August 19, 1982

Docket No. 50-213
LS05-82-08-041

Mr. W. G. Counsil, Vice President
Nuclear Engineering and Operations
Connecticut Yankee Atomic Power Co.
Post Office Box 270
Hartford, Connecticut 06101

Dear Mr. Counsil:

SUBJECT: SEP TOPIC XV-17, STEAM GENERATOR TUBE RUPTURE (SYSTEM AND RADIOLOGICAL CONSEQUENCES) HADDAM NECK PLANT

By letter dated April 7, 1982, you submitted a safety assessment report for both the Systems and Radiological Consequences of Topic XV-17. The staff has reviewed these assessments and our conclusions are presented in the enclosed safety evaluation reports (SERs).

Systems

As noted in the enclosed evaluation, it is the staff's position that the licensee should provide additional justification that the assumed operator action times for charging pump initiation, auxiliary feedwater initiation and isolation, are appropriate to provide a conservative assessment of system performance and potential radiological consequences. The licensee should provide their emergency prodedures for this event. The licensee should also reconcile the mass imbalance and slowly falling pressurizer level with the assertion that there is no upper head voiding.

Radiological Consequences

For reasons given in the SER it is the staff's position that the Westinghouse Standard Technical Specification requirements for primary and secondary coolant iodine specific activity as well as associated surveillance requirements should be adopted and implemented.

Add: Gary Staley
T. Michaels
S. Brown

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The need to actually implement changes will be determined during the Integrated Safety Assessment. The evaluation may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the Integrated Assessment for this plant is completed.

Sincerely,

Original signed by:

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosures:
As stated

cc w/enclosures:
See next page

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OFFICE	SEPB	SEPB	SEPB	SEPB	SEPB	ORB#5	ORB#5
SURNAME	EMcKenna:bl	TMichaels	SBrown	CGrimes	WRussell	CTropf	DCrutchfield
DATE	8/11/82	8/11/82	8/12/82	8/16/82	8/16/82	8/17/82	8/18/82

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

Mr. W. G. Counsil

cc
Day, Berry & Howard
Counselors at Law
One Constitution Plaza
Hartford, Connecticut 06103

Superintendent
Haddam Neck Plant
RFD #1
Post Office Box 127E
East Hampton, Connecticut 06424

Mr. Richard R. Laudenat
Manager, Generation Facilities Licensing
Northeast Utilities Service Company
P. O. Box 270
Hartford, Connecticut 06101

Board of Selectmen
Town Hall
Haddam, Connecticut 06103

State of Connecticut
OFfice of Policy and Management
ATTN: Under Secretary Energy
Division
80 Washington Street
Hartford, Connecticut 06115

U. S. Environmental Protection Agency
Region I Office
ATTN: Regional Radiation Representative
JFK Federal Building
Boston, Massachusetts 02203

Resident Inspector
Haddam Neck Nuclear Power Station
c/o U. S. NRC
East Haddam Post Office
East Haddam, Connecticut 06423

Ronald C. Haynes, Regional Administrator
Nuclear Regulatory Commission, Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

ENCLOSURE

HADDAM NECK PLANT

SEP TOPIC XV-17

Steam Generator Tube Failure

(Systems)

I. INTRODUCTION

In the event of a steam generator tube rupture (SGTR), primary coolant will be discharged to the steam generator secondary coolant until the primary system pressure is reduced to equal the pressure in the steam generator secondary side. The discharge of primary coolant can result in a radioactive release to the environment via the turbine condenser air ejector or from the relief and safety valves if offsite power is not available and the condenser is isolated, or if appropriate actions are not taken by the operator to prevent discharge through these potential release paths.

II. REVIEW CRITERIA

Section 50.34 of 10 CFR Part 50 requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. The steam generator tube rupture is one of the postulated accidents used to evaluate the adequacy of these structures, systems, and components with respect to public health and safety.

