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

Cook Nuclear Plant Unit 2 was originally supplied with fuel by Westinghouse Electric Co. It was later refueled with replacement fuel supplied by Exxon Nuclear Company (later Advanced Nuclear Fuels Corporation [ANF] and now Siemens Nuclear Power Corporation). Most recently, Vantage 5 replacement fuel from Westinghouse is used for reload fresh fuel, and beginning with Cycle 21, all fresh fuel will be clad with **Optimized ZIRLO™** material.

This section discusses loss-of-coolant accident analyses applicable to the current Westinghouse Vantage 5 fuel.

Loss-of-coolant accidents (LOCAs) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. The Donald C. Cook Nuclear Plant Unit 2 Emergency Core Cooling System (ECCS) has been designed to mitigate the effects of postulated LOCAs by providing a sufficient amount of borated water to protect the fuel in the reactor core.

In order to assure effective long-term core cooling, certain operator actions are assumed. These actions are principally

1. to switch the ECCS from the injection phase to the recirculation phase,
2. to place the reactor coolant pumps in a condition where they can most effectively aid core cooling, and
3. to switch the ECCS from cold leg recirculation to hot leg recirculation at the appropriate time to prevent boron precipitation.

All of these items and other appropriate actions are specified in plant procedures. Long term cooling includes long-term criticality control, which is discussed in more detail in Unit 1 Section 14.3.5.

14.3.1 Large Break Loss-Of-Coolant-Accident Analyses

14.3.1.1 General

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the Donald C. Cook Nuclear Plant Unit 2, but is postulated as a conservative design basis.

When the Final Acceptance Criteria (FAC) governing the loss-of-coolant accident (LOCA) for Light Water Reactors was issued in Appendix K of 10 CFR 50.46, both the Nuclear Regulatory Commission (NRC) and the industry recognized that the stipulations of Appendix K were highly conservative. That is, using the then accepted analysis methods, the performance of the Emergency Core Cooling System (ECCS) would be conservatively underestimated, resulting in predicted Peak Cladding Temperatures (PCTs) much higher than expected. At that time, however, the degree of conservatism in the analysis could not be quantified.

As a result, the NRC began a large-scale confirmatory research program with the following objectives:

1. Identify, through separate effects and integral effects experiments, the degree of conservatism in those models permitted in the Appendix K rule. In this fashion, those areas in which a purposely prescriptive approach was used in the Appendix K rule could be quantified with additional data so that a less prescriptive future approach might be allowed.
2. Develop improved thermal-hydraulic computer codes and models so that more accurate and realistic accident analysis calculations could be performed. The purpose of this research was to develop an accurate predictive capability so that the uncertainties in the ECCS performance and the degree of conservatism with respect to the Appendix K limits could be quantified.

Since that time, the NRC and the nuclear industry have sponsored reactor safety research programs directed at meeting the above two objectives. The overall results have quantified the conservatism in the Appendix K rule for LOCA analyses and confirmed that some relaxation of the rule can be made without a loss in safety to the public. It was also found that some plants were being restricted in operating flexibility by the overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in SECY-83-472. The SECY-83-472 approach retained those features of Appendix K that were legal requirements, but permitted applicants to use best-estimate thermal-hydraulic models in their ECCS evaluation model. Thus, SECY-83-472 represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1998, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models", to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. Under the amended rules, best-estimate thermal- hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the LOCA analysis, an assessment of the uncertainty of the best-estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance criteria of 10 CFR 50.46. Further guidance for the use of best-estimate codes is provided in Regulatory Guide 1.157.

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (NUREG/CR-5249). This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three- and four-loop Pressurized Water Reactor (PWR) plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and has been approved by the NRC (WCAP-12945-P-A).

Westinghouse subsequently developed an alternative uncertainty methodology called ASTRUM, which stands for Automated Statistical TReatment of Uncertainty Method (WCAP-16009-P-A). This method is still based on the CQD methodology and follows the steps in the CSAU methodology (NUREG/CR-5249). However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing applications in WCAP-16009-P-A (WCAP-16009-P-A).

The three 10 CFR 50.46 criteria (peak cladding temperature, maximum local oxidation, and core-wide oxidation) are satisfied by running a sufficient number of WCOBRA/TRAC calculations (sample size). In particular, the statistical theory predicts that 124 calculations are required to simultaneously bound the 95th percentile values of three parameters with a 95-percent confidence level.

This analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A, as applicable to the ASTRUM methodology. Section 13-3 of WCAP-16009-P-A was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the

ASTRUM Final Safety Evaluation Report appended to this topical report. Additionally, Westinghouse analyzed the D. C. Cook Unit 2 LBLOCA using a plant-specific adaptation of the ASTRUM methodology. The analysis was performed in compliance with all of the conditions and limitations identified in NRC Safety Evaluation approving ASTRUM (WCAP-16009-P- A). The plant-specific adaptation of ASTRUM better models the downcomer region by increasing the number of circumferential noding stacks from four to twelve. This finer nodalization has been assessed against experimental data, as described in "WCOBRA/TRAC Validation with revised Downcomer Noding for D. C. Cook Unit 1 and 2", which was submitted to the NRC in Reference 9 and approved by the NRC in Reference 10.

14.3.1.2 Method of Analysis

The methods used in the application of WCOBRA/TRAC to the large break LOCA with ASTRUM are described in WCAP-12945-P-A and WCAP-16009-P-A. A detailed assessment of the computer code WCOBRA/TRAC was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the ability of the code to predict key physical phenomena in a PWR large break LOCA. Modeling of a PWR introduces additional uncertainties which are identified and quantified in the plant-specific analysis. WCOBRA/TRAC MOD7A was used for the execution of ASTRUM for D. C. Cook Unit 2 (WCAP-16009-P-A).

WCOBRA/TRAC combines two-fluid, three-field, multi-dimensional fluid equations used in the vessel with one-dimensional drift-flux equations used in the loops to allow a complete and detailed simulation of a PWR. This best-estimate computer code contains the following features:

1. Ability to model transient three-dimensional flows in different geometries inside the vessel
2. Ability to model thermal and mechanical non-equilibrium between phases
3. Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes
4. Ability to represent important reactor components such as fuel rods, steam generators, reactor coolant pumps, etc.

A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state, initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are discussed in the next section. Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood proceeds continuously, using the same computer code (WCOBRA/TRAC) and the same modeling assumptions. Containment pressure is modeled with the BREAK component using a time dependent pressure table. Containment pressure is calculated using the LOTIC code (WCAP-8354-P-A) and mass and energy releases from the WCOBRA/TRAC calculation.

The final step of the best-estimate methodology, in which all uncertainties of the LOCA parameters are accounted for to estimate a PCT, Local Maximum Oxidation (LMO), and Core-Wide Oxidation (CWO) at 95-percent probability (and 95-percent confidence level), is described in the following sections.

1. Plant Model Development:

   In this step, a WCOBRA/TRAC model of the plant is developed. A high level of noding detail is used in order to provide an accurate simulation of the transient. However, specific guidelines are followed to ensure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, except for specific areas dictated by hardware differences, such as in the upper plenum of the reactor vessel or the ECCS injection configuration.

