met and the initial incremental power reductions (or other remedial actions) are not successful in restoring this factor to within limits. Subsequent operation up to 50% RTP is only allowed for physics testing. The proposed ITS allows the plant to continue operation at the reduced power level initially required by the Action Statements. This change is acceptable because the ITS initial actions require the reactor power to be reduced to a level at which the reactor can operate safely. If the heat flux hot channel factor limits are not met the reactor power is reduced 1% for each 1% the limit is exceeded. This power reduction maintains an acceptable absolute power density. If the nuclear enthalpy rise hot channel factor limits are not met the reactor power is reduced to 50% RTP. Reducing power below 50% RTP increases the DNB margin and reduces the likelihood of violating the DNBR limit in steady state operation. For both of these conditions, the power remains below the power level required by the specifications until the hot channel factor is restored within its limits. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change allows the plant to continue to operate in MODE 1 at reduced power levels until the hot channel factor which exceeds its limit is restored to within limits. This change does not involve a significant increase in the probability or consequences of an accident previously evaluated since the power is reduced to a level at which absolute power density and DNBR margin are acceptable.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change makes the PI ITS consistent with the guidance of NUREG-1431 and does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident.

Specific NSHD for Change L3.2-18 (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

The TS are intended to maintain the hot channel factors within acceptable limits. When these limits are not met, the Action Statements require power reductions which compensate, to some degree, for the hot channel factor which exceeds its limits. Furthermore, CTS also allow some power operation, up to 50% RTP for the purpose of Physics Testing, prior to restoring the hot channel factors within their limits. Thus, the proposed change makes the PI ITS consistent with the guidance of NUREG-1431 and does not involve a significant reduction in margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.2-24

The CTS requirement for at least three operable excore channels to be within the target band is not included in the ITS. CTS also requires three excore channels to be within the target band when the power level is increased. Its would allow power increases providing two excore channels are not outside the band. This change is acceptable since the ITS, like the CTS, assures the plant is operated safely by requiring remedial actions when two excore channels indicate the AFD is outside the band. This change avoids possible operator confusion when a channel is inoperable.

When all four excore channels are operable the phrase "two excore channels indicating the AFD is outside the band" and the phrase "at least three operable excore channels shall be maintained within the target band" are functional equivalents. Therefore this NSHD only address the change if one excore channel is inoperable.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The excore detectors are not accident precursors and the proposed change does not alter the configuration or operation of any plant equipment. Thus, this change does not involve a significant increase in the probability of a previously evaluated accident.

   The AFD for each channel is used as an indicator of the total core AFD. It is the total core AFD that is assumed in the accident analyses with a magnitude that bounds the target AFD. The proposed change allows power ascension when one channel is outside the target band and one excore channel is inoperable. During normal plant maneuvers all excore channels indicate approximately the same AFD, thus the difference in the total core AFD when "three operable channels indicate within the band' and when "two operable channels indicate inside the band" will be small. This small difference in total core AFD will not be large enough to challenge the AFD assumptions in the safety analyses. Therefore, the proposed change does not involve a significant increase in the consequences of an accident.

Specific NSHD for Change L3.2-24 (continued)

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant, that is, no new or different type of equipment will be installed. This proposed change does not introduce any new mode of plant operation or change the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The proposed change does not alter the configuration or operation of any plant equipment. It allows power ascension when one excore channel is outside the target band and one excore channel is inoperable. During normal plant maneuvers all excore channels indicate approximately the same AFD, thus the difference in the total core AFD when "three operable channels indicate within the band' and when "two operable channels indicate inside the band" will be small. Since the difference in the total core AFD will be small there will not be a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.2-28

