September 30, 1982

Docket No. 50-213
LS05-82-09-094

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

Dear Mr. Counsil:

SUBJECT: SUMMARY OF SEP TOPIC DIFFERENCES - HADDAM NECK PLANT

Enclosure 1 is a listing of all of the SEP topics for which Haddam Neck does not meet the current licensing acceptance criteria. Enclosure 2 is a summary description of each topic difference, except for Topics III-6, "Seismic Design Considerations," and III-5.A, "Effects on Pipe Break on Structures, Components and Systems Inside Containment." The summary descriptions for these topics will soon be completed and issued. A full description of each of the differences may be found in the respective topic safety evaluation reports.

Some of the differences are based on draft safety evaluation reports. Therefore, the status of some of those topics and the summary of differences may be revised pending your confirmation that the facts upon which the staff based their draft evaluations are correct or require revision.

Sincerely,

Original signed by:

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

SE04
DSU USE(02)
ADD:
G. Staley

Enclosure:
As stated

cc w/enclosure:
See next page

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*See previous yellow for additional concurrences.

OFFICE	SEPB:DL	SEPB:DL	SEPB:DL	ORB#5:PM	ORB#5:BC	AD:SA:DL	
SURNAME	SBrown:dk*	RHermann*	WRussell	JLyons	DCrutchfield	FMiraglia	
DATE	9/28/82	9/29/82	9/29/82	9/30/82	9/30/82	9/30/82	

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Some of the differences are based on recently completed topic reviews. The safety evaluation reports for those topics will be issued within two weeks. Therefore, the status of some of those topics and the summary of differences may be revised pending your confirmation that the facts upon which the staff based their evaluations are correct or require revision.

Sincerely,

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
As stated

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DATE	9/13/82	9/29/82	9/ /82	9/ /82	9/ /82	9/ /82	

Mr. W. G. Counsil

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LISTING OF SEP TOPIC EVALUATIONS WITH DIFFERENCES FOR HADDAM NECK PLANT

Topic No.	Title
II-3.B(*)	Flooding Potential and Protection Requirements
II-3.B.1(*)	Capability of Operating Plant to Cope with Design Basis Flooding Conditions
II-3.C(*)	Safety Related Water Supply (UHS)
III-1 (S)(*)	Classification of Structures, Components and Systems (Seismic and Quality)
III-2(*)	Wind and Tornado Loadings
III-3.A	Effects of High Water Level on Structures
III-3.B(*)	Structural and Other Consequences of Failure of Underdrain Systems
III-3.C	Inservice Inspection of Water-Control Structures
III-4.A	Tornado Missiles
III-4.C(*)	Internally Generated Missiles
III-5.B	Pipe Break Outside Containment
III-7.B(*)	Design Codes, Design Criteria, Load Combinations and Reactor Cavity Design Criteria
III-8.A	Loose Parts Monitoring and Core Barrel Vibration Monitoring
III-10.A(*)	Thermal-Overload Protection for Motors of Motor-Operated Valves
III-10.B	Pump Flywheel Integrity
IV-2	Reactivity Control Systems Including Functional Design and Protection Against Single Failure
V-5	Reactor Coolant Pressure Boundary Leakage Detection

Topic No.	Title
V-6	Reactor Vessel Integrity
V-10.A	RHR Heat Exchanger Tube Failures
V-10.B	RHR Reliability
V-11.A	Requirements for Isolation of High and Low Pressure Systems
V-11.B(S)	RHR Interlock Requirements
VI-1	Organic Materials and Post-Accident Chemistry
VI-4(S)(*)	Containment Isolation System
VI-7.B	ESF Switchover from Injection to Recirculation Mode (Automatic ECCS Realignment)
VI-7.C.1	Independence of Redundant Onsite Power Systems
VI-10.A(*)	Testing of Reactor Trip System and Engineered Safety Features Including Response Time Testing
VII-1.A(*)	Isolation of Reactor Protection System from Non-Safety Systems, Including Qualification of Isolation Devices
VII-3(E)	Systems Required for Safe Shutdown
VIII-1.A	Potential Equipment Failures Associated with a Degraded Grid Voltage
VIII-2	Onsite Emergency Power System - Diesel Generator
VIII-3.A	Station Battery Test Requirements
VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation
IX-3	Station Service and Cooling Water Systems
IX-5	Ventilation Systems
XV-2(R)(*)	Spectrum of Steam System Piping Failures Inside and Outside Containment
XV-7	Reactor Coolant Pump Rotor Seizure and Shaft Break
XV-12(R)	Radiological Consequences of a Rod Ejection Accident

Topic No.	Title
XV-16(R)(*)	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment
XV-17(R,S)	Steam Generator Tube Rupture (PWR)
XV-19(R)	LOCAs Resulting from Spectrum of Postulated Piping Breaks Within the RCPB

Legend

E - Electrical aspects
R - Radiological aspects
S - Systems aspects
* - Draft SER

