II. Markup of Proposed Changes

See attached markup of proposed changes to Technical Specifications.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting* of 2485 psig ± ~~1~~ 3%.**

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

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*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Within ± 1% following pressurizer Code safety valve testing

REACTOR COOLANT SYSTEM

SAFETY VALVES

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting* of 2485 psig ± ~~1~~ 3%.**

APPLICABILITY: MODES 1, 2, and 3#.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Within ± 1% following pressurizer Code safety valve testing.

#Entry into this MODE is permitted for up to 24 hours to perform post-modification or post-maintenance testing to verify OPERABILITY of components. ACTION requirements shall not apply until OPERABILITY has been verified.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR-LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	87
2	65
3	43

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER					
Loop 1	Loop 2	Loop 3	Loop 4	LIFT SETTING* (± ~~1~~ 3%)**	ORIFICE SIZE
V6	V22	V36	V50	1185 psig	16.0 sq. in.
V7	V23	V37	V51	1203 psig	16.0 sq. in.
V8	V24	V38	V52	1220 psig	16.0 sq. in.
V9	V25	V39	V53	1238 psig	16.0 sq. in.
V10	V26	V40	V54	1255 psig	16.0 sq. in.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Within ±1% following main steam line Code safety valve testing

(1974 Edition, including the Summer 1975 Addenda)

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1320 psia) of its design pressure of 1200 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, ~~1971 Edition~~. The total relieving capacity for all valves on all of the steam lines is 1.839×10^7 lbs/hr which is 121% of the total secondary steam flow of 1.514×10^7 lbs/hr at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table ~~3.7-2~~. 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For four loop operation:

$$SP = \frac{(X) - (Y)(V)}{X} \times 109$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hr. and

Y = Maximum relieving capacity of any one safety valve in lbs/hr

III. Retype of Proposed Changes

See attached retype of proposed changes to Technical Specifications. The attached retype reflects the currently issued version of Technical Specifications. Pending Technical Specification Changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with Technical Specifications prior to issuance.

Revision bars are provided in the right hand margin to indicate a revision to the text. No revision bars are utilized when the page is changed solely to accommodate the shifting of text due to additions or deletions.

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting* of 2485 psig ± 3%.**

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**Within ±1% following pressurizer Code safety valve testing.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting* of 2485 psig ± 3%.**

APPLICABILITY: MODES 1, 2, and 3#.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**Within ±1% following pressurizer Code safety valve testing.

#Entry into this MODE is permitted for up to 24 hours to perform post-modification or post-maintenance testing to verify OPERABILITY of components. ACTION requirements shall not apply until OPERABILITY has been verified.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR-LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
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TABLE 3.7-2

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VALVE NUMBER

Loop 1	Loop 2	Loop 3	Loop 4	LIFT SETTING* (± 3%)**	ORIFICE SIZE
V6	V22	V36	V50	1185 psig	16.0 sq. in.
V7	V23	V37	V51	1203 psig	16.0 sq. in.
V8	V24	V38	V52	1220 psig	16.0 sq. in.
V9	V25	V39	V53	1238 psig	16.0 sq. in.
V10	V26	V40	V54	1255 psig	16.0 sq. in.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**Within ±1% following main steam line Code safety valve testing.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1320 psia) of its design pressure of 1200 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1974 Edition, including the Summer 1985 Addenda. The total relieving capacity for all valves on all of the steam lines is 1.839×10^7 lbs/hr which is 121% of the total secondary steam flow of 1.514×10^7 lbs/hr at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For four loop operation:

$$SP = \frac{(X) - (Y)(V)}{X} \times 109$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hr, and

Y = Maximum relieving capacity of any one safety valve in lbs/hr

IV. **Safety Assessment of Proposed Changes**

New Hampshire Yankee is proposing to revise the Seabrook Station Technical Specifications to allow a relaxation in the Pressurizer Safety Valve (PSV) and Main Steam Safety Valve (MSSV) setpoint tolerances to ± 3% for ASME Section XI testing acceptance criteria. The proposed Technical Specification changes also require that the PSV and MSSV setpoints be restored to within ± 1% of their nominal setpoints following testing. New Hampshire Yankee is also proposing to revise the BASES for Technical Specification 3/4.7.1.1 to specify the correct Edition of the ASME Boiler and Pressure Vessel Code Section III applicable to the MSSVs. Additionally, NHY is proposing to correct a typographical error in the BASES for Technical Specification 3/4.7.1.1. The BASES currently contain an incorrect reference to Technical Specification Table 3.7-2, whereas the correct reference is to Table 3.7-1.

