## 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 95$^{th}$ 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 $F^N_{\Delta H}$ reduction (1.61)
- Peaking factor bum down (Reduce both $F_Q$ and $F^N_{\Delta H}$ peaking factors as a function of burnup)
- Steam Generator Tube Plugging (1%)
- 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

### 14.3.1.7 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