May 13, 1991

Docket No. 52-002

Mr. E. H. Kennedy, Manager
Nuclear Systems Licensing
Combustion Engineering
1000 Prospect Hill Road
Post Office Box 500
Windsor, Connecticut 06095

Dear Mr. Kennedy:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON CESSAR-DC, SYSTEM 80+

Enclosed is a request for additional information based on a review by the Reactor Systems Branch of Chapter 15 of CESSAR-DC. Please respond within 90 days of receipt of this request.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P. L. 96-511.

Sincerely,

original signed by:

Thomas V. Wambach, Project Manager
Standardization Project Directorate
Division of Advanced Reactors and
Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

DISTRIBUTION:

Docket File	SSun, 8E23
NRC PDR	MRubin, 8E23
PDST R/F	TEssig, 10D4
PShea	KEccleston, 10D4
TWambach	
EJordan, MNBB3701	
OGC, 15B18	
ACRS (10), P-315	
DCrutchfield	
WTravers	

LA:PDST	PM:PDST	D:PDST
PShea	TWambach	CMiller
05/7/91	05/09/91	05/13/91

DOCUMENT NAME: LTR TO KENNEDY 04/23

DF03 1/1

cc: Mr. A. E. Scherer, Vice President
Nuclear Quality
ABB Combustion Engineering Nuclear Power
1000 Prospect Hill Road
Post Office Box 500
Windsor, Connecticut 06095-0500

Mr. C. B. Brinkman, Manager
Washington Nuclear Operations
Combustion Engineering, Inc.
12300 Twinbrook Parkway
Suite 330
Rockville, Maryland 20852

Mr. Stan Ritterbusch
Nuclear Licensing
Combustion Engineering
1000 Prospect Hill Road
Post Office Box 500
Windsor, Connecticut 06095-0500

ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION
ON THE DC APPLICATION FOR COMBUSTION ENGINEERING SYSTEM 80+ DESIGN
DOCKET NO. STN 52-002
CESSAR-DC

440.83 (6.3) A recent licensee event report from an operating reactor indicated that all of its safety injection pumps became inoperable due to freezing of the minimum recirculation lines to refueling water storage tank (RWST). The System 80+ design has minimum recirculation lines from the safety injection pumps (located outside containment) connected to the in-containment RWST. Provide a description of the procedures and System 80+ design features that would prevent the occurrence of frozen recirculation lines and associated safety injection lines.

440.84 (15.0) 10 CFR Part 50, Appendix A states that a single failure means an occurrence which results in the loss of capacity of a component to perform its intended safety functions. In other words, a single active failure means a failure of a safety grade active component. In CESSAR-DC, Table 15.0-4, you have provided a list of single failures considered in your safety analyses. Confirm that the components and systems listed in this table are designed to safety grade standards. Identify the items in this list that are non-safety grade and justify why a failure of these components would result in the most conservative analysis.

440.85 (15.0) CESSAR-DC, Table 15.0-4 states that a loss of offsite power greater than 3 seconds after turbine trip caused by reactor trip is considered as a single failure in your accident analyses. It is the staff position that a loss of offsite power (LOOP) should be assumed coincident with a reactor trip/turbine trip following an accident and a single failure of a safety grade active component is assumed to obtain the most limiting consequences of the event. The purpose of these licensing analyses is to document the bounding cases of design basis accidents that the System 80+ design could encounter with sufficient safety margin.

It is noted that in the design of operating reactors, there may be a few cases that a mechanistic approach of LOOP following a turbine trip with time delay is assumed in their safety analyses. The staff considers that it is not prudent to use this compromised assumption in the design of advanced reactors which are expected to offer further defense in depth for public health and safety.

440.86 (15.0) In CESSAR-DC, Chapter 15, expand the sequence of event descriptions of the limiting transients and accidents in each event category. Include the following:

(a) Identify all components and systems that are called upon in event mitigation that ensures safe shutdown of the plant.

(b) List non-safety grade components and systems that are called upon in the scenario of event mitigation per emergency operating procedures (EOPs) developed from the CE Emergency Procedure Guidelines (EPGs).

(c) Provide an assessment of the most limiting consequences of each event analyzed in CESSAR-DC in light of an actual event scenario using EOPs with failures of non-safety grade mitigating equipment that result in a time delay of initiation of safety grade mitigating components and systems.

(d) For condition II events (Moderate Frequency Events), provide transient DNB curves to demonstrate that the acceptance criteria for this class of event are met with sufficient margin and the final stabilized condition has been reached within the analyzed time period.

(e) Identify the most limiting single failure used for the analysis of each event with respect to different acceptance criteria of the event (e.g., peak RCS pressure, DNB, radiological consequences).