The Seabrook Station overpressure protection design incorporates three Code safety valves on the primary system pressurizer and a total of twenty Code safety valves on the four main steam lines (five per line) in the secondary system. The pressurizer safety valves (PSVs) were designed and manufactured to meet the 1971 Edition including the Winter 1972 Addenda of the ASME Code, Section III. The main steam safety valves (MSSVs) were designed and manufactured to meet the 1974 Edition including the Summer 1975 Addenda of the ASME Code, Section III. An ASME Code, Section III requirement for both the PSVs and the MSSVs is that they be designed to open within ± 1% of their set pressure. The current Technical Specification Limiting Conditions for Operation (LCO) for the PSVs and MSSVs also impose the tolerance of ± 1% on their set pressure.

The Technical Specification Surveillance Requirements for the PSVs and the MSSVs require that testing be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50.55a(g), except where specific written relief has been granted by the Commission. The PSVs and the MSSVs are tested to verify that their lift pressures and seat leakages are acceptable pursuant to the New Hampshire Yankee Inservice Test (IST) Program which complies with the ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition through the Summer 1983 Addenda. The NRC evaluation of the NHY IST Program is documented in NUREG-0896, Supplement No. 6, "Safety Evaluation Report Related to the Operation of Seabrook Station, Units 1 and 2", dated October 1986. The 1983 Edition of ASME Section XI does not specify a tolerance to be applied to safety valve lift pressure verification; therefore the tolerance (± 1%) prescribed in the LCO for the PSVs and MSSVs is utilized as the acceptance criteria for ASME Section XI testing. ASME Section XI Article IWV-3513 requires additional safety valve testing when a safety valve "fails to function properly". Currently, a PSV or MSSV which has a tested lift pressure outside the ± 1% tolerance specified in the LCO is determined to have failed to function properly, thereby requiring repair or replacement per IWV-3514 and testing of additional valves in the system per IWV-3513.

The 1989 Edition of the ASME Code, Section XI, requires that the PSVs and MSSVs be tested pursuant to the ASME/ANSI OM-1987, Part 1, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices." This standard allows the tested lift pressure to exceed the stamped set pressure by up to 3% before declaring a test failure. It also provides a guideline for testing additional valves when a valve exceeds the ± 3% tolerance. Therefore, increasing the PSV and MSSV setpoint tolerance to ± 3% for testing acceptance criteria is in compliance with the later ASME Code, Section XI requirements.

The proposed relaxation of the setpoint tolerances for the PSVs and the MSSVs has been determined to be in compliance with the 1989 Edition of the ASME Code, Section III, Subarticle NB-7410/NC-7410, which states that "The set pressure of at least one of the pressure relief devices connected to the system not be greater than the Design Pressure of any component within the pressure retaining boundary of the protected system". The Reactor Coolant System design pressure is 2485 psig (reference UFSAR Table 5.3-1) which corresponds to the setpoint of the PSVs. The Main Steam Supply System design pressure is 1185 psig (reference UFSAR Section 10.3.2.1) which corresponds to the Group 1 MSSVs which have the lowest opening setpoint.

It is important to note that NHY proposes to utilize the ± 3% tolerance for the "as-found" acceptance criteria for additional valve testing required by ASME Section XI, Subsection IWV-3513. The proposed Technical Specification revisions require that the PSV and MSSV setpoints be restored to within ± 1% of their nominal setpoints following testing.

