BORATION CONTROL

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (Continued)

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

Insert "B.3/4.1.1.3"

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS in MODES 1, 2, or 3, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT from expected operating conditions after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 22,000 gallons of 7000 ppm borated water from the boric acid storage tanks or a minimum contained volume of 477,000 gallons of 2000 ppm borated water from the refueling water storage tank (RWST).

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable in MODES 4, 5, and 6 provides assurance that a mass addition pressure transient can be relieved by operation of a single PORV or an RHR suction relief valve.

As a result of this, only one boron injection system is available. This is acceptable on the basis of the stable reactivity condition of the reactor, the emergency power supply requirement for the OPERABLE charging pump and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

Insert "B.3/4.1.1.3"

Prior to initial operation above 5% of RATED THERMAL POWER after each fuel loading, the MTC is measured as required by Surveillance Requirement 4.1.1.3.a. A measurement bias is derived from the difference between test measurement and test prediction. All predicted values of MTC for the cycle are conservatively corrected based on the measurement bias. The corrected predictions are then compared to the maximum upper limit of Technical Specification 3.1.1.3. Control rod withdrawal limits are established, if required, to assure all corrected values of predicted MTC will be less positive than the maximum upper limit required by Technical Specification 3.1.1.3.

Section 2

INCORE DETECTOR SYSTEM

Question

Explain how the fixed incore detector strings and the movable incore detectors are used to satisfy the OPERABILITY requirements of Technical Specification 3.3.3.2.

Response

The Incore Detector System consists of either a) fixed detector strings and their associated signal processing equipment, or b) movable incore detectors and their associated signal processing equipment. The OPERABILITY requirements of Technical Specification 3.3.3.2 may be satisfied by either the fixed detectors or the moveable detectors, but not by a combination of both.

A proposed enhancement to the Bases of Technical Specification 3/4.3.3.2 INCORE DETECTOR SYSTEM utilizing the information provided above is included on the following pages.

MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS (Continued)

and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 INCORE DETECTOR SYSTEM Insert "B.3/4.3.3.2"

The OPERABILITY of the Incore Detector System ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore detectors may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of Appendix A to 10 CFR Part 50.

Insert "B.3/4.3.3.2"

The Incore Detector System consists of either a) fixed detector strings and their associated signal processing, or b) movable incore detectors and their associated signal processing. OPERABILITY may be met by either fixed detectors or movable detectors but not by a combination of both.

Section 3

FLOW MEASUREMENT UNCERTAINTIES

Question

Explain how flow measurement uncertainties are used in demonstrating compliance with Technical Specification 3.2.5, DNB PARAMETERS.

Response

The Limiting Condition for Operation (LCO) on Reactor Coolant System (RCS) flow in Technical Specification 3.2.5 assures that RCS flow meets or exceeds the values used in safety analyses. Flow measurement accuracy is considered as follows:

RCS Flow ≥ Thermal Design Flow

A series of flow measurement uncertainties specific to Seabrook Station has been developed for use in plant operating procedures for comparing measured RCS flow to Thermal Design Flow (382,800 gpm). These measurement uncertainties are included in North Atlantic administrative procedures. Typically, the RCS flow rate determination required by Technical Specification 4.2.5.3 is performed at 90% Rated Thermal Power (RTP) using the Main Plant Computer System (MPCS) analog data as input with an allowance for 18 month instrument drift. The applicable measurement uncertainty is currently 3.27%. This uncertainty may be reduced to 2.5% by applying special instrument calibration to eliminate drift for certain instruments and performing the flow rate determination at higher power levels. The 12 hour surveillance requirement of Technical Specification 4.2.5.1 is performed using MCPS flow indication or Main Control Board indicators with an allowance for flow instrument normalization. The applicable measurement uncertainty for performing the 12 hour surveillance may be as high as 3.7%. In summary, the applicable flow measurement uncertainty used for comparison to Thermal Design Flow may vary from 2.5% to 3.7%. For example, with an applicable measurement uncertainty of 3.27%, the required value for measured flow is ≥ 395, 320 gpm (1.0327 x 382,800 gpm).

RCS Flow ≥ Minimum Measured Flow

In the analyses of DNB related events, a Minimum Measured Flow (392, 800 gpm) is used in lieu of the Thermal Design Flow. A conservative RCS flow measurement uncertainty of 5% is considered in the development of the design DNB limit value applied to these events. A separate allowance for flow measurement uncertainty when comparing RCS flow to Minimum Measured Flow is not required since a flow uncertainty of 5% is already included in the design DNB limit value and 5% bounds the range of applicable measurement uncertainties described above.