2. Determination of Plant Operating Conditions:

   In this step, the expected or desired operating range of the plant to which the analysis applies is established. The parameters considered are based on a "key LOCA parameters" list that was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen for input as initial conditions to the plant model. A transient is run utilizing these parameters and is known as the "initial transient". Next, several confirmatory runs are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. Because certain parameters are not included in the uncertainty analysis, these parameters are set at their bounding condition. This analysis is commonly referred to as the confirmatory analysis. The most limiting input conditions, based on these confirmatory runs, are then combined into the model that will represent the limiting state for the plant, which is the starting point for the assessment of uncertainties.

3. Assessment of Uncertainty:

   The ASTRUM methodology is based on order statistics. The technical basis of the order statistics is described in Section 11 of WCAP-16009-P-A. The determination of the PCT uncertainty, LMO uncertainty, and CWO uncertainty relies on a statistical sampling technique. According to the statistical theory, 124 WCOBRA/TRAC calculations are necessary to assess against the three 10 CFR 50.46 criteria (PCT, LMO, CWO).

The uncertainty contributors are sampled randomly from their respective distributions for each of the WCOBRA/TRAC calculations. The list of uncertainty parameters, which are randomly sampled for each time in the cycle, break type (split or double-ended guillotine), and break size for the split break are also sampled as uncertainty contributors within the ASTRUM methodology.

Results from the 124 calculations are tallied by ranking the PCT from highest to lowest. A similar procedure is repeated for LMO and CWO. The highest rank of PCT, LMO, and CWO will bound 95 percent of their respective populations with 95-percent confidence level.

4. Plant Operating Range:

The plant operating range over which the uncertainty evaluation applies is defined. Depending on the results obtained in the above uncertainty evaluation, this range may be the desired range or may be narrower for some parameters to gain additional margin.

14.3.1.3 Analysis Assumptions

The expected PCT and its uncertainty developed are valid for a range of plant operating conditions. The range of variation of the operating parameters has been accounted for in the uncertainty evaluation. Table 14.3.1-1 summarizes the operating ranges for D. C. Cook Unit 2 as defined for the proposed operating conditions, which are supported by the Best-Estimate LBLOCA analysis. Tables 14.3.1-2, 14.3.1-3, and 14.3.1-7 summarize the LBLOCA containment data used for calculating containment pressure. If operation is maintained within these ranges, the LBLOCA results developed in this report are considered to be valid. Note that some of these parameters vary over their range during normal operation within a fuel cycle (e.g., accumulator temperature) and other parameters are typically fixed during normal operation within a fuel cycle (full-power Tavg).

14.3.1.4 Design Basis Accident

The D. C. Cook Unit 2 PCT and LMO-limiting transient is a cold leg split break (effective break area = 1.049 times the cold leg area) which analyzes conditions that fall within those listed in

Table 14.3.1-1. The CWO-limiting transient is a cold leg double-ended guillotine break. Traditionally, cold leg breaks have been limiting for large break LOCA. Analysis experience indicates that this break location most likely causes conditions that result in flow stagnation to occur in the core. Scoping studies with WCOBRA/TRAC have confirmed that the cold leg remains the limiting break location (WCAP-12945-P-A).

The large break LOCA transient can be divided into convenient time periods in which specific phenomena occur, such as various hot assembly heatup and cooldown transients. For a typical large break, the blowdown period can be divided into the Critical Heat Flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflood, and long-term cooling periods. Specific important transient phenomena and heat transfer regimes are discussed below, with the transient results shown in Figures 14.3.1-1A through 14.3.1-1M. (The limiting case was chosen to show a conservative representation of the response to a large break LOCA.)

1. Critical Heat Flux (CHF) Phase:

   Immediately following the cold leg rupture, the break discharge rate is subcooled and high (Figure 14.3.1-1B). The regions of the RCS with the highest initial temperatures (core, upper plenum, upper head, and hot legs) begin to flash to steam, the core flow reverses and the fuel rods begin to undergo departure from nucleate boiling (DNB). The fuel cladding rapidly heats up (Figure 14.3.1-1A) while the core power shuts down due to voiding in the core. This phase is terminated when the water in the lower plenum and downcomer begins to flash (Figures 14.3.1-1F and 14.3.1-1L, respectively). The mixture swells and intact loop pumps, still rotating in single-phase liquid, push this two-phase mixture into the core.

2. Upward Core Flow Phase:

   Heat transfer is improved as the two-phase mixture is pushed into the core. This phase may be enhanced if the pumps are not degraded, or if the break discharge rate is low due to saturated fluid conditions at the break. If pump degradation is high or the break flow is large, the cooling effect due to upward flow may not be significant. Figure 14.3.1-1C shows the void fraction for one intact loop pump and the broken loop pump. The figure shows that the intact loop remains in single-phase liquid flow for several seconds, resulting in enhanced upward core flow cooling. This phase ends as the lower plenum mass is depleted, the loop flow becomes two-phase, and the pump head degrades.

3. Downward Core Flow Phase:

   The loop flow is pushed into the vessel by the intact loop pumps and decreases as the pump flow becomes two-phase. The break flow begins to dominate and pulls flow down through the core, up the downcomer to the broken loop cold leg, and out the break. While liquid and entrained liquid flow provide core cooling, the top of core vapor flow (Figure 14.3.1-1D) best illustrates this phase of core cooling. Once the system has depressurized to the accumulator pressure (Figure 14.3.1-1E), the accumulators begin to inject relatively cold borated water into the intact cold legs (Figure 14.3.1-1I). During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is calculated to be bypassed around the downcomer and out the break. As the system pressure continues to fall, the break flow, and consequently the downward core flow (i.e. reverse flow in the fuel bundle region), is reduced. The core begins to heat up as the system pressure approaches the containment pressure and the vessel begins to fill with ECCS water (Figure 14.3.1-1K).

4. Refill Period:

   As the refill period begins, the core begins a period of heatup and the vessel begins to fill with ECCS water (Figures 14.3.1-1H, 14.3.1-1I, and 14.3.1-1J). This period is characterized by a rapid increase in cladding temperatures at all elevations due to the lack of liquid and steam flow in the core region. This period continues until the lower plenum is filled and the bottom of the core begins to reflood and entrainment begins.

5. Reflood Period:

   During the early reflood phase, the accumulators begin to empty and nitrogen enters the system. This forces water into the core, which then boils, causing system re-pressurization and the lower core region begins to quench (Figure 14.3.1-1K). During this time, core cooling may increase due to vapor generation and liquid entrainment. During the reflood period, the core flow and temperatures are oscillatory as relatively cold water periodically rewets and quenches the hot fuel cladding, which generates steam and causes system re-pressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out of the break. This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force. From the later stage of blowdown to the beginning of reflood, the accumulators rapidly discharge borated cooling water into the RCS, filling the lower plenum and contributing to the filling of the downcomer. The pumped ECCS water aids in the filling of the downcomer and subsequently supplies water to maintain a full downcomer and complete the reflood period. As the quench front progresses up the core, the PCT location moves higher into the top core region (Figure 14.3.1-1M). Please note that PCT location plot is based on the core noding (approximately one node for every 1.9" of core elevation). As the vessel continues to fill (Figure 14.3.1-1G), the PCT location is cooled and the early reflood period is terminated.