(f) Expand the table of assumed initial conditions to include the worst initial conditions for each event considering events occurring at all modes of plant operation.

440.87 (15.0) Provide the basis for the assumption of the time delay (0.8 seconds) between the trip breaker opening and the starting of CEA motion. Are there any testing data to validate this assumption?

440.88 (15.0) Provide the basis for the assumption of the time period required (3.66 seconds) for 90 percent CEA insertion in the reactor core. Discuss the adequacy of 90 percent CEA insertion for a reactor trip relative to the required shutdown margin.

440.89 (15.0) Discuss the NRC review status of all the computer codes used in the transient and accident analyses documented in CESSAR-DC.

450.01 440.90 (15.0) A moderate frequency event in combination with any single active component failure, or single operator error, shall be considered as an infrequent event for which an estimate of the number of potential fuel failures shall be provided and the radiological consequences should be calculated. For such events, fuel failure must be assumed for all rods for which the DNBR falls below 1.24. The radiological consequences of the infrequent event should not exceed a small fraction of 10 CFR 100 limits. Confirm that the above acceptance criteria are used in assessment of the infrequent events for the System 80+ design.

440.91 (15.0) Most of the Chapter 15 and 6.3.3 (LOCA) analyses are performed based on the event being initiated at full power operation. The staff requires that CE provide an assessment on the consequences of the transients and accidents initiated at low power levels or lower modes of plant operation such as shutdown operations. This is required to demonstrate that the analyses performed in CESSAR-DC are the bounding cases for all modes of plant operation.

440.92 (15.0) The scheme of feed only good steam generator (FOGS) has been eliminated in the System 80+ design. So upon a secondary break, steam or feedwater line break, you are going to continue to dump emergency feedwater into the broken steam generator in the System 80+ design. Discuss the merits of this design change from the previous CE design in light of accident recovery and mitigation.

440.93 (15.1.3) Provide the results of an analysis of a postulated fail open event of all turbine bypass valves due to a single malfunction in the electrical system. This event should be treated as an event with moderate frequency of occurrence. The need of this analysis is due to plant operating experiences from the Palo Verde plant where a control system failure caused all steam dump valves to open.

440.94 (15.1.4) CESSAR-DC, Section 15.1.4 indicates that in the infrequent event of inadvertent opening of a steam generator atmospheric dump valve with single failure, the loss of control element drive mechanism control (CEDMC) is assumed as the limiting single failure. However, the CEDMC as described in CESSAR-DC Chapter 7 is not a safety related system. Thus, a failure of a safety grade active component should be assumed on top of the failure of (or non-credit for) non-safety grade systems in the analysis of this event.

450.02
440.95 (15.1.5) Provide a discussion of why the following cases are not analyzed in CESSAR-DC Section 15.1.5 relative to radiological consequences:

(a) A steam line break outside of containment upstream of the main steam isolation valve (MSIV) during full power operation with concurrent loss of offsite power in combination with a single failure, steam generator tube leakage at the allowable limit of the technical specifications, and a stuck CEA.

(b) A steam line break outside of containment upstream of the MSIV during zero power operation with offsite power available in combination with a single failure, steam generator tube leakage at the allowable limit of the technical specifications, and a stuck CEA.

440.96
450.03
(15.1.5) CESSAR-DC Section 15.1.5 states that the evaluation shows that for the full and zero power steam line break (SLB) without loss of offsite power (Cases 2 and 4), the most adverse effect is caused by failure of a MSIV to close on one of the steam lines on the intact steam generator following a main steam isolation signal (MSIS). Consequently, these cases assume steam continues to be released from the intact steam generator after MSIS at a rate of a maximum of 11 percent of plant design steam flow rate. Assuming failure of non-safety grade isolation valves (including the turbine stop valves) downstream of the safety grade MSIV, what is the rate of steam release from the intact steam generator after a MSIS? Assess the radiological consequences of the event scenario.

440.97
(15.1.5) In your analyses of all four SLB cases (Case 1 through Case 4) chosen to maximize the potential for a post-trip return to power, you have stated that since there is no return to power, the values of DNBR remain above those for which fuel damage would be indicated. The staff does not consider that the fact of no return to power would necessarily result in the values of DNBR remain above the minimum DNBR. The transient DNBR could be affected by rapid depressurization of RCS and loss of forced circulation following the assumed loss of offsite power. Discuss the above staff concern and provide transient DNBR curves for all SLB cases.

440.98
(15.1.5) CESSAR-DC Section 15.1.5 states that for a SLB outside the containment upstream of the MSIV, a failure of the MSIV in the steam line connecting the intact steam generator will not affect the radiological consequences of the event. Provide the results of a reanalysis of these events assuming failure of one MSIV in the steam line connecting the intact steam generator and failure of all the non-safety grade isolation valves (including turbine stop valve) downstream of the MSIVs.

