PACKAGE 3.6

CONTAINMENT SYSTEMS

CROSS - REFERENCE

CURRENT TECHNICAL SPECIFICATIONS

TO

IMPROVED TECHNICAL SPECIFICATIONS

List of Section Cross - References

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 1.0				
1.0	ACTIONS	G	1.1	ACTIONS
New		G	1.1	ACTUATION LOGIC TEST
New		G	1.1	AFD
1.0	ABSVZ INTEGRITY		Relocated - Bases	
1.0	CHANNEL CALIBRATION	G	1.1	CHANNEL CALIBRATION
1.0	CHANNEL CHECK	G	1.1	CHANNEL CHECK
1.0	CHANNEL FUNCTIONAL TEST	G	1.1	CHANNEL OPERATIONAL TEST
1.0	CHANNEL RESPONSE TEST		Deleted	
1.0	CONTAINMENT INTEGRITY		Relocated - Bases	
1.0	CORE ALTERATION	G	1.1	CORE ALTERATION

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
1.0	COLR	G	1.1	COLR
1.0	DOSE EQUIV I-131	G	1.1	DOSE EQUIV I-131
1.0	E-AVE DISINTEGRATION	G	1.1	E-AVE DISINTEGRA-TION
1.0	LSSS		Deleted	
New		G	1.1	LEAKAGE
1.0	MODE	G	1.1	MODE
1.0	OPERABLE	G	1.1	OPERABLE
1.0	PHYSICS TESTS	G	1.1	PHYSICS TESTS
1.0	PTLR	G	1.1	PTLR
1.0	PROTECTION INSTR. AND LOGIC		Deleted	
1.0	QPTR	G	1.1	QPTR
1.0	RTP	G	1.1	RTP

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
New	CHANNEL RESPONSE TEST	G	1.1	RTS RESPONSE TIME
1.0	REPORTABLE EVENT	G	Relocated - TRM	
1.0	SHIELD BLDG INTEGRIT	G	Relocated - Bases	
1.0	SDM	G	1.1	SDM
1.0	SOURCE CHECK		Deleted	
1.0	STAGGERED TEST BASIS	G	1.1	STAGGERED TEST BASIS
1.0	STARTUP		Deleted	
1.0	THERMAL POWER	G	1.1	THERMAL POWER
New		G	1.1	TADOT
New		G	1.2	
New		G	1.3	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
New		G	1.4	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 3.3				
3.3.A.1.a		LCO	3.5.4	
New		SR	3.5.4.1	
3.3.A.1.a		SR	3.5.4.1	
3.3.A.1.a		SR	3.5.4.2	
3.3.A.1.b		LCO	3.5.1	
3.3.A.1.b.(1)		SR	3.5.1.1	
New		SR	3.5.1.1	
New		SR	3.5.1.2	
3.3.A.1.b.(2)		SR	3.5.1.2	
3.3.A.1.b.(3)		SR	3.5.1.4	
3.3.A.1.b.(4)		SR	3.5.1.3	
New		SR	3.5.1.3	
New		SR	3.5.1.5	
3.3.A.1.c		LCO	3.5.2	
3.3.A.1.d		LCO	3.5.2	
3.3.A.1.e		LCO	3.5.2	
New		LCO	3.5.3	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
New		SR	3.5.2.1	
New		SR	3.5.2.2	
New		SR	3.5.2.3	
New		SR	3.5.2.8	
New		SR	3.5.3.1	
3.3.A.1.f		LCO	3.5.3	
3.3.A.1.f		(Partial)	Relocated - Bases	
3.3.A.1.g (1)		SR	3.5.2.1	
3.3.A.1.g (1)		SR	3.5.2.3	
3.3.A.1.g (1)		(Partial)	Relocated - TRM	
3.3.A.1.g (2)		SR	3.5.2.1	
3.3.A.1.g (2)		SR	3.5.2.3	
3.3.A.1.g (2)		(Partial)	Relocated - TRM	
3.3.A.1.g (3)			Deleted	
3.3.A.1.g (4)			Relocated - TRM	
3.3.A.2		LCO	3.5.2	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
3.3.A.2.a			Relocated - Bases	
3.3.A.2.b			Relocated - Bases	
3.3.A.2.c			Relocated - Bases	
3.3.A.2.d			Relocated - Bases	
3.3.A.2.e		LCO	3.5.1	
New		LCO	3.5.1	
3.3.A.2.f		LCO	3.5.2	
3.3.A.2.g			Relocated - TRM	
New		LCO	3.5.4	
3.3.A.3		LCO	3.4.12	
3.3.A.3		LCO	3.4.13	
3.3.A.4		LCO	3.4.13	
3.3.A.5		LCO	3.4.12	
3.3.A.5		LCO	3.4.13	
3.3.B.1.a		LCO	3.6.5	
3.3.B.1.b		LCO	3.6.5	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
3.3.B.1.c		LCO	3.6.6	
3.3.B.1.c		(Partial)	Relocated - Bases	
3.3.B.1.d			Relocated - Bases	
3.3.B.1.e			Relocated - Bases	
New		LCO	3.6.5	
New		LCO	3.6.6	
3.3.B.2.a		LCO	3.6.5	
3.3.B.2.b		LCO	3.6.5	
New		SR	3.6.5.1	
3.3.B.2.c		LCO	3.6.6	
New		SR	3.6.6.1	
New		SR	3.6.6.2	
3.3.C.1.a		LCO	3.7.7	
3.3.C.1.a.1		LCO	3.7.7	
3.3.C.1.a.2			Relocated - Bases	
3.3.C.1.b		LCO	3.7.7	
3.3.C.1.b.1		LCO	3.7.7	
3.3.C.1.b.2			Relocated - Bases	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
New		SR	3.7.7.1	
3.3.C.2			Relocated - Bases	
3.3.D.1		LCO	3.7.8	
3.3.D.1.a			Relocated - Bases	
3.3.D.1.b			Relocated - Bases	
3.3.D.1.c			Relocated - Bases	
3.3.D.1.d		LCO	3.7.8	
New		LCO	3.7.8	
New		SR	3.7.8.3	
3.3.D.2		LCO	3.7.9	
3.3.D.2		LCO	3.7.8	
3.3.D.2.a		LCO	3.7.8	
3.3.D.2.a.(1)			Relocated - SFDP	
3.3.D.2.a.(2)			Relocated - SFDP	
3.3.D.2.a(3)		LCO	3.7.8	
3.3.D.2.b		LCO	3.7.8	
3.3.D.2.b(1)			Relocated - SFDP	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
3.3.D.2.b(2)		LCO	3.7.8	
3.3.D.2.b(2)		(partial)	Relocated - SFDP	
New		SR	3.7.8.1	
3.3.D.2.c		LCO	3.7.9	
3.3.D.2.d		LCO	3.7.9	
3.3.D.2.e		LCO	3.7.9	
New		SR	3.7.9.1	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 3.6				
3.6.A.1		LCO	3.6.1	
3.6.A.2		LCO	3.6.1	
3.6.B.1		LCO	3.6.8	
3.6.B.1		(Partial)	Relocated - Bases	
3.6.B.2		LCO	3.6.8	
3.6.B.3		LCO	3.6.8	
3.6.C.1		LCO	3.6.3	
New		LCO	3.6.3	
3.6.C.2		LCO	3.6.3	
3.6.C.3.(a)		LCO	3.6.3	
3.6.C.3.(b)		LCO	3.6.3	
3.6.C.3.(c)		LCO	3.6.3	
New		LCO	3.6.3	
New		SR	3.6.3.1	
New		SR	3.6.3.3	
New		SR	3.6.3.4	
New		SR	3.6.3.5	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
3.6.D.1			Relocated - Bases	
3.6.D.2.a			Relocated - Bases	
3.6.D.2.b		SR	3.6.3.6	
3.6.D.2.c		LCO	3.3.5	
3.6.D.2.d		LCO	3.3.5 B	
3.6.D.2.e		SR	3.6.3.2	
3.6.E.1		LCO	3.7.12	
3.6.E.2			Relocated - TRM/Bases	
3.6.E.3			Relocated - TRM/Bases	
3.6.F.1		LCO	3.7.12	
3.6.F.1		(partial)	Relocated - TRM/Bases	
3.6.F.2		LCO	3.7.12	
New		LCO	3.7.12	
3.6.G		LCO	3.6.10	
New		SR	3.6.10.1	
3.6.H		LCO	3.6.9	
3.6.I		LCO	3.6.4	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
New		SR	3.6.4.1	
3.6.J		SR	3.6.1.2	
3.6.K		SR	3.6.1.3	
3.6.L		LCO	3.6.7	
3.6.M		LCO	3.6.2	
New		LCO	3.6.2	
New		SR	3.6.2.2	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 4.4				
4.4.A.1		SR	3.6.1.1	
4.4.A.2		SR	3.6.2.1	
4.4.A.3		SR	3.6.3.8	
4.4.B.1		SR	3.6.9.5	
4.4.B.1		(Partial)	Relocated - TRM	
4.4.B.2		SR	3.7.12.3	
4.4.B.2		(Partial)	Relocated - Bases	
4.4.B.3		SR	3.6.9.2	
4.4.B.3		SR	3.7.12.2	
4.4.B.3		(Partial)	Relocated - VFTP	
4.4.B.3.a			Relocated - VFTP	
4.4.B.3.b			Relocated - VFTP	
4.4.B.3.c		SR	3.6.9.3	
4.4.B.3.c		SR	3.7.12.4	
4.4.B.3.c		(Partial)	Relocated - Bases	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
4.4.B.3.c		(Partial)	Relocated - ODCM	
4.4.B.4.a		SR	3.6.9.2	
4.4.B.4.a		SR	3.7.12.2	
4.4.B.4.a		(Partial)	Relocated - VFTP	
4.4.B.4.b			Relocated - VFTP	
4.4.B.4.c			Relocated - VFTP	
4.4.B.4.d		SR	3.6.9.1	
4.4.B.4.d		SR	3.7.12.1	
4.4.B.5		SR	3.6.9.2	
4.4.B.5		SR	3.7.12.2	
4.4.B.5		(Partial)	Relocated - VFTP	
4.4.C		SR	3.6.1.1	
4.4.C		SR	3.6.8.1	
4.4.D			CTS Deleted	
4.4.E		SR	3.6.3.7	
4.4.E		SR	3.6.9.4	
4.4.E		SR	3.7.12.4	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
4.4.F		SR	3.6.5.7	
4.4.G		SR	3.6.1.2	
4.4.H		SR	3.6.1.3	
4.4.I.a		SR	3.6.7.1	
4.4.I.a		(Partial)	Relocated - Bases	
4.4.I.b		SR	3.6.7.2	
4.4.I.b		(Partial)	Relocated - Bases	
4.4.I.c		SR	3.6.7.3	
4.4.I.c		(Partial)	Relocated - Bases	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 4.5				
4.5.A.1.a		(Partial)	Relocated - Bases	
4.5.A.1.a		SR	3.5.2.6	
4.5.A.1.b			Relocated - Bases	
4.5.A.2.a		SR	3.6.5.6	
4.5.A.2.a		(Partial)	Relocated - Bases	
4.5.A.2.b		SR	3.6.5.8	
4.5.A.2.c			Relocated - Bases	
4.5.A.3		SR	3.6.5.3	
4.5.A.3		(Partial)	Relocated - Bases	
4.5.A.4.a		SR	3.7.7.2	
4.5.A.4.a		SR	3.7.7.3	
4.5.A.4.b			Relocated - Bases	
4.5.A.5.a		SR	3.7.8.5	
4.5.A.5.a		SR	3.7.8.6	
4.5.A.5.a		(Partial)	Relocated - Bases	
4.5.A.5.b			Relocated - TRM	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
4.5.B.1.a		(Partial)	Relocated - IST	
4.5.B.1.a		SR	3.5.2.4	
4.5.B.1.a		SR	3.6.5.4	
4.5.B.1.b		SR	3.7.8.2	
4.5.B.1.b		(Partial)	Relocated - Bases	
4.5.B.1.c		SR	3.7.8.4	
4.5.B.1.c		(Partial)	Relocated - Bases	
4.5.B.2		SR	3.6.5.2	
4.5.B.2		(Partial)	Relocated - Bases	
4.5.B.3.a			Relocated - IST	
4.5.B.3.b			Relocated - IST	
4.5.B.3.c			Deleted by Boric Acid LAR	
4.5.B.3.d			Relocated - IST	
4.5.B.3.e		SR	3.7.8.5	
4.5.B.3.e		(Partial)	Relocated - Bases	
4.5.B.3.f		SR	3.5.2.5	
4.5.B.3.f		SR	3.6.5.5	
4.5.B.3.f		SR	3.6.6.4	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
4.5.B.3.f		SR	3.7.7.2	
4.5.B.3.f		SR	3.7.8.5	
4.5.B.3.g.1			Relocated - TRM	
4.5.B.3.g.2			Relocated - TRM	
4.5.B.3.g.3		SR	3.5.2.7	
4.5.B.3.h			Relocated - TRM	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Figure				
Figure 2.1-1		FIGURE	2.1.1-1	
Figure 3.1-3		FIGURE	3.4.17-1	
Figure 3.8-1		FIGURE	3.7.17-1	
Figure 3.8-2		FIGURE	3.7.17-2	
Figure 3.10-1			Relocated - COLR	
Figure 4.4-1			Relocated - TRM	
Figure 5.6-1		FIGURE	4.3.1-1	
Figure 5.6-2		FIGURE	4.3.1-2	
Figure 5.6-3		FIGURE	4.3.1-3	
Figure 5.6-4		FIGURE	4.3.1-4	
Figure 5.6-5		FIGURE	4.3.1-5	
Figure 5.6-6		FIGURE	4.3.1-6	
Figure 5.6-7		FIGURE	4.3.1-7	
Figure 5.6-8		FIGURE	4.3.1-8	
Figure 5.6-9		FIGURE	4.3.1-9	
Figure 5.6-10		FIGURE	4.3.1-10	
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
Figure 5.6-12		FIGURE	4.3.1-12	

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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
Table 3.5-2A	11	TABLE	3.3.1-1	9
Table 3.5-2A	12	TABLE	3.3.1-1	10
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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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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
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

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-2B	5a	LCO	3.7.2	
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	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
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	

Current Technical Specification Cross-Reference

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

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.15-1	9	TABLE	3.3.3-1	9
Table 3.15-1	10	TABLE	3.3.3-1	10
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	
Table 3.15-1	Action a1	LCO	3.3.3 C	
Table 3.15-1	Action a2	LCO	3.3.3 D	
Table 3.15-1	Action a2	LCO	3.3.3 I	
Table 3.15-1	Action a3	LCO	3.3.3 D	
Table 3.15-1	Action a3	LCO	3.3.3 J	
Table 3.15-1	Action a4	LCO	3.3.3 E	
Table 3.15-1	Action a4	LCO	3.3.3 I	
Table 3.15-1	Action a5	LCO	3.3.3 B	
Table 3.15-1	Action a5	LCO	3.3.3 C	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.15-1	Action a5	LCO	3.3.3	
Table 3.15-1	Action a6	LCO	3.3.3 F	
Table 3.15-1	Action a6	LCO	3.3.3 G	
Table 3.15-1	Action a6	LCO	3.3.3 I	
Table 3.15-1	New Cond	LCO	3.3.3 H	
Table 3.15-1	Action b	TABLE	3.3.3-1	Note a
Table 3.15-1	Action c	TABLE	3.3.3-1	Note b
Table 3.15-1	New Note	TABLE	3.3.3-1	Note c
Table 4.1-1A	1	TABLE	3.3.1-1	1
Table 4.1-1A	2a	TABLE	3.3.1-1	2a
Table 4.1-1A	2a	TABLE	3.3.1-1	6
Table 4.1-1A	2a	TABLE	3.3.1-1	7
Table 4.1-1A	2b	TABLE	3.3.1-1	2b
Table 4.1-1A	3	TABLE	3.3.1-1	3a
Table 4.1-1A	4	TABLE	3.3.1-1	3b
Table 4.1-1A	5	TABLE	3.3.1-1	4
Table 4.1-1A	6	TABLE	3.3.1-1	5
Table 4.1-1A	7	TABLE	3.3.1-1	6
Table 4.1-1A	8	TABLE	3.3.1-1	7

