CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT
362 INJUN HOLLOW ROAD • EAST HAMPTON, CT 06424-3099

February 20, 1997
Docket No. 50-213
B16172

Re: RAI for RCS
Overpressure

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

Haddam Neck Station
Response to
Request for Additional Information On Isolated Loops

In a letter dated December 27, 1996,(1) the NRC Staff requested additional information to assess the conditions associated with a July 1994 Reactor Coolant System (RCS) overpressurization event at the Haddam Neck Station.

The purpose of this letter is for Connecticut Yankee Atomic Power Company (CYAPCO) to provide the Staff with the requested additional information. Enclosed as Attachments 1 and 2 are the responses to the five requests.

If you should have any questions on the information contained herein, please contact Mr. G. P. van Noordennen at (860) 267-3938.

Very truly yours,

Connecticut Yankee Atomic Power Company

T. C. Feigenbaum
Executive Vice President and
Chief Nuclear Officer

cc: H. J. Miller, Regional Administrator, Region I
M. B. Fairtile, NRC Project Manager, Haddam Neck Plant
W. J. Raymond, Senior Resident Inspector, Haddam Neck Plant

260017

A001/1

(1) USNRC letter to B. D. Kenyon, "Request for Additional Information (TAC NO. M97181)", dated December 27, 1996

1028-3 REV 2-91

Docket No. 50-213

B16172

Attachment 1

Haddam Neck Plant

Response to

Request for Additional Information On Isolated Loops

February 1997

Attachment 1
Haddam Neck Plant
Response to
Request for Additional Information On Isolated Loops

Restatement of Request

The NRC staff is examining the conditions associated with a July 1994 event in which an isolated reactor coolant loop apparently reached 1000 psig and 106 °F. The Staff believes this issue has potential generic implications and requests the following additional information.

Request #1

Explain the evolution of the July 1994 event. In particular, note what the RCS and the secondary side conditions were at the time the loop isolation valves were closed, what caused the temperature and pressure to reach their eventual values of 1000 psig and 106 °F, and what factors (if any) limited the pressure increase.

Response to Request #1

On July 11, 1994, the loop isolation valves had been closed immediately following a trip in response to a loss of oil from #3 Reactor Coolant Pump (RCP). A small fire occurred from oil that leaked from the #3 RCP upper oil reservoir onto hot components. The Reactor Coolant System (RCS) loop was isolated and cooled down to prevent reignition. The plant cooled down RCS loop #3 by using the bleed and feed method of cooling.

On July 13, 1994, the RCS was in Mode 3, hot standby with RCS loop #3 reactor coolant loop isolated for retest of the #3 RCP. RCS loops #1, #2, and #4 were all at normal operating pressure and normal operating temperature. RCS loop #3 was 106°F and 1000 psig. The secondary side, also at 106°F, was isolated, at atmospheric pressure, with the main steam non-return valves closed.

The cooldown and pressurization of reactor coolant loop #3 was performed in accordance with Temporary Procedure Change (TPC) (#94-340) to NOP 2.4-4, "Cooldown of an Isolated Loop". The purpose of increasing pressure in RCS loop #3 was to place the RCP #3 motor on the upper thrust shoes for breakaway torque testing.

CYAPCO dedicated an operator to monitor and maintain pressure and temperature within the limits specified in NOP 2.4-4, "Cooldown of an Isolated Loop".

Request #2

State any operational restraints (i.e. unacceptable pressure/temperature regimes or relief valve capability) which may have been established by the design code of record for all ferritic components of an isolated reactor coolant loop. Show that the conditions (1000 psi at a temperature of 106 °F) associated with the July 1994 event are consistent with these design considerations.

Response to Request #2

No operational restraints were established by the design code of record for all ferritic components of an isolated reactor coolant loop. The conditions of 1000 psi at 106 °F are consistent with CYAPCO Haddam Neck Plant Technical Specifications Section 3/4.7.2, Steam Generator Pressure/Temperature Limitation, Limiting Condition for Operation.

Request #3

Demonstrate that the calculations performed to analyze the July 1994 event by the methods of ASME Section III (or Section XI) Appendix G are correct. Provide additional detail on the geometry of steam generator channel head including head diameters and wall thicknesses, the results of the channel head stress analysis, the assumed flaw orientation and size, the material properties for the ferritic materials in the channel head, and the loading conditions (pressure and temperature) during the July 1994 event. This information will permit the staff to independently assess the significance of the event.

Response to Request #3

The calculations performed to analyze the July 1994 event were performed by Westinghouse in accordance with the methodology described in Appendix G of the ASME, Section III Code. A summary description of the results of this calculation is provided in Attachment 2 (letter from D. E. Prager, Westinghouse, to N. F. Azevedo). The calculations were reviewed and accepted by the CYAPCO Engineering Staff.

