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14.3 REACTOR COOLANT SYSTEM PIPE RUPTURE (LOSS OF COOLANT ACCIDENT)

14.3.3 Core and Internals Integrity Analysis

The accident analysis for 15x15 Upgrade fuel is presented in Section 3.5.1. All of this material is considered to complement additional material on core and intervals integrity analysis presented in earlier chapters of this updated safety analysis report.

The information presented below reflects the analyses that were performed to support the rerating and temperature reduction program for Donald C. Cook Nuclear Plant Units 1 and 2 (Reference 15). The following information reflects the most recent reactor vessel internals analyses.

14.3.3.1 Input Assumptions and Results for LOCA Hydraulic Forcing Function Evaluation

For the Cook Nuclear Plant units a mechanistic fracture evaluation (Reference 11) was performed which demonstrated that the analytical conditions and margins against crack extension satisfy the criteria established by the staff so that the potential for rupture is low so that breaks in the main reactor coolant piping up to and including a break equivalent in size to the rupture of the largest pipe need not be postulated as a design basis for defining structural loads on or within the reactor vessel and the rest of the RCS main loops. This evaluation, corresponding Unit 2 license amendment No. 76 (Reference 14) and the NRC's revision to GDC-4 allow the exemption of postulating pipe ruptures of the primary loop for the Cook Nuclear Plant Units 1 and 2 Reduced Temperature and Pressure and Rerating Programs. The original qualification for the reactor internals is based on a double-ended severance of a reactor coolant system pipe, (WCAP-7332-L). However, in order to verify that the forcing functions based on the revised operating parameters are bounded by those previously analyzed, the next most limiting branch line break was analyzed. This comparison established that the limiting branch line loads for the accumulator line rupture are less severe than the reactor pressure vessel inlet nozzle (RPVIN) double-ended rupture in the original analysis.

Subsequently, the NRC modified 10 CFR 50 General Design Criteria 4, and published in the Federal Register (Vol. 52, No. 207) on October 27, 1987 its final rule, "Modification of General Design Criteria 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures". This change to the rule allows use of leak before break (LBB) technology for excluding from the design basis the dynamic effects of postulated ruptures in primary coolant loop piping in pressurized water reactors. Westinghouse prepared WCAP-15131 to account for the changes to GDC-4, the effects of thermal aging of primary pipe and Unit 1 and 2 replacement steam generators. The analysis concluded that the LBB criterion remains valid for the Unit 1 and 2 current loading conditions and as a result, dynamic effects of reactor coolant system primary loop postulated pipe breaks need not be considered in the structural design basis for Unit 2 uprating and for Units 1 and 2 replacement steam generators conditions (Reference 18). The Westinghouse analysis has been reviewed and accepted by the NRC for incorporation into the D.C. Cook license basis (Reference 17).

The Cook Nuclear Plant Units 1 and 2 Reduced Temperature and Pressure and Rerating LOCA hydraulic forces analysis considered a break in the largest branch line connecting to the Reactor Coolant System (RCS). In the case for the Cook Nuclear Plant units the most limiting line would be the 10-inch accumulator line. This break is considered to be most limiting based on break area and sensitivity studies which demonstrate that breaks in the cold leg produce greater peak loads than those postulated elsewhere in the RCS. In order to compare the results of the accumulator line break directly and permit the use of consistent methodologies another break of the reactor coolant loop piping was analyzed. This analysis assumed a one hundred square inch reactor pressure vessel inlet nozzle (RPVIN) break, which is greater in size than the accumulator line break and is located closer to the vessel and internals. The one-hundred square inch break size was determined to be the maximum size of a RPVIN break due to limiting displacement conditions such as pipe supports, pipe restraints, rigidity of the RCS piping, and physical barriers such as the reactor vessel cavity wall. One-hundred square inches is, therefore, the effective break opening of the reactor vessel inlet nozzle rupture. This break size was used to determine a base line data point to evaluate the sensitivity to other break sizes; it was not needed to determine the acceptability of the reactor vessel internals for a LOCA with leak-before-break acceptance criteria.