Enclosure 2

SUMMARY DESCRIPTION

OF SEP TOPIC DIFFERENCES FOR

HADDAM NECK PLANT

TOPIC NO.	TITLE
II-3.B	Flooding Potential and Protection Requirements
II-3.B.1	Capability of Operating Plant to Cope with Design Basis Flooding Conditions
II-3.C	Safety Related Water Supply (Ultimate Heat Sink)
III-3.B	Structural and Other Consequences of Failure of Underdrain Systems

10 CFR 50 (GDC 2 and 44) and 10 CFR 100, as implemented by SRP Sections 2.4.3, 2.4.4, 2.4.5, 2.4.7, 2.4.10, 2.4.11, and 9.2.5, Regulatory Guides 1.27 and 1.59, and ANSI N170, require, in part, that structures, systems and components (including the utlimate heat sink) important to safety be designed to withstand the effects of floods and consider hydrologic characteristics in the evaluation of the site.

The staff has determined that the following items do not meet current licensing acceptance criteria:

1. Roof Flooding - The design live load for the service building roof could be exceeded during rainfalls less severe than the PMP.

2. Site Flooding - The original site design basis flood on the Connecticut River was 19.5 ft msl with maximum external protection designed to be at elevation 21.5 ft msl. The probable maximum flood (PMF) on the Connecticut River is estimated to have flood elevation of 39.5 ft msl at the site and a standard project flood (SPF) is estimated to have an elevation of approximately 23.2 ft msl. Failure of upstream dams either during a PMF or as a separate flood producing event has not been addressed by the licensee.

   However, protection to 39.5 ft msl is not practical and thus the licensee has proposed protection to 30 ft msl which is the highest protection possible if building walls are able to structurally withstand the flood waters. This level is 6.8 ft greater than the SPF, but 9.5 ft less than PMF. Protection to only 30 ft msl would not meet current NRC criteria.

3. Groundwater - The maximum groundwater elevation for hydrostatic load will be the PMF level (39.5 ft msl). The normal high groundwater elevation for use in combination with appropriate seismic conditions is plant grade (21.0 ft msl). No credit is given for control of groundwater levels by the underdrain system.

4. Emergency Procedures - The licensee's proposed emergency flood procedure does not provide protection to the current NRC licensing flood level (PMF - elevation 39.5 ft msl). Recommendations for upgrading the emergency procedure to provide protection to 30.0 ft msl are given in the TER appended to the Report.

5. Ultimate Heat Sink - The Haddam Neck ultimate heat sink complex would not function during two postulated low water events in the Connecticut River. Full compliance with Regulatory Guide 1.27 has not been demonstrated.

6. Underdrain System - The mat sump system is not safety grade, and failure could enable groundwater rise to plant grade (see SEP Topic II-3.B). An evaluation under SEP Topic III-3.A using new groundwater elevation at plant grade is recommended.

TOPIC NO.	TITLE
III-1	Classification of Structures, Components and Systems (Seismic and Quality)

10 CFR 50 (GDC 1), as implemented by Regulatory Guide 1.26, requires that structures, components and systems important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of safety functions to be performed. The codes used for the design, fabrication, erection, and testing of Haddam Neck were compared with current codes.

The review of this topic identified several systems and components where insufficient information is available to justify a conclusion that the quality standards imposed during plant construction meet quality standards required for new facilities.

The staff safety evaluation of August 12, 1982, requested the licensee to provide information in the following areas:

1. Radiography requirements
2. Fracture toughness
3. Valves
4. Pumps
5. Storage tanks
6. Piping
7. Codes and standards
8. Pressure vessels

TOPIC NO. TITLE

III-2 Wind and Tornado Loadings

10 CFR 50 (GDC 2), as implemented by SRP Sections 3.3.1, 3.3.2 and 3.8 and Regulatory Guides 1.76 and 1.117, requires, in part, that safety-related structures, components and systems be adequately designed to resist wind and tornado loadings, including tornado pressure drop loading.

In the staff's safety evaluation, it was concluded that portions of some structures cannot withstand the postulated design basis tornado loads of 300 mph winds and 2.25 psi pressure drop.

The licensee should either implement modifications for the following structures or portions of structures, or demonstrate that the consequences of their failure if subjected to tornado loads are acceptable:

1. Upper portion of the primary auxiliary building.

2. Ventilation stack.

3. Interior masonry walls protected by exterior walls with minimal tornado resistance (e.g., siding).

4. Auxiliary feedwater pumphouse (structural portion and siding system).

5. Screenwell house (structural portion and siding system).

6. Service building (structural portion and siding system).

7. Roof decks on Category 1 structures.

8. Siding system on any other Category 1 structures.

9. New and spent fuel pool superstructure.

For safety-related components not inside qualified structures, the licensee should either demonstrate acceptability for tornado loads or that the consequences of failure if subjected to tornado loads are acceptable.

It should be determined whether operating pipe reaction loads, thermal loads and snow loads were considered with wind in the original design. If these loads were not, the effect of combining them should be addressed.