440.99
(15.1.5) CESSAR-DC Tables 15.1.5-7 and 15.1.5-10 indicate that the assumed initial steam generator liquid inventory for a SLB outside the containment is less than one half of the steam generator liquid inventory assumed for a SLB inside the containment with essentially similar event scenario. Discuss why these assumptions are conservative for those analyzed cases considering an assumed high steam generator liquid inventory may result in high doses to the environment following a SLB outside the containment.

440.100
(15.1.5) NRC IE Information Notice No. 79-22 identified concerns for several non-safety grade control systems including (1) steam generator PORV (ADV for System 80+) control system, (2) pressurizer PORV (SDS for System 80+) control system, (3) main feedwater control system, and (4) automatic rod control system. These systems may not be properly qualified for adverse environment conditions, and they could potentially malfunction due to the

adverse environment caused by a high energy line break inside or outside of containment. Assess the consequences of the failure of the similar control systems in conjunction with each of the SLB cases analyzed in CESSAR-DC Section 15.1.5.

440.101 (15.2.3) CESSAR-DC Section 15.2.3 indicates that in an event of loss of condenser vacuum with a single failure, the maximum number of fuel pins calculated to experience DNB is no greater than 1.8 percent. Confirm for the above assessment that all fuel pins with a transient DNBR below 1.24 are assumed to experience DNB and fuel failure for the purpose of calculating radiological consequences.

440.102 (15.2.8) The analysis of a postulated main feedwater line break accident presented in CESSAR-DC Section 15.2.8 is not sufficiently conservative with respect to the time assumed for a loss of offsite power (LOOP), single failure, etc. Provide the results of a reanalysis of this event considering a LOOP coincident with reactor trip/turbine trip. A most limiting single failure such as a failure of a diesel, one train of the auxiliary feedwater system, etc., should be considered in this analysis.

440.103 (15.3.3) The analysis of a postulated single reactor coolant pump (RCP) rotor seizure event presented in CESSAR-DC Section 15.3.3 is not sufficiently conservative with respect to the time assumed for a loss of offsite power (LOOP), the time period for a stuck open ADV, etc. Provide the results of a reanalysis of this event considering a loss of off-site power (LOOP) coincident with a reactor trip/turbine trip following the event initiation. In calculating the total ADV opening time for the failed ADV, the 30 minute operator action to close the block valve upstream of the failed ADV should begin from the time of <u>discovering</u> the failure status when the subject ADV is required to close. This part of the scenario is described properly in the analysis of the SGTR event presented in CESSAR-DC Section 15.6.3.

440.104 (15.6.2) CESSAR-DC Table 15.6.2-2 indicated that the non-safety grade pressurizer heaters are assumed to function in accident mitigation. Provide the results of a reanalysis of the letdown line break accident assuming only safety-grade alarms, components, or systems available for accident mitigation.

440.105 450.04 (15.6.2) Discuss the radiological consequences for a small break of the letdown line outside containment assuming that the event does not result in actuation of safety-grade alarms to alert the operators.

450.05 440.106 (15.6.3) CESSAR-DC Section 15.6.3 presents the results of analyses for three cases of postulated steam generator tube rupture (SGTR) event. However, the staff noted that these analyses were performed with non-conservative assumptions such as the use of non-safety grade equipment (e.g., pressurizer heaters) for accident mitigation, the use of time delay on loss of offsite power during the event, and assumption of operator action in less than ten minutes, etc. Provide the results of a reanalysis of a postulated SGTR event including the following elements for a bounding case.

(1) Assumptions

(a) A double-ended tube rupture at the location which results in the worst radiological consequences.

(b) A loss of offsite power occurred coincident with a reactor trip/turbine trip following the event initiation.

(c) A single failure of an atmospheric steam dump valve (ADV) associated with the affected steam generator.

(d) The failed ADV sticks open at its fully open position.

(e) A maximum technical specification tube leakage in the unaffected steam generator (1 gpm) allowed by technical specifications.

(f) A maximum technical specification activity level in the primary coolant allowed by technical specifications.

(g) A maximum percentage of steam generator tubes plugged for both S/Gs allowed by technical specifications.

(h) Fuel failure for any fuel pin with MDNBR below 1.24 as a result of this event.

(i) A minimum of ten minutes operator action time to perform a simple action inside the control room after a clear guidance is available to the operator.

(j) A sensitivity study should be performed for the timing of operator actions to open ADVs following a SGTR. An early opening of ADVs may not be conservative with respect to the radiological consequences since a late opening of ADVs could delay the primary system depressurization and thus prolong the leak flow through the ruptured steam generator tube.