In addition to the postulated break location and area, the severity of the postulated pipe break with respect to the reactor vessel internals is a function of the decompression path through the reactor internals, the break opening time, and the operating conditions of the plant at the time of the break. The break opening time used in the analysis was a conservative one millisecond as mandated by the Nuclear Regulatory Commission (NRC) in their Topical Evaluation Report.

In order to provide a Cook Nuclear Plant specific analysis, the most limiting plant operating parameters proposed for a potential rerating for the Cook Nuclear Plant units (Table 14.3.3-1) were incorporated into the analysis, and the internals geometry, specifically the barrel baffle region former configuration, was explicitly modeled. The forcing functions were then generated for the 100 square inch RPVIN break, to establish a base case with an area characteristic of a primary loop break with current Westinghouse methodology, and for the accumulator line break, in order to establish the sensitivity of the resultant forces. As the hydraulic loads decay very quickly, only the first 0.5 second of the blowdown transient is of interest. The effects of the LOCA hydraulic loads on the internals beyond 0.5 second are considered insignificant.

The parameters chosen as most limiting incorporate a conservatively high power level (3588 MWt per Table 14.3.3-1) with respect to Cook Nuclear Plant Unit 1, as well as the upper bound primary pressure (2250 psia) and the lower bound vessel inlet temperature for the Rerating Program (511.7°F), since these are the conditions that force the maximum mass flow through the break.

14.3.3.2 Blowdown Model

The purpose of the hydraulic forces analysis is to generate the forcing functions and loads that occur on RCS components as a result of a postulated loss-of-coolant accident (LOCA). The hydraulic forcing functions and loads that occur as a result of a postulated loss-of-coolant accident are calculated through the use of the MULTIFLEX 1.0 evaluation model. The initial phase of this model makes use of the MULTIFLEX 1.0 (Reference 12) computer code to determine the transient pressures, mass velocities, and densities throughout the entire reactor coolant system (RCS) as a function of time. This is accomplished through the use of a detailed thermal-hydraulic model of the RCS. In the second phase of the analysis, the LATFORC and FORCE-2 codes (Appendices A and B of Reference 12) use the time-history values as computed by MULTIFLEX and calculate the LOCA hydraulic forces on the vessel, core-barrel and other internal components of interest. The following sections briefly describe these programs with additional details available in Reference 12.

14.3.3.3 Methodology

The MULTIFLEX 1.0 computer code calculates the thermal-hydraulic transient within the entire primary coolant system. It considers sub-cooled, transition, and two-phase (saturated) blowdown regimes, employing the method of characteristics to solve the conservation laws, assuming one-dimensionality of flow and homogeneity of the liquid-vapor mixture. With its ability to treat multiple flow branches and a large number of nodes, MULTIFLEX has the required flexibility to represent various flow passages within the primary reactor coolant system. Basically, the RCS is divided into subregions in which the fluid flows along the longitudinal axes. Each subregion is regarded as an equivalent pipe and a complex network of these equivalent pipes is then used to represent the entire primary RCS.

A coupled fluid-structure interaction is considered by accounting for the deflection of constraining boundaries, which are represented by separate spring-mass oscillator systems. For the analysis, the core support barrel is modeled as an equivalent beam (with the structural properties of the core barrel) in the plane parallel to the broken inlet nozzle. Horizontally, the barrel is divided into ten segments, with each segment consisting of three separate walls. Mass and stiffness matrices, obtained from independent modal analyses of the core barrel, are applied at each of ten mass point locations. Horizontal forces are then calculated by applying the spatial pressure variation to the wall area at each of the elevations representative on the ten mass points of the beam model. The resultant barrel motion is translated into an equivalent change in flow area in each downcomer annulus channel. At every time increment, the code iterates between the hydraulic and structural subroutines of the program at each location confined by a flexible wall.

