May 18, 1994

Docket No. 50-331

Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
IES Utilities Inc.
Post Office Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Liu:

SUBJECT: DUANE ARNOLD ENERGY CENTER - NRC RESEARCH STAFF'S RESPONSE TO DAEC'S COMMENTS ON THE JANUARY 1994 PRELIMINARY DRAFT REPORT LETTER

On March 25, 1994 IES Utilities provided comments on the draft report, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," dated January 20, 1994. The staff has reviewed these comments and the enclosure contains our response. As noted in the response, we plan to correct typographical errors and to incorporate pertinent information identified in the comments into the next draft NUREG/CR report.

Please note that clarifying information would be needed and further studies will be undertaken to fully evaluate several of the comments. The next "for comment" draft NUREG/CR report on this study is scheduled for release in July 1994.

Sincerely,

ORIGINAL SIGNED BY

Robert M. Pulsifer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosure:
Comments and Responses

cc w/enclosure:
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Enclosure
DAEC's Comments and NRC Research Staff Responses

COMMENT (1): Section 2.2, page 2-5.

The report refers to RHR/HPCI when it should be RHR/LPCS.

RESPONSE (1): It was a typographical error, and it has been corrected.

COMMENT (2): Section 2.4, page 2-5.

The report states that the model described in NUREG-0897 was used to define a zone of insulation destruction. However, extending the zone of destruction to an L/D of 7 is not consistent with the guidance of NUREG-0897 for BWRs which states "BWR jet expansion fields decay more rapidly" than PWR fields which extend to an L/D of 7.

RESPONSE (2): NUREG-0897 does not limit the zone of destruction in PWRs to 7 L/D. However, it is recognized that BWR jet expansion will result in lower pressures at 7 L/Ds than PWRs.

COMMENT (3): Section 5.5, page 2-7.

The report states that suppression pool instabilities caused by chugging and steam condensation will cause further disintegration of the insulation debris and will cause the debris to remain suspended indefinitely. Testing at the Mark I containment Full Scale Test Facility, as discussed in NUREG-0661, demonstrated that the chugging phenomena is only associated with small breaks and that for large break LOCAs steam condensation oscillations take place for a maximum of 100 seconds. The assumption that these phenomena will continue to destroy the insulation in the pool and will keep the insulation suspended indefinitely is not consistent with MARK I test data or NUKON test data.

RESPONSE (3): The suppression pool transport aspect of the study had not been started when the preliminary draft report was issued, and therefore conservative modelling assumptions were used. The Mark I tests and the views provided by the DAEC staff will be taken into account in the modelling of suppression pool transport phenomena, which will be included in the final report.

COMMENT (4): Section 3.2.1, page 3-3.

The report incorrectly states that HPCI injects into the B main steam line. HPCI injects into the A feedwater line. The HPCI steam supply line taps into the B main steam line.

The report incorrectly states that the HPCI steam supply line is normally depressurized.

It is not clear what actual reactor vessel level the report is referring to as Level 1 and Level 2. This should be clarified so that the initiation sequence can be properly verified.

RESPONSE (4): This information was used as background information and did not effect the calculations in the study. This error will be corrected in the final report.

This error will be corrected. The NRC staff and SEA representatives would like to discuss how much of this piping is pressurized during operation. The welds on pressurized piping are used to estimate the initiating frequency and the amount of debris generated. Currently 323 welds are used in the analysis.

This statement was included as background information and did not effect the calculations in the study. This statement will be deleted from the report.

COMMENT (5): Section 3.2.2, page 3-3.

ZS-1907 and ZS-2008 are position indications on normally open manual valves. These are not motor operated valves. This incorrect assumption resulted in the amount of pressurized piping used in the analysis being incorrect.

RESPONSE (5): This error will be corrected. (see RESPONSE 4)

COMMENT (6): Section 3.2.3, page 3-3.

ZS-2142 and ZS-2143 are position indications on normally open manual valves, not motor operated valves. This error resulted in the length of pressurized core spray piping being incorrect.

RESPONSE (6): This error will be corrected. (see RESPONSE 4)

COMMENT (7): Section 3.3.1, page 3-4.

The report states that MO-4629 is closed during normal operation. This is incorrect. MO-4629 and MO-4630 are open during normal operation to minimize thermal stresses on the bypass lines.

