TVA

Tennessee Valley Authority, 1101 Market Street, LP 5A, Chattanooga, Tennessee 37402-2801

December 21, 2011

10 CFR Part 50

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Project Number 785

**SUBJECT: TENNESSEE VALLEY AUTHORITY (TVA)
CLINCH RIVER CONSTRUCTION PERMIT (CRCP) PROJECT
FOURTH REGULATORY FRAMEWORK WORKSHOP**

References:
1) TVA letter to NRC dated August 24, 2011, "Clinch River Construction Permit (CRCP) Project, First Regulatory Framework Workshop"
2) TVA letter to NRC dated October 14, 2011, "Second Regulatory Framework Workshop"
3) TVA letter to NRC dated November 10, 2011, "Third Regulatory Framework Workshop"

Please find attached the Regulatory Framework Documents and Section Outlines for the following sections to be presented at the public meeting:

PSAR Chapters 7, 8, 12, 13, 14, 17, 18, 19, and Part 5

TVA and Generation mPower have worked closely together to develop these Regulatory Framework Documents and Section Outlines, and look forward to receiving the NRC Staff's feedback at the January 24, 2012 public meeting. Please contact Thomas Spink at (423) 751-7062 if you have questions.

Sincerely,

Jack A. Bailey

Jack A. Bailey
Vice President, Nuclear Generation Development
Nuclear Generation Development and Construction

Attachment
cc: See Page 2

D104
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cc: w/Attachment

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Attachment

Regulatory Framework Documents and Section Outlines

Chapter 7

Chapter 8

Chapter 12

Chapter 13

Part 5

Chapter 14

Chapter 17

Chapter 18

Chapter 19

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1 70	Changes to the Standard Plant Design	Related Sections	Key Issues
7.1 Instrumentation and Controls - Introduction	PSAR	10 CFR Part 50 10 CFR 50 App. A GDC 1 10 CFR 100 10 CFR 50.55a(a)(1)	No	RG 1.70, RG 1.206	7.1 Table 7-1 Appendix 7.1-A Appendix 7.1-B Appendix 7.1-C Appendix 7.1-D	RG 1.152 SECY-93-087 (plus SRM) DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	IEEE Std. 603-1991 and correction sheet January 30, 1995 IEEE Std. 7-4 3.2-2003	Yes. Chapter 7 will conform to current regulatory guidance	No	Chapters 1 through 6, 8 through 10, and 12 through 19	Assess & apply any I&C lessons learned from Fukushima Daichi power station. Also address: 1. Compliance with IEEE 603-1991. 2. Development of defense-in-depth and diversity technical report for the PSAR. 3. See Sections 7.4, 7 8 and 7.9 4. RTNSS issues 5. DAC/ITAAC for Digital I&C systems (RG 1.70 does not recognize RTNSS, DAC or ITAAC). 6. Digital I&C Software Quality Assurance (RG 1.152), Diversity, and Cyber-Security (RG 5 71) in the PSAR RFDs, and what is required for the CPA. The I&C design will be informed by the latest issued version of the IEEE Standards and ISGs Commitments to the NRC, however, will be based upon guidance endorsed in RGs, BTPs and SRP Sections issued 6 months prior to the submittal of the PSAR. A Cyber Security Plan Report will be submitted separately.
	DCD	10 CFR 50 or 10 CFR Part 52 10 CFR 50 App A GDC 1 10 CFR 100 10 CFR 50 67 10 CFR 52.47(b)(1) 10 CFR 50 55a(a)(1)	No	RG 1.206	7.1 Table 7-1 Appendix 7.1-A Appendix 7.1-B Appendix 7.1-C Appendix 7.1-D	RG 1 152 SECY-93-087 (plus SRM) DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	IEEE Std 603-1991 and correction sheet January 30, 1995 IEEE Std 7-4 3.2-2003	N/A	N/A	Chapters 1 through 6, 8 through 10, and 12 through 19	See above
	FSAR	10 CFR 50 10 CFR 50 App A GDC 1 10 CFR 100 10 CFR 50 67 10 CFR 50 55a(a)(1)	No	RG 1 206 Note: RG 1 206 specifically states that section 7.1.2 of the FSAR should include the specific information identified in NUREG-0800, SRP Chapter 7, Appendix 7.1-A	7.1 Table 7-1 Appendix 7 1-A Appendix 7 1-B Appendix 7.1-C Appendix 7.1-D	RG 1.152 SECY-93-087 (plus SRM) DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	IEEE Std. 603-1991 and correction sheet January 30, 1995 IEEE Std 7-4 3 2-2003	N/A	No	Chapters 1 through 6, 8 through 10, and 12 through 19	See above

Note:
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
7.2 Reactor Trip System	PSAR	10 CFR 50 10 CFR 50 App. A GDC 1, 2, 4, 10, 13, 15, 19-25, 29 10 CFR 100 10 CFR 50.55a(a)(1) 10 CFR 50.55a(h) 10 CFR 50.55a(h)(2)	No	RG 1 206	7 2 BTP 7-3, BTP 7-5, BTP 7-8, BTP 7-9, BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1.22, RG 1.47, RG 1.53, RG 1.62, RG 1.75, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1 204 SRM on SECY-93-087 NUREG/CR-6303 DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	IEEE Std. 379-2000 IEEE Std 603-1991 IEEE Std. 7-4 3.2-2003 IEEE Std 323-2003 IEEE Std 338-1987 IEEE Std. 384-1992 IEEE Std 473-1985 IEEE Std. 497-2002 IEEE Std. 518-1982 IEEE Std. 610.12-1990 IEEE Std 665-1995 IEEE Std 666-1991 IEEE Std. 729-1983 IEEE Std. 828-1990 IEEE Std. 830-1993 IEEE Std 1008-1987 IEEE Std 1012-1998 IEEE Std. 1028-1997 IEEE Std. 1050-1996 IEEE Std. 1074-1995 IEEE Std C62.23-1995 IEEE Std C62.41-1991 IEEE Std C62.45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std. 1042-1987 MIL-STD-461E	Yes. Chapter 7 will conform to current regulatory guidance.	No	Chapters 1 through 6, 8 through 10, and 12 through 19	Assess & apply any I&C lessons learned from Fukushima Daichi power station. Also address: 1. Compliance with IEEE 603-1991. 2. Development of defense-in-depth and diversity technical report for the PSAR 3 See Sections 7.4, 7.8 and 7.9 4 RTNSS issues 5. DAC/ITAAC for Digital I&C systems (RG 1.70 does not recognize RTNSS, DAC or ITAAC) 6. Digital I&C Software Quality Assurance (RG 1.152), Diversity, and Cyber-Security (RG 5 71) in the PSAR RFDs, and what is required for the CPA The I&C design will be informed by the latest issued version of the IEEE Standards and ISGs Commitments to the NRC will be based upon guidance endorsed in RGs, BTPs and SRP sections issued 6 months prior to the submittal of the PSAR. A Cyber Security Plan Report will be submitted separately. As part of the standard plant design, an alternate review process to BTP 7-14 will be developed and pursued with the NRC for the mPower digital I&C systems software quality assurance program. The approach will be that B&W will comply with BTP 7-14, or with an alternate process deemed acceptable to the NRC (in this case, the software development process described in the capability maturity model and integration (CMMI) developed and assessed by the Software Engineering Institute) It is recognized that implementation of an alternate approach to BTP 7-14 would require NRC concurrence and potential revisions to the SRP.
	DCD	10 CFR 50 or 10 CFR Part 52 10 CFR 50 App A GDC 1, 2, 4, 10, 13, 15, 19-25, 29 10 CFR 100 10 CFR 50 67 10 CFR 50 55a(a)(1) 10 CFR 50.55a(h) 10 CFR 50.55a(h)(2) 10 CFR 50 34(f) 10 CFR 50 34(f)(2)(v) 10 CFR 50 34(f)(2)(xxiv) 10 CFR 52 47(b)(1)	No	RG 1 206	7.2 BTP 7-3, BTP 7-5, BTP 7-8, BTP 7-9, BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1 22, RG 1 47, RG 1.53, RG 1.62, RG 1.75, RG 1 105, RG 1.118, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 SRM on SECY-93-087 NUREG/CR-6303 DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	IEEE Std 379-2000 IEEE Std 603-1991 IEEE Std 7-4.3 2-2003 IEEE Std. 323-2003 IEEE Std 338-1987 IEEE Std. 384-1992 IEEE Std. 473-1985 IEEE Std 497-2002 IEEE Std 518-1982 IEEE Std 610 12-1990 IEEE Std 665-1995 IEEE Std 666-1991 IEEE Std 729-1983 IEEE Std 828-1990 IEEE Std 830-1993 IEEE Std. 1008-1987 IEEE Std 1012-1998 IEEE Std 1028-1997 IEEE Std. 1050-1996 IEEE Std. 1074-1995 IEEE Std C62 23-1995 IEEE Std C62.41-1991 IEEE Std. C62.45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std 829-1983 ANSI/IEEE Std 1042-1987 MIL-STD-461E	N/A	N/A	Chapters 1 through 6, 8 through 10, and 12 through 19	See above

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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1 70	Changes to the Standard Plant Design	Related Sections	Key Issues
	FSAR	10 CFR Part 50 10 CFR 50 App A GDC 1, 2, 4, 10, 13, 15, 19-25, 29 10 CFR 100 10 CFR 50 67 10 CFR 50 55a(a)(1) 10 CFR 50.55a(h) 10 CFR 50 55a(h)(2) 10 CFR 50 34(f) 10 CFR 50 34(f)(2)(v) 10 CFR 50 34(f)(2)(xxiii)	No	RG 1 206	7.2 BTP 7-3, BTP 7-5, BTP 7-8, BTP 7-9, BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1 22, RG 1.47, RG 1 53, RG 1 62, RG 1.75, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1 168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 SRM on SECY-93-087 NUREG/CR-6303 DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	Same as DCD	N/A	No	Chapters 1 through 6, 8 through 10, and 12 through 19	See above

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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
7.3 Engineered Safety Feature Systems	PSAR	10 CFR Part 50 10 CFR 50 App. A GDC 1,2,4, 10, 13, 15, 16, 19-24, 29, 33-35, 38, 41, 44 10 CFR 50.55a(a)(1) 10 CFR 50.55a(h) 10 CFR 50 55a(h)(2)	No	RG 1.206	7.3 BTP 7-3, BTP 7-4, BTP 7-6, BTP 7-8, BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1.22, RG 1.47, RG 1.53, RG 1.62, RG 1.75, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1 204 SRM on SECY-93-087 NUREG/CR-6303 DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	IEEE Std 379-2000 IEEE Std 603-1991 IEEE Std 7-4.3 2-2003 IEEE Std. 323-2003 IEEE Std. 338-1987 IEEE Std 384-1992 IEEE Std 473-1985 IEEE Std 497-2002 IEEE Std 518-1982 IEEE Std 610 12-1990 IEEE Std 665-1995 IEEE Std 666-1991 IEEE Std 729-1983 IEEE Std. 828-1990 IEEE Std. 830-1993 IEEE Std. 1008-1987 IEEE Std 1012-1998 IEEE Std 1028-1997 IEEE Std 1050-1996 IEEE Std 1074-1995 IEEE Std. C62 23-1995 IEEE Std. C62 41-1991 IEEE Std. C62.45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std 1042-1987 MIL-STD-461E	Yes Chapter 7 will conform to current regulatory guidance.	No	Chapters 1 through 6, 8 through 10, and 12 through 19	Assess & apply any I&C lessons learned from Fukushima Daichi power station. Also address: 1. Compliance with IEEE 603-1991. 2. Development of defense-in-depth and diversity technical report for the PSAR 3 See Sections 7.4, 7.8 and 7.9 4. RTNSS issues 5 DAC/ITAAC for Digital I&C systems (RG 1.70 does not recognize RTNSS, DAC or ITAAC) 6. Digital I&C Software Quality Assurance (RG 1.152), Diversity, and Cyber-Security (RG 5.71) in the PSAR RFDs, and what is required for the CPA. The I&C design will be informed by the latest issued version of the IEEE Standards and ISGs Commitments to the NRC, however, will be based upon guidance endorsed in RGs, BTPs and SRP Sections issued 6 months prior to the submittal of the PSAR. A Cyber Security Plan Report will be submitted separately. As part of the standard plant design, an alternate review process to BTP 7-14 will be developed and pursued with the NRC for the mPower digital I&C systems software quality assurance program. The approach will be that B&W will comply with BTP 7-14, or with an alternate process deemed acceptable to the NRC (in this case, the software development process described in the capability maturity model and integration (CMMI) developed and assessed by the Software Engineering Institute). It is recognized that implementation of an alternate approach to BTP 7-14 would require NRC concurrence and potential revisions to the SRP.

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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
	DCD	10 CFR 50 or 10 CFR Part 52 10 CFR 50 App. A GDC 1,2,4, 10, 13, 15, 16, 19-24, 29, 33-35, 38, 41, 44 10 CFR 50 55a(a)(1) 10 CFR 50.55a(h) 10 CFR 50.55a(h)(2) 10 CFR 50.34(f) 10 CFR 50 34(f)(2)(v) 10 CFR 50 34(f)(2)(xii) 10 CFR 50.34(f)(2)(xiv) 10 CFR 52 47(b)(1)	No	RG 1.206	7.3 BTP 7-3, BTP 7-4, BTP 7-6, BTP 7-8, BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1.22, RG 1.47, RG 1.53, RG 1.62, RG 1.75, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 SRM on SECY-93-087 NUREG/CR-6303 DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	IEEE Std 379-2000 IEEE Std 603-1991 IEEE Std 7-4.3.2-2003 IEEE Std. 323-2003 IEEE Std. 338-1987 IEEE Std 384-1992 IEEE Std 473-1985 IEEE Std 497-2002 IEEE Std 518-1982 IEEE Std 610.12-1990 IEEE Std 665-1995 IEEE Std 666-1991 IEEE Std. 729-1983 IEEE Std. 828-1990 IEEE Std. 830-1993 IEEE Std 1008-1987 IEEE Std 1012-1998 IEEE Std 1028-1997 IEEE Std 1050-1996 IEEE Std 1074-1995 IEEE Std C62.23-1995 IEEE Std C62.41-1991 IEEE Std C62.45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std 1042-1987 MIL-STD-461E	N/A	N/A	Chapters 1 through 6, 8 through 10, and 12 through 19	See above
	FSAR	10 CFR 50 10 CFR 50 App. A GDC 1,2,4, 10, 13, 15, 16, 19-24, 29, 33-35, 38, 41, 44 10 CFR 50.55a(a)(1) 10 CFR 50 55a(h) 10 CFR 50 55a(h)(2) 10 CFR 50 34(f) 10 CFR 50 34(f)(2)(v) 10 CFR 50.34(f)(2)(xii) 10 CFR 50.34(f)(2)(xiv)	No	RG 1.206	7.3 BTP 7-3, BTP 7-4, BTP 7-6, BTP 7-8, BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1.22, RG 1.47, RG 1.53, RG 1.62, RG 1.75, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 SRM on SECY-93-087 NUREG/CR-6303 DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	Same as DCD	N/A	No	Chapters 1 through 6, 8 through 10, and 12 through 19	See above

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Regulatory Framework Document
NRC Version

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1 70	Changes to the Standard Plant Design	Related Sections	Key Issues
7 4 Systems Required for Safe Shutdown	PSAR	10 CFR Part 50 10 CFR 50, App. A, GDC 1, 2, 4, 13, 19, 24, 34, 35, and 38 10 CFR 50 55a(h) 10 CFR 50 55a(a)(1)	No	RG 1 206	7.4 BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-21	RG 1.53, RG 1.75, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1 172, RG 1.173, RG 1 180, RG 1.189, RG 1.204 DI&C-ISG-2	IEEE Std 603-1991 IEEE Std 7-4.3.2-2003 IEEE Std 338-1987 IEEE Std. 384-1992 IEEE Std. 473-1985 IEEE Std. 518-1982 IEEE Std. 610.12-1990 IEEE Std. 665-1995 IEEE Std 666-1991 IEEE Std 729-1983 IEEE Std. 828-1990 IEEE Std. 830-1993 IEEE Std 1008-1987 IEEE Std 1012-1998 IEEE Std 1028-1997 IEEE Std. 1050-1996 IEEE Std 1074-1995 IEEE Std C62.23-1995 IEEE Std C62.41-1991 IEEE Std C62.45-1992 ANSI/IEEE Std 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std. 1042-1987 IEEEStd. 242-2001 IEEEStd. 383-2003 IEEEStd 634-2004 IEEEStd 690-2004 IEEEStd 835-1994 IEEEStd 1202-2006 ANSI/IEEE Std C2-2007 MIL-STD-461E	Yes. Chapter 7 will conform to current regulatory guidance	No	Chapters 1 through 6, 8 through 10, and 12 through 19	Assess & apply any I&C lessons learned from Fukushima Daiichi power station. Also address: 1. Compliance with IEEE 603-1991. 2. Development of defense-in-depth and diversity technical report for the PSAR. 3 See Sections 7 4, 7.8 and 7.9 4. RTNSS issues 5. DAC/ITAAC for Digital I&C systems (RG 1.70 does not recognize RTNSS, DAC or ITAAC) 6 Digital I&C Software Quality Assurance (RG 1.152), Diversity, and Cyber-Security (RG 5 71) in the PSAR RFDs, and what is required for the CPA. The I&C design will be informed by the latest issued version of the IEEE Standards and ISGs Commitments to the NRC, however, will be based upon guidance endorsed in RGs, BTPs and SRP Sections issued 6 months prior to the submittal of the PSAR. A Cyber Security Plan Report will be submitted separately As part of the standard plant design, an alternate review process to BTP 7-14 will be developed and pursued with the NRC for the mPower digital I&C systems software quality assurance program. The approach will be that B&W will comply with BTP 7-14, or with an alternate process deemed acceptable to the NRC (in this case, the software development process described in the capability maturity model and integration (CMMI) developed and assessed by the Software Engineering Institute) It is recognized that implementation of an alternate approach to BTP 7-14 would require NRC concurrence and potential revisions to the SRP.

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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
	DCD	10 CFR Part 50 or 10 CFR Part 52 10 CFR 50, App. A, GDC 1, 2, 4, 13, 19, 24, 34, 35, and 38 10 CFR 50 55a(h) 10 CFR 50 34(f)(2)(xx) 10 CFR 52 47(b)(1) 10 CFR 50.55a(a)(1)	No	RG 1.206	7.4 BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-21	RG 1 53, RG 1.75, RG 1.105, RG 1.118, RG 1 151, RG 1.152, RG 1 168, RG 1 169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.189, RG 1 204 DI&C-ISG-2	IEEE Std 603-1991 IEEE Std 7-4 3 2-2003 IEEE Std 338-1987 IEEE Std. 384-1992 IEEE Std. 473-1985 IEEE Std. 518-1982 IEEE Std. 610.12-1990 IEEE Std 665-1995 IEEE Std. 666-1991 IEEE Std 729-1983 IEEE Std. 828-1990 IEEE Std. 830-1993 IEEE Std 1008-1987 IEEE Std 1012-1998 IEEE Std 1028-1997 IEEE Std. 1050-1996 IEEE Std 1074-1995 IEEE Std C62.23-1995 IEEE Std C62 41-1991 IEEE Std. C62 45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std. 1042-1987 IEEEStd. 242-2001 IEEEStd. 383-2003 IEEEStd 634-2004 IEEEStd. 690-2004 IEEEStd 835-1994 IEEEStd. 1202-2006 ANSI/IEEE Std. C2-2007 MIL-STD-461E	N/A	N/A	Chapters 1 through 6, 8 through 10, and 12 through 19	See above
	FSAR	10 CFR Part 50 10 CFR 50, App A, GDC 1, 2, 4, 13, 19, 24, 34, 35, and 38 10 CFR 50 55a(h) 10 CFR 50 34(f)(2)(xx) 10 CFR 50 55a(a)(1)	No	RG 1.206	7.4 BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-21	RG 1 53, RG 1.75, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1 168, RG 1.169, RG 1 170, RG 1.171, RG 1 172, RG 1.173, RG 1 180, RG 1.189, RG 1.204 DI&C-ISG-2	Same as DCD	N/A	No	Chapters 1 through 6, 8 through 10, and 12 through 19	See above

Note
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1 70	Changes to the Standard Plant Design	Related Sections	Key Issues
7 5 Information Systems Important to Safety	PSAR	10 CFR Part 50 10 CFR 50, App. A, GDC 1, 2, 4, 13, 19, 24 10 CFR 50 55a(a)(1) 10 CFR 50 55a(h)	No	RG 1.206	7.5 BTP 7-10, BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-21	RG 1 47, RG 1 53, RG 1.75, RG 1.97, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168 RG 1.169, RG 1.170, RG 1.171, RG 1 172, RG 1.173, RG 1.180, RG 1.204, SRM to SCY 93-087	IEEE Std 497-2002 IEEE Std 603-1991 IEEE Std 7-4.3 2-2003 IEEE Std. 323-2003 IEEE Std. 338-1987 IEEE Std. 384-1992 IEEE Std. 473-1985 IEEE Std 518-1982 IEEE Std 610.12-1990 IEEE Std 665-1995 IEEE Std. 666-1991 IEEE Std 729-1983 IEEE Std 828-1990 IEEE Std 830-1993 IEEE Std 1008-1987 IEEE Std 1012-1998 IEEE Std 1028-1997 IEEE Std 1050-1996 IEEE Std 1074-1995 IEEE Std C62.23-1995 IEEE Std. C62.41-1991 IEEE Std C62 45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std 829-1983 ANSI/IEEE Std. 1042-1987 MIL-STD-461E	Yes Chapter 7 will conform to current regulatory guidance.	No	Chapters 1 through 6, 8 through 10, and 12 through 19	Assess & apply any I&C lessons learned from Fukushima Daiichi power station Also address 1. Compliance with IEEE 603-1991. 2. Development of defense-in-depth and diversity technical report for the PSAR. 3 See Sections 7.4, 7 8 and 7.9 4 RTNSS issues 5. DAC/ITAAC for Digital I&C systems (RG 1.70 does not recognize RTNSS, DAC or ITAAC) 6. Digital I&C Software Quality Assurance (RG 1.152), Diversity, and Cyber-Security (RG 5 71) in the PSAR RFDs, and what is required for the CPA The I&C design will be informed by the latest issued version of the IEEE Standards and ISGs. Commitments to the NRC, however, will be based upon guidance endorsed in RGs, BTPs and SRP Sections issued 6 months prior to the submittal of the PSAR. A Cyber Security Plan Report will be submitted separately As part of the standard plant design, an alternate review process to BTP 7-14 will be developed and pursued with the NRC for the mPower digital I&C systems software quality assurance program. The approach will be that B&W will comply with BTP 7-14, or with an alternate process deemed acceptable to the NRC (in this case, the software development process described in the capability maturity model and integration (CMMI) developed and assessed by the Software Engineering Institute). It is recognized that implementation of an alternate approach to BTP 7-14 would require NRC concurrence and potential revisions to the SRP.

Note.
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
	DCD	10 CFR Part 50 or 10 CFR Part 52 10 CFR 50, App A, GDC 1, 2, 4, 13, 19, 24 10 CFR 50.34(b) 10 CFR 50 34(f)(2)(v) 50.34(f)(2)(xi) 50 34(f)(2)(xii) 50 34(f)(2)(xvii) 50.34(f)(2)(xviii) 50.34(f)(2)(xix) 50.34(f)(2)(xx) 10 CFR 50 49 10 CFR 50 55a(a)(1) 10 CFR 50.55a(h) 10 CFR 52.47(b)(1)	No	RG 1 206	7.5 BTP 7-10, BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-21	RG 1.47, RG 1.53, RG 1.75, RG 1.97, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168 RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204, SRM to SCY 93-087	IEEE Std 497-2002 IEEE Std 603-1991 IEEE Std 7-4 3.2-2003 IEEE Std. 323-2003 IEEE Std. 338-1987 IEEE Std. 384-1992 IEEE Std. 473-1985 IEEE Std 518-1982 IEEE Std 610.12-1990 IEEE Std 665-1995 IEEE Std 666-1991 IEEE Std. 729-1983 IEEE Std 828-1990 IEEE Std 830-1993 IEEE Std 1008-1987 IEEE Std 1012-1998 IEEE Std. 1028-1997 IEEE Std 1050-1996 IEEE Std 1074-1995 IEEE Std. C62 23-1995 IEEE Std C62.41-1991 IEEE Std C62.45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std 1042-1987 MIL-STD-461E	N/A	N/A	Chapters 1 through 6, 8 through 10, and 12 through 19	See above
	FSAR	10 CFR Part 50 10 CFR 50, App. A. GDC 1, 2, 4, 13, 19, 24 10 CFR 50 34(b) 10 CFR 50 34(f)(2)(v) 50 34(f)(2)(xi) 50 34(f)(2)(xii) 50.34(f)(2)(xvii) 50 34(f)(2)(xviii) 50.34(f)(2)(xix) 50.34(f)(2)(xx) 10 CFR 50 49 10 CFR 50 55a(a)(1) 10 CFR 50 55a(h)	No	RG 1.206	7.5 BTP 7-10, BTP 7-11, BTP 7-12, BTP 7-13, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-21	RG 1 47, RG 1.53, RG 1.75, RG 1.97, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168 RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1 204, SRM to SCY 93-087	Same as DCD	N/A	No	Chapters 1 through 6, 8 through 10, and 12 through 19	See above

Note
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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
7.6 Interlock Systems Important to Safety	PSAR	10 CFR Part 50 10 CFR 50, App. A, GDC 1, 2, 4, 10, 13, 15, 16, 19, 24, 25, 28, 33, 34, 35, 38, 41, and 44 10 CFR 50 55a(a)(1) 10 CFR 50 55a(h)	No	RG 1.206	7 6 BTP 7-1, BTP 7-2, BTP 7-5, BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-21	RG 1.47, RG 1.53, RG 1.75, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 DI&C-ISG-04	IEEE Std 603-1991 IEEE Std 7-4 3 2-2003 IEEE Std. 323-2003 IEEE Std. 338-1987 IEEE Std. 384-1992 IEEE Std 473-1985 IEEE Std 518-1982 IEEE Std 610.12-1990 IEEE Std 665-1995 IEEE Std 666-1991 IEEE Std. 729-1983 IEEE Std. 828-1990 IEEE Std 830-1993 IEEE Std. 1008-1987 IEEE Std. 1012-1998 IEEE Std. 1028-1997 IEEE Std. 1050-1996 IEEE Std 1074-1995 IEEE Std. C62 23-1995 IEEE Std. C62.41-1991 IEEE Std C62.45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std. 1042-1987 MIL-STD-461E	Yes Chapter 7 will conform to current regulatory guidance	No	Chapters 1 through 6, 8 through 10, and 12 through 19	Assess & apply any I&C lessons learned from Fukushima Daiichi power station Also address 1. Compliance with IEEE 603-1991 2. Development of defense-in-depth and diversity technical report for the PSAR. 3 See Sections 7 4, 7 8 and 7 9 4. RTNSS issues 5 DAC/ITAAC for Digital I&C systems (RG 1.70 does not recognize RTNSS, DAC or ITAAC). 6 Digital I&C Software Quality Assurance (RG 1.152), Diversity, and Cyber-Security (RG 5.71) in the PSAR RFDs, and what is required for the CPA See above plus 7. Safe Shutdown There are several differences in the definition of safe shutdown for passive plants and what systems (non-safety or safety) are credited and this will have to be reconciled (SECY 94-084). The I&C design will be informed by the latest issued version of the IEEE Standards and ISGs Commitments to the NRC, however, will be based upon guidance endorsed in RGs, BTPs and SRP Sections issued 6 months prior to the submittal of the PSAR. A Cyber Security Plan Report will be submitted separately. As part of the standard plant design, an alternate review process to BTP 7-14 will be developed and pursued with the NRC for the mPower digital I&C systems software quality assurance program The approach will be that B&W will comply with BTP 7-14, or with an alternate process deemed acceptable to the NRC (in this case, the software development process described in the capability maturity model and integration (CMMI) developed and assessed by the Software Engineering Institute) It is recognized that implementation of an alternate approach to BTP 7-14 would require NRC concurrence and potential revisions to the SRP.

Note:
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1 70	Changes to the Standard Plant Design	Related Sections	Key Issues
	DCD	10 CFR Part 50 or 10 CFR Part 52 10 CFR 50. App. A. GDC 1, 2, 4, 10, 13, 15, 16, 19, 24, 25, 28, 33, 34, 35, 38, 41, and 44 10 CFR 50 55a(a)(1) 10 CFR 50.55a(h) 10 CFR 50.34(f)(2)(v) 10 CFR 52.47(b)(1)	No	RG 1 206	7.6 BTP 7-1, BTP 7-2, BTP 7-5, BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-21	RG 1 47, RG 1.53, RG 1.75, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 DI&C-ISG-4	IEEE Std 603-1991 IEEE Std 7-4 3 2-2003 IEEE Std 323-2003 IEEE Std 338-1987 IEEE Std 384-1992 IEEE Std 473-1985 IEEE Std. 518-1982 IEEE Std. 610 12-1990 IEEE Std 665-1995 IEEE Std 666-1991 IEEE Std. 729-1983 IEEE Std. 828-1990 IEEE Std. 830-1993 IEEE Std. 1008-1987 IEEE Std. 1012-1998 IEEE Std. 1028-1997 IEEE Std 1050-1996 IEEE Std 1074-1995 IEEE Std C62 23-1995 IEEE Std C62 41-1991 IEEE Std C62 45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std. 1042-1987 MIL-STD-461E	N/A	N/A	Chapters 1 through 6, 8 through 10, and 12 through 19	See above
	FSAR	10 CFR Part 50 10 CFR 50, App A, GDC 1, 2, 4, 10, 13, 15, 16, 19, 24, 25, 28, 33, 34, 35, 38, 41, and 44 10 CFR 50 55a(a)(1) 10 CFR 50 55a(h) 10 CFR 50 34(f)(2)(v)	No	RG 1 206	7 6 BTP 7-1, BTP 7-2, BTP 7-5, BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-21	RG 1.47, RG 1.53, RG 1.75, RG 1 105, RG 1.118, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1 180, RG 1.204 DI&C-ISG-4	Same as DCD	N/A	No	Chapters 1 through 6, 8 through 10, and 12 through 19	See above

Note.
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
7.7 Control Systems Not Required for Safety	PSAR	10 CFR Part 50 10 CFR 50.55a(a)(1) 10 CFR 50.55a(h) 10 CFR 50, App. A, GDC 1, 10, 13, 15, 19, 24, 28, 29, 44	No	RG 1.206	7.7 BTP 7-5, BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1.75, RG 1.105, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 SECY 93-087 and SRM dated 7/15/93 DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	IEEE Std 603-1991 IEEE 7-4 3.2-2003 IEEE Std. 384-1992 IEEE Std 473-1985 IEEE Std 518-1982 IEEE Std. 610.12-1990 IEEE Std. 665-1995 IEEE Std 666-1991 IEEE Std 729-1983 IEEE Std 828-1990 IEEE Std. 830-1993 IEEE Std. 1008-1987 IEEE Std 1012-1998 IEEE Std 1028-1997 IEEE Std 1050-1996 IEEE Std. 1074-1995 IEEE Std. C62.23-1995 IEEE Std. C62.41-1991 IEEE Std C62 45-1992 ANSI/IEEE Std 622-1987 ANSI/IEEE Std 829-1983 ANSI/IEEE Std 1042-1987 MIL-STD-461E	Yes. Chapter 7 will conform to current regulatory guidance.	No	Chapters 1 through 6, 8 through 10, and 12 through 19	Assess & apply any I&C lessons learned from Fukushima Daiichi power station. Also address: 1. Compliance with IEEE 603-1991. 2. Development of defense-in-depth and diversity technical report for the PSAR. 3 See Sections 7.4, 7.8 and 7.9 4. RTNSS issues 5 DAC/ITAAC for Digital I&C systems (RG 1.70 does not recognize RTNSS, DAC or ITAAC) 6. Digital I&C Software Quality Assurance (RG 1.152), Diversity, and Cyber-Security (RG 5 71) in the PSAR RFDs, and what is required for the CPA. The I&C design will be informed by the latest issued version of the IEEE Standards and ISGs. Commitments to the NRC, however, will be based upon guidance endorsed in RGs, BTPs and SRP Sections issued 6 months prior to the submittal of the PSAR. A Cyber Security Plan Report will be submitted separately. As part of the standard plant design, an alternate review process to BTP 7-14 will be developed and pursued with the NRC for the mPower digital I&C systems software quality assurance program. The approach will be that B&W will comply with BTP 7-14, or with an alternate process deemed acceptable to the NRC (in this case, the software development process described in the capability maturity model and integration (CMMI) developed and assessed by the Software Engineering Insitute). It is recognized that implementation of an alternate approach to BTP 7-14 would require NRC concurrence and potential revisions to the SRP.

Note.
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
	DCD	10 CFR Part 50 or 10 CFR Part 52 10 CFR 50.55a(a)(1) 10 CFR 50.55a(h) 10 CFR 50.34(f) 10 CFR 50.34(f)(2)(xxii) 10 CFR 52.47(b)(1) 10 CFR 50, App. A, GDC 1, 10, 13, 15, 19, 24, 28, 29, 44	No	RG 1.206	7.7 BTP 7-5, BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1.75, RG 1.105, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 SECY 93-087 and SRM dated 7/15/93 DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	IEEE Std 603-1991 IEEE 7-4.3.2-2003 IEEE Std. 384-1992 IEEE Std. 473-1985 IEEE Std 518-1982 IEEE Std. 610.12-1990 IEEE Std. 665-1995 IEEE Std. 666-1991 IEEE Std. 729-1983 IEEE Std. 828-1990 IEEE Std. 830-1993 IEEE Std 1008-1987 IEEE Std. 1012-1998 IEEE Std 1028-1997 IEEE Std. 1050-1996 IEEE Std 1074-1995 IEEE Std C62.23-1995 IEEE Std. C62.41-1991 IEEE Std C62.45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std. 1042-1987 MIL-STD-461E	N/A	N/A	Chapters 1 through 6, 8 through 10, and 12 through 19	See above
	FSAR	10 CFR Part 50 10 CFR 50.55a(a)(1) 10 CFR 50.55a(h) 10 CFR 50.34(f) 10 CFR 50.34(f)(2)(xxii) 10 CFR 50, App. A, GDC 1, 10, 13, 15, 19, 24, 28, 29, 44	No	RG 1.206	7.7 BTP 7-5, BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1.75, RG 1.105, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 SECY 93-087 and SRM dated 7/15/93 DI&C-ISG-1 DI&C-ISG-2 DI&C-ISG-4	Same as DCD	N/A	No	Chapters 1 through 6, 8 through 10, and 12 through 19	See above

Note
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
7.8 Diverse Instrumentation and Control Systems	PSAR	10 CFR Part 50 10 CFR 50, App. A, GDC 1,13, 19, 24 10 CFR 50.62 10 CFR 50.55a(a)(1) 10 CFR 50.55a(h)	No	RG 1.206	7.8 BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1.22, RG 1.62, RG 1.75, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 SECY 93-087 and SRM dated 7/15/93 DI&C-ISG-1 DI&C-ISG-4 GL 85-06	IEEE Std 603-1991 IEEE 7-4.3.2-2003 IEEE Std. 323-2003 IEEE Std. 338-1987 IEEE Std. 384-1992 IEEE Std 473-1985 IEEE Std 497-2002 IEEE Std. 518-1982 IEEE Std 610.12-1990 IEEE Std 665-1995 IEEE Std 666-1991 IEEE Std. 729-1983 IEEE Std 828-1990 IEEE Std. 830-1993 IEEE Std. 1008-1987 IEEE Std. 1012-1998 IEEE Std 1028-1997 IEEE Std 1050-1996 IEEE Std. 1074-1995 IEEE Std. C62.23-1995 IEEE Std C62.41-1991 IEEE Std C62.45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std. 1042-1987 MIL-STD-461E	Yes. Chapter 7 will conform to current regulatory guidance	No	Chapters 1 through 6, 8 through 10, and 12 through 19	Assess & apply any I&C lessons learned from Fukushima Daiichi power station. Also address: 1. Compliance with IEEE 603-1991. 2. Development of defense-in-depth and diversity technical report for the PSAR 3. See Sections 7.4, 7.8 and 7.9 4. RTNSS issues 5. DAC/ITAAC for Digital I&C systems (RG 1.70 does not recognize RTNSS, DAC or ITAAC) 6. Digital I&C Software Quality Assurance (RG 1.152), Diversity, and Cyber-Security (RG 5.71) in the PSAR RFDs, and what is required for the CPA 7. DCD Section 7.8 (Diversity Systems) is not part of the scope required by RG 1.70 for the PSAR. 8. The mPower design does not have a auxiliary feedwater system. Therefore, compliance with 10 CFR 50.62 (c) needs to be addressed. The overall ATWS mitigation strategy and approach will be included The I&C design will be informed by the latest issued version of the IEEE Standards and ISGs Commitments to the NRC, however, will be based upon guidance endorsed in RGs, BTPs and SRP Sections issued 6 months prior to the submittal of the PSAR. A Cyber Security Plan Report will be submitted separately. As part of the standard plant design, an alternate review process to BTP 7-14 will be developed and pursued with the NRC for the mPower digital I&C systems software quality assurance program. The approach will be that B&W will comply with BTP 7-14, or with an alternate process deemed acceptable to the NRC (in this case, the software development process described in the capability maturity model and integration (CMMI) developed and assessed by the Software Engineering Institute). It is recognized that implementation of an alternate approach to BTP 7-14 would require NRC concurrence and potential revisions to the SRP.