Question 4:

In group 12 on page 4 of Introduction and Description of Proposed Changes submitted with LAR 93-18, is not *Surveillance Requirement* 4.5.2.h.3) also being modified? In group 14 on page 4, is not *Administrative Control* 6.8.1.6.b also being modified?

Response 4:

Yes. *Surveillance Requirement* 4.5.2.h.3) and *Administrative Control* 6.8.1.6.b should be added to the list of modifications in groups 12 and 14, respectively.

Question 5:

In group 11 on page 4 of Introduction and Description of Proposed Changes submitted with LAR 93-18, the Pressurizer Pressure-Low Setpoint and Allowable Value in T.S. Table 3.3-4 are revised consistent with a revised safety analysis limit of 1665 psia. Describe the relationship between the safety analysis limit, the Setpoint, and the Allowable value.

Response 5:

The methodology which describes the relationship between the safety analysis limit, the Setpoint, and the Allowable Value is summarized in the *Bases* to Technical Specifications 3/4.3.1 and 3/4.3.2 Reactor Trip System and Engineered Safety Features Actuation System Instrumentation. The methodology is also described in UFSAR Section 7.1.2.1i and in the response to RAI 420-8. The methodology was applied in the initial setup of plant systems using plant-specific instrumentation parameters to derive Setpoints and Allowable Values.

Question 6:

In group 11 on page 4 of Introduction and Description of Proposed Changes submitted with LAR 93-18, "Values of Total Allowance (TA), Z, and Sensor Error (S) are deleted from the Technical Specifications consistent with the format used in NUREG 1431, Standard Technical Specifications Westinghouse Plants." Is this statement intended to apply to T.S. pages 3/4 3-24, 3/4 3-28, and 2-4?

Response 6:

No. The statement is intended to apply to the proposed changes shown on T.S. pages 2-4, and 3/4 3-24 only.

Section 4

PROPOSED TECHNICAL SPECIFICATIONS

The following information is provided in response to questions raised during the review of proposed changes to Seabrook Station *Technical Specifications* (T.S.) submitted with License Amendment Request LAR 93-18.

Question 1:

Define the OPERABILITY of the Incore Detector System.

Response 1:

OPERABILITY of the Incore Detector System (either fixed or movable detectors) is defined in T.S. 3.3.3.2. Although proposed changes are being made to this specification by LAR 93-18 to include APPLICABILITY to the FIDS Alarm, the definition of OPERABILITY for the fixed detectors remains unchanged from Amendment 27 which approved use of the Fixed Incore Detector System (FIDS) for performing incore power distribution surveillance.

Question 2:

How many detector locations must function to assure OPERABILITY?

Response 2:

From T.S. 3.3.3.2, it is seen that at least 75% of the detector locations and a minimum of 2 detector locations per core quadrant must be functioning for the Incore Detector System to be OPERABLE.

Question 3:

Discuss the wording of proposed Axial Flux Difference *Surveillance Requirement* 4.2.1.1.b in LAR 93-18.

Response 3:

Axial Flux Difference *Surveillance Requirement* 4.2.1.1.b in LAR 93-18 is unchanged from the current specification. The wording of this surveillance requirement is standard for PWRs of a design similar to Seabrook Station.

Question 4:

In group 12 on page 4 of Introduction and Description of Proposed Changes submitted with LAR 93-18, is not *Surveillance Requirement* 4.5.2.h.3) also being modified? In group 14 on page 4, is not *Administrative Control* 6.8.1.6.b also being modified?

Response 4:

Yes. *Surveillance Requirement* 4.5.2.h.3) and *Administrative Control* 6.8.1.6.b should be added to the list of modifications in groups 12 and 14, respectively.

Question 5:

In group 11 on page 4 of Introduction and Description of Proposed Changes submitted with LAR 93-18, the Pressurizer Pressure-Low Setpoint and Allowable Value in T.S. Table 3.3-4 are revised consistent with a revised safety analysis limit of 1665 psia. Describe the relationship between the safety analysis limit, the Setpoint, and the Allowable value.