RESPONSE (7): This error was corrected in the model before the report was issued, but not the text.

COMMENT (8): Section 3.3.2, page 3-7.

The feedwater piping material is incorrectly listed as 304SS. The actual piping material is carbon steel.

RESPONSE (8): Corrections will be made to indicate the correct material.

COMMENT (9): Section 3.5, page 3-18.

The torus diameter is incorrectly listed as 9.25 feet. The actual torus diameter is 25.67 feet.

RESPONSE (9): The error will be corrected. This correction will affect the suppression pool transport model, which was not included in the preliminary study. (see RESPONSE 3)

COMMENT (10): Figure 3-15, page 3-20.

The table in the figure has the strainer velocities for RHR and CS reversed. RHR should be 1.46 and CS should be 1.6.

RESPONSE (10): This was a typographical error. The correct values were used in the model and calculations.

COMMENT (11): Section 3.6, page 3-21.

LPCI injects into the recirculation discharge lines rather than the suction lines.

The assumption that all ECCS pumps are required to mitigate a LOCA is incorrect. The note that this assumption is consistent with the DAEC IPE model for large break LOCAs is also incorrect. In accordance with the DAEC IPE model only one low pressure ECCS pump (RHR or LPCS) is required to provide adequate core cooling.

Continuous ECCS flow of 25,000 gpm is not required to provide adequate core cooling in accordance with the DAEC licensing basis.

RESPONSE (11): The error will be corrected. This statement was included as background information and did not effect the calculations in the study.

The NRC staff agrees that all ECCS pumps are not needed to mitigate a LOCA, but believes that on an ECCS actuation all of the pumps will receive a start signal.

We are not aware of the procedures that would direct an operator to decrease ECCS flow for this event. Please provide us with excerpts of the procedures that directs the operator to reduce ECCS flow. The NRC staff is interested in the conditions and time frame that a reduction in flow would take place.

COMMENT (12): Figure 3-17, page 3-23.

The figure states that a value of 10 feet was conservatively chosen for CS required NPSH. The actual value chosen and the correct value is 15 feet.

RESPONSE (12): This was a typographical error and did not affect the results of the study.

COMMENT (13): Section 4.2.1.3, page 4-6.

The assumption that the main steam and feedwater welds have the same break frequency as 22" recirculation system welds is overly conservative and is not consistent with the guidance in NUREG-4792. These are carbon steel systems, not stainless steel, and have an analytical break frequency no higher than 1.0E-10/Rx-yr. This piping is not susceptible to IGSCC.

RESPONSE (13): Carbon steel is susceptible to corrosion/erosion, and the corrosion/erosion phenomenon was not evaluated in the NUREG/CR-4792 study. The break frequencies used in the study are intended to encompass any possible effects of corrosion/erosion.

COMMENT (14): Section 4.2.1, page 4-6.

No credit is given for the mitigation of IGSCC that is provided by Induction Heating Stress Improvement (IHSI) and Hydrogen Water Chemistry (HWC). This is contrary to the guidance contained in NUREG-0313 and Generic Letter 88-01.

RESPONSE (14): The values in NUREG/CR-4792 were lowered by a factor of ten to give credit for steps to mitigate IGSCC. Credit for ISI has lowered the weld break frequency to a value slightly above that for non-IGSCC susceptible stainless steel.

COMMENT (15): Table 4-3, page 4-10.

The total pipe break frequency estimate for the main steam system is added incorrectly. The total should be 1.8E-05, not 1.8E-4.

RESPONSE (15): This was a typographical error. The correct value was used in the calculations.

COMMENT (16): Section 5.2.2, page 5-4.

Extending Region I to an L/D of 3 to account for possible destruction to a pressure of 5 bar is overly conservative and is not supported by testing conducted on jacketed NUKON insulation at HDR (Heissdampfreaktor) and CEESI (Colorado Engineering Experiment Station, Inc).

The Region III outerbound at an L/D of 7 was originally established for break pressure of 150 bar and for unjacketed insulation. Experiments conducted at HDR and CEESI and the guidance contained in NUREG-0897 show that this is overly conservative for BWRs with jacketed insulation.

RESPONSE (16): The HDR data and the CEESI test data have been reviewed. We believe a L/D of 3 is appropriate for Region I when coupled with a destruction factor of 0.75.