Note:
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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
	DCD	10 CFR Part 50 or 10 CFR Part 52 10 CFR 50, App. A, GDC 1,13, 19, 24 10 CFR 50 62 10 CFR 50.55a(a)(1) 10 CFR 50 55a(h) 10 CFR 52 47(b)(1)	No	RG 1 206	7.8 BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1.22, RG 1 62, RG 1,75, RG 1.105, RG 1.118, RG 1.151, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 SECY 93-087 and SRM dated 7/15/93 DI&C-ISG-1 DI&C-ISG-4 GL 85-06	IEEE Std 603-1991 IEEE 7-4 3 2-2003 IEEE Std 323-2003 IEEE Std 338-1987 IEEE Std 384-1992 IEEE Std 473-1985 IEEE Std 497-2002 IEEE Std 518-1982 IEEE Std. 610.12-1990 IEEE Std. 665-1995 IEEE Std 666-1991 IEEE Std. 729-1983 IEEE Std 828-1990 IEEE Std. 830-1993 IEEE Std 1008-1987 IEEE Std 1012-1998 IEEE Std 1028-1997 IEEE Std. 1050-1996 IEEE Std 1074-1995 IEEE Std C62 23-1995 IEEE Std C62.41-1991 IEEE Std. C62 45-1992 ANSI/IEEE Std 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std 1042-1987 MIL-STD-461E	N/A	N/A	Chapters 1 through 6, 8 through 10, and 12 through 19	See above
	FSAR	10 CFR Part 50 10 CFR 50, App. A, GDC 1,13, 19, 24 10 CFR 50 62 10 CFR 50 55a(a)(1) 10 CFR 50 55a(h)	No	RG 1 206	7.8 BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	RG 1.22, RG 1.62, RG 1.75, RG 1.105, RG 1.118, RG 1.151, RG 1 152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1 204 SECY 93-087 and SRM dated 7/15/93 DI&C-ISG-1 DI&C-ISG-4 GL 85-06	Same as DCD	N/A	No	Chapters 1 through 6, 8 through 10, and 12 through 19	See above

Note
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1 70	Changes to the Standard Plant Design	Related Sections	Key Issues
7.9 Data Communication Systems	PSAR	10 CFR Part 50 10 CFR 50, App. A, GDC 1, 2, 4, 13, 19, 21, 22, 23, 24, 29 10 CFR 50 55a(a)(1) 10 CFR 50 55a(h) 10 CFR 50 55a(h)(2)	No	RG 1 206	7.9 BTP 7-8, BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	NUREG/CR-6082 RG 1 22, RG 1 47, RG 1 53, RG 1.75, RG 1.105, RG 1.118, RG 1 152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1 204 SECY-93-087 and SRM dated July 15, 1993 DI&C-ISG-2 DI&C-ISG-4	IEEE Std 379-2000 IEEE Std 603-1991 IEEE Std 7-4 3 2-2003 IEEE Std. 323-2003 IEEE Std. 338-1987 IEEE Std 384-1992 IEEE Std 473-1985 IEEE Std 518-1982 IEEE Std 610.12-1990 IEEE Std 665-1995 IEEE Std 666-1991 IEEE Std 729-1983 IEEE Std. 828-1990 IEEE Std. 830-1993 IEEE Std 1008-1987 IEEE Std 1012-1998 IEEE Std 1028-1997 IEEE Std 1050-1996 IEEE Std. 1074-1995 IEEE Std. C62.23-1995 IEEE Std C62 41-1991 IEEE Std C62.45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std 829-1983 ANSI/IEEE Std. 1042-1987 MIL-STD-461E	Yes Chapter 7 will conform to current regulatory guidance	No	Chapters 1 through 6, 8 through 10, and 12 through 19	Assess & apply any I&C lessons learned from Fukushima Daiichi power station. Also address: 1. Compliance with IEEE 603-1991 2. Development of defense-in-depth and diversity technical report for the PSAR 3. See Sections 7 4, 7 8 and 7.9 4. RTNSS issues 5 DAC/ITAAC for Digital I&C systems (RG 1 70 does not recognize RTNSS, DAC or ITAAC) 6. Digital I&C Software Quality Assurance (RG 1 152), Diversity, and Cyber-Security (RG 5.71) in the PSAR RFDs, and what is required for the CPA 7. DCD Section 7.9 (Data Communication Systems) is not part of the scope required by RG 1.70 for the PSAR The I&C design will be informed by the latest issued version of the IEEE Standards and ISGs. Commitments to the NRC, however, will be based upon guidance endorsed in RGs, BTPs and SRP Sections issued 6 months prior to the submittal of the PSAR. A Cyber Security Plan Report will be submitted separately. As part of the standard plant design, an alternate review process to BTP 7-14 will be developed and pursued with the NRC for the mPower digital I&C systems software quality assurance program. The approach will be that B&W will comply with BTP 7-14, or with an alternate process deemed acceptable to the NRC (in this case, the software development process described in the capability maturity model and integration (CMMI) developed and assessed by the Software Engineering Institute). It is recognized that implementation of an alternate approach to BTP 7-14 would require NRC concurrence and potential revisions to the SRP.

Note:
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
	DCD	10 CFR Part 50 or 10 CFR Part 52 10 CFR 50, App A, GDC 1, 2, 4, 13, 19, 21, 22, 23, 24, 29 10 CFR 50.55a(a)(1) 10 CFR 50 55a(h) 10 CFR 50.34(f) 10 CFR 50 34(f)(2)(v) 10 CFR 50.55a(h)(2) 10 CFR 52.47(b)(1)	No	RG 1.206	7.9 BTP 7-8, BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	NUREG/CR-6082 RG 1.22, RG 1.47, RG 1.53, RG 1.75, RG 1.105, RG 1.118, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 SECY-93-087 and SRM dated July 15, 1993 DI&C-ISG-2 DI&C-ISG-4	IEEE Std 379-2000 IEEE Std 603-1991 IEEE Std 7-4 3.2-2003 IEEE Std. 323-2003 IEEE Std. 338-1987 IEEE Std 384-1992 IEEE Std 473-1985 IEEE Std 518-1982 IEEE Std. 610.12-1990 IEEE Std. 665-1995 IEEE Std. 666-1991 IEEE Std 729-1983 IEEE Std. 828-1990 IEEE Std. 830-1993 IEEE Std 1008-1987 IEEE Std 1012-1998 IEEE Std 1028-1997 IEEE Std. 1050-1996 IEEE Std. 1074-1995 IEEE Std C62 23-1995 IEEE Std C62 41-1991 IEEE Std C62.45-1992 ANSI/IEEE Std. 622-1987 ANSI/IEEE Std. 829-1983 ANSI/IEEE Std. 1042-1987 MIL-STD-461E	N/A	N/A	Chapters 1 through 6, 8 through 10, and 12 through 19	See above
	FSAR	10 CFR Part 50 10 CFR 50, App. A. GDC 1, 2, 4, 13, 19, 21, 22, 23, 24, 29 10 CFR 50 55a(a)(1) 10 CFR 50 55a(h) 10 CFR 50 34(f) 10 CFR 50 34(f)(2)(v) 10 CFR 50.55a(h)(2)	No	RG 1.206	7.9 BTP 7-8, BTP 7-11, BTP 7-12, BTP 7-14, BTP 7-17, BTP 7-18, BTP 7-19, BTP 7-21	NUREG/CR-6082 RG 1.22, RG 1.47, RG 1.53, RG 1.75, RG 1.105, RG 1.118, RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, RG 1.173, RG 1.180, RG 1.204 SECY-93-087 and SRM dated July 15, 1993 DI&C-ISG-2 DI&C-ISG-4	Same as DCD	N/A	No	Chapters 1 through 6, 8 through 10, and 12 through 19	See above

Note.
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

CLINCH RIVER
REGULATORY FRAMEWORK DOCUMENTS
Chapter 7 Outline

7.1 Instrumentation and Controls - Introduction

7.1.1 Identification of Safety-Related Systems

PSAR The Clinch River PSAR Section 7.1.1 provides the following information regarding the Identification of Safety – Related Systems:

- a preliminary list all instrumentation, control, and supporting systems that are safety-related, including alarm, communication, and display instrumentation
- identification of, as applicable, the intended designers responsible for providing the I&C designs included for the facility
- a preliminary list of systems that are identical to those of a nuclear power plant of similar design that has recently received a COL license, design approval, or design certification (Address this by indicating that plant is based upon the standard mPower design)
- identification of systems that are expected to be different and discuss the differences and their effects on safety-related systems
- identification of any I&C that is credited in safety analysis but not intended to be safety-related (e.g., turbine trip sensors)
- identification of any exception to standards

DCD The mPower standard plant I&C design Section 7.1.1 provides the following information regarding the Identification of Safety-Related Systems:

- a list all instrumentation, control, and supporting systems that are safety-related, including alarm, communication, and display instrumentation
- identification of, as applicable, the designers responsible for providing the I&C designs included for the facility
- identification of a list of systems that are identical to those of a nuclear power plant of similar design that has recently received a COL license, design approval, or design certification
- identification of systems that are different and discuss the differences and their effects on safety-related systems
- identification of any I&C that is credited in safety analysis but not intended to be safety-related (e.g., turbine trip sensors)
- identification of any exception to standards
- identification of applicable DACs

FSAR Same contents as mPower standard plant DCD Section 7.1.1, and a commitment to close all applicable DACs prior to issuance of OL.

7.1.2 Identification of Safety Criteria

PSAR The Clinch River PSAR Section 7.1.2 provides the following preliminary information regarding the Identification of Safety Criteria:

- a regulatory requirements applicability matrix, for systems listed in Section 7.1.1, that includes, as available, information identified in NUREG-0800, SRP Chapter 7, Appendix 7.1-A for all known:
  - design bases,
  - criteria,
  - regulatory guides,
  - standards,
  - other documents to be implemented in the design.
- description of the technical design bases for all protection system functions, including the reactor trip function, ESF, emergency power, interlocks, bypasses, and equipment protection
- diversity requirements
- Note: The acceptance criteria and guidelines given in SRP Appendix 7.1-A are divided into four categories—
  - the regulations in 10 CFR 50.55a(h) including guidance in IEEE Std 603-1991,
  - the GDCs of Appendix A to 10 CFR Part 50,
  - regulatory guides (including endorsed industry codes and standards),
  - SRP Chapter 7 branch technical positions (BTPs) (10 CFR 50.34(h), conformance with the SRP).
- NUREGs identified in Section 7 SRP (e.g., NUREG/CR-6303) need to be reviewed for applicability.
- identification of any exception to standards

DCD The mPower standard plant I&C design Section 7.2 provides the following information regarding the Identification of Safety Criteria:

- a regulatory requirements applicability matrix, for systems listed in Section 7.1.1, that includes information identified in NUREG-0800, SRP Chapter 7, Appendix 7.1-A:
  - design bases,
  - criteria,
  - regulatory guides,
  - standards,
  - other documents to be implemented in the design
- description of the technical design bases for all protection system functions, including the reactor trip function, ESF, emergency power, interlocks, bypasses, and equipment protection
- diversity requirements

7.1.2 Identification of Safety Criteria (cont.)

DCD (cont.)

- Note: The acceptance criteria and guidelines given in SRP Appendix 7.1-A are divided into four categories
  - the regulations in 10 CFR 50.55a(h) including guidance in IEEE Std 603-1991,
  - the GDCs of Appendix A to 10 CFR Part 50,
  - regulatory guides (including endorsed industry codes and standards),
  - SRP Chapter 7 branch technical positions (BTPs) (10 CFR 50.34(h), conformance with the SRP).
- NUREGs identified in Section 7 SRP (e.g., NUREG/CR-6303) need to be reviewed for applicability.
- identification of any exception to standards
- identification of applicable DACs

FSAR Same contents as plant mPower standard plant DCD Section 7.1.2, and a commitment to close all applicable DACs prior to issuance of OL.

7.2 Reactor Trip System

PSAR The Clinch River PSAR Section 7.2 provides the following information regarding the Reactor Trip System:

7.2.1 Description

7.2.1.1 System Description:

- a description of the reactor trip system that includes:
  - initiating circuits
  - logic
  - bypasses
  - interlocks
  - redundancy
  - diversity
  - defense-in-depth design features
  - actuated devices
  - manual trip
  - response time
  - division independence
  - self diagnostics
  - control system interaction
  - testing features
- identification and description of any supporting systems
- identification of those parts of the reactor trip system that are not required for safety
- a description of how ATWS mitigation system interfaces with RTS
- a summary of BTP 7-19 analysis

7.2.1.2 Design Basis Information:

- a discussion of all topics listed in Appendix C.I.7-B to RG 1.206
- the following major design considerations are emphasized:
  - single-failure criterion
  - quality of components and modules
  - independence
  - defense in depth and diversity
  - system testing and inoperable surveillance
  - use of digital systems (guidance in SRP Chapter 7, Appendix 7.0-A)
  - setpoint determination
  - equipment qualification
- preliminary system drawings:
  - logic diagrams
  - block diagram
  - piping and instrumentation diagrams
  - location layout drawings of all reactor trip systems and supporting systems
- a commitment to supplement the application with final design drawings

7.2 Reactor Trip System (cont.)

PSAR (cont.)

7.2.2 Analysis

- analyses, including a failure mode and effects analysis (System level analysis at PSAR Stage)
- a discussion of how the requirements of the GDCs and IEEE Std 603-1991, IEEE Std 7-4.3.2-2003 and other applicable regulatory guides and appropriate criteria and standards are satisfied
- identification of any exception to standards
- considerations of instrumentation installed to prevent or mitigate the consequences of applicable Chapter 15 events
- a discussion of the need for and method of changing to more restrictive trip setpoints during abnormal operating conditions
- references, as appropriate, to other sections of the PSAR for discussions of supporting systems

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DCD

The mPower standard plant I&C design Section 7.2 provides the following information regarding the Reactor Trip System:

7.2.1 Description

7.2.1.1 System Description:

- a description of the reactor trip system that includes:
  - initiation circuits
  - logic
  - bypasses
  - interlocks
  - redundancy
  - diversity
  - defense-in-depth design features
  - actuated devices
  - manual trip
  - response time
  - division independence
  - self diagnostics
  - control system interaction
  - testing features
- identification and description of supporting systems
- identification of those parts of the reactor trip system that are not required for safety
- a description of how ATWS mitigation system interfaces with RTS
- a summary of BTP 7-19 analysis
- identification of applicable DACs

7.2 Reactor Trip System (cont.)

DCD (cont.)

7.2.1.2 Design Basis Information:

- a discussion of all topics listed in Appendix C.I.7-B to RG 1.206
- the following major design considerations are emphasized:
  - single-failure criterion
  - quality of components and modules
  - independence
  - defense in depth and diversity
  - system testing and inoperable surveillance
  - use of digital systems (guidance in SRP Chapter 7, Appendix 7.0-A)
  - setpoint determination
  - equipment qualification
- final system drawings:
  - logic diagrams
  - block diagram
  - piping and instrumentation diagrams
  - location layout drawings of all reactor trip systems and supporting systems

7.2.2 Analysis.

- analyses, including a failure mode and effects analysis (Component level analysis at DCD stage)
- a discussion of how the requirements of the GDCs and IEEE Std 603-1991, IEEE Std 7-4.3.2-2003 and other applicable regulatory guides and appropriate criteria and standards are satisfied
- identification of any exception to standards
- considerations of instrumentation installed to prevent or mitigate the consequences of applicable Chapter 15 events
- a discussion of the need for and method of changing to more restrictive trip setpoints during abnormal operating conditions
- references, as appropriate, to other sections of the DCD for discussions of supporting systems

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FSAR Same contents as mPower plant DCD Section 7.2 with the following supplemental information:

- updated diagrams and figures to reflect the final design
- a commitment to close of all applicable DACs prior to issuance of OL

7.3 Engineered Safety Features Systems

PSAR The Clinch River PSAR Section 7.3 provides the following information regarding the Engineered Safety Features Systems:

7.3.1 Description

7.3.1.1 System Description:

- a description of the I&Cs associated with the ESFs, including:
  - initiation circuits
  - logic
  - bypasses
  - interlocks
  - sequencing
  - redundancy
  - diversity
  - defense-in-depth design features
  - actuated devices
  - manual trip
  - response time
  - division independence
  - self diagnostics
  - control system interaction
  - testing features
- identification of system safety classification
- identification of RTNSS functions/equipment with reference to Chapter 19, Appendix 19B
- identification and description of any supporting systems
- identification of those parts of the ESF system not required for safety
- a description of how ATWS mitigation system interfaces with ESF
- a summary of BTP 7-19 analysis

7.3.1.2 Design Basis Information:

- a discussion of all topics listed in Appendix C.I.7-B to RG 1.206
- the following major design considerations are emphasized:
  - single-failure criterion
  - quality of components and modules
  - independence
  - defense-in-depth and diversity
  - system testing and inoperable surveillance
  - use of digital systems (guidance in SRP Chapter 7, Appendix 7.0-A)
  - setpoint determination
  - ESF control systems
  - equipment qualification
- preliminary system drawings for all ESF Systems and supporting systems:
  - logic diagrams
  - piping and instrumentation diagrams
  - location layout drawings

7.3 Engineered Safety Features Systems (cont.)

PSAR (cont.)

- a commitment to supplement the application, as necessary, with final design drawings

7.3.2 Analysis:

- analyses, including a failure mode and effects analysis (System level analysis at PSAR stage)
- a discussion of how the requirements of the GDCs and the guidance in IEEE Std 603-1991, and IEEE Std 7-4.3.2-2003, other applicable regulatory guides and other appropriate criteria and standards are satisfied
- these analyses include considerations of applicable Chapter 15 events
- identification of any exception to standards
- a discussion of the method for periodic testing of ESF I&C equipment and the effects on system integrity during testing

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DCD The mPower standard plant I&C design Section 7.3 provides the following information regarding the Identification of Engineered Safety Features Systems:

7.3.1 Description

7.3.1.1 System Description:

- a description of the I&Cs associated with the ESFs, including:
  - initiation circuits
  - logic
  - bypasses
  - interlocks
  - sequencing
  - redundancy
  - diversity
  - defense-in-depth design features
  - actuated devices
  - manual trip
  - response time
  - division independence
  - self diagnostics
  - control system interaction
  - testing features
- identification of system safety classification
- identification of RTNSS functions/equipment with reference to Chapter 19, Appendix 19B
- identification and description any supporting systems
- identification of those parts of the ESF system not required for safety
- a description of how ATWS mitigation system interfaces with ESF
- a summary of BTP 7-19 analysis
- identification of applicable DACs

7.3 Engineered Safety Features Systems (cont.)

DCD (cont.)

7.3.1.2 Design Basis Information:

- a discussion of all topics listed in Appendix C.I.7-B to RG 1.206
- the following major design considerations are emphasized:
  - single-failure criterion
  - quality of components and modules
  - independence
  - defense-in-depth and diversity
  - system testing and inoperable surveillance
  - use of digital systems (guidance in SRP Chapter 7, Appendix 7.0-A)
  - setpoint determination
  - ESF control systems
  - equipment qualification
- final system drawings for all ESF Systems and supporting systems:
  - logic diagrams
  - piping and instrumentation diagrams
  - location layout drawings

7.3.2 Analysis:

- analyses, including a failure mode and effects analysis (Component level analysis at DCD stage)
- a discussion of how the requirements of the GDCs and the guidance in IEEE Std 603-1991, and IEEE Std 7-4.3.2-2003, and other applicable regulatory guides and other appropriate criteria and standards are satisfied
- these analyses include considerations of applicable Chapter 15 events
- identification of any exception to standards
- a discussion of the method for periodic testing of ESF I&C equipment and the effects on system integrity during testing

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FSAR

Same contents as mPower plant DCD Section 7.3 with the following supplemental information:

- updated diagrams and figures for the final design
- a commitment to close all applicable DACs prior to issuance of OL

7.4 Systems Required for Safe Shutdown

PSAR The Clinch River PSAR Section 7.4 provides the following information regarding the Identification of Systems Required for Safe Shutdown:

7.4.1 Description:

- a description of the systems that are needed for safe shutdown of the plant, including:
  - initiating circuits
  - logic
  - bypasses
  - interlocks
  - redundancy
  - diversity
  - defense-in-depth design features
  - actuated devices
- a description of required controls and indications
- identification and description of any supporting systems
- a discussion of all topics listed in Appendix C.I.7-B to RG 1.206
- the following major design considerations are emphasized:
  - I&C systems required for safety shutdown
  - single-failure criterion
  - quality of components and modules
  - independence
  - periodic testing
  - use of digital systems (guidance in SRP Chapter 7, Appendix 7.0-A)
- locations of remote shutdown equipment
- transfer and isolation of remote shutdown equipment
- for remote shutdown capability:
  - description of the provisions taken in accordance with GDC 19, to provide the required equipment outside the control room to achieve and maintain hot and cold shutdown conditions
  - description of appropriate displays so that the operator can monitor the status of the shutdown
  - discussion of provisions for strict administrative controls limiting access to remote shutdown stations
  - operator response time to implement remote shutdown
- preliminary system drawings for all Safe shutdown systems and supporting systems:
  - logic diagrams
  - piping and instrumentation diagrams
  - location layout drawings
- a commitment to supplement the application, as necessary, with final design drawings.

7.4 Systems Required for Safe Shutdown (cont.)

PSAR (cont.)

7.4.2 Analysis:
- analyses that demonstrate how it has satisfied the requirements of the GDCs and 10 CFR 50.55a(h) (IEEE 603-1991 as applicable)
- justification for any deviation from meeting NRC regulations
- identification of any exception to standards
- considerations of instrumentation installed to permit a safe shutdown
- discussion of the need for and method of changing to more restrictive trip setpoints during abnormal operating conditions, such as operation with fewer than all reactor coolant loops operating
- reference, as appropriate, to other sections of the PSAR for supporting systems

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DCD

The mPower standard plant I&C design Section 7.4 provides the following information regarding the Identification of Systems Require for Safe Shutdown:

7.4.1 Description:
- a description of the systems that are needed for safe shutdown of the plant, including:
  - initiating circuits
  - logic
  - bypasses
  - interlocks
  - redundancy
  - diversity
  - defense-in-depth design features
  - actuated devices
- a description of required controls and indications
- identification and description of any supporting systems
- identification of applicable DACs
- a discussion of all topics listed in Appendix C.I.7-B to RG 1.206
- the following major design considerations are emphasized:
  - I&C systems required for safety shutdown
  - single-failure criterion
  - quality of components and modules
  - independence
  - periodic testing
  - use of digital systems (guidance in SRP Chapter 7, Appendix 7.0-A)
- locations of remote shutdown equipment
- transfer and isolation of remote shutdown equipment
- for remote shutdown capability:
  - description of the provisions taken in accordance with GDC 19, to provide the required equipment outside the control room to achieve and maintain hot and cold shutdown conditions

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7.4 Systems Required for Safe Shutdown (cont.)

DCD (cont.)

- description of appropriate displays so that the operator can monitor the status of the shutdown
- a discussion of provisions for strict administrative controls limiting access to remote shutdown stations
- operator response time to implement remote shutdown

- final system drawings for all safe shutdown systems and supporting systems:
  - logic diagrams
  - piping and instrumentation diagrams
  - location layout drawings

7.4.2 Analysis:

- analyses that demonstrate how it has satisfied the requirements of the GDCs and 10 CFR 50.55a(h) (IEEE 603-1991 as applicable)
- justification for any deviation from meeting NRC regulations
- identification of any exception to standards
- considerations of instrumentation installed to permit a safe shutdown
- discussion of the need for and method of changing to more restrictive trip setpoints during abnormal operating conditions, such as operation with fewer than all reactor coolant loops operating
- reference, as appropriate, to other sections of the DCD for supporting systems

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FSAR Same contents as mPower plant DCD Section 7.4 with the following supplemental information:

- updated diagrams and figures to reflect the final design
- a commitment to close all applicable DACs prior to issuance of OL

7.5 Information Systems Important to Safety

PSAR The Clinch River PSAR Section 7.5 provides the following information regarding the Information Systems Important to Safety:

7.5.1 Description:
- a description of the following instrumentation systems that provide information to enable the operator to perform required safety functions:
  - accident monitoring instrumentation (RG 1.97)
  - bypassed and inoperable status indication for safety systems (RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems")
  - plant annunciators (use of digital systems; see SRP Appendix 7.0-A)
- preliminary system drawings for all information systems important to safety:
  - logic diagrams
  - piping and instrumentation diagrams
  - location layout drawings
- a commitment to supplement the application with final design drawings
- reference to Chapter 18 for a discussion of the minimum inventory of human-system interfaces (i.e., alarms, controls, and displays)

7.5.2 Analysis:
- an analysis to demonstrate that the operator has:
  - sufficient information to perform required manual safety functions (e.g.)
    - safe control rod patterns
    - manual ESF operations
    - possible unanticipated post-accident operations
    - monitoring the status of safety equipment
  - sufficient time to make reasoned judgments and take action where operator action is essential for maintaining the plant in a safe condition
- appropriate safety criteria are identified and compliance with the criteria is demonstrated
- identification of any exception to standards
- information readouts and indications provided to the operator for monitoring conditions in the following are provided:
  - reactor
  - RCS
  - containment and safety-related process systems, including ESFs
- information available to the operator include all operating conditions of the plant, including:
  - Anticipated Operational Occurrences (AOO)
  - accident and post-accident conditions (including information from instrumentation that follows the course of accidents)

7.5 Information Systems Important to Safety (cont.)

PSAR (cont.)

- information includes:
  - design criteria
  - type of information to be displayed: the number of channels provided and their range, accuracy
  - location
  - discussion of the adequacy of the design
- range and accuracy is consistent with system requirements defined in the PSAR

DCD

The mPower standard plant I&C design Section 7. 5 provides the following information regarding Information Systems Important to Safety:

7.5.1 Description:

- a description of the following instrumentation systems that provide information to enable the operator to perform required safety functions:
  - accident monitoring instrumentation (RG 1.97)
  - bypassed and inoperable status indication for safety systems (RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems")
  - plant annunciators (use of digital systems; see SRP Appendix 7.0-A)
  - safety parameter displays (10 CFR 50.34, "Contents of applications; technical information," requirement related to TMI)
  - information systems associated with the emergency response facilities and nuclear data link (10 CFR 50.34 requirement related to TMI)
- reference to Chapter 18 for a discussion of the minimum inventory of human-system interfaces (i.e., alarms, controls, and displays)
- final system drawings for all information systems important to safety:
  - logic diagrams
  - piping and instrumentation diagrams
  - location layout drawings
- identification of applicable DACs

7.5.2 Analysis:

- an analysis to demonstrate that the operator has:
  - sufficient information to perform required manual safety functions (e.g.)
    - safe control rod patterns
    - manual ESF operations
    - possible unanticipated post-accident operations
    - monitoring the status of safety equipment
  - sufficient time to make reasoned judgments and take action where operator action is essential for maintaining the plant in a safe condition

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7.5 Information Systems Important to Safety (cont.)

DCD (cont.)

- appropriate safety criteria are identified and compliance with the criteria is demonstrated
- identification of any exception to standards
- information readouts and indications are provided to the operator for monitoring conditions in the following:
  - reactor
  - RCS
  - containment and safety-related process systems, including ESFs
- information available to the operator includes all operating conditions of the plant, as well as:
  - Anticipated Operational Occurrences (AOO)
  - accident and postaccident conditions (including information from instrumentation that follows the course of accidents)
- information includes:
  - design criteria
  - type of information to be displayed: the number of channels provided and their range, accuracy
  - location
  - discussion of the adequacy of the design
- range and accuracy is consistent with system requirements defined in the DCD

FSAR Same contents as mPower plant DCD Section 7.5 with the following supplemental information:

- updated diagrams and figures to reflect the final design
- any site-specific post-accident monitoring instrumentation
- a commitment to close all applicable DACs prior to issuance of OL

7.6 Interlock Systems Important to Safety

PSAR The Clinch River PSAR Section 7.6 provides the following information regarding the Interlock Systems Important to Safety:

- information describing all other instrumentation systems or any of their supporting systems required for safety that are not addressed in the following sections:
  - reactor trip system
  - ESF systems
  - safe shutdown systems
  - information system
- these other systems include:
  - interlock systems to prevent overpressurization of low-pressure systems when these systems are connected to high-pressure systems
  - interlocks to prevent overpressurizing the primary coolant system during low-temperature operations
  - interlocks to preclude inadvertent inter-ties between redundant or diverse safety systems for the purposes of testing or maintenance

7.6.1 Description:

- a description of all systems required for safety not already discussed in PSAR Sections 7.2 through 7.5, including:
  - Initiating circuits
  - logic
  - bypasses
  - interlocks
  - redundancy
  - diversity
  - defense-in-depth design features
  - actuated devices
- a description of the designed low power interlocks
- identification and description of any supporting systems reference to descriptions in other sections of the PSAR as appropriate
- design-basis information required by IEEE Std 603-1991 as applicable
- preliminary system drawings for interlock systems important to safety:
  - logic diagrams
  - piping and instrumentation diagrams
  - location layout drawings
- a commitment to supplement the application, as necessary, with final design drawings

7.6.2 Analysis:

- analyses which demonstrate how the applicable requirements of the GDC and IEEE Std 603-1991 have been satisfied and the extent to which they have satisfied applicable regulatory guides and other criteria and standards

7.6 Interlock Systems Important to Safety (cont.)

PSAR (cont.)

- identification of any exception to standards
- the following considerations are provided:
  - interlocks to prevent overpressurization of low-pressure systems
  - interlocks to prevent overpressurization of the primary coolant system during low-temperature operations of the reactor vessel
  - interlocks for ECCS accumulator valves
  - Interlocks required to preclude inadvertent inter-ties between redundant or diverse safety systems
- reference provided to other sections of the PSAR for supporting systems and analyses

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DCD

The mPower standard plant I&C design Section 7.6 provides the following information regarding the Interlock Systems Important to Safety:

- information describing all other instrumentation systems or any of their supporting systems required for safety that are not addressed in the following sections:
  - reactor trip system
  - ESF systems
  - safe shutdown systems
  - information system
- these other systems include:
  - interlock systems to prevent overpressurization of low-pressure systems when these systems are connected to high-pressure systems
  - interlocks to prevent overpressurizing the primary coolant system during low-temperature operations
  - interlocks to preclude inadvertent inter-ties between redundant or diverse safety systems for the purposes of testing or maintenance

7.6.1 Description:

- a description of all systems required for safety not already discussed in DCD Sections 7.2 through 7.5, including:
  - Initiating circuits
  - logic
  - bypasses
  - interlocks
  - redundancy
  - diversity
  - defense-in-depth design features
  - actuated devices
- a description of the designed low power interlocks
- identification and description of any supporting systems

7.6 Interlock Systems Important to Safety (cont.)

DCD (cont.)

- reference to descriptions in other sections of the DCD as appropriate
- design-basis information required by IEEE Std 603-1991 as applicable
- system drawings for interlock systems important to safety:
  - logic diagrams
  - piping and instrumentation diagrams
  - location layout drawings
- Identification of applicable DACs

7.6.2 Analysis:

- analyses which demonstrate how the applicable requirements of the GDC and IEEE Std 603-1991 have been satisfied and the extent to which they have satisfied applicable regulatory guides and other appropriate criteria and standards
- identification of any exception to standards
- the following considerations are provided:
  - interlocks to prevent overpressurization of low-pressure systems
  - interlocks to prevent overpressurization of the primary coolant system during low-temperature operations of the reactor vessel
  - interlocks for ECCS accumulator valves
  - interlocks required to preclude inadvertent inter-ties between redundant or diverse safety systems
- reference provided to other sections of the DCD for supporting systems and analyses.

FSAR Same contents as mPower plant DCD Section 7.6 with the following supplemental information:

- updated diagrams and figures to reflect the final design
- a commitment to close all applicable DACs prior to issuance of OL

7.7 Control Systems Not Required for Safety

PSAR The Clinch River PSAR Section 7.7 provides the following information regarding Control Systems Not Required for Safety (at high level at PSAR stage):

7.7.1 Description:

- descriptions of those control systems that can, through failure of normal operation, or inadvertent operation, affect the performance of critical safety functions
- an analysis confirming that the:
  - design of these control systems conforms to the acceptance criteria and guidelines
  - controlled variables can be maintained within prescribed operating ranges
  - effects of operation or failure of these systems are bounded by the accident analyses in Chapter 15 of the PSAR

7.7.2 Design-Basis Information:

- a discussion of the applicable topics identified in SRP Table 7-1
- the following major design considerations are emphasized:
  - design bases: The control systems include the necessary features for manual and automatic control of process variables within prescribed normal operating limits.
  - safety classification: The plant accident analysis in Chapter 15 of the PSAR does not rely on the operability of any control system function to assure safety.
  - effects of control system operation on accidents: The safety analysis considers the effects of both control system action and inaction in assessing the transient response of the plant for accidents and AOO.
  - effects of control system failures: The failure of any control system component or any auxiliary supporting system for control systems does not cause plant conditions more severe than those described in the analysis of AOO in Chapter 15 of the PSAR. The PSAR addresses failure modes that can be associated with digital systems such as software design errors and random hardware failures.
  - effects of control system failures caused by accidents: The consequential effects of AOO and accidents does not lead to control systems failures that would result in consequences more severe than those described in the analysis in Chapter 15 of the PSAR.
  - environmental control system: The I&C systems includes environmental controls as necessary to protect equipment from environmental extremes. This includes, for example, heat tracing for safety instruments and instrument sensing lines, as discussed in RG 1.151 and cabinet cooling fans.

7.7 Control Systems Not Required for Safety (cont.)

PSAR (cont.)

- use of digital systems: To minimize the potential for control system failures that could challenge safety systems, control system software has been developed using a structured process similar to that applied to safety system software. Elements of the process have been tailored to account for the lower safety significance of control system software.
- independence: The independence of safety system functions have been addressed.
- defense-in-depth and diversity: Control system elements credited in the defense-in-depth and diversity analyses have been addressed.
- potential for inadvertent actuation: Control system designs limit the potential for inadvertent actuation and challenges of safety system functions.
- control of access: Physical and electronic access to digital computer-based control system software and data to prevent changes by unauthorized personnel is controlled. Controls address access via network connections and via maintenance equipment.

- preliminary system drawings:
  - logic diagrams
  - piping and instrumentation diagrams
  - location layout drawings a commitment to supplement the application, as necessary, with final design drawings

7.7.3 Analysis:

- analyses provided which demonstrates that these systems are not required for safety
- demonstrates that the protection systems are capable of coping with all (including gross) failure modes of the control systems
- a description of failure analysis methodology and criteria for nonsafety-related systems, as applicable
- identification of any exception to standards

DCD The mPower standard plant I&C design Section 7.7 provides the following information regarding Control Systems Not Required for Safety:

7.7.1 Description:

- descriptions of those control systems that can, through failure of normal operation, or inadvertent operation, affect the performance of critical safety functions
- an analysis confirming that the:
  - design of these control systems conforms to the acceptance criteria and guidelines
  - controlled variables can be maintained within prescribed operating ranges

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7.7 Control Systems Not Required for Safety (cont.)

DCD (cont.)

- effects of operation or failure of these systems are bounded by the accident analyses in Chapter 15

7.7.2 Design-Basis Information:

- a discussion of the applicable topics identified in SRP Table 7-1
- The following major design considerations are emphasized:
  - design bases: The control systems should include the necessary features for manual and automatic control of process variables within prescribed normal operating limits.
  - safety classification: The plant accident analysis in Chapter 15 does not rely on the operability of any control system function to assure safety.
  - effects of control system operation on accidents: The safety analysis does not consider the effects of both control system action and inaction in assessing the transient response of the plant for accidents and AOO.
  - effects of control system failures: The failure of any control system component or any auxiliary supporting system for control systems does not cause plant conditions more severe than those described in the analysis of AOO in Chapter 15. The DCD addresses failure modes that can be associated with digital systems such as software design errors and random hardware failures.
  - effects of control system failures caused by accidents: The consequential effects of AOO and accidents does not lead to control systems failures that would result in consequences more severe than those described in the analysis in Chapter 15.
  - environmental control system: The I&C systems includes environmental controls as necessary to protect equipment from environmental extremes. This includes, for example, heat tracing for safety instruments and instrument sensing lines, as discussed in RG 1.151 and cabinet cooling fans.
  - use of digital systems: To minimize the potential for control system failures that could challenge safety systems, control system software has been developed using a structured process similar to that applied to safety system software. Elements of the process have been tailored to account for the lower safety significance of control system software.
  - independence: The independence of safety system functions have been addressed.
  - defense-in-depth and diversity: Control system elements credited in the defense-in-depth and diversity analyses have been addressed.
  - potential for inadvertent actuation: Control system designs limit the potential for inadvertent actuation and challenges of safety system functions.
  - control of access: Physical and electronic access to digital computer-based control system software and data to prevent changes by unauthorized personnel is controlled. Controls address access via network connections and via maintenance equipment.

7.7 Control Systems Not Required for Safety (cont.)

DCD (cont.)