Response 5:

The methodology which describes the relationship between the safety analysis limit, the Setpoint, and the Allowable Value is summarized in the *Bases* to Technical Specifications 3/4.3.1 and 3/4.3.2 Reactor Trip System and Engineered Safety Features Actuation System Instrumentation. The methodology is also described in WCAP-9712, C.R. Tuley, D.R. Sharp, and R.B. Miller, "Westinghouse Setpoint Methodology for Protection and Control Systems", 1981. The methodology has previously been applied by the licensee using plant-specific instrumentation parameters to derive Setpoints and Allowable Values.

Question 6:

In group 11 on page 4 of Introduction and Description of Proposed Changes submitted with LAR 93-18, "Values of Total Allowance (TA), Z, and Sensor Error (S) are deleted from the Technical Specifications consistent with the format used in NUREG 1431, Standard Technical Specifications Westinghouse Plants." Is this statement intended to apply to T.S. pages 3/4 3-24, 3/4 3-28, and 2-4?

Response 6:

No. The statement is intended to apply to the proposed changes shown on T.S. pages 2-4, and 3/4 3-24 only.

Section 5

YAEC-1871: "SAFETY ANALYSIS IN SUPPORT OF WIDE-BAND OPERATION AND CORE DESIGN ENHANCEMENTS FOR SEABROOK STATION"

The following information is provided in response to questions raised during the review of YAEC-1871. This report was submitted as part of the supporting information for License Amendment Request LAR 93-18.

Question 1:

Section 3.8 of YAEC-1871 states "The assumed uncertainty on pressurizer pressure has been increased from ±30 psi to ±50 psi. This change is reflected in Technical Specification 3.2.5, DNB Parameters." Why is this change made?

Response 1:

This change is made to facilitate continued Seabrook Station compliance with the *Surveillance Requirement* of Technical Specification 3.2.5 as it pertains to the *Limiting Condition for Operation* (LCO) placed on pressurizer pressure. The increased uncertainty allows flexibility in selection of the instrumentation used to measure the process variable. Results of the safety analysis in YAEC-1871 demonstrate that the relaxation in pressurizer pressure uncertainty does not reduce any margin of safety as defined in the *Bases* to the Technical Specifications.

Question 2:

Section 3.9 of YAEC-1871 states "The analysis values of the low pressurizer pressure safety injection actuation setpoint and actuation delay (offsite power unavailable) has been changed from 1760 psia and 27 seconds to 1665 psia and 30 seconds." Why are these changes made?

Response 2:

The reduction in the low pressure safety injection actuation setpoint will reduce the probability of unnecessary safety injection actuation following a reactor trip. The increase in the actuation delay will facilitate continued Seabrook Station compliance with this safety analysis input parameter. Results of the safety analysis in YAEC-1871 demonstrate that the reduction in the low pressurizer pressure safety injection actuation setpoint and the increase in the actuation delay time do not reduce any margin of safety as defined in the *Bases* to the Technical Specifications.

Question 3:

Section 3.10 of YAEC-1871 states "The analysis values for Emergency Core Cooling System pumped injection performance characteristics have been revised to more accurately represent the specific Seabrook System." Why are these changes made?

Response 3:

The proposed changes will facilitate continued Seabrook Station compliance with *Surveillance Requirement* 4.5.2.f.3 as it pertains to the differential pressure of a Residual Heat Removal pump on recirculation flow. The revised injection curves have been incorporated in all analyses presented in YAEC-1871. Results of the safety analysis in YAEC-1871 demonstrate that the revised injection curves do not reduce any margin of safety as defined in the *Bases* to the Technical Specifications.

Question 4:

Section 3.11 of YAEC-1871 states "The assumed Emergency Feedwater (EFW) temperature has been increased from 88°F to 100°F to be consistent with the design limit for the EFW pumps. The EFW System actuation time delay has been increased from 60 seconds to 75 seconds." Why are these changes made?

Response 4:

The source of water for the EFW System is the Condensate Storage Tank. The Condensate Storage Tank temperature is monitored. The EFW System actuation time delay is measured as part of periodic performance tests. Both actions are taken to assure compliance with these safety analysis input parameters. The changes in these parameters are made to provide operating margin while facilitating continued Seabrook Station compliance with the safety analysis. Results of the safety analysis in YAEC-1871 demonstrate that the increases in EFW temperature and actuation time delay do not reduce any margin of safety as defined in the *Bases* to the Technical Specifications.