(k) Factor in the steps necessary to prevent the steam generator overfill during the event.

(l) Only the safety-grade components and systems are credited in the accident mitigation. All the non-safety grade equipment are assumed to fail when they are called upon to function per the emergency operating procedure. In the sequence of events, a time delay of the initiation of safety-grade mitigation system should be conservatively assumed to consider the time spent in attempting the use of non-safety grade equipment by operating procedures.

(2) Information Needed to Present the Results of the SGTR Analysis

(a) Sequence of events on a time scale from the initiation of the tube rupture to the shutdown entry conditions.

(b) Operator action times including identifying of the faulted SG, isolation of the faulted SG, initiation of cooldown, depressurization, etc. This should also be presented on a time scale.

(c) Discuss the issue of a potential steam generator overfill, including the integrity of the steam lines under a water filled condition and, if applicable, the effects of liquid flow through the ADV or safety valves, since they were not designed for these service conditions. In assessing steam generator overfill, a most limiting single failure should be considered such as the auxiliary feedwater control valve fully open, an ADV associated with the affected steam generator closed, etc. Also, the effects of continuously feeding the ruptured steam generator in System 80+ design should be addressed.

(d) Major transient curves including RCS pressure, secondary system pressure, DNBR, steam generator water level, leak rates for the affected and intact steam generators, RCS temperature, pressurizer volume, total steam flow, feedwater flow rate, etc.

(e) Amount of fuel failure based on DNBR.

(f) Calculated radiological consequences as compared to the limit set forth in 10 CFR Part 100 (2 hours and 8 hours), including pre-accident and coincident iodine spiking and noble gas inventory based on technical specification limits.

(g) Radiological parameters and curves including mass flow loss with respect to time, flashed fractions, and partition and decontamination factors in accordance with SRP Section 15.6.3.

(h) Discussion of the consequences of a postulated tube rupture at the top of the tube bundle, as it has occurred at North Anna plant.

440.107
450.06
(15.6.3)

CE has indicated in March 6, 1991 ACRS meeting on System 80+ design that the radiological consequences of a steam generator tube rupture for the System 80 design is much higher than that for System 80+ design. This is because the Chi over Q value used for System 80 dose calculation is about five times higher than that used for System 80+ design. Discuss the discrepancy of these assumptions considering the System 80+ design could be built in the areas with the highest Chi over Q value. Provide appropriate interface criteria to ensure that site parameters do not exceed system 80+ analysis assumptions in this area.

440.108
450.07
(15.6.3) Discuss the radiological consequences for a small rupture in a steam generator if the event does not result in actuation of the reactor protection system and safety grade alarms to alert the operators.

440.109
(15.6.3) Provide the results of an analysis for the potential boron dilution event during the recovering phase following a SGTR when backfill from the secondary system through the ruptured steam generator occurred.

440.110
(6.3/15.6) Requirements for and analyses of safety injection systems (SISs) generally assume relatively short periods for operation of the SIS, on the order of several hours, up to perhaps one day. It must be recognized, however, that decay removal must continue to be provided after this initial period has passed, possibly for days, weeks, or even months. Under such circumstances, questions of reliability and maintainability become important. The staff is concerned that very-long-term post-LOCA cooling is not being adequately considered in the design of SISs, and is evaluating how such cooling might be incorporated into advanced reactor designs. The discussion in Sections 6.3 and 15.6 should be expanded and clarified to address the following items.

(1) Identify how long term decay heat is transported to the ultimate heat sink. Include in this discussion the potential for cross-connects between heat removal components that may improve overall system reliability.

(2) Identify what equipment is necessary for long-term post-LOCA cooling, and what are the projected mission times for the required equipment over the spectrum of accidents analyzed. Justify the mission times assumed.

(3) Where non-safety-related equipment is identified for use in long-term cooling, what reliability criteria are assumed in determining the availability of this equipment?

(4) In the event of severe fuel damage to part of the core, considerable activity, and possibly fuel debris, may be transported into the SIS, with deleterious effects on system components. How will maintenance or repair be performed in a potential high-radiation environment?

(5) Even without fuel damage, for long mission times, there is the possibility that key components, e.g., pumps and heat exchangers, will require maintenance and/or repair. How is this accommodated in the SIS requirements and in the long-term cooling plan?

(6) Has the necessity for very-long-term Post-LOCA decay heat removal been considered in your PRA? If not, why is this omission appropriate?

440.111 (15.8) Please provide a schedule for providing an ATWS analysis to demonstrate that the System 80+ ATWS response is within the bound considered by the staff during the deliberations leading to the ATWS Rule (10 CFR 50.62). This should include the analyses referenced on page B-92 which demonstrates that loss of feedwater with failure of turbine trip is the limiting peak pressure event.