15.0 SAFETY ANALYSES

15.1 Introduction

This chapter provides analyses of the plant's responses to postulated disturbances in process variables and postulated equipment failures or malfunctions, determines their consequences, and evaluates the capability of the plant to control or accommodate these events. These analyses help determine the limiting conditions for operation, limiting safety system settings, and design specifications for safety-related components and systems.

The analyses in this chapter includes a discussion of: (1) the classification of the transients and accidents and their results in the context of a sufficiently broad spectrum of initiating events and postulated equipment failures, (2) the frequency of occurrence for initiating events for anticipated operational occurrences and highly unlikely accidents, (3) plant characteristics considered in the safety evaluation, (4) assumed protection system actions, (5) evaluation of individual initiating events and systems that operate to reduce the probability of occurrence of specific events, and (6) analysis of anticipated transients without scram. The safety analyses provide a significant contribution to the selection of limiting conditions for plant operation, limiting safety system settings, and design specifications for plant components and systems from the standpoint of public health and safety.

15.2 Summary of Application

Chapter 15 of the Fermi 3 Combined License (COL) Final Safety Analysis Report (FSAR), Revision 7 incorporates by reference, with no departures, Chapter 15 of Revision 10 of the certified Economic Simplified Boiling-Water Reactor (ESBWR) Design Control Document (DCD). In addition, in FSAR Chapter 15, the applicant provides the following:

Supplemental Information

- STD SUP 15.3-1

The applicant states that the procedures will discuss the use of nuclear instrumentation to aid in detecting a possible mislocated fuel bundle after a fueling operation.

- EF3 SUP 15.4-1

In a letter dated August 22, 2014 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML14237A333), the applicant provides supplemental information in Section 15.4.1 of the COL FSAR, which discusses administrative control of certain doors or personnel air locks during movement of irradiated fuel, as related to the design basis accident control room habitability dose analysis for the fuel handling accident.

15.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is in NUREG–1966, "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor." In addition, STD SUP 15.3-1 is subject to the requirements of General Design Criterion (GDC) 13, "Instrumentation and Control," and the relevant guidance of the Commission regulations in the acceptance criteria of Section 15.4.7, "Inadvertent Loading and Operation of a

Fuel Assembly in an Improper Position," of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, (LWR Edition)," the Standard Review Plan (SRP).

15.4 Technical Evaluation

As documented in NUREG–1966, U.S. Nuclear Regulatory Commission (NRC) staff reviewed and approved Chapter 15 of the certified ESBWR DCD. The staff reviewed Chapter 15 of the Fermi 3 COL FSAR and checked the referenced ESBWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ESBWR DCD represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and the information incorporated by reference address the required information related to safety analyses.

The staff reviewed the information in the COL FSAR:

Supplemental Information

- STD SUP 15.3-1

The applicant states that the procedures will detail the use of nuclear instrumentation in helping to detect a possible mislocated fuel bundle after fuel loading. The staff found the supplemental information acceptable because it is consistent with the acceptance criteria in SRP Section 15.4.7, which states that plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel-loading errors after fueling operations, in order to meet the requirements of GDC 13.

- EF3 SUP 15.4-1

By a letter dated August 22, 2014 (ADAMS Accession No. ML14237A333), the applicant provided supplemental information to clarify operator actions that are related to the analysis of the design basis fuel handling accident (FHA) radiological consequences in the Fermi Unit 3 control room. Specifically, the applicant proposes to add the following site-specific supplemental information to the next revision of FSAR Subsection 15.4.1.2.3, "Identification of Operator Actions:"

> During movement of irradiated fuel, doors or personnel air locks on the east sides of the Reactor Building or Fuel Building could act as a point source that could result in control room χ/Q values that are higher than the ESBWR χ/Q values for a release in the Reactor Building or Fuel Building (See Subsection 2A.2.5). Therefore, the doors and personnel air locks on the east sides of the Reactor Building and Fuel Building are administratively controlled to remain closed during movement of irradiated fuel. Administrative control of these doors and air locks ensures that the control room habitability dose analysis for the fuel handling accident (FHA) incorporated by reference from ESBWR DCD Section 15.4.1 is bounding for Fermi Unit 3 and control room doses do not exceed the requirements of GDC 19 in the event of a FHA.

[1] See *"Finality of Referenced NRC Approvals"* in SER Section 1.2.2 for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