Question 5:

Reference 10 in YAEC-1871 is a report on fuel densification. Reference 15 is a letter to the NRC responding to concerns regarding FROSSTEY-2. Reference 16 is a letter to Vermont Yankee Nuclear Power Corporation including the NRC Safety Evaluation Report for FROSSTEY-2. Provide a reference describing the approved analysis methodology using FROSSTEY.

Response 5:

The approved analysis methodology using FROSSTEY-2 is described in References 3 through 14 in this supplementary information package. The FROSSTEY-2 code consists of the original FROSSTEY code along with a series of letters documenting changes. Reference 14 is the NRC Safety Evaluation Report (SER).

Question 6:

Section 5.0.1 of YAEC-1871 states that a 2% uncertainty on core power is applied for calorimetric error, yet in previous material Yankee states that the applied uncertainty on RCS flow measurement is in the range of 2.5% to 3.7%. Explain this difference.

Response 6:

Core power at Seabrook Station is determined from a calorimetric measurement using secondary plant parameters. The plant-specific uncertainty in this measurement is bounded by the standard 2% allowance used in safety analyses and quoted in Section 5.0.1 of YAEC-1871. RCS flow is then determined from core power (with its uncertainty), RCS loop ΔT measurements (additional uncertainty), and RCS flow indicator normalization (additional uncertainty). Since the determination of RCS flow inherently involves the use of core power with its uncertainty along with other uncertainties, the applied flow measurement uncertainty will be greater than the core power uncertainty. The range of applicable RCS flow measurement uncertainties is 2.5% to 3.7% as discussed previously in this package of supplementary information for LAR 93-18.

Question 7:

Section 5.1.3 of YAEC-1871 discusses "Excessive Increase in Secondary Steam Flow". On page 5.1-5, Yankee states "The Reactor Control System is designed to accommodate a 10 percent step load increase or a 5 percent per minute ramp load increase in the range of 15 percent to 100 percent of full power." On page 5.1-6, Yankee states "Four cases are analyzed to demonstrate the plant behavior following a 10 percent step load increase from rated load." Why is the 5%/min ramp load increase event not analyzed?

Response 7:

Section 5.1.3 of YAEC-1871 provides an evaluation of plant response to initiating events characterized as "Excessive Increase in Secondary Steam Flow". These events are considered to be ANS Condition II - Faults of Moderate Frequency. These faults, at worst, result in actuation of the Reactor Protection System with the plant being capable of returning to operation. The limiting event of this type was previously identified by the NSSS vendor as a 10% step load increase from rated load with analysis documented in UFSAR Section 15.1.3. Section 5.1.3 of YAEC-1871 provides a reanalysis of the limiting 10% step load increase from rated load. Results demonstrate that the acceptance criteria for this type of event continue to be met using:

1) the changes to analysis input parameters and assumptions discussed in Section 3.0 of YAEC-1871; and,

2) the analysis methodology discussed in Section 2.0 of YAEC-1871.

Additional justification for not providing analysis of the 5%/min ramp load increase in Section 5.1.3 is presented below.

As stated in Section 5.1.3 of YAEC-1871, with the turbine governor valves fully open, the maximum steam flow will be 105% of rated. However, Yankee has conservatively assumed that 110% of rated steam flow can be attained in 0.0 seconds. With an assumed ramp load increase of 5%/min from rated load, steam flow will attain 110% in 120 seconds unless a plant trip occurs sooner. The effect of the slower ramp load increase on each of the four step load cases analyzed in Section 5.1.3 is evaluated as follows:

1. Reactor Control in Manual With the Most Positive MTC

   Plant parameters change gradually until a reactor trip on low pressurizer pressure occurs. The PMTC causes core power to continuously decrease prior to trip. The decreasing core power causes the minimum DNBR to continuously increase from its initial value. A ramp load increase would produce a slower transient; the minimum DNBR would still occur at the start of the event; and, the minimum DNBR would be identical to the corresponding case with a step load increase.

2. Reactor Control in Manual With the Most Negative MTC

   Plant parameters change gradually until a new equilibrium operating condition is reached. The minimum DNBR occurs at the new equilibrium operating condition. The ramp load increase would produce a slower transient; the new equilibrium operating condition would be identical to the case with a step load increase; and, the minimum DNBR would be identical to the corresponding case with a step load increase.