- final system drawings:
  - logic diagrams
  - piping and instrumentation diagrams
  - location layout drawings

7.7.3 Analysis:

- analyses provided which demonstrate that these systems are not required for safety
- demonstrated that the protection systems are capable of coping with all (including gross) failure modes of the control systems
- demonstrated compliance with appropriate safety criteria
- a description of failure analysis methodology and criteria for nonsafety-related systems, as applicable
- identification of any exception to standards
- identification of applicable DACs

FSAR Same contents as mPower plant DCD Section 7.7 with the following supplemental information:

- updated diagrams and figures to reflect the final design
- a commitment to close all applicable DACs prior to issuance of OL

7.8 Diverse Instrumentation and Control Systems

PSAR The Clinch River Diverse Actuation System (DAS) is a non-safety I&C System that provides a diverse backup to the primary reactor protection and engineered safeguards features actuation systems. The primary purpose of the DAS is to mitigate the effects of a postulated common cause failure of the primary digital RPS/ESFAS to perform their required safety functions as discussed in the SRM on SECY-93-087, Item II.Q. The ATWS mitigation system logic and functions are contained in the DAS. The scope of the additional DAS functions is determined by performing a comprehensive diversity and defense-in-depth analysis in accordance with the requirements discussed in BTP 7-19. The Clinch River PSAR Section 7.8 provides the following information regarding Diverse Instrumentation and Control Systems:

7.8.1 System Description:

- the diverse I&C systems and include:
    - initiating circuits
    - logic
    - bypasses
    - interlocks
    - redundancy
    - diversity
    - defense-in-depth design features
    - actuated devices
- identification of system safety classification
- identification of RTNSS functions/equipment with reference to Chapter 19, Appendix 19B
- identification and description of any supporting systems
- reference to descriptions in other PSAR sections
- description provided for mitigation functions for ATWS and diverse manual controls and diverse display provisions addressed
- preliminary system drawings for all diverse I&C systems:
    - logic diagrams
    - piping and instrumentation diagrams
    - location layout drawings
- a commitment to supplement the application, as necessary, with final design drawings

7.8.2 Analysis:

- analyses (at system level) provided which demonstrate:
    - conformance of the proposed diverse I&C system with the requirements of 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants". Due to the uniqueness of the mPower design, it does not explicitly comply with the requirements of 10 CFR 50.62. However, the overall ATWS mitigation strategy and approach will be included.

7.8 Diverse Instrumentation and Control Systems (cont.)

PSAR (cont.)

- the adequacy of manual controls and displays supporting control room operator actions to place the nuclear plant in a hot shutdown condition and to perform reactivity control, core heat removal, reactor coolant inventory control, containment isolation, and containment integrity actions, and
- for plant designs using digital computer-based protection systems, the conformance of the proposed diverse I&C system with the guidance of SRP Chapter 7, BTP 7-19, "Guidance for Evaluation of Diversity and Defense-in- Depth in Digital Computer-Based Instrumentation and Control Systems," Revision 5, issued February 2007.

- identification of any exception to standards

DCD

The mPower standard plant Diverse Actuation System (DAS) is a non-safety I&C System that provides a diverse backup to the primary reactor protection and engineered safeguards features actuation systems. The primary purpose of the DAS is to mitigate the effects of a postulated common cause failure of the primary digital RPS/ESFAS to perform their required safety functions as discussed in the SRM on SECY-93-087, Item II.Q. The ATWS mitigation system logic and functions are contained in the DAS. The scope of the additional DAS functions is determined by performing a comprehensive diversity and defense-in-depth analysis in accordance with the requirements discussed in BTP 7-19. The mPower standard plant I&C design Section 7.8 provides the following information regarding Diverse Instrumentation and Control Systems:

7.8.1 System Description:

- the diverse I&C systems include:
  - initiating circuits
  - logic
  - bypasses
  - interlocks
  - redundancy
  - diversity
  - defense-in-depth design features
  - actuated devices
- identification of system safety classification
- identification of RTNSS functions/equipment with reference to Chapter 19, Appendix 19B
- identification and description of any supporting systems
- reference to descriptions in other DCD sections
- description provided for mitigation functions for ATWS and diverse manual controls and diverse display provisions addressed
- final system drawings for all diverse I&C systems:
  - logic diagrams
  - piping and instrumentation diagrams

7.8 Diverse Instrumentation and Control Systems (cont.)

DCD (cont.)

- location layout drawings

7.8.2 Analysis:

- a diversity analysis in accordance with BTP 7-19 to address and mitigate consequences of a common-cause failure
- analyses (at component level) provided which demonstrate:
  - conformance of the proposed diverse I&C system with the requirements of 10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,”. Due to the uniqueness of the mPower design, it does not explicitly comply with the requirements of 10 CFR 50.62. However, the overall ATWS mitigation strategy and approach will be included
  - the adequacy of manual controls and displays supporting control room operator actions to place the nuclear plant in a hot shutdown condition and to perform reactivity control, core heat removal, reactor coolant inventory control, containment isolation, and containment integrity actions, and
  - for plant designs using digital computer-based protection systems, the conformance of the proposed diverse I&C system with the guidance of SRP Chapter 7, BTP 7-19, “Guidance for Evaluation of Diversity and Defense-in- Depth in Digital Computer-Based Instrumentation and Control Systems,” Revision 5, issued February 2007.
- identification of any exception to standards
- identification of applicable DACs

FSAR Same contents as mPower plant DCD Section 7.8 with the following supplemental information:

- updated diagrams and figures to reflect the final design
- a commitment to close all applicable DACs prior to issuance of OL

7.9 Data Communication Systems

PSAR The Clinch River PSAR Section 7.9 provides the following information regarding Data Communication Systems:

7.9.1 System Description:
- all data communication systems (DCSs) (safety and nonsafety) that are part of or support the systems described in PSAR Sections 7.2 through 7.8
- the scope and depth of the system description varies according to the system's importance to safety
- communication between systems and communication between computers within a system are provided

7.9.2 Design Basis Information:
- the applicable criteria are addressed according to the importance to safety of the system
- the following major DCS design considerations are emphasized:
  - quality of components and modules
  - DCS software quality (see SRP Chapter 7, BTP 7-14, "Guidance on Software Reviews for Digital Computer-Based Instrumentation and Controls Systems," Revision 5, issued February 2007)
  - protocol selected for the DCS meets the supported systems performance requirements, which include the following:
    - real-time performance
    - system deterministic timing
    - time delays within the DCS
    - data rates
    - data bandwidths
    - interfaces with other DCSs
    - DCS test results commensurate with the system requirements
    - communication protocols
  - the potential hazards to the DCS are addressed, including inadvertent actuation, error recovery, self-testing, and surveillance testing
  - to control access, the DCS does not present an electronic path by which unauthorized personnel can change plant software or display erroneous status information to the operators
  - the appropriate channel assignments to individual communication subsystems are addressed to ensure that the assignments meet both redundancy and diversity requirements within the supported systems. (Note: The use of a DCS as a single path for multiple signals or data raises particular concerns regarding extensive consequential failure as the result of a single failure.)
  - independence based on requirements in IEEE Std 603-1991
  - protection system has been designed to fail into a safe state or a state demonstrated to be acceptable on some other defined basis (GDC 23)
  - system testing and surveillances are addressed

7.9 Data Communication Systems (cont.)

PSAR (cont.)

- the status of DCSs in the design of bypass and inoperable status indications are addressed
- data communication media does not present a fault propagation path for environmental effects so as to limit susceptibility to electromagnetic and radiofrequency interference, (e.g., high-energy electrical faults and lightning) from one redundant portion of a system to another, or from another system to a safety system
- defense-in-depth and diversity analyses address each potential failure mode
- design considers the exposure of DCSs to seismic hazards. (Note: If data communication or multiplexer equipment connected to the safety system is located in a nonseismic Category I structure, simultaneous seismic destruction or perturbation can affect the DCS equipment.)

- preliminary system drawings for the DCSs include:
  - layout drawings
  - network routing information
- a commitment to supplement the application, as necessary, with final design drawings

7.9.3 Analysis:

- Analyses provided which demonstrate:
  - DCS systems conform to the recommendations in the regulatory guides and industry codes and standards applicable to these systems, are in conformance with the guidance of GDC 1 and meet the requirements of 10 CFR 50.55a(a)(1)
  - operability of supporting data communication clearly affects the operability of supported I&C safety functions
  - means and criteria for determining if a function has failed as a result of communications failure are described
- identification of any exception to standards

DCD The mPower standard plant I&C design Section 7.9 provides the following information regarding Data Communication Systems:

7.9.1 System Description:

- DCSs (safety and nonsafety) that are part of or support the systems described in Sections 7.2 through 7.8
- scope and depth of the system description will vary according to the system's importance to safety
- communication between systems and communication between computers within a system are provided

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7.9 Data Communication Systems (cont.)

DCD (cont.)

7.9.2 Design Basis Information:

- applicable criteria are addressed according to the importance to safety of the system
- the following major DCS design considerations are emphasized:
  - quality of components and modules
  - DCS software quality (see SRP Chapter 7, BTP 7-14, "Guidance on Software Reviews for Digital Computer-Based Instrumentation and Controls Systems," Revision 5, issued February 2007)
  - protocol selected for the DCS meets the supported systems performance requirements, which include the following:
    - real-time performance
    - system deterministic timing
    - time delays within the DCS
    - data rates
    - data bandwidths
    - interfaces with other DCSs
    - DCS test results commensurate with the system requirements
    - communication protocols
  - the potential hazards to the DCS are addressed, including inadvertent actuation, error recovery, self-testing, and surveillance testing
  - to control access, the DCS does not present an electronic path by which unauthorized personnel can change plant software or display erroneous status information to the operators
  - the appropriate channel assignments to individual communication subsystems are addressed to ensure that the assignments meet both redundancy and diversity requirements within the supported systems (Note: The use of a DCS as a single path for multiple signals or data raises particular concerns regarding extensive consequential failure as the result of a single failure.)
  - independence based on requirements in IEEE Std 603-1991
  - protection system has been designed to fail into a safe state or a state demonstrated to be acceptable on some other defined basis (GDC 23)
  - system testing and surveillances are addressed
  - the status of DCSs in the design of bypass and inoperable status indications are addressed
  - data communication media does not present a fault propagation path for environmental effects so as to limit susceptibility to electromagnetic and radiofrequency interference, (e.g., high-energy electrical faults and lightning) from one redundant portion of a system to another, or from another system to a safety system
  - defense-in-depth and diversity analyses address each potential failure mode

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7.9 Data Communication Systems (cont.)

DCD (cont.)

  - design considers the exposure of DCSs to seismic hazards. If data communication or multiplexer equipment connected to the safety system is located in a nonseismic Category I structure, simultaneous seismic destruction or perturbation can affect the DCS equipment
- final system drawings for the DCSs include:
  - layout drawings
  - network routing information

7.9.3 Analysis:
- Analyses provided which demonstrate:
  - DCS systems conform to the recommendations in the regulatory guides and industry codes and standards applicable to these systems, are in conformance with the guidance of GDC 1 and meet the requirements of 10 CFR 50.55a(a)(1)
  - operability of supporting data communication clearly affects the operability of supported I&C safety functions
  - means and criteria for determining if a function has failed as a result of communications failure are described
- identification of any exception to standards
- identification of applicable DACs

FSAR Same contents as mPower plant DCD Section 7.9 with the following supplemental information:
- updated diagrams and figures to reflect the final design
- a commitment to close all applicable DACs prior to issuance of OL

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
8.1 Introduction	PSAR	See Sections 8 2, 8.3 1, 8 3 2, and 8 4	See Sections 8 2, 8 3.1, 8 3 2, and 8 4	RG 1.70	8.1	See Sections 8 2, 8 3 1. 8.3.2, and 8 4	See Sections 8 2, 8 3 1, 8.3 2, and 8 4	No	No	See Sections 8 2, 8.3 1, 8.3.2, and 8 4	8 2, 8.3.1, 8.3 2, 8 4
	DCD			RG 1 206	8 1			N/A	N/A		8 2, 8.3.1, 8 3.2, 8 4
	FSAR			RG 1.206	8.1			N/A	No		8 2, 8 3.1, 8 3 2, 8 4
8 2 Offsite Power System	PSAR	10 CFR 50, App A, GDC 2, 4, 5, 17, 18, 33, 34, 35, 38, 41, and 44 10 CFR 50.34(f)(2)(v) 10 CFR 50 63 10 CFR 50 65 (a)(4)	Yes - GDC 17	RG 1.70	8 2 BTP 8-3 BTP 8-6 BTP 8-8	RGs 1 32, 1.155, 1.160, 1.182, 1 204 GL 2006-02 GL 2007-01 NUREG-1784 NUREG-1793 NUREG/CR-7000 SECY-90-016, SECY-91-078, SECY-94-084, SECY-95-132, SECY-05-0227 IN 97-05, IN 98-02 IN 98-07, IN 2000-06 IN 2002-12, IN 2005-15 IN 2006-06, IN 2006-18 RIS 2000-24 RIS 2004-05	IEEE 80-2000, IEEE 242-2001, IEEE 308-2001 IEEE 338-1987, IEEE 399-1997, IEEE 605-2008 IEEE 665-1995, IEEE 666-1991, IEEE 693-2005 IEEE 741-1986, IEEE 741-2007, IEEE 765-2006, IEEE 837-2002 IEEE 998-1996, IEEE 1050-1996, IEEE 1100-1999 IEEE 1584-2002, IEEE C2-2007, IEEE C37.04-1999 IEEE C37.06-2009, IEEE C37.09-1999, IEEE C37.010-1999 IEEE C37.013-1997, IEEE C37.30-1997, IEEE C37.32-2002 IEEE C37.34-1994, IEEE C37.106-2003, IEEE C57.13-2008 IEEE C57.13.5-2009, IEEE C57 13.6-2005, IEEE C62.11-2005 IEEE C62.23-1995, IEEE C93.1-1999 NUMARC 8700, Rev 1 NUMARC 93-01, Rev 4 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	No	Yes - Switchyard voltage is different; 161KV versus 230KV	Similar to the AP1000 design - an exemption to the GDC 17 requirement for two physically independent offsite circuits for a passive reactor design is planned. Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena. Track NRC issuance of Rev. 1 to RG 1.93, Availability of Electric Power Sources (DG-1244)	4 6, 5.4, 6 2, 6.3, 6 5, 6.7, 7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 8 1, 8 3.1, 8 3.2, 8 4, 9 1, 9 2, 9.3, 9 4, 9.5, 10 4, 13 5, 14 2, 16, 17
	DCD	10 CFR 50, App. A, GDC 2, 4, 5, 17, 18, 33, 34, 35, 38, 41, and 44 10 CFR 50 34(f)(2)(v) 10 CFR 50 63 10 CFR 52.47(b)(1) 10 CFR 50 65 (a)(4)	Yes - GDC 17	RG 1.206	8 2 BTP 8-3 BTP 8-6 BTP 8-8	RGs 1.32, 1 155, 1.160, 1 182, 1 204 GL 2006-02 GL 2007-01 NUREG-1784 NUREG-1793 NUREG/CR-7000 SECY-90-016, SECY-91-078, SECY-94-084, SECY-95-132, SECY-05-0227 IN 97-05, IN 98-02 IN 98-07, IN 2000-06 IN 2002-12, IN 2005-15 IN 2006-06, IN 2006-18 RIS 2000-24 RIS 2004-05	IEEE 80-2000, IEEE 242-2001, IEEE 308-2001, IEEE 338-1987 IEEE 399-1997, IEEE 605-2008, IEEE 665-1995, IEEE 666-1991 IEEE 693-2005, IEEE 741-1986, IEEE 741-2007, IEEE 765-2006, IEEE 837-2002, IEEE 998-1996, IEEE 1050-1996, IEEE 1100-1999, IEEE 1584-2002, IEEE C2-2007, IEEE C37.04-1999, IEEE C37.06-2009, IEEE C37.09-1999, IEEE C37.010-1999, IEEE C37 013-1997 IEEE C37 30-1997, IEEE C37.32-2002 IEEE C37.34-1994, IEEE C37.106-2003 IEEE C57.13-2008, IEEE C57.13.5-2009 IEEE C57.13.6-2005, IEEE C62 11-2005 IEEE C62.23-1995, IEEE C93 1-1999 NUMARC 8700, Rev 1 NUMARC 93-01, Rev 4 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	N/A	N/A	Similar to the AP1000 design - an exemption to the GDC 17 requirement for two physically independent offsite circuits for a passive reactor design is planned Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena. Track NRC issuance of Rev. 1 to RG 1.93, Availability of Electric Power Sources (DG-1244)	4 6, 5 4, 6 2, 6 3, 6 5, 6 7, 7 2, 7 3, 7 4, 7.5, 7.6, 7.7, 8.1, 8 3.1, 8.3.2, 8 4, 9 1, 9 2, 9.3, 9.4, 9.5, 10 4, 13 5, 14 2, 14.3, 16, 17

Note:
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Clinch River Regulatory Framework Document
NRC Version

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
8 2 Offsite Power System	FSAR	10 CFR 50, App A, GDC 2,4, 5, 17, 18, 33, 34, 35, 38, 41, and 44 10 CFR 50 34(f)(2)(v) 10 CFR 50.63 10 CFR 50.65(a)(4)	Yes - GDC 17	RG 1 206	8 2 BTP 8-3 BTP 8-6 BTP 8-8	RGs 1.32, 1.155, 1 160, 1 182, 1 204 GL 2006-02 GL 2007-01 NUREG-1784 NUREG-1793 NUREG/CR-7000 SECY-90-016, SECY-91-078, SECY-94-084, SECY-95-132, SECY-05-0227 IN 97-05, IN 98-02 IN 98-07, IN 2000-06 IN 2002-12, IN 2005-15 IN 2006-06, IN 2006-18 RIS 2000-24 RIS 2004-05	IEEE 80-2000, IEEE 242-2001, IEEE 308-2001 IEEE 338-1987, IEEE 399-1997, IEEE 605-2008 IEEE 666-1991, IEEE 693-2005, IEEE 741-1986 IEEE 741-2007, IEEE 765-2006, IEEE 837-2002, IEEE 998-1996, IEEE 1050-1996, IEEE 1100-1999, IEEE 1584-2002, IEEE C2-2007 IEEE C37.04-1999, IEEE C37.06-2009 IEEE C37.09-1999, IEEE C37.010-1999 IEEE C37.013-1997, IEEE C37.30-1997 IEEE C37.32-2002, IEEE C37.34-1994 IEEE C37.106-2003, IEEE C57.13-2008 IEEE C57.13.5-2009, IEEE C57.13 6-2005 IEEE C62.11-2005, IEEE C62 23-1995 IEEE C93.1-1999 NUMARC 8700, Rev 1 NUMARC 93-01, Rev 4 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	N/A	Yes - Switchyard voltage is different; 161KV versus 230KV	Similar to the AP1000 design - an exemption to the GDC 17 requirement for two physically independent offsite circuits for a passive reactor design is planned. Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena. Track NRC issuance of Rev. 1 to RG 1.93, Availability of Electric Power Sources (DG-1244)	4.6, 5.4, 6.2, 6.3, 6.5, 6 7, 7 2, 7.3, 7 4, 7.5, 7 6, 7.7, 8.1, 8.3.1, 8.3.2, 8 4, 9.1, 9.2, 9 3, 9 4, 9.5, 10 4, 13 5, 14.2, 16, 17

Note
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Clinch River Regulatory Framework Document
NRC Version

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
8 3 1	PSAR	10 CFR 50, App. A, GDC 2, 4, 5, 17,18, 33, 34, 35, 38, 41, 44 and 50 10 CFR 50.34(f)(2)(v) 10 CFR 50.34(f)(2)(xiii) 10 CFR 50.34(f)(2)(xx) 10 CFR 50.55a(h) 10 CFR 50 63 10 CFR 50 65(a)(4)	No	RG 1.70	8.3 1 BTP 8-1 BTP 8-2 BTP 8-4 BTP 8-5 BTP 8-6 BTP 8-7 BTP 8-8	RGs 1.6, 1.9, 1.32, 1.47, 1 53, 1.63, 1.75, 1 81, 1.106, 1.118, 1.153, 1.155, 1.160, 1.182, 1.204 NUREG-0718 & NUREG-0737: I.D 3, II E 3.1, II G.1 NUREG-1784 NUREG-1793, NUREG-1801 NUREG/CR-0660 NUREG/CR-6866 NUREG/CR-7000 SECY 90-016 SECY 91-078 SECY 94-084 SECY 95-132 SECY 05-227 GL 1996-01, GL 2007-01 IN 98-38, IN 99-13 IN 00-06, IN 00-14 IN 02-01, IN 02-04 IN 02-12, IN 05-04 IN 06-18	IEEE 141-1993, IEEE 142-1991, IEEE 242-2001 IEEE 308-2001, IEEE 317-1983, IEEE 338-1987 IEEE 379-2000, IEEE 384-1992, IEEE 387-1995 IEEE 399-1997, IEEE 420-2001, IEEE 446-1995 IEEE 519-1992, IEEE 577-2004, IEEE 603-1991 (w/1995 correction sheet) IEEE 622-1987, IEEE 628-2001, IEEE 665-1995 IEEE 666-1991, IEEE 690-2004, IEEE 741-1986, IEEE 741-2007 IEEE 765-2006, IEEE 835-1994, IEEE 1050-1996 IEEE 1185-1994, IEEE 1584-2002, IEEE C37.06-2009 IEEE C37.013-1997, IEEE C37.13-2008 IEEE C37.16-2009, IEEE C37.17-1997 IEEE C37.20.1-2002, IEEE C37.20.2-1999 IEEE C37.20.3-2001, IEEE C37.23-2003 IEEE C37.46-2010, IEEE C37.47-2000 IEEE C37.90-2005, IEEE C37.91-2008 IEEE C37.96-2000, IEEE C37.101-1993 IEEE C37.102-2006, IEEE C37.106-2003 IEEE C37.121-1989, IEEE C50.13-2005 IEEE C57.12.00-2010, IEEE C57.12.01-2005 IEEE C57.12.10-2010, IEEE C57 12.22-1993 IEEE C57.12.55-1987, IEEE C57 12.51-2008 IEEE C57.116-1989, IEEE C62.11-2005, IEEE C62.23-1995 NUMARC 93-01 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	No	No	Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena. Track NRC issuance of RIS 2011-XX, Adequacy of Station Electric Distribution System Voltages Track NRC issuance of Rev. 1 to RG 1.93, Availability of Electric Power Sources (DG-1244) and Rev. 2 to RG 1.106, Thermal Overload Protection for Electric Motors on MOVs (DG-1264)	3 4, 3 5, 3 6, 3.11, 4.6, 5 4, 6 2, 6.3, 6 5, 6.7, 7.2, 7.3, 7.4, 7 5, 7.6, 7.7, 8.1, 8 2, 8 3 2, 8.4, 9.1, 9.2, 9.3, 9 4, 9.5, 10 4, 13.5, 14 2, 16, 17
AC Power Systems (onsite)	DCD	10 CFR 50, App. A, GDC 2, 4, 5, 17,18, 33, 34, 35, 38, 41, 44 and 50 10 CFR 50 34(f)(2)(v) 10 CFR 50 34(f)(2)(xiii) 10 CFR 50.34(f)(2)(xx) 10 CFR 50.55a(h) 10 CFR 50 63 10 CFR 50.65(a)(4)	No	RG 1 206	8 3.1 BTP 8-1 BTP 8-2 BTP 8-4 BTP 8-5 BTP 8-6 BTP 8-7 BTP 8-8	RGs 1.6, 1.9, 1.32, 1 47, 1.53, 1.63, 1.75, 1 81, 1.106, 1.118, 1.153, 1.155, 1.160, 1.182, 1 204 NUREG-0718 & NUREG-0737: I.D.3, II E 3 1, II G.1 NUREG-1784 NUREG-1793, NUREG-1801 NUREG/CR-0660 NUREG/CR-6866 NUREG/CR-7000 SECY 90-016 SECY 91-078 SECY 94-084 SECY 95-132 SECY 05-227 GL 1996-01, GL 2007-01 IN 98-38, IN 99-13 IN 00-06, IN 00-14 IN 02-01, IN 02-04 IN 02-12, IN 05-04 IN 06-18	IEEE 141-1993, IEEE 142-1991, IEEE 242-2001 IEEE 308-2001, IEEE 317-1983, IEEE 338-1987 IEEE 379-2000, IEEE 384-1992, IEEE 387-1995 IEEE 399-1997, IEEE 420-2001, IEEE 446-1995 IEEE 519-1992, IEEE 577-2004, IEEE 603-1991 (w/1995 correction sheet) IEEE 622-1987, IEEE 628-2001, IEEE 665-1995 IEEE 666-1991, IEEE 690-2004, IEEE 741-1986, IEEE 741-2007 IEEE 765-2006, IEEE 835-1994, IEEE 1050-1996 IEEE 1185-1994, IEEE 1584-2002, IEEE C37.06-2009 IEEE C37.013-1997, IEEE C37.13-2008 IEEE C37.16-2009, IEEE C37.17-1997 IEEE C37.20.1-2002, IEEE C37 20.2-1999 IEEE C37.20.3-2001, IEEE C37.23-2003 IEEE C37.46-2010, IEEE C37.47-2000 IEEE C37.90-2005, IEEE C37.91-2008 IEEE C37.96-2000, IEEE C37 101-1993 IEEE C37.102-2006, IEEE C37.106-2003 IEEE C37.121-1989, IEEE C50 13-2005 IEEE C57.12.00-2010, IEEE C57 12.01-2005 IEEE C57.12 10-2010, IEEE C57 12.22-1993 IEEE C57.12.55-1987, IEEE C57 12.51-2008 IEEE C57.116-1989, IEEE C62.11-2005, IEEE C62.23-1995 NUMARC 93-01 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	NA	NA	Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena Track NRC issuance of RIS 2011-XX, Adequacy of Station Electric Distribution System Voltages Track NRC issuance of Rev. 1 to RG 1.93, Availability of Electric Power Sources (DG-1244) and Rev. 2 to RG 1.106, Thermal Overload Protection for Electric Motors on MOVs (DG-1264)	3 4, 3 5, 3 6, 3.11, 4.6, 5.4, 6.2, 6 3, 6.5, 6 7, 7 2, 7 3, 7 4, 7.5, 7 6, 7 7, 8.1, 8 2, 8 3 2, 8 4, 9.1, 9 2, 9 3, 9 4, 9 5, 10 4, 13 5, 14.2, 14.3, 16, 17

Note
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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
8.3.1 AC Power Systems (onsite)	FSAR	10 CFR 50, App. A, GDC 2, 4, 5, 17, 18, 33, 34, 35, 38, 41, 44 and 50 10 CFR 50.34(f)(2)(v) 10 CFR 50.34(f)(2)(xiii) 10 CFR 50.34(f)(2)(xx) 10 CFR 50.55a(h) 10 CFR 50.63 10 CFR 50 65(a)(4)	No	RG 1 206	8.3.1 BTP 8-1 BTP 8-2 BTP 8-4 BTP 8-5 BTP 8-6 BTP 8-7 BTP 8-8	RGs 1.6, 1.9, 1 32, 1 47, 1.53, 1.63, 1.75, 1 81, 1.106, 1.118, 1.153, 1.155, 1.160, 1.182, 1.204 NUREG-0718 & NUREG-0737: I D 3, II E.3 1, II.G.1 NUREG-1784 NUREG-1793, NUREG-1801 NUREG/CR-0660 NUREG/CR-6866 NUREG/CR-7000 SECY 90-016 SECY 91-078 SECY 94-084 SECY 95-132 SECY 05-227 GL 1996-01, GL 2007-01 IN 98-38, IN 99-13 IN 00-06, IN 00-14 IN 02-01, IN 02-04 IN 02-12, IN 05-04 IN 06-18	IEEE 141-1993, IEEE 142-1991, IEEE 242-2001 IEEE 308-2001, IEEE 317-1983, IEEE 338-1987 IEEE 379-2000, IEEE 384-1992, IEEE 387-1995 IEEE 399-1997, IEEE 420-2001, IEEE 446-1995 IEEE 519-1992, IEEE 577-2004, IEEE 603-1991 (w/1995 correction sheet) IEEE 622-1987, IEEE 628-2001, IEEE 665-1995 IEEE 666-1991, IEEE 690-2004, IEEE 741-1986, IEEE 741-2007 IEEE 765-2006, IEEE 835-1994, IEEE 1050-1996 IEEE 1185-1994, IEEE 1584-2002, IEEE C37.06-2009 IEEE C37.013-1997, IEEE C37 13-2008 IEEE C37.16-2009, IEEE C37.17-1997 IEEE C37.20.1-2002, IEEE C37.20.2-1999 IEEE C37.20.3-2001, IEEE C37.23-2003 IEEE C37.46-2010, IEEE C37.47-2000 IEEE C37.90-2005, IEEE C37.91-2008 IEEE C37.96-2000, IEEE C37.101-1993 IEEE C37.102-2006, IEEE C37.106-2003 IEEE C37.121-1989, IEEE C50.13-2005 IEEE C57.12.00-2010, IEEE C57.12.01-2005 IEEE C57.12.10-2010, IEEE C57.12.22-1993 IEEE C57.12.55-1987, IEEE C57.12.51-2008 IEEE C57.116-1989, IEEE C62.11-2005, IEEE C62 23-1995 NUMARC 93-01 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	NA	No	Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena. Track NRC issuance of RIS 2011-XX, Adequacy of Station Electric Distribution System Voltages Track NRC issuance of Rev. 1 to RG 1.93, Availability of Electric Power Sources (DG-1244) and Rev. 2 to RG 1.106, Thermal Overload Protection for Electric Motors on MOVs (DG-1264)	3.4, 3 5, 3 6, 3.11, 4.6, 5 4, 6 2, 6.3, 6.5, 6.7, 7 2, 7.3, 7.4, 7.5, 7 6, 7.7, 8.1, 8 2, 8 3 2, 8.4, 9 1, 9.2, 9.3, 9 4, 9.5, 10 4, 13 5, 14 2, 16, 17

Note:
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
8.3 2 DC Power Systems (onsite)	PSAR	10 CFR 50, App A, GDC 2, 4, 5, 17, 18, 33, 34, 35, 38, 41, 44 and 50 10 CFR 50.34(f)(2)(v) 10 CFR 50 55a(h) 10 CFR 50.63 10 CFR 50 65(a)(4)	No	RG 1.70	8.3 2 BTP 8-5	RGs 1.6, 1.32, 1 47, 1.53, 1.63, 1.75, 1 81, 1.106, 1.118, 1 128, 1.129, 1.153, 1.155, 1.160, 1.182 NUREG-0718, I.D 3 NUREG-1793 IN 97-21	IEEE 242-2001, IEEE 308-2001 IEEE 338-1987, IEEE 379-2000 IEEE 384-1992, IEEE 450-2002 IEEE 484-2002, IEEE 485-1997 IEEE 603-1991 (w/1995 correction sheet), IEEE 741-1986 IEEE 741-2007, IEEE 946-2004, IEEE 1184-2006 IEEE 1187-2002, IEEE 1189-2007 IEEE 1375-1998, IEEE C37.17-1997 NUMARC 8700, Rev. 1 NUMARC 93-01, Rev 4 NUMARC 9700 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	No	No	Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena. Track NRC issuance of RIS 2011-XX, Adequacy of Station Electric Distribution System Voltages Track NRC issuance of Rev. 2 to RG 1 106, Thermal Overload Protection for Electric Motors on MOVs (DG-1264)	3 4, 3.5 3 6, 3.11 4.6, 5 4 6 2, 6 3 6.5, 6 7 7 2, 7.3 7.4, 7 5 7 6, 7.7 8.1, 8 2 8.3 1, 8 4 9 1, 9.2 9 3, 9 4 9 5, 10 4 13 5, 14.2 16, 17
	DCD	10 CFR 50, App A, GDC 2, 4, 5, 17, 18, 33, 34, 35, 38, 41, 44 and 50 10 CFR 50.34(f)(2)(v) 10 CFR 50.55a(h) 10 CFR 50 63 10 CFR 50.65(a)(4) 10 CFR 52.47(b)(1)	No	RG 1.206	8.3 2 BTP 8-5	RGs 1.6, 1.32, 1 47, 1 53, 1 63, 1.75, 1.81, 1.106, 1.118, 1.128, 1.129, 1.153, 1.155, 1.160, 1.182 NUREG-0718, I D.3 NUREG-1793 IN 97-21	IEEE 242-2001, IEEE 308-2001 IEEE 338-1987, IEEE 379-2000 IEEE 384-1992, IEEE 450-2002 IEEE 484-2002, IEEE 485-1997 IEEE 603-1991 (w/1995 correction sheet), IEEE 741-1986 IEEE 741-2007, IEEE 946-2004, IEEE 1184-2006 IEEE 1187-2002, IEEE 1189-2007 IEEE 1375-1998, IEEE C37.17-1997 NUMARC 8700, Rev. 1 NUMARC 93-01, Rev 4 NUMARC 9700 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	NA	NA	Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena Track NRC issuance of RIS 2011-XX, Adequacy of Station Electric Distribution System Voltages Track NRC issuance of Rev. 2 to RG 1.106, Thermal Overload Protection for Electric Motors on MOVs (DG-1264)	3.4, 3 5 3 6, 3 11 4 6, 5 4 6 2, 6.3 6 5, 6 7 7.2, 7.3 7.4, 7.5 7 6, 7.7 8 1, 8 2 8.3 1, 8.4 9.1, 9 2 9 3, 9 4 9.5, 10 4 13 5, 14 2 14.3 16, 17
	FSAR	10 CFR 50, App. A, GDC 2, 4, 5, 17, 18, 33, 34, 35, 38, 41, 44 and 50 10 CFR 50 34(f)(2)(v) 10 CFR 50 55a(h) 10 CFR 50 63 10 CFR 50 65 (a)(4)	No	RG 1 206	8 3 2 BTP 8-5	RGs 1 6, 1.32, 1.47, 1 53, 1 63, 1.75, 1 81, 1.106, 1.118, 1.128, 1.129, 1.153, 1.155, 1.160, 1.182 NUREG-0718, I D.3 NUREG-1793 IN 97-21	IEEE 242-2001, IEEE 308-2001 IEEE 338-1987, IEEE 379-2000 IEEE 384-1992, IEEE 450-2002 IEEE 484-2002, IEEE 485-1997 IEEE 603-1991 (w/1995 correction sheet), IEEE 741-1986 IEEE 741-2007, IEEE 946-2004, IEEE 1184-2006 IEEE 1187-2002, IEEE 1189-2007 IEEE 1375-1998, IEEE C37.17-1997 NUMARC 8700, Rev. 1 NUMARC 93-01, Rev 4 NUMARC 9700 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	NA	No	Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena Track NRC issuance of RIS 2011-XX, Adequacy of Station Electric Distribution System Voltages Track NRC issuance of Rev. 2 to RG 1.106, Thermal Overload Protection for Electric Motors on MOVs (DG-1264)	3 4, 3 5 3 6, 3 11 4 6 5 4 6 2, 6 3 6 5, 6.7 7.2, 7 3, 7.4, 7.5, 7.6, 7 7 8 1, 8 2 8 3 1, 8.4 9.1, 9 2 9.3, 9.4 9.5, 10 4 13 5, 14 2 16, 17

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RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Clinch River Regulatory Framework Document
NRC Version

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
8 4 Station Blackout	PSAR	10 CFR 50, App A, GDC 17, 18 10 CFR 50 63 10 CFR 50 65(a)(4)	Yes - GDC 17	RG 1.70 NOTE: PSAR Section 8 4 is not included in RG 1.70. This section will address current regulatory guidance on station blackout.	8 4	RGs 1.9, 1.155, 1.160, 1.182 NUREG-0737, Supplement 1 NUREG-0933, USI A-44 NUREG-1032 NUREG-1776 NUREG-1784 NUREG-1793 NUREG/CR-6890 SECY 90-016 SECY 91-078 SECY 94-084 SECY 95-132 GL 2006-02 IN 97-21, IN 97-05 IN 98-07, IN 00-06, IN 06-06 RIS 2000-24 & 2004-05 TI 2515/120	IEEE 308 -2001 IEEE 387 -1995 IEEE 485 -1997 IEEE 765 -2006 NUMARC 8700, Rev. 1 NUMARC 93-01, Rev. 4 NSAC-108 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	Yes - Section 8.4 will address Station Blackout requirements issued subsequent to RG 1.70	No	Similar to the AP1000 design - an exemption to the GDC 17 requirement for two physically independent offsite circuits for a passive reactor design is planned Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena.	4.6, 5.4 6.2, 6.3 6.5, 6.7 7 2 8 1, 8 2 8 3 1, 8.3.2 9.1, 9 2 9 3, 9 4 9 5, 10 4 13 5, 14 2 16, 17
	DCD	10 CFR 50, App. A, GDC 17, 18 10 CFR 50 63 10 CFR 50 65 (a)(4) 10 CFR 52.47(b)(1)	Yes - GDC 17	RG 1 206	8 4	RGs 1 9, 1.155, 1 160, 1.182 NUREG-0737, Supplement 1 NUREG-0933, USI A-44 NUREG-1032 NUREG-1776 NUREG-1784 NUREG-1793 NUREG/CR-6890 SECY 90-016 SECY 91-078 SECY 94-084 SECY 95-132 GL 2006-02 IN 97-21, IN 97-05 IN 98-07, IN 00-06, IN 06-06 RIS 2000-24 & 2004-05 TI 2515/120	IEEE 308 -2001 IEEE 387 -1995 IEEE 485 -1997 IEEE 765 -2006 NUMARC 8700, Rev. 1 NUMARC 93-01, Rev. 4 NSAC-108 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	NA	NA	Similar to the AP1000 design - an exemption to the GDC 17 requirement for two physically independent offsite circuits for a passive reactor design is planned. Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena.	4 6, 5 4 6 2, 6 3 6 5, 6 7 7.2 8 1, 8 2 8.3 1, 8 3 2 9 1, 9 2 9.3, 9 4 9 5, 10 4 13 5, 14.2 14.3 16, 17
	FSAR	10 CFR 50, App. A, GDC 17, 18 10 CFR 50.63 10 CFR 50.65 (a)(4)	Yes - GDC 17	RG 1.206	8 4	RGs 1.9, 1.155, 1 160, 1.182 NUREG-0737, Supplement 1 NUREG-0933, USI A-44 NUREG-1032 NUREG-1776 NUREG-1784 NUREG-1793 NUREG/CR-6890 SECY 90-016 SECY 91-078 SECY 94-084 SECY 95-132 GL 2006-02 IN 97-21, IN 97-05 IN 98-07, IN 00-06, IN 06-06 RIS 2000-24 & 2004-05 TI 2515/120	IEEE 308 -2001 IEEE 387 -1995 IEEE 485 -1997 IEEE 765 -2006 NUMARC 8700, Rev 1 NUMARC 93-01, Rev. 4 NSAC-108 ICEA P32-282-2007, NEMA WC-51-2009 NEMA SG 4-2009, NEMA MG 1-2009 NFPA 70-2011, NFPA 72-2010 NFPA 110-2010	NA	No	Similar to the AP1000 design - an exemption to the GDC 17 requirement for two physically independent offsite circuits for a passive reactor design is planned. Assess and apply lessons learned from the Fukushima Daiichi power station regarding offsite power systems operation during and after earthquakes or other natural phenomena.	4 6, 5 4 6 2, 6 3 6 5, 6.7 7.2 8 1, 8.2 8 3.1, 8.3 2 9.1, 9 2 9 3, 9 4 9.5, 10 4 13 5, 14.2 16, 17

Note.
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Chapter 8 Outline

8 Electric Power

8.1 Introduction

PSAR PSAR Chapter 8 provides information on the functional adequacy of the offsite and onsite power systems, including the ability to mitigate station blackout. PSAR Section 8.1 provides an introduction to the following topics:

- the local utility infrastructure, the TVA grid and the interconnection(s) to the Clinch River Plant site, including the connection of the unit switchyard(s) to the local utility
- the offsite and onsite electric systems
- the alternate AC power sources and provisions for mitigating station blackout and the ability of plant to run in the island mode
- compliance to applicable regulatory guides, branch technical positions, generic letters, and industry standards

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DCD DCD Chapter 8 provides information on the functional adequacy of the offsite and onsite power systems, including the ability to mitigate a station blackout. DCD Section 8.1 provides summary information on the following topics:

- connection of the unit switchyard(s) to the local utility
- offsite electric power systems
- onsite electric systems, including provisions to mitigate station blackout and ability to run in the island mode
- alternate AC power sources
- safety loads, the safety functions performed, and the type of electric power required by each safety load
- design bases, criteria, regulatory guides, standards, and other documents to be used in the design of safety-related and important-to-safety electric power systems
- extent to which the overall design conforms to each appropriate regulatory criteria, e.g., NRC regulatory guides (RGs), branch technical positions (BTPs), generic letters (GLs), and industry standards

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Chapter 8 Outline

8.1 Introduction (cont.)

FSAR Same contents as mPower standard plant DCD Section 8.1 updated with site-specific information if required.