The impact of the relaxed PSV and MSSV setpoint tolerance on the licensing basis analysis documented in the Seabrook Station Updated Final Safety Analysis Report (UFSAR), Chapter 15, has been reviewed by Yankee Atomic Electric Company (YAEC) and documented in topical report YAEC-1847 "Seabrook Station Code Safety Valve Setpoint Tolerance Relaxation". A copy of YAEC-1847 is enclosed in Section VIII herein. YAEC-1847 demonstrates that the licensing basis criteria are still met when the relaxed Code safety valve tolerance of ± 3% is assumed. YAEC-1847 demonstrates the following:

1. For events where Departure From Nucleate Boiling Ratio (DNBR) is a concern, there will be no reduction in the calculated minimum DNBR,

2. For events where overpressurization is a concern, the safety limits are not exceeded, and

3. For events where offsite doses are a concern, the safety limits are not exceeded, and

4. For LOCA events, the acceptance criteria for Emergency Core Cooling System (ECCS) performance are not exceeded.

The proposed Code safety valve setpoint tolerance relaxation does not affect UFSAR Departure From Nucleate Boiling (DNB) evaluations. The UFSAR DNB evaluations take credit for operation of the pressurizer Power Operated Relief Valves (PORVs) which have a setpoint of 2400 psia. This setpoint is lower than the proposed lower limit of 2425 psia for the PSVs. The UFSAR conservatively assumes PORV operation because lower Reactor Coolant System pressures yield more limiting values of DNBR.

The YAEC-1847 evaluation of the lower limit of the relaxed PSV and MSSV setpoint tolerance, - 3%, concludes that there will be no increase in the frequency of challenges to either the PSVs or the MSSVs due to the relaxed setpoint tolerance. A margin of 25 psi remains between the PORV setpoints and the relaxed PSV setpoints. The proposed lower limit of the PSV setpoint (- 3% tolerance) is 2425 psia. This PSV setpoint is sufficiently above the PORV setpoint of 2400 psia to prevent unnecessary challenges to the PSVs. A margin of 24 psi remains between the Atmospheric Steam Dump Valve (ASDV) setpoints and the relaxed Group 1 MSSV setpoints. The Group 1 MSSVs, which have the lowest opening setpoint, have a proposed lower limit setpoint (- 3% tolerance) of 1164 psia. This is sufficiently above the ASDV opening setpoint of 1140 psia to prevent unnecessary challenges to the MSSVs. YAEC-1847 also concluded that the proposed lower limit of the PSV setpoint, 2425 psia, will not affect the automatic reactor trip on high pressu pressure which occurs at 2400 psia.

The YAEC-1847 evaluation of the relaxed Code safety valve setpoint tolerance change considered each of the non-LOCA transient events documented in Chapter 15 of the UFSAR. For most of the UFSAR Chapter 15 non-LOCA transient events it is apparent that a relaxation in the PSV and MSSV setpoint tolerance cannot have an effect on the existing UFSAR analysis as for example when the opening setpoint is not challenged. The following UFSAR Chapter 15 non-LOCA transient events are evaluated in YAEC-1847:

Reduction in Feedwater Temperature
Increase in Feedwater Flow
Excessive Increase in Steam Flow
Inadvertent Opening of a Steam Generator Relief or Safety Valve
Steam Line Break
Loss of External Load
Turbine Trip
Inadvertent Closure of MSIVs
Loss of Condenser Vacuum
Loss of Off-Site Power
Loss of Feedwater Flow
Feedwater Line Break
Partial and Complete Loss of Reactor Coolant Flow
Reactor Coolant Pump Shaft Seizure
Reactor Coolant Pump Shaft Break
Uncontrolled RCCA Withdrawal From Subcritical Condition
Uncontrolled RCCA Withdrawal at Power
RCCA Misoperation
Startup of an Inactive Reactor Coolant Pump
Boron Dilution
Fuel Loading Error
RCCA Ejection
Inadvertent Operation of ECCS During Power Operation
C''CS Malfunction That Increases Reactor Coolant Inventory
Decrease in Reactor Coolant Inventory (non-LOCA)

The most detailed YAEC-1847 evaluation of the non-LOCA transient events was performed for the limiting pressurization transient, the Turbine Trip. This event was simulated using the RETRAN02 MOD5 computer code. This computer code has been approved by the NRC for this type of system transient as documented in NRC Safety