3. Reactor Control in Automatic With the Most Positive MTC

   The combination of control rod motion and PMTC causes oscillatory responses in plant parameters. The oscillations are damped and power eventually settles at 110% of initial. The minimum DNBR occurs during one of the upward swings in core power. A ramp load increase would produce a slower transient; control rod motion would be more effective; the magnitude of the core power swings would be reduced; and the minimum DNBR would be less limiting than the corresponding case with a step load increase.

4. Reactor Control in Automatic With the Most Negative MTC

   The negative MTC allows control rod motion to be effective. Plant parameters change gradually with a slight power overshoot until a new equilibrium operating condition is reached. The minimum DNBR occurs during the power overshoot. A ramp load increase would produce a slower transient; control rod motion would be more effective; the magnitude of the power overshoot would be reduced; and the minimum DNBR would be less limiting than the corresponding case with a step load increase.

In summary, a 5%/min ramp load increase event would produce a minimum DNBR which is either identical to or less limiting than a 10% step load increase.

The analysis of the 10% step load increase in Section 5.1.3 contains several bounding assumptions for the single purpose of demonstrating compliance with the acceptance criteria for ANS Condition II faults. These bounding assumptions render the analysis inappropriate for demonstrating the capability of the Reactor Control System to accommodate a 10% step load increase or a 5%/minute ramp load increase. The statement on page 5.1-5 of Section 5.1.3 regarding the Reactor Control System is taken from UFSAR Section 15.1.3 and is provided as a reaffirmation of the control system design. The changes to analysis input parameters and assumptions and the use of Yankee's analysis methodology do not impact the design of the Reactor Control System.

Question 8:

Se 5.3.1 of YAEC-1871 discusses "Partial Loss of Forced Reactor Coolant Flow". On page 5.3-3, Yan e states "The most negative value of the Doppler defect is used. A conservatively high positive moderator temperature coefficient is assumed since this results in the maximum core power during the initial part of the transient when the minimum DNBR is reached." Should a value other than the most negative Doppler defect have been used? Also, why was a conservatively high positive MTC used when the proposed changes to the Technical Specifications limit the MTC to zero at full power?

Response 8:

As part of the calculations supporting the loss of RCS flow event analyses documented in YAEC-1871, Yankee previously performed a sensitivity study on the effect of the Doppler defect. The study showed 1) the minimum DNBR was not strongly sensitive to the Doppler defect and 2) the most negative defect yields the most limiting minimum DNBR. Prior to trip, fuel temperature increases due to the flow coastdown. The most negative Doppler defect inserts negative reactivity. After trip, fuel temperature decreases due to the decrease in power. The most negative Doppler defect inserts positive reactivity and delays the post trip power decrease. The minimum DNBR occurs during the power decrease just after trip.

Although the proposed changes to the Seabrook Station Technical Specifications submitted in LAR 93-18 limit the MTC to zero at full power, Yankee has assumed a conservative positive value for the MTC in many of the safety analyses documented in YAEC-1871. The assumed PMTC value includes an allowance for measurement uncertainty.

Question 9:

Section 5.3.3 of YAEC-1871 discusses "Reactor Coolant Pump Shaft Seizure (Locked Rotor)". Clarify the discussion on page 5.3-7 regarding the treatment of uncertainties.

Response 9:

A deterministic treatment of uncertainties on plant parameters is used to calculate a conservative RCS pressure response for evaluating challenges to RCS integrity. Nominal plant parameters are otherwise used in DNBR evaluations consistent with the use of the Revised Thermal Design Procedure.

Question 10:

Section 5.3.3 of YAEC-1871 discusses "Reactor Coolant Pump Shaft Seizure (Locked Rotor)". Regarding Evaluation of DNB in the Core During the Accident on page 5.3-8, DNB is assumed to occur at the start of the transient to calculate a conservative temperature response. What were the actual results of the DNBR evaluation?

Response 10:

The peak clad surface temperature and the fraction of failed fuel derived from the DNB evaluation for this event are bounded by those reported for the Reactor Coolant Pump Shaft Seizure (Locked Rotor) Followed by Loss of Offsite Power in Section 5.3.4 of YAEC-1871. In conclusion, the peak clad surface

temperature is less than 1104°F (taken from Table 5.3-2) and the fraction of failed fuel is predicted to be less than 8% (taken from Radiological Consequences on page 5.3-11).

