14.0 SAFETY ANALYSIS [UNIT 1]

This chapter presents an evaluation of the safety aspects of Unit 1 of the Donald C. Cook Nuclear Plant and demonstrates that Unit 1 can be operated safely even if highly unlikely occurrences are postulated. It also shows that radiation exposures to the public and control room as a result of these highly unlikely occurrences do not exceed the limits of Regulatory Guide 1.183 and 10 CFR 50.67 .

Cook Nuclear Plant Unit 1 is currently loaded with fuel manufactured by Westinghouse Electric Corporation, and this chapter reports on those safety analyses performed in support of operation with the current Westinghouse fuel. The current safety analyses additionally support operation over a range of reactor coolant system (RCS) temperatures as described in Table 14.1-1.

This chapter is divided into four sections and two appendices. The first three sections each deal with a different licensing basis category of fault condition, and the last is concerned with analyses performed in support of environmental qualification of structures, systems, and components.* The ANS Conditions II, III, and IV are based on the anticipated frequency of their occurrence and are related to the licensing basis categories as described below. There are four ANS fault conditions: Condition I, Condition II, Condition III, and Condition IV. ANS Condition I occurrences do not require a safety analysis because they represent normal operational transients.

ANS Condition II occurrences are faults that may occur with moderate frequency during the life of the plant. They are accommodated with, at most, a reactor shutdown with the plant being capable of returning to operation after a corrective action. In addition, no ANS Condition II occurrence shall cause consequential loss of function of fuel cladding and reactor coolant system barriers.

ANS Condition III occurrences are faults that may occur very infrequently during the life of the plant. They may be accompanied by the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use

* The three categories of fault conditions analyzed in this chapter do not have a one to one correspondence with the ANS Conditions II, III, and IV, but each fault condition in each category is also identified as either ANS Condition II, III, or IV.

of those areas beyond the exclusion radius. An ANS Condition III occurrence will not, by itself, generate an ANS Condition IV occurrence or result in a consequential loss of function of the reactor coolant system or containment barriers.

ANS Condition IV occurrences are faults that are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic occurrences that must be designed against and represent limiting design cases. ANS Condition IV occurrences shall not cause a fission product release to the environment resulting in a radiation exposure to the public in excess of the guidelines in Regulatory Guide 1.183 and 10 CFR 50.67. A single ANS Condition IV occurrence shall not cause a consequential loss of required functions of systems needed to cope with the fault including those of the emergency core cooling system (ECCS) and the containment.

Section 14.1 Core and Coolant Boundary Protection Analysis

The majority of the fault conditions discussed in this section are ANS Condition II occurrences. Section 14.1 also includes an ANS Condition III occurrence, complete loss of forced reactor coolant (Section 14.1.6) and an ANS Condition IV occurrence, locked rotor (Section 14.1.6).

Section 14.2 Standby Safeguards Analysis

The fault conditions listed in this section are very infrequent and may lead to a breach of fission product barriers. Section 14.2 includes occurrences other than ANS Condition III occurrences, such as rupture of a control rod drive mechanism housing (Section 14.2.6), and rupture of a steam line (Section 14.2.5), which are ANS Condition IV occurrences.

Section 14.3 Reactor Coolant System Pipe Rupture (Loss of Coolant Accident or LOCA)

The occurrence discussed in this section is a rupture of a reactor coolant pipe including the double-ended severance of the largest pipe in the reactor coolant system (large break LOCA or LBLOCA), which is the worst conceivable and therefore is used as a basis for the design of engineered safeguards. Section 14.3 also includes small break LOCA (SBLOCA) (Section 14.3.2), which is an ANS Condition III occurrence.

SECTION 14.4 ENVIRONMENTAL QUALIFICATION ANALYSES

This section discusses where analyses applicable to the environmental qualification of equipment important to safety may be found. It also provides one example of such analyses, that associated with high energy line breaks outside of containment.

Changes from Base Accident Analyses: Note Concerning Tables and Figures

If an evaluation is needed for some change from a base accident analysis of record, and the evaluation changes information appearing in the UFSAR, a text description of the evaluation is provided in the appropriate text section of Chapter 14. However, unless specifically indicated otherwise, the associated tables and figures for the accident analysis are taken from the base accident analysis of record, and not any subsequent specific evaluation.

