Bepto

September 10, 1982

Docket No. 50-29
LS05-82-09-031

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SUMMARY OF SEP TOPIC DIFFERENCES - YANKEE NUCLEAR POWER STATION

Enclosure 1 is a listing of all of the SEP topics for which Yankee did not meet the current licensing acceptance criteria. Enclosure 2 is a summary description of each topic difference, except for Topics II-4.E, "Dam Integrity," and III-6, "Seismic Design Considerations." The summary descriptions for these two 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 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.

SE04

As previously discussed, we will meet with you on September 13 and 14, 1982, in Bethesda, to discuss your proposed actions on the enclosed topic differences. Following that meeting, your are requested to formally submit your proposed actions to resolve the differences.

DSU USE (11)

Sincerely,

Ralph Caruso, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

8209230224 820910
PDR ADOCK 05000029
P PDR

Enclosures:
As stated

cc w/enclosures:
See next page

OFFICE	SEPB:DL	SEPB:DL	SEPB:DL	ORB#5:PM	ORB#5:BC	AD:SA:DL
SURNAME	MBoyle:dk	CGrimes	WRussell	RCaruso	DCrutchfield	TIppolito
DATE	9/3/82	9/3/82	9/3/82	9/7/82	9/9/82	9/ /82

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

cc
Mr. James E. Tribble, President
Yankee Atomic Electric Company
25 Research Drive
Westborough, Massachusetts 01581

Chairman
Board of Selectmen
Town of Rowe
Rowe, Massachusetts 01367

Energy Facilities Siting Council
14th Floor
One Ashburton Place
Boston, Massachusetts 02108

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

Resident Inspector
Yankee Rowe Nuclear Power Station
c/o U.S. NRC
Post Office Box 28
Monroe Bridge, Massachusetts 01350

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

Enclosure 1

LISTING OF SEP TOPIC EVALUATIONS WITH DIFFERENCES FOR YANKEE NUCLEAR POWER STATION

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)
II-4.E	Dam Integrity
II-4.F	Settlement of Foundations and Buried Equipment
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.C	Inservice Inspection of Water-Control Structures
III-4.A	Tornado Missiles
III-5.A	Effects of Pipe Break on Structures, Systems and Components Inside Containment
III-5.B	Pipe Break Outside Containment
III-6	Seismic Design Considerations
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
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 (E,S)	RHR Interlock Requirements
VI-1	Organic Materials and Post-Accident Chemistry
VI-4 (S,*)	Containment Isolation System
VI-7.A.3	ECCS Actuation System
VI-10.A	Testing c 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-3.B	DC Power System Bus Voltage Monitoring and Annunciation
VIII-4	Electrical Penetrations of Reactor Containment
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-4	Loss of Non-Emergency AC Power to the Station Auxiliaries
XV-7	Reactor Coolant Pump Rotor Seizure and Shaft Break

Topic No.	Title
XV-16	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment
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

YANKEE NUCLEAR POWER STATION

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)

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. Groundwater - The licensee has submitted fragmented information on site groundwater. The information is insufficient to support estimation of maximum groundwater levels for use in structural evaluations. Therefore, the staff concludes that current NRC criteria would require that groundwater must be assumed at ground surface elevation for structural evaluations. (Topic II-3.B.)

2. Deerfield River Flooding - Harriman Dam will be overtopped and thus assumed to fail if average basin rainfall exceeds about 13 inches in 24 hours. Our estimate of the Probable Maximum Precipitation (PMP), the current NRC design basis rainfall, is 18.9 inches in 24 hours. The dam could also fail for other reasons, such as: seismic, piping or foundation failure. The failure of Harriman Dam for any reason would result in a flood water elevation at the Yankee site of at least 1174 feet msl or more than 40 feet over plant grade. In addition, a PMP centered over the drainage area between Sherman and Harriman Dams would overtop (elevation 1133.3 ft msl) and presumably breach Sherman Dam. If Sherman Dam does not breach, these flood levels could also inundate the Yankee site. We conclude that current provisions for flooding at the Yankee site do not meet current licensing criteria. (Topic II-3.B.)

3. Local Site Flooding - A PMP centered over tributary number 1, south of the Yankee Plant, could result in runoff across the plant area that would be up to three feet deep. This does not meet current NRC licensing criteria. (Topic II-3.B.)