A second cladding heatup transient may occur due to excessive boiling in the downcomer. The mixing of ECCS water with hot water and steam from the core, in addition to the continued heat transfer from the vessel and its components, reduces the subcooling of ECCS water in the lower plenum and downcomer. The saturation temperature is dictated by the containment pressure. If the liquid temperature in the downcomer reaches saturation, subsequent heat transfer from the vessel and other structures will cause boiling and level swell in the downcomer (Figure 14.3.1-1L). The downcomer liquid will spill out of the broken cold leg and reduce the driving head, which can reduce the reflood rate, causing a late reflood heatup at the upper core elevations.

14.3.1.5 Post LOCA Analyses

The post LOCA analyses contained in Unit 1 Section 14.3.1.5 applies to Unit 2.

14.3.1.6 Post Analysis of Record Evaluations

In addition to the analyses presented in this section, evaluations and reanalyses may be performed as needed to address computer code errors and emergent issues, or to support plant changes. The issues or changes are evaluated, and the impact on the Peak Cladding Temperature (PCT) is determined. The resultant increase or decrease in PCT is applied to the analysis of record PCT.

Subsequent to submittal of the Unit 2 Best-Estimate LBLOCA License Amendment Request (LAR) to the NRC for review and approval, it was discovered that the LOTIC2 containment calculations (Figures 14.3.1-3 and 14.3.1-4) did not include safety injection (SI) spilled mass and energy releases in the containment backpressure calculation. A conversion error in the energy releases was also discovered (Reference 12).

The addition of the SI mass and energy releases and correction of the energy conversion error causes the LOTIC2 predicted containment backpressure to decrease, which is in the non-conservative direction for Large-Break LOCA analyses. In order to gain back margin to offset the effect of these errors in the containment backpressure calculation, the Containment Spray (CTS) temperature was increased from 45°F (Table 14.3.1-2) to 65°F which is conservative relative to the minimum CTS design temperature of 70°F. With the revised CTS temperature, the containment pressure used in the Best-Estimate LBLOCA was confirmed to be conservatively low, leading to a PCT impact of 0°F (Reference 12).

Optimized ZIRLO™ cladding has been evaluated and found to be acceptable.

The PCT, including all penalties and benefits is presented in Table 14.3.1-6 for the large break LOCA. The current PCT is demonstrated to be less than the 10 CFR 50.46(b) requirement of 2200°F.

In addition, 10 CFR 50.46 requires that licensees assess and report the effect of changes to or errors in the evaluation model used in the large break LOCA analysis. These reports constitute addenda to the analysis of record provided in the UFSAR until the overall changes become significant as defined by 10 CFR 50.46. If the assessed changes or errors in the evaluation model results in significant changes in calculated PCT, a schedule for formal reanalysis or other action as needed to show compliance will be addressed in the report to the NRC.

Finally, the criteria of 10 CFR 50.46 requires that holders and users of the evaluation models establish a number of definitions and processes for assessing changes in the models or their use. Westinghouse, in consultation with the PWR Owner's Group (PWROG), has developed an approach for compliance with the reporting requirements. This approach is documented in WCAP-13451, *Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting.* D. C. Cook intends to provide future reports required by 10 CFR 50.46 consistent with the approach described in WCAP-13451.

14.3.1.6.1 Thermal Conductivity Degradation Error Resolution

Thermal Conductivity Degradation (TCD) is a physical phenomenon in which the ability of the fuel pellet to transfer heat is reduced as burnup increases. Because of the reduced ability to transfer heat out of the pellet TCD results in higher initial steady state fuel temperatures than would otherwise be expected. The impacts of TCD on the AOR LBLOCA PCT were evaluated and it was found that for Unit 2 PCT increased by 73°F. In order to show compliance with the

10CFR50.46(b) requirement of PCT < 2200°F, it was necessary to credit conservatisms in the analysis and input parameters. This was done by modifying the input parameters documented in Table 14.3.1-1 in the following way:

- ECCS flow (Increased assumed RWST level for NPSH calculation),
- SI temperature (70°F ≤ SI Temp ≤ 100°F)
- SI delay time (17 seconds with offsite power and 28 seconds with LOOP)
- Peaking factor $F_{\Delta H}^{N}$ reduction (1.61)
- Peaking factor bum down (Reduce both F_Q and $F_{\Delta H}^{N}$ peaking factors as a function of burnup)
- Steam Generator Tube Plugging (1.5%)
- Hot full power nominal T_{ave} (574°F)
- and accumulator temperature (60°F ≤ T_{ACC} ≤ 115°F)

The benefit from crediting the conservatisms listed above resulted in a decrease in PCT of −239°F. The total impact on PCT including TCD as well as crediting conservatisms is an integrated PCT of 1941°F, which is less than the 10CFR50.46(b) requirement of 2200°F. The results of the TCD evaluation were documented in response to a 10CFR50.54(f) request, and transmitted to the NRC in letter AEP-NRC-2012-13

14.3.1.6.2 Changes to Grid Blockage Ratio and Porosity

A change in the methodology used to calculate the grid blockage ratio and porosity for 17x17 OFA fuel resulted in a change to the grid inputs used in the Unit 2 ASTRUM analysis. Grid inputs affect heat transfer in the core during a large break LOCA. The revised Grid Blockage Ratio and Porosity has been evaluated to have a 16°F penalty, as noted in Table 14.3.1-6.

14.3.1.6.3 Revised Heat Transfer Multiplier Distributions

Errors were discovered in the heat transfer multiplier distributions, including errors in the grid locations specified in the WCOBRA/TRAC models for the G2 Refill and G2 Reflood tests, and errors in processing test data used to develop the reflood heat transfer multiplier distribution. Therefore, the blow-down heat-up, blowdown cooling, refill, and reflood heat transfer multiplier distributions were redeveloped. For the reflood heat transfer multiplier development, the evaluation time windows for each set of test experimental data and each test simulation were separately defined based on the time at which the test or simulation exhibited dispersed flow film boiling heat transfer conditions characteristic of the reflood time period. The revised heat transfer multiplier distributions have been evaluated for impact and found to have a 3°F benefit for the PCT, as noted in Table 14.3.1-6.

14.3.1.6.4 HOTSPOT Burst Strain Error Correction

An error in the application of the burst strain was discovered in HOTSPOT. The outer radius of the cladding, after burst occurs, should be calculated based on the burst strain, and the inner radius of the cladding should be calculated based on the outer radius. In HOTSPOT, the burst strain is applied to the calculation of the cladding inner radius. The cladding outer radius is then calculated based on the inner radius. As such, the burst strain is incorrectly applied to the inner radius rather than the outer radius, which impacts the resulting cladding geometry at the burst elevation after burst occurs. Correction of the erroneous calculation results in thinner cladding at the burst node and more fuel relocating into the burst node, leading to an increase in the PCT at the burst node. The penalty was evaluated to have a PCT impact of 13°F as noted in Table 14.3.1-6

14.3.1.6.5 Upflow Conversion

The upflow conversion program implements a field modification of the reactor vessel lower internals assembly to reduce the potential of fuel rod failures due to baffle joint jetting. This modification changes the flow paths during normal operation as well as accident scenarios and represents a change to the BE LBLOCA licensing basis. The upflow conversion evaluation considers the new plant configuration, the upflow conversion design input changes and the effect of TCD over the life of the fuel. Relative to the original TCD evaluation (Section 14.3. 1.6.1), the offsetting input margins that were updated are maintained. The upflow conversion estimate of effect on the PCT is determined based on the difference between parametric run sets. HOTSPOT executions are performed for each WC/T case to consider the effect of local uncertainties for both IFBA and non-IFBA fuel. The upflow conversion has been determined to be a 37°F penalty for the PCT, as noted in Table 14.3.1-6.

