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January 29, 1993

Docket No. 50-213
B14298

Re: SEP III-5.A
ISAP Topic 1.31

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Gentlemen:

Haddam Neck Plant
SEP Topic III-5.A
High Energy Pipe Break Inside Containment

In the NRC Staff letter of January 24, 1992,[1] Systematic Evaluation Program (SEP) Topic III-5.A, "High Energy Pipe Break Inside Containment" was identified as being reviewed by NRC Staff. In subsequent discussions between the NRC Staff and Connecticut Yankee Atomic Power Company (CYAPCO), the NRC Staff requested a summary of the ten open items which were identified in Section 4.8 of NUREG-0826.[2]

CYAPCO originally provided the justifications or clarifications requested by the Reference (2) report in a letter to the NRC Staff dated December 17, 1984.[3] Since the December 17, 1984, letter, various modifications and commitments have been made by CYAPCO which impact the information provided previously. Therefore, this letter serves to update the justification or clarification required to address the ten open items from Reference (2) and supports final closeout of SEP Topic III-5.A.

(1) Cascading Effect Sequences

The investigation results presented in CYAPCO's December 17, 1984, letter showed that cascading is controlled by separation of piping or by

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(1) A. Wang letter to J. F. Opeka, "Haddam Neck Plant--Closeout and Status of Various Systematic Evaluation Program Issues," dated January 24, 1992.

(2) Integrated Plant Safety Assessment, Haddam Neck Plant, NUREG-0826, June, 1983.

(3) W. G. Counsil letter to J. A. Zwolinksi, "SEP Topic III-5-A, High Energy Pipe Break Inside Containment," dated December 17, 1984.

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physical barriers. Due to the position that cascading is controlled, CYAPCO considers this item resolved.

(2) Jet Impingement from Circumferential Breaks

This open item involves a request from the Staff for additional information regarding the consideration of jet impingement effects as a result of circumferential breaks. CYAPCO's December 17, 1984, letter confirms that CYAPCO's high energy pipe breaks (HEPB) analysis considered jet impingement from both longitudinal and circumferential breaks. Additionally, the jet was assumed to travel the arc defined by the whipping pipe. Therefore, CYAPCO considers this item resolved.

(3) Strain Level Functionability Criteria

This item concerns allowable strain levels utilized in the SEP HEPB analysis. As concluded in CYAPCO's December 17, 1984, letter, this concern is not applicable to the SEP HEPB study and should be disregarded.

(4) Containment Integrity Criteria

This open item involves a request from the Staff for additional justification concerning containment integrity when subjected to pipe whip and jet impingement.

CYAPCO's December 17, 1984, letter evaluated the effects of a feedwater line break on the containment structure utilizing engineering judgement. It was concluded that the containment structure was capable of withstanding the resultant jet loads without degrading the integrity of the liner plate. However, a detailed analysis indicating the technical bases for determining the limiting pipe break and evaluating the effects of the break was not prepared.

As part of the current HEPB reverification program, a detailed HEPB evaluation was performed on the main steam and feedwater lines inside containment to evaluate the effects of a pipe break on the containment structure. The calculation developed mainsteam and feedwater jet impingement loads and reviewed the potential for pipe interaction with the containment structure.

The evaluation concluded that both the mainsteam and feedwater piping would not impact the containment wall following a pipe break. This conclusion was based on calculations and evaluations which considered the piping geometry and location, the support scheme, and the location of the pipe breaks.

The effects of the jet impingement loads on the containment structure, including the containment liner, were evaluated. It was determined that

the containment structure is capable of withstanding the resultant jet loads without degrading the integrity of the liner plate.

(5) Jet Impingement Effects on Target Piping

In the NRC's October 12, 1982, letter,[4] the Staff indicated that it was not clear how CYAPCO utilized size differential criteria in the jet impingement effects evaluation provided in CYAPCO's September 17, 1982,[5] letter.

CYAPCO's December 17, 1984, letter indicates that an evaluation of jet impingement of piping targets was performed in order to resolve this open item.

CYAPCO's December 17, 1984, letter concluded that jet impingement would not result in the loss of the three available shutdown methods. This is due primarily to the physical separation of required safety systems. This further strengthens our position in this area. Therefore, CYAPCO considers this item resolved.