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1A	9	TABLE	3.3.1-1	8a
Table 4.1-1A	10	TABLE	3.3.1-1	8b
Table 4.1-1A	11	TABLE	3.3.1-1	9
Table 4.1-1A	12	TABLE	3.3.1-1	10
Table 4.1-1A	13	TABLE	3.3.1-1	14
Table 4.1-1A	14	TABLE	3.3.1-1	13
Table 4.1-1A	15	TABLE	3.3.1-1	12
Table 4.1-1A	16a	TABLE	3.3.1-1	11a
Table 4.1-1A	16b	TABLE	3.3.1-1	11b
Table 4.1-1A	17	TABLE	3.3.1-1	15
Table 4.1-1A	18	TABLE	3.3.1-1	19
Table 4.1-1A	19	TABLE	3.3.1-1	17
Table 4.1-1A	20	TABLE	3.3.1-1	17
Table 4.1-1A	New Func	TABLE	3.3.1-1	16
Table 4.1-1A	New Func	TABLE	3.3.1-1	18
Table 4.1-1A	Note 1	TABLE	3.3.1-1	Note a
Table 4.1-1A	Note 2	TABLE	3.3.1-1	Note d
Table 4.1-1A	Note 3	TABLE	3.3.1-1	Note b
Table 4.1-1A	Note 4	SR	3.3.1.8	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1A	Note 4a	SR	3.3.1.15	
Table 4.1-1A	Note 5	SR	3.3.1.2	
Table 4.1-1A	Note 6	SR	3.3.1.3	
Table 4.1-1A	Note 7	SR	3.3.1.3	
Table 4.1-1A	Note 7	SR	3.3.1.11	
Table 4.1-1A	Note 8	SR	3.3.1.6	
Table 4.1-1A	Note 9	SR	3.3.1.4	
Table 4.1-1A	Note 9	SR	3.3.1.5	
Table 4.1-1A	Note 10	SR	3.3.1.8	
Table 4.1-1A	Note 10	(Partial)	Relocated - Bases	
Table 4.1-1A	Note 11	SR	3.3.1.9	
Table 4.1-1A	Note 11	SR	3.3.1.15	
Table 4.1-1A	Note 12	TABLE	3.3.1-1	18
Table 4.1-1A	Note 13		Relocated - Bases	
Table 4.1-1A	Note 14		Relocated - Bases	
Table 4.1-1A	Note 15	TABLE	3.3.1-1	17
Table 4.1-1A	Note 16	TABLE	3.3.1-1	Note i
Table 4.1-1A	New Note	SR	3.3.1.4	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1A	Note 17	SR	3.3.1.8	
Table 4.1-1A	Note 18		Relocated - TRM	
Table 4.1-1A	New Note	SR	3.3.1.16	
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note c
Table 4.1-1A	New Note	SR	3.3.1.16	
Table 4.1-1A	New Note	SR	3.3.1.10	
Table 4.1-1A	New Note	SR	3.3.1.11	
Table 4.1-1A	New Note	SR	3.3.1.12	
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note e
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note f
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note g
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note h
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note j
Table 4.1-1B	1a	TABLE	3.3.2-1	1a
Table 4.1-1B	1b	TABLE	3.3.2-1	1c
Table 4.1-1B	1c	TABLE	3.3.2-1	1e
Table 4.1-1B	1d	TABLE	3.3.2-1	1d
Table 4.1-1B	1e	TABLE	3.3.2-1	1b

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1B	2a	TABLE	3.3.2-1	2a
Table 4.1-1B	2b	TABLE	3.3.2-1	2c
Table 4.1-1B	2c	TABLE	3.3.2-1	2b
Table 4.1-1B	3a	TABLE	3.3.2-1	3c
Table 4.1-1B	3b	TABLE	3.3.2-1	3a
Table 4.1-1B	3c	TABLE	3.3.2-1	3b
Table 4.1-1B	4a	TABLE	3.3.5-1	5
Table 4.1-1B	4b	TABLE	3.3.5-1	1
Table 4.1-1B	4b	SR	3.3.5.4	
Table 4.1-1B	4c	TABLE	3.3.5-1	6
Table 4.1-1B	4d	TABLE	3.3.5-1	4
Table 4.1-1B	4e	TABLE	3.3.5-1	3
Table 4.1-1B	4e	SR	3.3.5.1	
Table 4.1-1B	4e	SR	3.3.5.3	
Table 4.1-1B	4e	SR	3.3.5.5	
Table 4.1-1B	4f	TABLE	3.3.5-1	2
Table 4.1-1B	4f	SR	3.3.5.2	
Table 4.1-1B	5a	SR	3.7.2.1	
Table 4.1-1B	5a	(partial)	Relocated - IST	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1B	5b	TABLE	3.3.2-1	4b
Table 4.1-1B	5c	TABLE	3.3.2-1	4d
Table 4.1-1B	5d	TABLE	3.3.2-1	4c
Table 4.1-1B	5e	TABLE	3.3.2-1	4a
Table 4.1-1B	6a	TABLE	3.3.2-1	5b
Table 4.1-1B	6b	TABLE	3.3.2-1	5c
Table 4.1-1B	6c		Relocated - TRM	
Table 4.1-1B	6d	TABLE	3.3.2-1	5a
Table 4.1-1B	7a		Relocated - TRM	
Table 4.1-1B	7b	TABLE	3.3.2-1	6b
Table 4.1-1B	7c	TABLE	3.3.2-1	6d
Table 4.1-1B	7c	TABLE	3.3.2-1	Note f
Table 4.1-1B	7d	TABLE	3.3.2-1	6e
Table 4.1-1B	7e	TABLE	3.3.2-1	6c
Table 4.1-1B	7f	TABLE	3.3.2-1	6a
Table 4.1-1B	8	SR	3.3.4.2	
Table 4.1-1B	8	SR	3.3.4.1	
Table 4.1-1B	Note 20	SR	3.3.2.5	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1B	Note 21	TABLE	3.3.2-1	Note a
Table 4.1-1B	Note 22	SR	3.3.2.2	
Table 4.1-1B	Note 23	TABLE	3.3.2-1	Note c
Table 4.1-1B	Note 23	LCO	3.7.2	
Table 4.1-1B	Note 24	TABLE	3.3.5-1	Note d
Table 4.1-1B	Note 25		Deleted	
Table 4.1-1B	Note 26	LCO	3.3.5-1	
Table 4.1-1B	New Note	TABLE	3.3.2-1	Note e
Table 4.1-1B	7d	TABLE	3.3.2-1	Note g
Table 4.1-1C	1		Relocated - TRM	
Table 4.1-1C	2	SR	3.1.4.1	
Table 4.1-1C	2	SR	3.1.7.1	
Table 4.1-1C	2	(Partial)	Relocated - TRM	
Table 4.1-1C	2	(Partial)	Deleted	
Table 4.1-1C	3		Relocated - TRM	
Table 4.1-1C	4		Relocated - TRM	
Table 4.1-1C	5		Deleted - Boric Acid LAR	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1C	6		Relocated - TRM	
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	
Table 4.1-1C	10	SR	3.6.8.2	
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	
Table 4.1-1C	16		Relocated - TRM	
Table 4.1-1C	17		Relocated - TRM	
Table 4.1-1C	18	SR	3.3.1.12	
Table 4.1-1C	19		Relocated - TRM	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1C	20		Relocated - TRM	
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	
Table 4.1-1C	22		CTS Deleted	
Table 4.1-1C	23		CTS Deleted	
Table 4.1-1C	24		Relocated - TRM	
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	
Table 4.1-1C	25	SR	3.4.12.5	
Table 4.1-1C	25	SR	3.4.13.5	
Table 4.1-1C	25	SR	3.4.13.6	
Table 4.1-1C	26		Relocated - TRM	
Table 4.1-1C	27		Relocated - TRM	
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
Table 4.1-1C	29	SR	3.3.3.2	
Table 4.1-1C	29	(Partial)	Relocated - TRM	
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	
Table 4.1-1C	Note 31		Deleted	
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	
Table 4.1-1C	Note 38	SR	3.4.12.4	
Table 4.1-1C	Note 38	SR	3.4.13.5	
Table 4.1-1C	Note 39	SR	3.6.8.2	
Table 4.1-1C	Note 39	SR	3.6.8.1	
Table 4.1-1C	New Note	SR	3.3.3.3	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-2A	1	SR	3.1.4.3	
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	
Table 4.1-2A	4	SR	3.7.1.1	
Table 4.1-2A	5	SR	3.9.2.1	
Table 4.1-2A	6	SR	3.4.11.1	
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	
Table 4.1-2B	1	SR	3.4.17.1	
Table 4.1-2B	2	SR	3.4.17.2	
Table 4.1-2B	3	SR	3.4.17.3	
Table 4.1-2B	4a	LCO	3.4.17	
Table 4.1-2B	4b	SR	3.4.17.2	
Table 4.1-2B	5		Relocated - TRM	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-2B	6		Relocated - TRM	
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	
Table 4.1-2B	10		Deleted by Boric Acid LAR	
Table 4.1-2B	11	SR	3.6.6.3	
Table 4.1-2B	12	SR	3.5.1.4	
Table 4.1-2B	13	SR	3.7.16.1	
Table 4.1-2B	14		Relocated - TRM	
Table 4.1-2B	15	SR	3.7.14.1	
Table 4.1-2B	16		Relocated - TRM	
Table 4.1-2B	Note 1	SR	3.4.17.3	
Table 4.1-2B	Note 2		Relocated - TRM	
Table 4.1-2B	Note 3	SR	3.9.1.1	
Table 4.1-2B	Note 4		Relocated - TRM	
Table 4.1-2B	Note 5		Deleted	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-2B	Note 6		Relocated - TRM	
Table 4.2-1	1	G	5.5.6	
Table 4.12-1		G	5.5.8	
Table 4.12-2		G	5.5.8	
Table 4.13-1			Relocated - TRM	

PACAKGE 3.6

CONTAINMENT SYSTEMS

CROSS - REFERENCE

IMPROVED TECHNICAL SPECIFICATIONS

TO

CURRENT TECHNICAL SPECIFICATIONS

Section Cross - Reference

Section 3.6

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 3.6				
3.6.1		LCO	1.0	
3.6.1		LCO	3.6.A.1	
3.6.1		LCO	3.6.A.2	
3.6.1.1		SR	4.4.A.1	
3.6.1.1		SR	4.4.C	
3.6.1.2		SR	3.6.J	
3.6.1.2		SR	4.4.G	
3.6.1.3		SR	3.6.K	
3.6.1.3		SR	4.4.H	
3.6.2		LCO	1.0	
3.6.2		LCO	3.6.M	
3.6.2		LCO	New	
3.6.2.1		SR	4.4.A.2	
3.6.2.2		SR	New	
3.6.3		LCO	1.0	
3.6.3		LCO	3.6.C.1	
3.6.3		LCO	3.6.D.1	

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item Number	Section Type	CTS Section	CTS Table Item Number
3.6.3		LCO	3.6.D.2	
3.6.3		LCO	New	
3.6.3.1		SR	New	
3.6.3.2		SR	3.6.D.2.e	
3.6.3.3		SR	New	
3.6.3.4		SR	New	
3.6.3.5		SR	New	
3.6.3.6		SR	3.6.D.2.b	
3.6.3.7		SR	4.4.E	
3.6.3.8		SR	4.4.A.3	
3.6.4		LCO	3.6.I.1	
3.6.4		LCO	3.6.I.2	
3.6.4.1		SR	New	
3.6.5		LCO	3.3.B.1.a	
3.6.5		LCO	3.3.B.1.b	
3.6.5		LCO	3.3.B.2.a	
3.6.5		LCO	New	
3.6.5		LCO	3.3.B.2.b	
3.6.5.1		SR	New	

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item Number	Section Type	CTS Section	CTS Table Item Number
3.6.5.2		SR	4.5.B.2	
3.6.5.3		SR	4.5.A.3	
3.6.5.4		SR	4.5.B.1.a	
3.6.5.5		SR	4.5.B.3.f	
3.6.5.6		SR	4.5.A.2.a	
3.6.5.7		SR	4.4.F	
3.6.5.8		SR	4.5.A.2.b	
3.6.6		LCO	3.3.B.1.c	
3.6.6		LCO	3.3.B.2.c	
3.6.6		LCO	New	
3.6.6.1		SR	New	
3.6.6.2		SR	New	
3.6.6.3		SR	Table 4.1-2B	11
3.6.6.4		SR	4.5.B.3.f	
3.6.7		LCO	3.6.L	
3.6.7.1		SR	4.4.I.a	
3.6.7.2		SR	4.4.I.b	
3.6.7.3		SR	4.4.I.c	
3.6.8		LCO	3.6.B.3	

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item Number	Section Type	CTS Section	CTS Table Item Number
3.6.8		LCO	3.6.B.2	
3.6.8.1		SR	Table 3.5-1	7
3.6.8.1		SR	Table 4.1-1C	10
3.6.8.1		SR	Table 4.1-1C	Note 39
3.6.8.1		SR	4.4.C	
3.6.8.2		SR	Table 4.1-1C	10
3.6.8.2		SR	Table 4.1-1C	Note 39
3.6.9		LCO	3.6.H	
3.6.9.1		SR	4.4.B.4.d	
3.6.9.2		SR	4.4.B.3	
3.6.9.2		SR	4.4.B.5	
3.6.9.3		SR	4.4.B.3.c	
3.6.9.4		SR	4.4.E	
3.6.9.5		SR	4.4.B.1	
3.6.10		LCO	1.0	
3.6.10		LCO	3.6.G	
3.6.10.1		SR	New	
3.6.8.1		SR	Table 3.5-1	7

ITS PACKAGE CONTENTS

Package:

3.7

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
7. Part G NSHD for changes to PI CTS
8. Cross-Reference CTS to ITS
9. Cross-Reference ITS to CTS

PACKAGE 3.7

PLANT SYSTEMS

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

3.7
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 3.7

PLANT SYSTEMS

PART B

PROPOSED PRAIRIE ISLAND IMPROVED TECHNICAL SPECIFICATIONS AND BASES

List of Pages

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Five MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSSV inoperable.	A.1 Restore inoperable MSSV to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	----------------------------NOTE---------------------------- Only required to be performed in MODES 1 and 2. --- Verify each MSSV lift setpoint per Table 3.7.1-1 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ±1%.	In accordance with the Inservice Testing Program

Table 3.7.1-1
Main Steam Safety Valve Lift Settings

VALVE NUMBER				
Unit 1 Steam Generator: #11	Unit 1 Steam Generator: #12	Unit 2 Steam Generator: #21	Unit 2 Steam Generator: #22	LIFT SETTING (psig ± 3%)
RS-21-1	RS-21-6	RS-21-11	RS-21-16	1077
RS-21-2	RS-21-7	RS-21-12	RS-21-17	1093
RS-21-3	RS-21-8	RS-21-13	RS-21-18	1110
RS-21-4	RS-21-9	RS-21-14	RS-21-19	1120
RS-21-5	RS-21-10	RS-21-15	RS-21-20	1131

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when both MSIVs are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
C. One MSIV inoperable in MODE 2 or 3.	C.1 Close MSIV.	8 hours
	<u>AND</u>	
	C.2 Verify MSIV is closed.	Once per 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	---------------------------NOTE--------------------------- Only required to be performed in MODES 1 and 2. -- Verify the isolation time of each MSIV is ≤ 5.0 seconds.	In accordance with the Inservice Testing Program
SR 3.7.2.2	---------------------------NOTE--------------------------- Only required to be performed in MODES 1 and 2. -- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Regulation Valves (MFRVs) and MFRV Bypass Valves

LCO 3.7.3 Two MFRVs and two MFRV bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---

1. Separate Condition entry is allowed for each valve.
2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both MFRVs inoperable.	A.1 Close and place in manual or isolate flow through MFRV(s).	72 hours
	<u>AND</u>	
	A.2 Verify MFRV(s) closed and in manual or flow through MFRV(s) isolated.	Once per 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or both MFRV bypass valves inoperable.	B.1 Close and place in manual or isolate flow through bypass valve(s).	72 hours
	<u>AND</u>	
	B.2 Verify bypass valve(s) closed and in manual or flow through valve(s) isolated.	Once per 7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	C.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the isolation time of each MFRV and MFRV bypass valve is within limits.	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify each MFRV and MFRV bypass valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.4 Steam Generator (SG) Power Operated Relief Valves (PORVs)

LCO 3.7.4 Two SG PORVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SG PORV inoperable.	A.1 -------------NOTE------------ LCO 3.0.4 is not applicable. -------------------------------- Restore SG PORV to OPERABLE status.	7 days
B. Two SG PORVs inoperable.	B.1 Restore one SG PORV to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	C.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each SG PORV.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Two AFW trains shall be OPERABLE.

