PSNH

Public Service of New Hampshire

SEABROOK STATION
Engineering Office:
1671 Worcester Road
Framingham, Massachusetts 01701
(617) - 872 - 8100

bcc: J.P. Cady, A.C. Cerne, S.D. Floyd, J.H. Herrin, G.F. McDonald, D.N. Merrill, D.E. Moody, NRC Chrono, R.P. Pizzuti, Projects-All, Projects-SLA, Ropes&Gray (Dignan/Ritsher/Gad), A.M. Shepard, J.W. Singleton, T.F. B4.2.99, G.S. Thomas, J.E. Tribble, UE&C&W (SB-15531), ASLB, R.J. Harrison, L. Walsh, A.E. Ladieu, P. Swanson, P.L. Anderson, W.N. Fadden, G. Tsouderos

April 8, 1983

SBN-498
T.F. B4.2.99

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing

References: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) USNRC Letter, dated February 8, 1983, "Resolution of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,' (Generic Letter No. 83-10c)," D. G. Eisenhut to All Applicants with Westinghouse (W) Designed Nuclear Steam Supply Systems (NSSS)

Subject: Response to NRC Generic Letter No. 83-10c

Dear Sir:

We have enclosed a detailed description of our plan for resolution of TMI Action Item II.K.3.5 in response to Generic Letter 83-10c [Reference (b)].

As is evident in the enclosed response, we are participants in the Westinghouse Owners Group effort to resolve this item and therefore, our schedules for plant specific submittals are tied to schedules developed by the Owners Group.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

J. DeVincentis

J. DeVincentis
Project Manager

ALL/fsf

cc: Louie Wheeler, Project Manager
Division of Licensing

Atomic Safety and Licensing Board Service List

PLAN FOR RESOLUTION OF TMI ACTION ITEM II.K.3.5

"AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"

INTRODUCTION

The criteria for resolution of TMI Action Plan Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" were stated in letters from Mr. Darrell G. Eisenhut of the Nuclear Regulatory Commission to all Applicants and Licensees with Westinghouse designed Nuclear Steam Supply Systems (83-10 c and d), dated February 8, 1983. The following represents the plan for demonstrating compliance with those criteria. In order to avoid confusion, the overall philosophy and plan will first be stated. Then, each section of the attachment to NRC Letters 83-10 c and d will be addressed as to how the overall plan responds to each NRC criteria.

OVERALL PLAN

In the four years that have passed since the event at Three Mile Island, Westinghouse and the Westinghouse Owners Group have held steadfastly to several positions relative to post-accident Reactor Coolant Pump (RCP) operation. First, there are small break LOCAs for which delayed RCP trip can result in higher fuel cladding temperatures and a greater extent of zircalloy-water reaction. Using the conservative evaluation model, analyses for these LOCAs result in a violation of the Emergency Core Cooling System (ECCS) Acceptance Criteria as stated in 10CFR50.46. The currently approved Westinghouse Evaluation Model for small break LOCAs was used to perform these analyses and found acceptable for use by the NRC in Letters 83-10 c and d. Therefore, to be consistent with the conservative analyses performed, the RCPs should be tripped if indications of a small break LOCA exist.

Secondly, Westinghouse and the Westinghouse Owners Group have always felt that the RCPs should remain operational for non-LOCA transients and accidents where their operation is beneficial to accident mitigation and recovery. This position was taken even though a design basis for the plant is a loss of off-site power. Plant safety is demonstrated in the Final Safety Analysis Reports for all plants for all transients and accidents using the most conservative assumption for Reactor Coolant Pump operation.