Evaluation Report dated November 1, 1991. Comparisons of the UFSAR Turbine Trip analysis with the RETRAN analysis of the Turbine Trip event were performed by YAEC to demonstrate that the RETRAN model provides comparable results. Agreement with the UFSAR results was very close, with the RETRAN analysis results slightly overpredicting the peak pressure as compared to the UFSAR. Evaluations with the relaxed PSV and MSSV setpoint tolerances were then performed with the RETRAN model demonstrating that the peak pressure remains well below the Condition II limit (Events of Moderate Frequency) of 110% of design pressure, or 2750 psia for the primary system and 1320 psia for the secondary system.

The impact of the proposed MSSV setpoint tolerance relaxation on the UFSAR design basis LOCA events were evaluated (the PSVs are not challenged in a LOCA transient therefore the proposed PSV setpoint tolerance relaxation does not affect LOCA analyses). The limiting LOCA analysis for Seabrook Station is a large break LOCA event yielding a Peak Clad Temperature (PCT) of 2041.2 °F. The limiting large break LOCA event is not affected by the proposed MSSV setpoint tolerance relaxation because the MSSVs are not challenged in this event as a result of the drawdown of secondary system pressure by the primary system pressure. The proposed relaxation of the MSSV setpoint tolerance does however affect the results of the limiting small break LOCA analysis which does predict a challenge to the MSSVs. The UFSAR evaluation of the limiting small break LOCA event specifies a PCT of 1790.0 °F (reference UFSAR Sec. 15.6). The current small break LOCA PCT including margin allocations reported to the NRC by NHY on July 26, 1991 (Reference NYN-91120) is 1973.2 °F. The YAEC-1847 evaluation of the Code safety valve setpoint tolerance relaxation specifies an increase in the limiting small break LOCA PCT of about 2.5 °F. YAEC-1847 recommends that a conservative PCT penalty of 5 °F be applied to the small break LOCA PCT result and be tracked in accordance with 10CFR50.46 reporting requirements. This increase in PCT is less than 50 °F and therefore is not a "significant" change as defined by 10CFR50.46. The revised small break LOCA PCT value of 1978.2 °F remains below the 2200 °F limit as well as below the large break LOCA PCT value of 2041.2 °F. NHY will report the small break LOCA PCT increase resulting from the proposed Code safety valve setpoint tolerance relaxation in its 10CFR50.46 annual report subsequent to NRC approval of this license amendment request.

The impact of the proposed Code safety valve setpoint tolerance relaxation on the design basis Steam Generator Tube Rupture (SGTR) radiological consequences is evaluated in YAEC-1847. An overall increase in Steam Generator steam mass release is predicted due to the increased MSSV "open" time from 20 seconds to 31.5 seconds. The MSSV "open" time occurs immediately after the reactor/turbine trip which is early on in the SGTR event and lasting only approximately 30 seconds. YAEC-1847 predicts that an incremental dose increase of approximately 2% will occur due to the slightly longer MSSV "open" time. The incremental dose increase is small, to the extent that it is within the round-off error of the predicted Exclusion Area Boundary two hour dose or the Low Population Zone eight hour dose. The design basis SGTR radiological consequences are specified in NHY's letter dated April 16, 1991, "Analysis of a postulated Design Basis Steam Generator Tube Rupture for Seabrook Station" (Reference NYN-91061)

V. Determination of Significant Hazards for License Amendment Request 91-11

(1) The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

New Hampshire Yankee is proposing to revise the Seabrook Station Technical Specifications to allow a relaxation in the Pressurizer Safety Valve (PSV) and Main Steam Safety Valve (MSSV) setpoint tolerances to ± 3% for ASME Section XI testing acceptance criteria. The proposed Technical Specification changes also require that the PSV and MSSV setpoints be restored to within ± 1% of their nominal setpoints following testing. New Hampshire Yankee is also proposing to revise the BASES for Technical Specification 3/4.7.1.1 to specify the correct Edition of the ASME Boiler and Pressure Vessel Code, Section III applicable to the MSSVs. Additionally, NHY is proposing to correct a typographical error in the BASES for Technical Specification 3/4.7.1.1. The BASES currently contain an incorrect reference to Technical Specification Table 3.7-2, whereas the correct reference is to Table 3.7-1.