--NOTE--
AFW trains may be considered OPERABLE during alignment and operation for steam generator level control if it is capable of being manually realigned to the AFW mode of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable. <u>OR</u> ------------NOTE----------- Only applicable if MODE 2 has not been entered following refueling. ------------------------------- One turbine driven AFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One AFW train inoperable for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours
C. Required Action and associated Completion Time for Condition A or B not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 4.	12 hours
D. Two AFW trains inoperable.	D.1 -------------NOTE------------ LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ---------------------------------	
	Initiate action to restore one AFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	-----------------------------NOTE----------------------------- AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control if it is capable of being manually realigned to the AFW mode of operation. -- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	-----------------------------NOTE----------------------------- Not required to be performed for the turbine driven AFW pump until prior to exceeding 10% RTP or within 72 hours after RCS temperature > 350°F. -- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.3	----------------------------NOTE--------------------------- AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. --- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.5.4	----------------------------NOTE--------------------------- AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. --- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tanks (CSTs)

LCO 3.7.6 The CSTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
Conditions and Required Actions applicable to both units.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CSTs inoperable.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours AND Once per 12 hours thereafter
	AND	
	A.2 Restore CSTs to OPERABLE status.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 Verify CSTs contents ≥ 100,000 gal per operating unit.	12 hours

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CC) System

LCO 3.7.7 Two CC trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CC train inoperable.	A.1 ------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CC. ------------------------------- Restore CC train to OPERABLE status.	 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-----------------------------NOTE----------------------------- Isolation of CC flow to individual components does not render the CC System inoperable. --- Verify each CC manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	-----------------------------NOTE----------------------------- This SR only applies to those valves required to align CC System to support the safety injection or recirculation phase of emergency core cooling. --- Verify each CC automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.7.3	Verify each CC pump starts automatically on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.8 Cooling Water (CL) System

LCO 3.7.8 Two CL trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No safeguards CL pumps OPERABLE for one train.	A.1 -------------NOTES------------ 1. Unit 1 enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-MODES 1, 2, 3, and 4," for emergency diesel generator made inoperable by CL System. 2. Both units enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4," for residual heat removal loops made inoperable by CL System.	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	3. This Condition may not exist > 7 days in any consecutive 30 day period. ----------------------------------	
	A.1 Restore one safeguards CL pump to OPERABLE status.	7 days
B. One CL supply header inoperable.	----------------NOTES---------------- 1. Unit 1 enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-MODES 1, 2, 3, and 4," for emergency diesel generator made inoperable by CL System. 2. Both units enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4," for residual heat removal loops made inoperable by CL System. ---	
	B.1 Verify vertical motor driven CL pump OPERABLE.	4 hours
	AND	
	B.2 Restore CL supply header to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met.	----------------NOTE---------------- Applicable to both units. --	
	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 5.	36 hours
D. Diesel driven CL pumps stored fuel oil supply < 19,500 gal and > 17,000 gal.	D.1 Restore fuel oil supply to within limits.	48 hours
E. Diesel driven CL pumps stored fuel oil supply < 17,000 gal. OR Required Action and associated Completion Time of Condition D not met.	E.1 Declare diesel driven CL pumps inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	-----------------------------NOTE----------------------------- Isolation of CL flow to individual components does not render the CL System inoperable. -- Verify each CL System manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.8.2	Verify each diesel driven CL pump starts and assumes load within one minute.	31 days
SR 3.7.8.3	Verify stored diesel driven CL pumps fuel oil supply $\geq$ 19,500 gal.	31 days
SR 3.7.8.4	Verify OPERABILITY of vertical motor driven CL pump.	92 days
SR 3.7.8.5	Verify each CL System automatic valve required to mitigate accidents that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.8.6	Verify the diesel driven and vertical motor driven CL pumps start automatically on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.9 Emergency Cooling Water (CL) Supply

LCO 3.7.9 The Emergency CL supply shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---
Conditions and Required Actions applicable to both units.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One safeguards traveling screen inoperable.	A.1 ------------NOTE------------ Not applicable during periods of testing for < 24 hours. ---------------------------------	
	Verify one emergency bay sluice gate open.	4 hours
	AND	
	A.2 Restore safeguards traveling screen to OPERABLE status.	90 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Both safeguards traveling screens inoperable.	B.1 Verify one emergency bay sluice gate open.	1 hour
	AND	
	B.2 Restore one safeguards traveling screen to OPERABLE status.	7 days
C. Emergency CL Line inoperable.	C.1 Verify one emergency bay sluice gate open.	1 hour
	AND	
	C.2 Restore Emergency CL Line to OPERABLE status.	7 days
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify safeguards traveling screens OPERABLE.	92 days

3.7 PLANT SYSTEMS

3.7.10 Control Room Special Ventilation System (CRSVS)

LCO 3.7.10 Two CRSVS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----
Conditions and Required Actions applicable to both units.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRSVS train inoperable.	A.1 Restore CRSVS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	C.1	Place OPERABLE CRSVS train in emergency mode.	Immediately
	<u>OR</u>		
	C.2	Suspend movement of irradiated fuel assemblies.	Immediately
D. Two CRSVS trains inoperable during movement of irradiated fuel assemblies.	D.1	Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CRSVS trains inoperable in MODE 1, 2, 3, or 4.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRSVS train ≥ 15 minutes.	31 days
SR 3.7.10.2	Perform required CRSVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each CRSVS train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.10.4	Verify the CRSVS fan in each train delivers 3600 to 4400 cfm.	24 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.11 Safeguards Chilled Water System (SCWS)

LCO 3.7.11 Two SCWS loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

------------------------------NOTE------------------------------
Conditions and Required Actions applicable to both units.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SCWS loop inoperable.	A.1 Restore SCWS loop to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	C.1 Place OPERABLE SCWS loop in operation. <u>OR</u> C.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
D. Two SCWS loops inoperable during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately
E. Two SCWS loops inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify each SCWS loop actuates on an actual or simulated actuation signal.	24 months on a STAGGERED TEST BASIS
SR 3.7.11.2	Verify SCWS components OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program.

3.7 PLANT SYSTEMS

3.7.12 Auxiliary Building Special Ventilation System (ABSVS)

LCO 3.7.12 Two ABSVS trains shall be OPERABLE.

-----NOTE-----
The ABSV boundary may be opened under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTE-----
Conditions and Required Actions applicable to both units.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ABSVS train inoperable.	A.1 Restore ABSVS train to OPERABLE status.	7 days
B. Two ABSVS trains inoperable due to inoperable ABSVS boundary.	B.1 Restore ABSVS boundary to OPERABLE status.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each ABSVS train for ≥ 10 hours with the heaters operating.	31 days
SR 3.7.12.2	Perform required ABSVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.3	Verify each ABSVS train can produce a negative pressure within 6 minutes after initiation.	92 days
SR 3.7.12.4	Verify each ABSVS train actuates on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool Special Ventilation System (SFPSVS)

LCO 3.7.13 Two SFPSVS trains shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pool enclosure.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SFPSVS train inoperable.	A.1 Restore SFPSVS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Place OPERABLE SFPSVS train in operation. <u>OR</u> B.2 Suspend movement of irradiated fuel assemblies in the spent fuel pool enclosure.	Immediately Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Both SFPSVS trains inoperable.	C.1 Suspend movement of irradiated fuel assemblies in the spent fuel pool enclosure.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each SFPSVS train for ≥ 10 hours with the heaters operating.	31 days
SR 3.7.13.2	Perform required SFPSVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	Verify each SFPSVS train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.13.4	Verify the SFPSVS fan in each train delivers 4680 to 5720 cfm.	24 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.14 Secondary Specific Activity

LCO 3.7.14 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3.	6 hours
	AND	
	A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.	31 days

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Storage Pool Water Level

LCO 3.7.15 The spent fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1 ------------NOTE------------ LCO 3.0.3 is not applicable. -------------------------------- Suspend movement of irradiated fuel assemblies in the spent fuel storage pool.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel storage pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Storage Pool Boron Concentration

LCO 3.7.16 The spent fuel storage pool boron concentration shall be ≥ 1800 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool boron concentration not within limit.	-----------------NOTE---------------- LCO 3.0.3 is not applicable. ---	
	A.1 Suspend movement of fuel assemblies in the spent fuel storage pool.	Immediately
	<u>AND</u>	
	A.2 Initiate action to restore spent fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the spent fuel storage pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Pool Storage

LCO 3.7.17 The combination of initial enrichment, burnup and decay time of each fuel assembly stored in the spent fuel pool shall be within the Unrestricted Region of Figure 3.7.17-1 or Figure 3.7.17-2, as applicable, or in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 -------------NOTE------------- LCO 3.0.3 is not applicable. --------------------------------- Initiate action to move the noncomplying fuel assembly to an acceptable location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means the initial enrichment, burnup and decay time of the fuel assembly is in accordance with Figure 3.7.17-1 or Figure 3.7.17-2, as applicable, or Specification 4.3.1.1.	Prior to storing or moving the fuel assembly
SR 3.7.17.2	Verify spent fuel pool inventory.	Within 7 days after completion of a spent fuel pool fuel handling campaign

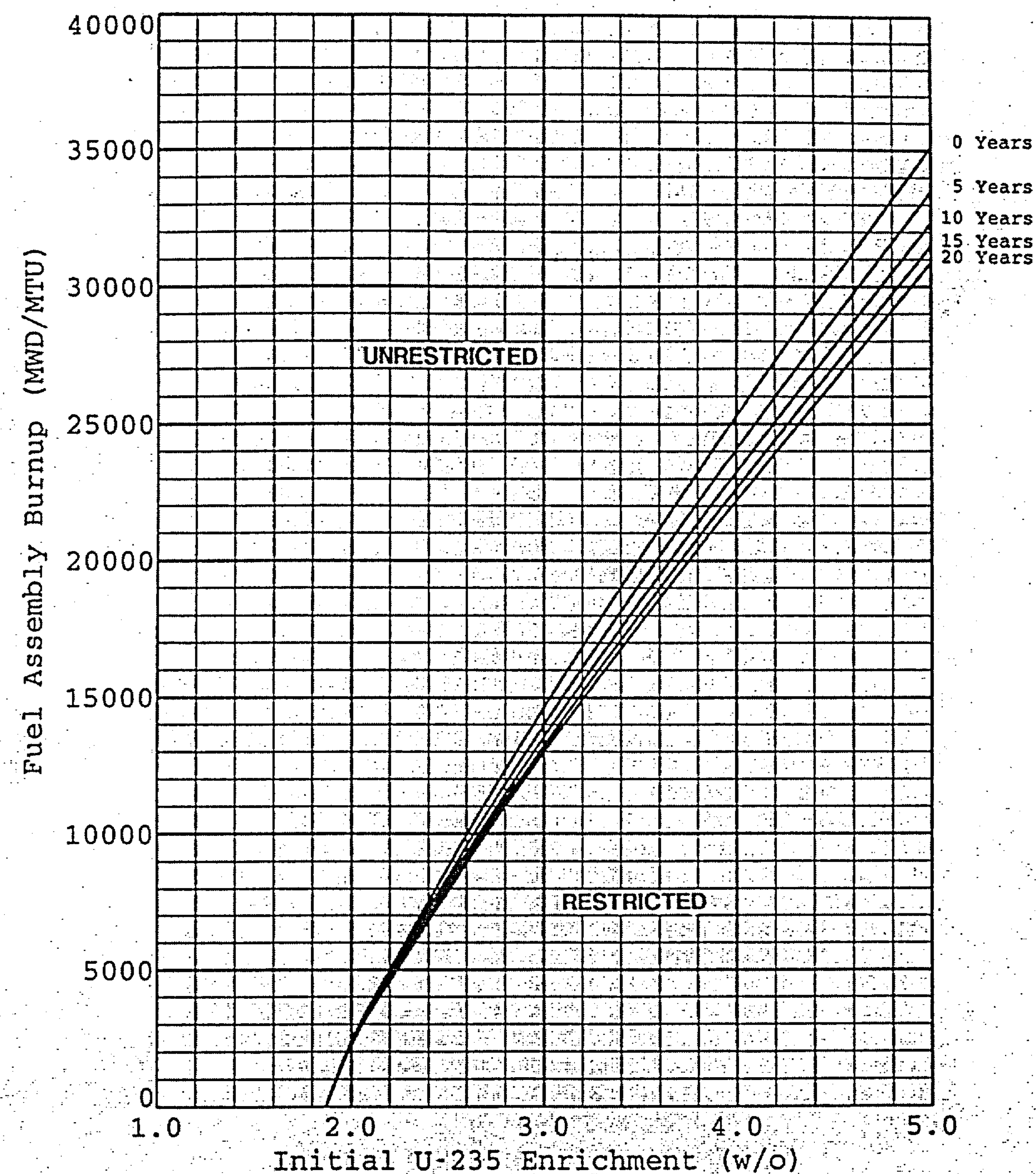

Figure 3.7.17-1
Spent Fuel Pool Unrestricted Region Burnup and Decay Time Requirements-OFA Fuel

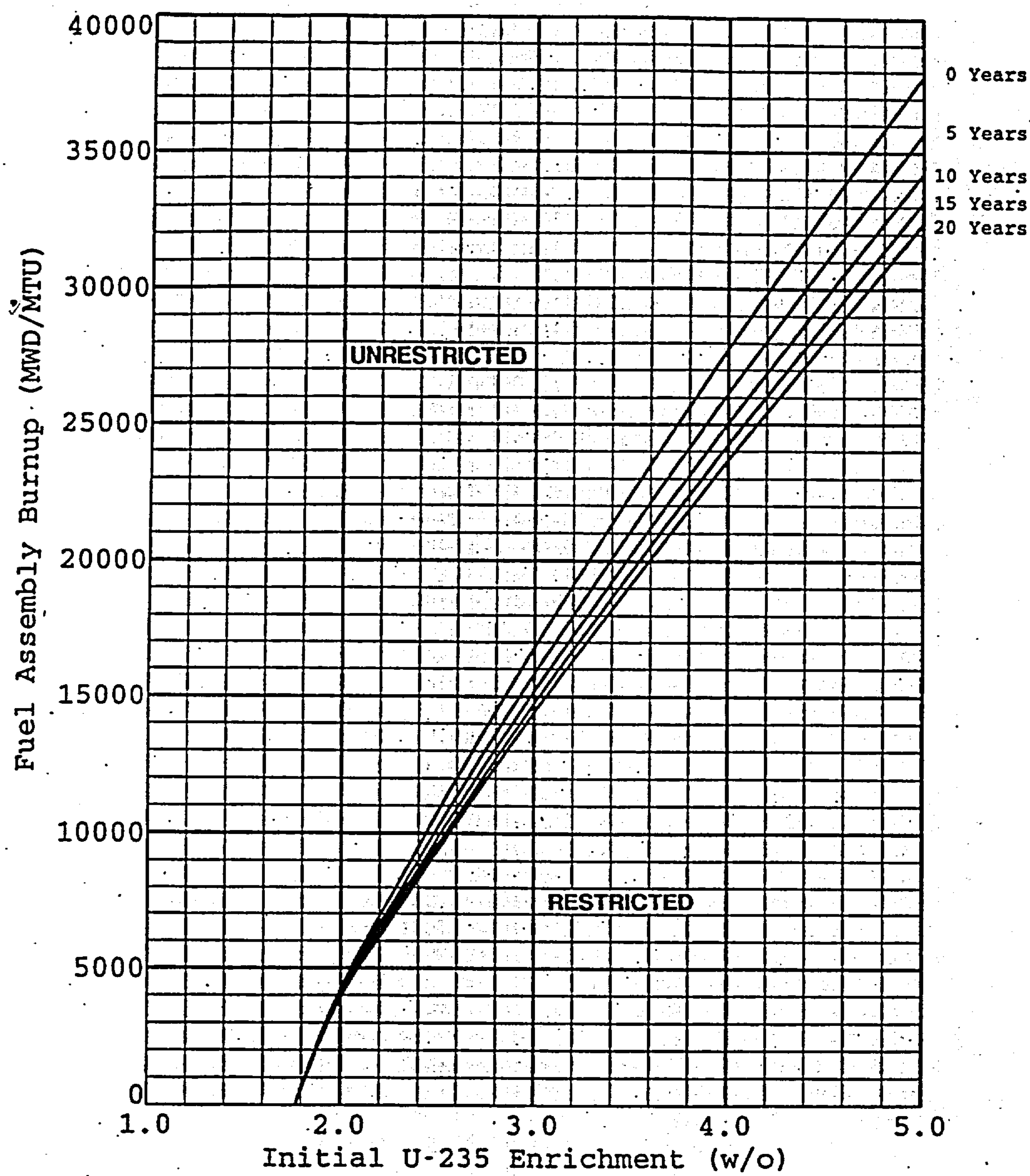

Figure 3.7.17-2
Spent Fuel Pool Unrestricted Region Burnup and Decay Time Requirements-STD Fuel

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the USAR (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-1 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

Normal functioning of a MSSV is expected to involve some "simmering" which does not make the valve inoperable.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis. The accident analysis requires five MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% RTP.