The proposed change allows an additional 15 minutes to restore AFD within limits when operating at or above 90% power. This change is acceptable since AFD changes slowly and the probability of an accident during this time is very low. A 15 minute time period is very short and in custom TS terminology is generally considered the same as "immediately". This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change extends the completion time for restoring the AFD to within limits by 15 minutes. This change does not involve a significant increase in the probability of an accident previously evaluated because the probability of a design basis event during this additional 15 minute time period is extremely low. The completion time for restoring the AFD is not assumed to mitigate any analyzed event. Although the AFD is an initial condition of the design basis events, the increase in the completion time has an insignificant impact on consequences of an event compared to the benefit derived from the avoidance of an unnecessary plant transient (reduction in power). Therefore, this change does not involve a significant increase in the probability or consequences of a previously analyzed accident.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant, that is, no new or different type of equipment will be installed. This proposed change does not introduce any new mode of plant operation or change the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

<u>**Specific NSHD for Change L3.2-28**</u> (continued)

3. <u>The proposed amendment will not involve a significant reduction in the margin of safety.</u>

    This change extends, by 15 minutes, the time required to reduce power if AFD does not meet its limits. A 15 minute time period is very short and requiring power reduction in a shorter time period may introduce other safety concerns. Often in custom TS, 15 minutes is considered the same as "immediately". By allowing an additional 15 minutes an unnecessary plant transient (reduction in power) may be avoided. Therefore, in consideration of the offsetting changes, the proposed change does not result in a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.2-32

The proposed change removes the CTS requirement to reduce the high neutron flux setpoint when AFD limits are not met. This change is acceptable since the ITS required power reduction below 50% RTP places the reactor in a safe condition and the AFD inputs into the overpower delta-T and overtemperature delta -T trip functions provide protection against power excursions. Therefore, the risk of a reactor trip caused by the setpoint reduction is not justified by the potential consequences of not reducing the trip setpoints. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change involves compensatory actions required for the condition where AFD is outside limits. As such, the proposed change will not affect the probability of any initiating events assumed in the safety analyses. Since the proposed ITS will continue to provide an acceptable level of protection for transients involving conditions where AFD is outside the limits, the proposed change does not affect the consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of a previously analyzed accident.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant, that is, no new or different type of equipment will be installed. This proposed change does not introduce any new mode of plant operation or change the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Specific NSHD for Change L3.2-32 (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   This change eliminates the requirement to reduce the power range neutron flux trip setpoints which also results in reduced potential for reactor trip due to the act of changing the trip setpoint. The ITS requirement to reduce power to less than or equal to 50% RTP provides an acceptable level of protection when AFD is outside limits. Thus the proposed change does not result in a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.2-44

CTS Action Statements for QPTR outside its limits are based on a different philosophy for remediation and therefore are completely different than the Action Statements in NUREG-1431. In deference to NUREG-1431, the CTS Action Statements have been replaced in their entirety. This change is acceptable since the ITS, based on NUREG-1431, requires actions which maintain the reactor in a safe configuration. This change is consistent with NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change deletes CTS Action Statements and in their place provides NUREG-1431 Action Statements which provide equivalent reactor safety. Since the new Action Statements maintain the reactor in a safe configuration, these new action are not new accident initiators and they do not involve a significant increase in the probability of an accident previously evaluated. Similarly, since the new actions are intended to provide an equivalent level of reactor safety, this change does not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   This proposed change does not involve a physical alteration of the plant, that is, no new or different type of equipment will be installed. The proposed change only changes the Action Statements to provide protection equivalent to the CTS when QPTR exceeds its limit. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Specific NSHD for Change L3.2-44 (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The proposed change replaces CTS actions with Action Statements from NUREG-1431 which are intended to provide equivalent reactor protection. These new Action Statements require power reductions, similar to CTS requirements. If QPTR is not restored to within its limits, core re-evaluation is required. Thus, safe operation within the new Action Statements is assured. Therefore, this change does not involve a significant reduction in a margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

ENVIRONMENTAL ASSESSMENT

The Nuclear Management Company has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration, or

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51 Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51 Section 51.22(b), an environmental assessment of the proposed changes is not required.