4. Roof Flooding - Only the turbine building roof is suceptible to rainfall accumulations that could exceed the design capacity of 40 psf (7.7 inches of ponded water). With roof drains assumed blocked, the PMP could pond to the top of the parapets which would exceed the design basis by 173%. (Topic II-3.B.)

5. Emergency Procedures and Technical Specifications - Emergency plans and technical specifications may be required for the ultimate heat sink and for Deerfield River flooding, dependent on the resolution of these issues during the integrated assessment. The feasibility of emergency plans and technical specifications will be determined at that time. (Topic II-3.B.1.)

6. Ultimate Heat Sink (UHS) - The normal UHS water supply (Sherman Pond) does not meet current NRC criteria. The acceptability of other sources is reviewed under other topics. (Topic II-3.C.)

TOPIC NO. TITLE

II-4.F Settlement of Foundations and Buried Equipment

10 CFR 50 (GDC 2) and 10 CFR 100 (Appendix A), as implemented by SRP Section 2.5.4 and Regulatory Guides 1.127, 1.132 and 1.138, require in part, that structures, systems and components important to safety be designed to withstand normal effects without loss of safety function and state the requirements of the geologic data necessary to establish site suitability.

Based on the staff's evaluation, the following items were found not to be in conformance with current criteria:

1. The licensee should investigate the liquefaction potential of submerged backfill material between the underlying lodgement till and the ground surface, and its potential effect on safety related structures.
2. The licensee should further investigate the reasons for the observed cracks in the walls of the Spent Fuel Pool Building in order to assure that the cracks are not caused by differential settlement of foundations and that these cracks do not pose any safety hazard.

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 Yankee 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 June 18, 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. Turbine
2. Diesel generator building
3. Primary auxiliary building
4. Control room
5. Siding and decking

For the following structures or components, the licensee should either demonstrate acceptability for tornado loads or that the consequences of failure if subjected to tornado loads are acceptable:

1. Chimney
2. Safety-related components not inside structures

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

Factors of safety used in the original design should be provided in order to determine whether the bearing stress increase for wind design is acceptable or not.

Where significantly lower capacities have been provided by the licensee than have been calculated by the staff for tornado dynamic pressure, the licensee should provide the bases for those capacities to clarify the discrepancies.

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 40 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.

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

1. Harriman Dam is an essential flood control structure and should be included in the inspection program.

2. The future inspection program should incorporate the items identified in the SER.

3. The inspection program should be constructed using the approach given in Regulatory Guide 1.127.

4. YAEC should perform additional inspections immediately after extreme events.

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 Yankee does not meet the current licensing criteria for tornado missile protection in the following areas:

1. The main steam and main feedwater systems exterior to the vapor container.
2. Atmospheric dump valves, steam generator vents and hoggers.
3. Auxiliary feedwater system.
4. Demineralized water storage tank and primary water storage tank.
5. Service water system.
6. Chemical and volume control.
7. Emergency power system.
8. Shutdown cooling system.
9. Component cooling system.
10. Pressure control and relief system blowdown.
11. Compressed air system.
12. Instrumentation for the safe shutdown equipment.
13. Reactivity control system.
14. Control room.
15. Radwaste treatment system.

TOPIC NO.	TITLE
III-5.A	Effects of Pipe Break on Structures, Components and Systems Inside Containment

10 CFR 50 (GDC 4), as implemented by Regulatory Guide 1.46 and SRP Section 3.6.2, requires, in part, that structures, components and systems important to safety be appropriately protected against dynamic effects, such as pipe whip and discharging fluids that may result from equipment failures. The effects of pipe breaks inside containment was not a part of the original design basis of Yankee.

The staff has determined that Yankee is adeqately protected against the dynamic effects of pipe break inside containment except for the following areas which require further evaluation.

1. Clarification of assumptions used in the jet impingement and pipe whip evaluations.
2. Evaluation of thrust forces on steam generator due to main steam or feedwater line breaks.
3. Evaluation of effects of jet impingement on blister 12E.
4. Evaluation of pipe whip interactions on loop compartment walls from postulated breaks in large RCS piping.

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 Yankee is adequately protected against the dynamic effects of pipe break outside containment except for the following two areas which require further evaluation.

1. Effects of main steam line break on adjacent hydraulic non-return valve operator.
2. Jet impingement on the switchgear room block wall from a break in the No. 2 feedwater heater extraction steam line on the mezzanine level.

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 Strutures at Yankee to determine if any of the structural elements for which a concern exists are a part of the facility design of Yankee. For those that are, assess the impact of the code changes on margins of safety on a plant specific basis, and

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 insure 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.