Chapter 8 Outline

8.2 Offsite Power System

PSAR PSAR Section 8.2 provides information on the offsite power system, which encompasses the connections to the grid, transmission lines (overhead or underground), transmission line towers, transformers, switchyard components and control systems, switchyard battery systems, etc., including the main step-up transformer, Unit Auxiliary Transformer (UAT), and Reserve Auxiliary Transformer (RAT).

Preliminary information is provided in Section 8.2 on the following topics:

- offsite power line(s) coming from the transmission network to the plant switchyard
- single designated offsite power circuit provided from the transmission network with sufficient capacity and capability to power safety systems under normal, abnormal, and accident conditions (passive designs are exempted from providing two offsite power sources required by GDC 17)
- Failure Mode and Effects Analysis (FMEA)
- compliance with regulatory requirements and guidance for the preferred power source
- description of how testing is performed on the offsite power system components to demonstrate compliance with the design requirements and applicable regulations, and identification of the potential effects that must be considered during testing, the margins being applied, and how the design incorporates these requirements for offsite power supplies, including, but not limited to, high-voltage transmission networks, medium-voltage distribution networks, switchyard equipment (bus work, transformers, circuit breakers, disconnect switches, surge protective devices, control, communication, grounding, and lightning systems), switching capacitors, and voltage control equipment between the switchyard and the plant.
- planned offsite power system component testing to demonstrate compliance with design requirements and applicable regulations
- potential adverse effects considered during testing while equipment is energized
- location of rights-of-way, transmission towers, voltage level, and length of each transmission line from the site to the first major substation that connects the line to the grid (preferred power source)
- layout drawings of the circuit(s) connecting the onsite distribution system to the preferred power supply
- grid stability including equipment that must be considered for review and approval by the appropriate grid reliability planning and coordination organization(s)

8.2 Offsite Power System (cont.)

PSAR (cont.)

- real time analysis tools to assess the impact of the loss or unavailability of various transmission system elements and the electrical condition of the transmission system
- how the stability of the grid is continuously studied as the loads grow and more transmission lines and generators are added
- the approving grid organization for reliability studies and any potential limits that may be imposed on operation
- grid availability and reliability
- testing program to ensure compliance of electric power systems with regulatory requirements
- compliance with regulatory requirements and guidance including: GDC 2, GDC 4, GDC 5, GDC 17, and GDC 18, for the preferred power source

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DCD

DCD Section 8.2 provides information on the offsite power system, which encompasses the connections to the grid, transmission lines (overhead or underground), transmission line towers, transformers, switchyard components and control systems, switchyard battery systems, etc., including the main step-up transformer, UAT and RAT. Section 8.2 provides detailed information on the following topics:

- capacity and electrical characteristics of transformers, breakers, busses, and preferred power source to demonstrate that there is adequate capability to supply maximum connected load during all plant conditions
- Failure Mode and Effects Analysis (FMEA)
- compliance with regulatory requirements and guidance including: GDC 2, GDC 4, GDC 5, GDC 17, and GDC 18, for the preferred power source
- description of how testing is performed on the offsite power system components to demonstrate compliance with the design requirements and applicable regulations, and identification of the potential effects that must be considered during testing, the margins being applied, and how the design incorporates these requirements for offsite power supplies, including, but not limited to, high-voltage transmission networks, medium-voltage distribution networks, switchyard equipment (bus work, transformers, circuit breakers, disconnect switches, surge protective devices, control, communication, grounding, and lightning systems), switching capacitors, and voltage control equipment between the switchyard and the plant.
- provision for offsite power system components testing to demonstrate compliance with design requirements and applicable regulations

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Chapter 8 Outline

8.2 Offsite Power System (cont.)

DCD (cont.)

- potential effects that must be considered during testing, the margins being applied, and how the design incorporates these requirements for offsite power supplies, include: high-voltage transmission networks, medium-voltage distribution networks, switchyard equipment (bus work, transformers, circuit breakers, disconnect switches, surge protective devices, control, communication, grounding, and lightning systems), switching capacitors, and voltage control equipment between the switchyard and the plant
- layout drawings of the circuit(s) connecting the onsite distribution system to the preferred power supply
- ability of the plant to operate in the islanding mode to provide auxiliary house loads, especially when there is loss of off-site power or loss of grid power

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FSAR Same contents as mPower standard plant DCD Section 8.2 with the following supplemental information:

- site-specific information on the offsite power system
- design of the ac power transmission system and its testing and inspection plan
- offsite power system components testing to demonstrate compliance with design requirements and applicable regulations
- potential effects that must be considered during testing, the margins being applied, and how the design incorporates these requirements for offsite power supplies, include: high-voltage transmission networks, medium-voltage distribution networks, switchyard equipment (bus work, transformers, circuit breakers, disconnect switches, surge protective devices, control, communication, grounding, and lightning systems), switching capacitors, and voltage control equipment between the switchyard and the plant
- maximum and minimum switchyard voltage that must be maintained by the transmission system provider/operator (TSP/TSO) without any reactive power support from the nuclear power plant and actions that will be taken by the plant operator when these voltages can not be maintained
- formal agreement or protocol between the nuclear power plant and the capability of the TSP/TSO regarding the preferred offsite system support plant startup, and shut down under normal and emergency conditions
- the TSP's ability to analyze contingencies on the grid involving the largest generation unit outage, critical transmission line outage, and other contingencies under varying power flows in response to market conditions and system demands

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Chapter 8 Outline

8.2 Offsite Power System (cont.)

FSAR (cont.)

- TSO's real time analysis tool used to determine the impact of the loss or unavailability of various transmission system elements on the condition of the transmission system
- TVA protocols for remaining cognizant of grid vulnerabilities in order to make informed decisions regarding maintenance activities that are critical to the plant's electrical system (10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants")
- the need for grid stability and to continuously study the grid as loads grow and transmission lines and generators are added
- acceptance criteria required for the continued safe operation and stability of the grid
- the organization that approves grid reliability studies and potential limits that may be imposed on operation
- grid availability, including the frequency, duration, and causes of outages over the past 20 years for both the transmission system accepting the unit's output and the transmission system providing the preferred power for the unit's loads
- grid stability analysis showing that loss of the largest single supply to the grid does not result in the complete loss of preferred power
- analysis showing that the grid remains stable for the minimum time required following a turbine trip to support assumptions made in safety analyses
- updated loads based on vendor supplied data
- technical interfaces including those for ac power requirements from offsite and the analysis of the offsite transmission system and the setting of protective devices

Chapter 8 Outline

8.3 Onsite Power Systems

PSAR PSAR Section 8.3 provides information on the AC and DC onsite power systems, including standby power sources, distribution systems, and auxiliary supporting systems. Preliminary information is provided in Section 8.3 on the following topics:

- onsite power systems, emphasizing those portions that are safety-related
- onsite power systems needed to support risk-important, non-safety related active systems
- functional logic diagrams, electrical single-line diagrams, tables, physical arrangement drawings,
- redundancy in the standby power systems with regard to both power sources and associated distribution systems
- instrumentation systems and control devices for the Class 1E loads and power systems
- physical and electrical redundancy, independence, and conformance with single failure criteria
- capacity and capability of safety-related components and systems to accomplish the necessary safety functions
- onsite standby and auxiliary power sources (e.g., diesel generators)
- bases and design criteria that address the following:
  - electric motors including temperature monitoring devices
  - interrupting capacity of switchgear, load centers, control centers, and distribution panels
  - electric circuit protection and grounding
  - ambient room temperature for proper equipment operation
- electrical power system calculations and distribution system studies including:
  - load flow/voltage regulation studies
  - short-circuit studies
  - equipment sizing studies including cable ampacity sizing
  - insulation coordination (surge and lightning protection)
- continuous and short-term ratings of batteries and battery chargers and voltage recovering characteristics of batteries and battery chargers
- dc system load performance including voltage profile curves, discharge rate curves, and temperature effect curves

Chapter 8 Outline

8.3 Onsite Power Systems (cont.)

PSAR (cont.)

- compliance with the GDCs, RGs, and other applicable guidance for both ac and dc systems

DCD

DCD Section 8.3 provides information on the onsite AC and DC power systems including standby power sources, distribution systems, and auxiliary supporting systems. Detailed information is provided in Section 8.3 on the following topics:

- onsite power systems, emphasizing those portions that are safety-related
- onsite power systems needed to support risk-important, non-safety related active systems identified through the regulatory treatment of non-safety systems process
- functional logic diagrams, electrical single-line diagrams, tables, physical arrangement drawings, and electrical schematics, describing the design of the electrical distribution systems, including grounding and lightning protection
- functional requirements of the onsite power system (including equipment capacities and the operational environment of the onsite power system)
- redundancy in the standby power systems and associated distribution systems, and how safety-related and important to safety loads are distributed between redundant safety divisions or groups
- how instrumentation systems and control devices for the Class 1E loads and power systems are supplied from the related redundant distribution systems
- the physical separation and electrical redundancy, conformance with single failure criteria, independence, and the capacity and capability of safety-related components and systems to accomplish necessary safety functions
- design aspects of the onsite standby and auxiliary power sources (e.g., diesel generators):
  - starting initiating circuits
  - electrical starting mechanism and system
  - tripping devices
  - interlocks and permissives
  - load shedding circuits
  - testability
  - fuel oil storage and transfer system, including capacity

8.3 Onsite Power Systems (cont.)

DCD (cont.)

- cooling and heating systems
- instrumentation and control systems, including status alarms and indications, with assigned power supply
- prototype qualification program

- bases and design criteria that establish the following:
  - motor ratings
  - minimum motor accelerating voltage
  - motor starting torque
  - minimum motor torque margin over pump torque through accelerating period at minimum applied voltage
  - motor insulation
  - temperature monitoring devices provided in large-horsepower motors
  - interrupting capacity of switchgear, load centers, control centers, and distribution panels
  - electric circuit protection
  - grounding requirements
  - forced cooling requirements
- electrical power system calculations and distribution system studies including:
  - load flow/voltage regulation studies and under/overvoltage protection
  - short-circuit studies
  - equipment sizing studies including cable ampacity
  - insulation coordination (surge and lightning protection)
- dc power system redundancy with regard to power sources and distribution loads
- continuous and short-term ratings of batteries and battery chargers and the voltage recovering characteristics of batteries and battery chargers
- dc power system capability to perform its safety function in the event of a single-failure
- electrical connections between redundant dc system components and their physical arrangement
- suitability of batteries and battery chargers as dc power supplies to inverters that power safety-related instrumentation to perform the intended function

Chapter 8 Outline

8.3 Onsite Power Systems (cont.)

DCD (cont.)

- dc system performance characteristic curves including voltage profile curves, discharge rate curves, and temperature effect curves
- for connection of non-safety loads to the dc system, provide the sizing of batteries and battery chargers to accommodate the added non-Class 1E loads
- for both ac and dc systems, compliance with the GDCs (2, 4, 5, 17, 18, and 50) and the extent to which RGs and other applicable guidance are followed
- ground fault clearance on the Class 1E dc system

FSAR Same contents as mPower standard plant DCD Section 8.3 updated with site-specific information as applicable and with the following supplemental information:

- periodic inspection for sulfated battery plates or other anomalous conditions
- battery maintenance and surveillance
- periodic testing of penetration protective devices
- diesel generator operation, inspection, and maintenance in accordance with manufacturer's recommendations
- periodic testing on the battery chargers and voltage regulating transformers

8.4 Station Blackout

PSAR Section 8.4 addresses the plant's ability to mitigate a Station Blackout (SBO) event for a specified duration. Preliminary information is provided on the following topics:

- compliance with 10 CFR 50.63
- procedures and training planned for plant operators
- discuss the plant's ability to perform its safety functions without reliance on ac power for 72 hours after an initiating SBO event
- the ability to power safe shutdown loads after the initial 72-hour period
- discuss the minimum required duration for operating on safety-related batteries and identify the paths available to recharge the batteries
- compliance to applicable regulatory criteria, GDCs, and guidance for SBO

DCD Section 8.4 addresses the plant's ability to mitigate a station blackout event for a specified duration. Detailed information is provided on the following topics:

- compliance with 10 CFR 50.63
- islanding capability and how this feature mitigates loss of off site power events
- plant's ability to withstand an SBO event until an ac power source can be brought on line to support long term (greater than 72 hours) core cooling
- provide the minimum required duration for operating only on safety-related batteries and identify the paths available to recharge the batteries
- compliance with applicable regulatory criteria, GDCs (17 & 18), and industry guidance

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FSAR Same contents as mPower standard plant DCD Section 8.4 updated with site-specific information as applicable and with the following supplemental information:

- SBO mitigation procedures and training provided for plant operators, including per NUMARC 87-00:
  - restoration of onsite power sources,
  - ac power restoration (e.g., coordination with transmission system load dispatcher), and
  - severe weather guidance (e.g., identification of actions to prepare for the onset of severe weather such as an impending tornado), as applicable

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Clinch River Regulatory Framework Document
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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
12.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	PSAR	10 CFR 19.12 10 CFR 20 1101(b) 10 CFR 20.1406	No	RG 1.70	12.1	RG 1 8 RG 8.8 RG 8 10 RG 8 27 NUREG-1736 NUREG-1757 NUREG/CR-3587 DC/COL-ISG-6	NEI 07-08A	No	No	None	12.2 12 3 12.4 12.5
	DCD	10 CFR 19.12 10 CFR 20 1101(b) 10 CFR 20.1406 10 CFR 52 47(b)(1)	No	RG 1 206	12.1	RG 1 8 RG 8 8 RG 8 10 RG 8 27 RG 8 38 NUREG-1736 NUREG-1757 NUREG/CR-3587 DC/COL-ISG-6	NEI 07-08A	N/A	N/A	None	12.2 12.3 12.4 12 5 14.3
	FSAR	10 CFR 19 12 10 CFR 20.1101(b) 10 CFR 20 1406	No	RG 1 206	12.1	RG 1 8 RG 8 8 RG 8.10 RG 8 27 RG 8.38 NUREG-1736 NUREG-1757 NUREG/CR-3587 DC/COL-ISG-6 Operational Considerations: RGs 8 2, 8.7, 8 9, 8.13, 8.15, 8 20, 8 25, 8 26, 8 28, 8 34, 8 35, 8 36	NEI 07-08A	N/A	N/A	None	12.2 12.3 12 4 12.5

Note.
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
12.2 Radiation Sources	PSAR	10 CFR 19 12 10 CFR 20 1101(b) 10 CFR 20 1201 10 CFR 20.1202 10 CFR 20.1203 10 CFR 20.1204 10 CFR 20.1206 10 CFR 20.1207 10 CFR 20.1301 10 CFR 20.1801 10 CFR 50, App A, GDCs 19 & 61 10 CFR 50.34(f)	No	RG 1.70 (Note: PSAR Section 12 2 will reference ANSI/ANS 18.1 which superseded ANSI N237)	12.2	RG 1 7 RG 1.112 RG 1.183 NUREG-0737, II.B.2	NEI 07-08A ANSI /ANS 18 1-1999 (ANSI N237 was superseded by ANSI/ANS 18 1)	No	No	None	11 1 11 2 11 3 11 4 12.1 12 3 12 4
	DCD	10 CFR 19.12 10 CFR 20.1101(b) 10 CFR 20.1201 10 CFR 20.1202 10 CFR 20.1203 10 CFR 20.1204 10 CFR 20.1206 10 CFR 20.1207 10 CFR 20.1301 10 CFR 20.1801 10 CFR 50, App A, GDCs 19 & 61 10 CFR 50.34(f) 10 CFR 52 47(b)(1)	No	RG 1.206	12.2	RG 1.7 RG 1.112 RG 1.183 NUREG-0737, II B.2	NEI 07-08A ANSI/ANS 18 1-1999	N/A	N/A	None	11.1 11 2 11 3 11.4 12.1 12.3 12 4
	FSAR	10 CFR 19.12 10 CFR 20.1101(b) 10 CFR 20.1201 10 CFR 20.1202 10 CFR 20.1203 10 CFR 20 1204 10 CFR 20 1206 10 CFR 20.1207 10 CFR 20.1301 10 CFR 20.1801 10 CFR 50, App A, GDCs 19 & 61 10 CFR 50.34(f)	No	RG 1.206	12.2	RG 1.7 RG 1.112 RG 1.183 NUREG-0737, II.B 2	NEI 07-08A ANSI/ANS 18 1-1999	N/A	N/A	None	11 1 11.2 11.3 11 4 12 1 12 3 12 4

Note:
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
12.3 Radiation Protection Design Features	PSAR	10 CFR 19.12 10 CFR 20 1101(b) 10 CFR 20.1201, 10 CFR 20 1202 10 CFR 20.1203, 10 CFR 20.1204 10 CFR 20.1301, 10 CFR 20.1302 10 CFR 20.1406 10 CFR 20.1501 10 CFR 20.1601, 10 CFR 20.1602 10 CFR 20.1701, 10 CFR 20.1702 10 CFR 20.1801 10 CFR 20.1901, 10 CFR 20 1902 10 CFR 20.1903, 10 CFR 20 1904 10 CFR 20.1905 10 CFR 50 34(f) 10 CFR 50 68 10 CFR 50, App A, GDCs 19, 61, 63	No	RG 1.70	12.3	RG 1.7 RG 1.21 RG 1.52 RG 1.69 RG 1.97 RG 1.183 RG 4 21 RG 8 2 RG 8 8 RG 8 10 RG 8.19 RG 8.25 RG 8.38 NUREG-0737, II B 2, II.F.1.3, III.D.3 3 NUREG-1430 NUREG/CR-3587	NEI 07-08A NEI 08-08A ANSI 6.4-2006 ANSI/ANS-HPSSC-6 81-1981 ANSI/HPS N13.1-1999	No	No	None	12 1 12.2 12 4
	DCD	10 CFR 19.12 10 CFR 20.1101(b) 10 CFR 20.1201, 10 CFR 20.1202 10 CFR 20.1203, 10 CFR 20.1204 10 CFR 20.1301, 10 CFR 20 1302 10 CFR 20.1406 10 CFR 20.1501 10 CFR 20.1601, 10 CFR 20.1602 10 CFR 20.1701, 10 CFR 20.1702 10 CFR 20.1801 10 CFR 20.1901, 10 CFR 20.1902 10 CFR 20.1903, 10 CFR 20.1904 10 CFR 20.1905 10 CFR 50 34(f)(2)(vii) 10 CFR 50 68 10 CFR 50, App A, GDCs 19, 61, 63 10 CFR 52 47(b)(1)	No	RG 1.206	12.3	RG 1.7 RG 1 21 RG 1.52 RG 1 69 RG 1.97 RG 1.183 RG 4.21 RG 8.2 RG 8 8 RG 8.10 RG 8.19 RG 8 25 RG 8 38 NUREG-0737, II.B 2, II.F.1.3, III.D.3.3 NUREG1430 NUREG/CR-3587	NEI 07-08A NEI 08-08A ANSI 6 4-2006 ANSI/ANS-HPSSC-6 81-1981 ANSI/HPS N13.1-1999	N/A	N/A	None	12.1 12 2 12.4
	FSAR	10 CFR 19.12 10 CFR 20.1101(b) 10 CFR 20.1201, 10 CFR 20 1202 10 CFR 20.1203, 10 CFR 20.1204 10 CFR 20.1301, 10 CFR 20 1302 10 CFR 20.1406 10 CFR 20.1501 10 CFR 20.1601, 10 CFR 20 1602 10 CFR 20 1701, 10 CFR 20.1702 10 CFR 20.1801 10 CFR 20.1901, 10 CFR 20 1902 10 CFR 20.1903, 10 CFR 20 1904 10 CFR 20.1905 10 CFR 50.34(f)(2)(vii) 10 CFR 50 68 10 CFR 50, App A, GDCs 19, 61, 63	No	RG 1.206	12.3	RG 1 7 RG 1.21 RG 1.52 RG 1 69 RG 1.97 RG 1.183 RG 4 21 RG 8 2 RG 8 8 RG 8 10 RG 8 19 RG 8.25 RG 8.38 NUREG-0737, II.B.2, II.F.1.3, III.D 3 3 NUREG-1430 NUREG/CR-3587	NEI 07-08A NEI 08-08A ANSI 6 4-2006 ANSI/ANS-HPSSC-6 81-1981 ANSI/HPS N13.1-1999	N/A	N/A	None	12.1 12 2 12.4

Note:
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
12 4 Dose Assessment	PSAR	10 CFR 19 12 10 CFR 20 1101(b) 10 CFR 20 1201, 10 CFR 20.1202 10 CFR 20.1203, 10 CFR 20.1204 10 CFR 20 1301 10 CFR 20.1406 10 CFR 20.1302 10 CFR 20.1701, 10 CFR 20 1702 10 CFR 20.1601, 10 CFR 20.1602 10 CFR 20.1801 10 CFR 20.1901, 10 CFR 20.1902 10 CFR 20.1903, 10 CFR 20.1904 10 CFR 20.1905 10 CFR 50.34(f) 10 CFR 50 68 10 CFR 50, App A, GDCs 19, 61, 63	No	RG 1 70	12 4	RG 1.7 RG 1 52 RG 1 69 RG 1.97 RG 1 183 RG 8.2 RG 8 8 RG 8.10 RG 8 19 RG 8 25 RG 8.38 NUREG-0737, II.F.1.3, III.D.3 3 NUREG-1430	NEI 07-08A ANSI 6 4-2006 ANSI/HPS N13 1-1999	No	No	None	12.1 12.2 12 3
	DCD	10 CFR 19.12 10 CFR 20.1101(b) 10 CFR 20.1201, 10 CFR 20.1202 10 CFR 20.1203, 10 CFR 20.1204 10 CFR 20.1301, 10 CFR 20 1302 10 CFR 20.1406 10 CFR 20.1601, 10 CFR 20.1602 10 CFR 20.1701, 10 CFR 20.1702 10 CFR 20.1801 10 CFR 20.1901, 10 CFR 20.1902 10 CFR 20.1903, 10 CFR 20 1904 10 CFR 20.1905 10 CFR 50.34(f) 10 CFR 50 68 10 CFR 50, App A, GDCs 19, 61, 63 10 CFR 52 47(b)(1)	No	RG 1 206	12 4	RG 1.7 RG 1.52 RG 1.69 RG 1.97 RG 1.183 RG 8.2 RG 8 8 RG 8.10 RG 8.19 RG 8.25 RG 8.38 NUREG-0737, II.F.1.3, III D.3 3 NUREG-1430	NEI 07-08A ANSI 6 4-2006 ANSI/HPS N13 1-1999	N/A	N/A	None	12 1 12.2 12.3
	FSAR	10 CFR 19.12, 10 CFR 19.13 10 CFR 20.1101 10 CFR 20.1201, 10 CFR 20.1202 10 CFR 20.1203, 10 CFR 20 1204 10 CFR 20.1206, 10 CFR 20.1207 10 CFR 20 1208 10 CFR 20 1301, 10 CFR 20.1302 10 CFR 20.1406 10 CFR 20.1501, 10 CFR 20 1502 10 CFR 20.1601, 10 CFR 20 1602 10 CFR 20.1701, 10 CFR 20 1702 10 CFR 20.1703, 10 CFR 20 1801 10 CFR 20.1901, 10 CFR 20.1902 10 CFR 20.1903, 10 CFR 20 1904 10 CFR 20.1905, 10 CFR 20 1906 10 CFR 20 2001, 10 CFR 20 2006 10 CFR 20 2101, 10 CFR 20.2102 10 CFR 20 2103, 10 CFR 20 2104 10 CFR 20 2105 10 CFR 20.2201, 10 CFR 20.2202 10 CFR 20 2203, 10 CFR 20 2204 10 CFR 20 2205, 10 CFR 20 2206 10 CFR 50.34(f) 10 CFR 50.68 10 CFR 50, App A, GDCs 19, 61, 63	No	RG 1.206	12.4	RG 1.7 RG 1.52 RG 1.69 RG 1.97 RG 1.183 RG 8 2 RG 8 8 RG 8 10 RG 8.19 RG 8 25 RG 8.38 NUREG-0737,II.B 3, II F.1.3, III D.3 3 NUREG-1430	NEI 07-08A ANSI 6 4-2006 ANSI/HPS N13 1-1999	N/A	N/A	None	12.1 12.2 12 3

Note.
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
12 5 Operational Radiation Protection Program	PSAR	10 CFR 19.12 10 CFR 20 1101(b) 10 CFR 20.1201 10 CFR 20.1202 10 CFR 20.1203 10 CFR 20.1204 10 CFR 20.1206 10 CFR 20.1207 10 CFR 20.1301 10 CFR 20.1801 10 CFR 50.120 10 CFR 50, App A, GDC 64 10 CFR 50 34(f) 10 CFR 71 5, 10 CFR 71.101 10 CFR71, Subparts G and H	No	RG 1 70 (Note: PSAR Section 12.5 will be titled "Operational Radiation Protection Program consistent with RG 1 206 in lieu of "Health Physics Program")	12 5	RG 1 8, RG 1.33, RG 1 97 RG 4 21 RG 8 2, RG 8 4, RG 8.7, RG 8 8, RG 8 9 RG 8.10, RG 8.13, RG 8.15 RG 8 20, RG 8.25, RG 8.26 RG 8 27, RG 8 29, RG 8.34 RG 8 35, RG 8.36, RG 8.38 NUREG/CR-0041 NUREG-0737 NUREG-0731 NUREG-1736 NUREG/CR-3587 IM-2504	ANS/HPS N13.11-2001 ANS/HPS N13 14-1994 ANS/HPS N13.30-1996 ANS/HPS N13 42-1997 ANSI N13 6-1999 ANSI N42.17A-1989 ANS N42.20-2003 ANS N42.28-2002 ANSI N323A-1997 IEEE 309-1991 NEI 07-03A NEI 08-08A	No	No	None	12 2 12 3 12.4 13 4
	DCD	10 CFR 19.12 10 CFR 20 1101(b) 10 CFR 20.1201 10 CFR 20 1202 10 CFR 20.1203 10 CFR 20.1204 10 CFR 20.1206 10 CFR 20.1207 10 CFR 20.1301 10 CFR 20.1801 10 CFR 50.120 10 CFR 50, App A, GDC 64 10 CFR 50.34(f) 10 CFR 71.5, 10 CFR 71.101 10 CFR71, Subparts G and H 10 CFR 52 47(b)(1)	No	RG 1 206	12 5	RG 1 8, RG 1.33, RG 1.97 RG 4.21 RG 8 2, RG 8 4, RG 8.7, RG 8.8, RG 8 9 RG 8.10, RG 8 13, RG 8.15 RG 8 20, RG 8.25, RG 8.26 RG 8 27, RG 8.29, RG 8.34 RG 8 35, RG 8.36, RG 8.38 NUREG/CR-0041 NUREG-0737 NUREG-0731 NUREG-1736 NUREG/CR-3587 IM-2504	ANS/HPS N13.11-2001 ANS/HPS N13.14-1994 ANS/HPS N13 30-1996 ANS/HPS N13 42-1997 ANSI N13 6-1999 ANSI N42.17A-1989 ANS N42 20-2003 ANS N42 28-2002 ANSI N323A-1997 IEEE 309-1991 NEI 07-03A NEI 08-08A	N/A	N/A	None	12.2 12.3 12.4 13 4
	FSAR	10 CFR 19.12 10 CFR 20.1101(b) 10 CFR 20.1201 10 CFR 20.1202 10 CFR 20 1203 10 CFR 20.1204 10 CFR 20 1206 10 CFR 20.1207 10 CFR 20.1301 10 CFR 20 1801 10 CFR 50 120 10 CFR 50, App A, GDC 64 10 CFR 50 34(f) 10 CFR 71.5, 10 CFR 71.101 10 CFR71, Subparts G and H	No	RG 1.206	12 5	RG 1.8, RG 1 33, RG 1.97 RG 4.21 RG 8 2, RG 8 4, RG 8 7, RG 8 8, RG 8.9 RG 8.10, RG 8 13, RG 8 15 RG 8 20, RG 8 25, RG 8 26 RG 8.27, RG 8 29, RG 8 34 RG 8.35, RG 8 36, RG 8 38 NUREG/CR-0041 NUREG-0737 NUREG-0731 NUREG-1736 NUREG/CR-3587 IM-2504	ANS/HPS N13 11-2001 ANS/HPS N13.14-1994 ANS/HPS N13 30-1996 ANS/HPS N13 42-1997 ANSI N13 6-1999 ANSI N42.17A-1989 ANS N42 20-2003 ANS N42.28-2002 ANSI N323A-1997 IEEE 309-1991 NEI 07-03A NEI 08-08A	N/A	N/A	None	12 2 12 3 12 4 13 4

Note.
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Chapter 12 - Section Outline

12.1 Ensuring that Occupational Radiation Exposures are As Low As Is Reasonably Achievable

PSAR The Clinch River ALARA program ensures that the preliminary facility design implements ALARA principles and is capable of minimizing public and radiological worker doses including:

- policy considerations for ensuring that occupational radiation exposures are ALARA, including organizational aspects, design and construction considerations, and decommissioning based on 10 CFR Part 20.1406 and applicable guidelines provided in RGs 1.8, 8.8, and 8.10, and ISG-06
- management commitment to implement an ALARA policy with the following major elements:
  - establishment of an ALARA program including requirements, procedures, goals, and expectations
  - personnel training in ALARA principles
  - implementation of ALARA principles in facility layout and design and in equipment specification documents
  - verification through reviews of implementing procedures and design documents
- description of how the design process incorporates lessons learned for ensuring ALARA including mechanisms in design reviews
- discussion of how ALARA design guidance is factored into the initial plant design to reduce maintenance needs, radiation levels, and minimize exposure of plant personnel
- methods used to develop detailed operational plans and procedures for ensuring that occupational radiation exposures are ALARA, including design changes and decommissioning
- establishment of ALARA objectives for contamination control and to minimize radioactive waste
- compliance with regulatory requirements in 10 CFR Part 20.1406 and applicable guidelines provided in RGs 1.8, 8.8, 8.10, and ISG-06

DCD The mPower standard plant ALARA program ensures that the detailed facility design implements ALARA principles and is capable of minimizing public and radiological worker doses including:

- policy considerations for ensuring that occupational radiation exposures are ALARA, including organizational aspects, design and construction considerations, and decommissioning based on 10 CFR Part 20.1406 and applicable guidelines provided in RGs 1.8, 8.8, and 8.10, and ISG-06

12.1 Ensuring that Occupational Radiation Exposures are As Low As Is Reasonably Achievable (cont.)

DCD (cont.)

- management commitment to implement an ALARA policy with the following major elements:
  - establishment of an ALARA program including requirements, procedures, goals, and expectations
  - personnel training in ALARA principles
  - implementation of ALARA principles in facility layout and design and in equipment specification documents
  - verification through reviews of implementing procedures and design documents
- description of how the design process incorporates lessons learned for ensuring ALARA including mechanisms in design reviews
- discussion of how ALARA design guidance is factored into the initial plant design to reduce maintenance needs, radiation levels, and minimize exposure of plant personnel
- methods used to develop detailed operational plans and procedures for ensuring that occupational radiation exposures are ALARA, including design changes and decommissioning
- establishment of ALARA objectives for contamination control and to minimize radioactive waste
- compliance with regulatory requirements in 10 CFR Part 20.1406 and applicable guidelines provided in RGs 1.8, 8.8, 8.10, and ISG-06
- identification of COL Items related to operational considerations for ensuring occupational radiation exposures are ALARA

FSAR Same contents as PSAR Section 12.1 including:

- additional details regarding management commitments, policies, and organizational structure to ensure occupational exposures are ALARA during plant operation and decommissioning
- compliance with design, procurement, and operational aspects of regulatory requirements in 10 CFR 20 and guidance in provided in RGs 1.8, 8.2, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36 and 8.38
- establishment of criteria and/or conditions under which operating procedures and techniques are provided to ensure that occupational radiation exposures are ALARA
- plans to perform construction inspections to verify that ALARA and shielding features are placed or installed as designed
- description of verification process used to ensure that ALARA and shielding features can perform their intended function
- description of specific exposure control techniques

12.2 Radiation Sources

PSAR A preliminary description of the sources of radiation for Clinch River during normal plant operations and accident conditions that provide the bases for radiation protection shielding design calculations and dose assessments is provided in PSAR Section 12.2 including:

- description of contained sources for both full power and shutdown operations, and accident conditions, including reactor core, spent fuel storage pool, and auxiliary systems
- description of sources that are contained in equipment in the radioactive waste management systems that are not described in Chapter 11
- sources of airborne radioactive material in equipment cubicles, corridors, and operating areas that are not described in Chapter 11
- models and parameters used for calculating airborne radioactivity concentrations
- description of airborne radioactive material sources, such as:
  - leakage in equipment cubicles, corridors, and operating areas
  - reactor vessel head removal
  - relief valve discharges
  - movement of spent fuel
- determination of concentrations of airborne radioactive material expected during normal operation, anticipated operational occurrence (AOO), and accident conditions for equipment cubicles, corridors, and operating areas
- table(s) providing radiation source parameters, including isotopic composition, source strength and geometry, and source location
- compliance with ANSI/ANS 18.1

DCD A detailed description of the sources of radiation for the mPower standard plant during normal plant operations and accident conditions that provide the bases for radiation protection shielding design calculations and dose assessments is provided in DCD Section 12.2 including:

- description of contained sources for both full power and shutdown operations, and accident conditions, including reactor core, spent fuel storage pool, and auxiliary systems
- description of sources that are contained in equipment in the radioactive waste management systems that are not described in Chapter 11
- sources of airborne radioactive material in equipment cubicles, corridors, and operating areas that are not described in Chapter 11
- models and parameters used for calculating airborne radioactivity concentrations
- description of airborne radioactive material sources, such as:
  - leakage in equipment cubicles, corridors, and operating areas
  - reactor vessel head removal
  - relief valve discharges
  - movement of spent fuel

12.2 Radiation Sources (cont.)

DCD (cont.)

- determination of concentrations of airborne radioactive material expected during normal operation, anticipated operational occurrence (AOO), and accident conditions for equipment cubicles, corridors, and operating areas
- table(s) providing radiation source parameters, including isotopic composition, source strength and geometry, and source location
- compliance with ANSI/ANS 18.1
- identification of a COL Information Item for an applicant to address any additional contained radiation sources not identified in DCD Section 12.2, including radiation sources used for instrument calibration or radiography

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FSAR Same contents as mPower standard plant DCD Section 12.2 including:

- updated information on expected use of additional contained radiation sources and/or radioactive materials, including radiation sources used for instrument calibration or radiography that will be addressed in a separate Part 30 license for byproduct material

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12.3 Radiation Protection Design Features

PSAR Preliminary radiation protection equipment and facility design features used to ensure that occupational radiation exposures are ALARA and to control access to radiologically restricted areas are described in PSAR Section 12.3, including the following:

- description of facility layout and equipment design features that support operations, as well as maintenance and inspection activities
- description of methods for reducing the production, distribution, and retention of activation products through design, material selection, water chemistry, decontamination procedures, etc.
- shielding design information for each of the radiation sources identified in PSAR Chapter 11 and Section 12.2, including the criteria for penetrations, materials, the method by which the shield parameters (attenuation coefficients, buildup factors, etc.) were determined, and the assumptions, codes, and techniques used in the shielding calculations
- description of plant vital areas based on requirements in 10 CFR 50.34(f)(2)(vii) and the criteria in Item II.B.2 of NUREG-0737
- description of ventilation systems that control the airborne concentration of radioactivity in equipment cubicles, corridors, and operating areas normally occupied by operating personnel
- air cleaning system design, including housings showing filter mountings, access doors, aisle space, service galleries, and provisions for testing, isolation, instrumentation, and decontamination
- description of how access to areas inside plant structures and plant yard areas is regulated and controlled by radiation zoning and access control
- plant arrangement drawings, as needed, of radiation zones, access controls, ingress and egress paths, etc.
- criteria for selection and placement of fixed area radiation monitoring instrumentation and continuous airborne radioactivity monitoring instrumentation
- description of area radiation and airborne radioactivity monitoring instrumentation
- description of in-containment high-range radiation monitoring capability following an accident based on requirements in 10 CFR 50.34(f)(2)(xvii) and the criteria in Item II.F.1 of NUREG-0737
- discussion of compliance with applicable regulatory requirements under 10 CFR 20.1406 and regulatory guidance, including concrete radiation shields, special protective features, and design features to minimize contamination and facilitate decommissioning
- tables providing:
  - radiation protection design features and parameters
  - shielding parameters
  - radiation zones
  - radiation monitor locations and service conditions

12.3 Radiation Protection Design Features (cont.)

PSAR (cont.)