14.3.1.7 Conclusions

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

(b)(1) The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting case is 2107°F, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Cladding Temperature less than 2200°F", is demonstrated. The results are shown in Table 14.3.1-5.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile LMO at the 95-percent confidence level. Since the resulting LMO for the limiting case is 9.7 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Local Maximum Oxidation of the cladding less than 17 percent of the total cladding thickness before oxidation", is demonstrated. The results are shown in Table 14.3.1-5.

(b)(3) The limiting core-wide oxidation corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. The limiting Hot Assembly Rod (HAR) total maximum oxidation is 0.55 percent. A detailed CWO calculation takes advantage of the core power census that includes many lower power assemblies. Because there is significant margin to the regulatory limit, the CWO value can be conservatively chosen as that calculated for the limiting HAR. A detailed CWO calculation is therefore not needed because the outcome will always be less than 0.55 percent. Since the resulting CWO is 0.55 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume", is demonstrated. The results are shown in Table 14.3.1-5.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The approved methodology (WCAP-12945-P-A) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the 44 assemblies in the low-power channel. This situation has not been calculated to occur for D. C. Cook Unit 2. Therefore, acceptance criterion (b)(4) is satisfied.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the EGGS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The manual actions that are currently in place to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology (WCAP-16009-P-A).

Based on the ASTRUM Analysis results (Table 14.3.1-5), it is concluded that D. C. Cook Unit 2 continues to maintain a margin of safety to the limits prescribed by 10 CFR 50.46. A time sequence of events for the limiting case is given in Table 14.3.1-8.

14.3.1.8 References for Section 14.3.1

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors", 10 CFR 50.46 and Appendix K of 10 CFR 50. Federal Register, Volume 39, Number 3, January 4, 1974.
2. SECY-83-472, Information Report from W.J. Dircks to the Commissioners, "Emergency Core Cooling System Analysis Methods", November 17, 1983.
3. Regulatory Guide 1.157, Best-Estimate Calculations of Emergency Core Cooling System Performance, USNRC, May 1989.
4. NUREG/CR-5249, Qualifying Reactor Safety Margins: Application of Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large Break Loss-of-Coolant-Accident, B. Boyack, et. al., 1989.
5. Bajorek, S.M., et. al., 1998, "Code Qualification Document for Best-Estimate LOCA Analysis", WCAP-12945-P-A, Volume 1, Revision 2 and Volumes 2 through 5, Revision 1.
6. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Westinghouse Proprietary}, in conjunction with Licensing Amendment Request WCOBRA/TRAC Validation with revised Downcomer Noding for D. C. Cook Unit 1 and 2," November 2007.
7. WCAP-8355, Supplement 1, May 1975, WCAP-8354 (Proprietary), "Long-Term Ice Condenser Containment LOTIC Code Supplement 1," July 1974.
8. WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting", October 1992.
9. Donald C. Cook, Unit 1 - License Amendment Request Regarding Large Break Loss-of-Coolant Accident Analysis Methodology," ADAMS Accession Number ML080090268, December 27, 2007.

10. "Donald C. Cook Nuclear Plant, Unit 1 - Issuance of Amendment to Renewed Facility Operating License Regarding use of the Westinghouse ASTRUM Large Break Loss-of-Coolant Accident Analysis Methodology," TAC MD7556, ADAMS Accession Number ML082670351, October 17, 2008.
11. "Donald C. Cook Nuclear Plant, Unit 2 (CNP-2) - Issuance of Amendment to Adopt A New Large-Break Loss-of-Coolant Accident Analysis (TAC No. ME1017)," ADAMS Accession Number ML1 10730783, March 31, 2011.
12. Westinghouse Letter NF-AE-11-53, "Evaluation Options for the LOTIC2 Safety Injection Spill Mass and Energy Error (IR # 10-218-M021)," Revision 0, May 20, 2011.
13. Good, B. F. , Allen, J. J., and Szweda, N. A., "Reactor Internals Upflow Conversion Program Engineering Report For Donald C. Cook Generation Station Unit 2," WCAP-18282-P, Rev 2, March 2018 (Proprietary).

14.3.2 Loss of Reactor Coolant From Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

14.3.2.1 Identification of Causes and Accident Description

A loss of coolant accident (LOCA) is defined as a rupture of the reactor coolant system (RCS) piping or of any line connected to the system up to the first closed valve. Ruptures of small cross section will cause loss of the coolant at a rate that can be accommodated by the charging pumps that would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown.

Should a larger break occur, depressurization of the reactor coolant system causes fluid to flow to the reactor coolant system from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the pressurizer low-pressure trip setpoint is reached. The consequences of the accident are limited in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay.
2. Injection of borated water ensures sufficient flooding of the core to prevent excessive cladding temperatures.

Before the break occurs, the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals and the vessel continues to be transferred to the reactor coolant system. The heat transfer between the reactor coolant system and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, system pressure increases and steam dumping may occur. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves. Emergency feedwater flow via the Auxiliary Feedwater (AFW) pumps is initiated on the reactor trip signal. The secondary flow aids in the reduction of reactor coolant system pressure. When the RCS depressurizes to the accumulator gas cover pressure, the accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped at the initiation of the accident and effects of pump coast down are included in the blowdown analyses.

14.3.2.2 Analysis of Effects and Consequences

14.3.2.2.1 Method of Analysis

For small breaks (less than 1.0 ft^2), the NOTRUMP (References 7, 8 and 10) digital computer code is employed to calculate the transient depressurization of the reactor coolant system as well as to describe the mass and enthalpy of the flow through the break.

14.3.2.2.2 Small Break LOCA Analysis Using NOTRUMP

The NOTRUMP and small break version of LOCTA-IV (References 7, 8 and 10) computer codes are used in the analysis of loss-of-coolant accidents due to small breaks in the RCS. The NOTRUMP computer .code is a one-dimensional general network code incorporating a number of advanced features. Among these are the calculations of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA emergency core cooling system (ECCS) evaluation model (NOTRUMP-EM) was developed to determine the RCS response to design basis small break LOCAs, and to address NRC concerns expressed in NUREG-0611 (Reference 4). The NOTRUMP-EM was modified in Reference 10 to incorporate modeling of safety injection in the broken loop and the COSI, condensation model.

The reactor coolant system model is nodalized into volumes interconnected by flow paths. The NOTRUMP code includes an option to utilize the N-loop model, which explicitly models one broken loop and each of three intact loops. A standard analysis would normally use the lumped loop model with one broken loop and one intact loop representing three intact loops. The N-loop model is used here primarily to model asymmetric safety injection. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multinode capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations. Detailed descriptions of the NOTRUMP code and the evaluation model are provided in References 7, 8 and 10.