(6) Effects on Instrumentation

CYAPCO's December 17, 1984, letter provided an evaluation of the minimum instrumentation required for safe shutdown. The analysis assumed a worst case incident where nonphysically separated instrumentation are rendered inoperable by a single HEPB. The following instrumentation was addressed:

a) Pressurizer Level
b) Pressurizer Pressure
c) Steam Generator Level
d) Loop T_H or Core Exit Thermocouples
e) Refueling Water Storage Tank Level, Volume Control Tank Level, and Demineralized Water Storage Tank Level
f) Pressurizer Relief Valve Monitors, Containment Water Level, and Containment High Range Radiation Detectors

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(4) D. M. Crutchfield letter to W. G. Counsil, "SEP Topic III-5.A, Effects of Pipe Break on Structures, Systems and Components Inside Containment," dated October 12, 1982.

(5) W. G. Counsil letter to D. M. Crutchfield "SEP Topic III-5.A, High Energy Break Inside Containment," dated September 17, 1982.

The matrix below shows which of the above instruments is required during each of the three shutdown methods (main feedwater, auxiliary feedwater, and feed-and-bleed). However, the matrix does not include instrumentation required due to the HEPB. For example, steam generator pressure and containment pressure would be required for a steam line break in addition to instruments required for the shutdown method employed.

Shutdown Method	Required Instruments
1 (Main Feedwater)	a) Pressurizer Level b) Pressurizer Pressure c) Steam Generator Level d) Loop T_H or Core Exit Thermocouples
2 (Auxiliary Feedwater)	a) Pressurizer Level b) Pressurizer Pressure c) Steam Generator Level d) Loop T_H or Core Exit Thermocouples e) Demineralized Water Storage Tank
3 (Feed-and-Bleed)	a) Pressurizer Level b) Pressurizer Pressure c) Loop T_H or Core Exit Thermocouples d) Refueling Water Storage Tank Level and Volume Control Tank Level e) Pressurizer Relief Valve Monitors, Containment Water Level, and Containment High Range Radiation Detectors f) Containment Pressure g) Reactor Coolant System (RCS) Wide-Range Press

The analysis for pressurizer level (Item a) in CYAPCO's December 17, 1984, letter indicated that Plant Shutdown Method 3 would be relied upon in case of loss of pressurizer level instrumentation. Subsequently, as indicated under the Integrated Safety Assessment Program (ISAP)

Topic 2.04 in CYAPCO's March 2, 1989,[6] letter, CYAPCO replaced transmitters for main control board-mounted equipment for the three loops of pressurizer level during the Cycle 14 refueling outage. In Enclosure 1, page 12, of the NRC's letter of March 21, 1990,[7] the Staff acknowledged and accepted that the reactor control and protection system instrumentation modifications did not reduce the degree of independence, separation, and isolation provided in the original design. This supports the analysis presented in CYAPCO's December 17, 1984, letter. Note, however, that plant Emergency Operating Procedures (EOP) would not require the operator to use feed-and-bleed for loss of pressurizer level indication due to a HEPB. Even if there is a loss of pressurizer pressure or level indication, CYAPCO would continue to use main or auxiliary feedwater and rely on other indications of satisfactory primary system heat removal. CYAPCO would attempt to use every other alternative, including use of a condensate pump to feed the secondary side of the steam generators, prior to feed-and-bleed. CYAPCO considers feed-and-bleed to be the option of "last choice," when all other options have been exhausted. The Haddam Neck Plant EOPs support this philosophy. However, the termination criteria for safety injection is based in part on regaining pressurizer level. Losing pressurizer level indication will certainly inhibit safety injection termination. This means that in effect, a partial feed-and-bleed (i.e., the operator will inject but will not open the power operated relief valves) is used in conjunction with Plant Shutdown Methods 1 and 2. However, it is Plant Shutdown Methods 1 and 2 that are being relied upon to remove decay heat.

The analysis for pressurizer pressure (Item b) indicated that RCS pressure instruments would be utilized in case of loss of pressurizer pressure instrumentation. Subsequently, as indicated in CYAPCO's March 2, 1989, letter, CYAPCO replaced transmitters and main control board-mounted equipment for the existing three loops of pressurizer pressure and added a fourth channel of pressurizer pressure. As was the case for pressurizer level, this modification to pressurizer pressure did not reduce the degree of independence, separation, or isolation provided in the original design. Therefore, the analysis presented in CYAPCO's December 17, 1984, letter remains valid. RCS pressure instruments would be utilized in case pressurizer pressure instrumentation is rendered inoperable by a single HEPB.