The licensee should demonstrate that foundations and soil capacities are greater than original design and that they are not limiting.

TOPIC NO.	TITLE
III-3.A	Effect of High Water Level on Structures

10 CFR 50 (GDC 2), as implemented by SRP Section 2.4.12, requires, in part, that the plant be designed for high water levels, including the dynamic effects.

On the basis of SEP Topics II-3.A and II-3.B, the design basis flood level is expected to be 20 feet over plant grade. For this flood level, the Category I structures will be damaged and some possibly destroyed. The levels of damage may vary, but the overall conclusion is that the postulated situation would be structurally unacceptable.

TOPIC NO.	TITLE
III-3.C	Inservice Inspection of Water-Control Structures

10 CFR 50 (GDC 1, 2 and 44) and 10 CFR 100 (Appendix A), as implemented by SRP Sections 2.5.4 and 2.5.5, and Regulatory Guides 1.27, 1.28, 1.59, 1.127, and 1.132, require, in part, that water-control structures built for use in conjunction with a nuclear power plant, whose failure could cause adverse radiological consequences, be inspected routinely.

Haddam Neck has met the acceptance criteria for this topic with the following exceptions:

1. Comprehensive report forms should be developed to convey field inspection information to the appropriate inspection program manager.

2. Criteria for initiating "special inspections" should be developed to ascertain the integrity of structures after the occurrence of extreme environmental event.

3. Inspection frequencies for each item should be established and included in the formal documentation.

4. Inspections should be performed by qualified technical personnel and directed by qualified engineering personnel.

5. A program for technical review and evaluation of inspection reports should be established.

TOPIC NO. TITLE

III-4.A Tornado Missiles

10 CFR 50 (GDC 2 and 4), as implemented by SRP 3.5.1.4 and Regulatory Guides 1.13, 1.27, 1.76, and 1.117, requires, in part, that structures, components and systems essential to safety be designed to withstand natural phenomena, such as tornadoes and their missiles.

Based upon the staff review, we conclude that Haddam Neck does not meet the current licensing criteria for tornado missile protection in the following areas:

1. Atmospheric dump valve (ADV) and associated steam vent path piping located in the auxiliary feedwater building.

2. Main steam and feedwater isolation valves.

3. Auxiliary feedwater system.

4. Water sources - demineralized water storage tank, primary water storage tank and primary water transfer pump.

5. Service water system.

6. Emergency switchgear room including portions of the emergency power distribution system.

7. Safe shutdown instrumentation.

8. Control air system.

9. Control rod drive system.

10. Life support equipment for the control room.

TOPIC NO.	TITLE
III-4.C	Internally Generated Missiles

10 CFR 50 (GDC 4), as implemented by SRP Sections 3.5.1.1 and 3.5.1.2 and Regulatory Guides 1.13 and 1.27, requires in part, that components and systems essential to safety be protected from internally generated missiles.

Based on our review of the systems and components needed to perform safety functions, we conclude that the design of protection from internally generated missiles meet the intent of current licensing criteria, except that the essential 480 volt switchgear and the station batteries are not adequately protected from the internally generated missiles.

TOPIC NO.	TITLE
III-5.B	Pipe Break Outside Containment

10 CFR 50 (GDC 4), as implemented by SRP Sections 3.6.1 and 3.6.2 and BTP MEB 3-1 and ASB 3-1, requires, in part, that structures, components and systems important to safety be appropriately protected against dynamic effects, including the effects of pipe whipping and discharging fluids, that may result from equipment failures.

The staff has determined that Haddam Neck Plant is adequately protected against the dynamic effects of pipe break outside containment except for the following four issues which remain to be resolved.

1. Verification that flooding and spray effects of leakage cracks have been fully addressed.

2. Evaluation of postulated breaks in the auxiliary feedwater system.

3. Clarification of the jet impingement criteria utilized in the evaluation of piping in the primary auxiliary building.

4. Evaluation of the effects of turbine extraction steam line breaks on the switchgear room.

TOPIC NO.	TITLE
III-7.B	Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria

10 CFR 50 (GDC 1, 2 and 4), as implemented by SRP Section 3.8, requires, in part, that structures, components and systems be designed for the loading that will be imposed on them and that they conform to applicable codes and standards.

Code, load and load combination changes affecting specific types of structural elements have been identified where existing safety margins in structures are significantly reduced from that which would be required by current versions of the applicable codes and standards. The differences between plant design and current licensing criteria should be resolved as follows:

1. Review of Seismic Category I Structures at Haddam Neck to determine if any of the structural elements for which a concern exists are a part of the facility design of Haddam Neck. For those that are, assess the impact of the code changes on margins of safety on a plant specific basis.

2. Examine on a sampling basis the margins of safety of Seismic Category I Structures for loads and load combinations not covered by another SEP topic and denoted by "Ax" in the SER. (The load tables should be reviewed to assure their technical accuracy concerning applicability of the loads for each of the structures and their significance. The Category I structures considered should be reviewed to assure completeness.)