Fuel rod heat-up calculations are performed with the LOCTA-IV (Reference 2) code using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow, and mixture heights as boundary conditions.

Description of Inputs and Initial Conditions

Figure 14.3.2-5 depicts the hot rod axial power shape used to perform the small break analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for small break LOCAs because it minimizes coolant level swell, while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The small break LOCA analysis assumes the core· continues to operate at full rated power until the control rods are completely inserted.

Safety injection systems consist of gas pressurized accumulator tanks and pumped injection systems. Minimum emergency core cooling system availability is assumed for the analysis from one Charging (CHG) pump, one High Head Safety Injection (HHSI) pump, and one residual heat removal (RHR) pump. Assumed pumped safety injection characteristics as a function of RCS pressure used as boundary conditions in the analysis are shown in Figures 14.3.2-1 through -4 and in Tables 14.3.2-4 through -7. For the break sizes less than 8.75 inches the broken loop safety injection flow is assumed to spill to RCS pressure. For the 8.75-inch break case, the broken loop safety injection flow from the RHR and HHSI pumps are assumed to spill to the containment backpressure of 0 psig and the broken loop safety injection flow from the CHG pump is assumed to spill to RCS pressure. The safety injection flow rates presented are consistent with opening the high head safety injection system cross-tie valve during ECCS injection. Safety injection is delayed 54 seconds after the occurrence of the injection signal to conservatively account for diesel generator startup and emergency power bus loading in case of a loss of offsite power coincident with an accident. During switchover from ECCS injection phase to ECCS recirculation phase the RHR flow is re-aligned to the sump, and as a result, an interruption in RHR flow for up to 5 minutes may occur. The analysis accounts for the RHR delay for break cases in which the RCS depressurizes to the RHR cut-in pressure.

The analysis supports operation for a full-power vessel average temperature range of 547.6°F to 578.1ºF (with +4.1ºF / -5.6ºF uncertainty) and nominal pressurizer pressure of 2100 psia and 2250 psia (with ±62.6 psi uncertainty). A list of input assumptions used in the analysis is provided in Table 14.3.2-1.

14.3.2.3 Results

Generic analyses using NOTRUMP (References 7 and 8) were performed and are presented in WCAP-11145 (Reference 9). Those results demonstrate that in a comparison of cold leg, hot leg and pump suction leg break locations, the cold leg break location is limiting. To insure that the worst possible small break size has been identified, calculations were performed using the NOTRUMP-EM for a spectrum of small breaks (1.5-, 2-, 3-, 4- and 6-inch equivalent diameter cold leg breaks) and includes an 8.75-inch equivalent diameter accumulator line break. The results of the small break LOCA analysis are summarized in Table 14.3.2-3, while the key transient event times are listed in Table 14.3.2-2. The limiting break was found to be a 4-inch diameter cold leg break. The maximum fuel cladding temperature attained during the transient was 1274°F.

14.3.2.3.1 Limiting Break Results

Figures 14.3.2-6 through -14 show the following for the limiting 4-inch break transient, respectively:

- Reactor Coolant System Pressure
- Core Mixture Level
- Clad Temperature at Peak Clad Temperature Elevation
- Vapor Mass Flow Rate Out of Top of Core
- Clad Surface Heat Transfer Coefficient at Peak Clad Temperature Elevation
- Fluid Temperature at Peak Clad Temperature Elevation
- Total Break Flow and Safety Injection Flow
- Total Reactor Coolant System Mass
- Top of Core Vapor Temperature

During the initial period of the small break, normal upward flow is maintained through the core and core heat is adequately removed. At the low heat generation rates following reactor trip, the fuel rods continue to be well cooled as long as the core is covered by a two-phase mixture level. Core uncovery begins during the injection phase of the transient at 864 seconds (Figure 14.3.2-7). From the clad temperature transient for the 4-inch break calculation shown in Figure 14.3.2-8, it is seen that the peak clad temperatures occurs near the time at which the core is most deeply uncovered when the top of the core is steam cooled. This time is also accompanied by the highest vapor superheating above the mixture level. From Figure 14.3.2-7 and Table 14.3.2-2 it can be seen that the core mixture level has completely recovered at 2830 seconds and continues to increase until the end of the calculated transient time. A comparison of the total break flow and safety injection flow shown in Figure 14.3.2-12 shows that at the time the transient was terminated the safety injection flow being delivered to the RCS exceeded the mass flow out the break.

14.3.2.3.2 Additional Break Cases

Summaries of the transient responses for the non-limiting break cases (1.5, 2, 3, 6 and 8.75 inches) are shown in Figures 14.3.2-15 through -31. The 1.5- and 8.75-inch breaks showed no core uncovery and the 6-inch break showed minimal uncovery; therefore, fuel rod heat-up calculations were not performed for these three cases. The plots for each of the additional non-limiting break cases include:

- Reactor Coolant System Pressure
- Core Mixture Level
- Top of Core Vapor Temperature
- Clad Temperature at Peak Clad Temperature Elevation (2 and 3 inch cases only)

The fuel rod heat-up results for each of the additional breaks considered, as seen in Table 14.3.2-3, are less than the limiting 4-inch break case (Note: the 3-inch break has equivalent transient oxidation as the 4-inch break).

14.3.2.4 Conclusions

The small break LOCA analysis considered a spectrum of cold leg breaks of 1.5-, 2-, 3-, 4-, 6- and 8.75-inch diameters. The analysis resulted in the limiting PCT of 1274°F for the 4-inch break and a maximum local transient oxidation of 0.11% calculated at beginning of life (BOL) for the 3- and 4-inch breaks. The analysis is applicable to core power up to and including 3612 MWt (3600 MWt plus 0.34% uncertainty) with both the HHSI and RHR cross-tie valves open during the injection phase and with HHSI cross-tie valves open and RHR cross-tie valves closed during the recirculation phase.

The analysis presented herein shows that the accumulator and SI subsystems of the ECCS, together with the heat removal capability of the steam generators, provide sufficient core heat removal capability to maintain the calculated PCT for small break LOCA below the required limit of 10 CFR 50.46 (Reference 1). Furthermore, the analysis shows that the local cladding oxidation and core wide average oxidation, including consideration of pre-existing and post-LOCA oxidation, are less than the 10 CFR 50.46 (Reference 1) limits. Note that the core wide average oxidation results illustrate that the total hydrogen generation is less than 1%.

Table 14.3.2-3 provides the summary of the results for the small break LOCA analysis including PCT, maximum local transient oxidation and total hydrogen generation.

14.3.2.5 Additional Evaluations

14.3.2.5.1 Optimized ZIRLO™ Cladding

Optimized ZIRLO™ has been evaluated and found to be acceptable.

14.3.2.5.2 Supplemental Calculations to Support Upflow Conversion

SBLOCA calculations using the NOTRUMP-EM (described in Section 14.3.2.2) were performed to determine the effect of converting the barrel/baffle region from a downflow configuration to an upflow configuration. In order to assess the impact of implementing the Upflow Conversion, the impact on the limiting 4-inch break is assessed. The key transient event times are listed in Table 14.3.2-2a, while the results of the evaluation are summarized in Table 14.3.2-3a. The upflow conversion has been determined to be a +75°F penalty for the PCT as noted ln Table 14.3.2-8. The maximum fuel cladding temperature attained during the transient was 1348.7°F.