Additional details on the geometry of the steam generator channel head include:

- Head Diameter = 8 ' 11 3/8 "
- Wall Thickness = 5.0 "

Request #4

State any corrective actions that were taken as a result of the July 1994 event to prevent conditions which could lead to nonductile failure of the reactor coolant loop when the loops are isolated. Based upon the analysis performed by Westinghouse for Northeast Utilities, the staff expects that the licensee would have taken corrective actions to ensure that pressure/temperature conditions would not occur which could challenge the integrity of any loop in a subsequent event. However, it appears from the information received to date that no precautions against overpressurization were in place during the August 1996 event.

Response to Request #4

The affected reactor coolant loop, in the "July 1994 event," was isolated from the reactor vessel and the pressure was intentionally increased to 1000 psi as a planned evolution to allow for retesting of RCP #3. The Pressure/Temperature (P/T) limit curves provided in Plant Technical Specification Section 3/4.4.9 were developed to protect the belt line region of the reactor vessel. Since the affected portion of the RCS was isolated from the reactor vessel it was concluded, at the time of the event, that the Technical Specification P/T limits did not apply to the isolated loop. This conclusion was subsequently reversed following additional CYAPCO reviews of this issue. As a result of the questions identified during these reviews, CYAPCO concluded that the additional structural evaluations were required to ensure compliance with 10CFR50 Appendix G. These structural evaluations were performed by Westinghouse to demonstrate that the 1000 psi pressure and 106°F temperature did not exceed the 10 CFR 50, Appendix G limits. (See attachment 2)

As a result of follow up reviews it was recommended that the Haddam Neck Technical Specifications be modified to included a 10 CFR 50, Appendix G limit curve for an isolated loop since the current Technical Specifications did not explicitly address P/T limits for an isolated loop. The proposed Technical Specification change was not submitted to the NRC when the decision was made not to restart the Haddam Neck Plant.

For the August 1996 event, the plant was shutdown with all four loops isolated and cooled down. In this instance, the operators inappropriately isolated overpressure protection in an effort to reduce suspected leakage from the loop into the reactor coolant system. This is a different issue than the "July 1994 event" in which only one loop was isolated, cooled down and pressurized.

Request # 5

Provide information on any other events of this type that may have occurred previously at Haddam Neck or other pressurize water reactors with loop isolation capability.

Response to Request #5

No other events of this type occurred previously at Haddam Neck. CYAPCO does not have the industry information available to provide the NRC on any events of this type that may have occurred previously at other pressurized water reactors with loop isolation capability.

Docket No. 50-213
B16172

Attachment 2

Haddam Neck Plant

Response to

Request for Additional Information On Isolated Loops

February 1997

MSE-SMT-275(94)

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

July 26, 1994

Nelson Azevedo
Northeast Utilities Service Co.
170 Selden St.
Berlin, CT. 06037

Dear Mr. Azevedo:

We are enclosing pressure-temperature limit curves for the isolated loop of the Haddam Neck plant, along with a brief report providing the background and technical basis for the curves. The curves demonstrate that the inadvertant pressurization of the inactive loop to 1000 psi at 106°F was acceptable.

If you have questions or need further information, please call me at 412-374-6494.

Very Truly Yours

Westinghouse Electric Corporation

D E Prager
D. E. Prager

Attachment to
MSE/SMT/275

Pressure and Temperature Limitations for the Isolated Loop: Haddam Neck

To ensure that adequate margins of safety are maintained for the loop which is isolated, a pressure-temperature limit curve was developed, using the same philosophy and margins which are required for the primary system during heatup and cooldown operations. Specifically, the guidelines and requirements of Section III Appendix G of the ASME Code were used. The same appendix appears as Appendix G of Section XI.

When the loop is isolated, the reactor vessel is no longer part of the loop. Since it is the limiting location for generation of the technical specification pressure temperature limit curves, a new limiting location must be determined. Review of the isolated loop revealed that the highest stressed region would be in the channel head of the steam generator, near its intersection with the tubesheet. This region was used to generate a curve for allowable pressure in the isolated loop as a function of temperature.

The guidelines which were used in the analysis were to assume a flaw in this region and impose a safety factor of 1.5 on the pressure stresses. The reference flaw was oriented circumferentially, perpendicular to the maximum principle stress in the region, which is axial, and results from the discontinuity at the channel head-tubesheet junction. The shape of the flaw was assumed to be semi-elliptic, with length six times depth, and the depth was taken as 25 percent of the wall thickness (t=5.0 inches).

The material properties used in developing the curve were for the cast channel head material, which is grade SA266 WCC. Fracture toughness data for this material have been obtained, and the toughness has been found to be similar to the steels for which the ASME Code K_{IC} and K_{IR} curves were developed. The RT_{NDT} values for the channel head and the tubesheet were known to be less than or equal to 60°F, the material ordering requirement. An attempt was made to find the actual data for the channel head and tubesheet from material test certificates, but only the tubesheet data are available. These results are provided in Table 1. Therefore 60°F was used. It is worthy of note that the stresses are highest in the weld region, which has an RT_{NDT} value of less than 10°F. For conservatism, the highest stresses were combined with the highest RT_{NDT} value. The reference fracture toughness curve of Appendix G, the K_{IR} curve, was used.