BASES

APPLICABLE SAFETY ANALYSES (continued)

By relieving steam the MSSVs prevent RCS overpressurization. The limiting events, described in the USAR, (Ref. 3), that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, such as the full power turbine trip without steam dump, and increasing core heat flux events, such as the rod cluster control assembly (RCCA) withdrawal at power.

The safety analyses demonstrate that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. If a minimum (least negative or most positive) reactivity feedback is assumed, the reactor is tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure.

The transient response for the slow and fast RCCA withdrawal at power events also present no hazard to the integrity of the RCS or the Main Steam System. Diverse reactor trip inputs from nuclear instrumentation and pressurizer level and pressure are assumed to shut down the reactor when the associated trip setpoint is reached. In this analysis, the pressurizer safety valves open and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2. The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, relieve steam generator overpressure, and close when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

BASES

LCO (continued)

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4, 5, and 6, there are no credible transients requiring the MSSVs.

The energy content in the steam generators is sufficiently low in MODES 5 and 6 that they cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

A.1

With one MSSV inoperable, restore OPERABILITY of the inoperable MSSV within 4 hours.

Continued operation with less than all five MSSVs OPERABLE for each steam generator is not permitted since safety analyses supporting such operation have not been performed.

B.1 and B.2

If the MSSV cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

BASES

ACTIONS

B.1 and B.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-1 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to within a nominal ± 1% of their setpoint during the Surveillance. The lift settings, according to Table 3.7.1-1, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. USAR, Section 11.4.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. USAR, Section 14.4.
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. ANSI/ASME OM-1-1987.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a main steam line break (MSLB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Steam Dump System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by any of the following signals:

a. High-High Containment Pressure;

b. High Steam Flow and Low-Low T_{avg} with Safety Injection; and

c. High-High Steam Flow with Safety Injection.

The MSIVs fail closed on loss of air.

Each MSIV has a MSIV bypass valve. These valves are normally used for warming steam lines and equalizing pressure across the MSIVs. These bypass valves are normally closed at power.

The MSIVs and MSIV bypass valves may be operated manually.

BASES

BACKGROUND (continued)	In addition to the fast-closing stop valve, each steam line has a downstream non-return check valve (NRCV). The four valves (one MSIV and one NRCV in each of two lines) prevent blowdown of more than one steam generator for any break location even if one valve fails to close. A description of the MSIVs and NRCVs is found in the USAR (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the USAR (Ref. 2). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV or NRCV to close).

The limiting case for the containment analysis is the main steam line break (MSLB) inside containment, with offsite power available following turbine trip, and failure of a safeguards train. At lower power, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

The analysis of several different SLB events are performed to demonstrate that the acceptance criteria listed in the USAR are satisfied.

Events evaluated include:

a. Containment response due to a large SLB inside of containment;

b. Core response due to a large SLB inside of containment;

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Small SLB; and

d. Core response due to a SLB outside of containment to support the voltage-based steam generator tube repair criteria (Ref. 3).

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. A MSLB inside containment. For this accident scenario, steam is discharged into containment from both steam generators until the NRCV on the broken line (or MSIV on the intact line) closes. After the valve closes, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header between the closed valve and the affected steam generator. Closure of the NRCV in the affected line (or the MSIV in the intact line) isolates the break from the unaffected steam generator.

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the NRCV in the affected line (or the MSIV in the intact line) isolates the break and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIV downstream of the ruptured steam generator isolates the ruptured steam generator from the intact steam generator to minimize radiological releases.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

This LCO requires that both MSIVs be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on a main steam isolation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, normally the MSIVs are closed, and the steam generator energy is low. In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs and considering the redundancy of the NRCV.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional passive means for containment isolation.

BASES

ACTIONS (continued)

B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered unless both MSIVs are closed. The Completion Times are reasonable, based on operating experience, to reach MODE 2 in an orderly manner without challenging unit systems.

C.1 and C.2

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIV may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is consistent with that allowed in Condition A for one MSIV inoperable.

For an inoperable MSIV that cannot be restored to OPERABLE status within the specified Completion Time, but is closed, the inoperable MSIV must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve

BASES

ACTIONS

D.1 and D.2 (continued)

this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5 seconds. The MSIV isolation time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of valve closure when the unit is generating power. As the MSIVs are not tested at power, they are deferred from the ASME Code (Ref. 5) requirements during operation in MODE 1 or 2. Since this test is performed by manually closing the valve, this test also verifies that the MSIV manual switches are functional.

The Frequency is in accordance with the Inservice Testing Program.

This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies each MSIV can close on an actual or simulated main steam isolation signal. This Surveillance is normally

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.2 (continued)

performed upon returning the plant to operation following a refueling outage.

The Frequency of MSIV testing is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 11.7.

2. USAR, Section 14.5.

3. License Amendment 133/125, issued November 18, 1997, "Voltage-based Steam Generator Tube Repair Criteria."

4. 10 CFR 100.11.

5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulation Valves (MFRVs) and MFRV Bypass Valves

BASES

BACKGROUND

The MFRVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a steam or feedwater line break. The key safety-significant functions of the MFRVs are to prevent:

a. Overfill of the steam generators;

b. Excessive cooldown of the Reactor Coolant System; and

c. Overpressurization of the containment following a main feedwater line break (FWLB) or main steam line break (MSLB).

Closure of the MFRVs and associated bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring downstream of the main feedwater isolation valves (MFIVs) or MFRVs. Credit is taken for only the MFRVs and the MFRV bypass valves since the closure times are shorter than those of the MFIVs. The MFIVs are treated solely as containment isolation valves in accordance with LCO 3.6.3, "Containment Isolation Valves". Check valves in the feedwater lines terminate FWLBs upstream of the MFIVs and MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFRVs will be mitigated by their closure. Closure of the MFRVs and MFRV bypass valves effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

The MFRVs, MFRV bypass valves, and piping upstream of the MFIVs are nonsafety related. In the event of a secondary side pipe rupture inside containment, the MFRVs, MFRV bypass valves,

BASES

BACKGROUND (continued)

check valves and the main feedwater pump trip limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

One MFRV and its MFRV bypass valve are located on each MFW line, outside but close to containment. The check valves are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MFIV or MFRV closure. The piping volume from the valves to the steam generators must be accounted for in calculating mass and energy releases. This line must be refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

The MFRVs and MFRV bypass valves close due to the following automatic feedwater isolation (FWI) signals:

a. Low T_{avg} coincident with reactor trip (P-4) (MFRV only);

b. Steam generator water level high-high signal; and

c. Safety injection.

The MFRVs and MFRV bypass valves may also be operated manually.

In addition to the MFIVs, the MFRVs and MFRV bypass valves, a check valve inside containment is available. The check valve isolates the feedwater line, penetrating containment, and ensures that the consequences of events do not exceed the capacity of the containment heat removal systems.

A description of the MFRVs is found in the USAR (Ref. 1).

BASES (continued)

APPLICABLE SAFETY ANALYSES	The design basis of the MFRVs is established by the analyses for the large Main Steam Line Break (MSLB). Closure of the MFRVs and associated bypass valves, and trip of the main feedwater pumps may be relied on to terminate feedwater flow during a MSLB for the core and containment response analyses. The MSLB core and containment response analyses bound the accident analysis for the large FWLB. Some leakage through the MFRVs and associated bypass valves is anticipated when control board instrumentation indicates that the valves have closed. This leakage has been conservatively bounded by the MSLB analyses. Failure of a MFRV or the MFRV bypass valves to close following a MSLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following a MSLB or FWLB event. The MFRVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO ensures that the MFRVs and the MFRV bypass valves will isolate MFW flow to the steam generators, following a FWLB or MSLB. This LCO requires that two MFRVs and associated MFRV bypass valves be OPERABLE. The MFRVs and the associated MFRV bypass valves are considered OPERABLE when feedwater isolation times are within limits and they close on a FWI signal. When control board instrumentation indicates that these valves have fully closed, the valves are OPERABLE since leakage through the closed valves has been conservatively bounded by the MSLB analyses, and therefore, they are performing their safety function. Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or

BASES

LCO (continued)	FWLB inside containment. Since a feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.
APPLICABILITY	The MFRVs and the MFRV bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. In MODES 1, 2, and 3, the MFRVs and the MFRV bypass valves are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed, they are already performing their safety function. In MODES 4, 5, and 6, steam generator energy is low. In addition the MFRVs and the MFRV bypass valves are normally closed since MFW is not required.
ACTIONS	The ACTIONS table is modified by two Notes. Note 1 specifies separate Condition entry is allowed for each valve. Note 2 specifies LCO 3.0.4 does not apply. <u>A.1 and A.2</u> With one MFRV in one or both flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close and place in manual or to isolate flow through inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function. Similarly, if the feedwater flow path to containment is isolated using, as an example, the MFIV, the required safety function is being met.

BASES

ACTIONS

A.1 and A.2 (continued)

The 72 hour Completion Time takes into account the redundancy afforded by the remaining valves in the feedwater line and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFRVs, that are closed and in manual or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that the valves are closed or isolated.

B.1 and B.2

With one MFRV bypass valve in one or both flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close and place in manual or to isolate flow through inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function. Similarly, if the feedwater flow path to containment is isolated using, as an example, the MFIV, the required safety function is being met.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining valves in the feedwater line and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

BASES

ACTIONS

B.1 and B.2 (continued)

Inoperable MFRV bypass valves that are closed and placed in manual or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

C.1 and C.2

If the MFRV(s) or the MFRV bypass valve(s) cannot be restored to OPERABLE status, closed, isolated, or the flow path through the valve isolated within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFRV and MFRV bypass valve is within limits set by the Inservice Testing Program. The MFRV isolation times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. This is consistent with the ASME Code (Ref. 2) periodic stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.2

This SR verifies that each MFRV and MFRV bypass valve can close on an actual or simulated FWI signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Frequency for this SR is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 11.9.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.4 Steam Generator (SG) Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The SG PORVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Dump System to the condenser be unavailable, as discussed in the USAR (Ref. 1). Cooldown is performed in conjunction with the Auxiliary Feedwater System providing makeup water to the steam generators.

One SG PORV is provided for each steam generator. Each SG PORV line consists of one SG PORV and an associated block valve.

The upstream manual block valves permit SG PORV testing at power and provide an alternate means of isolation. The SG PORVs are equipped with pneumatic controllers to permit control of the cooldown rate.

A description of the SG PORVs is found in References 1 and 2.

APPLICABLE SAFETY ANALYSES

Automatic operation of the SG PORVs is not credited in the safety analyses. Rather, the SG PORVs may provide mitigation for accidents involving use of main steam safety valves.

In the steam generator tube rupture (SGTR) accident analysis presented in Reference 2, the SG PORV in the unaffected steam generator is assumed to be used by the operator to cool down the unit for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a SGTR event, the operator is required to perform a limited cooldown to establish adequate subcooling as a necessary

BASES

APPLICABLE SAFETY ANALYSES (continued)	step prior to terminating the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for a SGTR is more critical than the time required to cool down for this event and also for other accidents.
	The SG PORVs are equipped with manual block valves in the event a SG PORV spuriously fails open or fails to close during use.
	The SG PORVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two SG PORVs are required to be OPERABLE to ensure that at least one SG PORV is available to conduct a unit cooldown following a SGTR.
	Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the Steam Dump System.
	A SG PORV is considered OPERABLE when it is capable of being remotely operated and when its associated block valve is open.
APPLICABILITY	In MODES 1, 2, and 3, the SG PORVs are required to be OPERABLE.
	In MODE 4, a steam generator and the SG PORV are not relied upon for heat removal.
	In MODE 5 or 6, a SGTR is not a credible event.
ACTIONS	<u>A.1</u>
	With one required SG PORV inoperable, action must be taken to restore OPERABLE status within 7 days.

BASES

ACTIONS

A.1 (continued)

The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE SG PORV, Steam Dump System, and MSSVs.

Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

B.1

With two SG PORVs inoperable, action must be taken to restore one SG PORV to OPERABLE status. Since the block valve can be closed to isolate a SG PORV, some repairs may be possible with the unit at power.

The 24 hour Completion Time is reasonable to repair an inoperable SG PORV, based on the availability of the Steam Dump System and MSSVs, and the low probability of an event occurring during this period that would require the SG PORV.

C.1 and C.2

If the SG PORV cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR ensures that the SG PORVs are tested through a full control cycle in accordance with the Inservice Testing Program. The SG PORV is isolated by the block valve for this test. Performance of inservice testing or use of a SG PORV during a unit cooldown may satisfy this requirement.

Operating experience has shown that these components usually pass the Surveillance when performed in accordance with the Inservice Testing Program. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 11.4.
2. USAR, Section 14.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply.

The AFW system is configured into two redundant trains. One train has a turbine driven AFW pump; the other has a motor driven AFW pump. Each AFW pump feeds the designated unit's two steam generators. In addition, each motor driven pump has the capability to be realigned locally to feed the other unit's steam generators.

The AFW pumps take suction from:

a. The nonsafety-related condensate storage tank (CST) supply header (LCO 3.7.6); or

b. The safety-related Cooling Water System (LCO 3.7.8).

The AFW pumps supply water to the steam generator secondary side via connections to the main feedwater (MFW) piping adjacent to the steam generators inside containment.

The steam generators function as a heat sink for core decay heat. The heat load may be dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1), steam generator power operated relief valves (SG PORVs) (LCO 3.7.4), or steam dump valve.

If the main condenser is available, steam may be released via the steam dump valve. Each unit's AFW System consists of:

BASES

BACKGROUND (continued)

a. One motor driven AFW pump;

b. One turbine driven AFW pump;

c. Steam generator AFW motor-operated supply valves; and

d. Steam generator AFW motor-operated throttle valves.

These components are configured to provide a flow path from each pump to both steam generators for the specific unit.

Each motor driven or turbine driven AFW pump can provide 100% of the required AFW flow capacity to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system.

The turbine driven AFW pump receives steam from both main steam lines upstream of the main steam isolation valves. Each steam feed line will supply 100% of the requirements of the turbine driven AFW pump. An air operated valve downstream of the motor operated valves from each loop allows passage of steam to the turbine driven AFW pump when required. The air supply to the valve is controlled by a normally open DC solenoid valve designed such that failure of either the air supply or control power would cause the respective valve to open, starting the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit operation in MODES 2 and 3. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions.

The AFW System is designed to supply sufficient water to the steam generators to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies

BASES

BACKGROUND (continued)

sufficient water to cool the unit to RHR entry conditions, with steam released through the SG PORVs or steam dump valve.

The following safety signals automatically initiate an AFW pump start signal:

a. Low-low water level in either steam generator; and

b. Safety injection.

Additionally, the following signals initiate an AFW pump start signal:

a. Trip of both main feedwater pumps (bypassed during startup and shutdown operation);

b. Loss of both 4 kV normal buses (turbine driven AFW pump only); and

c. Manually either local or remote.

Depending on pump type, the motor will start or the turbine steam admission air operated control valve will open.

The AFW System is discussed in the USAR (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event involving loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus margin for uncertainty and accumulation.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting plant condition which imposes safety-related performance requirements on the design of the AFW System is the loss of MFW as described in References 1 and 3.

The low-low steam generator level signal automatically actuates the motor and turbine driven AFW pumps and associated air operated valve and controls when required to ensure an adequate feedwater supply to the steam generators during loss of offsite power. Normally open motor operated valves are provided for each AFW line to allow throttling of the AFW flow from each AFW pump to each steam generator when required.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary.

Two independent AFW pumps in two diverse trains are required to be OPERABLE to ensure the availability of decay heat removal capability for all events accompanied by a loss of main feedwater and a single failure.