- figures providing:
  - plant arrangement and post accident vital areas
  - radiation zone maps (normal and post accident – at different times)
  - radiation protection around the main control room
  - radiation protection around very high radiation areas such as the fuel transfer tube
  - typical air handing unit layout
  - location of radiation sources and fixed radiation monitors

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DCD

Detailed radiation protection equipment and facility design features used to ensure that occupational radiation exposures are ALARA and to control access to radiologically restricted areas in the mPower standard plant are described in DCD Section 12.3, including:

- description of facility layout and equipment design features that support operations, as well as maintenance and inspection activities
- description of methods for reducing the production, distribution, and retention of activation products through design, material selection, water chemistry, decontamination procedures, etc.
- shielding design information for each of the radiation sources identified in DCD Chapter 11 and Section 12.2, including the criteria for penetrations, materials, the method by which the shield parameters (attenuation coefficients, buildup factors, etc.) were determined, and the assumptions, codes, and techniques used in the shielding calculations
- description of plant vital areas based on requirements in 10 CFR 50.34(f)(2)(vii) and the criteria in Item II.B.2 of NUREG-0737
- description of ventilation systems that control the airborne concentration of radioactivity in equipment cubicles, corridors, and operating areas normally occupied by operating personnel
- air cleaning system design, including housings showing filter mountings, access doors, aisle space, service galleries, and provisions for testing, isolation, instrumentation, and decontamination
- description of how access to areas inside plant structures and plant yard areas is regulated and controlled by radiation zoning and access control
- criteria for selection and placement of fixed area radiation monitoring instrumentation and continuous airborne radioactivity monitoring instrumentation
- description of area radiation and airborne radioactivity monitoring instrumentation
- description of in-containment high-range radiation monitoring capability following an accident based on requirements in 10 CFR 50.34(f)(2)(xvii) and the criteria in Item II.F.1 of NUREG-0737

12.3 Radiation Protection Design Features (cont.)

DCD (cont.)

- demonstration of compliance with applicable regulatory requirements under 10 CFR 20.1406 and regulatory guidance, including concrete radiation shields, special protective features, and design features to minimize contamination and facilitate decommissioning
- tables providing:
  - radiation protection design features and parameters
  - shielding parameters
  - radiation zones
  - radiation monitor locations and service conditions
- figures providing:
  - plant arrangement and post accident vital areas
  - radiation zone maps (normal and post accident – at different times)
  - radiation protection around the main control room
  - radiation protection around very high radiation areas such as the fuel transfer tube
  - typical air handing unit layout
  - location of radiation sources and fixed radiation monitors

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FSAR Same contents as mPower standard plant DCD Section 12.3 including:

- demonstration of compliance with applicable regulatory requirements under 10 CFR 20.1406 and regulatory guidance, including concrete radiation shields, special protective features, and design features to minimize contamination and facilitate decommissioning
- description of use of portable instruments, including associated training and procedures, to determine airborne iodine concentrations in areas where plant personnel may be present during an accident
- updated plant arrangement drawings, as needed, of radiation zones, access controls, ingress and egress paths, etc.
- administrative controls and physical access controls to very high radiation areas (e.g., the reactor cavity and the fuel transfer tube)
- tables providing updated information:
  - radiation protection design features and parameters
  - shielding parameters
  - radiation zones
  - radiation monitor locations and service conditions
- figures providing updated information:
  - plant arrangement and post accident vital areas
  - radiation zone maps (normal and post accident – at different times)
  - radiation protection around the main control room
  - radiation protection around very high radiation areas such as the fuel transfer tube
  - typical air handing unit layout
  - location of radiation sources and fixed radiation monitors

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12.4 Dose Assessment

PSAR The objectives and criteria for design dose rates are discussed in PSAR Section 12.4 for Clinch River, including methods to estimate annual doses associated with operation, normal and special maintenance, radwaste handling, refueling, and inservice inspection. List typical job activities for these work categories. Preliminary information includes:

- list of typical job activities that would normally be performed along with the associated annual collective dose estimates
- list of expected number of personnel, occupancy times, and average dose rates used to determine the annual collective dose estimates
- estimated person-hours of occupancy and estimated personnel inhalation exposures for areas with expected airborne radioactivity concentrations during normal operation and AOO, including the bases, models, and assumptions used to determine these values
- mission dose (dose to access the area, perform necessary functions, and exit the area) for each vital area that may require occupancy to enable an operator to aid in mitigating or recovering from an accident
- design bases, models, and assumptions for vital area mission doses, including the occupancy time spent in each vital area and the post-accident dose rates for each vital area and the access route
- verification that the dose guidelines in GDC 19 are not exceeded during the course of an accident
- design bases, models, and assumptions for public and construction worker estimated dose values
- demonstration of compliance with applicable regulatory requirements and RG 8.19
- tables providing:
  - estimated man-hours of occupancy and estimated inhalation exposures to plant personnel in areas with expected airborne radioactivity concentrations
  - estimated annual dose at boundary of restricted area, site boundary and at various locations within the area where new SMR modules are constructed
  - estimated annual doses to construction workers due to radiation from existing operating SMR plant(s) and the annual man-rem doses associated with such construction, including models, assumptions, and input data
  - estimated annual doses to a member of the public

12.4 Dose Assessment (cont.)

DCD The objectives and criteria for design dose rates are discussed in DCD Section 12.4 for the mPower standard plant, including methods to estimate annual doses associated with operation, normal and special maintenance, radwaste handling, refueling, and inservice inspection. List typical job activities for these work categories. Detailed information includes:

- list of typical job activities that would normally be performed along with the associated annual collective dose estimates
- list of expected number of personnel, occupancy times, and average dose rates used to determine the annual collective dose estimates
- estimated person-hours of occupancy and estimated personnel inhalation exposures for areas with expected airborne radioactivity concentrations during normal operation and AOO, including the bases, models, and assumptions used to determine these values
- mission dose (dose to access the area, perform necessary functions, and exit the area) for each vital area that may require occupancy to enable an operator to aid in mitigating or recovering from an accident
- design bases, models, and assumptions for vital area mission doses, including the occupancy time spent in each vital area and the post-accident dose rates for each vital area and the access route
- verification that the dose guidelines in GDC 19 are not exceeded during the course of an accident
- design bases, models, and assumptions for public and construction worker estimated dose values
- demonstration of compliance with applicable regulatory requirements and RG 8.19
- tables providing:
  - estimated man-hours of occupancy and estimated inhalation exposures to plant personnel in areas with expected airborne radioactivity concentrations
  - estimated annual dose at boundary of restricted area, site boundary and at various locations within the area where new SMR modules are constructed
  - estimated annual doses to construction workers due to radiation from existing operating SMR plant(s) and the annual man-rem doses associated with such construction, including models, assumptions, and input data
  - estimated annual doses to a member of the public

12.4 Dose Assessment (cont.)

FSAR Same contents as mPower standard plant PSAR and DCD 12.4 Sections, updated to include:

- site specific estimated annual doses to construction workers in a new unit construction area, as a result of radiation from onsite radiation sources from existing operating plants
- site specific estimated annual doses to onsite personnel and the offsite members of the public as a result of radiation from onsite radiation sources and activities outside the scope of the DC
- tables providing updated information:
  - estimated annual total body dose and maximum organ dose to a member of the public
  - estimated annual doses to construction workers as a result of radiation from onsite radiation sources including existing operating plants

12.5 Operational Radiation Protection Program

PSAR A preliminary description of the Operational Radiation Protection Program for the Clinch River Plant is described in PSAR Section 12.5. The program will be implemented through procedures and work controls that ensure that radiation protection measures are employed for the protection of workers, the public, and the environment including:

- description of the administrative organization, procedures and training, including the authority and responsibility of each position identified
- criteria for selection of fixed and portable laboratory equipment and instrumentation for performing surveys, for airborne monitoring and sampling, for area monitoring, and for personnel monitoring
- description of the instrument storage, calibration, and maintenance facilities
- description and location of the radiation protection facilities
- discussion on conformance to applicable requirements in 10 CFR Parts 20, 70 and 71, NUREG/CR-0041, and RGs 1.8, 1.16, 1.33, 1.39, 1.97, 8.2, 8.3, 8.4, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.29, 8.32, 8.34, 8.35, 8.36, 8.38

DCD The Operational Radiation Protection Program is site-specific and is briefly addressed as part of the mPower standard plant design. DCD Section 12.5 describes the ORPP and identifies appropriate COL Applicant items.

FSAR Same contents as Clinch River PSAR Section 12.5 updated to include

- detailed information related to implementation of an Operational Radiation Protection Program, including organizational aspects, facilities, instrumentation and equipment, procedures, and training
- description of implementing procedures and work controls necessary to ensure that radiation protection measures are employed for the protection of workers, the public, and the environment including:
  - describe management and staff authorities and responsibilities per 10 CFR 20.1101 and 20.2102
  - describe the portable air sampling and analysis system ability to determine airborne radionuclide concentrations during and following an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737
- conformance to applicable requirements in 10 CFR Parts 20, 70 and 71, NUREG/CR-0041, and RGs 1.8, 1.16, 1.33, 1.39, 1.97, 8.2, 8.3, 8.4, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.29, 8.32, 8.34, 8.35, 8.36, 8.38

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
13.1 Organizational Structure of Applicant	PSAR	10 CFR 50.34 10 CFR 50.40(b) 10 CFR 50.54(j)-(m)	None	RG 1.70	13.1.1 13.1.2 13.1.3	RGs 1 8, 1.33, 1.68, 1.114, 1.189 GL 86-04 NUREG-0711 NUREG-0718, Item II J.3 1 NUREG-0737, TMI Items I.A 1.1 and I.A.1.3	ANSI/ANS 3.1-1993 ANSI N18.7-1976/ANS-3 2	Yes - Additional information will be provided consistent with current requirements issued subsequent to RG 1.70	No	None	9 5, 12 5, 13.2, 13 3, 13 4, 13 5, 13 6, 17, 18
	DCD	10 CFR 50 34 10 CFR 50 40(b) 10 CFR 50 54(j)-(m)	None	RG 1 206	13 1.1 13.1.2 13.1.3	RGs 1.8, 1 33, 1 68, 1.114, 1.189 GL 86-04 NUREG-0711 NUREG-0718, Item II.J 3.1 NUREG-0737, TMI Items I A 1.1 and I A.1.3 IMC-2504	ANSI/ANS 3.1-1993 ANSI N18.7-1976/ANS-3.2 NEI Position Paper on Control Room Staffing for Small Reactors, dated September 23, 2011 ANS Report, "Interim Report of the American Nuclear Society President's Special Committee on Small and Medium Sized Reactor (SMR) Generic Licensing Issues," July 2010	N/A	N/A	Generation mPower to develop justification to support an applicant's submittal of an exemption request to reduce Control Room Staffing	9 5, 12.5, 13 2, 13 3, 13.4, 13 5, 13.6, 17, 18
	FSAR	10 CFR 50 34 10 CFR 50 40(b) 10 CFR 50.54(j)-(m)	10 CFR 50.54(m)	RG 1.206	13 1.1 13.1.2 13.1.3	RGs 1 8, 1.33, 1.68, 1.114, 1.189 GL 86-04 NUREG-0694, Items I.B.1.2 and I C.3 NUREG-0711 NUREG-0718, Item II.J.3.1 NUREG-0737, TMI Items I.A.1.1 and I A 1 3 NUREG-1791 NUREG/CR-6838 SECY-11-098	ANSI/ANS 3.1-1993 ANSI N18.7-1976/ANS-3 2 NEI Position Paper on Control Room Staffing for Small Reactors, dated September 23, 2011 ANS Report, "Interim Report of the American Nuclear Society President's Special Committee on Small and Medium Sized Reactor (SMR) Generic Licensing Issues," July 2010	N/A	No	Generation mPower to develop justification to support an applicant's submittal of an exemption request to reduce Control Room Staffing	9 5, 12 5, 13 2, 13 3, 13 4, 13 5, 13.6, 17, 18

Note 1: RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
	PSAR	10 CFR 19.12 10 CFR 26 21 10 CFR 50 34 10 CFR 50.40(b) 10 CFR 50 54(i - m) 10 CFR 50.120 10 CFR 50, App E 10 CFR 55 4 10 CFR 55.31 10 CFR 55 41 10 CFR 55 43 10 CFR 55 45 10 CFR 55 46 10 CFR 55 59	None	RG 1.70	13 2.1 13 2.2	RGs 1.8, 1.149, 1.189 GL 86-04 NUREGs-0711, -1021, -1220	ANSI/ANS 3 1-1993 NEI 06-13A	Yes - Additional information will be consistent with current requirements	No	None	9 5, 12.5, 13.1, 13.3, 13 4, 13 5, 13 6, 18
13 2 Training	DCD	10 CFR 19.12 10 CFR 26 21 10 CFR 50.34 10 CFR 50.40(b) 10 CFR 50 54(i - m) 10 CFR 50.120 10 CFR 50, App E 10 CFR 52.78 10 CFR 55 4 10 CFR 55.31 10 CFR 55 41 10 CFR 55 43 10 CFR 55 45 10 CFR 55 46 10 CFR 55 59	None	RG 1.206	13 2.1 13.2.2	RGs 1 8, 1.149, 1.189 GL 86-04 NUREGs-0711, -1021, -1220 IMC-2504	ANSI/ANS 3.1-1993 NEI 06-13A	N/A	N/A	None	9 5.1, 12.5, 13.1, 13.3, 13 4, 13 5, 13 6, 18
	FSAR	10 CFR 19 12 10 CFR 26 21 10 CFR 50 34 10 CFR 50 40(b) 10 CFR 50 54(i - m) 10 CFR 50.120 10 CFR 50, App E 10 CFR 55 4 10 CFR 55 31 10 CFR 55 41 10 CFR 55 43 10 CFR 55 45 10 CFR 55 46 10 CFR 55.59	None	RG 1 206	13.2 1 13 2 2	RGs 1 8, 1.149, 1.189 GL 86-04 NUREGs-0711, -1021, -1220	ANSI/ANS 3 1-1993 NEI 06-13A	N/A	No	None	9.5, 12 5, 13 1, 13 2, 13 4, 13 5, 13 6, 18

Note 1: RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
13.3 Emergency Planning	PSAR	10 CFR 50.33 10 CFR 50.34 10 CFR 50.47 10 CFR 50.54 10 CFR 50.72 10 CFR 50, App A 10 CFR 50, App. E 10 CFR 73.71 10 CFR 100, 100.1, 100 3, 100 20, 100 21(g) 44 CFR 350 44 CFR 351, 352, and 353	None	RG 1 206	13 3	RGs 1.23, 1 97, 1 101, 4.7, 5 62 NUREGs-0396, -0654, -0660, -0696, -0711, -0718, -0737, -0814, -0835, -0933, -1022, -1394, -1793 NUREG/CR-4831, -6863, -6864 BLs 79-18, 80-15, 05-02 IEC 80-09 GLs 82-33, 91-14 RISs 2000-08, 2000-11, 2001-16, 2002-01, 2002-16, 2002-21, 2003-12, 2003-18, 2004-13, 2004-15, 2005-02, 2005-08, 2006-03, 2006-12 ALs 94-04, 94-07, 94-16 INs 81-34, 85-41, 85-44, 85-52, 85-80, 86-18, 96-43, 86-55, 86-98, 87-54, 87-58, 88-15, 89-72, 90-74, 91-64, 91-77, 92-32, 92-38, 93-53, 93-81, 93-94, 94-27, 95-23, 95-48, 96-19, 97-05, 98-20, 02-14, 02-25, 04-19, 05-06, 05-19 SECY-91-041, SECY 05-197 (and SRM on SECY-05-0197) EPPOS 1, 2, 3, 5 NSIR/DPR-ISG-01	NEI 99-01	Yes - PSAR will reflect additional emergency planning requirements issued since RG 1.70	No	Track NRC Issuance of draft NSIR/DPR-ISG-01, Interim Staff Guidance on Emergency Planning for Nuclear Power Plants, and proposed Revision 2 to RG 5 62 (DG-5062) Additional emergency preparedness regulatory actions may be needed as a result of Fukishima Lessons Learned	2.1, 2.3, 2 4.13, 6 4, 7.5, 9 5 2, 12.3, 12 4, 13 2, 13 4, 13 6, 18, and Part 4
	DCD	10 CFR 50.33 10 CFR 50.34 10 CFR 50 47 10 CFR 50.54 10 CFR 50.72 10 CFR 50, App. A 10 CFR 50, App. E 10 CFR 52 47 10 CFR 52 48 10 CFR 73.71 10 CFR 100, 100 1, 100.3, 100 20, 100 21(g) 44 CFR 350, 351, 352, and 353	None	RG 1 206	13 3	RGs 1.23, 1.97, 1.101, 4.7, 5.62 NUREGs-0396, -0654, -0660, -0696, -0711, -0718, -0737, -0814, -0835, -0933, -1022, -1394, -1793 NUREG/CR-4831, -6863, -6864 BLs 79-18, 80-15, 05-02 IEC 80-09 GLs 82-33, 91-14 RISs 2000-08, 2000-11, 2001-16, 2002-01, 2002-16, 2002-21, 2003-12, 2003-18, 2004-13, 2004-15, 2005-02, 2005-08, 2006-03, 2006-12 ALs 94-04, 94-07, 94-16 INs 81-34, 85-41, 85-44, 85-52, 85-80, 86-18, 96-43, 86-55, 86-98, 87-54, 87-58, 88-15, 89-72, 90-74, 91-64, 91-77, 92-32, 92-38, 93-53, 93-81, 93-94, 94-27, 95-23, 95-48, 96-19, 97-05, 98-20, 02-14, 02-25, 04-19, 05-06, 05-19 SECY-91-041, SECY 05-197 (and SRM on SECY-05-0197) EPPOS 1, 2, 3, 5 NSIR/DPR-ISG-01	NEI 99-01	N/A	N/A	Track NRC Issuance of draft NSIR/DPR-ISG-01, Interim Staff Guidance on Emergency Planning for Nuclear Power Plants, and proposed Revision 2 to RG 5 62 (DG-5062) Additional emergency preparedness regulatory actions may be needed as a result of Fukishima Lessons Learned	6 4, 7.5, 9 5, 12.3, 12.4, 13 2, 13.4, 13.6, 14.3, 18
	FSAR	10 CFR 50.33 10 CFR 50 34 10 CFR 50 47 10 CFR 50 54 10 CFR 50.72 10 CFR 50, App A 10 CFR 50, App E 10 CFR 73.71 10 CFR 100, 100 1, 100 3, 100 20, 100 21(g) 44 CFR 350, 351, 352, and 353	None	RG 1 206	13.3	RGs 1.23, 1.97, 1.101, 4 7, 5 62 NUREGs-0396, -0654, -0660, -0696, -0711, -0718, -0737, -0814, -0835, -0933, -1022, -1394, -1793 NUREG/CR-4831, -6863, -6864 BLs 79-18, 80-15, 05-02 IEC 80-09 GLs 82-33, 91-14 RISs 2000-08, 2000-11, 2001-16, 2002-01, 2002-16, 2002-21, 2003-12, 2003-18, 2004-13, 2004-15, 2005-02, 2005-08, 2006-03, 2006-12 ALs 94-04, 94-07, 94-16 INs 81-34, 85-41, 85-44, 85-52, 85-80, 86-18, 96-43, 86-55, 86-98, 87-54, 87-58, 88-15, 89-72, 90-74, 91-64, 91-77, 92-32, 92-38, 93-53, 93-81, 93-94, 94-27, 95-23, 95-48, 96-19, 97-05, 98-20, 02-14, 02-25, 04-19, 05-06, 05-19 SECY-91-041, SECY 05-197 (and SRM on SECY-05-0197) EPPOS 1, 2, 3, 5 NSIR/DPR-ISG-01	NEI 99-01	N/A	No	Track NRC Issuance of draft NSIR/DPR-ISG-01, Interim Staff Guidance on Emergency Planning for Nuclear Power Plants, and proposed Revision 2 to RG 5 62 (DG-5062) Additional emergency preparedness regulatory actions may be needed as a result of Fukishima Lessons Learned	2.1, 2 3, 2.4.13, 6 4, 7.5, 9.5 2, 12.3, 12.4, 13.2.2, 13 4, 13 6, 18, and Part 4

Note 1: RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
13 4 Operational Program Implementation	PSAR	10 CFR 50 54(p) 10 CFR 50 90	None	RG 1 206	13 4	SECY-05-0197 SRM on SECY-05-0197	ANSI N18.7-1976/ANS-3 2	Yes - Additional information on implementation of an Operational Program will be provided consistent with current requirements	No	None	17.5
	DCD	10 CFR 50 54(p) 10 CFR 50.90	None	RG 1 206	13 4	SECY-05-0197 SRM on SECY-05-0197	ANSI N18 7-1976/ANS-3 2	N/A	N/A	None	14.3, 17.5
	FSAR	10 CFR 50 54(p) 10 CFR 50.90	None	RG 1 206	13 4	SECY-05-0197 SRM on SECY-05-0197	ANSI N18 7-1976/ANS-3 2	N/A	No	None	17.5
13.5 Plant Procedures	PSAR	10 CFR 26 10 CFR 50 34 10 CFR 50 40 10 CFR 50 54 10 CFR 50 65 10 CFR 50, App. A, GDC 1 10 CFR 50, App. B, Criteria V, VI, XI 10 CFR 73 58	None	RG 1.70	13.5.1.1 13.5 2.1	RGs 1.33, 1.114 NUREGs-0578, -0660, -0694, -0711, -0737 and Supplement 1, -0899, -1358 and Supplement 1 GLs 82-33, 83-05, 83-22, 83-23, 83-28, 83-31, 90-03	ANSI/ANS 3 1-1993 ANSI N18.7-1976/ANS-3 2 ASME B30 2-2005	Yes - Additional information on plant procedures will be provided consistent with current requirements	No	Track NRC issuance of draft SRP 13 5.2 2, "Maintenance and Other Operating Procedures"	17 4, 17.6, 18
	DCD	10 CFR 26 10 CFR 50.34 10 CFR 50.40 10 CFR 50 54 10 CFR 50.65 10 CFR 50, App. A, GDC 1 10 CFR 50, App B, Criteria V, VI, XI 10 CFR 52 10 CFR 73 58	None	RG 1 206	13 5 1.1 13.5 2.1	RGs 1.33, 1.114 NUREGs-0578, -0660, -0694, -0711, -0737 and Supplement 1, -0899, -1358 and Supplement 1 GLs 82-33, 83-05, 83-22, 83-23, 83-28, 83-31, 90-03	ANSI/ANS 3.1-1993 ANSI N18.7-1976/ANS-3 2 ASME B30 2-2005	N/A	N/A	Track NRC issuance of draft SRP 13.5.2 2, "Maintenance and Other Operating Procedures"	14 3, 17 4, 17 6, 18
	FSAR	10 CFR 26 10 CFR 50 34 10 CFR 50 40 10 CFR 50 54 10 CFR 50 65 10 CFR 50, App. A, GDC 1 10 CFR 50, App. B, Criteria V, VI, XI 10 CFR 73 58	None	RG 1.206	13.5 1.1 13 5.2.1	RGs 1.33, 1.114 NUREGs-0578, -0660, -0694, -0711, -0737 and Supplement 1, -0899, -1358 and Supplement 1 GLs 82-33, 83-05, 83-22, 83-23, 83-28, 83-31, 90-03	ANSI/ANS 3 1-1993 ANSI N18.7-1976/ANS-3.2 ASME B30.2-2005	N/A	No	Track NRC issuance of draft SRP 13.5 2 2, "Maintenance and Other Operating Procedures"	17 4, 17.6, 18

Note 1: RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
13 6 Security	PSAR	10 CFR 11.11 10 CFR 26 10 CFR 50 34, 54, 70 10 CFR 73.1 10 CFR 73 2 10 CFR 73 21 10 CFR 73 22 10 CFR 73 23 10 CFR 73 54, 55, 56, 57, 58 10 CFR 73 70, 71 10 CFR 73, App. A, B, C, G, and H 10 CFR 74 10 CFR 100 10 CFR 100 21	None	RG 1.206	13 6	RGs 1.91, 4.7, 5.7, 5 44, 5 54, 5 65, 5 66, 5 68, 5.69, 5.71, 5.74, 5.75, 5.76, 5 77 DGs-5019, -5035 (SRI) NUREG-0509, -1226, -1267 NUREG/CR-1198, -1345, -1381, -2585, -2643, -4250, -6190 RISs 2003-06, 2005-09 IMC-2201 DC/COL-ISG-016	NEI 03-09 NEI 03-12 NEI 08-09 NEI 09-01	Yes - PSAR will address current regulatory requirements on security issued since RG 1.70	No	A determination is needed to establish whether or not a Security Plan is required at the construction permit stage. 10 CFR 50.34 indicates it's needed with the FSAR, not the PSAR. Approval of mPower Security Topical Reports. NRC acceptance of proposed security staffing for mPower. Track NRC Issuance of draft RGs DG-5019 and DG-5035 (SRI)	13 4 and Part 5
	DCD	10 CFR 11.11 10 CFR 26 10 CFR 50.34, 54 10 CFR 52 47(b)(1) 10 CFR 73.1 10 CFR 73.2 10 CFR 73 21 10 CFR 73.22 10 CFR 73 23 10 CFR 73 45(g)(4)(i) 10 CFR 73.54, 55, 56, 57, 58 10 CFR 73.70, 71 10 CFR 73, App. A, B, C, G, and H 10 CFR 74 10 CFR 100 10 CFR 100 21	None	RG 1 206	9.5 2, 9.5 3, 13.6, 13 6 2, 13 6 6, 14.3.12	RGs 1.91, 4.7, 5.7, 5.12, 5 44, 5.54, 5 62, 5.65, 5 68, 5 69, 5.71, 5.74, 5.76, 5 77, 5.79 Draft DGs-5019, -5035 (SRI) NUREG-0509, -1226, -1267 NUREG/CR-1198, -1345, -1381, -2585, -2643, -4250, -6190 RISs 2003-06, 2005-04 IMC-2201 IMC-2504 DC/COL-ISG-016	NEI 03-12 NEI 08-09	N/A	N/A	Approval of mPower Security Technical/Topical Reports Track NRC Issuance of draft RGs DG-5019 and DG-5035 (SRI)	9 5 2, 9 5 3, 13 4, 13 6.1, 13 6 2, 13 6 3, 14.3
	FSAR	10 CFR 11.11 10 CFR 26 10 CFR 50 34, 54 10 CFR 50.70 10 CFR 73.1 10 CFR 73.12 10 CFR 73.21 10 CFR 73 22 10 CFR 73.23 10 CFR 73.54, 55, 56, 57, 58 10 CFR 73 70, 71 10 CFR 73, App. A, B, C, G, and H 10 CFR 74 10 CFR 100 10 CFR 100 21	None	RG 1.206	13 6 13 6.1 13 6 6	RGs 1.91, 4.7, 5.7, 5.44, 5.54, 5.65, 5.66, 5 68, 5 69, 5.71, 5.74, 5.75, 5.76, 5.77 DGs-5019, -5035 (SRI) NUREG-0509, -1226, -1267 NUREG/CR-1198, -1345, -1381, -2585, -2643, -4250, -6190 RISs 2003-06, 2005-09 IMC-2201 DC/COL-ISG-016	NEI 03-09 NEI 03-12 NEI 06-12 NEI 08-09 NEI 09-01	N/A	No	Approval of mPower Security Topical Reports. NRC acceptance of proposed security staffing for mPower. Track NRC Issuance of draft RGs DG-5019 and DG-5035 (SRI)	13 4 and Part 5

Note 1: RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
13.7 Fitness-For-Duty	PSAR	10 CFR 26	None	RG 1.206	N/A	GL 91-16 IN 09-28 SECY-00-0022 SECY-05-0074 SECY-06-0244 RIS 2011-08	NEI 06-06	Yes - PSAR will address current requirements on fitness-for-duty issued subsequent to RG 1.70	No	Track NRC issuance of draft SRP 13 7, "Fitness-for-Duty," and 13.7 2, "Fitness-for-Duty (Construction)"	13 4, 13.5
	DCD	10 CFR 26	None	RG 1.206	N/A	GL 91-16 IN 09-28 SECY-00-0022 SECY-05-0074 SECY-06-0244 RIS 2011-08	N/A	N/A	N/A	None	13 4, 13 5, 14.3
	FSAR	10 CFR 26	None	RG 1.206	N/A	RG 5 73 GL 91-16 IN 09-28 SECY-00-0022 SECY-05-0074 SECY-06-0244 RIS 2011-08	NEI 06-11	N/A	No	Track NRC issuance of draft SRP 13 7, "Fitness-for-Duty," and 13.7.2, "Fitness-for-Duty (Construction)"	13 4, 13 5

Note 1: RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Chapter 13 Outline

13 Conduct of Operations

13.1 Organizational Structure of Applicant

PSAR PSAR Section 13.1 provides information related to operational plans for the Clinch River SMR Plant to provide reasonable assurance that the plant's organization will be of sufficient size and technical competence to conduct operations in a manner that protects public health and safety, including the following information:

- description of the management and technical support organization
- the various organization's (e.g., applicant, NSSS supplier, architect/engineer) past experiences in the design, construction, and operation of nuclear power plants and past experience in activities of similar scope and complexity
- description of the operating organization
- description of the qualifications of plant personnel

DCD DCD Section 13.1 identifies the following COL Information Items related to operational plans for the mPower standard plant design to provide reasonable assurance that a COL applicant's organization will be of sufficient size and technical competence to conduct plant operations in a manner that protects public health and safety:

- description of management and technical support organization, including functional interfaces and responsibilities
- description of the organization's past experience in the design, construction, and operation of nuclear power plants and past experience in activities of similar scope and complexity
- description of the operating organization, including information that the organizational structure is consistent with the human-system interface design assumptions described in Chapter 18
- description of the qualifications of plant personnel for identified positions or classes of positions

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FSAR FSAR Section 13.1 provides updated information from PSAR Section 13.1 with the following supplemental information:

- documentation that demonstrates that the organizational structure is consistent with the human-system interface design assumptions described in Chapter 18
- organizational charts reflecting corporate and site staffing
- identification of principal contract relationships and responsibilities
- details of the structure, functions, and responsibilities of the onsite organization established to operate and maintain the plant

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Chapter 13 Outline

13.2 Training

PSAR PSAR Section 13.2 provides a description of the training program development process for licensed reactor operators, senior reactor operators, and non-licensed personnel, including the following information:

- reference to NEI 06-13A, Template for an Industry Training Program Description
- reference to PSAR Section 13.4 for schedule milestones for training implementation

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DCD DCD Section 13.2 provides reference to a designer's input guidelines for the mPower standard plant on how training insights will be passed from the designer to the operator, including addressing the elements of the training program for the operations personnel who participate as subjects in Human Factors Engineering validation and verification. In addition, DCD Section 13.2 identifies the following COL Information Items for an applicant to provide:

- description of the reactor operator training program
- description of the non-licensed plant staff training program
- reference to NEI 06-13A, Template for an Industry Training Program Description, which provides a complete generic training program description for use with COL applications

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FSAR Same contents as DCD Section 13.2 with updated information from PSAR Section 13.2 and the following supplemental information:

- details on the education, training, and experience requirements established for each management, operating, technical, and maintenance position category
- proposed means for evaluating the training program effectiveness
- contingency plans for individuals applying for licenses in case fuel loading is delayed
- details on the schedule of each part of the training program for each functional group of employees in the organization in relation to preoperational testing, fuel loading, initial core criticality, and operations

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Chapter 13 Outline

13.3 Emergency Planning

PSAR PSAR Section 13.3 provides the proposed plans for coping with emergencies in accordance with applicable regulatory requirements and guidance, including the following information:

- reference to Part 4 of the Construction Permit Application for a site-specific Emergency Plan, including:
    - proposed design features, facilities, functions, and equipment necessary for implementation of the emergency plan
    - conformance to the requirements in Appendix E to CFR Part 50
    - provisions for establishing a Technical Support Center (TSC) at the Clinch River Site consistent with the intent of the applicable functional criteria in NUREG-0696, including HVAC design, radiation dose criteria, staffing, and communication interface requirements
    - provisions for establishing a Operations Support Center (OSC) at the Clinch River site
    - description of an Emergency Operations Facility (EOF), including staffing provisions consistent with NUREG-0654/FEMA-REP-1
    - approach for implementation of the Emergency Response Data System (ERDS) in accordance with 10 CFR 50, Appendix E and the Safety Parameter Display System (SPDS) designed consistent with the functional criteria in NUREG-0654/FEMA REP-1
    - provisions for establishing data communications between the Control Room, TSC, EOF, OSC, NRC, and State and local authorities, as appropriate
    - approach for a comprehensive emergency plan for a multi-unit site
    - approach for an emergency classification and action level scheme in accordance with 10 CFR 50.47(b)(4)
    - reference to PSAR Chapter 2 for a discussion on how the exclusion area and low population zone (LPZ) for the Clinch River site comply with 10 CFR Part 100
- reference to PSAR Section 13.4 for schedule milestones for emergency planning implementation

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DCD DCD Section 13.3 provides a description of mPower standard plant design features, facilities, functions, and equipment necessary for implementation of an applicant's emergency plan, including identification of the following COL Information Items:

- development of a site-specific Emergency Plan in accordance with 10 CFR 50.47 and 10 CFR 50, Appendix E
- description of a TSC consistent with the intent of the applicable functional criteria in NUREG-0696, including HVAC design, radiation dose criteria, staffing, and communication interfaces requirements

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13.3 Emergency Planning (cont.)

DCD (cont.)

- provisions for establishing a OSC, EOF, and associated communication interfaces (e.g., ERDS and SPDS) between the Control Room, TSC, OSC, and NRC, and to State and local authorities, as appropriate
- extent of compliance to 10 CFR 100 with respect to exclusion area boundary and the low population zone

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FSAR Same contents as DCD Section 13.3 with updated information from PSAR Section 13.3 and the following supplemental information:

- reference to Part 4 of the Operating License Application for a comprehensive site-specific Emergency Plan, including information to address compliance with regulatory requirements and identification of emergency planning implementing procedures (EPIPs)
- description of extent to which existing emergency plan and emergency provisions are credited to the new emergency plan, for multi-unit sites

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Chapter 13 Outline

13.4 Operational Program Implementation

PSAR PSAR Section 13.4 provides a description of operational programs that are required by regulations, as defined in SECY-05-0197, and the schedule for the implementation as described below:

- commitments for implementation of operational programs that are required by regulation
- table that provides reference to applicable PSAR Sections that includes preliminary descriptions of the operational programs (as applicable) at a functional level and preliminary implementation milestones for the operational programs

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DCD DCD Section 13.4 identifies the following COL Information Items for an applicant to address the operational programs that are required by regulations as defined in SECY-05-0197:

- descriptions of the operational programs at a functional level
- implementation milestones for the operational programs

FSAR FSAR Section 13.4 provides updated information from PSAR Section 13.4 with the following supplemental information:

- table of operational programs required by NRC regulation and program implementation milestones (e.g., prior to radioactive sources on site, prior to fuel on site, prior to fuel load, at fuel load, prior to exceeding 5-percent power, prior to first shipment of radioactive waste, etc.)