The allowable pressure as a function of temperature was determined by calculation of the stress intensity factor, using the well known expression of Newman and Raju*. The allowable pressure at a given temperature was then found as the pressure at which the calculated stress intensity factor (including the 1.5 safety factor) was equal to the reference fracture

---

*Newman, J.C., and Raju, I.S., "Stress Intensity Factors for Internal Surface Cracks in Cyulindrical Pressure Vessels", Trans ASME Journal of Pressure Vessel Technology, Volume 102, 1980, pp. 342-346.

toughness. The curve of allowable pressure as a function of temperature is given in Figure 1. Also contained in this figure is another pressure temperature curve based on a more realistic assessment of the fracture toughness, the K_{Ic} curve of the ASME Code Section XI. This more realistic curve is presented in the figure as a dashed line, and shows that a margin of at least 40% exists for the curve to be used for the isolated loop.

TABLE 1
CHARPY DATA FROM THE HADDAM NECK TUBESHEET

Tube Plate #	Heat No.	Test Temperature	RT_{NDT}(°F)	Charpy Date (ft-lb)
16A460-1	X53139, X42820, X32028	10	30	18.0, 14.0, 17.0
16A460-2	125J235	10	30	19.4, 27.8, 23.5
16A460-3	ZV1167	10	30	39.0, 40.5, 42.0
16A460-4	ZV/1258, BV/1226, SV/1581	10	30	55.0, 58.5, 35.0

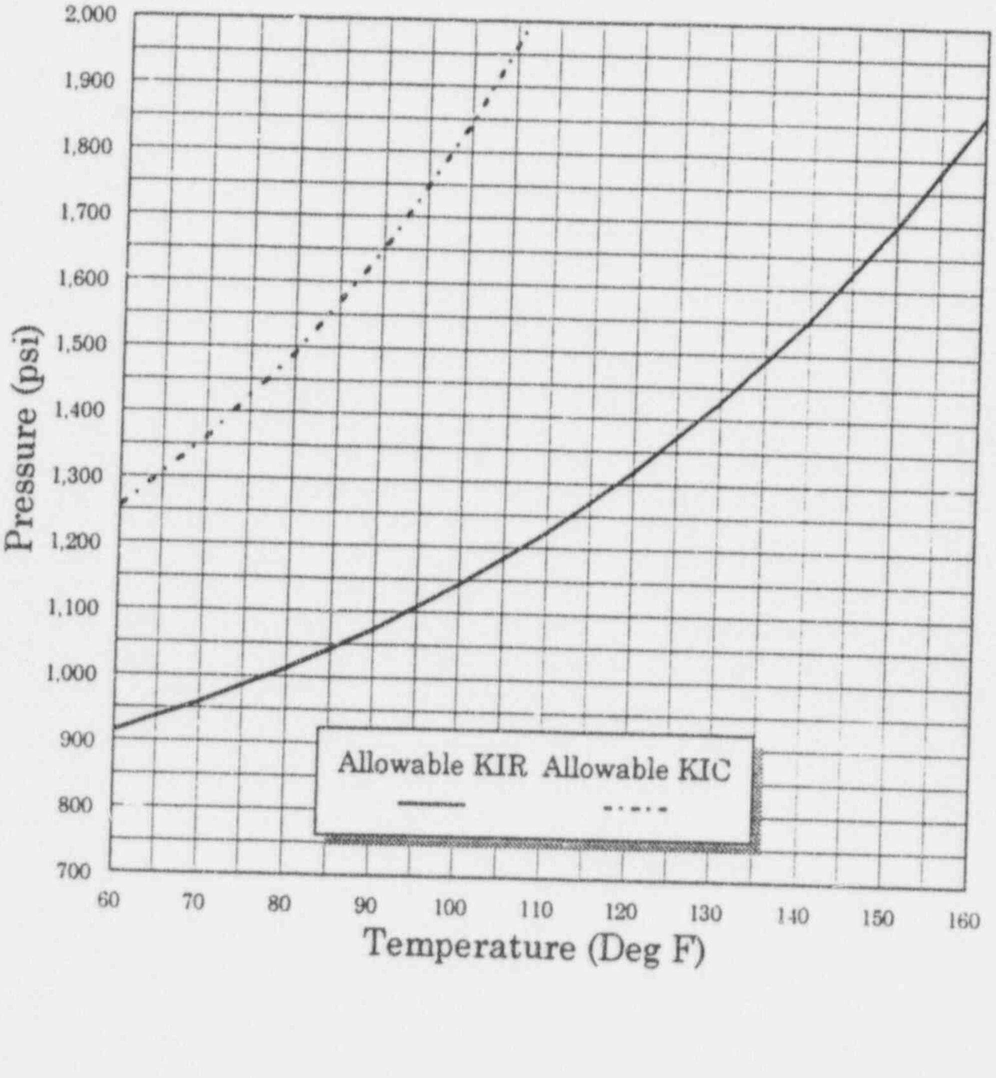

Figure 1: Pressure-Temperature Limit Curves for Isolated Loop: Haddam Neck