Plots of the fol1owing parameters are shown in Figures 14.3.2-32a through 14.3.2-32d for the limiting 4-inch break transient.

- Reactor Coolant System Pressure
- Core Mixture Level
- Top of Core Vapor Temperature
- Clad Temperature at Peak Clad Temperature Elevation

The results of the Upflow conversion calculations show that the requirements of 10 CFR 50.46 are still met: total oxidation is less than 17%, core wide oxidation is less than 1%, and peak cladding temperature is less than 2200°F.

14.3.2.6 References for Section 14.3.2

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," 10 CFR 50.46, August 2007 and "ECCS Evaluation Models," Appendix K of 10 CFR 50, June 2000.

2. Bordelon, F. M., et al., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301 (Proprietary), June 1974.

3. Not Used.

4. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, January 1980.

5. Not Used.

6. Not Used.

7. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary), August 1985.

8. Lee, N., et al, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary), August 1985.

9. Rupprecht, S. D., et al, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A (Proprietary), October 1986.

10. Thompson, C. M. et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety 1njection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary), July 1997.

14.3.3 Core and Internals Integrity Analysis

The original material in this section was filed for both Units 1 and 2 when it was thought that both units would be identical. This original material can be found in Appendix 14G of the Unit 1 updated FSAR. Sections 3.1 through 3.4 of the Unit 2 UFSAR incorporate the change to Vantage 5 fuel. Section 3.2.2 describes the design basis, design loading conditions, design loading categories, design criteria basis, and vibration considerations for the reactor vessel internals. In addition, additional studies in this area were performed to support the Vantage 5 fuel upgrade. The results of these studies, which are now completed, are discussed in the following paragraphs.

The analyses that were performed to support the rerating and temperature reduction for Donald C. Cook Nuclear Plant Units 1 and 2 is discussed in Section 14.3.3 of the Unit 1 FSAR. The evaluations that were performed to support the Donald C. Cook Nuclear Plant Unit 2 Vantage 5 fuel upgrade with Intermediate Flow Mixing grids (IFM) are discussed below. Reloading a reactor with fuel other than that for which a plant was originally designed, and operating a plant at conditions other than those considered in the original design, require that the reactor vessel system/fuel interface be thoroughly addressed in order to assure compatibility of the replacement fuel/core and to assure that the structural integrity of the reactor vessel system is not adversely affected. In addition, thermal-hydraulic analyses are required to determine plant specific core bypass flows, pressure drops, and upper head temperature in order to provide input to emergency core cooling system (ECCS) and non-LOCA accident analyses, as well as Nuclear Steam Supply System (NSSS) performance evaluations. Generally, the areas of concern most affected by changes in fuel design and system operating conditions are as follows:

- reactor hydraulic characteristics
- baffle gap flow leakage
- reactor vessel and internals structural integrity
- rod control cluster assembly (RCCA) scram performance (which is affected by a fuel change but not a rerating)

14.3.3.1 LOCA Evaluations

The finite element models shown in figures 14.3.3-1 through 14.3.3-5 were used to perform the LOCA analysis. Since Cook Nuclear Plant Unit 2 has leak-before-break (LBB) exemption, the

LOCA analyses due to main line breaks for the reactor pressure vessel system are not required. The next limiting breaks to be considered are the branch line breaks, which consist of

a. accumulator line,

b. pressurizer surge line, and

c. residual heat removal (RHR) line.

Of these branch line breaks, the most limiting break considered for the dynamic analysis of Donald C. Cook Nuclear Plant Unit 2 reactor pressure vessel system was the accumulator line (cold leg) break.

Following a postulated LOCA pipe rupture, forces are imposed on the reactor vessel and its internals. These forces result from the release of the pressurized primary system coolant, and - for guillotine pipe breaks - from the disturbance of the mechanical equilibrium in the piping system prior to the rupture. The release of pressurized coolant results in traveling depressurization waves in the primary system. These depressurization waves are characterized by a wavefront with low pressure on one side and high pressure on the other. The wavefront translates and reflects throughout the primary system until the system is completely depressurized. The rapid depressurization results in transient hydraulic loads on the mechanical equipment of the system.

The LOCA loads applied to the Cook Nuclear Plant Unit 2 reactor pressure vessel system consist of

1. reactor internal hydraulic loads (vertical and horizontal), and

2. reactor coolant loop mechanical loads.

All the loads are calculated individually and combined in a time-history manner.

14.3.3.2 Reactor Pressure Vessel (RPV) Internal Hydraulic Loads

Depressurization waves propagate from the postulated break location into the reactor vessel through either a cold leg or a hot leg nozzle. Figures 14.3.3.7 and 14.3.3.8 depict the possible wave propagation paths for waves entering from the RPV cold leg or hot leg, respectively. The following paragraphs describe the depressurization wave path in the reactor vessel for a break in either the cold leg or hot leg piping of the reactor coolant system. After a postulated break in the reactor inlet pipe, the depressurization path for waves entering the reactor vessel is through the nozzle which contains the broken pipe and into the region between the core barrel and reactor vessel (Figure 14.3.3-7). This region is called the downcomer annulus. The initial waves propagate up, around, and down the downcomer annulus, then up through the region circumferentially enclosed by the core barrel; that is, the fuel region.

The region of the downcomer annulus close to the break depressurizes rapidly but, because of restricted flow areas and finite wave speed (approximately 3,000 feet per second), the opposite side of the core barrel remains at a high pressure. This results in a net horizontal force on the core barrel and reactor pressure vessel (RPV). As the depressurization wave propagates around the downcomer annulus and up through the core, the barrel differential pressure reduces, and similarly, the resulting hydraulic forces drop.

In the case of a postulated break in the reactor outlet pipe, the waves follow a dissimilar depressurization path, passing through the outlet nozzle and directly into the upper internals region, depressurizing the core and entering the downcomer annulus from the bottom exit of the core barrel (Figure 14.3.3-8). Thus, after an RPV outlet nozzle break, the downcomer annulus would be depressurized with very little difference in pressure across the outside diameter of the core barrel.

A hot leg break produces less horizontal force because the depressurization wave travels directly to the inside of the core barrel (so that the downcomer annulus is not directly involved) and internal differential pressures are not as large as for a cold leg break. Since the differential pressure is less for a hot leg break, the horizontal force applied to the core barrel is less for a hot leg break than for a cold leg break. For breaks in both the hot leg and cold leg, the depressurization waves would continue to propagate by reflection and translation through the reactor vessel and loops.

The MULTIFLEX computer code calculates the hydraulic transients within the entire primary coolant system. It considers subcooled, transition, and two-phase (saturated) blowdown regimes. The MULTIFLEX program employs the method of characteristics to solve the conservation laws, and assumes one-dimensionality of flow and homogeneity of the liquid-vapor mixture.