The HDR data and the CEESI test data have been reviewed. We believe that from 5 L/D to 7 L/D is an appropriate boundary for Region III when coupled to a destruction factor of 0.40.

COMMENT (17): Section 5.2.2, page 5-6.

The fractions of transportable debris generated which are used in the study for regions I, II, and III are conservative and are not supported by experimental data using jacketed NUKON insulation.

RESPONSE (17): The destruction factors were chosen based on SEA's interpretation of the HDR and CEESI tests. The staff considers the destruction factors reasonable estimates and not overly conservative.

COMMENT (18): Section 5.2.3.1, page 5-6.

The assumption that only the pipe in which the break occurs is targeted by the jet may be nonconservative.

RESPONSE (18): The January 1994 model has been revised to include "targeted" pipes. Targets other than pipes were not included in the revised model. The revision did affect the results of the study and re-enforced its findings. These results were reported at the March 30, 1994, public meeting.

COMMENT (19): Section 5.2.3, page 5-7.

Neglecting the effects of "shadowing" may result in the assumed generation of overly conservative amounts of insulation debris.

RESPONSE (19): The NRC staff does not believe the exclusion of "shadowing" is overly conservative.

COMMENT (20): Tables 5-1, 5-2, and 5-3.

Numerous target lengths which are listed in these tables are not physically possible due to the plant layout (i.e., it is not possible for the piping to fall within the 90 degree cone or the piping does not exist). This results in assuming that conservative amounts of insulation debris are generated. The 90° cone model has been replaced with a spherical model.

RESPONSE (20): The NRC staff is unable to determine the significance of this comment without additional information. Please identify the welds and region that are incorrect. Also identify (estimate) the "correct" lengths. The lengths used in the study were estimated from the drawings provided to SEA.

COMMENT (21): Section 5.3.1, page 5-15.

The statement that the transport models proposed in USI A-43 are not applicable to BWRs is incorrect. These transport models can be used to analyze the debris flow and settling on the drywell floor to the vents and in the suppression pool during the recirculation phase.

RESPONSE (21): The transport models in USI A-43 are being revised to reflect recent events and studies (including foreign events and studies). Also, the PWR transport model does not address the effects of suppression pool dynamics on debris transport.

COMMENT (22): Section 5.3.1, page 5-17.

The values which are used for the debris transport fractions are estimates which are not backed up by experimental data or the event at Barsebäck and may be excessively conservative. A reduction of these fractions by only 5% will reduce the conditional blockage probability by over 40%.

RESPONSE (22): The drywell transport factors are not based on experimental data, but they are based on the event at Barsebäck. The NRC staff does not believe the values are excessively conservative.

COMMENT (23): Section 5.4, page 5-18.

The chugging phenomena is not associated with large break LOCAs. Also, steam condensation oscillations take place for a maximum of 100 seconds.

The assumption that all of the debris within the pool is deposited on the strainers is grossly conservative. The NUKON insulation has a negative buoyancy and suppression pool velocities during the recirculation phase will not be large enough to transport all insulation debris to the strainers.

RESPONSE (23): See RESPONSE 3.

COMMENT (24): Section 5.5.3, page 5-24.

The draft report states that the sensitivity analysis is documented in section 10.2.3. The report does not contain a section 10.2.3.

RESPONSE (24): This was a typographical error. The analysis is contained in section 7.3.

COMMENT (25): Section 5.5, page 5-24.

The report states that the available NPSH using atmospheric pressure and a 120 degree pool temperature is about 24 and 32 feet of water for LPCI and CS respectively. These values are incorrect for a 120 degree pool temperature. For example, when the pool temperature is 120 degrees, the NPSH available for LPCI is about 34 feet and for CS is about 36 feet.

RESPONSE (25): The $NPSH_{available}$ is based on licensing conditions - not actual accident conditions. Further studies will be carried out to estimate the effect associated $NPSH_{available}$ based on accident conditions, when the information is provided.

COMMENT (26): Section 5.6, page 5-25.

The assumption that adequate core cooling is lost when strainer head loss equals 14 feet is incorrect since the core spray pumps have an NPSH margin of 17 feet and one core spray pump can provide adequate core cooling after the core is initially reflooded.

RESPONSE (26): The NRC staff is reviewing this assumption.