Section 50.36 of 10 CFR requires the Technical Specifications to include safety limits which protect the integrity of the physical barriers which guard against the uncontrolled release of radioactivity.

In addition, 10 CFR Part 100.11 provides an acceptable dose consequence limit for reactor siting.

III. RELATED SAFETY TOPICS

Topic V-8, "Steam Generator Integrity" ensures that acceptable levels of integrity of the steam generator are maintained in accordance with current criteria. Various other SEP topics evaluate such items as ESF systems.

IV. REVIEW GUIDELINES

The review of the consequences was conducted in accordance with Standard Review Plan 15.6.3. The plant is considered adequately designed against a steam generator tube failure if system and operator response is possible such that the resulting doses at the exclusion area and low population zone boundaries are less than a small fraction of the 10 CFR Part 100 exposure guidelines, and are within the 10 CFR Part 100 guidelines for the case of a preaccident iodine spike. The calculated doses based on the plant response are the subject of a separate evaluation.

V. EVALUATION

The steam generator tube rupture event was analyzed assuming an initial power level of 102%, single failure of the only atmospheric dump valve (failure to open) and loss of offsite power concurrent with the tube rupture. The loss of offsite power is assumed to cause a reactor trip, reactor coolant pump trip and feedwater pump trip. A single charging pump is assumed to be manually restarted by the operator at 60 seconds into the transient following loss of offsite power. The auxiliary feedwater pumps are assumed to be automatically started on low steam generator water level at 350 seconds (the licensee has assumed that the main feed pump breakers fail to start the auxiliary feed pumps following loss of offsite power as they are designed to do). The licensee's analysis also assumes the operator manually stops auxiliary feed flow to the affected steam generator at the same time (350 seconds). Isolation of the affected steam generator and cooldown initiation occurs at 1800 seconds. The leak flow from the broken tube is assumed to continue until the affected steam generator and the primary pressures equalize at about 7500 seconds.

The licensee has stated that these assumptions were made to provide a conservative assessment of the potential radiological consequences of this event. In particular, the assumptions were selected to maximize break flow, minimize auxiliary feedwater flow to the affected generator so as to minimize the dilution and to maximize steam release from the affected steam generator.

As part of the SEP review of Haddam Neck, we have evaluated the licensee's analysis and discussed with the licensee the assumptions used and the results. Based on our evaluation, we cannot conclude that the assumed operator actions are suitably conservative, or that the predictions of system performance acceptably bound the range of possible system conditions. In particular, the licensee's assumption of operator action within 60 seconds to restart the charging pump should be justified. The licensee stated that this action was necessary to regain pressurizer level for pressure control and that this assumption is conservative because it maintains high primary pressure and thus a high leak flow rate. However, if the pressurizer level drops to a very low level if no charging flow is assumed, voids may be formed elsewhere in the primary system. Pressure control system may then be difficult, especially with the assumed loss of offsite power. The licensee should justify this assumed operator action and associated time. In the absence of justification, we believe the operator action times presented in ANSI draft standard N660 should be used.

Furthermore, we have evaluated the licensee's automatic auxiliary feedwater initiation at about 350 seconds, and the assumption that the operator manually isolates auxiliary feedwater flow to the damaged steam generator at that time. Based on our review, we cannot conclude that the assumed operator action times are acceptable.

Additionally, we do not believe that the results of the analyses are conservative regarding the assumption of failure of the main feed pump breakers to initiate auxiliary feedwater flow.

Additional feedwater to the affected generator would result in a faster increase in steam generator level. The partitioning effect of the generator would diminish when the generator level rises to the separator region. Furthermore, if the steam generator fills too far, water may overflow into the steam lines and may then overstress the lines or be released as water through the steam relief/safety valves.

The licensee should justify the assumptions concerning auxiliary feedwater flow to the affected generator or reanalyze the event using operator action times consistent with ANSI N660 and assuming auxiliary feedwater is initiated by the first initiating event (i.e., main feed pump breaker opening).

