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

August 24, 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 FIRST REGULATORY FRAMEWORK WORKSHOP**

References:
1) TVA letter to NRC dated November 5, 2010, "Key Assumptions Letter for the Possible Licensing and Construction of Small Modular Reactor Modules at the Clinch River Site"
2) TVA letter to NRC dated December 22, 2010, "Addendum to the Key Assumptions Letter for the Possible Licensing and Construction of Small Modular Reactor Modules at the Clinch River Site"

In References 1 and 2, TVA communicated to the NRC its Key Licensing Assumptions related to evaluating the feasibility of deploying small modular reactor (SMR) modules at the proposed Clinch River site utilizing the Babcock & Wilcox (B&W) mPower design as the technology of choice. These Key Assumptions identified that TVA would utilize its established Regulatory Framework process to assure that the regulations applicable to a 10CFR Part 50 Construction Permit Application are satisfactorily addressed, and to define the appropriate level of detail to be provided. 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 a future public meeting.

Material to be presented at the first Regulatory Framework Workshop is provided in the following attachments:

Attachment 1: Regulatory Framework Workshop Presentation slides
Attachment 2: PSAR Section 9.1 Regulatory Framework Document (RFD) and Section Outline
Attachment 3: PSAR Subsection 9.1.2 Illustrative Mock-up
Attachment 4: PSAR Section 5.3 RFD and Section Outline
Attachment 5: PSAR Subsection 2.4.12 RFD and Section Outline

D104
NRO

We look forward to scheduling this first Regulatory Framework Workshop. Please contact Thomas Spink at (423)-751-7062 if you have questions.

Sincerely,

Andrea L. Sterdis
Senior Manager, Strategic nuclear Expansion
Nuclear Generation Development and Construction

Attachments
cc: See Page 3

cc: w/Attachments

R.W. Borchardt
U.S. Nuclear Regulatory Commission
One White Flint North, 16E15
11555 Rockville Pike
Rockville, Maryland 20852-2378

Michael Johnson
U. S. Nuclear Regulatory Commission
Two White Flint North, 6F15
11545 Rockville Pike
Rockville, Maryland 20852-2738

Michael Mayfield
U.S. Nuclear Regulatory Commission
Two White Flint North, 6E4
11545 Rockville Pike
Rockville, Maryland 20852-2738

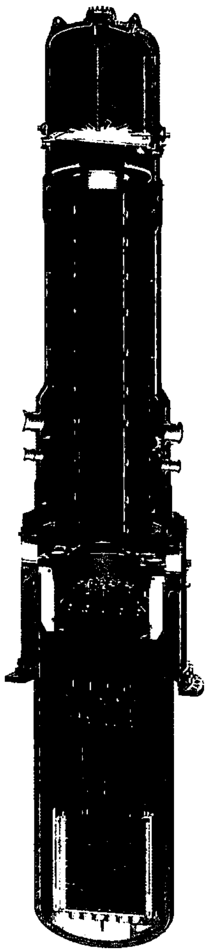
Victor McCree
U.S. NRC Region II
Marquis One Tower
245 Peachtree Center Avenue N.E., Suite 1200
Atlanta, GA 30303

# Attachment 1

Regulatory Framework Workshop Presentation Slides

TVA

generation mPower





# *Clinch River Construction Permit Application Development*

First Regulatory Framework Workshop
September XX, 2011

# Agenda

- Introduction
- Workshop Objectives
- Background
- Regulatory Framework Process
- Specific Workshop Reviews
  - PSAR Section 9.1 RFD – Fuel Storage and Handling
  - PSAR Section 9.1.2 Mockup – New and Spent Fuel Storage
  - PSAR Section 2.4.12 - Groundwater
  - PSAR Section 5.3 – Reactor Vessel
- Conclusion

# Workshop Objectives

- Present initial Clinch River Licensing Baseline
  - Regulatory Framework Documents
  - Section Outlines
- Develop understanding of NRC CPA level of detail needs
- Engage NRC Staff in discussions on RFD/Section Outline content
- Obtain NRC agreement on Identified Issues
- Identify areas for future interaction

**Goal: NRC Acceptance of Licensing Baseline for CPA**

# Key Assumptions

- 10 CFR Part 50 licensing process
- PSAR level of detail consistent with RG 1.70, Revision 3 and organizational structure of Standard Review Plan
  - Utilize the Regulatory Framework process
  - Address 10 CFR Part 52 requirements, as applicable
  - Develop Environmental Report consistent NUREG 1555
  - Evaluate SRP revision in effect 6 months prior to CPA submittal
- One Design – One Review
- NRC would inspect B&W as a vendor
- Initial test program will inform future ITAAC

**CP and OL will address applicable Regulations**

# TVA Chooses 10 CFR Part 50 for First-of-a-Kind (FOAK) SMR Project

- Experience with licensing process - know how it will work throughout
- Less cost and potentially less time to get to point where you can construct – CP issuance
- Modifications during construction easier to accommodate – useful for first-of-a-kind
- Testing and verification of design established later versus defining completely upfront
- Regulator has opportunity to evaluate as-built plant prior to operating license issuance

10CFR52 still appropriate and preferred for standardized deployment after FOAK

# TVA's Regulatory Framework Approach

- TVA Regulatory Framework History
  - Browns Ferry Unit 1
  - Watts Bar Unit 2
  - Bellefonte Units 1 and 2
- Establish clear understanding of licensing basis applicable to a Construction Permit determination finding to address the following:
  - Current regulations
  - Current regulatory guidance
  - NRC generic communications and unresolved safety issues
- For Clinch River, the Regulatory Framework will be a living database system to guide both development and implementation of the licensing basis

# Establishing Regulatory Alignment Consistent with NRC Requirements and Guidance

generation mPower

## Key Assumptions

- Use of Part 50
- RG 1.70/SRP
- Timing of SAMDA
- One design/one review

**NRC Response Received**

## Regulatory Framework

- Regulatory Assessment
- CP Application Level of Detail
- OL/Design Certification Link
- Detailed at Section Level

**NRC Interaction**

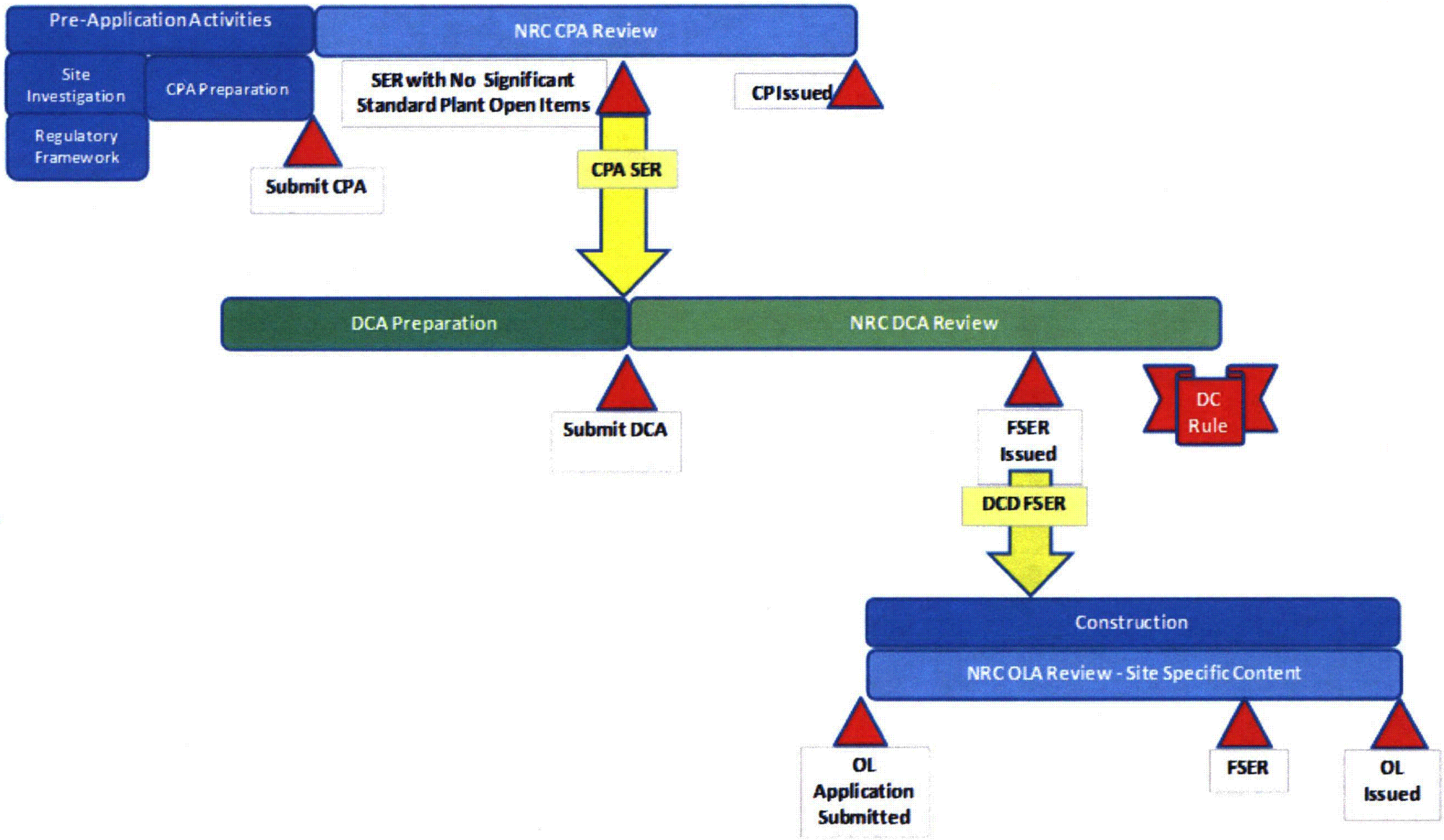
## Construction Permit Application

- PSAR
- ER
- Emergency Plan
- Security Plan
- General and Administrative

# Regulatory Framework Purpose

- Current Regulations and Regulatory Guidance will be addressed
  - CP PSAR
  - DCD (Standard Plant Design)
  - OL FSAR
- Establish Licensing Baseline for Construction Permit
  - Regulations
  - Regulatory Guidance
  - Generic Communications
  - Level of Detail
- Illustrate alignment between CPA, B&W NE mPower™ Design Certification Application (DCA), and the OLA
- Provide input to CPA development schedule

