February 2, 2001

MEMORANDUM TO: File

FROM: John F. Stang, Senior Project Manager, Section 1 ***/RA/***
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ACCEPTANCE REVIEW REGARDING LICENSE AMENDMENT REQUEST, “CONTROL ROOM HABITABILITY AND RESPONSE TO GENERIC LETTER (GL) 99-02 - LABORATORY TESTING OF NUCLEAR-GRADE ACTIVATED CHARCOAL,” DATED JUNE 12, 2000 (TAC NOS. MA9394 AND MA9395)

During the review of the subject proposed license amendment and response to GL 99-02, the staff determined additional information was necessary to complete its review. Attached is the draft request for additional information (RAI). In accordance with Nuclear Reactor Regulation (NRR) Office Letter 803, the draft RAI will be E-Mailed to the licensee and a conference call will be arranged to discuss the RAI. Once the Nuclear Regulatory Commission (NRC) staff and the licensee have a common understanding of the information required, the RAI will be issued formally to the licensee.

Docket Nos. 50-315 and 50-316

Attachment: As Stated

ACCEPTANCE REVIEW FOR
D. C. COOK UNITS 1 AND 2
SUBMITTAL C06000-13 CONTROL ROOM HABITABILITY , DATED JUNE 12, 2000

1. In numerous locations, your submittal references NUREG-1465 and Draft Guide-1081 as basis for your submittal. Please provide a commitment to the applicable provisions of Regulatory Guide (RG) 1.183, in lieu of the NUREG-1465 and DG-1081 referenced in your submittal, identifying proposed alternatives, if any, for staff consideration.

   (The staff used some information from NUREG-1465 as part of the basis for the development of the regulatory guidance in DG-1081 and the final RG 1.183. However, the staff has not endorsed NUREG-1465 for use by currently licensed power reactors since NUREG-1465 is not specifically applicable to currently licensed power reactors, especially those with fuel burnups in excess of 40 GWD/MTU. It is the staff's intent that the guidance of RG 1.183 be used by licensees in preparing their initial application under 10 CFR 50.67 and that guidance, less any approved alternatives, would become the facility's AST design basis.)

2. DG-1081 was published for public comment in December 1999, and the final guide RG-1.183 was issued in July 2000. Your submittal was dated June 2000. In addressing the public comments and preparing the final guide, several analysis assumptions in DG-1081 were revised. As such, some assumptions identified in your submittal differ from those deemed acceptable in RG 1.183. For many of these differences, the staff believes that your submitted analyses could be shown to be bounding using the outdated assumption, and as such, it may be possible to incorporate the updated assumption in your design basis without resubmitting the analysis. Please compare your analysis assumptions against those provided in RG 1.183 and indicate your intent to either update the assumption or retain the assumption as a proposed alternative to RG 1.183. Provide a justification for each such proposed alternative.

3. Your analyses incorporated revised atmospheric dispersion (X/Q) values calculated using the ARCON96 computer code. The staff considers this to be a change in analysis methodology requiring staff approval. Please provide sufficient information for the staff to evaluate the acceptability of your X/Q values. The information should include:

   a. Confirmation that the meteorological data input to ARCON96 was collected by the site's meteorological instrumentation as described in the updated final safety analysis report (UFSAR) or T/S and subject to 10 CFR Part 50, Appendix B quality assurance requirements.

   b. Unit 1 and Unit 2 release point and receptor configuration information (e.g., height, velocity, distances, direction, etc.), release mode (e.g., ground, elevated, surface), and meteorological sensor configuration, as input to ARCON96.

   c. A floppy disk containing the meteorological data input to ARCON96, in the ARCON96 input data format.