The peak clad surface temperature is conservatively calculated assuming DNB occurs coincident with the locked rotor.

The failed fuel fraction is calculated by performing a DNB analysis with a variety of power distributions and assuming that all rods which pass through DNB become failed rods. The time of DNB for this analysis depends on the assumed power distribution and was found to vary from 1.0 seconds to 1.5 seconds after the locked rotor.

Question 11:

Provide a discussion of flow reversal in the faulted RCS coolant loop, if this occurs, for the Locked Rotor event with offsite power available (Section 5.3.3) and with subsequent loss of offsite power (Section 5.3.4).

Response 11:

For the locked rotor with offsite power available, the response of the faulted loop coolant flow is illustrated in the bottom part of Figure 5.3-5, Sheet 1. Continued operation of the Reactor Coolant Pumps (RCPs) in the unfaulted loops causes the faulted loop flow to reverse.

For the locked rotor with a loss of offsite power, the response of the faulted loop coolant flow is illustrated in the bottom part of Figure 5.3-6, Sheet 1. Initially, operation of the RCPs in the unfaulted loops causes the faulted loop flow to reverse. Following the loss of offsite power the RCPs in the unfaulted loops coast down and the magnitude of the reverse flow in the faulted loop is reduced.

Question 12:

Pertaining to the analysis of the Locked Rotor event in YAEC-1871 Sections 5.3.3 and 5.3.4, provide a statement of compliance to TMI Action Item II.K.3.5 as required by the Standard Review Plan.

Response 12:

Seabrook Station's compliance with TMI Action Plan Requirements are discussed in Updated Final Safety Analysis Report (UFSAR), Section 1.9. The compliance is not affected by the changes to analysis input parameters and assumptions discussed in Section 3.0 of YAEC-1871, and the analysis methodology discussed in Section 2.0 of YAEC-1871. The discussion in UFSAR Section 1.9 is repeated below in abbreviated form:

Task II.K.3.5 Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident (NUREG-0737)

Position:

Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution ... Until a better solution is found, the reactor coolant pumps should be tripped automatically in case of small break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623.

Response:

... Westinghouse (in support of the Westinghouse Owners Group) performed an analysis of delayed ... RCP trip during small break LOCAs. This analysis is ... the basis for the Westinghouse position ... i.e., automatic RCP trip is not necessary since sufficient time is available for manual tripping of the RCPs.

...

The Revision 1 to the WOG Emergency Response Guidelines has been implemented into the plant specific procedures. The training program includes instruction to operators in their responsibility for performing RCP trip in the event of a small break LOCA. In particular, the operators are trained in prioritization of actions following engineered safety features actuation. The instrumentation the operators use to determine the need for RCP trip is part of the instrumentation used for Inadequate Core Cooling ...

In light of the above information, Seabrook Station does not consider that design modifications are necessary.

Question 13:

In relation to the radiological consequences of the locked rotor event with subsequent loss of offsite power, page 5.3-11 of YAEC-1871 states "The offsite whole body gamma doses for this event are 1.4 rem and 1.9 rem at the EAB and LPZ, respectively." Are these doses reversed? If not, discuss why the dose to the LPZ is greater than the dose to the EAB.

Response 13:

The magnitude of the EAB and LPZ whole body doses (as well as the thyroid doses) are approximately equal because they are integrated doses over different time periods. The EAB dose is by definition a 2 hour dose based on the 0 to 2 hour release of fission products. The LPZ dose is an 8 hour integrated dose based on an 8 hour release period. The pathway to the environment is an assumed 1 gpm leak from the primary side to the secondary side for an 8 hour time period. The greater dispersion afforded by the LPZ distance (approximately a factor of 2 for the time periods of interest) is offset by the larger release inventory (approximately a factor of 4).

Question 14:

Regarding the conclusion of the analysis of the locked rotor with loss of offsite power on page 5.3-11 of YAEC-1871, state the value of the peak RCS pressure.

Response 14:

The calculated peak RCS pressure for the locked rotor with loss of offsite power is shown in Table 5.3-2 of YAEC-1871. The value is 2604 psia.

Question 15:

Provide a Table which cross-references the transient analyses discussed in the Standard Review Plan (NUREG 0800) and the analysis sections in YAEC-1871.

Response 15:

The enclosed table provides a cross reference between transients discussed in the Standard Review Plan (NUREG 0800) and the corresponding analysis sections in YAEC-1871. Transients are listed in the order used in NUREG 0800.