# CPA – DCA – OLA Alignment



# B&W NE mPower™ High Level Requirements

generation mPower

- 150 MWE Nominal Output per Module
- 60-year plant design life
- NSSS Forging Diameter Allows Domestic Forgings and Unrestricted Rail Shipment
- Passive Safety Requirements – Emergency (Diesel) Power Not Required
  - Minimize Primary Coolant Penetrations, Maximize Elevation of Penetrations
  - Large Reactor Coolant Inventory
  - Low Core Power Density
- Standard Fuel (less than 5% $U^{235}$)
- Long Fuel Cycle, 4 Year Core Life

# B&W NE mPower™
# High Level Requirements

generation mPower

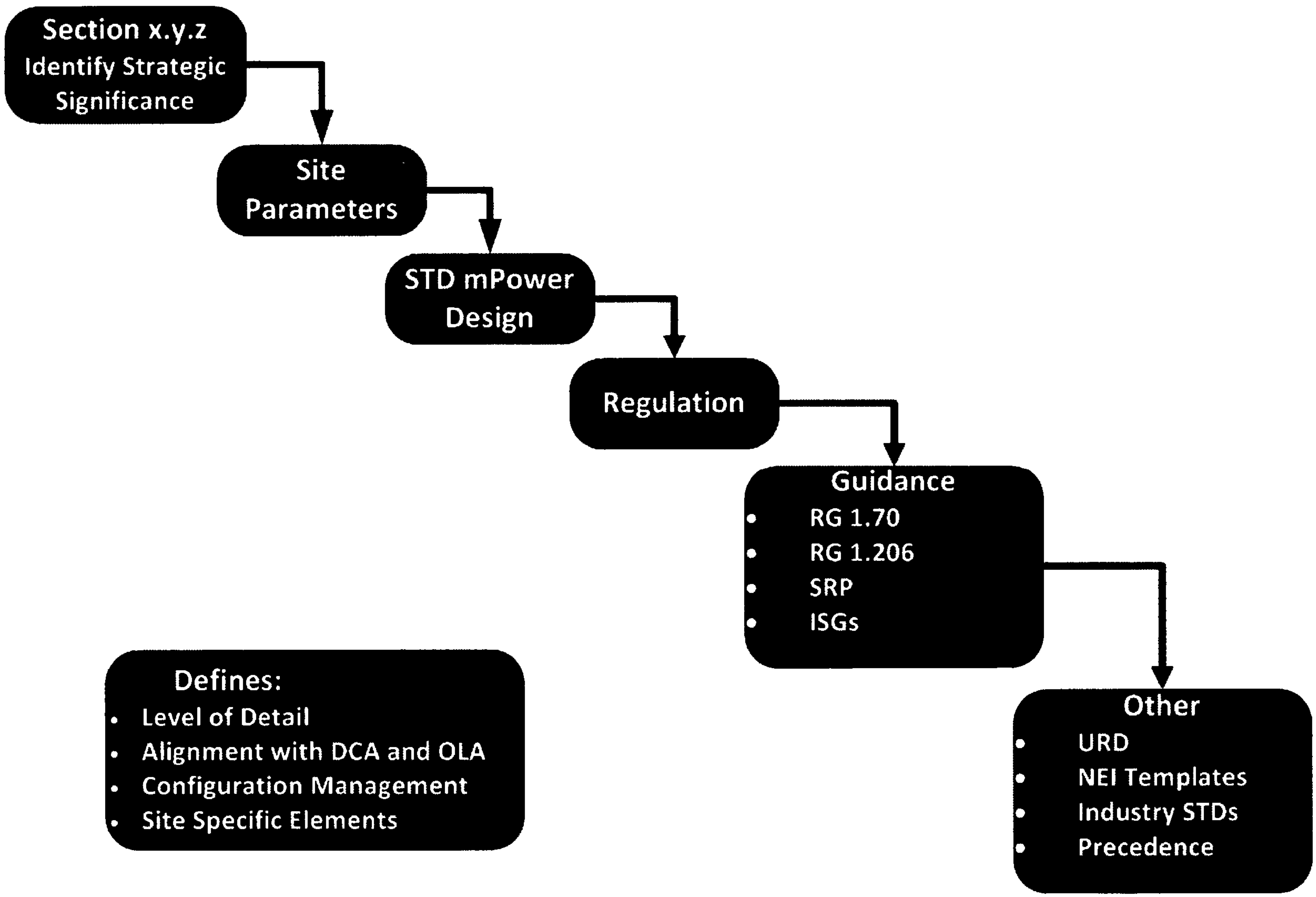
- No Soluble Boron in Primary System for Normal Reactivity Control
- Conventional/Off-the-Shelf Balance of Plant Systems and Components
- Accommodate Air-Cooled and Water-Cooled Condensers
- Flexible Grid Interface (50 Hz or 60 Hz)
- Digital Instrumentation and Controls Compliant with NRC Regulations

# Regulatory Framework Process Description

generation mPower

- Production
- Integration
- Standard Plant – Site Specific
- Review

# Regulatory Framework Process



# Regulatory Framework Process

generation mPower

| Submittal Document | Regulatory Requirements | Proposed Exemptions | Regulatory Basis for Section Content | NUREG-0800 (SRP) Section | Regulatory Guidance |
|---|---|---|---|---|---|
| PSAR | | | | | |
| DCD | | | | | |
| FSAR | | | | | |

# Regulatory Framework Process

generation mPower

| Submittal Document | Industry Guidance | CPA Information Beyond RG 1.70 | Changes to the Standard Plant Design | Related Sections |
|---|---|---|---|---|
| PSAR | | | | |
| DCD | | | | |
| FSAR | | | | |

# Section Outline

- PSAR
  - Summary description of PSAR content
- DCD
  - Summary description of mPower Standard Plant DCD
- FSAR
  - Summary description of FSAR content

# General CP Guidance

generation mPower

- 10CFR50.35(a) Issuance of CP

(1) The proposed facility design will be described, including
  - Principal architectural and engineering design criteria, and
  - Major features or components incorporated for protection of public health and safety

(2) Further technical or design information as may be required to complete the safety analysis, which can reasonably be left for later consideration, will be provided in FSAR.

(3) A description of the research and development program to be conducted to resolve any safety questions for safety features and components requiring such a program.

# General CP Guidance

- Standard Review Plan Subsection III, Review Procedures

  The procedures in Subsection III of the SRP are used during the CP review to confirm that the design criteria and bases and the preliminary design as specified in the PSAR meet the acceptance criteria given in Subsection II of the SRP.

# Regulatory Framework

- Chapter 1 Strategy
  - Rollup of other Chapters/Sections
  - General Arrangement Drawings
    - Fire protection Zones
    - Radiation Zones
  - Regulatory Guide conformance
- Proprietary Information
  - All proprietary Information will be included in a separate part of the application
  - Includes: Business sensitive, security related (SUNSI), and business proprietary

# Regulatory Framework Schedule

- Proposed Schedule for Future Workshops
  - October 13
  - November 15

- Other planned meetings
  - Subsurface Investigation Plans
  - Low Level Radioactive Waste
  - Post Fukushima

# Regulatory Framework Documents

generation mPower

- Section 9.1 – Fuel Storage and Handling
  - Regulatory Framework Matrix
  - Section Outline
  - Section 9.1.2, “New and Spent Fuel Storage” Mockup

# Regulatory Framework Documents

generation mPower

- Section 2.4.12 - Groundwater
  - Regulatory Framework Matrix
  - Section Outline

# Regulatory Framework Documents

generation *mPower*

- Section 5.3 – Reactor Vessel
  - Regulatory Framework Matrix
  - Section Outline

# Conclusion

- Conclusion

- Questions?

# Attachment 2

PSAR Section 9.1 Regulatory Framework Document (RFD) and Section Outline

**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 |
|---|---|---|---|---|---|---|---|---|---|---|
| 9.1.1<br><br>Criticality Safety of Fresh and Spent Fuel Storage and Handling | PSAR | 10 CFR 50, App. A, GDC 62<br>10 CFR 50.68 | No | RG 1.70<br><br>Note: Section 9.1.1 of RG 1.70 only addresses New Fuel Storage; however, consistent with RG 1.206 format, PSAR Section 9.1.1 will address both new and spent fuel storage criticality analysis, with the following exceptions:<br><br>Information on governing codes for design, ability to withstand external loads and forces, and safety implications related to sharing (for multiunit facilities) will be provided in PSAR Section 9.1.2. | 9.1.1 | RG 1.13 | ANSI/ANS 57.1, 1992 (R2005)<br>ANSI/ANS 57.2, 1983<br>ANSI/ANS 57.3, 1983<br>ANSI/ANS 8.1, 1998 (R2007)<br>ANSI/ANS 8.24, 2007<br>ANSI/ANS 8.17, 2004 | Yes<br><br>Also need to address criticality accident requirements per 10 CFR 50.68 (1998) | No | 3.2.1, 3.8.4 |
| | DCD | 10 CFR 50, App. A, GDC 62<br>10 CFR 50.68<br>10 CFR 52.47(b)(1) | No | RG 1.206 | 9.1.1 | RG 1.13 | ANSI/ANS 57.1, 1992 (R2005)<br>ANSI/ANS 57.2, 1983<br>ANSI/ANS 57.3, 1983<br>ANSI/ANS 8.1, 1998 (R2007)<br>ANSI/ANS 8.24, 2007<br>ANSI/ANS 8.17, 2004 | N/A | N/A | 3.2.1, 3.8.4 |
| | FSAR | 10 CFR 50, App. A, GDC 62<br>10 CFR 50.68 | No | RG 1.206 - Same contents as DCD | 9.1.1 | RG 1.13 | ANSI/ANS 57.1, 1992 (R2005)<br>ANSI/ANS 57.2, 1983<br>ANSI/ANS 57.3, 1983<br>ANSI/ANS 8.1, 1998 (R2007)<br>ANSI/ANS 8.24, 2007<br>ANSI/ANS 8.17, 2004 | N/A | No | 3.2.1, 3.8.4 |
| 9.1.2<br><br>New and Spent Fuel Storage | PSAR | 10 CFR 50, App. A, GDC 2, 4, 5, 61, 63<br>10 CFR 20.1101(b)<br>10 CFR 50.68 | No | RG 1.70<br><br>Note: Section 9.1.2 of RG 1.70 only addresses Spent Fuel Storage; however, consistent with RG 1.206 format, PSAR Section 9.1.2 will address both new and spent fuel storage and include information on governing codes for design, ability to withstand external loads and forces, and safety implications related to sharing (for multiunit facilities) | 9.1.2, 3.8.4 App. D | RGs 1.13, 1.29, 1.52, 1.115, 1.117, 1.140, 8.8 NUREG/CR-5912 | ANSI/ANS 57.2, 1983<br>ANSI/ANS 57.3, 1983 | Yes<br><br>Note that criticality accident requirements will be addressed in Section 9.1.1 | No | 3.2, 3.3, 3.4, 3.5, 6.6, 9.1.1, 12.3, 16 |
| | DCD | 10 CFR 50, App. A, GDC 2, 4, 5, 61, 63<br>10 CFR 20.1101(b)<br>10 CFR 50.68<br>10 CFR 52.47(b)(1) | No | RG 1.206 | 9.1.2, 3.8.4 App. D | RGs 1.13, 1.29, 1.52, 1.115, 1.117, 1.140, 8.8 NUREG/CR-5912 | ANSI/ANS 57.2, 1983<br>ANSI/ANS 57.3, 1983 | N/A | N/A | 3.2, 3.3, 3.4, 3.5, 6.6, 9.1.1, 12.3, 16 |