The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that one motor driven AFW pump be OPERABLE and capable of supplying AFW to both steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to both steam generators. The piping, valves, instrumentation, and controls in the required flow paths, required for the system to perform the safety related function, also are required to

BASES

LCO (continued)

be OPERABLE. The normal (Condensate Storage Tanks (CSTs)) and backup (Cooling Water System) water supplies to the AFW pumps must also be OPERABLE. OPERABILITY requirements for the CSTs are specified in LCO 3.7.6, "Condensate Storage Tanks (CSTs)."

The LCO is modified by a Note indicating that an AFW train may be considered OPERABLE during alignment and operation for steam generator level control if capable of being manually realigned to the AFW mode of operation.

During operation in MODES 2 and 3, the AFW pump discharge motor operated valves used for throttling may be less than full open. The Shutdown-Auto mode of control may be used during such operations. This control mode bypasses the AFW pump start due to both MFW pumps being tripped or shutdown.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to provide heat removal . In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODES 4, 5 or 6, the AFW System is not required to perform a safety function.

ACTIONS

A.1

If one of the two steam supplies to the turbine driven AFW train is inoperable, or if a turbine driven pump is inoperable while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

BASES

ACTIONS

A.1 (continued)

a. For the inoperability of a steam supply to the turbine driven AFW pump, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump;

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling outage, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation; and

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of the redundant OPERABLE motor driven AFW pump, and due to the low probability of an event requiring the use of the turbine driven AFW pump.

Condition A is modified by a Note which limits the applicability of the Condition when the unit has not entered MODE 2 following a refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

B.1

With one of the required AFW trains (pump or flow path) inoperable for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

BASES

ACTIONS (continued)

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

If both AFW trains are inoperable, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR verifies the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths thereby providing assurance that the proper flow paths

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1 (continued)

will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during MODES 2, 3 and 4 operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Differential pressure is a normal test of centrifugal pump

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.2 (continued)

performance required by Section XI of the ASME Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 2) satisfies this requirement. The Inservice Testing Program specifies the Frequency for testing each pump. This test is considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. This deferral is based on the inservice testing requirements not met; all other requirements for OPERABILITY must be satisfied.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated safety injection signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This test is considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.3 (continued)

the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during MODES 2, 3 and 4 operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

SR 3.7.5.4

This SR verifies that the AFW pumps will start when required by demonstrating that each AFW pump starts automatically on an actual or simulated AFW pump start signal. Since this test is performed during unit shutdown, the turbine driven AFW pump is not actually started, but the components necessary to assure it starts on an actual or simulated AFW pump start signal are demonstrated to be OPERABLE. This test is considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.4 (continued)

This SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during MODES 2, 3 and 4 operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

REFERENCES

1. USAR, Section 11.9.
2. ASME, Boiler and Pressure Vessel Code, Section XI.
3. USAR, Section 14.4.

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tanks (CSTs)

BASES

BACKGROUND

Three 150,000 gallon CSTs are shared via a common header between the 2 units. Unit 1 has 1 tank (11) and Unit 2 has 2 tanks (21 and 22). The CSTs provide a nonsafety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS).

A backup safety grade source of water is provided by the safety-related portion of the Cooling Water (CL) System (LCO 3.7.8) via either the Emergency Cooling Water Line or the emergency bay sluice gates.

Since water supplied from the CL System is of lower purity, its use is considered less desirable under normal conditions than the higher purity condensate water from the CSTs. However, if needed, the operator can lineup the Cooling Water supply by opening the associated CL supply motor operated valve (MOV) and closing the associated CST supply MOV for each auxiliary feedwater pump.

The CSTs provide a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves, the steam generator power operated relief valves or the atmospheric dump valve. Each AFW pump operates with a continuous recirculation to a CST.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam dump valve. The condensed steam may be returned to the CSTs by the condensate pump. This has the advantage of conserving condensate while minimizing releases to the environment.

BASES

BACKGROUND (continued)

Although the CSTs are a principal secondary side water source for removing residual heat from the RCS, they are not designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. However, the backup CL safety-related source is designed to withstand such phenomena.

A description of the CSTs is found in the USAR (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CSTs may provide high purity cooling water to remove decay heat and to cool down the unit following events in the accident analysis as discussed in the USAR (Ref. 2).

The 100,000 gallon CSTs volume requirement for each unit in MODE 1, 2, or 3 is sufficient to:

a. Remove the decay heat generated by one reactor in the first 24 hours after shutdown; and

b. Ensure sufficient water is available to cool down a reactor from 547°F to 350°F using natural circulation at 25°F/hour.

These calculations take into account the decay heat and reactor coolant system stored energy (Ref. 2).

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The CSTs are considered OPERABLE when the CSTs' contents have at least 100,000 gallons per operating unit (MODES 1, 2, or 3).

This basis is established in Reference 2 and exceeds the volume required by the accident analysis.

BASES

LCO (continued)	The OPERABILITY of the CSTs is determined by maintaining the tank level at or above the minimum required level.
APPLICABILITY	In MODES 1, 2, and 3 the CSTs are required to be OPERABLE. In MODES 4, 5, or 6, the CSTs are not required because the AFW System is not required.
ACTIONS	A NOTE preceding the ACTIONS Table specifies the Conditions and Required Actions are applicable to both units. The common CST header and the sharing of the backup safety-related portion of the CL supply between the two units provide the basis for this requirement.

A.1 and A.2

If the CSTs are not OPERABLE (e.g., level is not within limits), the OPERABILITY of the backup safety-related portion of the CL supply should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup safety-related portion of the CL supply must include verification that the flow paths from the backup water supply to the AFW pumps are OPERABLE in accordance with LCO 3.7.8. The CSTs must be restored to OPERABLE status within 7 days.

The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup safety-related portion of the Cooling Water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup safety-related portion of the CL supply being available, and the low probability of an event occurring during this time period requiring the CSTs.

BASES

ACTIONS (continued)

B.1 and B.2

If the CSTs cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply.

To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CSTs contain the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks.

Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.

REFERENCES

1. USAR, Section 11.9.

2. USAR, Section 14.4.

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CC) System

BASES

BACKGROUND

The CC System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CC System also provides this function for various nonessential components, as well as the spent fuel storage pool. The CC System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Cooling Water System, and thus to the environment.

Each unit's CC System is arranged as two parallel cooling loops, and has isolable nonsafety related components. Each safety related train includes a full capacity pump, supply from a common unit-specific surge tank, heat exchanger, piping, valves, and instrumentation.

The CC systems have the capability to be cross-connected between loops and between the two units at the CC pump suction and discharge. This design feature is not used during normal operation but does allow added flexibility of pump and heat exchanger combinations during abnormal conditions.

During operation the outlet CC temperature from the CC heat exchanger is normally maintained between 80°F and 105°F. During operation CC water is circulated through the shell side of the CC heat exchanger and then to the various system components at a maximum temperature of 120°F for 2 hours (Ref. 1).

Each safety related train is powered from a separate bus. A surge tank in the system is provided to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal, and all nonessential

BACKROUND (continued)

components are isolated. In addition, an automatic low pressure pump start can avert a reactor coolant pump seal failure during a loss of offsite power event (Ref. 1).

Additional information on the design and operation of the system, along with a list of the components served, is presented in the USAR (Ref. 1).

The principal safety related function of the CC System is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System during post accident containment sump recirculation.

APPLICABLE SAFETY ANALYSES

The design basis of the CC System is for one CC train to remove the post loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase.

During an accident, a design CC inlet temperature to the major heat exchangers of 95°F was assumed. The RHR heat exchanger during DBA long-term conditions is the primary heat load. The required post accident heat removal rate is in the same range as the required rate during MODES 1 or 2, but less than that needed during normal MODE 4 condition. In a normal MODE 4 cooldown from 350°F to 200°F, more equipment is expected to be operating than during a post accident condition or cooldown. The time required to cool from 350°F to 200°F is a function of the number of CC and RHR trains operating (Ref. 1).

The CC System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power. One CC train is sufficient to remove decay heat during subsequent operations.

The CC System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

The CC trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one CC train is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two trains of CC must be OPERABLE. At least one CC train will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.

A CC train is considered OPERABLE when:

a. The pump is OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of CC from other components or systems may render those components or systems inoperable but does not affect the OPERABILITY of the CC System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CC System is a normally operating system, which must be prepared to perform its post accident safety functions, including but not limited to RCS heat removal, which is achieved by cooling the RHR heat exchanger.

In MODE 5 or 6, the OPERABILITY requirements of the CC System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6,

BASES

ACTIONS

A.1 (continued)

"RCS Loops-MODE 4," be entered if an inoperable CC train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one CC train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CC train is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CC train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CC flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CC System.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1 (continued)

This SR verifies the correct alignment for manual, air operated, and automatic valves in the CC flow path. This provides assurance that the proper flow paths exist for CC operation.

This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. Control room indication may be used to fulfill this SR.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CC valves on an actual or simulated safety injection actuation signal.

The CC System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This test is considered satisfactory if control board indication and visual observation of the equipment demonstrate that all components have operated properly.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.2 (continued)

The 24 month Frequency is based on engineering judgement and to allow performance of this Surveillance under the conditions that apply during a unit outage. Although this SR may be performed during normal power operation, there may be plant conditions when it is advantageous to perform this Surveillance during a unit outage.

Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the Frequency is acceptable from a reliability standpoint.

This SR is modified by a Note stating that the SR applies to those valves required to align CC System to support the safety injection or recirculation phases of emergency core cooling.

SR 3.7.7.3

The CC pumps may be actuated by either a safety injection signal or system low pressure. This SR verifies proper automatic operation of the CC pumps on an actual or simulated safety injection actuation signal and verifies proper automatic operation of the CC pumps on an actual or simulated low pressure actuation signal. This test is considered satisfactory if control board indication and visual observation of the equipment demonstrate that all components have operated properly.

The CC System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation.

The 24 month Frequency is based on engineering judgement and to allow performance of this Surveillance under the conditions that apply during a unit outage. Although this SR may be performed during normal power operation, there may be plant conditions when it is advantageous to perform this Surveillance during a unit outage.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.7.3 (continued) Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the Frequency is acceptable from a reliability standpoint.
REFERENCES	1. USAR, Section 10.4.

B 3.7 PLANT SYSTEMS

B 3.7.8 Cooling Water (CL) System

BASES

BACKGROUND

The CL System is a shared system common to both units. The CL System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the CL System also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The CL System consists of a common CL pump discharge header for the five CL (2 nonsafeguards, 2 safeguards, 1 that can be designated as safeguards or nonsafeguards) pumps that directs flow into two separate, 100% capacity, CL headers. Each header then supplies loops in the turbine and auxiliary buildings and containments for the two units.

Each safeguards CL train consists of:

a. One 100% capacity vertical safeguards pump (12 or 121 for Train A; 22 or 121 for Train B);

b. A header; and

c. Piping, valving, instrumentation and controls.

The vertical motor driven pump (121) may be directed to supply either CL header when aligned in its safeguards mode of operation. In this case, the vertical motor driven pump (121) may replace a vertical diesel driven pump.

The vertical motor driven pump may be powered from a pre-selected independent power source (one of the Unit 2 redundant

BASES

BACKGROUND (continued)

safeguards 4 kV buses and associated diesel generator, i.e., 121 CL pump can be aligned to fulfill a Train A or Train B function).

A single CL pump can provide sufficient cooling in one unit during the injection and recirculation phases of a postulated loss of coolant accident plus sufficient cooling to maintain the second unit in a safe shutdown condition.

The CL pump discharge header contains redundant motor-operated header isolation valves (MV-32034, MV-32035, MV-32036, and MV-32037) that assure at least one OPERABLE safeguards pump is aligned to each safeguards supply header when functioning under accident conditions.

The safeguards diesel driven pumps and the vertical motor driven pump (when aligned in the safeguards mode) supply the safeguards components after being automatically started upon receipt of a safety injection or header low pressure signal.

Principal post accident heat loads supplied by the CL System include Unit 1 diesel generators, control room chillers, component cooling (CC) heat exchangers, containment fan coil units, and the nonsafeguards instrument air compressors.

The cooling water supplied to all safeguards and nonsafeguards equipment from supply header A is normally discharged through the Train A CL return header to the Unit 1 Circulating Water (CW) return header. The cooling water supplied to all safeguards and nonsafeguards equipment from supply header B is normally returned through the Train B CL return header to the Unit 2 CW discharge header. The auxiliary feed pumps, safeguards traveling screens, and filtered water supplies do not have return lines.

The two CL return headers are connected through two normally closed, motor-operated isolation valves. An emergency dump to

BASES

BACKGROUND (continued)

grade is connected between the isolation valves. The dump to grade requires manual actuation of the motor valve, either locally or from the main control room. Each of the return headers discharges to a standpipe in the turbine building which directs the cooling water to the CW discharge piping. Each of the standpipes is equipped with an overflow line to the ground outside the turbine building.

The CL System also provides the backup safeguards water supply to the Auxiliary Feedwater System (LCO 3.7.5).

The CL System, in conjunction with the CC System, also cools the unit from residual heat removal (RHR) entry conditions to MODE 5 during normal and post accident operations, as discussed in the USAR (Ref. 1). The time required for this evolution is a function of the number of CC and RHR System trains that are operating. One CL train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6.

Additional information about the design and operation of the CL System, along with a list of the components served, is presented in the USAR (Refs. 1 and 2).

APPLICABLE SAFETY ANALYSES

The design basis of the CL System is to maintain cooling for the heat loads of one unit in MODE 3 and the second unit in long term post accident condition.

One CL train, in conjunction with the CC System and a 100% capacity containment cooling system, has the capability to remove long term core decay heat following a design basis LOCA as discussed in the USAR (Ref. 2). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the Emergency Core Cooling System (ECCS) pumps. The CL System is designed to perform its function with a single

BASES

APPLICABLE SAFETY ANALYSES (continued)

failure of any active component, assuming the loss of offsite power. This assumes a maximum CL temperature of 95°F occurring simultaneously with design heat loads for the system.

The CL system satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two CL trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

A CL train is considered OPERABLE when:

a. The safeguards CL pump, aligned to the train, is OPERABLE;

b. The associated header is OPERABLE; and

c. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

A diesel-driven safeguards CL pump is considered OPERABLE when:

a. The pump can meet the design flow/pressure requirements in accordance with the Inservice Testing Program;

b. The associated piping, valves, auxiliaries, and instrumentation and controls required to perform the safety related function are OPERABLE; and

c. There is a minimum fuel oil supply of 19,000 gallons available for the diesel-driven safeguards pumps.

The 121 CL pump starts during low header pressure conditions and it functions as a backup source replacing a diesel driven safeguards

BASES

LCO (continued)

CL pump. In this latter case, additional requirements for OPERABILITY are specified.

121 CL pump is considered OPERABLE when:

a. The pump can meet the design flow/pressure requirements in accordance with the Inservice Testing Program; and

b. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

121 CL pump is considered OPERABLE as the safeguards substitute for 12 diesel driven CL pump when:

a. The pump can meet the design flow/pressure requirements in accordance with the Inservice Testing Program;

b. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE;

c. MV-32037 or MV-32036 are closed and the associated breaker is locked in OFF position;

d. MV-32034 and MV-32035 are open and both breakers are locked in the OFF position; and

e. Bus 27 is supplied from Bus 25.

121 CL pump is considered OPERABLE as the safeguards substitute for 22 diesel driven CL pump when:

a. The pump can meet the design flow/pressure requirements in accordance with the Inservice Testing Program;

b. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE;

BASES

LCO (continued)

c. MV-32034 or MV-32035 are closed and the associated breaker is locked in OFF position;

d. MV-32036 and MV-32037 are open and both breakers are locked in the OFF position; and

e. Bus 27 is supplied from Bus 26.

A header is considered to be OPERABLE when the associated piping, valves, and instrumentation and controls can perform the required safety related functions:

a. Provide flow and cooling for the required safeguards components supplied from the header; and

b. Provide necessary isolation functions required for the header during a safeguards actuation.

Removal of return header piping or components from service does not automatically make the system inoperable. Factors to consider during an OPERABILITY determination are:

a. If the piping or component inoperability results in an individual component being incapable of heat removal, the individual component is to be considered inoperable;

b. If the piping or component inoperability results in required components in a train being incapable of heat removal, the train is to be considered inoperable; and

c. If cooling flow for the required components can be maintained by opening the emergency dump to grade path, by routing to the other unit's discharge header, or overflow from the turbine building standpipes, the train or components are not considered inoperable.