Chapter 13 Outline

13.5 Plant Procedures

PSAR PSAR Section 13.5 provides a description of the administrative and operating procedures that the operating plant staff uses to assure that routine operating, off-normal and emergency activities are conducted in a safe manner, based on the following information:

- brief description of the process for the development, review and approval of procedures, and revisions, prior to use
- listing and preliminary schedule for the preparation of appropriate written administrative procedures, plant operating, maintenance, and emergency operating procedures

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DCD DCD Section 13.5 provides reference to designer's input guidelines for the mPower standard plant that provides generic information on the design and development of the plant operating procedures, including emergency response guidelines and emergency operating procedures. In addition, DCD Section 13.5 identifies the following COL Information Items for an applicant to address administrative and operating procedures that the operating plant staff will use to assure that routine operating, off-normal and emergency activities are conducted in a safe manner for the mPower standard plant, including:

- description of process for the development, review and approval of procedures, and revisions prior to use
- nature and content, and the schedule for the preparation, of appropriate written administrative procedures
- description of plant operating, maintenance, and emergency operating procedures, including procedure classification system

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FSAR Same contents as DCD Section 13.5 with updated information from PSAR Section 13.5 and the following supplemental information:

- updated description of plant operating, administrative, maintenance, and emergency operating procedures, including procedure classification system
- reference to Chapter 17 for Quality Assurance Program that governs procedural document control

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Chapter 13 Outline

13.6 Security

PSAR PSAR Section 13.6 provides reference to Part 5 of the Clinch River SMR Plant Construction Permit Application (CPA), which includes:

- elements of the individual security plans (e.g., Physical Security Plan, the Training and Qualification Plan, and the Safeguards Contingency Plan), as required by 10 CFR 73
- reference to NEI 09-01, "Security Measures During New Reactor Construction Template" (formerly NEI 03-12, Appendix F)
- commitment to provide the following separate licensing documents when the Operating License Application is submitted:
  - a Cyber Security Plan based on Regulatory Guide 5.71 and NEI 08-09, "Cyber Security Plan Template,"
  - a document called the "Loss of Large Areas of the Plant Mitigative Strategies Description," which addresses large fires and explosions as required by 10 CFR 50.54(hh)(2) based on Interim Staff Guidance ISG-016 and NEI 06-12, "B.5.b Phase 2 & 3 Submittal Guideline Template"

DCD DCD Section 13.6 addresses the following information regarding the elements of the individual security plans (e.g., physical security, training and qualification, and safeguards contingency) for the mPower standard plant:

- description of design features for the mPower plant that enhance security as described in B&W Technical Report R0003-08-002708-000(P), "Physical Security Design and Program Considerations"
- reference to the mPower standard plant security assessment as described in B&W Technical Report XXX, "Security Assessment, Safeguards Information," which addresses identification of vital equipment, development of target sets, vulnerability assessments, defensive analyses, design features to enhance security, portions of NRC Orders to current operating plants that impact the B&W mPower plant design, and other security features that establish the security system design
- identification of COL Information Items for an applicant to address, including:
  - development of a site-specific security assessment and security plan that adequately demonstrates how the performance requirements are met for the initial implementation of the security program
  - development of a Cyber Security Plan

FSAR Same contents as DCD Section 13.6 and updated information from PSAR Section 13.6 with the following supplemental information:

Chapter 13 Outline

FSAR (cont.)

- reference to Part 5 of the Clinch River SMR Plant Operating License Application, which provides a description of the elements of the individual security plans (i.e., Physical Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan), as required by 10 CFR 73
- reference to separate licensing document called the Cyber Security Plan as required by 10 CFR 73 based on NEI 08-09, "Cyber Security Plan Template"
- reference to separate licensing document called the "Loss of Large Areas of the Plant Mitigative Strategies Description," which addresses large fires and explosions as required by 10 CFR 50.54(hh)(2) (Originally B.5.b. Order) based on NEI 06-12 "B.5.b Phase 2 & 3 Submittal Guideline Template," and Interim Staff Guidance ISG-016 "Compliance with 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d) Loss of Large Areas of the Plant due to Explosions or Fires from a Beyond-Design Basis Event"

Chapter 13 Outline

13.7 Fitness-For-Duty

PSAR PSAR Section 13.7 provides the following information regarding implementation of the Fitness-For-Duty (FFD) Program for the Clinch River SMR Plant:

- description of the FFD Program during construction as required by 10 CFR Part 26, including reference to Section 13.4 for implementation plans
- reference to NEI 06-06, Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites

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DCD DCD Section 13.7 identifies the following COL Information Item for an applicant to provide:

- description of a FFD Program and its implementation as required by 10 CFR Part 26

FSAR FSAR Section 13.7 provides the following information regarding implementation of the FFD for the Clinch River SMR Plant:

- description of the FFD Program during plant operations as required by 10 CFR Part 26, including reference to Section 13.4 for implementation plans

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Clinch River Regulatory Framework Document
NRC Version

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
Part 5: Safeguards and Security Plans	CPA	10 CFR 11.11 10 CFR 26, Subpart K 10 CFR 50.34 10 CFR 50.54 10 CFR 73.1 10 CFR 73.2 10 CFR 73.21 10 CFR 73.22 10 CFR 73.23 10 CFR 73.54 10 CFR 73.55 10 CFR 73 56 10 CFR 73 57 10 CFR 73.58 10 CFR 73.70 10 CFR 73.71 10 CFR 73, App. A 10 CFR 73, App. B 10 CFR 73, App. C 10 CFR 73, App. G 10 CFR 73, App. H 10 CFR 74 10 CFR 100 10 CFR 100.21	None	RG 1.206	2.1.1, 2.1.2, 13.1.1, 13.1.2, 13.1.3, 13.5.1.1, 13 6, 13 6.1 14.3.12	RGs 1.91, 4.7, 5.7, 5.12, 5.20, 5.43, 5.44, 5.54, 5.62, 5.65, 5 66, 5 68, 5.69, 5.71, 5.73, 5.74, 5.75, 5.76, 5.77, 5.79 Draft DG-5019 NUREG-0509, -1226, -1267 NUREG/BR-0252 NUREG/CR-0485, -1198, -1345, -1381, -2585, -2643, -4250, -6190 RISs 2003-06, 2005-04 IMC-2201	NUMARC 89-01 NEI 03-09, 03-12, 06-11, 09-01	Yes - CPA will contain updated security requirements based on current regulations	N/A	A determination is needed to establish whether or not a Security Plan is required at the construction permit stage. 10 CFR 50.34 indicates it's needed with the FSAR, not the PSAR. Approval of mPower Security Technical/Topical Reports. NRC acceptance of proposed security staffing for mPower.	2.1.1, 2.1.2, 13.1.1, 13.1.2, 13.3.3, 13 4, 13.5.1.1, 13.6.1, 13.6.2, 13.6.3
	DCD (Section 13.6)	10 CFR 11.11 10 CFR 26 10 CFR 50.34 10 CFR 50.54 10 CFR 52.47(b)(1) 10 CFR 73.1 10 CFR 73 2 10 CFR 73 21 10 CFR 73 22 10 CFR 73 23 10 CFR 73.45(g)(4)(i) 10 CFR 73.54 10 CFR 73.55 10 CFR 73.56 10 CFR 73.57 10 CFR 73.58 10 CFR 73.70 10 CFR 73.71 10 CFR 73, App. A 10 CFR 73, App. B 10 CFR 73, App. C 10 CFR 73, App. G 10 CFR 73, App. H 10 CFR 74 10 CFR 100 10 CFR 100.21	None	RG 1 206	9.5.2, 9.5.3, 13.6, 13.6 2, 13.6 6 14.3 12	RGs 1.91, 4 7, 5.7, 5.12, 5 44, 5.54, 5.62, 5.65, 5.68, 5.69, 5.71, 5.74, 5 76, 5.77, 5 79 ISG-016 Draft DG-5019, -5035 (SRI) NUREG-0509, -1226, -1267 NUREG/CR-1198, -1345, -1381, -2585, -2643, -4250, -6190 RISs 2003-06, 2005-04 IMC-2201 IMC-2504	NEI 03-12, 08-09	N/A	N/A	Approval of mPower Security Technical/Topical Reports	9.5.2, 9.5 3, 13.4, 13.6.1, 13 6 2, 13 6.3, 14 3

Note 1: RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
Part 5: Safeguards and Security Plans	OLA	10 CFR 11.11 10 CFR 26 10 CFR 50.34 10 CFR 50.54 10 CFR 73.1 10 CFR 73.2 10 CFR 73.21 10 CFR 73.22 10 CFR 73.23 10 CFR 73 45(g)(4)(i) 10 CFR 73.54 10 CFR 73.55 10 CFR 73.56 10 CFR 73 57 10 CFR 73.58 10 CFR 73.70 10 CFR 73.71 10 CFR 73, App. A 10 CFR 73, App. B 10 CFR 73, App. C 10 CFR 73, App. G 10 CFR 73, App. H 10 CFR 74 10 CFR 100 10 CFR 100.21	None	RG 1.206	2.1.1, 2.1.2, 9.5.2, 9.5.3, 13.1.1, 13.1.2, 13.1.3, 13.4, 13.5.1.1, 13.6, 13.6.1, 13.6.6, 14.3.12	RGs 1.91, 4.7, 5.7, 5 12, 5.20, 5.43, 5 44, 5 54, 5.62, 5.65, 5.66, 5.68, 5.69, 5.71, 5.73, 5.74, 5.75, 5.76, 5.77, 5.79 ISG-016 Draft DG-5019, -5035 (SRI) NUREG-0509, 1226, -1267 NUREG/BR-0252 NUREG/CR-0485, -1198, -1345, -1381, -2585, -2643, -4250, -6190 RISs 2003-06, 2005-04 IMC-2201	NUMARC 89-01 NEI 03-09, 03-12, 06-11, 06-12, 08-09, 09-01	N/A	No	Approval of mPower Security Technical/Topical Reports. NRC acceptance of proposed security staffing for mPower.	2.1.1, 2.1.2, 9.5.2, 9.5.3, 13.1.1, 13.1.2, 13.3.3, 13.4, 13.5.1.1, 13.6.1, 13 6.2, 13.6.3

Note 1: RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

CLINCH RIVER
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Part 5 – Safeguards and Security Plans Outline

Part 5 Safeguards and Security Plans

CPA Part 5 CPA Part 5 provides a description of the elements of the individual security plans (e.g., Physical Security Plan, the Training and Qualification Plan, and the Safeguards Contingency Plan), as required by 10 CFR 73, for the Clinch River site. Part 5 consists of the following elements:

- description of the proposed site security provisions to be implemented during construction
- preliminary version of the security plans consistent with the following NRC-endorsed NEI templates:
  - NEI 03-12, "Physical Security Plan Generic Template",
  - NEI 09-01, "Security Measures During New Reactor Construction Template" (formerly NEI 03-12, Appendix F), and
  - NEI 03-09, "Security Officer Training Program Template"
- preliminary description of design features of the mPower plant that enhance security as described in B&W Technical Report R0003-08-002708-000(P) "Physical Security Design and Program Considerations"
- reference to the mPower standard plant security assessment as described in B&W Technical Report XXX, "Security Assessment, Safeguards Information," which addresses identification of vital equipment, development of target sets, security assessments, protective strategies, design features to enhance security, portions of NRC Orders to current operating plants that impact the B&W mPower plant design, and other security features that establish the security system design
- reference to PSAR Section 13.6, as appropriate

DCD 13.6 A separate Security Plan is not part of the DCA; however, DCD Section 13.6 addresses physical security for the mPower standard plant design as summarized below:

- description of design features of the mPower plant that enhance security as described in B&W Technical Report R0003-08-002708-000(P) "Physical Security Design and Program Considerations"
- reference to the mPower standard plant security assessment as described in B&W Technical Report XXX, "Security Assessment, Safeguards Information," which addresses identification of vital equipment, development of target sets, security assessments, protective strategies, design features to enhance security, portions of NRC Orders to current operating plants that impact the B&W mPower plant design, and other security features that establish the security system design
- identification of COL Information Items for an applicant to address, including:
  - development of a site-specific security assessment and security plan that adequately demonstrates how the performance requirements are met for the initial implementation of the security program
  - development of the Cyber Security Plan

Part 5 Safeguards and Security Plans (cont.)

OLA Part 5 OLA Part 5 provides a complete description of the elements of the individual security plans (e.g., Physical Security Plan, the Training and Qualification Plan, and the Safeguards Contingency Plan) as required by 10 CFR 73 for the Clinch River site. Part 5 consists of the following elements:

- description of the proposed site security provisions implemented during construction
- detailed security plans consistent with the following NRC-endorsed NEI templates:
  - NEI 03-12, "Physical Security Plan Generic Template",
  - NEI 09-01, "Security Measures During New Reactor Construction Template" (formerly NEI 03-12, Appendix F), and
  - NEI 03-09, "Security Officer Training Program Template"
- separate licensing document called the Cyber Security Plan as required by 10 CFR 73 based on NEI 08-09, "Cyber Security Plan Template"
- separate licensing document called the "Loss of Large Areas of the Plant Mitigative Strategies Description," which addresses large fires and explosions as required by 10 CFR 50.54(hh)(2) (Originally B.5.b. Order) based on NEI 06-12, "B.5.b Phase 2 & 3 Submittal Guideline Template," and Interim Staff Guidance ISG-016, "Compliance with 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d) Loss of Large Areas of the Plant due to Explosions or Fires from a Beyond-Design Basis Event"

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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
14.1 Specific Information to be Addressed for the Initial Plant Test Program	PSAR	10 CFR 30 53(c) 10 CFR 50, App B, Section XI 10 CFR 50, App J, Section III.A 4 10 CFR 50.34(b)(6)(iii)	None	RG 1.70	14.2	RG 1 9 RG 1.20 RG 1.30 RG 1.37 RG 1.41 RG 1.52 RG 1.68 RG 1.68.2 RG 1.68 3 RG 1.78 RG 1.79 RG 1.128 RG 1 140 NUREG-0660 - Action Plan Item I G 1 NUREG-0694 NUREG-0737 - TMI Item I.G 1	None	N/A	No	None	No
	DCD	Section 14 1 is not applicable to a DCD submitted under Part 52.	N/A	N/A	NA	N/A	N/A	N/A	N/A	N/A	N/A
	FSAR	This section is not applicable for an FSAR.	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
14 2 Initial Plant Test Program	PSAR	PSAR Section 14 2 is not applicable to the Clinch River PSAR	N/A	RG 1.70	N/A	N/A	N/A	N/A	N/A	N/A	N/A
	DCD	10 CFR 30 53(c) 10 CFR 50, App B, Section XI 10 CFR 50, App J, Section III.A 4 10 CFR 50.34(b)(6)(iii) 10 CFR 52 47(b)(1)	None	RG 1 206	14 2	RG 1.9 RG 1 20 RG 1.30 RG 1.37 RG 1 41 RG 1.52 RG 1 68 RG 1.68 2 RG 1.68.3 RG 1.78 RG 1.79 RG 1.128 RG 1.140 NUREG-0660 - Action Plan Item I G 1 NUREG-0694 NUREG-0737 - TMI Item I.G.1 IMC 2504	None	N/A	N/A	None	3 5, 3.7, 3.8, 3 9 4.3, 4.4, 4 6 5 1, 5 2, 5 4 6.2, 6.3, 6 4 7.1, 7.2, 7.3, 7.5, 7.6, 7.7 8.2, 8.3 9.1, 9 2, 9.3, 9.4, 9 5 10.2, 10.3, 10 4 11.2, 11.3, 11.4, 11.5
	FSAR	10 CFR 30.53(c) 10 CFR 50, App B, Section XI 10 CFR 50, App J, Section III.A 4 10 CFR 50 34(b)(6)(iii)	None	RG 1.206	14 2	RG 1.9 RG 1 20 RG 1.30 RG 1.37 RG 1 41 RG 1.52 RG 1.68 RG 1 68 2 RG 1.68.3 RG 1 78 RG 1 79 RG 1.128 RG 1 140 NUREG-0660 - Action Plan Item I.G.1 NUREG-0694 NUREG-0737 - TMI Item I.G.1	None	N/A	No	None	3.5, 3.7, 3.8, 3 9 4 3, 4 4, 4.6 5.1, 5.2, 5 4 6 2, 6 3, 6.4 7.1, 7.2, 7.3, 7.5, 7.6, 7.7 8 2, 8 3 9.1, 9.2, 9.3, 9 4, 9.5 10 2, 10.3, 10 4 11 2, 11.3, 11.4, 11.5 15 3

Note.
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal

Clinch River Regulatory Framework Document
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Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
14 3 Inspection, Test, Analysis, and Acceptance Criteria	PSAR	PSAR Section 14 3 is not applicable to a PSAR submitted under 10 CFR Part 50	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
	DCD	10 CFR 20 10 CFR 50, Appendix A 10 CFR 50 43 10 CFR 50 55a 10 CFR 50 63 10 CFR 52 47(a)(26) 10 CFR 52 47(b)(1) 10 CFR 52 63 10 CFR 73 55 10 CFR 73 70(e) 40 CFR 190	None	RG 1 206	14.2 14 3 14.3 2 14 3 3 14.3 4 14 3 5 14 3 6 14 3 7 14 3 8 14 3 9 14 3 10 14 3.11 14 3 12 17.4	BTP 7-14, BTP 7-19 RGs 1.9, 1 20, 1.30, 1.37, 1.41, 1.52, 1 68, 1 68 2, 1 68 3, 1.79, 1 97, 1.128, 1.136, 1.152, NUREG-0554 NUREG-0696 Supplement 1 NUREG-0737 IMC-2503 IP 65001.17 RIS 2008-05 SECY-05-0197 and SRM on SECY-05-0197 SECY-90-016 as modified by SRM to SECY-93-087 SECY-95-132 SECY-94-084 SECY-97-044 SECY-92-137 SECY-02-059 SECY-92-053 3/26/09 NRC Letter, "PS-ITAAC Related to New Plant Construction	ASME BPVC 2007 with 2008 Addenda IEEE 603 - 1991 NEI 12/19/09 Letter, "Security ITAAC Related to New Plant Construction" IEEE 308-2001	N/A	No	None	3 5, 3.7, 3 8, 3 9 4 3, 4 4, 4 6 5.1, 5 2, 5 4 6 2, 6 3, 6 4 7.1, 7 2, 7.3, 7.5, 7.6, 7.7 8.2, 8 3 9 1, 9 2, 9 3, 9 4, 9 5 10 2, 10.3, 10 4 11 2, 11.3, 11 4, 11.5 17 4
	FSAR	FSAR Section 14.3 is not applicable to an FSAR submitted under 10 CFR Part 50.	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Note:
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

14.0 Verification Programs

14.1 Specific Information to be Addressed for the Initial Plant Test Program

PSAR The Clinch River Initial Test Program (ITP) addresses the plan for preoperational and startup testing. The following elements of the ITP are discussed in PSAR Section 14.1:

- scope of test program, including major phases of the ITP and the overall test objectives and general prerequisites
- organizations and their general responsibilities for the development, approval, conduct, and review of tests
- methods and objectives of pre-operational and/or startup testing of plant design features that are special, unique, or first-of-a-kind
- plans for using applicable regulatory guides in initial test program (e.g., RG 1.68)
- plans for use of available information on reactor operating experiences
- overall schedule relative to expected fuel load date for developing and conducting major phases of the test program
- schedule for training of plant operating/technical staff or reference to appropriate sections of Chapter 13
- plans pertaining to trial use of plant operating and emergency procedures during the ITP
- general plans for assignment of additional personnel during each major phase of test program, including general responsibilities, summary of interrelationships and interfaces of responsible organizations
- general qualifications and approximate schedule (relative to fuel load) for augmenting staff personnel

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DCD DCD Section 14.1 includes a statement that the contents of this section are not applicable to the mPower standard plant design.

FSAR FSAR Section 14.1 includes a statement that the contents of this section are not applicable to the Clinch River FSAR.

14.2 Initial Plant Test Program

PSAR PSAR Section 14.2 includes a statement that the contents of this section are not applicable to the Clinch River PSAR.

DCD The mPower standard plant Initial Test Program (ITP) provides information on the plans for conducting preoperational and startup testing of structures, systems, and components (SSCs) and design features for both the nuclear portion of the facility and the balance of plant. The following elements of the ITP are discussed in DCD Section 14.2:

- Summary of Test Programs and Objectives
  - objectives for each major phase of test program
  - criteria for selection of plant features to be tested
  - conformance of test programs with RG 1.68, and the RGs listed in RG 1.68 that are referenced in SRP Section 14.2, including proposed exceptions or alternatives along with justifications

- Test Procedures
  - general guidance to control ITP activities
  - general guidance for the review of relevant operating and testing experience at other facilities
  - general guidance related to the extent that test program will use plant operating/emergency/surveillance procedures in testing
  - test abstracts of SSC and unique design features that will be tested, including objectives, tests, and acceptance criteria

- Summary description of initial fuel loading, initial criticality, low power, and power ascension testing

- Individual test descriptions, including abstracts of planned tests to demonstrate/verify the performance capabilities of functions:
  - used for safe shutdown and cooldown of reactor under normal conditions and for maintenance in extended shutdown period
  - used for safe shutdown and cooldown under transient conditions and postulated accident conditions and for maintenance following such conditions
  - used for establishing conformance with Safety Limits (SL) or Limiting Conditions of Operations (LCO) in Technical Specifications (TS)
  - classified as Engineered Safety Features (ESF) or used to support or ensure operations within ESF design limits
  - assumed or credited in accident analysis
  - used to process, store, control, measure, or limit the release of radioactive materials.
  - used in special low power testing program if relied on to provide technical information for resolution of TMI Action Item I.G.1 identified as risk significant in design-specific PRA

14.2 Initial Plant Test Program (cont.)

DCD (cont.)

- Description of first-of-a-kind design features related to the mPower standard plant, including functional test requirements and acceptance criteria

- Reference to Tier 1 for a description of the following elements of the ITP:
  - general description of the preoperational and power ascension test programs.
  - major program documents that define how ITP will be conducted and controlled
  - key elements of the ITP to ensure a COL applicant does not unilaterally initiate changes in the conduct of the ITP
  - provisions to ensure test procedures and test specification are made available to the NRC

- Identification of the following COL Information Items to be addressed by an applicant:
  - summary of test program and objectives
  - organization and staffing
  - test procedures
  - conduct of test program
  - review, evaluation, and approval of test results
  - test records
  - test program's conformance with regulatory guides
  - utilization of reactor operating and test experience in test program development
  - trial use of plant operating and emergency procedures
  - initial fuel loading and initial criticality
  - test program schedule and sequence
  - individual test descriptions

FSAR

The Clinch River SMR Plant ITP provides information on the plans for conducting preoperational and startup testing of SSCs and design features for both the nuclear portion of the facility and the balance of plant. The following elements of the ITP are discussed in FSAR Section 14.2:

- Summary of Test Programs and Objectives
  - objectives for each major phase of test program
  - criteria for selection of plant features to be tested
  - conformance of test programs with RG 1.68, and the RGs listed in RG 1.68 that are referenced in SRP Section 14.2, including proposed exceptions or alternatives along with justifications

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14.2 Initial Plant Test Program (cont.)

FSAR (cont.)

- Organization and Staffing
  - description of organization and augmenting organizations that will manage, supervise, or execute any phase of the test program, including the authorities, responsibilities, and degree of participation of each organization
  - discussion on the operating/technical staff participation in each phase of the test program, including personnel qualifications for management, development, or conduct of each test phase
  - description of training program for personnel involved with the preoperational and startup testing

- Test Procedures
  - process for development, review, and approval of test procedures, including organizations/personnel responsibilities and how the design organizations will participate in establishing test performance or acceptance criteria
  - process for ensuring that test procedures include appropriate prerequisites, objectives, safety precautions, initial test conditions, methods to direct and control test performance, and appropriate acceptance criteria
  - description of test procedure format consistent with RG 1.68, or justification for any exceptions, suitable for NRC review

- Conduct of Test Program
  - administrative controls that will govern conduct of test phases
  - controls to ensure prerequisites are satisfied
  - methods for initiating plant modifications/maintenance determined to be required by test program
  - methods that will be used to ensure retesting following modifications/maintenance, including involvement of design organizations in the review/approval of plant modifications
  - administrative controls pertaining to adherence to approved test procedures and methods for effecting changes

- Review, Evaluation, and Approval of Test Results
  - measures established for review, evaluation, and approval of test results
  - controls to ensure notification of affected/responsible organizations/personnel when test acceptance criteria not met as well as controls to resolve such matters
  - plans to approve test data for each major test phase before proceeding to next test phase, and approval of test data at each power test level plateau, before increasing power

14.2 Initial Plant Test Program (cont.)

FSAR (cont.)

- Test Records
  - requirements for the disposition of test procedures and data following completion of the test program
  - provisions to retain test reports that include test procedures and results as part of the plant historical records

- Conformance of ITP with Regulatory Guides
  - table(s) providing a list of the regulatory guides used for the ITP, including identification of any exceptions and justification for alternatives

- Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program
  - description of program for reviewing available information on reactor operating experience in the development of tests, including status of program, the organizations participating and their roles, and conclusions/findings from this review
  - summary description of preoperational and/or startup testing planned for unique or first-of-a-kind principal mPower standard plant design features, including test method, objective, and frequency

- Trial Use of Plant Operating and Emergency Procedures
  - schedule for development of plant procedures, including description of how the procedures will be tested during the ITP
  - identification of specific operator training to be conducted during the special low-power testing program

- Initial Fuel Loading and Initial Criticality
  - description of plans for initial fuel loading and initial criticality, including the safety and precautionary measures to be established for safe plant operation

- Test Program Schedule and Sequence
  - schedule, relative to fuel loading, for conducting each major phase of the test program
  - identification of tests required for individual SSCs, including their sequence, required to be completed prior to initial fuel load
  - schedule for development of test procedures, including timeframes the procedures will be available for NRC review

- Individual Test Descriptions
  - individual abstracts of tests to be conducted for specified SSCs as defined in DCD Section 14.2 with an identification of each test by title, prerequisites, major plant operating conditions, and summary description of the test method, test objectives, and acceptance criteria for each test

14.3 Inspection, Test, Analysis, and Acceptance Criteria

PSAR PSAR Section 14.3 includes a statement that the Clinch River license application is submitted in accordance with 10 CFR Part 50; therefore, an ITAAC Program is not applicable and is not included in the PSAR.

DCD The mPower standard plant inspections, tests, analyses, and acceptance criteria (ITAAC) are described in DCD Section 14.3. The following elements of ITAAC are discussed:

- description of the bases, processes, and selection criteria used to develop the Tier 1 information and ITAAC in the following areas:
  - Structural and Systems Engineering
  - Piping Systems and Components
  - Reactor Systems
  - Instrumentation and Controls
  - Electrical Systems
  - Plant Systems
  - Radiation Protection
  - Human Factors Engineering
  - Emergency Planning
  - Containment Systems
  - Physical Security Hardware
  - Initial Plant Test Program
  - Design Reliability Assurance Program

- identification of COL Information Items for an applicant to provide the following information:
  - site-specific ITAAC, as needed
  - plant-specific emergency planning ITAAC not addressed in the DCD
  - plant-specific physical security hardware ITAAC not addressed in the DCD

FSAR FSAR Section 14.3 includes a statement that the Clinch River license application is submitted in accordance with 10 CFR Part 50, therefore an ITAAC Program is not applicable and is not included in the FSAR.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
17.1 Quality Assurance During the Design and Construction Phases	PSAR	See Section 17.5	See Section 17.5	See Section 17.5	See Section 17.5	See Section 17.5	See Section 17 5	See Section 17.5	No	See Section 17.5	See Section 17.5
	DCD	See Section 17.5	See Section 17.5	See Section 17.5	See Section 17 5	See Section 17 5	See Section 17.5	See Section 17.5	N/A	See Section 17.5	See Section 17.5
	FSAR	N/A	N/A	FSAR Section 17.1 contains a statement that the contents of this section are addressed in PSAR Section 17 5 and are not applicable to the FSAR	N/A	N/A	N/A	N/A	No	N/A	N/A
17.2 Quality Assurance During the Operations Phase	PSAR	N/A	N/A	PSAR Section 17.2 contains a statement that the contents of this section are not applicable for a PSAR but will be addressed in the FSAR.	N/A	N/A	N/A	N/A	No	N/A	N/A
	DCD	See Section 17.5	See Section 17.5	See Section 17 5	See Section 17 5	See Section 17 5	See Section 17.5	See Section 17.5	N/A	See Section 17.5	See Section 17 5
	FSAR	See Section 17 5	See Section 17 5	See Section 17.5	See Section 17 5	See Section 17.5	See Section 17.5	See Section 17.5	No	See Section 17.5	See Section 17 5
17.3 Quality Assurance Program	PSAR	See Section 17.5	See Section 17.5	See Section 17 5	See Section 17 5	See Section 17 5	See Section 17.5	See Section 17.5	See Section 17.5	See Section 17.5	See Section 17.5
	DCD										
	FSAR										
17.4 Reliability Assurance Program	PSAR	10 CFR 50 65 10 CFR 50, App. B	None	RG 1.206	17 4	RG 1.160 ISG-018 SECYs 94-084, 95-132 NUREG/CR-5424	NUMARC 93-01 EPRI-1021413 EPRI-1021415 EPRI-1021416	Yes - Development of a Reliability Assurance Program not specified in RG 1.70	No	1) In response to the events at Fukishima, additional SSCs may be required as part of D-RAP	3.2, 17.5, 17 6, 19
	DCD	10 CFR 50 65 10 CFR 50, App. B 10 CFR 52.47	None	RG 1.206	17.4	RG 1 160 ISG-018 SECYs 94-084, 95-132 NUREG/CR-5424 IMC 2503 IP 65001	NUMARC 93-01 EPRI-1021413 EPRI-1021415 EPRI-1021416	N/A	N/A	1) In response to the events at Fukishima, additional SSCs may be required as part of D-RAP	3 2, 14.3, 17.5, 17 6, 19
	FSAR	10 CFR 50 65 10 CFR 50, App B	None	RG 1.206	17 4	RG 1.160 ISG-018 SECYs 94-084, 95-132 NUREG/CR-5424	NUMARC 93-01 EPRI-1021413 EPRI-1021415 EPRI-1021416	N/A	No	1) In response to the events at Fukishima, additional SSCs may be required as part of D-RAP	3 2, 17 5, 17 6, 19

Note:
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Key Issues	Related Sections
17.5 Quality Assurance Program Description	PSAR	10 CFR 21 10 CFR 50.34 10 CFR 50 48 10 CFR 50.54 10 CFR 50 55a 10 CFR 50.62 10 CFR 50.63 10 CFR 50.65 10 CFR 50.69 10 CFR 50, App. A, GDC 1 10 CFR 50, App. B	None	RG 1 206	17.5	RGs 1 26, 1 29 RG 1.28, Rev. 3 RIS 2000-18 GLs 85-06, 88-18, 89-02, 91-05 SECYs 94-084, 95-132	ASME NQA-1-1994 ANSI/ISO/IEC 17025, 2005 ANSI N45 2 EPRI NP-5652, 1988 NIRMA TG 11-1998, 15-1998, 16-1998, 21-1998	Yes - QA Program description will be based on current regulatory requirements	No	None	None
	DCD	10 CFR 21 10 CFR 50 34 10 CFR 50 48 10 CFR 50 54 10 CFR 50 55a 10 CFR 50 62 10 CFR 50 63 10 CFR 50 65 10 CFR 50 69 10 CFR 50, App A, GDC 1 10 CFR 50, App B 10 CFR 52.47	None	RG 1.206	17 5	RGs 1.26, 1.29 RG 1.28, Rev. 3 RIS 2000-18 GLs 85-06, 88-18, 89-02, 91-05 SECYs 94-084, 95-132, 05-197 IMC 2502 and 2504	ASME NQA-1-1994 ANSI/ISO/IEC 17025, 2005 EPRI NP-5652, 1988 NIRMA TG 11-1998, 15-1998, 16-1998, 21-1998	N/A	N/A	None	14 3
	FSAR	10 CFR 21 10 CFR 50.34 10 CFR 50 48 10 CFR 50.54 10 CFR 50.55a 10 CFR 50 62 10 CFR 50.63 10 CFR 50 65 10 CFR 50 69 10 CFR 50, App. A, GDC 1 10 CFR 50, App. B	None	RG 1 206	17.5	RGs 1 8, 1 26, 1 29, 1.33, 1.37, 1.155, 1.189 RG 1 28, Rev. 3 RIS 2000-18 GLs 85-06, 88-18, 89-02, 91-05 SECYs 94-084, 95-132	ASME NQA-1-1994 ANSI/ISO/IEC 17025, 2005 ANSI N45 2 EPRI NP-5652, 1988 NIRMA TG 11-1998, 15-1998, 16-1998, 21-1998	N/A	No	None	None
17 6 Maintenance Rule	PSAR	10 CFR 50 65 10 CFR 50 69	None	RG 1.206	17.6	RGs 1.160, 1.182, 1.187 GL 2006-02 IP 62706	NEI 96-07 NUMARC 91-06 NUMARC 93-01	Yes - Section 17.6 addresses the Maintenance Rule requirements issued subsequent to RG 1 70	No	1) In response to the events at Fukishima, additional SSCs may be required as part of the Maintenance Rule Program	3 2, 4.5, 5 4, 12, 13 4, 17 1, 17 2, 17 3, 17.4
	DCD	10 CFR 50 65 10 CFR 50 69	None	RG 1.206	17.6	RGs 1 160, 1 182, 1 187 GL 2006-02 SECY 05-197 IMC-2504 IP 62706	NEI 96-07 NUMARC 91-06 NUMARC 93-01	N/A	N/A	1) In response to the events at Fukishima, additional SSCs may be required as part of the Maintenance Rule Program	3.2, 4.5, 5.4, 12, 13 4, 17.1, 17.2, 17.3, 17.4
	FSAR	10 CFR 50 65 10 CFR 50.69	None	RG 1.206	17.6	RGs 1.160, 1.182, 1 187 GL 2006-02 IP 62706	NEI 96-07 NUMARC 91-06 NUMARC 93-01 NEI 07-02A	N/A	No	1) In response to the events at Fukishima, additional SSCs may be required as part of the Maintenance Rule Program	3.2, 4.5, 5 4, 12, 13.4, 17.1, 17 2, 17.3, 17.4

Note:
RG revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

Chapter 17 Outline

17 Quality Assurance and Reliability Assurance

17.1 Quality Assurance During the Design and Construction Phases

PSAR This section provides a reference to PSAR Section 17.5 for a description of the Quality Assurance Program during the design and construction phases of the Clinch River SMR Plant.

DCD This section provides a reference to DCD Section 17.5 for a description of the Quality Assurance Program for the design phase of the mPower standard plant and includes a COL Item for an applicant to describe the Quality Assurance Program for site-specific design and construction activities.

FSAR FSAR Section 17.1 contains a statement that the contents of this section are addressed in PSAR Section 17.5 and are not applicable to the FSAR.

Chapter 17 Outline

17.2 Quality Assurance During the Operations Phase

PSAR Section 17.2 contains a statement that this section is not applicable to the PSAR but will be addressed in the FSAR.

DCD This section includes a COL Item for the applicant to provide a description of Quality Assurance Program for operational activities.

FSAR This section provides a reference to FSAR Section 17.5 for a description of the Quality Assurance Program during the operations phase of the Clinch River SMR Plant.

Chapter 17 Outline

17.3 Quality Assurance Program

PSAR This section provides a reference to PSAR Section 17.5 for a description of the Quality Assurance Program for the Clinch River SMR Plant.

DCD This section provides a reference to DCD Section 17.5 for a description of the Quality Assurance Program for the mPower standard plant design.

FSAR This section provides a reference to FSAR Section 17.5 for a description of the Quality Assurance Program for the Clinch River SMR Plant.

Chapter 17 Outline

17.4 Reliability Assurance Program

PSAR PSAR Section 17.4 provides a description of the Design Reliability Assurance Program (D-RAP) and the preliminary program elements as described below:

- preliminary description of D-RAP, including scope, purpose and objectives
- identification of the responsible organization for D-RAP, organizational interfaces and responsibilities
- preliminary approach for application of D-RAP to risk-significant structures, systems, and components (SSCs)
- preliminary listing of D-RAP SSCs

DCD DCD Section 17.4 addresses the following information regarding the Reliability Assurance Program for the mPower standard plant:

- description of the D-RAP, including scope, purpose and objectives
- identification of the responsible organization for D-RAP, organizational interfaces and responsibilities
- description of the design and design change control process for configuration management, including procedures and instructions, corrective action process, records, and audit plans
- description of procedures and instructions for implementation of D-RAP and how they are developed and controlled
- description of the methodology for identification of SSCs within the scope of RAP, including the use of risk evaluations, operating experience, and an Expert Panel to assist with identification of risk-significant SSCs
- description of the corrective action process as it applies to the SSCs within the D-RAP scope
- description of how records related to the D-RAP are controlled and maintained
- description of the necessary measures and governing procedures to implement audits to verify conformance of activities covered by D-RAP
- summary of the roles, responsibilities, and qualification requirements for Expert Panel members
- a comprehensive list of RAP SSCs within the DCD scope and description of the configuration control process for maintaining the list of SSCs
- identification of COL Information Items for an applicant to develop and implement a site-specific D-RAP and integrate the RAP activities into plant-specific procurement and operational programs

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Chapter 17 Outline

17.4 Reliability Assurance Program (cont.)

FSAR Same contents as mPower standard plant DCD Section 17.4 with the following supplemental site-specific information:

- updated listing of site-specific SSCs that are risk-significant SSCs to be incorporated in the D-RAP
- description of how the RAP is integrated into operational programs, including the Maintenance Rule, surveillance testing, inservice inspection, inservice testing, and QA programs; and evolution of D-RAP into an Operational RAP
- description of the performance goals for the risk-significant SSCs, and their performance and condition monitoring
- description of processes that will continue to verify and evaluate the dominant failure modes and associated effects for RAP SSCs during the operations stage
- description of the graded approach to quality assurance controls for the non-safety-related, risk significant SSCs within the RAP

Chapter 17 Outline

17.5 Quality Assurance Program Description

PSAR PSAR Section 17.5 provides information regarding the establishment and implementation of a Quality Assurance Program applicable to activities during design, fabrication, construction, and testing of the Clinch River SMR Plant based on the following information:

- reference to the approved TVA Nuclear Quality Assurance Plan (TVA-NQA-PLN89-A)
- reference to the approved Quality Assurance Program of the major organizations participating in the design and construction activities
- description of the organizational structure responsible for development of the QA program

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DCD DCD Section 17.5 addresses the following information regarding the Quality Assurance Program Guidance for the mPower standard plant:

- reference to the approved BWNE Quality Assurance Topical Report that describes the Quality Assurance Program
- description of organizational structure responsible for development of the mPower standard plant design
- identification of a COL Information Item for an applicant to develop a Quality Assurance Program for site-specific design activities and for plant construction and operation

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FSAR Same contents as the PSAR Section 17.5 with the following plant-specific supplemental information:

- reference to the approved TVA Nuclear Quality Assurance Plan (TVA-NQA-PLN89-A) applicable for use during the design, fabrication, construction, testing, and operation phases
- delineation of quality assurance functions between the applicant and other organizations
- description of how the applicant will maintain responsibility for, and control over, those portions of the Quality Assurance Program that have been delegated
- identification of the responsible organization(s) and the process for verifying that delegated functions are effectively implemented
- identification of major work interfaces for activities affecting quality
- description of how clear and effective lines of communication between the applicant and its principal contractors are maintained to assure coordination and control of the QA program

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Chapter 17 Outline

17.6 Maintenance Rule

PSAR PSAR Section 17.6 provides the following information regarding the implementation of the Maintenance Rule requirements under 10 CFR 50.65:

- a regulatory commitment to establish a Maintenance Rule Program in the FSAR

DCD DCD Section 17.6 provides a COL Item for an applicant to provide a description of the Maintenance Rule Program and proposed implementation milestones necessary to meet the requirements of 10 CFR 50.65.