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(6) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Integrated Safety Assessment Program (ISAP)," dated March 2, 1989.

(7) A. B. Wang letter to E. J. Mroczka, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Reactor Protection System Upgrade (Phase One)," dated March 21, 1990.

The analysis for steam generator level (Item c) in CYAPCO's December 17, 1984, letter indicates that safe shutdown can be achieved if steam generator level instrumentation is lost due to a single HEPB. Modifications to the steam generator wide-range and narrow-range level have occurred since the CYAPCO December 17, 1984, letter transmittal. The modifications have, along with other benefits, improved the adherence to single failure criterion. Each steam generator now has two redundant Class 1E wide-range level detectors to provide wide-range level signals. Steam generator level instrumentation conduit runs in groups of two inside the crane wall, but together in the annulus region. A break in the annulus region has been identified which could disable the conduit which carries the wiring for the wide-range steam generator level transmitters (LT-1302-1A, 2A, 3A, and 4A) of all four steam generators. However, the redundant steam generator level channels will remain operable since the redundant channels have been routed taking into account HEPBs. Under Phase III of the reactor protection upgrade (planned for 1993 refueling outage) eight additional redundant Class 1E steam generator narrow-range level transmitters will be installed. Under the present system, each steam generator has one narrow-range level transmitter. After the Phase III modifications, there will be twelve transmitters in total (three per loop). However, the analysis provided in CYAPCO's December 17, 1984, letter remains valid (safe shutdown can be achieved if the subject instrumentation is lost).

The analysis presented in CYAPCO's December 17, 1984, letter for the instrumentation corresponding to items d through f above remains valid. No modifications performed since the submittal of CYAPCO's December 17, 1984, letter have affected the analysis results for this instrumentation.

(7) Main Coolant Loop Breaks

CYAPCO's December 17, 1984, letter indicated that a limited number of modifications to the RCS were identified as a result of the review of SEP Topic III-6. These modifications were included as part of ISAP (ISAP Topic 1.08). As indicated in CYAPCO's March 2, 1989, letter, all but one of the modifications were completed in the Cycle 14 refueling outage and Topic 1.08 is considered resolved. The only remaining modification as documented in CYAPCO's December 17, 1984, letter was the replacement of steam generator hold-down bolts. Per CYAPCO's letter of May 1, 1987,[8] the modifications specified for the steam generator hold-down bolts are not required (safety factors on the bolts are within allowable limits). Therefore, all required modifications to the RCS are complete.

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(8) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "RCS Seismic Reevaluation and Final Plans for Modifications," dated May 1, 1987.

CYAPCO's December 17, 1984, letter addressed the ability of the leakage monitoring systems to detect RCS leakage from the postulated circumferential throughwall flaw. CYAPCO's December 17, 1984, letter indicated that the conversion to Standard Technical Specifications would ensure that operability requirements and actions are addressed in the event of leakage detection system failure. Generic Letter 84-04[9] documents the Staff's response and agreement with the completion of the RCS seismic modifications as recommended by the SEP and the issuance of the leakage detection technical specifications. Generic Letter 84-04 concluded that Unresolved Safety Issue A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant System" was considered resolved.

(8) Plant Shutdown Method 3

CYAPCO's December 17, 1984, letter, responded to the Staff's request for information regarding Plant Shutdown Method 3 (Feed-and-Bleed). This shutdown method is to be used only when the reactor coolant pressure boundary remains intact and the steam generators or systems servicing them are not available to remove core decay heat.

Feed-and-bleed is needed in the unlikely event that either: (1) a pipe break outside containment were to compromise capability to inject either main or auxiliary feedwater into the steam generators, (2) a pipe break inside containment were to compromise capability to utilize the steam generators to remove core decay heat, or (3) some other failure resulted in the loss of steam generator decay heat removal.