The MULTIFLEX code considers a coupled fluid-structure interaction by accounting for the deflection of constraining boundaries, which are represented by separate spring-mass oscillator systems. A beam model of the core support barrel has been developed from the structural properties of the core barrel; in this model, the cylindrical barrel is vertically divided into various segments and the pressure as well as the wall motions are projected onto the plane parallel to the broken inlet nozzle. Horizontally, the barrel is divided into 10 segments; each segment consists of 3 separate walls. The spatial pressure variation at each time step is transformed into 10 horizontal forces, which act on the 10 mass points of the beam model. Each flexible wall is bounded on either side by a hydraulic flow path. The motion of the flexible walls is determined by solving the global equations of motion for the masses representing the forced vibration of an undamped beam.

14.3.3.3 Reactor Coolant Loop Mechanical Loads

The reactor coolant loop mechanical loads are applied to the RPV nozzles by the primary coolant loop piping. The loop mechanical loads result from the release of normal operating forces present in the pipe prior to the separation as well as transient hydraulic forces in the reactor coolant system. The magnitudes of the loop release forces are determined by performing a reactor coolant loop analysis for normal operating loads (pressure, thermal, and deadweight). The loads existing in the pipe at the postulated break location are calculated and are "released" at the initiation of the LOCA transient by application of the loads to the broken piping ends. These forces are applied with a ramp time of 1 millisecond because of the assumed instantaneous break opening time.

14.3.3.4 Seismic Evaluations

The non-linear dynamic seismic analysis of the reactor pressure vessel system includes the development of the system finite element model and the synthesized time history accelerations. Both of these developments for the time history seismic analysis are discussed in the following sections.

Mathematical Model of the Reactor Pressure Vessel

The mathematical model of the RPV is a three-dimensional nonlinear finite element model which represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. The model was developed using the WECAN computer code. The loop layout and model global coordinate system are shown in Figure 14.3.3-1. The model consists of three concentric structural submodels connected by nonlinear impact elements and stiffness matrices. The first submodel, shown in Figure 14.3.3-2 represents the reactor vessel shell and associated components. The reactor vessel support system is shown in Figure 14.3.3-3.

The second submodel, shown in Figure 14.3.3-4, represents the reactor core barrel, thermal shield, lower support plate, tie plates, and secondary core support components. This submodel is physically located inside the first, and is connected to it by a stiffness matrix at the internals support ledge. Core barrel to vessel shell impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radial support locations.

The third and innermost submodel, shown in Figure 14.3.3-5, represents the upper support plate, guide tubes, support columns, upper and lower core plates, and fuel. The third submodel is connected to the first and second by stiffness matrices and nonlinear elements.

Fluid-structure or hydro-elastic interaction is included in the reactor pressure vessel model for seismic evaluation. The horizontal hydro-elastic interaction is significant in the cylindrical fluid flow region between the core barrel and thermal shield and between the thermal shield and reactor vessel (the downcomer). Mass matrices with off-diagonal terms (horizontal degrees-of-freedom only) attach between nodes on the shells. The mass matrices are for the hydro-elastic interaction of two concentric cylinders as developed in Reference 10.

The matrices are a function of the properties of two cylinders with a fluid in the cylindrical annulus, specifically; inside and outside radius of the annulus, density of the fluid and length of the cylinders. Vertical segmentation of the reactor core barrel (RCB) allows inclusion of radii variations along the RCB height and approximates the effects of RCB beam deformation. These mass matrices were inserted between selected nodes on the core barrel and reactor vessel shell as shown in figure 14.3.3-6.

The WECAN computer code, which is used to determine the response of the reactor vessel and its internals, is a general purpose finite element code. In the finite element approach, the structure is divided into a finite number of members or elements. The inertia and stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Employing appropriate transformation, the element global matrices and arrays are then computed. Finally, the global element matrices and arrays are assembled into the global structural matrices and arrays, and used for dynamic solution of the differential equation of motion for the structure. WECAN solves the differential equation by using nonlinear modal superpositioning techniques.

For a time history response of the reactor pressure vessel and its internals under seismic excitations, synthesized time history accelerations are required. The synthesized time history accelerations used in Donald C. Cook Nuclear Plant Unit 2 RPV system analysis were based on the 'design' spectra in Reference 15. The time history accelerations were developed using DEBLIN2 Computer Code, Reference 11. In DEBLIN2, the spectrum amplification and suppression techniques are used to modify the initial transients supplied as input to the code as described in Reference 12. The records of a real earthquake, Taft, are the basis for the synthesized time history accelerations. The spectral characteristics of the synthesized time histories are similar to the original 'Taft' earthquake records. The spectrum ordinates are computed using suggested frequency intervals given in Regulatory Guide 1.122 (Reference 13). The spectra corresponding to the synthesized time history motions meet the acceptance criteria given in Standard Review Plan (SRP) 3.7.1 (Reference 14). Note that the input excitations which were developed are for ten (10) second long seismic events.

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

See the discussion, which is applicable to both units, that is presented in Section 14.3.3 of Unit 1 UFSAR.

14.3.3.5 Summary

The evaluation for the reactor internals combines the results of the LOCA and seismic evaluations to determine the combined loadings/stresses on the reactor internals. The results of the LOCA and seismic evaluation performed for the Donald C. Cook Nuclear Plant Unit 2 Replacement Reactor Vessel Head (RVCH), associated service structure enhancements, and Vantage 5 fuel upgrade with IFMs, concludes that the original design condition double-ended pipe break and seismic results are not exceeded. Therefore, the original design stresses and displacements for the reactor vessel internals are still applicable.

14.3.3.6 Asymmetric LOCA Loads and Mechanistic Fracture Evaluation

References (1), (2), (3), (4), (5), (6) and (7) discuss work done by a Westinghouse Owners' Group specifically formed to provide an analytical evaluation of the effects of certain postulated break loads on the reactor coolant system and internals. The evaluation program, which is now completed, was divided into three phases.

Phase "A" included data acquisition from the utilities, and review of structural and hydraulic parameters for potential grouping among generically similar plants. This phase was completed in July 1979, Phase "B" (Pipe Breaks Outside Reactor Cavity) consists of evaluation of structural integrity of the NSSS component supports for breaks outside the reactor cavity and development of specific plant qualification programs as required. Phase "B" also included work required as input for reactor vessel evaluations to be performed in Phase "C" (pipe breaks inside reactor cavity) and initiation of mechanistic pipe break analyses.

Westinghouse has completed the above evaluations and issued the following detailed reports:

1. "Westinghouse Owners' Group Asymmetric LOCA Loads Evaluation."
   Phase B: WCAP-9628 (Proprietary), WCAP-9662 (Non-Proprietary)
   Phase C: WCAP-9748 (Proprietary), WCAP-9749 (Non-Proprietary)

2. "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack."
   WCAP-9558 Rev. 2, May 1981 (Proprietary), WCAP-9570 (Non-Proprietary

3. "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation."
   WCAP 9787 (May 1981) (Proprietary)

4. Letter Report NS-ERP-2519, E.P. Rahe to D. G. Eisenhut (November 10, 1981), Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Components During the Westinghouse Presentation on September 25, 1981.

For the Donald C. Cook Nuclear Plant Units 1 and 2, the Phase "B" analysis concluded that the maximum stresses in the components and their supports were within the allowable stress limits and thereby the structural and operational integrity of the system components is assured and no modifications to the existing design are required.