TOPIC NO.	TITLE
III-8.A	Loose Parts Monitoring and Core Barrel Vibration Monitoring

10 CFR 50 (GDC 13), as implemented by Regulatory Guide 1.133, Revision 1, and SRP Section 4.4, prescribes a loose parts monitoring program for the primary system of light-water-cooled reactors.

Haddam Neck does not have a loose parts monitoring program that meets the criteria of Regulatory Guide 1.133.

TOPIC NO.	TITLE
III-10.A	Thermal-Overload Protection for Motors of Motor-Operated Valves

10 CFR 50 Appendix A (GDC 13, 21, 22, 23, and 29), as implemented by IEEE Std. 279-1971, requires, in part, that protective actions be reliable and precise and satisfy the single failure criterion using quality components. Regulatory Guide 1.106 presents the staff position on how thermal-overloads can be made to meet these requirements.

Thermal-overload protection for motor-operated valves at Haddam Neck does not satisfy current licensing requirements. Thermal-overload devices are not bypassed, no information is available to support adequacy of trip setpoints, and torque switches rather than limit switches are used to terminate valve travel.

TOPIC NO.	TITLE
III-10.B	Pump Flywheel Integrity

10 CFR 50 (GDC 4), as implemented by SRP 5.4.1.1 and Regulatory Guide 1.14 recommended, in part, methods to minimize the potential for failures of reactor coolant pump flywheels.

Adequate information to determine the extent of inspections was not provided.

TOPIC NO. TITLE

IV-2 Reactivity Control Systems Including Functional Design and Protective Against Single Failures

10 CFR 50 (GDC 25), as implemented by SRP Section 15.4.3, requires, in part, that the reactor protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal of control rods.

Based upon an audit review of the information provided by CYAPCo, the staff has determined that the following may occur as a result of single failures:

1. Two banks of control rods may move simultaneously instead of one bank.

2. Two subgroups of control rods could move simultaneously instead of one subgroup.

3. A cluster, subgroup, or bank of shutdown rods may not move when movement is commanded.

4. A cluster, clusters, subgroup, bank, or banks of control rods may not move when movement is commanded.

5. One bank of shutdown rods could move inadvertently.

6. A subgroup, bank or banks (in overlap region) of control rods could move inadvertently.

7. An individual shutdown rod or a cluster, subgroup, bank, or banks of shutdown rods could fall into the core.

8. An individual control rod of cluster, clusters, subgroup, bank, or banks of control rods could fall into the core.

It was the staff's conclusion that CYAPCo should revise the evaluation of Topic XV-8 to include the eight items listed above or show why these types of failures cannot occur at Haddam Neck.

TOPIC NO.	TITLE
V-5	Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

10 CFR 50 (GDC 30), as implemented by Regulatory Guide 1.45 and SRP Section 5.2.5, prescribes the types and sensitivity of systems, as well as their seismic, indication and testability criteria, necessary to detect leakage of primary reactor coolant to the containment or to other interconnected systems. Reliable and sensitive leakage detection systems are required in order to identify primary system leaks at an early stage before failure occurs.

Based upon our review of the information available for Haddam Neck, we have determined that the systems employed for the detection of leakage from the reactor coolant pressure boundary to the containment do not meet all of the recommendations of Regulatory Guide 1.45, specifically.

1. Information indicates that the systems incorporated for measurement of leakage from the reactor coolant pressure boundary to the containment do not conform with Regulatory Guide 1.45 criteria regarding sensitivity and seismic qualification.

2. Standard Technical Specification 3/4.4.6 and the corresponding surveillance requirements concerning the operability of the reactor coolant pressure boundary to the containment leakage detection systems (excluding the sump flow monitor) should be added to the Haddam Neck Technical Specifications. Also, the current Haddam Neck Technical Specification 3.14 should be revised to state that the sensitivities of the reactor coolant pressure boundary to containment leakage detection system is 1 gpm within 1 hour for Items 1, 4 and 7 in Table 1 of topic evaluation.

3. Information concerning the leakage detection systems for the detection of intersystem reactor coolant pressure boundary leakage and the reactor coolant inventory balance is incomplete. Therefore, we cannot determine the extent to which Regulatory Guide 1.45 is met.

TOPIC NO.	TITLE
V-6	Reactor Vessel Integrity

10 CFR 50.55a(c) requires that pressure vessels which are part of the reactor coolant pressure boundary meet the requirements for Class A vessels set forth in Section III of the ASME Boiler and Pressure Vessel Code, applicable Code Cases, and Addenda.

The staff has recommended the following actions be taken in order to assure continued acceptability of reactor vessel materials throughout the expected plant service life:

1. The last two material surveillance capsules removed from Haddam Neck contained no weld metal samples. Therefore, it is recommended that another capsule be removed in the next several years. This capsule should contain weld metal specimens.

2. The present pressure-temperature operating limits are based on the extrapolation of data obtained from the material surveillance program. Since a capsule subjected to relatively high fluences has recently been removed from the vessel, we should have in the near future a better data base to estimate the amount of radiation damage. Therefore, the staff should review again the pressure-temperature operating limits when the test results on the recently removed capsule become available.