**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 |
|---|---|---|---|---|---|---|---|---|---|---|
| | FSAR | 10 CFR 50, App. A, GDC 2, 4, 5, 61, 63<br>10 CFR 20.1101(b)<br>10 CFR 50.68 | No | RG 1.206 - Same contents as DCD | 9.1.2, 3.8.4 App. D | RGs 1.13, 1.29, 1.52, 1.115, 1.117, 1.140, 8.8<br>NUREG/CR-5912 | ANSI/ANS 57.2, 1983<br>ANSI/ANS 57.3, 1983 | N/A | No | 3.2, 3.3, 3.4, 3.5, 6.6, 9.1.1, 12.3, 16 |

**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 |
|---|---|---|---|---|---|---|---|---|---|---|
| 9.1.3<br><br>Spent Fuel Pool Cooling and Cleanup System | PSAR | 10 CFR 50, App. A, GDC 2, 4, 5, 61, 63<br>10 CFR 20.1101(b)<br>10 CFR 50.68 | No | RG 1.70 | 9.1.3 | RGs 1.13, 1.26, 1.29, 1.52, 8.8 | ANSI/ANS 57.2, 1983 | No | No | 3.3, 3.4, 3.5, 3.6, 3.7, 3.8, 3.9, 6.6, 11.1, 11.2, 16 |
| | DCD | 10 CFR 50, App. A, GDC 2, 4, 5, 61, 63, 10 CFR 20.1101(b), 10 CFR 50.68, 10 CFR 52.47(b)(1) | No | RG 1.206 | 9.1.3 | RGs 1.13, 1.26, 1.29, 1.52, 8.8 | ANSI/ANS 57.2, 1983 | N/A | N/A | 3.3, 3.4, 3.5, 3.6, 3.7, 3.8, 3.9, 6.6, 11.1, 11.2, 16 |
| | FSAR | 10 CFR 50, App. A, GDC 2, 4, 5, 61, 63, 10 CFR 20.1101(b), 10 CFR 50.68 | No | RG 1.206 - Same contents as DCD | 9.1.3 | RGs 1.13, 1.26, 1.29, 1.52, 8.8 | ANSI/ANS 57.2, 1983 | N/A | No | 3.3, 3.4, 3.5, 3.6, 3.7, 3.8, 3.9, 6.6, 11.1, 11.2, 16 |
| 9.1.4<br><br>Light Load Handling System (Related to Refueling) | PSAR | 10 CFR 50, App. A, GDC 2, 5, 61, 62 | No | RG 1.70<br><br>Note: Section 9.1.4 of RG 1.70 addresses Fuel Handling Systems, both overhead and light load handling systems. Consistent with RG 1.206 format, PSAR Section 9.1.4 will only address light load handling system design aspects, while the overhead heavy load handling system design aspects will be addressed in PSAR Section 9.1.5. | 9.1.4 | RG 1.29 | ANSI/ANS 57.1, 1992 (R2005) | No | No | 3.2, 3.7, 3.8, 3.9 |
| | DCD | 10 CFR 50, App. A, GDC 2, 5, 61, 62<br>10 CFR 52.47(b)(1) | No | RG 1.206 | 9.1.4 | RG 1.29 | ANSI/ANS 57.1, 1992 (R2005) | N/A | N/A | 3.2, 3.7, 3.8, 3.9 |
| | FSAR | 10 CFR 50, App. A, GDC 2, 5, 61, 62 | No | RG 1.206 - Same contents as DCD | 9.1.4 | RG 1.29 | ANSI/ANS 57.1, 1992 (R2005) | N/A | No | 3.2, 3.7, 3.8, 3.9 |

**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 |
|---|---|---|---|---|---|---|---|---|---|---|
| 9.1.5<br><br>Overhead Heavy Load Handling System | PSAR | 10 CFR 50, App. A, GDC 1, 2, 4, 5 | No | RG 1.70<br><br>Note: Section 9.1.5 is not included in RG 1.70. However, consistent with RG 1.206 format, PSAR Section 9.1.5 will address overhead heavy load handling systems while the light load handling systems will be addressed in PSAR Section 9.1.4. | 9.1.5 | RGs 1.13, 1.29, NUREG-0554 NUREG-0612 RIS 2005-25, Suppl. 1 | ANSI/ANS 57.1, 1992 (R2005)<br>ANSI N14.6, 1993<br>ASME B30.2, 2005<br>ASME B30.9, 2010<br>ASME NOG-1, 2010 | No | No | 3.2, 3.7, 3.8, 3.9 |
| | DCD | 10 CFR 50, App. A, GDC 1, 2, 4, 5<br>10 CFR 52.47(b)(1) | No | RG 1.206 | 9.1.5 | RGs 1.13, 1.29 NUREG-0554 NUREG-0612 RIS 2005-25, Suppl. 1 | ANSI/ANS 57.1, 1992 (R2005)<br>ANSI N14.6, 1993<br>ASME B30.9, 2010<br>ASME NOG-1, 2010<br>ASME B30.2, 2005 | N/A | N/A | 3.2, 3.7, 3.8, 3.9 |
| | FSAR | 10 CFR 50, App. A, GDC 1, 2, 4, 5 | No | RG 1.206 - Same contents as DCD | 9.1.5 | RGs 1.13, 1.29, NUREG-0554, NUREG-0612 RIS 2005-25, Suppl. 1 | ANSI/ANS 57.1, 1992 (R2005)<br>ANSI N14.6, 1993<br>ASME B30.9, 2010<br>ASME NOG-1, 2010<br>ASME B30.2, 2005 | N/A | No | 3.2, 3.7, 3.8, 3.9 |

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

# Section 9.1 Outline

## 9.1 Fuel Storage and Handling

### 9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling

**PSAR** New and spent fuel storage facilities are designed to maintain the required degree of subcriticality. Criticality is precluded by adequate design of fuel handling and storage facilities and by administrative control features. Criticality safety is demonstrated based on the following information:

- description and preliminary design bases of the new and spent fuel storage facilities
- proposed approach to addressing criticality accident requirements under 10 CFR 50.68(b)
- quantity of fuel to be stored
- description of proposed storage racks
- proposed approach for maintaining a subcritical array
- assumptions and proposed methodology for the criticality safety analysis which will conform to applicable ANSI standards
- figure(s) providing location of new and spent fuel storage facilities

---

**DCD** The new and spent fuel storage facilities for the mPower standard plant are designed to prevent a criticality accident by physical systems or processes using geometrically safe configurations. Criticality safety is demonstrated based on the following information:

- final description, design bases and safety evaluation of the new and spent fuel storage facilities
- approach to addressing criticality accident requirements under 10 CFR 50.68(b)
- quantity of fuel to be stored
- approach for maintaining a subcritical array
- degree of subcriticality provided for all normal and credible abnormal conditions
- design data for the fuel and fuel storage racks
- methods, approximations, assumptions, and design tolerances used in the criticality safety analysis conforming to applicable ANSI standards based on use of **[---]** computer code and **[---]** validation methodology
- reference to technical report(s) for the new and spent fuel rack criticality safety analysis
- figure(s) showing general arrangements and location of new and spent fuel storage facilities

---

**FSAR** Same contents as mPower standard plant DCD Section 9.1.1.

---

# Section 9.1 Outline

## 9.1.2 New and Spent Fuel Storage

**PSAR** The new and spent fuel storage facilities are designed to maintain the fuel assemblies in a safe and subcritical array during all credible storage conditions and are designed to provide a safe means of loading the spent fuel assemblies into shipping or storage casks. The description and preliminary design bases for the new and spent fuel storage facilities are provided in PSAR Section 9.1.1. Specific aspects of the new and spent fuel storage facilities are described below:

- governing codes for the design of the new and spent fuel storage facilities, including design aspects for protection from natural phenomena, internal and external missiles
- quantity of new and spent fuel to be stored
- fuel rack seismic classification
- proposed configuration of the storage facilities
- description of proposed new and spent fuel storage racks
- proposed approach for new and spent fuel rack structural dynamic and stress analysis conforming to RG 1.124 for components designed by the linear elastic method
- proposed measures to prevent drainage of spent fuel storage areas
- proposed measures to prevent flooding of dry fuel storage areas
- proposed approach for natural coolant circulation through the spent fuel storage racks
- proposed features for radiological shielding and spent fuel pool monitoring
- proposed provisions to preclude heavy load drops on spent fuel
- description of material compatibility aspects
- proposed approach for thermal-hydraulic analysis
- safety implications related to sharing fuel handling area systems/components for two mPower modules (as applicable)
- reference to PSAR Section 12.3 for proposed provisions for radiation monitoring
- reference to PSAR Section 9.4 for description of fuel handling area ventilation system
- identification of required inspection(s) applicable to the spent fuel pool liner and spent fuel storage racks
- preliminary table(s) showing loads and load combinations
- figure(s) showing facility design, including fuel handling area layout and new and spent fuel rack layout

---

**DCD** The new and spent fuel storage facilities for the mPower standard plant are designed to maintain the fuel assemblies in a safe and subcritical array during all credible storage conditions and are designed to provide a safe means of loading the spent fuel assemblies into shipping or storage casks. The description and design bases for the new and spent fuel storage facilities are provided in DCD Section 9.1.1. Specific aspects of the new and spent fuel storage facilities are described below:

## Section 9.1 Outline

### 9.1.2 New and Spent Fuel Storage (cont.)

**DCD (cont.)**

- governing codes for the design of the new and spent fuel storage facilities, including design aspects for protection from natural phenomena, internal and external missiles
- quantity of new and spent fuel to be stored
- fuel rack seismic classification
- configuration of the storage facilities
- new and spent fuel storage rack design, including ability to withstand accident forces associated with fuel handling
- fuel assembly drop accident analysis (as required)
- measures to prevent drainage of spent fuel storage areas
- measures to prevent flooding of dry fuel storage areas
- effectiveness of natural coolant circulation through the spent fuel storage racks
- radiological shielding design considerations
- spent fuel pool monitoring requirements
- provisions to preclude heavy load drops on spent fuel
- material compatibility requirements
- spent fuel pool liner leak collection and control features
- provisions for inspection and testing spent fuel pool liner and spent fuel racks
- reference to technical report(s) on new and spent fuel rack structural dynamic and stress analysis
- reference to technical report(s) on spent fuel rack thermal-hydraulic analysis
- safety implications related to sharing fuel handling area systems/components for two mPower modules (as applicable)
- description of spent fuel rack coupon surveillance program
- reference to DCD Section 12.3 for radiation monitoring systems
- reference to DCD Section 9.4 for description of fuel handling area ventilation system
- table(s) showing storage rack design loads and load combinations
- figure(s) showing fuel storage rack layouts and spent fuel pool (SFP) layout

---

**FSAR** Same contents as mPower standard plant DCD Section 9.1.2 with the following supplemental information:

- inspection procedures for spent fuel pool liner and spent fuel racks

---

# Section 9.1 Outline

## 9.1.3 Spent Fuel Pool Cooling and Cleanup System

**PSAR** The spent fuel pool (SFP) cooling and cleanup system provides decay heat removal capabilities and maintains purity and clarity of the water in the SFP and spent fuel cask loading pit (when in service). Specific aspects of the preliminary design of the spent fuel cooling and cleanup system are described below:

- proposed system description, including provisions for continuous or intermittent cooling and cleanup of fission and corrosion products
- quantity of fuel to be cooled
- SFP makeup water requirements
- radiation shielding requirements
- description of proposed system instrumentation, including monitoring of water level, temperature, and radiation levels
- safety evaluation addressing capability for SFP cooling during all operating conditions, provisions to ensure water will not be lost at a rate greater than makeup capability, and keeping the fuel covered during all storage conditions
- preliminary system inspection and testing requirements
- table(s) listing the principal parameters of the SFP cooling and cleanup system
- preliminary figures showing fuel pool cooling and cleanup schematic features

---

**DCD** The spent fuel pool (SFP) cooling and cleanup system provides decay heat removal capabilities and maintains purity and clarity of the water in the SFP and spent fuel cask loading pit (when in service). Specific aspects of the SFP cooling and cleanup system are described below:

- detailed system description, including SFP cooling functions, heat generation rate of stored fuel, and heat removal design bases
- pool cleanliness requirements during normal operations
- SFP cooling purification capability for fuel handling and to demonstrate ALARA compliance
- safety evaluation describing capability to withstand design loads and forces, protection of essential components from natural phenomena effects, and provisions to collect system leakage
- provisions to collect system leakage and preclude inadvertent or accidental draining of SFP
- provisions to maintain tritium concentration in the spent fuel pool water
- system instrumentation requirements, including monitoring of water level, temperature, and radiation levels
- system inspection and testing requirements
- failure modes and effects analyses (FMEA) of SFP cooling and cleanup system, including makeup system
- table(s) providing design parameters and component data for the SFP cooling and cleanup system

---

## Section 9.1 Outline

### 9.1.3 Spent Fuel Pool Cooling and Cleanup System (cont.)

| | |
|---|---|
| **DCD (cont.)** | - figure(s) of SFP cooling and cleanup system piping design layout and instrumentation diagram |
| **FSAR** | Same contents as mPower standard plant DCD Section 9.1.3. |

# Section 9.1 Outline

## 9.1.4 Light Load Handling System (Related to Refueling)

**PSAR** The light load handling systems are designed for the handling and storage of new and spent fuel assemblies. Specific aspects of the light load handling systems are described below:

- preliminary description and design bases of the light load handling systems, including design codes and standards, compliance with applicable General Design Criteria and regulatory guidance, and seismic design aspects
- performance and load handling requirements, including features to prevent fuel handling accidents
- preliminary outline of procedures for new fuel receipt and storage, reactor refueling operations, and spent fuel storage
- proposed inspection, testing, and instrumentation requirements

---

**DCD** The mPower standard plant light load handling systems are designed for the handling and storage of new and spent fuel assemblies. Specific aspects of the light load handling system are described below:

- final description and design bases of the light load handling systems, including design codes and standards, seismic design, and compliance with applicable General Design Criteria and regulatory guidance
- performance and load handling requirements, load handling control features, including limit and safety devices, and provisions to prevent fuel handling accidents
- procedures for new fuel receipt and storage, reactor refueling operations, and spent fuel storage
- inspection, testing, and instrumentation requirements
- figure(s) of system components and associated tools
- a FMEA analysis to demonstrate individual subsystems and components, including controls and interlocks, are designed to meet the single-failure criterion

---

**FSAR** Same contents as mPower standard plant DCD Section 9.1.4 with the following supplemental information:

- inservice inspection program for the light load handling system

---

# Section 9.1 Outline

## 9.1.5 Overhead Heavy-Load Handling System

**PSAR** The overhead heavy load handling systems consist of several devices used for critical load handling evolutions, including removal of the upper reactor vessel. Specific aspects of the heavy load handling systems are described below:

- description and design bases of the overhead heavy load handling systems, including all principal components
- performance and load handling requirements, including features to prevent load drop accidents
- outline of heavy load handling procedures
- inspection, testing, and instrumentation requirements

---

**DCD** The mPower standard plant overhead heavy load handling systems consist of several devices used for critical load handling evolutions, including removal of the upper reactor vessel. Specific aspects of the heavy load handling systems are described below:

- detailed description and design bases of the overhead heavy load handling systems, including physical arrangements, loads to be handled, and design features or interlocks to restrict movement to areas away from stored fuel and equipment necessary for safe shutdown of the reactor
- principal component descriptions, including relevant design data, seismic category and quality class, and design codes and standards used for design, manufacture, testing, operation, and maintenance
- parameters defining the load that, if dropped, would cause greatest damage
- a safety evaluation in accordance with Section 5.1 of NUREG-0612 describing design features to preclude load drops, analyses of potential load drops, and general load handling practices
- inspection, testing, and instrumentation requirements
- table listing of the Nuclear Island heavy load handling systems
- figure(s) of the overhead handling system components, building layouts, and illustrations of special lifting devices
- a FMEA analysis to demonstrate individual subsystems and components, including controls and interlocks, are designed to meet the single-failure criterion

---

**FSAR** Same contents as mPower standard plant DCD Section 9.1.5 with the following supplemental information:

- inspection program for the overhead heavy load handling systems
- heavy load handling program, including associated procedural and administrative controls and applicable guidelines provided in RIS 2005-25, Supplement 1

---

# Attachment 3

PSAR Subsection 9.1.2 illustrative Mock-up

### 9.1.2 New and Spent Fuel Storage

#### 9.1.2.1 Design Bases

The new and spent fuel storage facilities are designed to maintain the fuel assemblies in a safe and subcritical array during all credible storage conditions. As described in Subsection 9.1.1, both facilities are located underground in the fuel handling area that is shared between two mPower reactor modules. The descriptive information provided in this section related to the fuel storage facilities, such as number of fuel assemblies, size of the spent fuel pool, etc. is representative of a single mPower reactor module.

The fuel handling area resides within the Reactor Service Building (RSB), which is a Seismic Category I structure designed to withstand the effects of natural phenomena without the loss of safety function, including earthquakes (Section 3.7), extreme winds and tornados (Section 3.3), external missiles (Section 3.5), and flooding conditions (Section 3.4). In addition, the storage facilities remain functional following other postulated hazards, such as fires, internal missiles, or pipe break events.

The quality group, seismic, and ASME Code classifications for the Clinch River fuel storage facilities, equipment, and systems are discussed in Section 3.2. The new and spent fuel racks are designed to meet Seismic Category I requirements. Criteria for the evaluation of the fuel rack stress analysis are provided in RG 1.124 and ASME Code Section III, Division I, Article NF-2000. The new and spent fuel rack design includes provisions to ensure that there is no increase in $k_{eff}$ beyond the safe limits specified in 10 CFR 50.68 under various operating and accident conditions. Further details regarding the approach and assumptions for the fuel rack criticality analysis are provided in Subsection 9.1.1.

##### 9.1.2.1.1 New Fuel Storage

The systems and equipment associated with the storage of new fuel are designed in accordance with the guidelines provided in ANSI/ANS-57.3 (Reference 9.1.2-1). The new fuel is stored in low density racks installed in a dry new fuel storage pit. The new fuel storage racks contain storage locations for **[---]** fuel assemblies, which corresponds to approximately a full core offload plus an additional **[---]** locations. The refueling cycle for each mPower reactor module is approximately 48 months.

##### 9.1.2.1.2 Spent Fuel Storage

The systems and equipment associated with the storage of spent fuel are designed in accordance with the guidelines of ANSI/ANS-57.2 (Reference 9.1.2-2) with the additions, clarifications, and exceptions provided in RG 1.13.

Spent fuel is stored in high density racks installed in the spent fuel pool (SFP) that can hold **[---]** fuel assemblies corresponding to the amount of spent fuel from approximately **[---]** years of operation at full power. The racks are designed to withstand the loading conditions provided in Table 9.1.2-1 based on NUREG-0800, SRP Section 3.8.4, Appendix D.

The spent fuel pool cooling and cleanup system (SFPCCS), described in Subsection 9.1.3, maintains the SFP water temperature below 120°F and removes impurities from the water. This system maintains the water temperature and water level within prescribed limits by removing decay heat generated by the stored spent fuel assemblies during normal conditions. In a loss of cooling scenario, SFP makeup is accomplished with a redundant, safety-related water makeup system which maintains water level in the pool. The makeup system has adequate capacity to replace evaporation and boiling losses from the SFP.