The licensee's analysis does not predict any void formation in the RCS during the event, but the pressurizer level response does not appear to reflect the mass flow in from the charging pump and the mass flow out through the broken tube. The level drops only about 10% over a 30 minute span, although a mass balance predicts a net loss of about 10 lbm/sec over this interval. The licensee should explain the predicted break flow and pressurizer level response.

VI. CONCLUSION

Therefore, in order to determine the ability of the plant to mitigate the consequences of a SGTR, we request that the licensee either provide the justification discussed above or reanalyze the event assuming operator actions consistent with ANSI-N660. The ANSI N660 times assumed should be consistent with the licensees event categorization of the SGTR event. Additionally, in order to better understand the operator actions and how they affect the plant, we request that the licensee submit emergency procedures for this event.

Until the above concerns are resolved, we cannot conclude that the predicted system performance provides a conservative basis for assessment of potential radiological consequences.

VII. REFERENCE

1. Letter from W. G. Counsil to D. M. Crutchfield, dated April 7, 1982.

HADDAM NECK NUCLEAR POWER PLANT

XV-17, STEAM GENERATOR TUBE FAILURE ACCIDENT

(RADIOLOGICAL CONSEQUENCES)

I. INTRODUCTION

Steam generator tube failures allow the escape of radioactivity from the reactor coolant system to the environment. SEP Topic XV-17 is intended to review the radiological consequences of a steam generator tube failure. The review encompasses those design features which limit the release of radioactivity, including the plant technical specifications associated with coolant activity concentrations.

II. REVIEW CRITERIA

Section 50.34 of 10 CFR Part 50 requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. The steam generator tube failure accident is one of the postulated accidents used to evaluate the adequacy of these structures, systems, and components with respect to public health and safety.

In addition, 10 CFR Part 100.11 provides dose consequence guidelines for reactor siting.

III. RELATED SAFETY TOPICS

Topic II-2.C, "Atmospheric Transport and Diffusion Characteristics for Accident Analysis" provides the meteorological data used to evaluate the potential offsite doses. The staff has received the information submitted for this topic and has calculated atmospheric dispersion factors which are acceptable for use in evaluating the radiological consequences of postulated accidents. Topic V-8, "Steam Generator Tube Integrity" ensures that acceptable levels of integrity of the steam generator are maintained in accordance with current criteria.

IV. REVIEW GUIDELINES

The review of the radiological consequences was conducted in accordance with Standard Review Plan (SRP) Section 15.6.3. The plant is considered adequately designed against a steam generator tube failure if calculations show that the resulting radiological consequences at the Exclusion Area and Low Population Zone Boundaries are less than a small fraction of the 10 CFR Part 100 exposure guidelines for the case of an accident induced iodine spike, and are within the 10 CFR Part 100 guidelines for the case of a preaccident iodine spike.

The offsite dose calculations are based on the Westinghouse Standard Technical Specification limits for primary and secondary coolant iodine concentrations.

V. EVALUATION

In a letter to NRC dated April 7, 1982, the licensee transmitted their analysis of the radiological consequences following a postulated steam generator tube rupture (SGTR) accident. In the analysis, the licensee assumed Westinghouse Standard Technical Specification (STS) values for primary and secondary coolant iodine activities and assumed that the atmospheric dump valve on the affected steam generator failed to open. The latter assumption permitted releases from the affected steam generator through the safety valve to the environment for a period of approximately 7500 seconds. The licensee's calculated radiological consequences are presented in Table 1, and are less than the acceptance criteria given in SRP Section 15.6.3 and the guideline values of 10 CFR Part 100.

In accordance with SRP Section 15.6.3 (Revision 2), the staff also performed an independent analysis of the radiological consequences following a SGTR accident. The licensee's current technical specifications for primary coolant activity do not contain either an equilibrium or a maximum limit for iodine activity, but rather contains an equilibrium limit for total activity (all isotopes with half-lives greater than 1/2 hour) in the coolant of 68/E μCi/ml.