TOPIC NO.	TITLE
V-10.A	Residual Heat Removal System Heat Exchanger Tube Failures

SRP Section 5.2.3 requires monitoring and sampling of the primary coolant system.

The Haddam Neck Technical Specifications (TS) do not presently contain any chemistry limits for primary coolant chemistry. The existing TS contain a limit for primary system activity, but none for dissolved chemicals such as chlorides or fluorides. Therefore, the licensee does not conform to SRP 5.2.3 which requires limitations on the concentrations of impurities in the reactor coolant and monitoring on a scheduled basis. The licensee should have a technical specification which requires monitoring for dissolved chemicals.

TOPIC NO.	TITLE
V-10.B	RHR System Reliability
V-11.B	RHR Interlock Requirements (Systems)

10 CFR 50 (GDC 34), as implemented by SRP 5.4.7 and Branch Technical Position RSB 5-1, requires, in part, that a system to remove residual heat be provided with suitable redundancy to assure that for onsite electrical power system operation the system safety function can be accomplished, assuming a single failure.

1. Because of the potential for Residual Heat Removal (RHR) overpressurization, the staff has determined that the following modifications should be considered for backfit during the integrated plant safety assessment:

   a. Interlocks on the RHR-to-core deluge motor-operated valves to prevent opening until RCS pressure is below design pressure.

   b. Modification of the technical specifications to require placing the overpressure protection system in operation whenever RHR cooling is in progress.

2. The staff concludes that the Haddam Neck systems fulfill the safety objectives of reliable plant shutdown capability using safety-grade equipment provided that plant operating procedures are modified to instruct operators how to perform shutdown and cooldown functions with the systems identified in the minimum systems list.

3. The staff noted during the safe shutdown evaluation that no Technical Specification requirement governs the allowed outage time of an ECCS train. The need for this requirement will be evaluated under SEP Topic XVI, "Technical Specifications."

4. Based on our review, the staff concludes that procedural shortcomings exist with respect to shutdown from outside the control room in the areas of maintenance of batteries for portable instruments, the assignment of shutdown duties for shift pesonnel and emergency communication methods. The licensee should modify his procedures to alleviate these shortcomings.

TOPIC NO.	TITLE
V-11.A	Requirements for Isolation of High and Low Pressure Systems
V-11.B	RHR Interlock Requirements (Electrical)

10 CFR 50 (GDC 15) as implemented by SRP Section 7.6 and BTP ICSB 3, requires that interlock systems important to safety be adequately designed to assure their availability in the event of an accident. This includes those systems with direct interface with the reactor coolant system which have design pressure ratings lower than the reactor coolant system design pressure.

Because of the severe consequences of a LOCA outside of containment the staff proposes that the SI system isolation valve control be modified to prevent opening if RCS pressure exceeds SI system design pressure as required by SRP 6.3.

The charging pump discharge valves do not satisfy the applicable criteria and modifications to these valves will be pursued under SEP Topic VI-4.

TOPIC NO.	TITLE
VI-1	Organic Materials and Post-Accident Chemistry

10 CFR 50 (GDC 1, 4, 14, 31, 35, 41, and Appendix B), as implemented by SRP Sections 6.1.1 and 6.1.2 and Regulatory Guide 1.54, requires, in part, that structures, systems and components important to safety be designed to accomodate the effects of and be compatible with the environmental conditions associated with normal operating and postulated accident conditions. In particular, paints and organic materials used inside containment and post-accident water chemistry should not adversely effect ESF functions.

Post Accident Chemistry - Based on the staff evaluations, we conclude that, although the Haddam Neck Plant can be operated with an acceptable degree of safety under normal conditions when containment spray and sump water recirculation are not required, the post accident water chemistry does not meet the acceptance criterion of Standard Review Plan Section 6.1.1 and Branch Technical Position MTEB 6-1 and is, therefore, not acceptable. In order to reduce the potential of stress corrosion cracking of the engineered safety feature equipment inside the containment following a design basis accident, the licensee should either show that the post accident water chemistry meets the acceptable criterion II.B.1 in Standard Review Plan Section 6.1.1 and Branch Technical Position MTEB 6-1, or provide an acceptable alternative.

TOPIC NO.	TITLE
VI-4	Containment Isolation System

10 CFR 50 (GDC 54, 55, 56, and 57), as implemented by SRP 6.2.4 and Regulatory Guides 1.11 and 1.141, establish explicit requirements for isolation valving in lines penetrating the containment. Specifically, they address the number and location of isolation valves (for example, redundant valving with one located inside containment and the other located outside containment), valve actuation provisions (for example, automatic or remote manual isolation valves), valve position (for example, locked closed, or the position of greater safety in the event of an accident or power failure) and valve type (for example, a simple check valve is not a permissible automatic isolation valve outside containment).