Yankee 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 Yankee 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 v

TOPIC NO.	TITLE
IV-2	Reactivity Control Systems Including Functional Design and Protection Against Single Failures

10 CFR 50 (GDC 25), as implemented by SRP Section 15.4.3, requires 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 YAEC, the staff has determined that the following may occur as a result of single failures:

1. A single rod may drop into the core.
2. A single rod may not move when movement is commanded.
3. A single rod may be inadvertently moved or malpositioned.
4. An entire group may drop into the core.
5. An entire group may not move when movement is commanded. This includes both automatic in (based on T_{ave}) and manual commands.
6. An entire group may be inadvertently moved or malpositioned. This includes the simultaneous movement of two groups when only one group is commanded and the inward movement of a group when the outward movement is selected.
7. All rods may drop into the core.
8. All rods may not move in when movement commanded. This includes any number of groups failing to move in when all rods are commanded to move.
9. All rods may be inadvertently moved in or malpositioned.
10. An entire group of rods may be withdrawn beyond the 485 MWt limit.

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

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 Yankee, 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, only two of the three recommended detection systems are present. Also, the presently installed systems are not seismically qualified, and, while the systems may be able to detect a one gallon per minute leak, the time required to detect most leaks is greater than one hour.

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 acceptabiity of reactor vessel materials thoughout the expected plant service life:

1. Samples from several welds made by the same technique and materials as the vessel beltline welds should be made and a chemical analysis performed on them.

2. Since many of the vessel welds cannot be examined in accordance with ASME Code Section XI rules, it is recommended that the use of acoustic emission techniques be considered as a means of verifying the integrity of the welds.

3. Following completion of USI A-11, "Reactor Vessel Materials Toughness," YAEC should submit a report to the NRC covering the items required by Appendix G to 10 CFR Part 50 for vessels containing ferritic pressure-retaining materials with Charpy Upper Shelf Energies of less than 50 ft-lbs.

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

SRP Section 9.2.1 requires that the service water system include the capability for detection and control of radioactive leakage into and out of the system and prevention of accidental releases to the environment.

Yankee's Shutdown Cooling System (SCS) is normally at a higher pressure than the component cooling water system (CCW). Therefore, a tube leakage in the SCS heat exchanger would result in contamination of the CCW system. Furthermore, because CCW heat exchangers are cooled by the service water system, there exists a possible pathway for contaminated primary coolant to leak to the ultimate heat sink and the environment.

The staff determined that there is no service water system monitor or alarm to alert plant operators to leakage of radioactive materials to the environment. In addition, there are no technical specification requirements for the operability and surveillance of the CCW system monitor.

TOPIC NO.	TITLE
V-10.B	RHR System Reliabiity
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 electric power system operation the system safety function can be accomplished, assuming a single failure. Redundancy to the Shutdown Cooling System (SCS) is provided by the low pressure surge tank (LPST) system. The staff has determined that the degree of redundancy provided by the SCS and LPST is acceptable; however, the following deviations exist which could impair the reliability of the system.

1. The SCS suction and discharge motor-operated isolation valves do not have position indication in the control room. The valves are operated from the primary auxiliary building (PAB) and cannot be operated from the control room.

2. There are no provisions to prevent damage to the SCS pump or LPST system cooling pump due to overheating, cavitation, or loss of adequate suction fluid.

3. In order to cool the reactor coolant system to the SCS initiation point and to initiate SCS operation, significant operator action must be performed from outside the control room.

10 CFR 50 (GDC 34) requires, in part, that a system to remove residual heat be provided with suitable isolation capabilities to assure the safety system function can be accomplished, assuming a single failure. The Yankee SCS suction and discharge (isolation) valves do not have any permissive interlocks or automatic closure features, and valve position indication is not provided in the control room. Also, the SCS isolation valves do not have automatic closure interlocks to close the valves during slow increases in reactor coolant system pressure.

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.

Yankee has two systems with a lower design pressure rating than the RCS that are directly connected to the RCS. These are the Reactor Heat Removal (RHR) and the Chemical Volume Control (CVCS) Systems.

The RHR system and CVCS are not in compliance with current licensing requirements for isolation of high and low pressure systems because the RHR system isolation valves do not have any interlocks to prevent opening when RCS pressure exceeds RHR system design pressure as required by BTP RSB 5-1.