ESBWR DCD, COL Item 2A.2-2-A, “Confirmation of the Reactor Building χ/Q Values,” gives guidance to COL applicants that if the site-specific point source control room receptor atmospheric dispersion factors (χ/Q) values for potential releases through doors or personnel air locks on the east sides of the reactor building (RB) and fuel building (FB) are greater than those used as site parameter values in the ESBWR DCD dose analysis for the FHA and would result in a higher radiological consequence than was reported in the DCD, then the affected doors or air locks are administratively controlled during movement of irradiated fuel. The applicant did not provide site-specific point source control room receptor χ/Q values for releases through the doors and air locks on the east sides of the RB and FB or a comparison to the values used in the ESBWR DCD for the FHA to make a determination whether the dose in the control room for the FHA would be higher than reported in the ESBWR DCD. Instead, the applicant stated in EF3 COL 2A.2-2-A that the affected doors and air locks are administratively controlled to remain closed during movement of irradiated fuel. This statement was repeated in Supplemental Item EF3 SUP 15.4-1 with additional information to include the relationship to the assumptions used in the FHA control room dose analysis. The staff finds the supplemental information acceptable because administrative control of the doors and air locks on the east side of the RB and FB during the movement of irradiated fuel provides assurance that in the event of an FHA, releases through the doors are sufficiently prevented so that the FHA dose analysis incorporated by reference from ESBWR DCD Tier 2 Section 15.4.1 is bounding for Fermi Unit 3. The applicant’s commitment to providing EF3 SUP 15.4-1 in a future revision of the FSAR is Confirmatory Item 15-1. The staff verified that FSAR Revision 7 includes EF3 SUP 15.4-1. Therefore, Confirmatory Item 15-1 is resolved.

Section 2.3.4, “Short-Term (Accident) Diffusion Estimates,” of this safety evaluation report (SER) discusses the staff’s evaluation of the radiological consequences associated with design-basis accidents (DBAs) and comparison of site characteristic atmospheric dispersion estimates to the DCD analysis assumptions as discussed in ESBWR COL Information Item 2.0-1-A, “Site Characteristics Demonstration.” The DBA radiological consequence analyses in the ESBWR DCD used design reference site parameter values for the offsite χ/Qs, in place of site characteristic (site-specific) values. The χ/Q values are the only input to the DBA radiological consequence analyses that are affected by the site characteristics. The applicant provided and discussed the Fermi 3 site characteristic short-term accident χ/Q values in FSAR Sections 2.3.4 and Appendix 2A in response to COL information items EF3 COL 2.0-10-A, “Short-Term Dispersion Estimates for Accidental Atmospheric Releases,” and EF3 COL 2A.2-1-A, “Confirmation of the ESBWR χ/Q Values.”

The estimated DBA dose for a particular site is affected by the site characteristics through the calculated χ/Q input to the analysis; therefore, the resulting dose would be different than that calculated generically for the ESBWR design in the DCD. All other inputs and assumptions in the radiological consequences analyses remain the same as those in the DCD. Smaller χ/Q values are associated with greater dilution capability, resulting in lower radiological doses. When comparing a DCD site parameter χ/Q value and a site characteristic χ/Q value, the site is acceptable for the design if the site characteristic χ/Q value is smaller than the site parameter χ/Q value. Such a comparison shows that the site has better dispersion characteristics than that required by the reactor design.

For each time averaging period, the Fermi 3 site characteristic offsite and control room short-term χ/Q values are less than the site parameter χ/Q values used by the ESBWR DCD, Revision 10, radiological consequence analysis for each of the DBAs. Since the result of the radiological consequence analysis for a DBA during any time period of radioactive material release from the plant is directly proportional to the χ/Q for that time period, and because the

Fermi 3 site characteristic χ/Q values are less than the comparable ESBWR DCD site parameter χ/Q values for all time periods and all accidents, the Fermi 3 site-specific total dose for each DBA is therefore less than the ESBWR DCD, Revision 10, generic total dose for each DBA. The ESBWR DCD, Revision 10, analyses show that the offsite, control room, and the technical support center (TSC) radiological consequences meet the regulatory dose requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 100.21, "Non-seismic siting criteria," 10 CFR 50.34(a)(1), and 10 CFR 52.79(a)(1) for offsite receptors, GDC 19 "Control room"; and Paragraph IV.E.8 of Appendix E "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," for the TSC. Because, the Fermi 3 site-specific DBA offsite, control room and TSC doses are less than those given in the ESBWR DCD, the applicant has sufficiently shown that the Fermi 3 DBA radiological consequences meet the requirements of 10 CFR 100.21, 10 CFR 50.34(a)(1), 10 CFR 52.79(a)(1), GDC 19 and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50.

Technical Specifications

COL application Part 4, "Technical Specifications," Section 5.6.3, "Core Operating Limit Report (COLR)," Item (c) states:

> The core operating limits shall be determined such that all applicable limits, (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The Fermi 3 technical specifications are evaluated in Chapter 16 of this SER.

15.5 Post Combined License Activities

The applicant states in Supplemental Information EF3 SUP 15.4-1 that the doors and personnel air locks on the east sides of the Reactor Building and Fuel Building are administratively controlled to remain closed during movement of irradiated fuel.

15.6 Conclusion

The NRC staff's finding related to information incorporated by reference is in NUREG–1966. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirms that the applicant addressed the required information, and no outstanding information is expected to be addressed in the COL FSAR related to this chapter. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Appendix E, "Design Certification Rule for the Economic Simplified Boiling-Water Reactor," Section VI.B.1, all nuclear safety issues relating to "Safety Analyses" that were incorporated by reference are resolved.

In addition, the staff finds that the additional information in the application meets the relevant NRC regulations and is consistent with the guidance in Chapter 15 of NUREG–0800.