At Haddam Neck, the staff determined that the licensee does not comply with current licensing criteria in the following areas:

1. Both containment isolation valves are located outside of containment.

2. Use of simple check valve outside containment as a containment isolation valve.

3. Use of remote manual valves without provisions to inform operator when isolation is required.

4. The use of hand operated manual valves for containment isolation with no indication that these valves are sealed closed or otherwise under administrative control.

5. Containment penetrations with no valving identified for isolation purposes.

6. Containment penetrations with only one valve identified as an isolation valve.

7. Use of blind flanges without indicating if barriers are leak tested.

TOPIC NO.	TITLE
VI-7.B	ESF Switchover from Injection to Recirculation Mode (Automatic ECCS Realignment)

10 CFR 50 (GDC 35) requires that a system to supply abundant emergency core cooling be provided.

At Haddam Neck the staff has determined that the licensee does not comply with current licensing criteria as follows:

1. The switchover from injection to recirculation in Haddam Neck is accomplished manually from the control room. The primary instrument for determining when to make the switchover does not satisfy the single failure criterion. Furthermore, present backup instrumentation (containment sump level) is not independent of the primary instrumentation. Accordingly, the primary instrumentation should be replaced by a Class 1E system satisfying the review guidelines.

2. There are no alarms to alert the operator to start the switchover when sufficient water has been pumped from the RWST.

3. The available time for the operator to detect the need for switchover and to complete the required actions is not consistent with the review guidelines.

4. The consequences of failing to complete the transfer before reaching the minimum RWST level have not been shown to be acceptable. The charging pumps, which take suction on the RWST during the injection mode and are thus susceptible to damage if the switchover is not completed before the tank level drops too low, are used for two-path recirculation.

TOPIC NO.	TITLE
VI-7.C.1	Independence of Redundant Onsite Power Systems

10 CFR 50 (GDC 17) as implemented by Regulatory Guide 1.6 and IEEE Std. 308-1974 requires that onsite electrical power supplies and their onsite distribution systems shall have sufficient independence to perform their safety function assuming a single failure.

The Haddam Neck onsite standby AC and DC power systems do not comply with current licensing criteria. In each case, a manual breaker exists which allows paralleling of the two power divisions; no interlocks or procedures prevent this. Additionally, the DC power system design permits all four inverters to be supplied from a single battery.

TOPIC NO. TITLE

VI-10.A Testing of Reactor Trip System and Engineered Safety Features, Including Response-Time Testing

10 CFR 50 (GDC 21), as implemented by IEEE Stds. 279-1971 and 338-1977, and Regulatory Guide 1.22, requires that the reactor protection system be designed to permit periodic testing of its functioning, including a capability to test channels independently.

It is the staff's position that the design of systems which are required for safety shall include provisions for periodic verification that the minimum performance of instruments and controls is not less than that which was assumed in the safety analysis. Therefore, the licensee should implement a program for response time testing of all reactor protection systems (including engineered safety features systems such as containment isolation). As a part of this program, the response time test requirements should be stated in the Technical Specifications in a manner simliar to that of the Standard Technical Specifications.

The staff proposes that the following corrections be made to existing programs by making suitable changes in the Haddam Neck test procedures and Technical Specifications:

1. The licensee should provide for calibration of the Low Pressurizer Pressure and High Steam Flow Channels.

2. The licensee should provide for functional tests of the following during reactor operations:

   a. Scram logic (both automatic and manual functions)
   b. Low pressurizer pressure
   c. High steam flow
   d. Steam-feedwater flow mismatch
   e. Low steam generator level

3. The licensee should provide for channel checks for low pressurizer and high steam flow channels.

4. The licensee should document the basis for the frequency of calibration, functional test, and channel check for each parameter required to protect the public health and safety.

5. For each parameter that is not tested during reactor operation, the licensee should provide the information specified in Position D.4 of R.G. 1.22.

6. The licensee should clarify the discrepancies in plant documentation that were identified by our contractor.

7. The licensee should design, provide suitable test equipment for, and conduct periodic response time tests of those channels and systems that are required for the protection of public health and safety.

TOPIC NO.	TITLE
VII-1.A	Isolation of Reactor Protection System From Non-Safety Systems, Including Qualifications of Isolation Devices

10 CFR 50 (GDC 24) as implemented by IEEE Std. 279-1971, requires that safety signals be isolated from non-safety signals and that no credible failure at the output of an isolation device shall prevent the associated protection system channel from meeting the minimum performance requirements specified in the design bases.

1. Isolation of RPS monitoring channels from remote meters, the data logger, and/or process recorders does not meet current licensing criteria in the following subsystems:

   a. Pressurizer pressure
   b. High pressurizer level
   c. Steam flow
   d. Feedwater flow
   e. Steam generator level

2. Isolation between the RPS and the following control circuits does not meet current licensing criteria:

   a. The computer which provides setpoints for reactor trip for variable low pressure also provides output signals to the rod control systems without isolation.

   b. The steam-feedwater flow mismatch system provides analog signals to the steam flow controller, the feedwater flow controller and the steam generator level controller without isolation.