FSAR FSAR Section 17.6 incorporates by reference NEI 07-02A, including the following supplemental information:

- description of the Maintenance Rule Program implementation in accordance with NUMARC 93-01 and RG 1.160
- process for determining which plant structures, systems, and components (SSCs) are included in the scope of the Maintenance Rule Program in accordance with 10 CFR 50.65
- the process for determining the safety/risk significance classification of SSCs within scope of the Maintenance Rule
- how the D-RAP SSC list will be integrated into the Maintenance Rule Program scope
- how SSCs within Maintenance Rule scope are monitored and periodically evaluated in accordance with 10 CFR 50.65
- the program for maintenance risk assessment and management in accordance with 10 CFR 50.65(a)(4)
- the program for training and qualification of personnel with Maintenance Rule-related responsibilities
- description of process for monitoring and tracking the performance and/or condition of SSCs as they become operational prior to fuel load

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18 0 Human Factors Engineering	PSAR	N/A	No	RG 1.206	N/A	N/A	N/A	Yes - RG 1.70 does not address HFE topics	No	6.3, 7.0, 13.1, 13.2, 13.5, 13.6, 14.3, 15.0	Need to address DAC/ITAAC for HFE Program
	DCD	N/A	No	RG 1.206	N/A	N/A	N/A	N/A	N/A	6.3, 7.0, 13.1, 13 2, 13.5, 13.6, 14.3, 15.0	Need to address DAC/ITAAC for HFE Program
	FSAR	N/A	No	RG 1.206	N/A	N/A	N/A	N/A	No	6.3, 7.0, 13.1, 13.2, 13.5, 13.6, 14.3, 15.0	Need to address DAC/ITAAC for HFE Program
18.1 Human Factors Engineering (HFE) Management	PSAR	10CFR50.54(i) through (m) - related to staffing	No	RG 1.206	18.0, Sections II.A.A.1 and II.A.A.5	RG 1.174, NUREG-0711 NUREG-0654, NUREG/CR-6838 NUREG-1791 SECY-05-197, SECY-11-0098 RG 1.8, RG 1.206 Section C.IV.4	ANSI/ANS 3.1, 1993 (R1999) - related to staffing NEI Position Paper on Control Room Staffing for Small Reactors, dated September 23, 2011 ANS Report, "Interim Report of the American Nuclear Society President's Special Committee on Small and Medium Sized Reactor (SMR) Generic Licensing Issues," July 2010	Yes - RG 1.70 does not address HFE topics	No	13.1	None
	DCD	10CFR50.34(f)(3)(vii) 10CFR50.54(i) through (m) - related to staffing	10CFR50.54(m)	RG 1.206	18.0, Sections II.A.A.1 and II.A.A.5	RG 1.174, NUREG-0711 NUREG-0654, NUREG/CR-6838 NUREG-1791 SECY-05-197, SECY-11-0098 RG 1.8, RG 1.149, RG 1.206 Section C.IV.4	ANSI/ANS 3.1, 1993 (R1999) - related to staffing NEI Position Paper on Control Room Staffing for Small Reactors, dated September 23, 2011 ANS Report, "Interim Report of the American Nuclear Society President's Special Committee on Small and Medium Sized Reactor (SMR) Generic Licensing Issues," July 2010	NA	N/A	13.1	Generation mPower provides initial minimum staffing requirements in the DCD submittal, demonstrates their acceptability during the design process, modifies them as required, and provides justification to support application for an exemption to 10CFR50.54(m), if validated mPower minimum manning is less than specified in 10CFR 50.54(m).
	FSAR	10CFR50.34(f)(3)(vii) 10CFR50.54(i) through (m) - related to staffing	10CFR50.54(m)	RG 1.206	18 0, Sections II.A.A.1 and II.A.A.5	RG 1.174, NUREG-0711 NUREG-0654, NUREG/CR-6838 NUREG-1791 SECY-05-197, SECY-11-0098 RG 1.8, RG 1.149, RG 1.206 Section C.IV 4	ANSI/ANS 3.1, 1993 (R1999) - related to staffing NEI Position Paper on Control Room Staffing for Small Reactors, dated September 23, 2011 ANS Report, "Interim Report of the American Nuclear Society President's Special Committee on Small and Medium Sized Reactor (SMR) Generic Licensing Issues," July 2010	NA	No	13.1	Generation mPower to develop justification to support an applicant's submittal of an exemption request to reduce Control Room Staffing, if needed

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
18.2 Operating Experience Review	PSAR	None	No	RG 1.206	18.0, Section II.A.A.2	NUREG-0711	None	Yes - RG 1 70 does not address HFE topics	No	None	None
	DCD	10CFR50.34(f)(3)(i) 10CFR52 47(a)(21)	No	RG 1.206	18.0, Section II A A 2	NUREG-0711, NUREG-0933	None	NA	N/A	None	NUREG-0800 incorrectly identifies 10CFR52.49(a)(21) - the correct reference is 10CFR52 47(a)(21) (see Regulatory Requirements column)
	FSAR	10CFR50.34(f)(3)(i)	No	RG 1.206	18.0, Section II.A.A.2	NUREG-0711	None	NA	No	None	None
18.3 Functional Requirements Analysis and Function Allocation	PSAR	None	No	RG 1.206	18 0, Section II.A A.3	NUREG-0711	None	Yes - RG 1.70 does not address HFE topics	No	None	None
	DCD	None	No	RG 1.206	18.0, Section II.A.A.3	NUREG-0711	None	NA	N/A	None	None
	FSAR	None	No	RG 1.206	18.0, Section II.A.A.3	NUREG-0711	None	NA	No	None	None

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
18.4 Task Analysis	PSAR	10CFR50.54(i) through (m) - related to staffing	No	RG 1.206	18.0, Sections II.A.A.4 and II.A.A.5	NUREG-0711 NUREG-0654, NUREG/CR-6838 NUREG-1791 SECY-05-197, SECY-11-0098 RG 1.8, RG 1.206 Section C.IV.4 DI&C-ISG-05	ANSI/ANS 3.1, 1993 (R1999) - related to staffing NEI Position Paper on Control Room Staffing for Small Reactors, dated September 23, 2011 ANS Report, "Interim Report of the American Nuclear Society President's Special Committee on Small and Medium Sized Reactor (SMR) Generic Licensing Issues," July 2010	Yes - RG 1.70 does not address HFE topics	No	None	None
	DCD	10CFR50.54(i) through (m) - related to staffing	10CFR50.54(m)	RG 1.206	18.0, Sections II.A.A.4 and II.A.A.5	NUREG-0711 NUREG-0654, NUREG/CR-6838 NUREG-1791 SECY-05-197, SECY-11-0098 RG 1.8, RG 1.149, RG 1.206 Section C.IV.4 DI&C-ISG-05	ANSI/ANS 3.1, 1993 (R1999) - related to staffing NEI Position Paper on Control Room Staffing for Small Reactors, dated September 23, 2011 ANS Report, "Interim Report of the American Nuclear Society President's Special Committee on Small and Medium Sized Reactor (SMR) Generic Licensing Issues," July 2010	NA	N/A	None	Generation mPower provides initial minimum staffing requirements in the DCD submittal, demonstrates their acceptability during the design process, modifies them as required, and provides justification to support application for an exemption to 10CFR50.54(m), if validated mPower minimum manning is less than specified in 10CFR 50.54(m).
	FSAR	10CFR50.54(i) through (m) - related to staffing	10CFR50.54(m)	RG 1.206	18.0, Sections II.A.A.4 and II.A.A.5	NUREG-0711 NUREG-0654, NUREG/CR-6838 NUREG-1791 SECY-05-197, SECY-11-0098 RG 1.8, RG 1.149, RG 1.206 Section C.IV.4 DI&C-ISG-05	ANSI/ANS 3.1, 1993 (R1999) - related to staffing NEI Position Paper on Control Room Staffing for Small Reactors, dated September 23, 2011 ANS Report, "Interim Report of the American Nuclear Society President's Special Committee on Small and Medium Sized Reactor (SMR) Generic Licensing Issues," July 2010	NA	No	None	Generation mPower to develop justification to support an applicant's submittal of an exemption request to reduce Control Room Staffing, if needed

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
18.5 Staffing and Qualification	PSAR	Included in Sections 18.1 and 18.4									
	DCD	Included in Sections 18.1 and 18.4									
	FSAR	Included in Sections 18.1 and 18 4									
18.6 Human Reliability Analysis	PSAR	None	No	RG 1.206	18.0, Section II.A.A.6	NUREG-0711, NUREG-1792, DI&C-ISG-02 DI&C-ISG-05 RG 1.206 - Sections C.II.1 and C.I.6.3.2.8	None	Yes - RG 1.70 does not address HFE topics	No	19 6.3.III.19	None
	DCD	10CFR50.34(f)(i), 10CFR52.47(b)(1), 10CFR52 79	No	RG 1.206	18.0, Section II.A.A.6	NUREG-0711, NUREG-0800 - Chapter 19 and Section 6.3.III.19, NUREG-1792, DI&C-ISG-02 DI&C-ISG-05 RG 1.206 - Sections C.II.1 and C.I.6.3.2.8	None	NA	N/A	19 6.3.III.19	None
	FSAR	10CFR50.34(f)(i),	No	RG 1.206	18.0, Section II.A.A 6	NUREG-0711, NUREG-0800 - Chapter 19 and Section 6.3.III.19, NUREG-1792, DI&C-ISG-02 DI&C-ISG-05 RG 1.206 - Sections C.II.1 and C.I.6.3.2.8	None	NA	No	19 6.3.III.19	None

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
18.7 Human-System Interface Design	PSAR	None	No	RG 1.206	18.0, Section II.A.A.7	RGs 1.22, 1.47, 1.62, 1.97, 1.105 NUREG-0696, NUREG-0711, NUREG-0700, Supplement 1 of NUREG-0737, NUREG-0800 Appendix 18-A (Replaced Section 3 DI&C ISG-05); NUREG-0835, NUREG-1342, NUREG-1852, NUREG/CR-6372, DI&C ISG-05, DI&C ISG-02 BTP 07-19	ANSI/ANS 58.8, 1994 (R2008) ANSI/ANS 3.5, 2009	Yes - RG 1.70 does not address HFE topics	No	7	None
	DCD	10CFR50.34(f)(2)Sub sections (i), (iii), (iv), (v), (xi), (xii), (xiii), (xv), (xvii), (xviii), (xix), (xxi), (xxiv), (xxv), & (xxvii)	No	RG 1.206	18.0, Section II.A.A.7	RGs 1.22, 1.47, 1.62, 1.97, 1.105 NUREG-0696, NUREG-0711, NUREG-0700, Supplement 1 of NUREG-0737, NUREG-0800 Appendix 18-A (Replaced Section 3 DI&C ISG-05); NUREG-0835, NUREG-1342, NUREG-1852, NUREG/CR-6372, DI&C ISG-05, DI&C ISG-02 BTP 07-19	ANSI/ANS 58.8, 1994 (R2008) ANSI/ANS 3.5, 2009	NA	N/A	7	None
	FSAR	10CFR50.34(f)(2)Sub sections (i), (iii), (iv), (v), (xi), (xii), (xiii), (xv), (xvii), (xviii), (xix), (xxi), (xxiv), (xxv), & (xxvii)	No	RG 1.206	18.0, Section II.A.A.7	RGs 1.22, 1.47, 1.62, 1.97, 1.105 NUREG-0696, NUREG-0711, NUREG-0700, Supplement 1 of NUREG-0737, NUREG-0800 Appendix 18-A (Replaced Section 3 DI&C ISG-05); NUREG-0835, NUREG-1342, NUREG-1852, NUREG/CR-6372, DI&C ISG-05, DI&C ISG-02 BTP 07-19	ANSI/ANS 58.8, 1994 (R2008) ANSI/ANS 3.5, 2009	NA		7	None

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
18.8 Procedure Development	PSAR	See Chapter 13									
	DCD	See Chapter 13									
	FSAR	See Chapter 13									
18.9 Training Program Development	PSAR	See Chapter 13									
	DCD	See Chapter 13									
	FSAR	See Chapter 13									
18.10 Verification and Validation	PSAR	None	No	RG 1.206	18.0, Section II.A.A.10	NUREG-0711, NUREG-6393	EPRI NP-3659 ANSI/ANS-58.6-1983	Yes - RG 1.70 does not address HFE topics	No	None	None
	DCD	None	No	RG 1.206	18.0, Section II.A.A.10	NUREG-0711, NUREG-6393	EPRI NP-3659 ANSI/ANS-58.6-1983	NA	N/A	None	None
	FSAR	None	No	RG 1.206	18.0, Section II.A.A.10	NUREG-0711, NUREG-6393	EPRI NP-3659 ANSI/ANS-58.6-1983	NA	No	None	None
18.11 Design Implementation	PSAR	None	No	RG 1.206	18.0, Section II.A.A.11	NUREG-0711, NUREG-0696 Supplement 1 to NUREG-0737	EPRI NP-3659 ANSI/ANS-58.6-1983	Yes - RG 1.70 does not address HFE topics	No	None	None
	DCD	None	No	RG 1.206	18.0, Section II.A.A.11	NUREG-0711, NUREG-0696 Supplement 1 to NUREG-0737	EPRI NP-3659 ANSI/ANS-58.6-1983	NA	N/A	None	None
	FSAR	None	No	RG 1.206	18.0, Section II.A.A.11	NUREG-0711, NUREG-0696 Supplement 1 to NUREG-0737	EPRI NP-3659 ANSI/ANS-58.6-1983	NA	No	None	None
18.12 Human Performance Monitoring	PSAR	None	No	RG 1.206	18.0, Section II.A.A.12	NUREG-0711	None	Yes - RG 1.70 does not address HFE topics	No	None	None
	DCD	None	No	RG 1.206	18.0, Section II.A.A.12	NUREG-0711	None	NA	N/A	None	None
	FSAR	None	No	RG 1.206	18.0, Section II.A.A.12	NUREG-0711	None	NA	No	None	None

CLINCH RIVER
REGULATORY FRAMEWORK DOCUMENTS
Chapter 18 Outline

18.0 Human Factors Engineering

PSAR PSAR Section 18.0 provides an introduction to Subsections 18.1 through 18.12 that address the Human Factors Engineering (HFE) Program elements. The HFE principles are incorporated into four major areas, namely, planning and analysis, design, verification and validation, and implementation and operation.

The elements that comprise the four areas are as follows:

- Planning and Analysis:
  - HFE Program Management
  - Operating Experience Review
  - Functional Requirements Analysis and Function Allocation
  - Task Analysis
  - Staffing and Qualification
  - Human Reliability Analysis
- Design:
  - Human-System Interface Design
  - Procedure Development
  - Training Program Development
- Verification and Validation:
  - Human Factors Verification and Validation
- Implementation and Operation:
  - Design Implementation
  - Human Performance Monitoring

Each of these elements is addressed individually in Sections 18.1 through 18.12.

DCD DCD Section 18.0 provides an introduction to Subsections 18.1 through 18.12 that address the Human Factors Engineering (HFE) Program elements. The HFE principles are incorporated into four major areas, namely, planning and analysis, design, verification and validation, and implementation and operation.

The elements that comprise the four areas are as follows:

- Planning and Analysis:
  - HFE Program Management
  - Operating Experience Review
  - Functional Requirements Analysis and Function Allocation

18.0 Human Factors Engineering (cont.)

DCD (cont.)

- Task Analysis
- Staffing and Qualification
- Human Reliability Analysis

– Design:
- Human-System Interface Design
- Procedure Development
- Training Program Development

– Verification and Validation:
- Human Factors Verification and Validation

– Implementation and Operation:
- Design Implementation
- Human Performance Monitoring

Each of these elements is addressed individually in Sections 18.1 through 18.12.

FSAR Same content as the mPower standard plant DCD Section 18.1 updated to reflect plant-specific information as applicable. Where information is not available for an element, an implementation plan and a schedule for addressing the element is included.

18.1 HFE Program Management

PSAR To the extent possible, the Clinch River PSAR HFE Program Management section incorporates by reference appropriate mPower technical reports related to HFE. For Section 18.1 the primary technical report is the Human Factors Engineering (HFE) Program Management Plan. A combination of references to technical reports and PSAR discussions addresses the following topics and goals of HFE Program:

- Goals of HFE Program
- Assumptions and constraints on designScope of HFE applicability (including main control room, remote shutdown station, technical support center, emergency operations center, and risk-important local control stations)
- Identification of plant personnel affected by HFE activities (all plant personnel who are expected to perform tasks related to plant safety are identified)
- HFE design team and organization including:
  - Areas of responsibility with respect to HFE program
  - Placement within the overall design organization
  - Authority to provide reasonable assurance that all of its area of responsibility area accomplished and to identify problems in the implementation of the overall plant design. Design team composition with respect to areas of expertise
  - Staffing in terms of job descriptions and assignments of team personnel
- Description of the HFE process and procedures listed in NUREG-0711, Section 2.4.3
- Description of means and process by which HFE issues are tracked to resolution
- Description of means used to document issues
- Description of procedures used to define individual responsibilities for issue identification, logging, tracking, analysis, and resolution acceptance
- Description of how each issue is tracked to completion to ensure it is appropriately addressed prior to fuel load
- Description of the general technical approach to address following HFE activities including the integration and scheduling of these activities with the overall design effort (the detailed objectives, scope, methodology, and results of these activities are described in subsequent sections of Chapter 18):
  - Operating experience review
  - Functional requirements analysis and functional allocation
  - Task analysis
  - Staffing and qualifications
  - Human reliability analysis
  - Human-system interface design

18.1 HFE Program Management (cont.)

PSAR (cont.)

- Procedure development
- Training program development
- Human factors verification and validation
- Design implementation
- Human performance monitoring

– Description of the following aspects of the HFE technical program:
  - General HFE requirements, standards, and specifications
  - General HFE facilities, equipment, tools, and techniques, such as simulators used in HFE program

– Description of the objectives of the staffing and qualifications analyses, including scope of analyses performed

– Discussion of analyses for the requirements for the number and qualification of personnel are preformed in a systematic manner that includes a thorough understanding of task requirements and applicable regulatory requirements

– Scope includes:
  - Number and qualifications of personnel, following guidance of RG 1.8 for full range of plant conditions and tasks including:
    – Operational (normal, abnormal, and emergency)
    – Plant maintenance and testing (including surveillances)
  - Personnel to be considered includes:
    – Licensed control room operators (10 CFR 55)
    – 10 CFR 50.120 Personnel including:
      - Non-licensed operators
      - Shift supervisor
      - Shift technical advisor
      - I&C technician
      - Electrical maintenance
      - Mechanical maintenance
      - Radiological protection technician
      - Chemistry technician
      - Engineering support personnel
    – Other personnel that perform tasks directly related to plant safety

18.1 HFE Program Management (cont.)

DCD The DCD HFE Program and Scope incorporates by reference mPower technical reports. For Section 18.1 the primary technical report is the Human Factors Engineering (HFE) Program Management Plan. A combination of references to technical reports and PSAR discussions addresses the following:

- Goals of HFE Program
- Assumptions and constraints on design
- Scope of HFE applicability (including main control room, remote shutdown station, technical support center, emergency operations center, and risk-important local control stations)
- Identification of plant personnel affected by HFE activities (all plant personnel who are expected to perform tasks related to plant safety are identified).
- HFE design team and organization including:
    - Areas of responsibility with respect to HFE program
    - Placement within the overall design organization
    - Authority to provide reasonable assurance that all of its area of responsibility area accomplished and to identify problems in the implementation of the overall plant design. Design team composition with respect to areas of expertise
    - Staffing in terms of job descriptions and assignments of team personnel
- Description of the HFE process and procedures listed in NUREG-0711, Section 2.4.3
- Description of means and process by which HFE issues are tracked to resolution
- Description of means used to document issues
- Description of procedures used to define individual responsibilities for issue identification, logging, tracking, analysis, and resolution acceptance.
- Description of how each issue is tracked to completion to ensure it is appropriately addressed prior to fuel load
- Description of the general technical approach to address following HFE activities including the integration and scheduling of these activities with the overall design effort (the detailed objectives, scope, methodology, and results of these activities are described in subsequent sections of Chapter 18):
    - Operating experience review
    - Functional requirements analysis and functional allocation
    - Task analysis
    - Staffing and qualifications
    - Human reliability analysis
    - Human-system interface design
    - Procedure development

18.1 HFE Program Management (cont.)

DCD (cont.)

- Training program development
- Human factors verification and validation
- Design implementation
- Human performance monitoring

– Description of the following aspects of the HFE technical program:
  - General HFE requirements, standards, and specifications
  - General HFE facilities, equipment, tools, and techniques, such as simulators used in HFE program

– Description of the objectives of the staffing and qualifications analyses, including scope of analyses performed

– Discussion of analyses for the requirements for the number and qualification of personnel are preformed in a systematic manner that includes a thorough understanding of task requirements and applicable regulatory requirements

– Scope includes:
  - Number and qualifications of personnel, following guidance of RG 1.8 for full range of plant conditions and tasks including:
    - – Operational (normal, abnormal, and emergency)
    - – Plant maintenance and testing (including surveillances)
  - Personnel to be considered includes:
    - – Licensed control room operators (10 CFR 55)
    - – 10 CFR 50.120 Personnel including:
      - Non-licensed operators
      - Shift supervisor
      - Shift technical advisor
      - I&C technician
      - Electrical maintenance
      - Mechanical maintenance
      - Radiological protection technician
      - Chemistry technician
      - Engineering support personnel
    - – Other personnel that perform tasks directly related to plant safety

– Identification of applicable design acceptance criteria (DACs)

FSAR Same contents as the mPower standard plant DCD Section 18.1 updated to reflect plant-specific information as applicable, and a commitment to close all applicable DACs prior to issuance of OL

18.2 Operating Experience Review

PSAR The Clinch River HFE Operations Experience Review (OER) incorporates by reference the related mPower technical reports. For Section 18.2 the primary technical report is the HFE Operating Experience Plan. A combination of references to technical reports and PSAR discussions addresses the topics and goals of OER Program. The PSAR includes a commitment to provide additional OER information in the FSAR.

DCD The DCD Operations Experience Review incorporates by reference the mPower technical reports. For Section 18.2 the primary technical report is the HFE Operating Experience Plan. A combination of reference to applicable technical reports and the DCD discussion covers the following topics:

- Description of the operating experience review and how it is used to identify HFE-related safety issues including:
  - Objectives and Scope
  - Methodology
  - Results
- Description of the objectives of the OER process and scope of the analyses performed including OER analyses related to the following topics:
  - Predecessor plant(s) and systems
  - Experience in industries with applicable systems
  - Industry HSI experience
  - Risk-important human actions (HAs)
  - Specifically identified industry issues
  - Issues identified by plant personnel
  - Issue resolution
- Description of the administrative procedures for evaluating operating, design, and construction experience, and for ensuring the applicable industry experiences will be provided in a timely manner to those designing and constructing the plant as required by 10 CFR 50.34(f)3)(i)
- Identification of previous or predecessor design/plant that is used as part of the design basis (if applicable). If there is more than one predecessor plant/design, the role of each is clearly defined.
  - Describe how HFE-related problems and issues for each predecessor plant/design are identified and analyzed to avoid problems in the new design
  - Address how positive features of previous plants/designs are identified, evaluated, and retained
  - Address human factor issues related to predecessor plants.

18.2 Operating Experience Review (cont.)

DCD (cont.)

- For Risk-Important Human Actions (HAs):
  - Identify HAs in predecessor plants and determine if they remain risk important in new design
  - For applicable HAs, identify the scenarios where HAs are relied upon during plant operations and state whether the actions are successfully completed, noting design aspects that help ensure success, or if errors resulted from HA, identify insights for improvement in human performance
  - Where HAs for new plant are different from predecessor plant, identify operational experience related to the different HAs
  - Identify HAs from OER that require special attention during design, noting insights that would be beneficial during HFE design and implementation
- For HFE Technology, describe:
  - Operating experience associated in the design
  - For new technology, obtain and describe experience from applications of the new technology, even if not from predecessor plant
- For recognized industry issues, describe how the design addresses recognized HFE issues, including issues in the following categories (experience in these areas since 1996 is reviewed as appropriate):
  - USI and GSI
  - TMI issues
  - NRC Generic Letters and Information Notices
  - Reports issued by the NRC Office for Analysis and Evaluation of Operational Data
  - Low power and shutdown operations
  - Operating plant event reports
- Issues identified by plant personnel interview conducted during the OER, to determine OE related to predecessor plants and systems are described, and information summarized as it relates to plant operations and HFE design areas listed in NUREG-0711, Section 3.4.1(4)
- Description of how OER issues are entered into HFE tracking system (Issue Analysis, Tracking, and Review)
- A summary of the results of the OER, including:
  - Source of materials
  - Sample of OER-identified issues along with resolution
- Reference to the database where issues are maintained
- Identification of applicable DACs

18.2 Operating Experience Review (cont.)

FSAR Same contents as the mPower standard plant DCD Section 18.3 updated to reflect plant-specific information as applicable, and a commitment to close all applicable DACs prior to issuance of OL

18.3 Functional Requirements Analysis and Function Allocation

PSAR The Clinch River PSAR Section 18.3 incorporates by reference mPower technical reports related to HFE. The primary technical report for this section is the Functional Requirements Analysis (FRA) Implementation Plan. The PSAR references the applicable technical reports and identifies a commitment to provide the details of the completed FRA and Function Allocation in the FSAR.

DCD The DCD Functional Requirements Analysis and Function Allocation section incorporates by reference mPower technical reports related to HFE. The primary technical report for this section is the Functional Requirements Analysis Implementation Plan. References to appropriate technical reports and DCD discussions in Section 18.3 address:

- Functional Requirements Analysis to include a description of objectives of functional requirements analysis and the scope of analysis performed
  - Objective is to document that allocation of functions to human and systems resources is allocated in a manner to take advantage of human strengths and avoid human weaknesses.
  - Scope includes identification and analyses of functions that must be performed to satisfy safety objectives.
- Function Allocation Analysis describes objectives of function allocation analysis and scope of analysis performed
  - Scope includes analysis of requirements for plant control and assignment of control functions to
    - Personnel
    - System elements (e.g., automatic control or passive systems)
    - Combination of personnel and system elements
- Methodology for functional requirements analysis includes:
  - Description of methodology used to perform functional requirements Analysis
  - Description of how functional requirements analysis will be kept current over the life of the plant
  - Description of functions and systems with comparison to reference plants/systems
  - For each safety function, clear definition of the set of plant system configurations or success paths for or capable of carrying out function
  - Description of each high-level function and related parameters
  - Documentation of technical basis for modification to high-level functions in the new design (compared to predecessor design)
  - Description of functional requirements analysis to show following criteria are met:
    - All high-level functions necessary to achieve safe operation are identified

18.3 Functional Requirements Analysis and Function Allocation (cont.)

DCD (cont.)

- All requirements of each high-level function are identified
- Methodology for Function Allocation Analysis including:
  - Description of methodology used to perform functional allocation analysis
  - Description of how the function allocation will be kept current over the plant's life cycle
  - Provide documented technical bases for all function allocations including:
    - Allocation criteria
    - Rationale
    - Analysis
  - Description of how OER is used to identify needed modifications to function allocation
  - Description of primary allocations to personnel as well as responsibilities if personnel assume manual control in event of automatic system failure
  - Description of verification of function allocation to show allocation of functions results in a coherent role for plant personnel
- Summary of the results of functional requirements analysis and function allocation. Summary includes discussion of:
  - Results of analyses using methodology discussed above.
  - Final safety functions along with analyses used to obtain those functions
  - Final plant function allocations along with analyses used to obtain those function allocations.
  - Reference to reports that contain more detailed analyses (if applicable)
- Identification of applicable DACs

FSAR Same contents as the mPower standard plant DCD Section 18.3 updated to reflect plant-specific information as applicable, and a commitment to close all applicable DACs prior to issuance of OL

18.4 Task Analysis

PSAR The Clinch River PSAR's Task Analysis section incorporates by reference mPower technical reports related to HFE. The primary technical report for this section is the Task Analysis Implementation Plan. The PSAR includes a commitment to the mPower technical report, and indicates that further information will be provided in the FSAR.

DCD The DCD Task Analysis section incorporates by reference mPower technical reports related to HFE. For Section 18.4 the primary technical report is the Task Analysis Implementation Plan. Reference to the applicable technical reports addresses the following program aspects:

- Description of objectives and scope of task analysis including:
  - Assumptions
  - Bounding Conditions
- Scope description addresses:
  - How representative and important operations, maintenance, test, inspection, and surveillance tests are selected
  - How range of operating modes included in analyses is selected
- Description of the use of PRA/human reliability analysis (HRA) for the identification of risk-important HAs, including monitoring and backup of automatic actions
- Description of methods used to analyze tasks including:
  - Means by which high-level descriptions are developed into detailed task requirements
  - Methods used to describe tasks and illustrate their relationship
  - Methods used allocate tasks to members of operating crew
  - How the skills necessary for task performance are determined
- Description of the minimum inventory of human-system interfaces (i.e., alarms, controls, and displays) needed to implement the plant's emergency operating procedures, bring the plant to a safe condition, and to carry out those operator actions shown to be risk important by the PRA (in accordance with DI&C-ISG-05 (Item 2))
- Description of methodology and criteria used to identify a minimum inventory of alarms, controls, and displays and the Human Factors and I&C criteria used to identify this minimum inventory
- Summary of results of task analysis, including identification of where and how the detailed results are documented
- Description of how the task analysis results are used as input to the design of HSIs, procedures, and training programs
- Description of the iterative nature of staffing analysis and how initial staffing goals are/have been reviewed and modified as the analyses with other HFE elements are/were completed
- Analysis and justification for exemption from 10 CFR 50.54 (i) through (m) requirements

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18.4 Task Analysis (cont.)

DCD (cont.)

- A summary including:
  - Results of analyses obtained using methodology discussed above
  - Sufficient detail to permit understanding how the methodology is implemented to provide results
- Final minimum staffing levels for all personnel identified in above scope
- Reference to additional reports as applicable
- Identification of applicable DACs

FSAR Same contents as the mPower standard plant DCD Section 18.4 updated to reflect plant-specific information as applicable, and a commitment to close all applicable DACs prior to issuance of OL

18.5 Staffing and Qualification

PSAR PSAR Section 18.5 related to the Clinch River Staffing and Qualification plans is addressed as part of HFE Program Management Plan and Task Analysis as discussed in PSAR Sections 18.1 and 18.4.

DCD DCD Section 18.5, Staffing and Qualification, for the mPower standard plant design is addressed as part of HFE Program Management Plan and Task Analysis as discussed in DCD Sections 18.1 and 18.4.

FSAR Same contents as the mPower standard plant DCD Section 18.5 updated to reflect plant-specific information as applicable

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18.6 Human Reliability Analysis

PSAR The Clinch River PSAR Section 18.6 contains a commitment to implement the mPower Technical Report, Human Reliability Implementation Plan, and address additional aspects of the Human Reliability Analysis (HRA) in the FSAR.

DCD DCD Section 18.6 addresses the following aspects of the HRA by a combination of reference to the technical report, Human Reliability Implementation Plan, including:

- Description of the objectives of the use of HRA in the HFE program
- In conjunction with Section 19, this section will discuss how HRA/PRA results have been incorporated into other activities of the HFE program so that risk-important HAs have been thoroughly addressed in plant design
- Description of the use of PRA/HRA to identify risk-important HAs including:
  - Various portion of the PRA considered in determining HAs including:
    - Level 1 (core damage)
    - Level 2 (release from containment)
    - Post-core damage actions
    - Internal and external event portions of the PRA
    - Low power and shutdown PRA
- Description of:
  - Importance Measures
  - HRA sensitivity analyses
  - Threshold criteria (with bases) used for risk-important HAs
  - Human actions related to passive systems and computer-based HSIs
  - Methodology by which PRA/HRA results and risk-important HAs are addressed by HFE design team to minimize likelihood of operator error and provide for error detection and recovery
  - Process with which HRA assumptions are validated during HFE design process
- Description of the minimum inventory of human-system interfaces (i.e., alarms, controls, and displays) needed to implement the plant's emergency operating procedures, bring the plant to a safe condition, and to carry out those operator actions shown to be risk important by the PRA (in accordance with DI&C-ISG-05 (Item 2))
- A list of risk-important HAs and summary of how the HAs and associated tasks and scenarios are addressed during design process to ensure HAs are supported by design and within human performance capabilities
- Discussion of results of the validation of HRA assumptions, and, as necessary, reference to reports that contain more detailed analyses
- Identification of applicable DACs

FSAR Same contents as the mPower standard plant DCD Section 18.6 updated to reflect plant-specific information as applicable, and a commitment to close all applicable DACs prior to issuance of OL

18.7 Human-System Interface Design

PSAR The Clinch River PSAR HSI Design section incorporates by reference mPower technical reports. For Section 18.7 the primary technical report is the Human System Interface Implementation Plan. The PSAR commits to the implementation of the applicable technical reports and to provide further information in the FSAR.

DCD The DCD HSI Design section incorporates by reference mPower technical reports. The primary technical report for this section is the Human System Interface Implementation Plan. The DCD references the appropriate technical reports. References to technical reports and the section content addresses the following topics of HSI Design:

- Description of the HSI design process and scope, through the systematic application of HFE principles and criteria, including translation of function and task requirements into detailed design of:
  - Alarms
  - Displays
  - Controls
  - Other aspects of HSI
- Description of the process by which HSI design requirements are developed, and HSI designs identified and refined
- HSI Design Inputs - Identification of source of information used as input to HSI design
- Concept of Operations – Description of operations used as basis for HSI design including:
  - Crew composition
  - Roles and responsibilities of individual crewmembers
  - Personnel interaction with plant automation
  - Use of control room resources
  - Methods to ensure good coordination of crewmember activities
- Functional Requirements Specification – description of functional requirements for HSI resources
- HSI Concept Design
  - Overview of the basis for the design that includes conceptual designs that are considered or predecessor designs that are used
- HSI Detailed Design and Integration
  - Description of HSI style guide including: scope, technical contents, procedures for use, and guide development and maintenance
  - Discussion of the following:
    - How design supports personnel in monitoring and controlling plant while minimizing interface demands
    - How design addresses safety parameter display system (SPDS) referenced in 10CFR50.34(f)(2)(iv)

18.7 Human-System Interface Design (cont.)

DCD (cont.)

- How design minimizes probability of error for risk-important HAs and provides error detection
- Basis for allocation of HSI functions to Main Control Room or LCS
- Basis for control room layout and organization of HSIs
- How control room supports a range of anticipated staffing
- How HSI characteristics mitigate excessive fatigue
- How HSI characteristics support human performance under full range of environmental conditions
- Means of inspection, maintenance, tests, and repair of HSI without interference with control room tasks

- HSI Tests and Evaluations – Description of tests and evaluations performed of HSI design/integration including:
  - Types of activities (trade-off evaluations and performance based tests) identifying factors in evaluating, selection criteria, results objectives and results
  - Methodology including test beds, performance measures and criteria, study participants, test design, and data analysis
  - Description of the use of the test results and evaluations, including specifically how identified problems/issues are resolved
- Description of the final HSI design addressing the considerations of the following:
  - Overview of HSI Design and Key Features
    - Description of the overall design concept and rationale including:
      - Main control room
      - Remote shutdown facility
      - LCS
      - Key features of the design (information display, "soft" controls, computer based procedures, alarm processing, control room layout)
      - Computer based procedures in accordance with DI&C-ISG-05 (Item 1)
  - Safety Aspects of the HSI – coordinate with Section 7 and describe plant –specific implementation of following safety aspects:
    - Safety function monitoring
    - Periodic testing of protective system actuations
    - Bypassed and inoperable status indication
    - Manual initiation of protective actions
    - Instrumentation to assess plant and environmental conditions (post accident)
    - Setpoints for safety-related instrumentation
    - HSI for TSC & EOF
    - Minimum inventory of fixed position alarms, controls, and

18.7 Human-System Interface Design (cont.)

DCD (cont.)

displays

- HSI Change Process – description of process after plant operation for:
  - HSI modification and updating
  - Temporary HSI changes
  - Operator defined HSI
  - Description of procedures for operator-initiated changes to HSI
  - Description of criteria used to determine when HSI change/modification requires formal engineering change process
- Identification of applicable DACs

FSAR Same contents as the mPower standard plant DCD Section 18.7 updated to reflect plant-specific information as applicable, and a commitment to close all applicable DACs prior to issuance of OL

18.8 Procedure Development

PSAR PSAR Section 18.8 related to Clinch River Procedure Development is addressed in Chapter 13.

DCD DCD Section 18.8, Procedure Development, for the mPower standard plant is addressed in Chapter 13.

FSAR Same contents as the mPower standard plant DCD Section 18.8 updated to reflect plant-specific information as applicable

18.9 Training Program Development

PSAR PSAR Section 18.9 related to Clinch River Training Program Development is addressed in Chapter 13.