The LATFORC computer code utilizes the MULTIFLEX generated field pressures, together with geometric vessel information (component radial and axial lengths), to determine the horizontal forces on the vessel wall, core barrel, and thermal shield. The LATFORC code represents the vessel region with a model that is consistent with the model used in the MULTIFLEX blowdown calculation. The downcomer annulus is subdivided into cylindrical segments, formed by dividing this region into circumferential and axial zones. The horizontal forces are resolved into x and y direction forces and added algebraically at each of the ten elevations. Note that the x-axis coincides with the axis of the broken loop's RPV inlet nozzle, and the positive direction is directed away from this nozzle.

The vertical hydraulic loads on the reactor vessel internals during blowdown are determined by the FORCE2 computer code utilizing a detailed geometric description of the vessel components, and the transient pressures and mass velocities computed by the MULTIFLEX code. Conservation of linear momentum forms the analytical basis for the derivation of the mathematical equations used in the FORCE2 code. In evaluating the vertical hydraulic loads on the reactor vessel internals, the following types of transient forces are considered:

- Pressure differential acting across the element.
- Flow stagnation on the element and unrecovered orifice losses across the element.
- Friction losses along the element.

These three types of forces are summed together to give the total force on each element. Individual forces on elements are further combined, depending upon what particular reactor vessel internals component is being considered, to yield the resultant vertical hydraulic load on that component.

14.3.3.4 Blowdown Evaluations - Conclusions for Rerating and Reduced Temperature

In support of the Rerating Program, LOCA Hydraulic Forcing Functions were generated for the accumulator branch line break and a 100 square inch reactor pressure vessel inlet nozzle break. The results of this analysis are summarized in Reference 13. The 100 square inch reactor pressure vessel inlet nozzle (RPVIN) break analysis was performed to represent the original design conditions of the Cook units. The original design condition, as stated in WCAP-7332-L, was a double-ended pipe break, which is a much more severe case than the 100 square inch RPVIN break.

A comparison of LOCA hydraulic forcing function for the accumulator branch line break and the 100 square inch RPVIN break shows the RPVIN break to be limiting with respect to the peak total horizontal and vertical forces.

Therefore, an evaluation of the Cook reactor internals for LOCA loads resulting from the rerating is not needed since the original design condition loads (double-ended pipe break) bound the accumulator branch line break loads. The use of the accumulator branch line break loads was allowed by the leak-before-break exemption.

14.3.3.5 Seismic Evaluation - For Rerating and Temperature Reduction

The dynamic response for the Cook RPV system will remain essentially unchanged for the parameters of the rerating program. The frequency of the components is a function of:

$$f_n = F(E/\rho)^{1/2}$$

where

F is a constant that varies from component to component

E is the Modulus of Elasticity

Therefore, the new frequency of the components is:

$$f_n * = f_n\left(\frac{E*/\rho *}{E/\rho}\right)^{1/2} = f_n\left[\left(\frac{E*/\rho *}{E/\rho}\right)\left(\frac{\rho}{\rho *}\right)\right]^{1/2}$$

where

$$\rho = \left(\rho_{metal} + \rho_{water}\right)$$

$$\rho *, f_n *, E* = \text{at rerated conditions}$$

Since the density of metal changes only slightly with temperature, the density change of the water is the only concern. It will be conservative to assume the metal density changes at the same rate as the water density with temperature.

The quantity of $\left(\frac{\rho *}{\rho}\right)\left(\frac{E}{E*}\right)$ was calculated to be equal to (1/1.03),

now:

$$f_n * = f_n\left(\frac{1}{1.03}\right)^{1/2} = .985 f_n$$

The frequency of the components will only decrease by approximately 1.5%. Therefore, the dynamic response of the Cook internals is essentially unchanged for the rerating conditions, and as a result the original seismic loads are still considered valid.

14.3.3.6 Leak-Before-Break Confirmation for Changes Due to SG Replacement Activities

As part of the effort to support taking credit for RCCA insertion for criticality control at the time the ECCS is realigned from cold leg recirculation to hot leg recirculation (Section 14.3.1 for both Units 1 and 2 for further discussion), a reanalysis was submitted to the NRC to demonstrate the continued applicability of the leak-before-break (LBB) technology for the Cook units. The NRC staff independently assessed the compliance of the reactor coolant system with the LBB criteria established in NUREG-1061, Volume 3. The NRC staff concluded in Reference 17 that the analyses and additional information submitted were sufficient to demonstrate that LBB behavior would be expected following the installation of the replacement steam generators (SGs).