COMPARISON OF ANALYZED TRANSIENTS STANDARD REVIEW PLAN (NUREG 0800) VERSUS YAEC-1871	
Standard Review Plan (NUREG 0800) Transients	**YAEC-1871 Transients**
15.1.1, 15.1.2, 15.1.3, 15.1.4 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Secondary Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	5.1.1 Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature 5.1.2 Feedwater System Malfunctions Causing an Increase in Feedwater Flow 5.1.3 Excessive Increase in Secondary Steam Flow 5.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve
15.1.5 Steam System Piping Failures	5.1.5 Steam System Piping Failure
15.2.1 - 15.2.5 Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	5.2.1 Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow 5.2.2 Loss of External Load 5.2.3 Turbine Trip 5.2.4 Inadvertent Closure of Main Steam Isolation Valves 5.2.5 Loss of Condenser Vacuum and Other Events Resulting in a Turbine Trip
15.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries	5.2.6 Loss of Nonemergency AC Power to the Plant Auxiliaries (Loss of Offsite Power)
15.2.7 Loss of Normal Feedwater Flow	5.2.7 Loss of Normal Feedwater Flow
15.2.8 Feedwater System Pipe Breaks	5.2.8 Feedwater System Pipe Break
15.3.1 - 15.3.2 Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	5.3.1 Partial Loss of Forced Reactor Coolant Flow 5.3.2 Complete Loss of Forced Reactor Coolant Flow
15.3.3 - 15.3.4 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	5.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor) 5.3.4 Reactor Coolant Pump Shaft Seizure (Locked Rotor) Followed by Loss of Offsite Power 5.3.5 Reactor Coolant Pump Shaft Break
15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical or Low Power Startup Condition	5.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical or Low Power Startup Condition
15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	5.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power
15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)	5.4.3 Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)

COMPARISON OF ANALYZED TRANSIENTS (CONTINUED)	
Standard Review Plan (NUREG 0800) Transients	**YAEC-1871 Transients**
15.4.4 - 15.4.5 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	5.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature
15.4.6 Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant	5.4.5 Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant
15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	5.4.6 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
15.4.8 Spectrum of Rod Ejection Accidents	5.4.7 Spectrum of Rod Cluster Control Assembly Ejection Accidents
15.4.9 Spectrum of Rod Drop Accidents (BWR)	Not Applicable
15.5.1 - 15.5.2 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	5.5.1 Inadvertent Operation of ECCS During Power Operation 5.5.2 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory
15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	5.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve
15.6.2 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	5.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment
15.6.3 Radiological Consequences of Steam Generator Tube Failure	5.6.3 Steam Generator Tube Rupture
15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	Not Applicable
15.6.5 Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	5.6.4 Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
15.7.1 Waste Gas System Failure	5.7 Radioactive Release From a System or Component
15.7.2 Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	5.7 Radioactive Release From a System or Component
15.7.3 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	5.7 Radioactive Release From a System or Component
15.7.4 Radiological Consequences of Fuel Handling Accidents	5.7 Radioactive Release From a System or Component
15.7.5 Spent Fuel Cask Drop Accidents	5.7 Radioactive Release From a System or Component
15.8 Anticipated Transients Without Scram	5.8 Anticipated Transients Without Scram

Question 16:

Is YAEC-1698, "Analysis of a Postulated Design Basis Steam generator Tube Rupture For The Seabrook Nuclear Power Station", available to the NRC?

Response 16:

Yes, North Atlantic submitted YAEC-1698 to the NRC on April 16, 1991 (North Atlantic Letter NYN-91061) to address NRC concerns related to analysis of the design basis Steam Generator Tube Rupture (SGTR). The SGTR analysis documented in YAEC-1698 was performed using methodology consistent with YAEC-1856P, "System Transient Analysis Methodology Using RETRAN for PWR Applications", which is Reference 9 in YAEC-1871. The changes to analysis input parameters and assumptions discussed in Section 3.0 of YAEC-1871 and, the analysis methodology discussed in Section 2.0 of YAEC-1871 do not impact the SGTR analysis in YAEC-1698. LAR 93-18 contains no proposed changes to the Seabrook Station Technical Specifications required to support the SGTR analysis in YAEC-1698. The SGTR analysis in YAEC-1698 remains valid and was incorporated by reference into YAEC-1871 to provide a complete safety analysis in support of LAR 93-18.