The impact of the relaxed PSV and MSSV setpoint tolerance on the licensing basis analysis documented in the Seabrook Station Updated Final Safety Analysis Report (UFSAR), Chapter 15, has been reviewed by Yankee Atomic Electric Company (YAEC) and documented in topical report YAEC-1847 "Seabrook Station Code Safety Valve Setpoint Tolerance Relaxation". YAEC-1847 demonstrates that the licensing basis criteria are still met when the relaxed Code safety valve tolerance of ± 3% is assumed. YAEC-1847 demonstrates the following:

1. For events where Departure From Nucleate Boiling Ratio (DNBR) is a concern, there will be no reduction in the calculated minimum DNBR,

2. For events where overpressurization is a concern, the safety limits are not exceeded, and

3. For events where offsite doses are a concern, the safety limits are not exceeded, and

4. For LOCA events, the acceptance criteria for Emergency Core Cooling System performance are not exceeded.

(2) The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The overpressure protection function of the PSVs and MSSVs is not affected by the proposed relaxation of their setpoint tolerance. The nominal setpoint on the PSVs and the MSSVs will not be changed. New Hampshire Yankee proposes to utilize the ± 3% tolerance for the "as-found" acceptance criteria for additional valve testing required by ASME Section XI, Article IWV-3513. The proposed Technical Specification revisions require that the PSV and MSSV setpoints be restored to within ± 1% of their nominal setpoints following testing. The evaluation of the proposed relaxation of the PSV and MSSV setpoint tolerances documented in YAEC-1847 demonstrates that new or different kinds of accidents are not created by the proposed changes.

(3) The proposed changes do not result in a significant reduction in the margin of safety.

The Seabrook Station overpressure protection design incorporates three Code safety valves on the primary system pressurizer and a total of twenty Code safety valves on the four main steam lines (five per line) in the secondary system. The proposed Technical Specification changes involve a relaxation of the setpoint tolerance for the Code safety valves in the primary and secondary systems. The YAEC-1847 evaluation of the Code safety valve setpoint tolerance relaxation demonstrates that peak pressures in the primary and secondary systems during the limiting pressurization transient (Turbine Trip) will remain within the limits for Condition II events (Faults of Moderate Frequency). The limiting small break LOCA event which involves the opening of the MSSVs is evaluated in YAEC-1847 which predicts a small increase in the Peak Clad Temperature (PCT). The small break LOCA PCT increase is not significant as defined by 10CFR50.46 nor are the PCT limits of 10CFR50.46 exceeded. YAEC-1847 also demonstrates that the frequency of challenges to the Code safety valves will not increase nor will the reactor trip on high pressurizer pressure be impacted as a result of the lowered setpoint tolerance (- 3%). Additionally, YAEC-1847 demonstrates that the radiological consequences associated with a Steam Generator Tube Rupture (SGTR) event are only minimally increased and remain within the round-off error applied to the SGTR radiological consequences which were delineated in NHY's April 16, 1991 letter to the NRC.

VI. Proposed Schedule for License Amendment Issuance and Effectiveness

New Hampshire Yankee (NHY) requests NRC review of License Amendment Request 91-11 and issuance of a license amendment having immediate effectiveness by August 31, 1992. This schedule is proposed in support of Code safety valve testing which is required to be performed pursuant to the NHY Inservice Test Program during the second refueling outage which is scheduled to begin in September 1992.

VII. <u>Environmental Impact Assessment</u>

New Hampshire Yankee (NHY) has reviewed the proposed license amendment against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, NHY concludes that the proposed change meets the criteria delineated in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

VIII. Other Supporting Information

Enclosure One:

Yankee Atomic Electric Company Topical Report YAEC-1847 "Seabrook Station Code Safety Valve Setpoint Tolerance Relaxation", February 28, 1992.

New Hampshire Yankee
May 5, 1992

<u>ENCLOSURE ONE TO NYN-92059</u>