BASES (continued)

APPLICABILITY

The CL System specification is applicable for single or two unit operation.

In MODES 1, 2, 3, and 4, the CL System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the CL System and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the CL System are determined by the systems it supports.

ACTIONS

A.1

If no safeguards CL pumps are OPERABLE for one train, action must be taken to restore one CL safeguards pump to OPERABLE status within 7 days.

Either the diesel-driven CL pump for the train may be restored to OPERABLE status, or the 121 CL pump may be aligned to fulfill the safeguards function for the train that has no OPERABLE safeguards CL pump.

The 7 day Completion Time is based on:

a. Low probability of loss of offsite power during the period;

b. The low probability of a DBA occurring during this time period;

c. The safeguards cooling capabilities afforded by the remaining OPERABLE train; and

d. The capability to route water from the non-safeguards pumps, if needed.

BASES

ACTIONS

A.1 (continued)

Required Action A.1 is modified by 3 notes. Note 1 requires Unit 1 entry into the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-Operating," for an emergency diesel generator made inoperable by the CL system. For Unit 1, the diesel generators are major heat loads supplied by the CL system. Thus, inoperability of two safeguards CL pumps will affect at least the heat loads on one CL header, including one Unit 1 diesel generator. Inability to adequately remove the heat from the diesel generator will render it inoperable.

Note 2 requires entry into the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4", for both units for the RHR loops made inoperable by the CL System. If either unit is in MODE 4, inoperability of two safeguards CL pumps may affect all the heat loads on one CL header, including a CC train and subsequently one RHR heat exchanger on each unit. Inability to adequately remove the heat from a RHR heat exchanger will render it inoperable.

Note 3 specifies that the Condition with no safeguard CL pumps OPERABLE for one train may not exist for more than 7 days in any consecutive 30 day period. If such a condition occurs, Condition C must be entered with the specified Required Action taken because the equipment reliability is less than considered acceptable.

B.1and B.2

If one CL supply header is inoperable, action must be taken to verify the vertical motor driven CL pump is OPERABLE within 4 hours, and restore the inoperable CL header to OPERABLE status within 72 hours.

BASES

ACTIONS

B.1and B.2 (continued)

Verification of pump OPERABILITY does not require the pump to be aligned and may be performed by administrative means. Completion of the CL pump surveillance tests is not required.

Conditions may occur in the CL System piping, valves, or instrumentation downstream of the supply header (e.g., closed or failed valves, failed piping, or instrumentation in a return header) that can result in the supply header being considered inoperable. In such cases, Condition B and related Required Actions shall apply.

In this Condition, the remaining OPERABLE CL header is adequate to perform the heat removal function. However, the overall redundancy is reduced because only a single CL train remains OPERABLE.

Required Action B.1 ensures that the 121 CL pump may be used to provide redundancy for the safeguards CL pump on the OPERABLE header. Required Action B.2 assures adequate system reliability is maintained.

Required Actions B.1 and B.2 are modified by two Notes.

The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-Operating," should be entered for Unit 1 since an inoperable CL train results in an inoperable emergency diesel generator.

The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4," should be entered if an inoperable CL train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

The 4 and 72 hour Completion Times are based on the redundant capabilities afforded by the OPERABLE train, and the low

BASES

ACTIONS

B.1and B.2 (continued)

probability of a DBA occurring during this time period. In addition, the 4 hour Completion Time for Required Actions B.1 and B.2 is within the time period anticipated to verify OPERABILITY of the required CL pump by administrative means.

C.1 and C.2

If at least one safeguards CL pump for a train or a CL supply header cannot be restored to OPERABLE status within the associated Completion Time, the units must be placed in a MODE in which the LCO does not apply. To achieve this status the units must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. Required Actions C.1 and C.2 are modified by a single note which specifies the actions are applicable to both units.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

In this Condition, the 14 day fuel oil supply for the diesel driven CL pumps is not available. However, the Condition is restricted to fuel oil supply reductions that maintain at least a 12 day supply. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank(s). A period of 48 hours is considered sufficient to complete restoration of the required supply prior to declaring the diesel driven CL pumps inoperable. This period is acceptable based on the remaining 12 day fuel oil supply, the fact

BASES

ACTIONS

D.1 (continued)

that procedures will be initiated to obtain replenishment, availability of the vertical motor driven CL pump and the low probability of an event during this brief period.

E.1

With the stored fuel oil supply not within the limits specified or Required Actions and associated Completion Times of Condition D not met, the diesel driven CL pumps may be incapable of performing their intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the CL System components or systems may render those components inoperable, but does not affect the OPERABILITY of the CL System.

This SR verifies the correct alignment for manual, power operated, and automatic valves in the CL System flow path to assure that the proper flow paths exist for CL System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. Control room indication may be used to fulfill this SR. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.2

This SR verifies each diesel driven CL pump can be started and be up to operating speed and assumes load within one minute to provide assurance that equipment would perform as expected in the safety analysis.

Diesel CL pump start will normally be initiated by the manual start switch. Once per calendar year, start should be initiated by use of the low pressure header pressure switch.

The 31 day Frequency is based on the experience that the CL pump usually passes the Surveillance when performed at this Frequency.

SR 3.7.8.3

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support the operation of one diesel driven CL pump for 14 days. The 14 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and plant operators would be aware of any large uses of fuel oil during this period.

SR 3.7.8.4

This SR verifies the vertical motor driven CL pump is OPERABLE to provide assurance that equipment, when lined up in the safeguards mode, will perform as expected in the safety analysis.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.4 (continued)

For this test, an acceptable level of performance shall be:

a. Pump starts and reaches required developed head; and

b. Control board indications and visual observations indicate that the pump is operating properly for at least 15 minutes.

The 92 day Frequency is based on the Inservice Testing Program requirements (Ref. 3).

Under some plant conditions, the vertical motor driven CL pump is required to operate to provide additional CL flow. When this pump is operated to support plant operations, this test can not be performed and this pump is considered inoperable as a safeguards CL pump.

SR 3.7.8.5

This SR verifies proper automatic operation of the CL System valves on an actual or simulated safety injection actuation signal, including those valves that isolate nonessential equipment from the system. The CL System is a normally operating system that is shared between the two units and cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

These tests demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry.

Unit 1 SI actuation circuits for Train A and Train B valves shall be tested during Unit 1 refueling outages. Unit 2 SI actuation circuits for Train A and Train B valves shall be tested during Unit 2 refueling outages.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.5 (continued)

A test is considered satisfactory if control board indication and visual observations indicate that all components have operated satisfactorily and if cooling water flow paths required for accident mitigation have been established.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during an outage of one unit (the other unit may be operating) and the potential for an unplanned transient in the unit affected by the tested components if the Surveillance were performed with that reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.6

The safeguards CL pumps may be actuated by either a safety injection (SI) signal or system low pressure. This SR verifies proper automatic operation of the diesel driven and vertical motor driven CL pumps on an actual or simulated safety injection actuation signal and verifies proper automatic operation of these pumps on an actual or simulated low pressure actuation signal. The CL is a normally operating system that cannot be fully actuated in a safeguards mode as part of normal testing during normal operation. A test is considered satisfactory if control board indication and visual observations indicate that all components have operated satisfactorily.

The 24 month Frequency is based on the need to perform the SI signal portion of this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.8.6 (continued) Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the Frequency is acceptable from a reliability standpoint.
REFERENCES	1. USAR, Section 10.4. 2. USAR, Section 6. 3. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.9 Emergency Cooling Water (CL) Supply

BASES

BACKGROUND

The Emergency CL Supply is designed to provide a supply of screened cooling water (CL) following an earthquake that destroys Dam No. 3, dropping the water level in the normal intake canal supply to the screenhouse. The Emergency CL Supply consists of the Emergency CL Line and the two Safeguards Traveling Screens.

The emergency pump bay, located within the safeguards section of the screen house, houses the safeguards traveling screens, the motor-driven vertical CL pump, and the diesel-driven vertical CL pumps. Two normally open sluice gates, one from each unit's circulating water (CW) pump bay, provide water to the vertical CL pumps. Sluice gates may be used to isolate the emergency bay from the CW pump bays.

Under design basis seismic event conditions, water will be supplied to the emergency bay through an Emergency Cooling Water Line. The 36-inch pipe, buried in the approach canal and Circulating Water Intake Canal bottom, directs water from the deepest part of the river to the emergency bay. The intake end of the pipe is covered with a screen to minimize the amount of trash drawn into the pipe. The Emergency Cooling Water Line is designed to provide adequate flow at the lowest possible water elevation resulting from loss of Dam No. 3. The pipe is buried approximately 40 feet below the Circulating Water Intake Canal water level.

Two safeguards traveling screens are designed to remove debris from the cooling water entering the emergency pump bay through the Emergency CL Line. Trash trays attached to the screens aid in carrying the trash to a trash trough. The screens have two speeds. The screens are backwashed with water supplied from the CL pump discharge.

BASES

BACKGROUND (continued)	Additional information on the design and operation of the Safeguards Traveling Screens and the Emergency CL Line can be found in Reference 1.
APPLICABLE SAFETY ANALYSES	The Design Basis Earthquake provides the basis for the Emergency CL Supply. This safety analysis assumes that Dam No. 3 is destroyed by the seismic event, such that supply through the Emergency CL Line is required. Under these conditions, trash removal by the safeguards traveling screens is required. The Emergency CL Supply satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This specification applies to single or dual unit operation. The Emergency CL Supply is a passive gravity fed supply to the safeguards CL pumps intended only for the case of a seismic event that destroys Dam No. 3 resulting in low level in the intake canal. In this low probability event, the safeguards traveling screens would be required to remove debris from the water supply to the pumps. Both safeguards traveling screens are required to be OPERABLE. A safeguards traveling screen is considered OPERABLE when: a. The valve, instrumentation and controls required to provide the screen backwash function are OPERABLE; and b. The safeguards traveling screen can turn.

BASES

LCO (continued)

Safeguards traveling screen OPERABILITY is not required for OPERABILITY of the safeguards CL pumps (LCO 3.7.8).

The Emergency CL Line is OPERABLE when a flow path through the pipe exists.

APPLICABILITY

With either unit in MODES 1, 2, 3, and 4, the Safeguards Traveling Screens and Emergency CL Line are required to support the OPERABILITY of the equipment serviced by the CL System during the design basis condition and required to be OPERABLE in these MODES.

With both units in MODE 5 or 6, the OPERABILITY requirements of the Emergency CL Supply are determined by the systems it supports. The design basis does not include shutdown conditions.

ACTIONS

The ACTIONS table is modified by a Note indicating that the Conditions and Required Actions are applicable to both units. This is appropriate since the Emergency CL Supply is shared by the two units.

A.1 and A.2

If one safeguards traveling screen is inoperable, action must be taken to verify an emergency bay sluice gate is open within 4 hours, and restore that safeguards traveling screen to OPERABLE status within 90 days.

BASES

ACTIONS

A.1 and A.2 (continued)

In this Condition, the remaining OPERABLE safeguards traveling screen or open emergency bay sluice gate is adequate to provide the CL supply to any of the three vertical CL pumps during any design basis condition.

Required Action A.1 is modified by a Note which states the action is not required during testing periods of less than 24 hours.

The 4 hour Completion Time is based on the redundant capability afforded by the OPERABLE safeguards traveling screen.

The 90 day Completion Time is based on:

a. The redundant capability afforded by the remaining OPERABLE safeguards traveling screen;

b. The low risk impact of an inoperable safeguards traveling screen; and

c. The low probability of a high magnitude earthquake that could destroy Dam No. 3 during this time interval.

B.1 and B.2

If both safeguards traveling screens are inoperable, action must be taken to verify one emergency bay sluice gate is open within 1 hour, and restore one safeguards traveling screen to OPERABLE status within 7 days.

In this Condition, the open emergency bay sluice gate is adequate to perform the CL supply function except in those cases where use of the Emergency CL Line is needed. As a result, overall reliability is reduced.

BASES

ACTIONS

B.1 and B.2 (continued)

The 7 day Completion Time is based on the low probability of a design basis earthquake occurring during this time interval.

C.1 and C.2

If the Emergency CL Line is inoperable, action must be taken to verify one emergency bay sluice gate is open within 1 hour, and restore the Emergency CL Line to OPERABLE status within 7 days.

The 1 hour and 7 day Completion Times are reasonable based on the low probability of a design basis earthquake occurring during the 7 days that the Emergency CL Line is inoperable , the availability of other water sources, and the time required to reasonably complete the Required Actions.

D.1 and D.2

If the Emergency CL Line or Safeguards Traveling Screen(s) cannot be restored to OPERABLE status within the associated Completion Time, the units must be placed in a MODE in which the LCO does not apply. To achieve this status, the units must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies that the safeguards traveling screens can adequately filter (screen) water and that screens can backwash as needed.

This SR verifies that:

a. The backwash supply valve will open;

b. Backwash water pressure is sufficient; and

c. The safeguards traveling screens can turn.

The 92 day Frequency is based on operating experience that demonstrates this interval is sufficient to ensure screen and support equipment reliability.

REFERENCES

1. USAR, Section 10.4.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Special Ventilation System (CRSVS)

BASES

BACKGROUND

The CRSVS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity.

The CRSVS consists of two independent, redundant trains that recirculate and filter the control room air. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a cleanup fan.

Ductwork, valves or dampers, and instrumentation also form part of the system.

The CRSVS is an emergency system, parts of which may also operate during normal unit operation.

Upon receipt of the actuating signal(s), normal air supply to the control room is isolated, and the stream of ventilation air is recirculated through the system filter trains. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

Actuation of the CRSVS in the emergency mode of operation, is initiated by:

a. High radiation in the control room ventilation duct; or

b. Safety injection signal.

Actuation of the system in the emergency mode of operation closes the unfiltered outside air intake and unfiltered exhaust dampers, and

BASES

BACKGROUND (continued)

aligns the system for recirculation of the control room air through the redundant trains of HEPA and the charcoal filters. The emergency operating condition initiates filtered ventilation of the air supply to the control room.

The CRSVS operation is discussed in the USAR (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CRSVS is designed in accordance with Seismic Category I requirements.

The CRSVS is designed to maintain the control room environment for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The CRSVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CRSVS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident fission product release presented in the USAR (Ref. 2). The CRSVS function also plays a significant role in protecting control room personnel during a fuel handling accident in the spent fuel pool enclosure or the containment and a main steam line break (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The worst case single active failure of a component of the CRSVS does not impair the ability of the system to perform its design function.

The CRSVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO applies to single or dual unit operation since there is a single CRSVS for both units.

Two independent and redundant CRSVS trains are required to be OPERABLE to ensure that at least one is available assuming a single failure disables the other train. Total system failure could result in exceeding the whole body dose limit of 5 rem (Ref. 3) to the control room operator during the worst 4 week exposure following a postulated accident.

The CRSVS is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A CRSVS train is OPERABLE when the associated:

a. Cleanup fan is OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions;

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained; and

d. instrumentation, including associated radiation monitor for starting the cleanup fan, is OPERABLE, or the system is aligned to perform its safety function and is operating.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

BASES

LCO (continued)	Opening a door for personnel ingress or egress does not make the control room ventilation zone boundary inoperable. Blocking a door open (e.g., for maintenance) without a person present to close the door requires entry into an ACTION.
APPLICABILITY	In MODES 1, 2, 3, and 4 for either unit, CRSVS must be OPERABLE to control operator exposure during and following a DBA. In addition, during movement of irradiated fuel assemblies , the CRSVS must be OPERABLE to cope with the release from a fuel handling accident.
ACTIONS	A Note preceding the Actions table identifies that all Conditions and Required Actions are applicable to both units. Because of the common Control Room, both units can be affected. <u>A.1</u> When one CRSVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CRSVS train is adequate to perform the control room protection function. However, the overall redundancy is reduced because only a single CRSVS train remains OPERABLE. The 7 day Completion Time is based on the low probability of a DBA or fuel handling accident occurring during this time period, and ability of the remaining train to provide the required capability.

BASES

ACTIONS (continued)

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CRSVS train cannot be restored to OPERABLE status within the required Completion Time, both units must be placed in a MODE that minimizes accident risk. To achieve this status, the units must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

If the inoperable CRSVS train cannot be restored to OPERABLE status within the required Completion Time, Required Action C.1 must be taken to immediately place the OPERABLE CRSVS train in the emergency mode. This is a reasonable action, based on engineering judgement, to assure the control room air is filtered in the event of an accident.