Even though the licensee used the Westinghouse STS limits on iodine, the absence of any plant technical specification limits on the iodine activity in the primary and secondary coolants makes it impossible to assure that operation of Haddam Neck would be restricted such that the exposure guidelines of 10 CFR Part 100 would not be exceeded in the event of a SGTR accident. The staff also used the Westinghouse STS coolant (primary and secondary) activity limits and recommends that they be implemented.

Because the staff is not completely satisfied with the licensee's evaluation in the areas of operator actions, primary system response and overfilling of the steam generator (see the systems portion of this SEP Topic), the staff performed an evaluation of the radiological consequences following a SGTR accident using what are believed to be conservative assumptions as outlined below.

For Case 1, the staff assumed that an iodine spike had occurred prior to the assumed SGTR accident and had raised the primary coolant specific activity to 60 μCi/gm of dose equivalent I-131 (DEI-131). The licensee has estimated that 194,000 pounds of primary coolant will be leaked to the secondary system during the 7500 second period of leakage. The staff has estimated the total integrated flow (using Figure 5 of the licensee's submittal) and concludes that 194,000 pounds is a reasonable estimate of the leakage for use in its calculation. The staff has assumed that all of the iodine in the primary-to-secondary leakage would be released to the environment via the stuck-open safety/relief valve. No credit for partitioning or iodine scrubbing in the affected steam generator was assumed.

To assume that all primary-to-secondary leakage is released to the environment for a steam generator tube rupture accident is extremely conservative for two reasons. First, the steam generator level usually increases instead of decreases and, therefore, some iodine partitioning and iodine scrubbing by the bulk water in the steam generator should occur. Second, if the steam generator overfills and begins releasing liquid through the safety valves, the iodine concentration in the water of the steam generator will not be as high as the initial iodine concentration in the primary system (which was assumed in this analysis) if the dilution effects of the initial mass in the affected steam generator and the incoming charging/safety injection water into the primary system are considered. Consideration of these effects will be determined from the licensee's response to the issues identified in the systems portion of this topic.

Because the licensee did not provide an estimate of the steam released during the plant cooldown, the staff estimated the amount of potential steam dump to the environment for the Haddam Neck plant and conservatively assumed that all the secondary steam released had an iodine specific activity equal to 0.1 uCi/gm DEI-131. The appropriate steam dump values used in the staff's calculations are presented in Table 2 of this evaluation.

Also as required in the Standard Review Plan, the staff assumed that the Haddam Neck Technical specification primary-to-secondary leak rate in the steam generators of 0.4 gpm existed in the unaffected steam generators. A decontamination factor of 100 was assumed for the leakage occurring in these steam generators.

The 0.4 gpm leakage is assumed to occur for a period of eight hours.

Using the conservative assumptions outlined above and the atmospheric dispersion factors calculated under SEP Topic II-2.C, the resultant offsite radiological consequences are less than the guideline values of 10 CFR Part 100 at the outer boundary of the Low Population Zone, but would greatly exceed the guideline values of 10 CFR Part 100 at the Exclusion Area Boundary.

In the second case (Case 2), the staff evaluated the radiological consequences of a SGTR accident assuming that the accident initiates an iodine spike which increases the iodine specific activity in the coolant at a rate of 30 μCi/gm-hr. Using the Westinghouse STS equilibrium value for primary coolant iodine activity of 1.0 μCi/gm (DEI-131), and the other appropriate assumptions described above, the resultant estimated offsite radiological consequences are estimated to be substantially greater than a small fraction of the 10 CFR Part 100 guideline values at both the Exclusion Area and Low Population Zone Boundaries and, therefore, exceed the acceptance criteria of SRP Section 15.6.3 (Revision 2).