TOPIC NO. TITLE

VII-3 Systems Required for Safe Shutdown

GDC 17, requires that offsite power be provided by two independent lines. One of these lines must be available immediately. At Haddam Neck, the two incoming lines (1772 and 1206) can be interconnected via a disconnect (389T399) or a tie breaker (2T3) between bus 1-2 and bus 1-3. These interconnections provide paths that could compromise independence. In addition, because line 1772 may not be synchronied with line 1206, a spurious closing of 2T3 may result in a loss of both lines and cause significant damage to the onsite distribution system.

The staff has not completed its review of how the Haddam Neck Plant meets GDC 17 and the resolution of these concerns will be addressed in the integrated assessment.

TOPIC NO.	TITLE
VIII-1.A	Potential Equipment Failures Associated With a Degraded Grid Voltage

10 CFR 50 (GDC 17), as implemented by IEEE Standards 279-1971 and 308-1977 and staff positions defined in an NRC Generic Letter to Connecticut Yankee Atomic Power Company, dated August 8, 1979, requires, in part, that an offsite electric power system be provided to permit funtioning of systems important to safety. This topic looks at the effects of a sustained degradation of the offsite power source voltage that could result in the loss of capability of redundant safety loads, their control circuitry and the associated electrical components required to perform safety functions.

The staff has reviewed and found with the exception of operating procedures, Haddam Neck's design is acceptable.

TOPIC NO.	TITLE
VIII-2	Onsite Emergency Power System - Diesel Generator

The review criteria are presented for Section 8.3.1 in Table 8-1 of the SRP.

The Haddam Neck design generator protective interlocks do not meet current licensing criteria.

TOPIC NO. TITLE

VIII-3.A Station Battery Test Requirements

IEEE Standard 450-1975, IEEE Standard 308-1974, BTP EICSB 6 and the "Standard Technical Specifications for Westinghouse Pressurized Water Reactors" (NUREG-0452). The required tests are as follows:

1. At least once per 18 months, during shutdown, a battery service test should be performed to verify that the battery capacity is adequate to supply and maintain in operable status all of the actual emergency loads for 2 hours.

2. At least once per 60 months, during shutdown, a battery discharge test should be performed to verify that the battery capacity is at least 80% of the manufacturer's rating.

The technical specifications for the Haddam Neck Plant do not include any requirements for station battery tests. Therefore, the Haddam Neck Plant does not comply with current licensing requirements for station battery tests.

TOPIC NO.	TITLE
VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation

10 CFR 50.55a(h), as implemented by SRP Section 8.3.2 and Regulatory Guide 1.47, requires that the dc power system be monitored to the extent that it is shown ready to perform its intended function. This monitoring is considered necessary in order to assure the design adequacy of the dc power system battery and bus voltage monitoring and annunciation schemes such that the operator can (1) prevent the loss of an emergency dc bus; or (2) take timely corrective action in the event of loss of an emergency dc bus.

The Haddam Neck Plant control room does not meet current licensing criteria. Specifically, the staff proposes that as a minimum, the following additional indications and alarms of the Class 1E dc power system(s) status shall be provided in the control room.

Battery current (ammeter-charge/discharge)
Battery charger output current (ammeter)
DC bus ground alarm (for ungrounded system)
Battery breaker(s) or fuse(s) open alarm
Battery charger output breaker(s) or fuse(s) open alarm
DC bus voltage (voltmeter)

TOPIC NO.	TITLE
IX-3	Station Service and Cooling Water Systems

10 CFR 50 (GDC 44, 45 and 46), as implemented by SRP Sections 9.2.1 and 9.2.2, requires that a cooling water system be provided, inspected and tested, and that the system be capable of transferring heat from structures, systems and components important to safety to the ultimate heat sink.

The staff has determined that the design of the service and cooling water systems is adequate, except for the following:

Component Cooling Water System - The need for system modification to eliminate potential passive single failures will be evaluated during the integrated assessment.

Service Water System - The licensee should verify that those motor-operated valves relied on for system isolation in the event of a loss of offsite AC power receive emergency power, have a fail closed design, or that sufficient time is available for operator action to close the valves.

The licensee should demonstrate by test or analysis that adequate procedures exist to balance system flow requirements and maintain system components below design thermal limits for a single active failure.

The licensee should demonstrate the ability to provide power to a second SWS pump with one pump out of service. (Assuming that the active failure was a diesel generator.)

The licensee should demonstrate that single passive failures (pipe break in containment fan cooler supply header) would not compromise containment integrity or core cooling in the event of a LOCA.

TOPIC NO.	TITLE
IX-5	Ventilation Systems

10 CFR 50 (GDC 5, 19, 60, and 61), as implementd by SRP Sections 9.4.1, 9.4.2, 9.4.3, 9.4.4, and 9.4.5, requires that ventilation systems be provided and have the capability to provide a safe environment for plant personnel and for the operation of engineered safety features.