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. Required Action C.2 places the plant in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

D.1

If two CRSVS trains are inoperable during movement of irradiated fuel assemblies, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room.

BASES

ACTIONS

D.1 (continued)

This places the plant in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CRSVS trains are inoperable in MODE 1, 2, 3, or 4, the CRSVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately for both units.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Each train must be operated for ≥ 15 minutes to demonstrate the system functions. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

SR 3.7.10.2

This SR verifies that the required CRSVS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.3

The CRSVS may be actuated by either a safety injection signal or a high radiation signal. This SR verifies that each CRSVS train starts and operates on an actual or simulated safety injection actuation signal and verifies each CRSVS train starts and operates on an actual or simulated high radiation signal. The Frequency of 24 months allows performance when a unit is shutdown.

SR 3.7.10.4

This SR verifies proper functioning of the CRSVS. During the emergency mode of operation, the CRSVS train is designed to provide 4000 ± 10% cfm .

The Frequency of 24 months on a STAGGERED TEST BASIS is consistent with industry component reliability experience.

REFERENCES

1. USAR, Section 10.3.
2. USAR, Section 14.9.
3. 10 CFR 50 Appendix A, GDC Criterion 19.

B 3.7 PLANT SYSTEMS

B 3.7.11 Safeguards Chilled Water System (SCWS)

BASES

BACKGROUND

The SCWS, a shared system between the two units, circulates chilled water to provide ambient air cooling to essential areas, including the control room, Unit 1 safeguards 4160 VAC and 480 VAC safeguards bus rooms, residual heat removal (RHR) pump pits, relay room, and the event monitoring equipment room. The system functions during normal plant operations and accident conditions. The system function is to remove heat generated by safety related equipment and accident conditions.

The SCWS consists of two separate, but normally cross-connected, closed 100% capacity loops. Each loop consists of a header with water chiller, expansion tank, chilled water pump, unit coolers, piping, valves, instrumentation, and controls.

A safety injection (SI) signal closes the control room chiller outlet cross-connect air operated control valves, splitting the two headers so that each header is then supplied by the associated chilled water pump and chiller.

The SCWS operation is discussed in the USAR (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the SCWS is to remove heat produced by equipment located in the various rooms during worst case heatup scenarios. The heat removal rates exceed the design basis heat generation rates in the control room, Unit 1 safeguards 4kV and 480 VAC rooms, relay room, computer room, RHR pits, and event monitoring equipment room.

In event of a single failure affecting one loop of safeguards chilled water, the alternate loop is able to meet required heat load demands.

The SCWS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

BASES (continued)

LCO

The SCWS is a shared system between the two units.

Two 100% capacity loops of the SCWS are required to be OPERABLE to ensure that at least one is available, assuming a single failure.

Even if both loops should fail, operator actions are available in procedures to provide sufficient cooling to these rooms. The RHR pumps, 480V buses and 4kV buses can perform their functions without an immediate need for equipment heat removal and their long term OPERABILITY is handled by procedures as discussed in Reference 1. The control room and relay room can provide their functions for a shorter time period before replacement heat removal is required and long term operability is handled by procedures.

The SCWS is considered to be OPERABLE when the individual components (chiller and chilled water pump) necessary to maintain the supplied components and rooms are OPERABLE in both loops. A loop is OPERABLE when:

a. Chiller is OPERABLE;

b. Chilled water pump is OPERABLE;

c. Loop separation function, required during an accident, is OPERABLE; and

d. Supplied components are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4 and during movement of irradiated fuel assemblies, the SCWS must be OPERABLE to ensure that the room temperatures will not exceed equipment operational requirements in the essential areas this system serves following an accident.

In MODES 5 and 6, the OPERABILITY requirements of the SCWS are determined by the systems it supports.

BASES (continued)

ACTIONS

The ACTIONS table is modified by a Note indicating that the Conditions and Required Actions are applicable for both units.

A.1

With one SCWS loop inoperable, action must be taken to restore OPERABLE status within 30 days.

In this Condition, the remaining OPERABLE SCWS loop is adequate to provide cooling. However, the overall reliability is reduced because a single failure in the OPERABLE SCWS loop could result in loss of SCWS function.

The 30 day Completion Time is based on the low probability of an event requiring SCWS loop separation, the consideration that the remaining loop can provide the required protection, and that alternate safety or nonsafety related cooling means are available.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable SCWS loop cannot be restored to OPERABLE status within the required Completion Time, the units must be placed in a MODE that minimizes the risk.

To achieve this status, the units must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS (continued)

C.1 and C.2

During movement of irradiated fuel, if the inoperable SCWS loop cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE SCWS loop must be placed in operation immediately. This action ensures that the required cooling function is provided.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. Required Action C.2 places the units in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

During movement of irradiated fuel assemblies, with two SCWS loops inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room.

This Action minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

If both SCWS loops are inoperable in MODE 1, 2, 3, or 4, the SCWS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the each SCWS loop actuates on an actual or simulated safety injection actuation signal.

The 24 month Frequency on a STAGGERED TEST BASIS is appropriate since significant degradation of the SCWS is slow and is not expected over this time period.

SR 3.7.11.2

This SR verifies that necessary components in each SCWS loop operate as required.

The Frequency required by the Inservice Testing program (Ref. 2) is appropriate since degradation of the SCWS could be detected in a timely manner for the components specified based on the known reliability of the components and the loop redundancy.

REFERENCES

1. USAR, Section 10.4.
2. Inservice Testing Program.

B 3.7 PLANT SYSTEMS

B 3.7.12 Auxiliary Building Special Ventilation System (ABSVS)

BASES

BACKGROUND

The ABSVS is a standby ventilation system, common to the two units, that is designed to collect and filter air from the Auxiliary Building Special Ventilation (ABSV) boundary following a loss of coolant accident (LOCA). The ABSV boundary contains those areas within the auxiliary building which have the potential for collecting significant containment leakage that could bypass the shield building and leakage from systems which could recirculate primary coolant during LOCA mitigation.

The ABSVS consists of two independent and redundant trains. Each train consists of a heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan.

Ductwork, dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the ABSV boundary following receipt of a safety injection (SI) signal.

The exhaust from the main condenser air ejector is directed to the ABSVS for filtering prior to exhausting from the plant via the shield building stack to mitigate steam generator tube leakage.

When the ABSVS actuates, the normal nonsafeguards supply and exhaust dampers close automatically, and the Auxiliary Building Normal Ventilation System supply and exhaust fans trip. The prefilters remove any large particles in the air, and with the heaters reduce the level of entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level.

BASES

BACKGROUND (continued)	The ABSVS would typically only be used for post accident atmospheric cleanup functions. The ABSVS and ABSV boundary are discussed in the USAR (References 1, 2 and 3).
APPLICABLE SAFETY ANALYSES	The design basis of the ABSVS is established by the large break LOCA. The potential leakage paths from the containment to the auxiliary building are discussed in Reference 1. The system evaluation assumes a passive failure of the ECCS outside containment, such as an RHR pump seal failure, during the recirculation mode (Ref. 4). In such a case, the system limits radioactive release to within the 10 CFR 100 (Ref. 5) limits. The analysis of the effects and consequences of a large break LOCA is presented in References 3 and 4. The ABSVS also actuates following a small break LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing. The ABSVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two independent and redundant trains of the ABSVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train. This OPERABILITY requirement ensures that the atmospheric releases, in the event of a Design Basis Accident (DBA) in containment, from ECCS pump leakage and containment leakage which bypasses the shield building would not result in doses exceeding 10 CFR 100 limits (Ref. 5). An ABSVS train is considered OPERABLE when its associated: a. Fan is OPERABLE;

BASES

LCO (continued)

b. HEPA filter and charcoal adsorbers are capable of passing their design flow and performing their filtration functions; and

c. Heater, ductwork, and dampers are OPERABLE and air circulation can be maintained.

The ABSV boundary is OPERABLE if both of the following conditions can be met:

a. Openings in the ABSV boundary are under direct administrative control and can be reduced to less than 10 square feet within 6 minutes following an accident; and

b. Dampers and actuation circuits that isolate the Auxiliary Building Normal Ventilation System following an accident are OPERABLE or can be manually isolated within 6 minutes following an accident.

To reduce leakage from the Auxiliary Building, the Turbine Building roof exhauster fans must be capable of being de-energized within 30 minutes following a loss of coolant accident.

The LCO is modified by a Note allowing the ABSV boundary to be opened under administrative controls. As discussed above, openings must be closed to less than 10 square feet within 6 minutes following an accident.

APPLICABILITY

In MODES 1, 2, 3, and 4 for either unit, the ABSVS is required to be OPERABLE.

When both units are in MODE 5 or 6, the ABSVS is not required to be OPERABLE.

BASES (continued)

ACTIONS

The Actions table is modified by a Note that indicates the Conditions and Required Actions are applicable to both units. This is appropriate because ABSVS is a shared system; thus ABSVS train inoperability can affect either or both units.

A.1

With one ABSVS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ABSVS function.

The 7 day Completion Time is appropriate because the ABSVS risk contribution is substantially less than that for the ECCS (72 hour Completion Time). The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

Concurrent failure of two ABSVS trains would result in the loss of functional capability; therefore, LCO 3.0.3 must be entered immediately.

B.1

With both ABSVS trains inoperable due to an inoperable ABSV boundary, action must be taken to restore OPERABLE status within 24 hours.

The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the availability of the ABSVS to provide a filtered release (albeit with potential for some unfiltered leakage).

If the ABSV boundary cannot be restored to OPERABLE status within the associated Completion Time, both units must be placed in a MODE in which the LCO does not apply.

BASES

ACTIONS (continued)

C.1 and C.2

If an ABSVS train cannot be restored to OPERABLE status or the ABSV boundary cannot be restored to OPERABLE status within the associated Completion Time, both units must be placed in a MODE in which the LCO does not apply.

To achieve this status, both units must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that each ABSVS train can be manually started, the associated filter heater energizes, and the filter units remain sufficiently dried out to ensure they can perform their function.

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Monthly heater operations, with air circulation through the filter, dries out any moisture that may have accumulated in the charcoal from humidity in the ambient air. Each ABSVS train must be operated ≥ 10 hours per month with the heaters energized. The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.2

This SR verifies that the required ABSVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP).

The VFTP includes testing HEPA filter performance, charcoal adsorbers efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations).

Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

This SR verifies proper functioning of the ABSVS by verifying the integrity of the ABSV boundary and the ability of the ABSVS to maintain a negative pressure with respect to potentially uncontaminated adjacent areas.

During the post accident mode of operation, the ABSVS is designed to maintain a slight negative pressure within the ABSV boundary with respect to the containment and shield building.

Each ABSVS train is started from the control room and the following are verified:

a. Associated Auxiliary Building Normal Ventilation System fans trip and dampers close; and

b. A measurable negative pressure is drawn within the ABSV boundary within 6 minutes after initiation, with a 10 square foot opening within the ABSV boundary.

The 92 day Frequency is based on the known reliability of equipment and the two train redundancy available.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.4

This SR verifies that each ABSVS train starts and operates on an actual or simulated safety injection actuation signal.

The 24 month Frequency is consistent with industry reliability experience for similar equipment. The 24 month Frequency is acceptable since this system usually passes the Surveillance when performed.

REFERENCES

1. USAR, Appendix G.
2. USAR, Section 10.3.
3. USAR, Section 14.
4. USAR, Section 6.7.
5. 10 CFR 100.11.

B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool Special Ventilation System (SFPSVS)

BASES

BACKGROUND

In this Specification, the spent fuel pool enclosure refers to the concrete building that contains the racks and storage pool used to store new and spent fuel.

SFPSVS refers to that portion of the Spent Fuel Special and Containment Inservice Purge system that provides the spent fuel pool enclosure air cleanup function.

The SFPSVS filters airborne radioactive particulates from the area of the spent fuel pool following a fuel handling accident in that area.

The Spent Fuel Pool Special Ventilation fans exhaust air to prefilter-absolute-charcoal (PAC) filters, then to the associated Shield Building vent stack (Unit 1 for Train A; Unit 2 for Train B).

The SFPSVS consists of two independent and redundant trains, each capable of meeting the design requirements.

Each train consists of a heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan.

Ductwork, dampers, and instrumentation also form part of the system. Heaters function to reduce the relative humidity of the airstream.

The system initiates filtered ventilation of the spent fuel pool enclosure following receipt of a high radiation signal from a radiation detector located in the exhaust ducting of the spent fuel pool normal ventilation system. One detector actuates Train A equipment; the other actuates Train B equipment.

BASES

BACKGROUND (continued)	The SFPSVS is a standby system. Upon receipt of the actuating signal, normal air supply to and discharge from the spent fuel pool ventilation system are isolated, and the stream of ventilation air discharges through the two SFPSVS filter trains. The prefilters remove any large particles in the air, and the heaters remove any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. The SFPSVS is discussed in the USAR (Refs. 1, 2, and 3).
APPLICABLE SAFETY ANALYSES	The SFPSVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), a fuel handling accident (FHA) in the spent fuel pool enclosure. LCO 3.9.4, "Containment Penetrations," separately addresses a fuel handling accident in containment. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that only one train of the SFPSVS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the spent fuel pool enclosure is determined for a fuel handling accident. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4). The SFPSVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two independent and redundant trains of the SFPSVS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure disables the other train. This OPERABILITY requirement ensures that the atmospheric release

BASES

LCO (continued)	from a fuel handling accident in the spent fuel pool enclosure would not result in doses exceeding the 10 CFR 100 limits. The SFPSVS is considered OPERABLE when the individual components necessary to control offsite exposure are OPERABLE in both trains. An SFPSVS train is considered OPERABLE when its associated: a. Fan is OPERABLE; b. HEPA filter and charcoal adsorber are capable of passing their design flow and performing their filtration function; c. Heater, ductwork, and dampers are OPERABLE; d. Spent Fuel Pool Normal Ventilation train radiation monitor is OPERABLE; and e. Spent Fuel Pool Normal Ventilation train is running. Opening a personnel door for personnel ingress or egress does not make the SFPSVS boundary inoperable. Blocking the door open is not allowed (Ref. 5).
APPLICABILITY	During movement of irradiated fuel in the spent fuel pool enclosure, the SFPSVS is required to be OPERABLE to alleviate the consequences of a fuel handling accident.
ACTIONS	The ACTIONS table is modified by a Note stating LCO 3.0.3 is not applicable. LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify

BASES

ACTIONS (continued)

any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shut down unnecessarily.

A.1

With one SFPSVS train inoperable, action must be taken to restore OPERABLE status within 7 days.

During this period, the remaining OPERABLE train is adequate to perform the SFPSVS function.

The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable SFPSVS train, and the remaining SFPSVS train providing the required protection.

B.1 and B.2

When Required Action A.1 cannot be completed within the required Completion Time, during movement of irradiated fuel assemblies in the spent fuel pool enclosure, the OPERABLE SFPSVS train must be started immediately or fuel movement suspended. This is a reasonable action, based on engineering judgement, to assure that spent fuel pool enclosure releases are filtered in the event of an accident.

An alternative to Required Action B.1 is to immediately suspend activities that could result in a release of radioactivity. Required Action B.2 places the plant in a condition that minimizes risk. If the system is not placed in operation, this action requires suspension of fuel movement, which precludes a fuel handling accident. This does not preclude the movement of fuel assemblies to a safe position.

BASES

ACTIONS (continued)

C.1

When two trains of the SFPSVS are inoperable during movement of irradiated fuel assemblies in the spent fuel pool enclosure, action must be taken immediately to suspend movement of irradiated fuel assemblies in the spent fuel pool enclosure. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that each SFPSVS train can be started, and that the associated filter units and heaters can perform their function.

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. Each SFPSVS train must be operated with heaters energized for ≥ 10 hours. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.13.2

This SR verifies that the required SFPSVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP).

The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.2 (continued)

Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.13.3

This SR verifies that each SFPSVS train starts and operates on an actual or simulated radiation monitor actuation signal.

The 24 month Frequency is consistent with the VFTP.

SR 3.7.13.4

This SR verifies the ability of the SFPSVS fan to maintain the design flow rate of 5200 ±10% cfm.