Evaluations have concluded that the LBB analyses remain acceptable for the period of extended operation as described in Chapter 15 of the UFSAR.

14.3.3.7 Asymmetric Loca Loads And Mechanistic Fracture Evaluation

References (3) through (9), (11), and (14) describe work done by a Westinghouse Owners' Utility Group specifically formed to provide an analytical evaluation of the effects of certain postulated break loads on the reactor coolant system and internals as well as the NRC safety evaluations. For a discussion of the group's work and results see Section 14.3.3, Unit 2.

The response of the reactor core and vessel internals under excitation produced by a simultaneous complete severance of a reactor coolant pipe and seismic excitation for typical 4-loop plant internals has been determined. A detailed description of the analysis applicable to the Donald C. Cook Nuclear Power Plant design appears in WCAP-7332-L (Reference 2), Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation (Westinghouse Proprietary).

Effect of the Replacement Steam Generators on Unit 1

A detailed evaluation or analysis of the effect of the RSG on the core and internals integrity analysis was not performed. However, since the RSG operating parameters (i.e., pressure, temperature, and flowrate), are similar, and it has been concluded that the existing LOCA analysis is applicable to the RSGs, there is no impact by the use of the RSGs on the material presented in this section.

14.3.3.8 References for Section 14.3.3

1. Reference deleted.
2. WCAP-7332-L, "Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation," G. J. Bohm, February 1970.
3. Letter from J. A. Tillinghast, Indiana & Michigan Power Co., to E. G. Case, NRC, dated March 8, 1978.
4. Letter from J. A. Tillinghast, Indiana & Michigan Power Co., to E. G. Case, NRC, dated May 15, 1978.
5. Letter from G. P. Maloney, Indiana & Michigan Power Co., to H. R. Denton, NRC, dated September 26, 1979, AEP:NRC:0137.
6. Letter from John E. Dolan, Indiana & Michigan Electric Co., to H. R. Denton, NRC, dated December 7, 1979, AEP:NRC:00137A.

7. Letter from R. S. Hunter, Indiana & Michigan Electric Co., to H. R. Denton, NRC, dated February 15, 1980, AEP:NRC:00137B.

8. Letter from R. S. Hunter, Indiana & Michigan Electric Co., to H. R. Denton, NRC, dated October 8, 1980, AEP:NRC:0137C.

9. Letter from M. P. Alexich, Indiana & Michigan Electric Co., to H. R. Denton, NRC, dated September 10, 1984, AEP:NRC:0137D.

10. DC-D-3195-368-SC, Calculation of "Structural Analysis of Reactor Coolant Loop Piping for Replacement of Steam Generators on D.C. Cook Units 1 and 2" revision 0 (Westinghouse Calculation No. CN-SMT-99-75).

11. WCAP-9558, Rev. 2, May 1982, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack".

12. Takeuchi, K., et al., MULTIFLEX, A Fortran-IV Computer Program for Analyzing Thermal Hydraulic Structure System Dynamics," WCAP-8708-P/A (Proprietary) and WCAP-8709-A (Non-Proprietary), September 1977.

13. 13 WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," October 1988.

14. NRC Unit 2 License Amendment No. 76 and associated SER, November 22, 1985.

15. WCAP-12135, "D. C. Cook Nuclear Plant Units 1 & 2, Rerating Engineering Report," September 1989.

16. WCAP-12828, "Reactor Pressure Vessel & Internals System Evaluations for the D. C. Cook Unit 2 Vantage 5 Fuel Upgrade with IFMs," December 1990.

17. Stang, J. F. (NRC), Letter to R. P. Powers (I&M) "Issuance of Amendments – Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6473 and MA6474)," dated December 23, 1999.

18. Westinghouse WCAP-15131 revision 1 "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the D.C. Cook Units 1 and 2 Nuclear Power Plants" dated October 1999.