Phase "C" (pipe breaks inside reactor cavity) involved the evaluation of the structural integrity of the NSSS components and supports, ECCS piping, fuel, internals and the CRDM, for breaks near the reactor vessel inlet nozzles. The evaluations performed in Phase "C" incorporate, as part of the analysis assumptions, the presence of break - limiting devices around the inlet nozzles, inside the concrete primary shield wall penetration. The assumption was that these restraints will limit the pipe break to less than or equal to 144 square inches. Westinghouse has completed a detailed evaluation of the NSSS components and their supports and a qualitative evaluation of the ECCS piping system in phase C of the evaluation program.

For both Units of the Donald C. Cook Nuclear Plant, this evaluation demonstrates the capability of the NSS System to withstand the effects of the postulated reactor vessel nozzle rupture loads. Additionally, it has been demonstrated that the appropriate systems and components will maintain their functional capability and insure a safe plant shutdown during the postulated design accident condition.

The primary shield wall structure was also evaluated by Westinghouse to account for the differential pressurization of the reactor cavity and was found capable of resisting the overturning and shearing forces. This completed the analytical evaluation of the effects of certain postulated break loads on the Reactor Coolant System and internals.

Concurrent with the Phase "B" and "C" evaluations, Westinghouse conducted experimental and analytical investigations to determine the need to include a guillotine rupture of the reactor coolant piping as a reasonable design basis for their plants. The mechanistic pipe break study of Phase "B" involved an analysis and test program to understand the fracture mechanics characteristics of a pipe break. Both aspects of this effort are now complete. This study showed that a through-wall thickness flaw existing in the pipe material (which is easily detectable by ultrasonic testing) will also have a limited crack growth under combined operating plus earthquake (SSE) loads; crack propagation will not result in a full size pipe break; the crack will remain stable. Analytical studies were also made to evaluate the effect of a crack suddenly appearing in the pipe. This analysis was done for both wrought and cast materials. In both cases the effect was minimal and there exists no danger of producing a double-ended circumferential break. The analytical and experimental work demonstrates

a. that flaws large enough to cause significant asymmetric loads will not occur,

b. that large margins against unstable crack extension exist for stainless steel PWR primary piping postulated to have large flaws and subjected to postulated safe shutdown earthquake and other plant loading, and that

c. flaws can be detected prior to unstable flaw growth leading to a double-ended circumferential break.

Adequate leakage detection systems are available to detect the primary coolant loop leakage.

The overall conclusion of this mechanistic fracture investigation is that under the worst combination of loadings, including the effects of the safe shutdown earthquake, a realistically postulated flaw will not propagate around the circumference of the pipe and cause a guillotine break. Any postulated flaw that might exist can be detected based on the leak criteria. The "leak-before-break" is thus considered a realistic criterion for the evaluation of the NSS System, and thus, it was concluded that pipe whip restraints and other protective measures against the dynamic effects of a break in the main coolant piping, as assumed in the phase C analysis, are not required.

The NRC reviewed the above-noted reports and issued Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops" dated February 1, 1984. This Generic Letter and its attachment documents NRC acceptance of the Westinghouse Owners Group evaluations and the basis for the leak-before-break criterion, and NRC is applicable to the plants noted in the Generic Letter. Donald C. Cook Nuclear Plant is one of the plants listed in the Generic Letter. These evaluations satisfactorily addressed unresolved safety issue A-2.

Based on the safety evaluation noted in Generic Letter 84-04, we submitted a plant specific evaluation (Reference 7). The resolution of unresolved Safety Issue A-2 and the NRC's acceptance of the leak-before-break criterion satisfactorily addressed license condition 2.c.(3)(a) on "Analysis of Reactor Vessel Supports and Internals." We therefore requested a license amendment to delete the license condition, and we requested exemption from the requirements of General Design Criterion No. 4.

Upon completion of the staff's review of the above plant-specific evaluation, the NRC issued license Amendment No. 76 to the Unit No. 2 operating license (dated November 22, 1985). The SER attached to the amendment noted that:

> The Westinghouse evaluations and the NRC SER noted in Generic Letter 84-04 are directly applicable to Donald C. Cook Nuclear Plant, and pipe whip restraints and other protective measures against the dynamic effects of a break in the main coolant piping are not required.
>
> The Westinghouse plant specific evaluations submitted by the Westinghouse Utility Group I&M Electric Company meet the requirements of the proposed rule on modifications of Criterion 4 of Appendix A to Part 50 that was published in the Federal Register (50 FR 27006) dated July 1, 1985.
>
> The evaluation submitted satisfactorily addresses the issue noted in the NRC unresolved safety issue USI A-2 and responds to the license condition 2.c.3.(a) and therefore the subject license condition is deleted.
>
> The leak detection system at Cook Plant meets the leak-before-break criterion established for leak detection systems i.e., one gpm in four hours.

The details of the NRC safety evaluations to

1. the Westinghouse Owners Group submittal and

2. the D. C. Cook Nuclear Plant specific submittal can be found in references (8) and (9), respectively.

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 References for Section 14.3.3

1. Letter from J. A. Tillinghast, Indiana & Michigan Power Co. to E. G. Case, NRC, dated March 8, 1978.
2. Letter from J. A. Tillinghast, Indiana & Michigan Power Co. to E. G. Case, NRC, dated May 15, 1978.
3. Letter from G. P. Maloney, Indiana & Michigan Power Co. to H. R. Denton, NRC, dated September 26, 1979, AEP:NRC:0137.
4. Letter from John E. Dolan, Indiana & Michigan Electric Co. to H. R. Denton, NRC, dated December 7, 1979, AEP:NRC:00137A.
5. Letter from R. S. Hunter, Indiana & Michigan Electric Co. to H. R. Denton, NRC, dated February 15, 1980, AEP:NRC:0137B.
6. Letter from R. S. Hunter, Indiana & Michigan Electric Co. to H. R. Denton, NRC, dated October 8, 1980, AEP:NRC:0137C.
7. Letter from M. P. Alexich, Indiana & Michigan Electric Co. to H. R. Denton, NRC dated September 10, 1984, AEP:NRC:0137D.
8. NRC Generic Letter 84-04, "Safe Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," Dated February 1, 1984. (See AEP:NRC:0137D)
9. Amendment No. 76 to Operating License No. DRR-74 Donald C. Cook Nuclear Plant No. 2 and attached SER dated November 22, 1985. (See AEP:NRC:0137G.)
10. Fritz, R. J., "The Effect of Liquids on the Dynamic Motions of Immersed Solids," Trans. ASME, Journal of Engineering for Industry, February 1972, pp 167-173.

11. Lin, C. W., "DEBLIN2 - A Computer Code to Synthesize Earthquake Acceleration Time Histories," WCAP-8867, November 1976 (Westinghouse Proprietary).

12. Tsai, Nien-Chiem, "Spectrum - Compatible Motions for Design Purposes," ASCE Journal of Engr. Mech. Div. 93, April 1972, pp. 345-356 (No. Em2).

13. U. S. Nuclear Regulatory Commission Office of Standard Development, Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," September 1976.

14. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, NUREG-75/087, Section 3.7.1, Seismic Impact, November 1975.

15. AEPSC Letter AEP2-W/0049: September 12, 1989.

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

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

18. "WECAN Users' Manual," Rev. W (Westinghouse Proprietary)