4. Your analyses incorporated an iodine flashing fraction of 10^{-4} for emergency core cooling system (ECCS) leakage, contrary to the default 10^{-1} assumption provided in RG 1.183. On Pages 5 and 6 of Attachment 1 to your submittal, you attempted to justify

these assumptions on an experiment reported in your existing final safety analysis report (FSAR), and on theoretical iodine partitioning of 10^{-8}. The staff does not believe that the provided justification supports the use of 10^{-4} for the ECCS flash fraction. Based on the description of the experiment, the staff questions whether the experimental drying to evaporation can appropriately model leakage that could be sprayed from the leakage paths, or as droplets fall through air and impinge on nearby surfaces. The staff also questions how well Eggleton's mathematical treatment of steady state vapor partial pressures between the gas and liquid phases can adequately model the more dynamic situation associated with leakage from pressurized systems as is the case here. Your submittal quoted partitioning of 10^{-8} which appears to be at odds with the abstract for Eggleton work which reports partitioning values ranging from 0.012 at high iodine concentrations and low pH to less than 0.0001 at high pH and low iodine concentrations. Please provide additional justification, including consideration of sump pH and area ventilation rates and iodine entrainment in evaporated vapor, in support of your assumption.

5. Your analyses addresses a small break loss-of-coolant accident (LOCA) event in which containment sprays do not start or are terminated early. Page 11 of 30 of DIT-B-00069-06 contains a note that states:

   > Per DG-1081 Appendix A, gap fractions from Table 3 can be used for small-break loss-of-coolant accident (SBLOCA) if no fuel melt is projected.

   While this provision may have been present in a pre-decisional version of the draft guide, this provision was not included in the draft guide published for public comment in December 1999, nor in the final regulatory guide published in July 2000. While the staff agrees with the conclusion that the fuel damage could be less than that assumed for a large-break LOCA, the staff expects the licensee to provide a technical justification for the amount of fuel damage being assumed. Please provide an acceptable basis for this conclusion. See §3.6 of RG 1.183.

6. On Page 7 of Attachment 1, you note your conclusion that the assumption of a constant break flow for 30 minutes is more limiting than using the actual operator response times. Although this assumption may be valid with regard to mass of reactor coolant system (RCS) transferred to the secondary, what is the sensitivity of other analysis parameters to delays in operator actions, such as break flow flashing fraction, steam release from the affected steam generator, tube uncovery? The staff is concerned that these other parameters, and the time-dependent buildup of RCS activity due to iodine spiking, could negate the apparent conservatism in the RCS mass transferred. Please confirm your conclusion relative to the postulated dose to the control room operators. Please explain how your amendment request dated October 24, 2000, on steam generator tube rupture (SGTR) analysis methodology affects this control room amendment request.

7. Contrary to the guidance of RG 1.183, in some of your analyses you have assumed an iodine spike duration of 6 hours based on the depletion of the 12 percent iodine gap inventory. The iodine spiking phenomenon is generally understood to be the result of RCS liquid flushing out suspended iodine salts from the fuel rod via pin hole leakage. The transfer of iodine from the pellet to the plenum region is dependent, in part, on

partial pressures of iodine in the gap and the pellet. In light of these considerations, please explain why basing your assumption on the gap inventory alone is appropriate.

8. Your analyses generally assume that it takes 30 days to cool down the plant via steam generator steam releases to reach conditions at which residual heat removal (RHR) can be initiated. Given a minimal cool down rate of 50 degrees/hour, the cooldown could be completed in 10-12 hours. The staff recognizes that the 30-day assumption is likely conservative, but an explanation of this assumption is requested.

9. §3.1.1 of Attachment 6, identifies the assumption that 3 percent of the gap activity is released from 30 seconds to 90 seconds and the remaining 2 percent of the gap is released over the next 28.5 minutes. RG 1.183 (and DG-1081) provided that the activity would be released from the core in a linear fashion over the duration of the release phase, or as an alternative, released instantaneously at the start of the particular release phase. Please provide a justification for this proposed alternative from RG 1.183.

10. §3.1.4 of Attachment 6, identifies that the sedimentation removal coefficient is conservatively assumed to be only 0.1 hr^{-1} and that sedimentation does not continue beyond a DF of 1000. Please justify the conservatism of these two assumptions against the DF factors presented in Table 20 of NUREG/CR-6189, "A simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," and the effective decontamination coefficients presented in Table 24 of the same document.