Unit 1 Replacement Steam Generators (RSGs)

All accident analyses presented in Chapter 14 of the D.C. Cook 1 UFSAR were evaluated to determine the effects of replacing the Westinghouse Series-51 OSGs with Babcock & Wilcox Canada RSGs. The objective of each evaluation was to demonstrate that the D.C. Cook Unit 1 plant response with the RSGs would meet all applicable UFSAR acceptance criteria. The plant parameters (e.g. reactivity coefficients, valve capacities, fluid volume, tube surface area, etc.) that affect the calculated approach to the acceptance criteria were identified. These plant parameters were compared to identify any changes from the original steam generators (OSG) to the RSG. Once the applicable steam generator parameter differences were determined for each event, an evaluation was performed to verify either that the accident analyses in the UFSAR bound the response with the RSGs or that the plant response would meet all acceptance criteria following steam generator replacement. Each event was evaluated based on all safety system actuation setpoints and initial conditions, as listed in this section, remaining unaltered following steam generator replacement. The evaluations are based on “10 percent” tube plugging criteria for the replacement steam generators. Initially, the majority of the evaluations are being performed against current analyses and license basis associated with the original steam generators at “30 percent” tube plugging criteria.

With respect to the 30 percent tube plugging analysis prepared to support the degrading OSGs (Reference 14.1.0.10.10), the RCS flow for the RSGs plugged at 10 percent is higher than the RCS flow for the OSGs plugged at 30 percent. As a result, the 30 percent tube plugging analysis is bounding for the RSGs at the maximum (10 percent) plugging level. In addition, the slightly lower flow resistance of the RSGs will result in the RSGs having a slightly higher best estimate flow at comparable plugging levels.

A section was added to each accident that summarizes the results of each evaluation performed for the Unit 1 RSGs and concludes that each analysis of record is applicable for Unit 1 with the RSGs.

Unit 1 MUR Program

Accident analyses presented in Chapter 14 of the D.C. Cook 1 UFSAR were analyzed or evaluated to determine the effects of implementing a Measurement Uncertainty Recapture (MUR) power uprate. The objective of each evaluation was to demonstrate that the D.C. Cook Unit 1 plant response at the MUR power uprate level would meet all applicable UFSAR acceptance criteria. The only significant plant parameters that changed for the MUR power uprate were initial power level and power measurement uncertainty. Other than the change in power level and associated measurement uncertainty, the safety system actuation setpoints and initial conditions were not changed from previous analyses in performing the evaluations and analyses to implement the MUR power uprate.

A section was added to each accident that summarizes the results of each evaluation performed for the Unit 1 MUR Program and concludes that each analysis of record is applicable for Unit 1 at the MUR uprated power level.

Unit 1 NOP/NOT Program

Accident analyses presented in Chapter 14 of the D.C. Cook Unit 1 UFSAR were analyzed or evaluated to determine the effects of returning to Reactor Coolant System (RCS) Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) conditions. The most significant change to plant parameters for the Return to RCS NOP/NOT program, beyond the increased RCS pressure (2250 psia) and analyzed full power primary vessel average temperature (575.4°F), was the removal of prior restrictions on the maximum allowable values for T' and T'' in the overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT) setpoint equations, respectively, which had been limited to a value less than the maximum full power primary vessel average temperature.

A section was added to each accident that summarizes the results of each evaluation performed for the Unit 1 Return to RCS NOP/NOT Program. The evaluations demonstrated that, with the exception of the Rupture of a Steam Pipe event, the analyses for the RCS reduced temperature and pressure operation at MUR conditions supporting up to 30% SGTP remained bounding of operation at the Return to RCS NOP/NOT conditions. For the Rupture of a Steam Pipe event, an explicit analysis was performed to demonstrate that adequate overpower protection is provided when the event is initiated from full power conditions.

The Rupture of a Steam Pipe at Full Power analysis only supports the RCS NOP pressure of 2250 psia; therefore, the non-LOCA safety analyses no longer support operation at the reduced RCS pressure of 2100 psia.