DCD DCD Section 18.9, Training Program Development, for the mPower standard plant is addressed in Chapter 13.

FSAR Same contents as the mPower standard plant DCD Section 18.9 updated to reflect plant-specific information as applicable

18.10 Verification and Validation

PSAR The Clinch River PSAR Section 18.10 incorporates by reference mPower technical report, HFE Verification and Validation (V&V) Implementation Plan, and includes a commitment to provide additional information in the FSAR.

DCD DCD Section 18.10 describes the following:

- V&V activities to confirm HSI conforms to HFE design principles, enables personnel to perform tasks to achieve plant safety/operational goals. Scope of V&V includes main control room, remote shutdown panel, and local stations associated with risk-informed HAs.
- Identification of which aspects of plant HFE are included in the HSI task support verification, HFE design verification and integrated system validation
- V&V of manual operator actions that are credited for safety I&C common cause failure, if any, based on the final analysis of diversity and defense-in-depth
- Methodology addresses:
  - Operational Conditions Sampling including:
    - Conditions that are representative of the range of events encountered during plant operation
    - Characteristics expected to contribute to system performance variation
    - Safety significance of HSI components
  - Design Verification providing description(s) of:
    - Inventory of all HSI components associated with personnel tasks for operational conditions within scope of V&V
    - HSI task support verification used to verify that HSI provides all of the alarms, information, and control capabilities required
    - HFE methods and appropriate used to verify that characteristics of the HSI, and the environment in which it is used, conform to HFE guidelines
    - How evaluation criteria are developed and how human engineering discrepancies (HEDs) are identified
  - Integrated System Validation
    - Evaluation of integration of hardware, software, and personnel elements supporting safe operation of the plant
    - Description of the methods for including following aspects of validation methodology:
      - Test Objectives
      - Validation test beds
      - Plant personnel
      - Scenario definition
      - Performance measurement (characteristics, selection, criteria)

18.10 Verification and Validation (cont.)

DCD (cont.)

- Test design
- Data analysis and interpretation
- Conclusions

– Description of how HEDs are identified during validation

- Human Engineering Discrepancy Resolution
  - – Description of the process by which HEDs are prioritized and resolved
  - – Identification of design changes made for HEDs along with current status
  - – For HEDs requiring design change, description of V&V evaluation criteria and compliance

– Summary of the V&V results including:

- Identification & resolution of HED
- Clear identification of V&V criteria that can't be evaluated until after fuel load, when V&V will be performed, and how it will be communicated to NRC
- Information on how detailed results are documented and how they can be accessed by NRC staff

– Identification of applicable DACs

FSAR Same contents as the mPower standard plant DCD Section 18.10 updated to reflect plant-specific information as applicable, and a commitment to close all applicable DACs prior to issuance of OL

18.11 Design Implementation

PSAR PSAR Section 18.11 describes the programmatic method that will be used for design implementation. The Clinch River PSAR incorporates by reference the mPower technical report, HFE Design Implementation Plan, and identifies a commitment to provide additional information in the FSAR.

DCD DCD Section 18.11 provides a description of the objectives and scope of design implementation including:

- How verification that as-built conforms to V&V design from HFE design process
- Objectives and scope of design implementation. Scope includes:
  - V&V of design aspects that can't be completed in HSI V&V
  - Confirmation that as-built HSI, procedures, and training conform to approved design
  - Confirmation that all HFE issues in tracking system are addressed
- Description of the design implantation including:
  - How aspects of design not addressed in V&V will be evaluated
  - How the final as-built HSIs, procedures and training will be compared with detailed design to verify they conform to design resulting from HFE design process and V&V activities
  - Process for correcting identified discrepancies
  - Justification for not changing design features that cause discrepancies
  - Process for ensuring HFE-related issues are documented, tracked, and resolved
- Description of final documentation that will be developed to show successful completion of the Section 18.11 activity "Design Implementation"
- Identification of applicable DACs

FSAR Same contents as the mPower standard plant DCD Section 18.11 updated to reflect plant-specific information as applicable, and a commitment to close all applicable DACs prior to issuance of OL

18.12 Human Performance Monitoring

PSAR PSAR Section 18.12 documents a human performance monitoring strategy for determining no significant safety degradation occurs due to changes made to the plant over time. This section contains a commitment to discuss the human performance monitoring program in the FSAR.

DCD The DCD Section 18.12 provides the following:

- Description of the objective and scope of the human performance monitoring program. The program description addresses how the program provides reasonable assurance of how the following criteria are met:
  - Design can be effectively used by personnel, including within control room, and between control room and LCS and support centers
  - Changes to HSIs, procedures, and training do not have adverse effects on personnel performance
  - HA can be accomplished within established time and performance criteria
  - Acceptable level of performance established during integrated system validation is maintained
- Description of methodology for human performance monitoring including:
  - Human performance monitoring strategy
  - How it trends human performance relative to changes implemented after startup
  - How it demonstrates that performance is consistent with that assumed in the various analyses conducted to justify changes
  - How the following criteria are met:
    - HAs are monitored commensurate with their safety importance
    - Feedback of information and corrective actions are accomplished in a timely manner
    - Degradation in performance can be detected and corrected before plant safety is compromised
    - Available information that most closely approximates performance data in actual conditions is used when plant or personnel performance under actual design conditions is not readily measurable
    - How the program provides for specific:
      - Cause determination
      - Trending of performance degradation and failures Determination of appropriate corrective actions
- Description of the documentation to be maintained after the program is implemented
- Identification of applicable DACs

18.12 Human Performance Monitoring (cont.)

FSAR	Same contents as the mPower standard plant DCD Section 18.11 updated to reflect plant-specific information as applicable, and a commitment to close all applicable DACs prior to issuance of OL

Clinch River Regulatory Framework Document
NRC Version

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
19.0 Probabilistic Risk Assessment and Severe Accident Evaluation	PSAR	10 CFR 50.34(f)(1)(i)	None	RG 1 206 - C I.19 Note: PSAR Section 19.0 will provide description of plans to prepare a plant-specific PRA as discussed in the NRC's 1/31/11 Response to the TVA Key Assumptions Letters dated 11/5/10 and 12/22/10.	19.0	RG 1.174 RG 1.200 SECY-90-016 and 6/26/90 SRM SECY-93-087 and 7/21/93 SRM SECY-96-128 and 1/15/97 SRM SECY-97-044 and 6/30/97 SRM NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50FR32138) NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (51FR28044) NRC Policy Statement on Nuclear Power Plant Standardization (52FR34884) NRC Policy Statement on Regulation of Advanced Nuclear Power Plants (59FR35461) NRC Policy Statement on Use of Nuclear PRA Methods in Nuclear Regulatory Activities (60FR42622)	ASME/ANS RA-S-2008 ASME/ANS RA-Sa-2009 ANSI-58.21 -2007	Yes - Provide PRA information based on current regulatory guidance	No	None	Commitments to the NRC with regard to PRA standards will be based upon guidance endorsed in RGs, BTPs, and SRP sections issued 6 months prior to the submittal of the PSAR
	DCD	10 CFR 52 47(8) 10 CFR 50.34(f)(1)(i) 10 CFR 52.47(a)(23) 10 CFR 52 47(a)(27)	None	RG 1 206 - C.I.19	19.0	RG 1.174 RG 1.200 SECY-90-016 and 6/26/90 SRM SECY-93-087 and 7/21/93 SRM SECY-96-128 and 1/15/97 SRM SECY-97-044 and 6/30/97 SRM NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50FR32138) NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (51FR28044) NRC Policy Statement on Nuclear Power Plant Standardization (52FR34884) NRC Policy Statement on Regulation of Advanced Nuclear Power Plants (59FR35461) NRC Policy Statement on Use of Nuclear PRA Methods in Nuclear Regulatory Activities (60FR42622)	ASME/ANS RA-S-2008 ASME/ANS RA-Sa-2009 ANSI-58 21-2007 NEI 00-02	N/A	N/A	None	Commitments to the NRC with regard to PRA standards will be based upon guidance endorsed in RGs, BTPs, and SRP sections issued 6 months prior to the submittal of the PSAR
	FSAR	10 CFR 50 34(f)(1)(i)	None	RG 1.206 - C.I.19	19 0 19.1	RG 1.174 RG 1 200 SECY-90-016 and 6/26/90 SRM SECY-93-087 and 7/21/93 SRM SECY-96-128 and 1/15/97 SRM SECY-97-044 and 6/30/97 SRM NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50FR32138) NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (51FR28044) NRC Policy Statement on Nuclear Power Plant Standardization (52FR34884) NRC Policy Statement on Regulation of Advanced Nuclear Power Plants (59FR35461) NRC Policy Statement on Use of Nuclear PRA Methods in Nuclear Regulatory Activities (60FR42622)	ASME/ANS RA-S-2008 ASME/ANS RA-Sa-2009 ANSI-58 21-2007 NEI 00-02	N/A	No	None	Commitments to the NRC with regard to PRA standards will be based upon guidance endorsed in RGs, BTPs, and SRP sections issued 6 months prior to the submittal of the PSAR

Note.
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

Clinch River Regulatory Framework Document
NRC Version

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
19.1 Probabilistic Risk Assessment	PSAR	10 CFR 50 34(f)(1)(i)	None	RG 1 206 - C.I.19 Note: PSAR Section 19 1 will provide an overview of plans to prepare a plant-specific PRA as discussed in the NRC's 1/31/11 Response to the TVA Key Assumptions Letters dated 11/5/10 and 12/22/10.	19 0	RG 1.174 RG 1.200 SECY-90-016 and 6/26/90 SRM SECY-93-087 and 7/21/93 SRM SECY-96-128 and 1/15/97 SRM SECY-97-044 and 6/30/97 SRM NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50FR32138) NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (51FR28044) NRC Policy Statement on Nuclear Power Plant Standardization (52FR34884) NRC Policy Statement on Regulation of Advanced Nuclear Power Plants (59FR35461) NRC Policy Statement on Use of Nuclear PRA Methods in Nuclear Regulatory Activities (60FR42622)	ASME/ANS RA-S-2008 ASME/ANS RA-Sa-2009 ANSI-58 21 -2007 NEI 00-02	Yes - Provide PRA information based on current regulatory guidance	No	3 2 3.9 16.1 17 4	None
	DCD	10 CFR 52 47(8) 10 CFR 50 34(f)(1)(i) 10 CFR 52.47(a)(23) 10 CFR 52 47(a)(27)	None	RG 1.206 - C.I.19	19 0 19.1	RG 1.174 RG 1.200 SECY-90-016 and 6/26/90 SRM SECY-93-087 and 7/21/93 SRM SECY-96-128 and 1/15/97 SRM SECY-97-044 and 6/30/97 SRM NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50FR32138) NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (51FR28044) NRC Policy Statement on Nuclear Power Plant Standardization (52FR34884) NRC Policy Statement on Regulation of Advanced Nuclear Power Plants (59FR35461) NRC Policy Statement on Use of Nuclear PRA Methods in Nuclear Regulatory Activities (60FR42622)	ASME/ANS RA-S-2008 ASME/ANS RA-Sa-2009 ANSI-58 21 -2007 NEI 00-02	N/A	N/A	3 2 3 9 16 1 17 4	None
	FSAR	10 CFR 50 34(f)(1)(i)	None	RG 1 206 - C.I.19	19.0 19.1	RG 1.174 RG 1 200 SECY-90-016 and 6/26/90 SRM SECY-93-087 and 7/21/93 SRM SECY-96-128 and 1/15/97 SRM SECY-97-044 and 6/30/97 SRM NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50FR32138) NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (51FR28044) NRC Policy Statement on Nuclear Power Plant Standardization (52FR34884) NRC Policy Statement on Regulation of Advanced Nuclear Power Plants (59FR35461) NRC Policy Statement on Use of Nuclear PRA Methods in Nuclear Regulatory Activities (60FR42622)	ASME/ANS RA-S-2008 ASME/ANS RA-Sa-2009 ANSI-58.21 -2007 NEI 00-02	N/A	No	3 2 3.9 16 1 17.4	None

Note:
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
	PSAR	10 CFR 50.34(f)(1)(i)	None	RG 1.206 - C.I.19 Note: PSAR Section 19.2 will provide an overview of plans to address severe accident issues and include a reference to the Environmental Report for a description of the methodology and criteria to be used for developing severe accident management design alternative (SAMDA) as discussed in the NRC's 1/31/11 Response to the TVA Key Assumptions Letters dated 11/5/10 and 12/22/10.	19.0	RG 1.174 RG 1.200 SECY-90-016 and 6/26/90 SRM SECY-93-087 and 7/21/93 SRM SECY-96-128 and 1/15/97 SRM SECY-97-044 and 6/30/97 SRM NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50FR32138) NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (51FR28044) NRC Policy Statement on Nuclear Power Plant Standardization (52FR34884) NRC Policy Statement on Regulation of Advanced Nuclear Power Plants (59FR35461) NRC Policy Statement on Use of Nuclear PRA Methods in Nuclear Regulatory Activities (60FR42622)	None	Yes - Provide PRA information based on current regulatory guidance	No	3.2 3.9 16.1 17.4	None
19.2 Severe Accident Evaluation	DCD	10 CFR 51.55 10 CFR 52.47(8) 10 CFR 50.34(f)(1)(i) 10 CFR 52.47(a)(23) 10 CFR 52.47(a)(27)	None	RG 1.206 - C.I.19	19.0 19.1	RG 1.174 RG 1.200 SECY-90-016 and 6/26/90 SRM SECY-93-087 and 7/21/93 SRM SECY-96-128 and 1/15/97 SRM SECY-97-044 and 6/30/97 SRM NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50FR32138) NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (51FR28044) NRC Policy Statement on Nuclear Power Plant Standardization (52FR34884) NRC Policy Statement on Regulation of Advanced Nuclear Power Plants (59FR35461) NRC Policy Statement on Use of Nuclear PRA Methods in Nuclear Regulatory Activities (60FR42622)	None	N/A	N/A	3.2 3.9 16.1 17.4	None
	FSAR	10 CFR 50.34(f)(1)(i)	None	RG 1.206 - C.I.19	19.0 19.1	RG 1.174 RG 1.200 SECY-90-016 and 6/26/90 SRM SECY-93-087 and 7/21/93 SRM SECY-96-128 and 1/15/97 SRM SECY-97-044 and 6/30/97 SRM NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50FR32138) NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (51FR28044) NRC Policy Statement on Nuclear Power Plant Standardization (52FR34884) NRC Policy Statement on Regulation of Advanced Nuclear Power Plants (59FR35461) NRC Policy Statement on Use of Nuclear PRA Methods in Nuclear Regulatory Activities (60FR42622)	None	N/A	No	3.2 3.9 16.1 17.4	None

Note:
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
19.3 Open and Confirmatory Items	PSAR	10 CFR 50.34(f)(1)(i)	None	RG 1 206 - C.I.19 Note: PSAR Section 19 3 will identify development of the plant-specific PRA as an Open Item.	19.0	N/A	N/A	Yes - Provide Open Items related to PRA based on current regulatory guidance	No	N/A	None
	DCD	10 CFR 52 47(a)(8) 10 CFR 50 34(f)(1)(i) 10 CFR 52 47(a)(23) 10 CFR 52.47(a)(27)	None	RG 1.206 - C I.19	19.0 19 1	N/A	N/A	N/A	N/A	3 2 3 9 16 1 17.4	None
	FSAR	10 CFR 50.34(f)(1)(i)	None	RG 1.206 - C I.19	19 0 19 1	N/A	N/A	N/A	No	3 2 3.9 16.1 17.4	None
19A Beyond Design Basis Aircraft Impact Assessment	PSAR	10 CFR 50.34(a)(13) 10 CFR 50.150	None	RG 1.206 - C.I.19	N/A - NUREG-0800 does not address large, commercial aircraft impact analysis requirements	RG 1.217	NEI 07-13	Yes - Address Beyond Design Basis Aircraft Impact Assessment per 10 CFR 50.150	No	1.2, 3.7, 3 8, 4.6, 5 4, 6 2, 6.3, 9.1, 9 5	None
	DCD	10 CFR 50.150 10 CFR 52 47(a)(28)	None	RG 1 206 - C I.19	N/A - NUREG-0800 does not address large, commercial aircraft impact analysis requirements	RG 1 217	NEI 07-13	N/A	N/A	1.2, 3.7. 3 8, 4.6, 5.4, 6 2, 6.3, 9.1, 9.5	None
	FSAR	10 CFR 50 34(b)(12) 10 CFR 50.150	None	RG 1.206 - C I.19	N/A - NUREG-0800 does not address large, commercial aircraft impact analysis requirements	RG 1 217	NEI 07-13	N/A	No	1.2, 3 7, 3.8, 4 6, 5 4, 6.2, 6 3, 9 1, 9.5	None
19B Regulatory Treatment of Non-Safety Systems	PSAR	10 CFR 50 34(f)(1)(i)	None	RG 1.206 - C IV 9	N/A - NUREG-0800 does not address issues related to Regulatory Treatment of Non-Safety Systems	SECY-94-084 SECY-93-087 SECY-95-132	IEEE-344-1987	Yes - RTNSS issue is beyond RG 1.70	No	3 9, 3.10, 3.11, 4 6, 6 2, 7.1, 7.7, 7.8, 9.1, 9 4, 9.5	None
	DCD	10 CFR 50.34(f)(1)(i) 10 CFR 52 47(a)(8) 10 CFR 52.47(a)(23) 10 CFR 52 47(a)(27)	None	RG 1 206 - C.IV.9	N/A - NUREG-0800 does not address issues related to Regulatory Treatment of Non-Safety Systems	SECY-94-084 SECY-93-087 SECY-95-132	IEEE-344-1987	N/A	N/A	3 9, 3.10, 3.11, 4 6, 6.2, 7.1, 7 7, 7.8, 9.1, 9 4, 9.5	None
	FSAR	10 CFR 50.34(f)(1)(i)	None	RG 1.206 - C.IV 9	N/A - NUREG-0800 does not address issues related to Regulatory Treatment of Non-Safety Systems	SECY-94-084 SECY-93-087 SECY-95-132	IEEE-344-1987	N/A	No	3 9, 3.10, 3.11, 4.6, 6 2, 7.1, 7.7, 7 8, 9.1, 9.4, 9 5	None

Note:
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal.

Section Number/Title	Submittal Document	Regulatory Requirements	Proposed Exemptions	Regulatory Basis for Section Content	NUREG-0800 (SRP) Section	Regulatory Guidance (See Note 1)	Industry Guidance	CPA Information Beyond RG 1.70	Changes to the Standard Plant Design	Related Sections	Key Issues
19C Availability Controls Manual	PSAR	N/A	No	RG 1 206, Section C.IV 9 Note: PSAR Appendix 19C will provide a preliminary version of a RTNSS Availability Control Manual consistent with RG 1 206, since mPower is passive plant design	N/A	SECY-94-084 SECY-95-132 SECY-96-128 SRM to SECY 94-084 dated June 30, 1994 NUREG/CR-6595, Rev 1 NRC Memorandum, L Joseph Callan (Executive Director for Operations) to Chairman Jackson, et al, "Implementation of Staff Position in SECY-96-128," June 23, 1997	None	Yes - RTNSS Information	No	19 2 19 6 19 7	There is a potential that Fukushima event may result in additional "deterministic" beyond design basis events and plant systems relied to cope with those events will need to be classified as RTNSS.
	DCD	N/A	No	RG1 206, Section C.IV.9	N/A	SECY-94-084 SECY-95-132 SECY-96-128 SRM to SECY 94-084 dated June 30, 1994 NUREG/CR-6595, Rev. 1 NRC Memorandum, L. Joseph Callan (Executive Director for Operations) to Chairman Jackson, et al, "Implementation of Staff Position in SECY-96-128," June 23, 1997	None	N/A	N/A	19 2 19 6 19 7	There is a potential that Fukushima event may result in additional "deterministic" beyond design basis events and plant systems relied to cope with those events will need to be classified as RTNSS.
	FSAR	N/A	No	RG1 206, Section C IV 9	N/A	SECY-94-084 SECY-95-132 SECY-96-128 SRM to SECY 94-084 dated June 30, 1994 NUREG/CR-6595, Rev. 1 NRC Memorandum, L Joseph Callan (Executive Director for Operations) to Chairman Jackson, et al. "Implementation of Staff Position in SECY-96-128," June 23, 1997	None	N/A	No	19.2 19.6 19.7	There is a potential that Fukushima event may result in additional "deterministic" beyond design basis events and plant systems relied to cope with those events will need to be classified as RTNSS

Note:
RG revisions are not identified as these will be consistent with the versions in effect 6 months prior to the PSAR submittal

19.0 Probabilistic Risk Assessment and Severe Accident Evaluation

PSAR The Clinch River SMR PSAR Section 19.0 provides an introduction to PRA and severe accident evaluation and includes the following information:

- the purpose and objectives of the plant-specific PRA and severe accident evaluations,
- the requirements in 10 CFR Part 52 (specific to PRA) and 10 CFR Part 50, as well as the related Commission policies and positions,
- the potential uses of PRA and severe accident evaluations, and
- identification of the structure of PSAR Chapter 19[1].

DCD DCD Section 19.0 provides an introduction to probabilistic risk assessment and severe accident evaluation and includes the following information:

- the purpose and objectives of the design-specific PRA and severe accident evaluations,
- the requirements in 10 CFR Part 52 and 10 CFR Part 50, as well as the related Commission policies and positions,
- uses of PRA and severe accident evaluations as follows:
  - During the design phase:
    i. identify and address potential design features and plant operational vulnerabilities, where a small number of failures could lead to core damage, containment failure, or large releases (e.g., assumed individual or common-cause failures could drive plant risk to unacceptable levels with respect to the Commission's goals, as presented below)
    ii. select among alternative features, operational strategies, and design options
    iii. Reduce or eliminate the significant risk contributors of existing operating plants that are applicable to the new design by introducing appropriate features and requirements
  - Identify risk-informed safety insights based on systematic evaluations of the risk associated with the design, construction, and operation of the plant in order to identify and describe the following:
    i. design's robustness, levels of defense-in-depth, and tolerance of severe accidents initiated by either internal or external events
    ii. risk significance of specific human errors associated with the design, including a characterization of significant human errors that may be used as an input to operator training programs and procedure refinement

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[1] Per agreement between NRC and TVA, the PRA and its results will not be included in the Clinch River PSAR [Reference: NRC Response Letter to TVA Key Assumptions Letters, dated 1/31/11]. The PSAR will contain TVA's methodology and criteria for developing a PRA.

19.0 Probabilistic Risk Assessment and Severe Accident Evaluation (cont.)

DCD (cont.)

- Demonstrate how the risk associated with the design compares against the Commission's goals of less than $1x10^{-4}$/year for core damage frequency and less than $1x10^{-6}$/year for large release frequency.
- Compare the design against the Commission's approved use of a containment performance goal, which includes (1) a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges and (2) a probabilistic goal that the conditional containment failure probability be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA
- Assess the balance of preventive and mitigative features of the design, including consistency with the Commission's guidance in SECY-93-087 and the associated SRM
- Demonstrate whether the plant design, including the impact of site-specific characteristics, represents a reduction in risk compared to existing operating plants
- Discuss how the results and insights of the PRA support the process used to demonstrate whether the RTNSS is sufficient and identify the SSCs included in RTNSS
- Discuss how the results and insights of the PRA identify and support the development of specifications and performance objectives for the plant design, construction, inspection, and operation, such as ITAAC; the RAP; TS; and COL action items and interface requirements

- identification of the structure of Chapter 19
- Adequate level of documentation of the PRA and severe accident evaluations to enable the NRC staff to determine the acceptability of the risks to public health and safety associated with operation of a proposed new plant

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FSAR Same contents as mPower standard plant DCD Section 19.0 with supplemental information on the plant-specific PRA and severe accident evaluations as applicable

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19.1 Probabilistic Risk Assessment

PSAR The PSAR Section 19.1 provides a summary description of methodology and tools to be used to produce the Clinch River SMR PRA. The PSAR also:

- describes the process and methodology to develop the mPower standard plant Level 1 PRA,
- describes the process and methodology to develop the mPower standard plant Level 2 PRA, and
- references Appendix 19A for discussion of beyond design basis aircraft impact assessment as required by 10 CFR 50.150.

DCD DCD Section 19.1 provides a description of the PRA and its results. The following information is provided in Section 19.1

- the uses and risk-informed applications of the PRA in the design phase (through design certification) and the DCD cross-references to specific program descriptions, as appropriate
- the technical adequacy of the PRA in the context of its uses and the risk-informed applications
- the scope of the PRA as a Level 1 and Level 2 PRA that includes internal and external events and addresses all plant operating modes for which standards exist
- characterization of the PRA's level of detail as sufficient to support the uses and applications that have been identified
- the methodologies, processes, analyses, and personnel associated with the PRA and how they comply with the provisions for nuclear plant quality assurance (e.g., Appendix B to 10 CFR Part 50)
- adherence to the recommendations provided in RG 1.200 pertaining to technical adequacy. Alternatively, the acceptability of, alternative measures for addressing PRA technical adequacy
- the key preventive features that are intended to minimize initiation of plant transients, arrest the progression of plant transients once they start, and prevent severe accidents (core damage)
- the key mitigative features that are intended to arrest progression of the core damage event and maintain the integrity of the reactor vessel and containment pressure boundary
- the mitigating features that are intended to terminate releases from containment and minimize offsite doses/consequences
- the features and requirements introduced to reduce or eliminate the known weaknesses and vulnerabilities in current reactor designs, including the effect of new design features and operational strategies on plant risk
- the PRA-based insights and assumptions used to develop design requirements
- the Level 1 internal events PRA for operations at power and its results, including:
  - the methodology used to develop the Level 1 PRA model
  - listing of the significant internal initiating events (including internal floods) that are addressed in the PRA

19.1 Probabilistic Risk Assessment (cont.)

DCD (cont.)

- the success criteria used to delineate accident sequences, discussion of how they were determined, and any T-H codes used
- summary of the significant accident sequences modeled in the PRA
- listing of the plant systems and associated functions that are included in the PRA model, and identify their interdependencies
- the source of all numerical data, especially for numerical data that is based on expert judgment or expert elicitation
- the PRA software platform used to construct the model
- the truncation frequency used to solve the PRA model
- the total mean core-damage frequency
- the "significant" core damage sequences, and their mean core-damage frequencies
- the significant internal initiating events, and their percent contributions to the total core-damage frequencies
- the significant functions, SSCs, and operator actions, and their risk importance measures
- the PRA "assumptions" and "PRA-based insights"
- the results and insights from importance, sensitivity, and uncertainty analyses

- the Level 2 internal events PRA for operations at power, and its results:
  - the frequency of containment failure (including containment bypass, early containment failure, intermediate containment failure, late containment failure, and containment isolation failure) and conditional containment failure probability
  - the interface with the core damage evaluation (Level I PRA)
  - the severe accident physical processes/phenomena and modeling
  - the success criteria used to delineate accident sequences, and a discussion of how they were determined, and any T-H codes used
  - the accident classes/release categories
  - characterization of the containment ultimate pressure capacity, its determination, and any computer codes used
  - listing of the plant systems and associated functions that are included in the Level 2 PRA model, and their interdependencies. One acceptable way to provide dependency information is to include a system dependency matrix
  - the total mean large release frequency and total mean conditional containment failure probability
  - the significant large release sequences, and their mean release frequencies

19.1 Probabilistic Risk Assessment (cont.)

DCD (cont.)

- listing of the significant internal initiating events, and provide their percent contributions to the total large release frequency
- the risk-significant functions, SSCs, phenomena and operator actions, and their risk measures
- characterization of the containment performance
- the PRA assumptions and PRA-based insights
- the results and insights from importance, sensitivity, and uncertainty analyses

- description of the external events evaluated and safety insights. If some external events were screened out or incorporated into other evaluations, the methods used to conduct the screening and bounding analyses is described. FSAR cross-references to specific external events, as appropriate, are included. The risk evaluation for following external events and results is included:
  - seismic margin analysis
  - the seismic risk evaluation for operations at power, including its results
  - the internal fire risk evaluation for operations at power, including its results
  - If not screened or bounded, risks for other external events such as high winds, tornados, external floods, and hurricanes are evaluated and results described.
- description of the risk evaluation for low power and shutdown operations PRA, including its results, including:
  - the other (non-full-power) modes of operation addressed in the risk evaluation
  - the methods used to conduct the grouping and bounding analyses, if the evaluation of some modes is incorporated into (or bounded by) the evaluations of other modes
  - the methodology used to develop the low-power and shutdown PRA models
  - the initiating events (internal and external) that are addressed in the PRA
  - the success criteria used to delineate accident sequences, a discussion of how they were determined, and any T-H codes used
  - Summary of the accident sequences modeled in the PRA
  - listing of the plant systems and associated functions that are included in the PRA model
  - the source of all numerical data, especially for numerical data that is based on expert judgment or expert elicitation
  - the PRA software platform used to construct the model
  - the truncation frequency used to solve the PRA model
  - the total mean core-damage frequency
  - For each plant operating state, the significant core-damage, large release, and offsite consequence (optional) sequences, and their mean values are described.

19.1 Probabilistic Risk Assessment (cont.)

DCD (cont.)

- For each plant operating state, the significant initiating events, including both internal and external events, are identified, and their percent contributions to the total core-damage frequency and the large release frequency are provided.
- For each plant operating state, the significant functions, SSCs, and operator actions, and their risk importance measures are provided.PRA assumptions and PRA-based insights
- the results and insights from importance, sensitivity, and uncertainty analyses

- discussion of PRA-based insights to programs and processes, including assumptions regarding SSC and operator performance and reliability, ITAAC, interface requirements, and cross-references to the specific sections that describe and evaluate each of the following:
  - PRA input to the reliability assurance program
  - PRA input to the regulatory treatment of nonsafety-related systems program
- description of the use of PRA during the OL application phase including:
  - the use in support of other licensee programs such as the human factors program and severe accident management program
  - specific risk-informed applications being implemented
  - cross-references to specific program descriptions (e.g., 10 CFR 50.69 implementation, NFPA-806 implementation), as appropriate
- description of the use of PRA during the construction phase from issuance of CP to initial fuel loading including:
  - cross-references to specific program descriptions
  - the use in support of other licensee programs (e.g., human factors program)
- maintenance of the PRA model to reflect plant changes such as maintenance updates, modifications, procedure changes, or performance data.
- upgrading of the PRA model to incorporate new methodologies or changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. A peer review will be completed after a PRA upgrade has been performed.

FSAR Same contents as the mPower standard plant DCD Section 19.1 with the following supplemental information:

- description of use of PRA during the construction phase from issuance of CP to initial fuel loading including specific risk-informed applications that will be implemented
- description of the use of PRA during the operational phase including:
  - how PRA would be used during plant operations (commencing with initial fuel loading and continuing through plant commercial operation) and FSAR cross-references to specific program descriptions

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19.1 Probabilistic Risk Assessment (cont.)

FSAR (cont.)

- how PRA would be used during plant operations to support of other licensee programs such as the Maintenance Rule, interface with the ROP, reliability assurance program, human factors program, and severe accident management program, including FSAR cross-references to specific program descriptions
- specific risk-informed applications that have been implemented during the operational phase including FSAR cross-references to specific program descriptions such as risk-informed ISI, risk-informed IST, 10 CFR 50.69 implementation, and NFPA-806 implementation

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19.2 Severe Accident Evaluation

PSAR PSAR Section 19.2 contains a summary description of methodology and tools to be used to develop the severe accident evaluations for the Clinch River SMR. It also provides a description of the methodology and criteria for developing severe accident management design alternatives (SAMDA).

DCD DCD Section 19.2 describes the design features to prevent and mitigate severe accidents, in accordance with the requirements in 10 CFR 52.47(23) (similar to operating license application requirement 10 CFR 52.79(a)(38)). These features specifically address the issues identified in SECY-90-016 and SECY-93-087, which the Commission approved in related SRMs dated June 26, 1990, and July 21, 1993, respectively, for prevention (e.g., ATWS, mid-loop operation, SBO, fire protection, and intersystem LOCA) and mitigation (e.g., hydrogen generation and control, core debris coolability, high-pressure core melt ejection, containment performance, dedicated containment vent penetration, equipment survivability).

The design satisfies the requirements of 10 CFR 52.47(8) (similar to operating license application requirement 10 CFR 52.79(a)(17), which invokes 10 CFR 50.34(f)(1)(i) to specify that a design-specific PRA should be performed to seek improvements in core heat removal system reliability and containment heat removal system reliability that are significant and practical and do not excessively impact the plant. The following is included in this section:

- a description of the severe accident evaluation.
- a deterministic evaluation to show how the plant's severe accident preventive features would cope with the following events:
  - Anticipated Transients without Scram (ATWS)
  - Mid-Loop Operations
  - Station Blackout (SBO)
  - Fire Protection
  - Intersystem Loss of Coolant Accident (LOCA)
  - Description of other Severe Accident Preventive Features
- a description of severe accident mitigation features including:
  - an Overview of the Containment Design
  - a description of Severe Accident Progression, both In-and Ex-Vessel
  - a description of Severe Accident Mitigation Features for External Reactor Vessel Cooling, Hydrogen Generation and Control, Core Debris Coolability, High-Pressure Melt Ejection, Fuel-Coolant Interactions, Containment Bypass (including Steam Generator Tube Rupture and Intersystem LOCA), Equipment Survivability, and Other Severe Accident Mitigation Features
- a discussion of the containment performance goals identified in SECY-93-087 and SECY-90-016, as approved by the associated SRMs.

19.2 Severe Accident Evaluation (cont.)

DCD (cont.)

- a description of actions taken during the course of an accident by the plant operating and technical staff to:
  - prevent core damage
  - terminate the progress of core damage if it begins an d retain the core within the reactor vessel
  - maintain containment integrity as long as possible
  - minimize offsite releases
- a description of how the requirement of 10 CFR 50.34(f)(1)(I) has been met

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FSAR Same contents as the mPower standard plant DCD Section 19.2 with site-specific supplemental information as applicable, and a discussion of SAMA if needed.

19.3 Open and Confirmatory Items

PSAR PSAR Section 19.3 is not applicable.

DCD DCD Section 19.3 contains subsections as follows:

- resolution of Open Items
- resolution of Confirmatory Items
- identification of COL Action Items

FSAR Same contents as the mPower standard plant DCD Section 19.3.

19A Beyond Design Basis Aircraft Impact Assessment

PSAR PSAR Appendix 19A provides a description of the methodology that is used to assess the key design features of the Clinch River SMR Plant that are relied upon to withstand the effects of a large, commercial aircraft impact as required by 10 CFR 50.150, including the following preliminary information:

- summary description of regulatory criteria and guidance related to beyond design basis aircraft impact assessments
- description of the aircraft impact assessment scope and methodology based on guidance provided in NEI 07-13

DCD DCD Appendix 19A provides a description of the key design features and functional capabilities of the mPower standard plant design that are relied upon to withstand the effects of a large, commercial aircraft impact as required by 10 CFR 50.150, including the following information:

- summary description of regulatory criteria and guidance related to beyond design basis aircraft impact assessments
- summary description of key design features and functional capabilities used to ensure that the reactor core remains cooled or the containment remains intact, and spent fuel cooling or spent fuel pool integrity is maintained following an aircraft impact with reduced operator action
- description of the aircraft impact assessment scope and methodology based on guidance provided in NEI 07-13
- results and conclusions of aircraft impact assessment

FSAR Same contents as the mPower Standard Plant DCD Appendix 19A with supplemental, site-specific information, if necessary.

19B Regulatory Treatment of Non-Safety Systems

PSAR PSAR Appendix 19B provides a description of the methodology that is used to demonstrate that the design of the Clinch River SMR plant adequately addresses regulatory treatment of non-safety systems (RTNSS) issues. The PSAR includes the following preliminary information:

- summary description of criteria and guidance related to RTNSS issues provided in SECY-94-084
- a summary description of the RTNSS evaluation scope and methodology based on guidance provided in SECY-94-084

DCD DCD Appendix 19B provides a description of the methodology that is used to demonstrate that the mPower standard plant design adequately addresses RTNSS issues. The DCD includes the following information:

- description of the criteria and guidance related to RTNSS issues provided in SECY-94-084
- description of the process and methodology used to determine whether regulatory oversight is needed for certain nonsafety-related systems
- description of the process and methodology used to identify risk important SSCs for regulatory oversight (if regulatory oversight is determined to be needed)
- description of the process and methodology used to determine the appropriate level of regulatory oversight for the identified SSCs commensurate with their risk importance
- results and conclusions of the RTNSS evaluation

FSAR Same contents as the mPower Standard Plant DCD Appendix 19B with supplemental, site-specific information, if necessary.

19C Availability Controls Manual

PSAR PSAR Appendix 19C provides a preliminary Availability Controls Manual (ACM) for those structures, systems, and components (SSCs) classified as RTNSS based on the criteria provided in RG 1.206, Section C.IV.9. The format of the RTNSS ACM is similar to the plant-specific TS to provide the following categories of information on the availability of nonsafety-related SSCs:

- Limiting Conditions for Operation (LCOs) and associated remedial actions
- Surveillance Requirements
- Design Features
- Administrative Controls, including requirements on effluents containing radioactive material

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DCD DCD Appendix 19C provides a generic ACM for the mPower standard plant SSCs classified as RTNSS based on the criteria provided in RG 1.206, Section C.IV.9. The format of the RTNSS ACM is similar to the generic TS to provide the following categories of information on the availability of nonsafety-related SSCs:

- Limiting Conditions for Operation (LCOs) and associated remedial actions
- Surveillance Requirements
- Design Features
- Administrative Controls, including requirements on effluents containing radioactive material

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FSAR Same contents as the mPower standard plant DCD Appendix 19C with updated site-specific and standard supplemental information, as needed

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