11. For the analyses that have credited iodine partitioning in the steam generators, was the impact of steam generator tube uncovery during the transient considered? Was this considered in determining the flash fraction? If not, why not?

12. §7.1.1 of Attachment 6, identifies the fuel handling accident decay time as 100 hours, Item F1 in DIT-B-00069-06 identifies the decay period as 168 hours. Please indicate which is the correct value.

13. There are discrepancies regarding the pool iodine decontamination factor in §7.1.3 of Attachment 6, and in item F6 in DIT-B-00069-06. Please resolve these differences and confirm your intent to limit iodine DF to an overall effective DF of 200, as identified in RG 1.183 (and DG-1081 as published for comment).

14. The 3rd and 4th paragraphs on page 27 of Attachment 6, appear to be addressing the same plant response but with different nomenclature. As we understand the system operation, the control room ventilation systems re-align on a safety injection signal, not a containment isolation signal as implied in the 3rd paragraph. Please confirm that the control room re-alignment occurs on an safety injection (SI) signal (e.g., low PZR pressure, low steamline pressure, high containment pressure, etc.).

15. Please explain the noble gas concentration adjustment described in the footnote to Table 5.

16. Items L43 and L44 in DIT-B-00069-06 identifies spray coverage for the three regions in the containment. This parameter was not addressed in the Attachment 6 discussion

and was not tabulated in Table 11 of Attachment 6. Please describe how the spray coverage was incorporated into the analysis.

17. The staff believes that the third commitment in Attachment 9 is in error when it references verification of the control room X/Q values. The Westinghouse letters in Attachment 6 indicate that it is the offsite X/Q values that need verification. Please confirm that the control room X/Q values used in the analyses performed in support of this amendment meet applicable quality assurance standards.

18. The staff has reviewed the information in Attachment 7 to your submittal. Item 6 on page 3 of this attachment addressed an issue related to design controls on changes made in the control room flow rates between 1982 and 1986, and whether or not the consequences of these changes were adequately evaluated. While your current re-analyses using the AST demonstrate compliance with GDC-19 (as revised in late 1999) this conclusion may not be applicable to the issue cited in 1986 since the source term and acceptance criterion were different. The staff expects to approve the current amendment request without accepting this item.

19. There appear to be discrepancies regarding the assumed volumes of the various regions of the containment between the text of Attachment 6, Table 11 of Attachment 6, and items L5 and L43 of DIT-B-00069-06. For example, the volume of the upper containment is given as 990,000 ft3 in Attachment 6, 900,000 ft3 in Table 11, 743,320 ft3 in item L5, and, 589,242 ft3 in item L43 (denominator of spray coverage equation). The volume of the lower containment is given as 330,000 ft3 in Attachment 6, 300,000 ft3 in Table 11, 296,767 ft3 in item L5, and, 157,164 ft3 in item L43. The volume of the annular region is given as 69,000 ft3 in Table 11, and 61,927 ft3 in item L5, and, 25.339 ft3 in item L43. Please provide a simplified sketch of the entire containment volume, illustrating the various physical regions of the entire containment, annotated with the assumed volume of each in ft3, and showing the boundaries of the regions as modeled in the analysis described in Attachment 6.

20. Provide a simplified sketch of the containment radioactivity transport for the design-basis accident (DBA) LOCA, showing all nodes (regions) and the transfers between nodes and removals from nodes, as modeled in Attachment 6. Explain how the model changes for the small-break LOCA and the rod ejection accident.

21 Please provide a description of the SBLOCA T/H analysis that was performed for determining the source term. Please include a summary of and justification for the initial assumptions used, the sequence of events, the criteria used for determining fuel pin failures and/or fuel melting, the technical basis supporting the decision criteria, and the results of the analysis from the standpoint of justifying the analysis as limiting with respect to source term.