A 24 month Frequency (on a STAGGERED TEST BASIS) is consistent with industry reliability experience for similar equipment. The 24 month Frequency on a STAGGERED TEST BASIS is acceptable since this system usually passes the Surveillance when performed.

REFERENCES

1. USAR, Section 7.5.
2. USAR, Section 10.3.
3. USAR, Section 14.5.
4. Regulatory Guide 1.25.
5. NSP Prairie Island Safety Evaluation 50-475, "Spent Fuel Pool Personnel Access Doors".

B 3.7 PLANT SYSTEMS

B 3.7.14 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment during normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 150 gpd tube leak (LCO 3.4.14, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.17, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours).

Limiting secondary specific activity also reduces site and exclusion area boundary (EAB) exposures in the event of a steam generator tube rupture (Ref. 1).

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB) outside of containment, as discussed in the USAR (Ref. 1) and NSP License Amendment Request correspondence (Ref. 2), assumes the initial secondary coolant specific activity to have a radioactive isotope

BASES

APPLICABLE SAFETY ANALYSES (continued)

concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of MSLB do not exceed a small fraction of the unit EAB limits of 10 CFR 100.11 for whole body and thyroid dose rates.

With the loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through the main steam safety valves (MSSVs) and steam generator power operated relief valve (SG PORV). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and SG PORV during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the EAB would be a very small fraction of Reference 3 requirements if the MSSVs open for 2 hours following a trip from full power.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 3).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within36 hours.

BASES

ACTIONS

A.1 and A.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE.

The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. USAR, Section 14.5.
2. Letter entitled "Response to Request for Additional Information Related to License Amendment Request Dated May 15, 1997 Incorporation of Voltage-Based Steam Generator Tube Repair Criteria (TAC Nos. M98944 and M98945)", Joel P. Sorensen (NSP) to US Nuclear Regulatory Commission, dated October 20, 1997.
3. 10 CFR 100.11.

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel storage pool design and a description of the Spent Fuel Pool Cooling and Cleanup System is given in the USAR (Ref. 1). The assumptions of the fuel handling accident are given in Reference 2.

APPLICABLE SAFETY ANALYSES

The minimum water level in the spent fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 3). The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100.11 limits.

According to Reference 3, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 3 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few

BASES

APPLICABLE SAFETY ANALYSES (continued)	rows fail from a hypothetical maximum drop. The Fuel Handling Accident is discussed in Reference 2. The spent fuel storage pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The spent fuel storage pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for spent fuel movement within the spent fuel storage pool.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool, since the potential for a release of fission products exists.
ACTIONS	A.1 Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel storage pool is immediately suspended. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel

BASES

ACTIONS

A.1 (continued)

assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies sufficient spent fuel storage pool water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the spent fuel storage pool is in equilibrium with the refueling cavity, and the level in the refueling cavity is checked daily in accordance with SR 3.9.2.1.

REFERENCES

1. USAR, Section 10.2.
2. USAR, Section 14.5.
3. Regulatory Guide 1.25, Rev. 0.

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

The spent fuel storage pool is a two compartment pool as described in Reference 1. These 2 compartments are referred to as Pool 1 and Pool 2. Pool 1 has up to 462 storage positions. Pool 2 has up to 1120 storage positions.

Either pool is designed to accommodate fuel of various initial enrichments (up to 5 weight percent (w/o)) which have accumulated minimum burnups and decay times within the unrestricted domain according to the applicable Figure 3.7.17-1 (OFA design) or Figure 3.7.17-2 (STD design), in the accompanying LCO. Fuel assemblies not meeting the criteria of the applicable Figure 3.7.17-1 or Figure 3.7.17-2 shall be stored in accordance with paragraph 4.3.1.1 in Technical Specifications Section 4.3, Fuel Storage.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 1.00 be evaluated in the absence of soluble boron. The double contingency principle discussed in Reference 2 and the April 1978 NRC letter (Ref. 3) allows credit for additional soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. Safe operation of the spent fuel pool may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.17, "Spent Fuel Pool Storage" and by maintaining boron concentration in accordance with this LCO.

BASES (continued)

APPLICABLE SAFETY ANALYSES

Most accident conditions in the spent fuel pool will not result in an increase in K_{eff} of the racks. Examples of those accident conditions which will not result in an increase in K_{eff} are a fuel assembly drop on the top of the racks, a fuel assembly drop between rack modules and wall (rack design precludes this condition), and a drop or placement of a fuel assembly into the cask loading area of the small pool. However, two accidents can be postulated which could increase reactivity. The first postulated accident would be a loss of the spent fuel pool cooling system and the second would be a misload of a fuel assembly into a cell for which the restrictions on location, enrichment, burnup, decay time or gadolinium credit are not satisfied. For an occurrence of these postulated accident conditions, the double contingency principle of Reference 2 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 750 ppm required to maintain K_{eff} less than 0.95 under normal conditions) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

Calculations were performed (Ref. 4) to determine the amount of soluble boron required to offset the highest reactivity increase caused by either of these postulated accidents and to maintain K_{eff} less than or equal to 0.95. It was found that a spent fuel pool boron concentration of 1300 ppm was adequate to mitigate these postulated criticality related accidents and to maintain K_{eff} less than or equal to 0.95. This specification ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by a mispositioned fuel assembly or a loss of spent fuel pool cooling. The 1800 ppm spent fuel pool boron concentration limit in this specification was chosen to be consistent with the boron concentration limit required for a spent fuel cask containing fuel. The 1800 ppm limit will ensure that K_{eff} for the

BASES

APPLICABLE SAFETY ANALYSES (continued)	spent fuel cask, including statistical probabilities, will be less than or equal to 0.95 for all postulated arrangements of fuel within the cask. Technical Specifications Section 4.3 requires that the spent fuel rack K_{eff} be less than or equal to 0.95 when flooded with water borated to 750 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 K_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration from 1800 ppm to 750 ppm is not a credible event. The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The fuel storage pool boron concentration is required to be ≥ 1800 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in References 4 and 5. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage, movement within the fuel storage pool, and for loading and unloading a spent fuel storage cask.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool.
ACTIONS	<u>A.1 and A.2</u> The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

BASES

ACTIONS

A.1 and A.2 (continued)

When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies that the concentration of boron in the spent fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

REFERENCES

1. USAR, Section 10.2.

2. ANSI/ANS-8.1-1983.

3. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978.

BASES

REFERENCES (continued)

4. "Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit", Westinghouse Commercial Nuclear Fuel Division, February 1997.

5. USAR, Section 14.5.

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Pool Storage

BASES

BACKGROUND

The spent fuel storage pool is a two compartment pool as described in the USAR (Ref. 1). These 2 compartments are referred to as Pool 1 and Pool 2. Fuel stored in the Prairie Island fuel storage pools include fuel with the:

a. OFA designation, which includes the Westinghouse OFA and Vantage Plus designs; and

b. STD designation, which includes the Westinghouse Standard and Exxon fuel designs.

Criticality considerations provide the primary basis for storage limitations.

Pool 1 may contain up to 462 storage positions, except when the pool is used for cask laydown. In the latter case, only 266 storage positions are available since 4 storage racks must be removed to accommodate the storage cask. Pool 2 has up to 1120 storage positions.

Pools 1 and 2 are designed to accommodate fuel of various initial enrichments (up to 5 weight percent (w/o)), which have accumulated minimum burnups and decay times within the unrestricted domain according to the applicable Figure 3.7.17-1 (OFA Fuel) or Figure 3.7.17-2 (STD Fuel), in the accompanying LCO.

Fuel assemblies not meeting the criteria of the applicable Figure 3.7.17-1 or Figure 3.7.17-2 shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.

BASES

BACKGROUND (continued)

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 1.00 be evaluated in the absence of soluble boron. The double contingency principle discussed in Reference 2 and the April 1978 NRC letter (Ref. 3) allows credit for additional soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. To mitigate postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the spent fuel pool may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO and maintaining boron concentration in accordance with LCO 3.7.16.

APPLICABLE SAFETY ANALYSES

The hypothetical criticality accidents can only take place during or as a result of the movement of an assembly (Ref. 4). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.16, "Fuel Storage Pool Boron Concentration") prevents criticality. By closely controlling the movement of each assembly and by verifying the appropriate checkerboarding after each fuel handling campaign, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for criticality accidents, the operation may be under the auspices of the accompanying LCO.

The spent fuel storage racks have been analyzed in accordance with the methodology contained in Reference 5. That methodology ensures that the spent fuel rack multiplication factor, K_{eff}, is less than 0.95 as recommended by ANSI 57.2-1983 (Ref. 6) and NRC guidance (Ref. 3). The codes, methods and techniques contained in the methodology are used to satisfy this criterion on K_{eff}. The resulting Prairie Island spent fuel rack criticality analysis allows for the storage of fuel assemblies with enrichments up to a maximum

BASES

APPLICABLE SAFETY ANALYSES (continued)

of 5.0 weight percent U-235 while maintaining $K_{eff} \leq 0.95$ including uncertainties and credit for soluble boron. In addition, sub-criticality of the pool ($K_{eff} < 1.0$) is assured on a 95/95 basis, without the presence of the soluble boron in the pool. Credit is taken for radioactive decay time of the spent fuel and for the presence of fuel rods containing gadolinium burnable poison.

The criticality analysis (Ref. 7) utilized the following storage configurations to ensure that the spent fuel pool will remain subcritical during the storage of fuel assemblies with all possible combinations of burnup and initial enrichment:

a. The first storage configuration utilizes a checkerboard loading pattern to accommodate new or low burnup fuel with a maximum enrichment of 5.0 w/o U-235. This configuration stores "burned" and "fresh" fuel assemblies in a 3x3 checkerboard pattern as shown in Figure 4.3-1. Fuel assemblies stored in "burned" cell locations are selected based on a combination of fuel assembly type, initial enrichment, discharge burnup and decay time (Figures 4.3-3 through 4.3-12). The criteria for the fuel stored in the "burned" locations is also dependent on the number of rods containing gadolinium in the center "fresh" fuel assembly. The use of empty cells is also an acceptable option for the "burned" cell locations. This will allow the storage of new or low burnup fuel assemblies in the outer rows of the spent fuel storage racks because the area outside the racks can be considered to be empty cells.

   Fuel assemblies that fall into the restricted range of Figures 3.7.17-1 or 3.7.17-2 are required to be stored in "fresh" cell locations as shown in Figure 4.3-1. The criteria included in Figures 3.7.17-1 and 3.7.17-2 for the selection of fuel assemblies to be stored in the "fresh" cell locations is based on a combination of fuel assembly type, initial enrichment, decay time and discharge burnup.

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. The second storage configuration does not utilize any special loading pattern. Fuel assemblies with burnup, initial enrichment and decay time which fall into the unrestricted range of Figures 3.7.17-1 or 3.7.17-2, as applicable, can be stored anywhere in the region with no special placement restrictions.

The burned/fresh fuel checkerboard region can be positioned anywhere within the spent fuel racks, but the boundary between the checkerboard region and the unrestricted region must be either:

a. Separated by a vacant row of cells; or

b. The interface must be configured such that there is one row carryover of the pattern of burned assemblies from the checkerboard region into the first row of the unrestricted region (Figure 4.3-2).

Specification 3.7.17 and Section 4.3 ensure that fuel is stored in the spent fuel racks in accordance with the storage configurations assumed in the spent fuel rack criticality analysis (Ref. 7).

The spent fuel pool criticality analysis addresses all the fuel types currently stored in the spent fuel pool and in use in the reactor. The fuel types considered in the analysis include the Westinghouse Standard (STD), OFA, and Vantage Plus designs, and the Exxon fuel assembly types in storage in the spent fuel pool. The OFA designation on the figures in Specification 3.7.17 and Section 4.3 bound all of the Westinghouse OFA and Vantage Plus fuel assemblies at Prairie Island. The STD designation on the figures in Specification 3.7.17 and Section 4.3 bound all of the Westinghouse STD and Exxon fuel assemblies at Prairie Island.

Most accident conditions in the spent fuel pool will not result in an increase in K_{eff} of the racks. Examples of those accident conditions

BASES

APPLICABLE SAFETY ANALYSES (continued)

which will not result in an increase in K_{eff} are:

a. A fuel assembly drop on the top of the racks;

b. A fuel assembly drop between rack modules and wall (rack design precludes this condition); and

c. A drop or placement of a fuel assembly into the cask loading area of the small pool.

However, two accidents can be postulated which could increase reactivity. The first postulated accident would be a loss of the spent fuel pool cooling system and the second would be a misload of a fuel assembly into a cell for which the restrictions on location, enrichment, burnup, decay time or gadolinium credit are not satisfied.

For an occurrence of these postulated accident conditions, the double contingency principle of Reference 2 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 750 ppm required to maintain K_{eff} less than 0.95 under normal conditions) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

Westinghouse Commercial Nuclear Fuel Division calculations (Ref. 7) were performed to determine the amount of soluble boron required to offset the highest reactivity increase caused by either of these postulated accidents and to maintain K_{eff} less than or equal to 0.95. It was found that a spent fuel pool boron concentration of 1300 ppm was adequate to mitigate these postulated criticality related accidents and to maintain K_{eff} less than or equal to 0.95.

Specification 3.7.16 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by

BASES

APPLICABLE SAFETY ANALYSES (continued)

a mispositioned fuel assembly or a loss of spent fuel pool cooling. The 1800 ppm spent fuel pool boron concentration limit in Specification 3.7.16 is consistent with the boron concentration limit required for a spent fuel cask containing fuel.

Section 4.3 requires that the spent fuel rack K_{eff} be less than or equal to 0.95 when flooded with water borated to 750 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 K_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration from 1800 ppm to 750 ppm is not a credible event.

When the requirements of Specification 3.7.17 are not met, immediate action must be taken to move any noncomplying fuel assembly to an acceptable location to preserve the double contingency principle assumption of the criticality accident analysis.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with the applicable Figure 3.7.17-1 (OFA Fuel) or Figure 3.7.17-2 (STD Fuel), in the accompanying LCO, ensure the k_{eff} of the spent fuel storage pool will always remain < 0.95, with credit given for boron in the water.

Fuel assemblies not meeting the criteria of the appropriate Figure 3.7.17-1 or Figure 3.7.17-2 shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.

BASES (continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the spent fuel storage pool is not in accordance with the applicable Figure 3.7.17-1 or Figure 3.7.17-2, or paragraph 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with the applicable Figure 3.7.17-1 or Figure 3.7.17-2 or Specification 4.3.1.1.

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with the applicable Figure 3.7.17-1 or Figure 3.7.17-2 in the accompanying LCO. For fuel assemblies in the restricted range of the applicable Figure 3.7.17-1 or Figure 3.7.17-2, performance of this SR will ensure compliance with Specification 4.3.1.1.

The Frequency of this SR is prior to storing or moving a fuel assembly.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.17.2

This SR verifies that the fuel assemblies in the spent fuel storage racks are stored in accordance with the requirements of LCO 3.7.17 and Section 4.3.1.1.

The intent of this SR is to not require completion of the spent fuel pool inventory verification during interruptions in fuel handling during a defined fuel handling campaign. No spent fuel pool inventory verification is required following fuel movements where no fuel assemblies are relocated to different spent fuel rack locations.

The Frequency of this SR requires performance within 7 days after the completion of any fuel handling campaign which involves:

a. The relocation of fuel assemblies within the spent fuel pool; or

b. The addition of fuel assemblies to the spent fuel pool.

The extent of a fuel handling campaign will be defined by plant administrative procedures. Examples of a fuel handling campaign would include all the fuel handling performed during a refueling outage or associated with the placement of new fuel into the spent fuel pool.

The 7 day allowance for completion of this SR provides adequate time for completion of the spent fuel pool inventory verification while minimizing the time a fuel assembly may be misloaded in the spent fuel pool. If a fuel assembly is misloaded during the fuel handling campaign, the minimum boron concentration required by LCO 3.7.16 will ensure that the spent fuel rack K_{eff} remains within limits until the spent fuel inventory verification is performed.

BASES (continued)

REFERENCES

1. USAR, Section 10.2.
2. ANSI/ANS-8.1-1983.
3. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978.
4. "Criticality Analysis of the Prairie Island Units 1 & 2 Fresh and Spent Fuel Racks", Westinghouse Commercial Nuclear Fuel Division, February 1993.
5. WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", Revision 1, November 1996.
6. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants", ANSI/ANS-57.2-1983, October 7, 1983.
7. "Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit", Westinghouse Commercial Nuclear Fuel Division, February 1997.