REFERENCES

1. North Atlantic Letter NYN-93160, Dated November 23, 1993, "License Amendment Request 93-18: Wide Band Operation and Core Design Enhancements (TAC No. M87849)", T.C. Feigenbaum to USNRC.

2. YAEC-1871, "Safety Analysis in Support of Wide-Band Operation and Core Design Enhancements for Seabrook Station", by A.E. Ladieu, September 1993.

3. S. P. Schultz and K. E. St.John, "Method for the Analysis of Oxide Fuel Rod Steady-State Thermal Effects (FROSSTEY) Code/Model Description Manual," YAEC-1249P, April 1981.

4. S. P. Schultz and K. E. St.John, "Method for the Analysis of Oxide Fuel Rod Steady-State Thermal Effects (FROSSTEY) Code Qualification and Application," YAEC-1265P, June 1981.

5. Letter, R. W. Capstick (VYNPC) to V. L. Rooney (USNRC), "Vermont Yankee LOCA Analysis Method FROSSTEY Fuel Performance Code (FROSSTEY2)," FVY 87-116.

6. Letter, M. B. Fairtile (USNRC) to R. W. Capstick (VYNPC), "Request for Additional Information - FROSSTEY2 Fuel Performance Code (TAC No. 68216)," May 2, 1989.

7. Letter, M. B. Fairtile (USNRC) to R. W. Capstick (VYNPC), " Second Request for Additional Information - FROSSTEY2 Fuel Performance Code (TAC No. 68216)," May 24, 1989.

8. Letter, M. B. Fairtile (USNRC) to R. W. Capstick (VYNPC), "Request for Additional Information - FROSSTEY2 Fuel Performance Code (TAC No. 68216)," March 9, 1990.

9. Letter, L. A. Tremblay (VYNPC) to Document Control Desk (USNRC), "Responses to Request for Additional Information on FROSSTEY2 Fuel Performance Code," BVY 91-024, March 6, 1991.

10. Letter, C. E. Beyer (PNL) to S. L. Wu (USNRC), August 19, 1991.

11. Letter, C. E. Beyer (PNL) to S. L. Wu (USNRC), December 23, 1991.

12. Letter, L. A. Tremblay (VYNPC) to Document Control Desk, "LOCA Related Responses to Open Issues on FROSSTEY2 Fuel Performance Code," BVY 92-39, March 27, 1992.

13. Letter, L. A. Tremblay (VYNPC) to Document Control Desk, "FROSSTEY2 Fuel Performance Code - Vermont Yankee Response to Remaining Concerns," BVY 92-54, May 15, 1992.

14. Letter, P. Sears (USNRC) to L. A. Tremblay (VYNPC), "Vermont Yankee Nuclear Power Station, Safety Evaluation of FROSSTEY-2 Computer Code (TAC NO. M68216)," September 24, 1992.

Section 6

10CFR50.46 LOCA MODEL ASSESSMENTS ON THE PCT MARGIN UTILIZATION

Question

Provide a reference for the COSI test results referred to in Attachment VIII of License Amendment Request 93-18.

Response

The Attachment to LAR 93-18 labeled as "10CFR50.46 LOCA Model Assessments on the PCT Margin Utilization" refers to COSI test results. These tests are described in the following reference:

> WCAP-11767, Dan J. Shimeck, "COSI SI/Steam Condensation Experiment Analysis", Dated March 1988, Submitted to the NRC January 1994

The following editorial corrections are provided. In the second sentence of item 1, the phrase "To offset this affect," should be replaced with "To offset this effect,". In item 2 the phrase "drift flux floe regime map" should be replaced with "drift flux flow regime map".

Section 7

SEABROOK STATION FUEL UPGRADE PROGRAM LOCA SAFETY ANALYSIS REPORT

The following provides clarification and editorial corrections to the Seabrook Station Fuel Upgrade Program Safety Analysis Report.

On page 6 of the LOCA report, Reference 4 is identified as a detailed description of the BASH code. "Reference 4" on page 6 of the LOCA report should read "Reference 6".

On page 82 of the LOCA report, small break LOCA sensitivity studies are incorrectly stated to be available in References 22 and 23. These sensitivity studies are correctly described in Reference 22 only. Reference 23 shown on page 116 is identical to Reference 13. Reference 23 should be omitted or marked "Not used".