22. The current licensing bases for DC Cook Units 1 and 2, use departure from nucleate boiling ratio (DNBR) as the criterion for determining the degree of fuel damage resulting from a locked rotor event. The licensee has not submitted either a request to modify its licensing basis or sufficient justification to demonstrate that the use of the 2700 °F criterion is appropriate. We note that the staff has not accepted the use of the 2700 °F criterion at other plants and further that the staff continues to believe that the DNBR

criterion is the appropriate criterion for determining the amount of fuel failure. If you choose to use a criterion other than DNBR, please provide the technical justification for that criterion. Also, the description provided for the locked rotor event indicates that no pins exceed the DBNR limit. However, the description of the analysis does not include sufficient information for the staff to conduct its review. Therefore, please provide a description of the analysis for the locked rotor event. Please include a summary of and justification for the initial assumptions used, the sequence of events, the criteria used for determining fuel pin failures and/or fuel melting, the technical basis supporting the decision criteria, and the results of the analysis from the standpoint of justifying the analysis as limiting with respect to source term.

23. There appear to be discrepancies regarding the assumed volumes of the various regions of the containment between the text of Attachment 6, Table 11 of Attachment 6, and items L5 and L43 of DIT-B-00069-06. For example, the volume of the upper containment is given as 990,000 ft3 in Attachment 6, 900,000 ft3 in Table 11, 743,320 ft3 in item L5, and, 589,242 ft3 in item L43 (denominator of spray coverage equation). The volume of the lower containment is given as 330,000 ft3 in Attachment 6, 300,000 ft3 in Table 11, 296,767 ft3 in item L5, and, 157,164 ft3 in item L43. The volume of the annular region is given as 69,000 ft3 in Table 11, and 61,927 ft3 in item L5, and, 25.339 ft3 in item L43. Please provide a simplified sketch of the entire containment volume, illustrating the various physical regions of the entire containment, annotated with the assumed volume of each in ft3, and showing the boundaries of the regions as modeled in the analysis described in Attachment 6.

24. Provide a simplified sketch of the containment radioactivity transport for the DBA LOCA, showing all nodes (regions) and the transfers between nodes and removals from nodes, as modeled in Attachment 6. Explain how the model changes for the small break LOCA and the rod ejection accident.

25. Requested Action 2 of GL 99-02 states, "If the system has a face velocity greater than 110 percent of 0.203 m/s [40 ft/min], then the revised TS should specify the face velocity."

    Please refer to or provide docketed information which indicates the <u>actual system face velocity and/or the actual residence time</u> for the control room emergency ventilation system (CREVS), engineered safety feature ventilation system (ESFVS), and SPVS and describes how it is calculated for these systems.

    The actual system face velocities can be calculated by dividing the maximum accident condition system flow rates specified in the technical specification (TS) (nominal + typically 10 percent upper value) by the total exposed surface area of the charcoal filter media. (The guidance on calculation of the residence times in ASME AG-1-1997, Division II, Sections FD and FE, Articles I-1000, or in ANSI N510-1975 can be used to calculate the actual system face velocities). It should be noted that the face velocity should be consistent with the bed depth and residence time. (Bed Depth = Face Velocity x Residence Time)

26. In order for the staff to verify that a safety factor as low as two is used, the staff needs to know the charcoal adsorber removal efficiencies which are credited in the current and proposed radiological accident analyses for organic iodide.

27. On page 19 of Attachment 1 to Letter C0600-13, it is stated that in case of CREVS the recent accident analyses assume 95 percent iodine removal efficiency for single-fan operation under normal system flow rate and 80 percent removal efficiency for two-fan operation at an increased face velocity during the first two hours of the accident. It is also stated that "...The 80 percent efficiency calculation includes a safety factor of two. To ensure the accident analysis assumptions remain valid for both single- and two-fan operation, the surveillance requirement is revised to demonstrate a penetration of less than or equal to 1 percent when tested at normal system flow rate."

    (a) Clarify how at 80 percent filter efficiency the safety factor of two is calculated.

    (b) For two-fan operation, what is actual increased maximum face velocity across the charcoal bed.

    (c) Explain how 80 percent filter efficiency at increased face velocity compares with 95 percent filter efficiency at normal system flow rate.

    (d) Demonstrate how the 1 percent penetration at normal system flow rate as the surveillance requirement bound both single- and two-fan operation cases.

February 2, 2001

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