The RHR system isolation valve control circuitry should be modified to prevent opening when RCS pressure exceeds RHR system design pressure as required by BTP RSB 5-1.

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 condition 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.

Yankee has been determined not to meet the following:

1. Post-Accident Water Chemistry

    a. There is no provision to control oxygen or chloride content in the sump water.

    b. During recirculation, sump water pH will be below 7.0.

2. Organic Materials

    a. Details of the routine inspection of painted surfaces should be provided. Acceptable procedures are presented in ANSI 101.2-1973.

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 Yankee, the staff has determined that the licensee does not comply with current licensing criteria in the following areas:

1. A lack of redundancy in the isolation provisions by only using a single isolation barrier in an open system;

2. The use of simple check valves outside containment as an automatic isolation valve;

3. No isolation provisions on safety valve discharge lines and instrument lines;

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. The use of local manual valves or remote manual valves for non-essential systems;

6. Certain engineered safety feature systems do not have remote manual isolation capability;

7. Insufficient information regarding the design and environmental conditions for the Low Pressure Surge Tank to ascertain its qualification as an extension of the containment boundary; and

8. Insufficient information regarding the design and isolation provisions for containment leg expansion joints to be qualified as containment penetrations.

TOPIC NO.	TITLE
VI-7.A.3	ECCS Actuation System

10 CFR 50 (GDC 37), as implemented by SRP 7.1 Appendix B, Branch Technical Position ICSB-25, and Regulatory Guide 1.22, requires that the ECCS be designed to permit periodic pressure and functional testing to assure operability and performance.

The Yankee Technical Specifications provides for the exclusion of testing automatic valves in the flowpath of the ECCS. The staff has determined that this is not acceptable and that the phrase "Excluding Automatic" should be deleted from the Technical Specifications.

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.

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.

The present design should be upgraded by substituting qualified isolators in the Nuclear Instrumentation, Main Coolant Flow and High Pressurizer Water Level Circuits where any of the following conditions are met:

1) Redundant channels could be connected to a defective circuit by operation of a switch, or

2) The wiring to the non-safety recorders and indicators does not satisfy the separation criteria of Regulatory Guide 1.75.

In addition, the design of the 36 volt power supplies and their source(s) of power may not be adequate to assure isolation between redundant coolant pressure channels. Also, since multiple sensors feed single logic amplifiers it is necessary to assure that a scram is generated upon the loss of any logic power source.

TOPIC NO.	TITLE
VII-3	Systems Required for Safe Shutdown

10 CFR 50 (GDC 2, 4, 13, 17, 19, 26, 34, 35, 44), as implemented in SRP Chapter 7, Branch Technical Position ICSB-18, IEEE Standard 279-1971 and Regulatory Guide 1.53, requires systems capable of safely shutting down the reactor in the presence of certain conditions.

Yankee satisfies all of the requirements for Safe Shutdown except for a lack of adequate electrical supply. An additional ac onsite source is required for the Shutdown Cooling System or Low Pressure Suction Tank valves and an additional set of instruments (from an independent Class 1E power source) is required for the component cooling water surge tank level.

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 Yankee Atomic Electric Company, dated June 3, 1977, requires, in part, that an offsite electric power system be provided to permit functioning 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 acceptable Yankee's proposal to use specific operator action under degraded grid conditions without an accident acceptable subject to completion of all proposed modifications and the institution of adequate procedures.

The licensee must commit to a submittal and review schedule for these procedures.

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 Yankee 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

TOPIC NO. TITLE

VIII-4 Electrical Penetrations of Reactor Containment

10 CFR 50 (GDC 50), as implemented by IEEE Standard 317 and Regulatory Guide 1.63, requires, in part, that reactor containment structure, including pentrations, be designed so that the containment structure can, without exceeding design leakage rate, accomodate the calculated pressure, temperature and other environmental conditions resulting from any loss of coolant accident.

As a result of our review we have concluded that adequate protection for the following electrical penetrations does not exist. The staff recommends that the following be implemented to resolve this topic:

1. The medium voltage penetration pairs for the reactor coolant pumps should be monitored and their feeders tripped automatically whenever either penetration fails to carry its normal share of the load.

2. Class 1E qualified low voltage circuits inside of containment should be identified and provided with a Class 1E isolation device.

3. All non-Class 1E low voltage circuits that were identified in your August 2, 1982 message should be a) qualified to Class 1E standards and provide with a Class 1E isolation devices, or b) each circuit should be provided with redundant Class 1E isolation devices, c) the circuits should be de-energized during reactor operation, or d) the existing breakers should be modified to trip on an accident signal.