The staff's and licensee's values for the thyroid doses at the Exclusion Area and the Low Population Zone Boundaries are presented in Table 1. The whole body doses are small and because they do not approach the acceptance limits, the whole body doses are not presented in Table 1.

VI. CONCLUSION

Based upon the licensee's calculations, the radiological consequences resulting from a steam generator tube rupture at the Exclusion Area and the Low Population Zone boundaries are less than a small fraction of the 10 CFR Part 100 exposure guideline values for the case of an accident induced iodine spike; and are within the 10 CFR Part 100 guideline values for the case of a preaccident iodine spike; and, therefore, comply with the acceptance criteria of SRP Section 15.6.3 (Revision 2). The licensee's calculations are based upon the Westinghouse Standard Technical Specification limits for primary and secondary coolant iodine concentrations and a plant specific specification for primary-to-secondary leakage in the steam generator.

Because the staff cannot conclude that the assumed operator actions are suitably conservative or that the predictions of system performance acceptably bound the range of possible system conditions, the staff performed a radiological consequence evaluation of the SGTR accident assuming two iodine spiking cases, the Westinghouse STS for primary and secondary coolant iodine, and extremely conservative iodine transport assumptions (i.e., all the iodine in the primary-to-secondary leakage into the affected steam generator is released to the environment). In summary, the staff's calculated radiological consequences at the Exclusion Area Boundary for Cases 1 and 2 exceed the guideline values of 10 CFR Part 100. The calculated radiological consequences at the Low Population Zone Boundary for Cases 1 and 2 are less than the guideline values of 10 CFR Part 100. However, for Case 2 the calculated radiological consequences exceed the acceptance criteria of SRP Section 15.6.3 (Revision 2).

It is the staff's position that the licensee should implement Westinghouse Standard Technical Specification requirements for primary and secondary coolant iodine specific activity as well as associated surveillance requirements.

With these requirements and subject to satisfactory resolution of the staff concerns identified in the systems discussion, the staff concludes that the radiological consequences of a steam generator tube rupture accident will be shown to be acceptable.

TABLE 1

THYROID DOSES FOLLOWING A SGTR ACCIDENT

	Thyroid Dose (Rem) EAB	LPZ
Licensee's Values		
Case 1 (Initial coolant activity = 60 μCi/gm DEI-131)	18	0.5
Case 2 (Accident induced iodine spike)	8.7	0.2
Staff Values		
Case 1 (Initial coolant activity = 60 μCi/gm DEI-131)	2290	148
Case 2 (Accident induced iodine spike)	1170	75

TABLE 2

ASSUMPTIONS MADE IN THE ANALYSIS OF THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE ACCIDENT

1. Reactor power = 1825 Mwt.
2. Loss of offsite power following the accident.
3. Iodine decontamination factor of 1 between the water and steam for primary-to-secondary leakage in the affected steam generator.
4. Iodine decontamination factor in the unaffected steam generators of 100.
5. Primary-to-secondary leak rate of 0.13 gpm to each of the unaffected steam generators for a period of 8 hours.
6. Primary coolant iodine specific activity prior to the accident of 60 μCi/gm DEI-131 for Case 1 and 1.0 μCi/gm DEI-131 for Case 2.
7. Secondary coolant iodine specific activity prior to the accident of 0.1 μCi/gm of DEI-131 (for both cases)
8. Iodine spiking factor in the primary coolant of 30 μCi/gm per hour DEI-131 for the first four hours (for Case 2, the accident induced iodine spike).
9. Atmospheric dispersion factors (sec/cubic meter) from SEP Topic II-2.C:

   Exclusion Area Boundary (0 - 2 hour) = 8.4 E-4/sec

   Low Population Zone Boundary (0 - 8 hour) = 5.4 E-5/sec
10. Total secondary coolant assumed to be released to the environment:

    0 - 2 hours = 300,000 lbs

    0 - 8 hours = 730,000 lbs