### 9.1.2.2 Facilities Description

The new and spent fuel storage facilities are located in the fuel handling area of the RSB, which is designed to Seismic Category I requirements. A representative layout of the fuel handling area is provided in Figure 9.1.2-1. The fuel handling area is shared between two mPower reactor modules as described in Section 1.2. The shared SSCs between the two reactor modules are:

- Light load handling equipment
- Overhead heavy load handling equipment
- Fuel handling area HVAC
- SFP cask loading pit
- Fire protection equipment

The following SSCs are not shared between the two mPower reactor modules:

- New fuel pit
- New fuel elevator
- Fuel transfer pit
- Spent fuel pool
- Spent fuel pool cooling and cleanup system (SFPCCS)
- Spent fuel pool makeup system
- Spent fuel pool drains
- Spent fuel pool leakage detection system

#### 9.1.2.2.1 New Fuel Storage

The new fuel is stored in low density racks which are designed for storage of the fuel with the maximum design basis enrichment. The racks in the new fuel pit consist of an array of cells interconnected to each other and to a thick base plate at the bottom elevation. New fuel racks are stored in a dry, unlined, reinforced concrete new fuel storage pit that is approximately **[---]** feet deep. The racks are supported by the pit floor and the storage pit is provided with a drain system, which is connected to the RSB sump to prevent the new fuel pit from being accidentally flooded.

The new fuel storage rack includes storage locations for **[---]** fuel assemblies with center-to-center spacing between adjacent fuel assemblies of **[---]** inches. This spacing provides a minimum separation between adjacent fuel assemblies which is sufficient to maintain a subcritical array as discussed in Section 9.1.1. The layout of the new fuel storage rack array is designed such that even if the pit becomes flooded with non-borated water, the fuel remains subcritical ($k_{eff}$ < 0.95). A typical storage rack layout is shown in Figure 9.1.2-2. These representative new fuel storage racks provide support for the fuel assemblies with ample clearance and lead-in to facilitate insertion of the new fuel assemblies. The new fuel assemblies are handled using a single-failure-proof new fuel gantry crane equipped with the new fuel handling tool, as described further in Subsection 9.1.4.

#### 9.1.2.2.2 Spent Fuel Storage

The spent fuel assemblies are stored in the spent fuel pool, which is an integral part of the RSB Seismic Category I structure. The pool is approximately **[---]** feet deep and is constructed of reinforced concrete with an attached stainless steel liner and concrete filled structural modules as described in Subsection 3.8.4. The minimum water volume of the SFP is **[---]** gallons of water. A SFP liner leak detection system and water level monitoring system are provided to detect leakage.

Spent fuel is stored in high density racks which include integral neutron absorbing material to maintain the required degree of subcriticality. The racks are designed to store fuel with the maximum design basis enrichment. Each free-standing rack consists of an array of cells interconnected to each other at several elevations and to a thick base plate at the bottom elevation. The spent fuel storage racks include storage locations for **[---]** fuel assemblies and **[---]** defective fuel assemblies. A typical spent fuel rack layout is shown in Figure 9.1.2-3.

The SFP is designed to include the following provisions to prevent inadvertent draining of water from the pool:

- piping penetrations for the drain and makeup lines are located such that draining of the SFP due to a break in a line or failure of a pump to stop is precluded
- the connection used for suction of the SFP cooling pumps is located below normal water level and above the level needed to provide sufficient water for shielding and for cooling of the fuel if the SFPCCS is unavailable as further described in Section 9.1.3
- a SFP liner leak detection system and water level monitoring system is provided to detect leakage

- a redundant, safety-related water makeup system is provided for each SFP

During refueling operations, the spent fuel assemblies are transferred through a fuel transfer tube that connects the refueling canal with the SFP through a gated opening. The spent fuel transfer operation is completed underwater, and the waterways are of sufficient depth to maintain a minimum of **[---]** feet of shielding water above the active fuel height of spent fuel assemblies. The refueling operations are handled by a single-failure-proof fuel handling crane as discussed in Section 9.1.4.

A shared spent fuel cask loading pit is located adjacent to the two SFPs and is accessible by another gated opening to each SFP. The cask loading pit is a stainless steel lined reinforced concrete structure that is also an integral part of the RSB Seismic Category I structure. The cask loading pit is normally closed and is opened only to provide underwater loading of spent fuel assemblies into a shipping cask and cask draining/decontamination either prior to shipment from the Clinch River site or to an independent spent fuel storage installation (ISFSI). The RSB gantry crane, as discussed in Section 9.1.5, is used for movement of a SFP shipping cask from the cask loading pit.

Ventilation for the fuel handling area is provided by a fuel handling area HVAC system designed to maintain slightly negative pressure with respect to adjacent areas in the RSB. This system includes air supply and filtration equipment. A discussion of the design and operation of the fuel handling area HVAC system is provided in Section 9.4. An engineered safety feature filtration system as described in RG 1.52 is not needed in the fuel handling area to limit offsite dose consequences from accident conditions as discussed in Chapter 15.

### 9.1.2.3 Safety Evaluation

The new and spent fuel storage facilities are designed consistent with the guidance provided in References 9.1.2-1 and 9.1.2-2, and RG 1.13 to maintain the fuel within the required degree of subcriticality and provide a safe means for load handling operations. In addition, the facility design complies with the regulatory requirements of GDC 2, 4, 5, 61, and 63, and meets the applicable provisions of RGs 1.29, 1.52, 1.115, 1.117, 1.140, and 8.8. Based on the design and operation of the single-failure-proof cranes and associated tools used to handle the new and spent fuel assemblies, adequate provisions are provided to ensure safe, efficient, and reliable fuel handling operations. Administrative controls are provided to ensure that new and spent fuel assemblies cannot be inserted into a location other than a location designed to receive an assembly.

The fuel handling area SSCs that are shared between mPower reactor modules are not used for accident mitigation and will not impair the safety functions of the new and spent fuel storage facilities. In addition, the physical routing of the shared systems does not pose a load drop hazard on the new or spent fuel storage facilities.

The new and spent fuel racks are designed to withstand normal and postulated dead loads, live loads, loads resulting from applicable thermal effects, and loads caused by an SSE. In addition, the spent fuel racks are designed to withstand the accident forces associated with fuel handling activities as discussed in Section 9.1.4. Structural materials selected for the rack construction are compatible with the storage environment and are corrosion resistant and will not contaminate the fuel assemblies. For the spent fuel racks, the selected absorber material is **[---]**. A program for monitoring the effectiveness of this martial to ensure the subcriticality requirements of the stored fuel are maintained will be established prior to plant operation.

Through adequate design and operational procedures, the design and operation of the storage facilities meet the requirements of 10 CFR 20.1101(b). The exposure of plant personnel to radiation is maintained below regulatory limits and in accordance with ALARA principles. A radiation monitoring system is provided in the fuel handling area to detect excessive radiation levels. To limit radiation doses to acceptable limits, the SFP is designed such that there are no drains, piping, or connections with other systems that could allow the water level to drain below approximately 10 feet above the top of the stored fuel assemblies consistent with the guidance in RG 1.13. A redundant, safety-related water makeup system provides the capability to maintain SFP level. The SFP storage rack arrays are also designed such that even in the event of loss of cooling in the SFP, boiling/natural convection will adequately cool the fuel. Additionally, the SFP capacity is such that the pool can boil for up to **[---]** days without adding fresh makeup water or uncovering the fuel.

To avoid the unnecessary buildup of radioactive material, smooth and non-porous surfaces are used for all components that come in contact with contaminated coolant (e.g., SFP liner and storage racks). Further details of radiological considerations, including those for the fuel handling area, are presented in Chapter 12.

### 9.1.2.4 References

9.1.2-1 ANSI/ANS-57.3-1983, Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants

9.1.2-2 ANSI/ANS-57.2-1983, Design Requirements for Light-Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants

**Table 9.1.2-1**
**Loads and Load Combinations for Fuel Racks**

| Load Combination | Acceptance Limit |
|---|---|
| {TBD} | {TBD} |

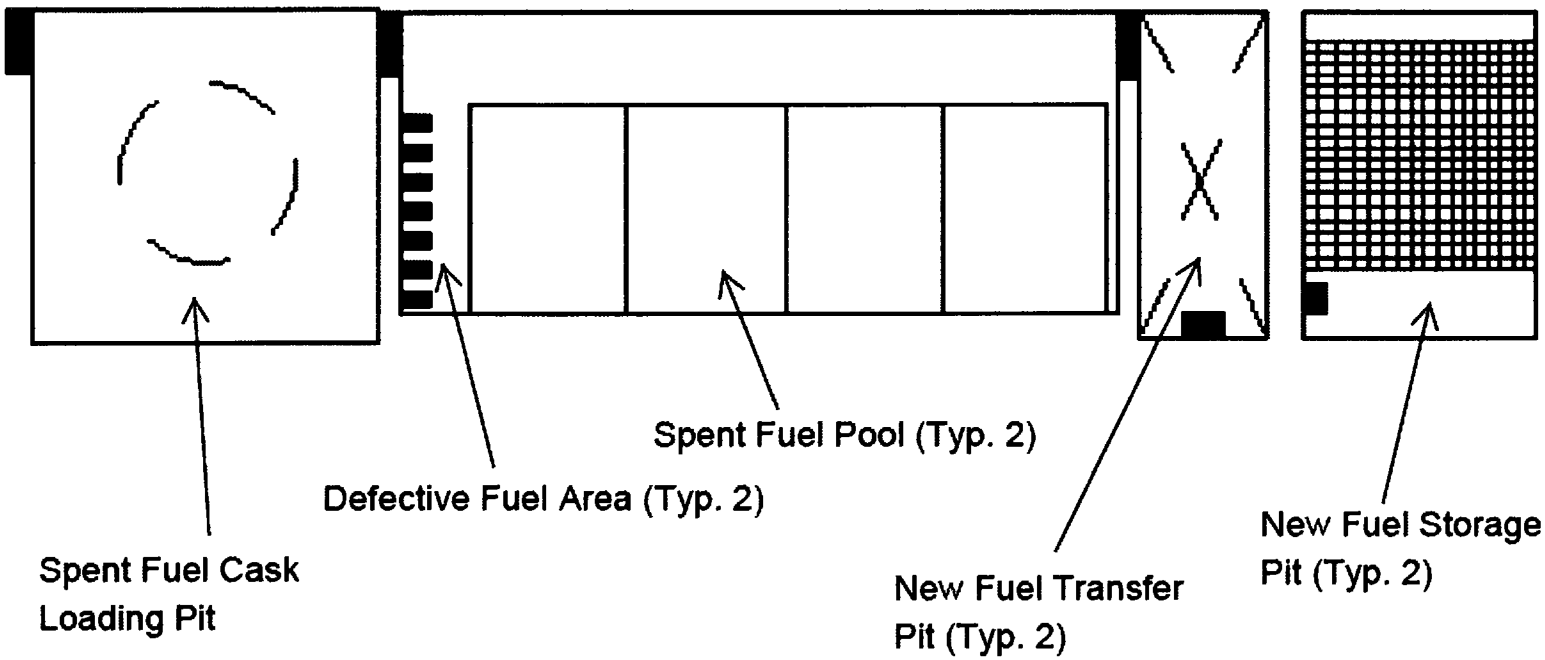



**Figure 9.1.2-1 Fuel Handling Area Representative Layout**

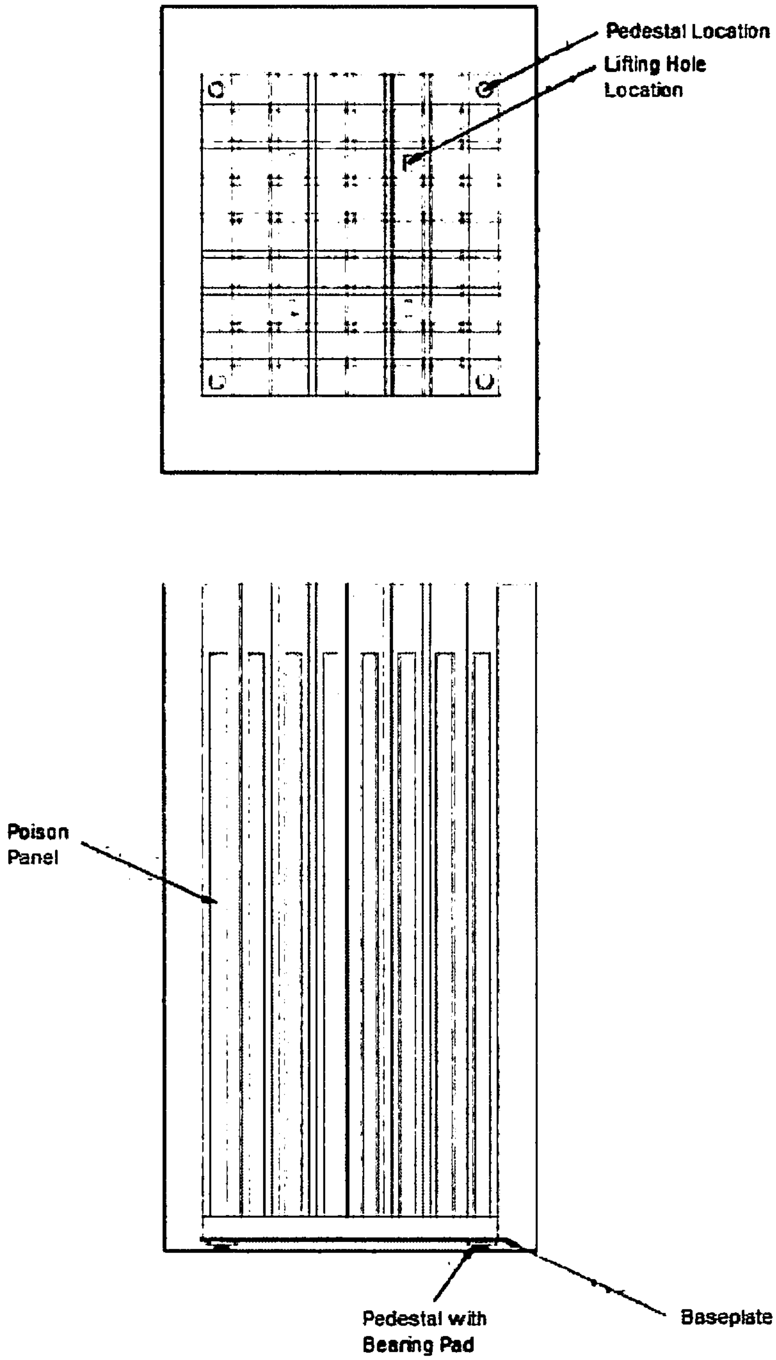



**Figure 9.1.2-2 New Fuel Rack Representative Layout**

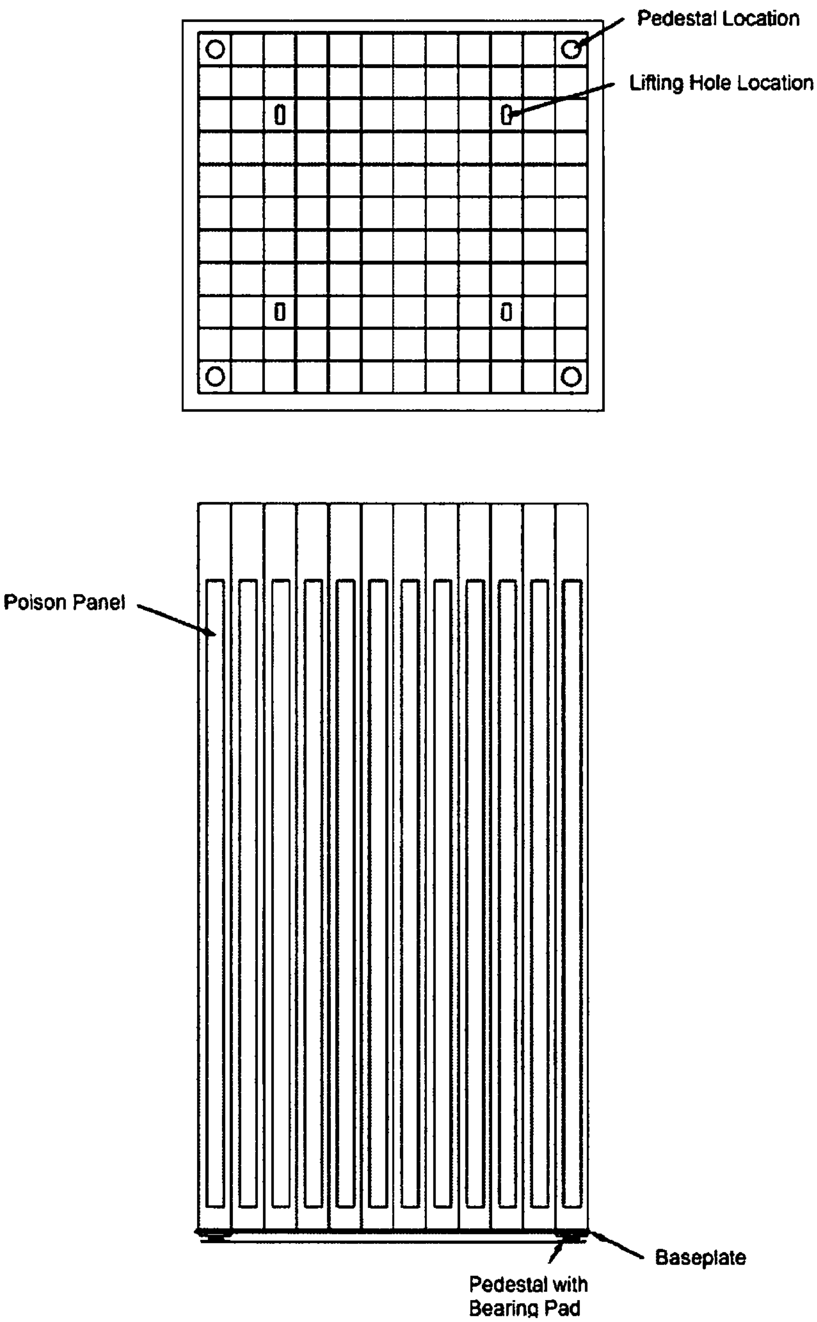



**Figure 9.1.2-3 Spent Fuel Storage Rack Representative Layout**

# Attachment 4

PSAR Section 5.3 RFD and Section Outline

**Clinch River Regulatory Framework Document**
**NRC Version**

| Section Number | 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 |
|---|---|---|---|---|---|---|---|---|---|---|
| 5.3.1<br><br>Reactor Vessel Materials | PSAR | 10 CFR 50, App. A, GDC 1, 4, 14, 30, 31, and 32<br>10 CFR 50, Appendices B, G and H<br>10 CFR 50.55a<br>10 CFR 50.60<br>10 CFR 50.61 | No | RG 1.70 | 5.3.1 | RGs 1.31, 1.34, 1.37, 1.43, 1.44, 1.50, 1.65, 1.71, 1.99 | ASTM E-1921, 10e1<br>ASTM E-1820, 09e1<br>EPRI Technical Report 1014986, December 2007<br>ASTM B-700, 08<br>EPRI 1015007, July 2008 | If none of the fracture toughness tests has been performed, the PSAR must contain a statement of the applicant's intention to perform this work In accordance with ASME Code Section III, NB-2300 and Appendix G of 10 CFR Part 50. The PSAR states the end-of-life fluence calculated for the vessel belt-line, the maximum predicted shift in reference transition temperature ($RT_{NDT}$), the number of capsules, the number and types of specimens to be placed in the capsules, and that the program is in compliance with ASTM E-185 and Appendix H, 10 CFR Part 50.(from SRP 5.3.1 Rev. 2, page 5.3.1-15) | No | 3.13<br>5.2<br>5.3 |
| | DCD | 10 CFR 50, App. A, GDC 1, 4, 14, 30, 31, and 32<br>10 CFR 50, Appendices B, G and H<br>10 CFR 50.55a<br>10 CFR 50.60<br>10 CFR 50.61<br>10 CFR 52.47(b)(1) | No | RG 1.206 Note: DCD will provide a commitment to document verification of the reactor vessel mechanical property and toughness test results in the FSAR to demonstrate that the material conforms to the regulatory guidance. | 5.3.1 | RGs 1.31, 1.34, 1.37, 1.43, 1.44, 1.50, 1.65 1.71, 1.99 SECY-05-197 | ASTM E-1921, 10e1<br>ASTM E-1820, 09e1<br>EPRI Technical Report 1014986, December 2007<br>ASTM B-700, 08<br>EPRI 1015007, July 2008 | N/A | N/A | 3.13<br>4.3<br>5.2<br>5.3 |
| | FSAR | 10 CFR 50, App. A, GDC 1, 4, 14, 30, 31, and 32<br>10 CFR 50 Appendices B, G and H<br>10 CFR 50.55a<br>10 CFR 50.60<br>10 CFR 50.61 | No | RG 1.206 | 5.3.1 | RGs 1.31, 1.34, 1.37, 1.43, 1.44, 1.50, 1.65, 1.71, 1.99 | ASTM E-1921, 10e1<br>ASTM E-1820, 09e1<br>EPRI Technical Report 1014986, December 2007<br>ASTM B-700, 08<br>EPRI 1015007, July 2008 | N/A | No | 3.13<br>4.3<br>5.2<br>5.3 |