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:

1. Component Cooling System - The licensee should verify that adequate procedures exist to ensure that emergency power is provided to this system in the event of an accident.

2. Service Water System - The licensee should verify the existence of procedures which would ensure that system flow requirements are balanced.

TOPIC NO. TITLE

IX-5 Ventilation Systems

10 CFR 50 (GDC 5, 19, 60, and 61), as implemented 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 Yankee ventilation systems meet the current acceptance criteria, except for the following:

1. Auxiliary and Radwaste Ventilation System - Failure of exhaust fan RF-11, which ventilates the radioactively clean portion of the primary auxiliary building, may allow area temperatures to rise enough, particularly during the summer, to adversely affect operation of safety-related equipment located in that vicinity. Therefore, this ventilation system does not satisfy single failure criterion.

   Louvers in the upper level of the primary auxiliary building which must open to vent the building in case of a steam line break were not described adequately to enable an assessment of functional redundancy. If the ventilation system is vulnerable to a single active failure of the louvers or their operating mechanism, a steam line break might cause the formation of a harsh environment around safety-related equipment located in the lower level of the building. The system would thus not satisfy the single-active-failure criterion. The licensee should submit clarifying information about the louvers to resolve this question and propose corrective action if required.

2. Diesel Generator Building Ventilation System - The licensee's evaluation did not address the effect of the most limiting single active failure of the system (four motor-operated dampers, ventilating unit UV-1, and roof exhaust fans PRV 1 and PRV 2, actuated by heat-sensitive switches) which ventilates the area containing the safety injection pumps, the No. 3 battery and charger, and switchgear. The system appears to lack the necessary redundancy. The licensee should review the system that ventilates this area to ensure that a single active failure cannot result in an unacceptable temperature rise.

3. Battery Rooms Ventilation System - This ventilation system services battery rooms 1 and 2 and should be evaluated for compliance with relevant acceptance criteria. The licensee has not yet submitted an evaluation of this system.

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

TOPIC NO.	TITLE
XV-4	Loss of Non-Emergency AC Power to the Station Auxiliaries

10 CFR 50 (GDC 10, 15 and 26), as implemented by SRP 15.2.6, 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.

YAEC has stated that the immediate reactor trip that takes place in the event of loss of non-emergency ac power causes this transient to be less severe than for a loss-of-load without a direct reactor scram event during which the reactor may not be tripped for 20 seconds after the loss of load. The additional energy, which is generated by the reactor during this 20 seconds, makes the transient, which follows a loss-of-load without a direct reactor scram, more severe than that for the loss of non-emergency ac power. However, this conclusion relies on having flow from two reactor coolant pumps for 30 to 60 seconds after the loss of ac power. The power for this flow would be obtained from the inertia of the generator and turbine during the coastdown. We have not obtained from the licensee any information regarding his calculated values of the RCS pump flows during the coastdown of the generator.

The justification for the stated generator inertial characteristics is an open item in Topic XV-7.

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.

In their safety evaluation, the staff has indicated that the results of the YAEC's analysis do not meet the acceptance criteria of SRP 15.3.1. We recommend that the licensee provide justification that the simultaneous coastdown of all four pumps is not an event of moderate frequency. This justification should include information on the inertial characteristics of the turbine to support the 30-60 second delay discussed above.

Further, as discussed in part (b) of this topic, the licensee should demonstrate, using an acceptable fuel damage model, that the radiological consequences of any loss of flow event are acceptable.

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 instrument line break is the limiting case for offsite doses. The calculated offsite doses exceed 10% of 10 CFR Part 100 guidelines and, therefore, do not meet the criteria of SRP 15.6.2.

The licensee is requested to provide analyses or data which could lead to a more detailed assessment of potential small line breaks in order to support a conclusion that doses would be unlikely to exceed SRP 15.6.2 guidelines.

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, rquires, 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.

The staff has determined that the calculated doses resulting from a loss-of-coolant accident exceed the dose guidelines of 10 CFR 100.11. A major contributor to the calculated dose is from the postulated leakage of recirculated core cooling water outside containment. It is reasonable to assume that the leakage could be reduced by appropriate surveillance and maintenance, and limited by Technical Specifications to lower values. Also, the postulated release of airborne iodine from this leakage could be reduced by orders of magnitude by filtering this release pathway.