The ventilation systems for the Haddam Neck Plant were found to be in conformance with criteria for this topic except for the following:

1. The spent fuel pool area ventilation system is neither single failure proof nor powered from emergency sources. To resolve this issue the licensee should either demonstrate that the results of a fuel handling accident without credit for area ventilation, are acceptable or propose corrective system modifications.

2. The primary auxiliary building ventilation system supply portion is not single failure proof. The licensee should evaluate the effects of degraded PAB ventilation on both equipment and personnel. If necessary corrective modifications should be provided.

3. The cable vault ventilation system is subject to several disabling failures. The licensee should either demonstrate that the operation of vital equipment located within this area would not be affected by loss of area ventilation or propose corrective system modifications.

4. The ventilation system associated with each of the emergency diesel generator rooms are subject to disabling single failures. The licensee should either demonstrate that the loss of ventilation will not significantly affect diesel generator availability or propose corrective modifications.

5. The switchgear room ventilation system is susceptible to disabling single failures. The licensee should either demonstrate that vital equipment located within this area would be unaffected by loss of area ventilation or propose corrective system modifications.

6. Supporting information to enable the staff to perform an independent assessment of the cable spreading areas, was not provided. The adequacy of room openings to maintain suitable service conditions should be evaluated.

TOPIC NO.	TITLE
XV-2	Spectrum of Steam System Piping Failures Inside and Outside Containment (Radiological Consequences)

10 CFR 100, as implemented by SRP Section 15.1.5, requires, in part, that the radiological consequences of a steam line break outside containment not exceed specific guidelines for the reactor site.

The staff has determined that Haddam Neck meets the acceptance criteria for this topic. However, this conclusion is based upon a staff analysis in which certain assumptions regarding the design of Hadadm Neck were made. Thus, we recommend that CYAPCo confirm these assumptions to support the validity of the staff evaluation.

TOPIC NO. TITLE

XV-7 Loss of Forced Coolant Flow, Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

10 CFR 50 (GDC 10, 15 and 26), as implemented by SRP Sections 15.3.1 and 15.3.2, requires, in part, that the reactor, reactor coolant system and reactivity control system be capable of operating to keep the plant within design margins even in the event of anticipated operational occurrences.

Based on the information provided, we cannot conclude that the Haddam Neck Plant meets the requirements of GDC 27, 28 and 10 CFR 100 if analyzed in accordance with SRP Sections 15.3.3 and 15.3.4.

TOPIC NO.	TITLE
XV-12	Radiological Consequences of a Rod Ejection Accident

10 CFR Part 100.11 provides dose guidelines for reactor siting against which calculated accident dose consequences may be compared.

The estimated low population zone thyroid doses are acceptable to licensing criteria. The estimated 2 hour EAB dose exceeds the criteria by 33% or 24 rem. However, because the percentage (10%) of failed fuel clad is conservative and because the dose model yields conservative estimates, it is the staff's judgement that an analysis using a DNB criteria would result in significantly lower estimations of failed fuel which would lead to lower doses. The need to perform a rod ejection accident evaluation to determine the number of fuel assemblies experiencing DNB will be determined during the integrated assessment.

TOPIC NO.	TITLE
XV-16	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

10 CFR Part 100, as implemented by Standard Review Plan 15.6.2, requires that the radiological consequences of failure of small lines carrying primary coolant outside containment be limited to small fractions of the exposure guidelines of 10 CFR Part 100.

Based on the staff's evaluation, the radiological consequences of small line failures outside containment are a small fraction of the 10 CFR 100 guidelines, provided that the Standard Technical Specifications for coolant activity are implemented in order to limit reactor water iodine concentrations.

TOPIC NO. TITLE

XV-17 Steam Generator Tube Rupture

Section 50.34 of 10 CFR Part 50 requires, in part, that each applicant for a 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.

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

Radiological Consequences - The staff's calculated radiological consequences at the exclusion area boundary exceed the guideline values of 10 CFR Part 100. The calculated radiological consequences at the low population zone boundary are less than the guideline values of 10 CFR Part 100. However, SRP 15.6.3 criteria was exceeded.

Systems - In order for the staff to determine the ability of the plant to mitigate the consequences of a SGTR, we request that the licensee either provide the justification 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.

TOPIC NO.	TITLE
XV-19	Loss of Coolant Accidents Resulting From a Spectrum of Piping Breaks Within the Reactor Coolant Pressure Boundary

10 CFR 100, as implemented by SRP Section 15.6.5 Appendices A and B, TID-14844 and Regulatory Guide 1.4, requires, in part, that exposure guidelines not be exceeded for design basis LOCA resulting in containment leakage or in leakage outside containment from the engineered safety features.

Based on the review of the licensee's analysis and our independent evaluations, we conclude that the offsite doses from a postulated design basis loss-of-coolant accident at Haddam Neck are within the guidelines of 10 CFR 100.11.

However, for the reasons set forth in the evaluation, the operation of the containment spray system to assure the effectiveness of the internal filter system will be considered in the integrated assessment of this plant.