Feed-and-bleed entails injecting water into the RCS with a charging pump (or high pressure safety injection pump) and allowing water to discharge from the pressurizer power operation relief valves into the pressurizer relief tank and then into the containment sump. The water would be drawn from the sump, cooled by the residual heat removal heat exchangers, and returned to the suction of the charging pumps (or high pressure safety injection pumps). This is the least desirable shutdown method.

Subsequent to the response provided in CYAPCO's December 17, 1984, letter, CYAPCO has qualified the low temperature overpressure protection system (piping and valves) so that it will be qualified for the temperature, pressure, and flow conditions that would exist late in a feed-and-bleed scenario. Feed-and-bleed modifications were addressed in Reference (7) as ISAP Topic 1.62.

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(9) A. B. Wang letter to E. J. Mroczka, "Generic Letter 84-04 (Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination and Postulated Pipe Breaks in PWR Primary Main Loops)," dated July 11, 1989.

(9) Main Steam/Main Feedwater Interactions

Per NRC's letter of October 12, 1982, and later per NUREG-0826, the Staff requested that CYAPCO clarify the apparent inconsistency between the matrix and interaction evaluation for the main steam line breaks as presented in CYAPCO's September 17, 1982, letter.

In the NRC Staff's letter dated October 12, 1982, the Staff noted that in Section VII.A of CYAPCO's September 17, 1982, letter, interactions between steam and feed lines from breaks in main steam lines from steam generators 2 and 3 are considered not credible. However, the Staff noted that interaction matrices provided in CYAPCO's September 17, 1982, letter showed that for line 24-SHP-601-2 (main steam from #2 generator), interactions are shown for WFPD-601-7 (feed for #1 generator) and WFPD-601-8 (feed for #2 generator).

CYAPCO has reviewed the situation and has determined that the scenario given in the interaction evaluation is in fact correct and that for a given main steam line break, the only potential feedwater line interaction occurs with the feedwater line corresponding to the same steam generator. Therefore, the matrix is in error but the evaluation is correct as stated.

(10) Core Deluge Piping Breaks Effects

This concern involves determining the most limiting single failure that could occur after a core deluge line break between the isolation valve and reactor vessel.

As presented in CYAPCO's December 17, 1984, letter, the Staff questioned whether the loss of the motor operated valve in the unaffected train would be the most limiting case.

The analyses performed in support of the permanent modifications and the in-house small break loss-of-coolant accident analyses lead us to conclude that failure of the motor-operated valves in the unaffected loop is not the worst single failure. However, CYAPCO has concluded that acceptable results would be shown for any postulated single failure. The analyses presented in CYAPCO's July 1984 letter[10] show that for a break of this size, adequate results are obtained for the injection phase without any credit from the low-pressure safety injection (LPSI) system. The limiting single failure for these analyses was the failure of a diesel generator. A postulated core deluge line break would behave differently in the recirculation phase than the same

(10) Northeast Utilities Service Company, "Calculative Methods for the Northeast Utilities Small-Break LOCA ECCS Evaluation Model, Volumes 1 and 2," July 1984.

size break occurring elsewhere in the RCS. This is due to the higher LPSI flowrate associated with a core deluge line break which causes an earlier time to recirculation as documented in CYAPCO's December 17, 1986, letter.[11] The long-term modifications described in CYAPCO's April 1, 1987, letter[12] have been analyzed and show acceptable results for all postulated single failures.

CYAPCO believes the information provided above should allow you to close the ten open items listed in Section 4.8 of NUREG-0826 and close SEP Topic III-5.A. Please feel free to contact my staff if you have any questions regarding this topic. As in the past, please provide us with written confirmation of closure of this SEP and ISAP topic.

Please advise if you have any questions.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

J. F. Opeka

J. F. Opeka
Executive Vice President

cc: T. T. Martin, Region I Administrator
A. B. Wang, NRC Project Manager, Haddam Neck Plant
W. J. Raymond, Senior Resident Inspector, Haddam Neck Plant

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(11) E. J. Mroczka letter to C. I. Grimes, "Proposed Amendment to Facility Operating License No. DPR-61, Revisions to Technical Specifications, Flow Control Valve Repositioning," December 17, 1986.

(12) E. J. Mroczka to U.S. Nuclear Regulatory Commission, "ECCS Modification—Request for Extension of Single Failure Exemption," dated April 1, 1987.