| Section Number | 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 |
|---|---|---|---|---|---|---|---|---|---|---|
| 5.3.2<br><br>Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses | PSAR | 10 CFR 50, App. A, GDCs 1, 4, 14, 31, and 32<br>10 CFR 50, App. G<br>10 CFR 50.55a<br>10 CFR 50.60<br>10 CFR 50.61 | No | RG 1.70 -- Note: Section 5.3.2 of RG 1.70 only addresses Pressure-Temperature Limits; however, consistent with RG 1.206 format, PSAR Section 5.3.2 will also address pressurized thermal shock (PTS) and Upper Shelf Energy (USE). | 5.3.2<br>BTP 5-3 | RG 1.99<br>RG 1.161<br>RG 1.190 | ORNL/NRC/LTR-03/03 | Provide a commitment that the fracture toughness of the ferritic materials in the RCPB will comply with the requirements of Appendix G to 10 CFR Part 50, as detailed in Section XI of the ASME Code, and that the materials in the beltline region of the RV will comply with the requirements of 10 CFR 50.61 and the guidance of RG 1.99 (SRP Section 5.3.2, page 5.3.2-8). | No | 4.3 (RPV wall fluence)<br>5.2<br>5.3 |
| | DCD | 10 CFR 50, App. A, GDCs 1, 4, 14, 31, and 32<br>10 CFR 50, App. G<br>10 CFR 50.55a<br>10 CFR 50.60<br>10 CFR 50.61<br>10 CFR 52.47(b)(1) | No | RG 1.206 Note: Availability of the actual test results will be dependent upon the fabrication/construction schedules. | 5.3.2<br>BTP 5-3 | RG 1.99<br>RG 1.161<br>RG 1.190 | ORNL/NRC/LTR-03/03 | N/A | N/A | 4.3 (RPV wall fluence)<br>5.2<br>5.3 |
| | FSAR | 10 CFR 50 App. A, GDCs 1, 4, 14, 31, and 32<br>10 CFR 50, App. G<br>10 CFR 50.55a<br>10 CFR 50.60<br>10 CFR 50.61 | No | RG 1.206 - Actual material toughness test results and the predicted effects of irradiation as well as P-T limits using real temperature. Note: Availability of the actual test results will be dependent upon the fabrication/construction schedules. | 5.3.2<br>BTP 5-3 | RG 1.99<br>RG 1.161<br>RG 1.190 | ORNL/NRC/LTR-03/03 | N/A | No | 4.3 (RPV wall fluence)<br>5.2<br>5.3 |

**Clinch River Regulatory Framework Document**
**NRC Version**

| Section Number | 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 |
|---|---|---|---|---|---|---|---|---|---|---|
| 5.3.3<br><br>Reactor Vessel Integrity | PSAR | 10 CFR 50, App. A, GDC 1, 4, 14, 30, 31, and 32<br>10 CFR 50, Appendices B, G and H<br>10 CFR 50.55a<br>10 CFR 50.60<br>10 CFR 50.61 | No | RG 1.70 - Note: PSAR Section 5.3.3 will also address threaded fasteners consistent with the contents of Section C.I.5.3.3.8 of RG 1.206. | 5.3.3 | RG 1.99<br><br>NUREG/CR-6909 | None | None | No | 3.13<br>5.2<br>5.3 |
| | DCD | 10 CFR 50, App. A, GDC 1, 4, 14, 30, 31, and 32<br>10 CFR 50, Apps. B, G and H<br>10 CFR 50.55a<br>10 CFR 50.60<br>10 CFR 50.61<br>10 CFR 52.47(b)(1) | No | RG 1.206 | 5.3.3 | RG 1.99 | None | N/A | N/A | 3.13<br>5.2<br>5.3 |
| | FSAR | 10 CFR 50, App. A, GDC 1, 4, 14, 30, 31, and 32<br>10 CFR 50, Appendices B, G and H<br>10 CFR 50.55a<br>10 CFR 50.60<br>10 CFR 50.61 | No | RG 1.206 | 5.3.3 | RG 1.99 | None | N/A | No | 3.13<br>5.2<br>5.3 |

Notes:
(1) RG Revisions are not identified as these will be consistent with the version in effect 6 months prior to the PSAR submittal.

## 5.3 Reactor Vessel

### 5.3.1 Reactor Vessel Materials

**PSAR** The reactor vessel materials are described to demonstrate that materials selected under certain specifications and criteria meet the regulatory requirements, codes, and standards consistent with their safety function. Appropriate reactor vessel material selection is demonstrated based on the following information:

- table with preliminary list of materials in the reactor vessel, applicable attachments, and appurtenances
- preliminary data on the weld materials, fabrication methods, and inspection techniques to be used for the reactor vessel and applicable attachments and appurtenances
- description of the applicable criteria for the reactor vessel materials, including provisions of ASME Code, Section III and requirements provided in 10 CFR 50, Appendix G
- description of special processes to be used for the manufacture of the product forms and methods to fabricate the vessel or any of its applicable attachments and appurtenances
- description of special procedures to be used for detecting surface and internal discontinuities and, as applicable, information related to relief requests that may be required if alternative procedures than those in ASME Section III are planned to be used
- description of plans for special controls on welding, composition, heat treatments, and similar processes covered by Regulatory Guides (RGs) 1.31, 1.34, 1.43, 1.44, 1.50, and 1.71
- description of the proposed fracture toughness testing and acceptance criteria specified for materials of the reactor vessel and appurtenances and a commitment that the tests will be performed in accordance with ASME Code Section III, NB-2300 and the requirements of Appendix G to 10 CFR Part 50
- proposed plans for material surveillance program to comply with ASTM E-185 and 10 CFR 50, Appendix H, including provisions for monitoring the vessel materials for a 60-year design life
- description of the materials and preliminary design information for the stud bolts, washers, nuts, and other fasteners for the reactor vessel closure covered by RG 1.65

---

**DCD** DCD Section 5.3.1 expands upon the information provided in PSAR Section 5.3.1 to provide a complete description of the mPower standard plant reactor vessel material information as listed below:

- table with list of materials in the reactor vessel, applicable attachments, and appurtenances

## 5.3.1 Reactor Vessel Materials (cont.)

**DCD (cont.)**

- tables of inspection plan for reactor vessel materials and reactor vessel welds
- data on the weld materials, fabrication methods, and inspection techniques to be used for the reactor vessel and applicable attachments and appurtenances
- discussion on how material specifications meet and/or exceed the applicable provisions of ASME Code, Section III and conform to requirements provided in 10 CFR 50, Appendix G
- description of special processes for the manufacture of the product forms and methods to fabricate the vessel or any of its applicable attachments and appurtenances
- description of special procedures for detecting surface and internal discontinuities and, as applicable, relief requests required for alternative procedures than those provided in ASME Section III
- description of special controls on welding, composition, heat treatments, and similar processes covered by RGs 1.31, 1.34, 1.43, 1.44, 1.50, and 1.71
- description of the fracture toughness for the reactor vessel materials and appurtenances, including the maximum reference temperature, $RT_{NDT}$, (i.e., higher value of the nil ductility temperature), fracture toughness tests and acceptance criteria, and compliance with ASME Section III, NB-2300 and Appendix G to 10 CFR Part 50
- description of the reactor vessel material surveillance program, as discussed in SECY-05-0197, and its implementation milestones to demonstrate compliance with ASTM E-185 and 10 CFR 50, Appendix H, including provisions for monitoring the vessel materials for a 60-year design life and information related to use of an integrated surveillance program for multiple reactors covered by RG 1.99
- for reactor vessel fasteners, sufficient details regarding the materials property requirements, nondestructive evaluation techniques, lubricants or surface treatments, and protection provisions to show how the specifications of Appendix I to Section III of the ASME Code are met; how the data required by Appendix IV to Section III of the ASME Code is provided; and how the recommendations in RG 1.65, or equivalent measures, are followed

---

**FSAR** Same contents as mPower standard plant DCD Section 5.3.1 and the following supplemental information:

- results of fracture toughness tests on all ferritic materials of the reactor vessel
- results of mechanical property and fracture toughness tests to demonstrate that the materials from which the reactor vessel fasteners are fabricated conform to the recommendations of RG 1.65 or their equivalent, depending on the fabrication/construction schedules as appropriate
- relief requests if alternative procedures than those in ASME Section III are needed for detecting reactor vessel surface and internal discontinuities
- description of reactor vessel material surveillance programs and implementation milestones as referenced in DCD Table [13.x]
- figure showing the location and orientation of surveillance capsules

## 5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses

**PSAR** To ensure adequate safety margins of structural integrity for the ferritic components of the reactor coolant pressure boundary (RCPB) and the reactor vessel beltline region materials, specific regulations have been established under 10 CFR 50.60 and 10 CFR 50.61 related to pressure-temperature (P-T) limits, Charpy upper-shelf energy values, and pressurized thermal shock (PTS). Design basis information regarding how these requirements are met are defined in this PSAR section, including:

-the bases for setting operational P-T limit curves for (1) preservice system hydrostatic tests, (2) inservice leak and hydrostatic tests, (3) normal operation, including heatup and cooldown, and (4) reactor core operation
-a commitment that the fracture toughness of the ferritic materials in the RCPB will comply with the requirements of Appendix G of 10 CFR Part 50, as detailed in Section XI of the ASME Code
-screening criteria for the reactor vessel beltline region, where the PTS reference temperature ($RT_{PTS}$) is not to exceed 270°F for forgings and 300°F for weld materials
- a commitment that materials in the beltline region of the reactor vessel will comply with the requirements of 10 CFR 50.61 and the guidance of RG 1.99

---

**DCD** DCD Section 5.3.2 expands upon the design bases information provided in PSAR Section 5.3.2 relative to the mPower standard plant design P-T limits, the reactor vessel beltline Charpy Upper-Shelf Energy (USE) values, and an assessment of the potential for PTS, including calculational methodology, assumptions, and results as described below:

- figure(s) of representative P-T limit curves for the mPower reactor standard design based on a 60-year design life, generated using the reactor vessel neutron fluence data, and as referenced in the Pressure and Temperature Limits Report (PTLR) and generic mPower standard plant Technical Specifications
- USE calculations based on material specifications for the reactor vessel beltline region weld and forging material as well as the neutron fluence
- required initial and end-of-license (EOL) USE values, where the EOL USE for the mPower reactor vessel base material and weld are identified
- projected USE value for a 60-year design life due to radiation embrittlement for the mPower reactor vessel in accordance with the requirements of RG 1.99
- table summarizing EOL $RT_{NDT}$ and USE for beltline materials
- description of procedures used to update P-T limits during operation that will address radiation effects and conformance with the recommendations of RG 1.190

---

## 5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses (cont.)

**FSAR** Same contents as mPower standard plant DCD Section 5.3.2 and the following supplemental information:

- reference to plant technical specifications or P-T limits report showing the P-T limits using real temperature for all required conditions, and the limiting $RT_{NDT}$ incorporates radiation effects
- calculation of USE at EOL based on plant-specific material property requirements and verified
- reference to operating procedures to be used to update the operating P-T limits, associated radiation effects, and conformance with recommendations of RG 1.190
- description of plant operating procedures that will ensure that the P-T limits will not be exceeded during any foreseeable upset condition
- PTS evaluation based on actual reactor vessel material property requirements and projected 60-year neutron fluence values for $RT_{PTS}$ and, if these values are projected to exceed the PTS screening criterion, an analysis and schedule for implementation of flux reduction programs or one of two options in 10 CFR 50.61 to meet the PTS requirements
- actual material toughness test results

## 5.3.3 Reactor Vessel Integrity

**PSAR** The design, fabrication, testing, and installation of the reactor vessel are carried out to satisfy the requirements of 10 CFR 50.55a, 10 CFR 50.61, and 10 CFR 50, Appendix A, GDC 1, 4, 14, 30, 31 and 32. Design and fabrication of the reactor vessel also satisfy requirements of ASME Code Section III. This section provides a summary of information related to the reactor vessel integrity as follows:

- description and schematic of the reactor vessel design, including materials, construction features, fabrication methods, and inspections
- table(s) providing a summary of applicable design codes and bases with reference to applicable table(s) in PSAR Section 5.3.1 and PSAR Section 3.13 on threaded fasteners.
- table of reactor vessel dimensions and major design parameters
- summary of the materials used and any special requirements to improve their properties or quality, emphasizing the reasons for selection and assurance of suitability
- summary of fabrication methods and the service history of vessels constructed using these methods and the vessel supplier's experience with the procedures.
- summary of inspection test methods and requirements with particular attention to the level of initial integrity and any methods that are in addition to the guidelines in Section III of the ASME Code
- summary of the means used to protect the vessel so that its as-manufactured integrity will be maintained during shipment and site installation with reference to PSAR Sections 5.2.3 and 5.3.1 as appropriate

---

**DCD** Same contents as the PSAR Section 5.3.3 and the following supplemental information:

- diagrams of the reactor vessel (side view and cross-sectional view)
- summary of the inservice Inspection (ISI) and reactor vessel material surveillance programs and an explanation of their adequacy relative to Appendix H to 10 CFR Part 50 and Section XI of the ASME Code
- summary of the operational limits to ensure reactor vessel safety and a basis for concluding that vessel integrity will be maintained during the most severe postulated transients and PTS events
- summary of requirements for ensuring the integrity of bolting and threaded fasteners with reference to information in DCD Section 3.13 as appropriate

---

## 5.3.3 Reactor Vessel Integrity (cont.)

**FSAR** Same contents as mPower standard plant DCD Section 5.3.3 and the following supplemental information:

- complete description of the ISI and reactor vessel material surveillance programs and reference to implementation milestones as defined in FSAR Table [13.x]
- summary of operational programs for ensuring the integrity of bolting and threaded fasteners with reference to information in FSAR Section 3.13 as appropriate

---

# Attachment 5

PSAR Subsection 2.4.12 RFD and Section Outline

# 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 |
|---|---|---|---|---|---|---|---|---|---|---|
| 2.4.12<br><br>Groundwater | PSAR | 10 CFR 100.20(c)(3)<br>10 CFR 100.23(d)<br>10 CFR 50.55a<br>10 CFR 50, Appendix A, GDC 2, 4, 5, 44 | No | RG 1.70 - with updated information as discussed in DC/COL-ISG-014<br><br>Note: Section 2.4.12 of RG 1.70 addresses "Dispersion, Dilution, and Travel Times of Accidental Releases of Liquid Effluents in Surface Waters" while 2.4.13 refers to "Groundwater". To be consistent with RG 1.206 format, PSAR Section 2.4.12 will address groundwater. | 2.4.12 | RG 1.27<br>DC/COL-ISG-014 | Technical Guidance Documents (latest versions as appropriate)<br>- ASTM D 4044<br>- ASTM D 4327<br>- ASTM D 4630<br>- ASTM D 5088<br>- ASTM D 5092<br>- ASTM D 5903<br>- ASTM D 6089<br>- ASTM D 6452<br>- ASTM D 6517<br>- ASTM D 7069<br>- EPA 160.1<br>- EPA Method 300.0<br>- EPA 310.1<br>- EPA Method 350.1<br>- EPA Method 6020A<br>NEI 07-07, August 2007 | No | No | PSAR 2.0, 2.5.1, 2.5.4<br>2.4.13<br>9.2.5<br>ER 2.3, 4.2, 5.2 |
| | DCD | 10 CFR 52.47(a)(1)<br>10 CFR 100 | No | RG 1.206 | 2.4.12<br>2.0 | N/A - Site-Specific | N/A - Site-Specific | N/A | N/A | DCD 2.0 |
| | FSAR | 10 CFR 100.20(c)(3)<br>10 CFR 100.23(d)<br>10 CFR 50.55a<br>10 CFR 50, Appendix A<br>GDC 2, 4, 5, 44 | No | RG 1.206 | 2.4.12 | RG 1.27<br>DC/COL-ISG-014 | Technical Guidance Documents (latest versions as appropriate)<br>- ASTM D 4044<br>- ASTM D 4327<br>- ASTM D 4630<br>- ASTM D 5088<br>- ASTM D 5092<br>- ASTM D 5903<br>- ASTM D 6089<br>- ASTM D 6452<br>- ASTM D 6517<br>- ASTM D 7069<br>- EPA 160.1<br>- EPA Method 300.0<br>- EPA 310.1<br>- EPA Method 350.1<br>- EPA Method 6020A<br>NEI 07-07, August 2007 | N/A | N/A | FSAR 2.0, 2.5.1, 2.5.4<br>2.4.13<br>9.2.5<br>ER 2.3 |

Notes:

(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**
**REGULATORY FRAMEWORK DOCUMENTS**

**Section 2.4.12 Outline**

## 2.4.12 Groundwater

The level of detail and contents of PSAR Section 2.4.12 (Groundwater) will be consistent with that provided in FSAR Section 2.4.12 for a Combined License (COL) Application. A subsurface investigation consisting of observation well installation; monthly water level measurements; aquifer slug, packer, and pumping tests will be performed. The information collected will support the development of the Clinch River site hydrogeologic conceptual model. In addition, a groundwater numerical flow model will be developed and used to assess the local groundwater resources that could be affected by the construction and operation of the mPower SMR. The information contained in PSAR 2.4.12 will also be used as input to the development of PSAR 2.4.13 (Accidental Release of Radioactive Liquid Effluent in Ground and Surface Waters).

---

**PSAR** This section describes the hydrogeologic conditions present at, and in the vicinity of, the Clinch River site, including regional and local groundwater resources that could be affected by the construction and operation of the mPower SMR. The regional and site-specific data on the physical and hydrogeologic characteristics of these groundwater resources is described as follows:

- regional and local historical and current water use
- regional and local projected water use
- types of groundwater use, wells, water storage facilities, and flow requirements of the plant
- regional and local identification of aquifers, formations, sources, and sinks
- hydrogeologic properties of regional and local formations and aquifers
- groundwater flow direction (gradients) and magnitude (velocities)

An evaluation of the hydrogeology in the Clinch River site area is performed by developing a hydrogeologic conceptual site model, which includes the following components:

- characteristics of vadose zone and aquifer structures or heterogeneity identification of preferential flow or barriers
- temporal and spatial variations of hydraulic gradients
- hydraulic properties, including the conductivity, storage, porosity transmissivity, and heterogeneity and anisotropy of these parameters
- aquifer boundary conditions
- aquifer usage information, location and production data for water-supply wells, and estimates of future uses
- groundwater recharge and discharge
- groundwater interaction with surface water
- groundwater quality
- quantitative description of groundwater flow
- groundwater level trend analysis

---

## 2.4.12 Groundwater (cont.)

**PSAR (cont.)** To simulate pre-construction and post-construction groundwater conditions and hydrogeologic impacts of plant operations at the Clinch River Site, a groundwater numerical model is developed to identify:

- construction dewatering requirements
- expected normal and maximum groundwater levels for safety-related structures; and
- subsurface groundwater migration pathways to support a conservative analysis of an accidental liquid effluent release (used in support of PSAR 2.4.13).

Results of the groundwater numerical model are provided along with reference to a technical report providing details of the numerical model and detailed results of groundwater analysis.

This section also provides the maximum site-specific groundwater elevations and the design bases for groundwater-induced hydrostatic loadings on subsurface portions of the Clinch River safety-related structures, systems, and components. During excavation and construction at the Clinch River site, the hydrostatic loading on the excavation and structures will be controlled by a temporary construction dewatering system as outlined below:

- description of the proposed construction dewatering system
- design bases for subsurface hydrostatic loadings assumed during construction
- figures showing proposed locations, components and features of the construction dewatering system

PSAR Section 2.4.12 includes a discussion on the need for a permanent dewatering system at the Clinch River site. If applicable, a description of the proposed permanent dewatering system is presented.

A description of the proposed plans, procedures, safeguards, and groundwater monitoring programs for the Clinch River site is also provided in this section.

---

**DCD** Reference to DCD Section 2.0 that provides a table on "Key Site Parameters," including the groundwater site parameter for the mPower standard plant design.

---

## 2.4.12 Groundwater (cont.)

**FSAR** Same contents as the Clinch River PSAR Section 2.4.12 and the following supplemental information:

- comparison of the Clinch River groundwater site characteristic with the groundwater site parameter for the mPower standard plant design
- confirmation of estimates of groundwater levels
- updated site model based on post-PSAR groundwater data
- final plans, procedures, safeguards, and groundwater monitoring programs to be used to protect present and projected groundwater users (as applicable)