In keeping with these two positions, a low RCS pressure (symptom based) RCP trip criterion was developed that provided an indication to the operator to trip the RCPs for small break LOCA but would not indicate a need to trip the RCP for the more likely non-LOCA transients and accidents where continued RCP operation is desirable. The basis for this criterion is included in the generic Emergency Response Guideline (ERG) Background Document (E-0 Basic Revision, Appendix A). Relevant information regarding the expected results of using the RCP trip criterion can be derived from the transients which resulted from the stuck open steam dump valve at North Anna in 1979, the steam generator tube rupture at Prairie Island in 1980, and the steam generator tube rupture at Ginna in 1982. The RCPs were tripped in all three cases. However, a study of the North Anna and Prairie Island transients indicated that RCP trip would not have been needed based on the application of the ERG trip criterion. The Ginna event, however, indicated a need to review the basis for the RCP trip criterion to allow continued RCP operation for a steam generator tube rupture for low head SI plants.

Thirdly, it has always been the position of Westinghouse and the Westinghouse Owners Group that if there is doubt as to what type of transient or accident is in progress, the RCPs should be tripped. Again, the plants are designed to mitigate the effects of all transients and accidents, even without RCP operation while maintaining a large margin of safety to the public. The existing emergency operating procedures reflect this design approach.

Lastly, it remains the position of Westinghouse and the Westinghouse Owners Group that RCP trip can be achieved safely and reliably by the operator when required. An adequate amount of time exists for operator action for the small break LOCAs of interest. The operators have been trained on the need for RCP trip and the emergency operating procedures give clear instructions on this matter. In fact, one of the initial operator activities is to check if indications exist that warrant RCP trip.

Westinghouse and the Westinghouse Owners Group will undertake a two part program to address the requirements of NRC Letters 83-10 c and d based on the aforementioned positions for the purpose of providing more uniform RCP trip criteria and methods of determining those criteria. In the first part of the program, revised RCP trip criteria will be developed which provides an indication to the operator to trip the RCPs for small break LOCAs requiring such action, but will allow continued RCP operation for steam generator tube ruptures, less than or equal to a double-ended tube rupture. The revised RCP trip criteria will also be evaluated against other non-LOCA transients and accidents where continued RCP operation is desirable in order to demonstrate that a need to trip the RCPs will not be indicated to the operator for the more likely cases. Since this study is to be utilized for emergency response guideline development, better estimate assumptions will be applied in the consideration of the more likely scenarios. The first part of the program will be completed and incorporated into Revision 1 of the Emergency Response Guidelines developed by Westinghouse for the Westinghouse Owners Group. The scheduled date for completion of Revision 1 is July 31, 1983.

The second part of the program is intended to provide the required justification for manual RCP trip. This part of the program must necessarily be done after the completion of the first part of the program. The schedule for completion of the second part of the program is the end of 1983.

The preferred and safest method of pump operation following a small break LOCA is to manually trip the RCPs before significant system voiding occurs.

No attempt will be made in this program to demonstrate the acceptability of continued RCP operation during a small break LOCA. Further, no request for an exemption to 10CFR50.46 will be made to allow continued RCP operation during a small break LOCA.

DETAILED RESPONSE TO NRC LETTERS 83-10 C AND D

Each of the requirements stated in the attachment to NRC Letters 83-10 c and d will now be discussed indicating clearly how they will be addressed. The organization of this section of the report parallels the attachment to NRC Letters 83-10 c and d.

I. Pump Operation Criteria Which Can Result in RCP Trip During Transients and Accidents.

1. Setpoints for RCP Trip

The Westinghouse Owners Group response to this section of requirements will be contained in Revision 1 to the Emergency Response Guidelines scheduled for July 31, 1983. Seabrook Station will utilize Revision 1 to the Emergency Response Guidelines in the establishment of setpoints for RCP trip. Completion of plant specific procedures for Seabrook Station based on the Westinghouse Owners Group Emergency Response Guidelines (including Rev. 1) is scheduled for December 1983.

a. As stated above, Westinghouse and the Westinghouse Owners Group are developing revised RCP trip criteria which will assure that the need to trip the RCPs will be indicated to the operator for LOCAs where RCP trip is considered necessary. The criteria will also ensure continued forced RCS flow for:

1. Steam generator tube rupture (up to the design basis, double-ended tube rupture).

2. The other more likely non-LOCA transients where forced circulation is desirable (e.g., steam line breaks equal to or smaller than 1 stuck open PORV).

NOTE: Event diagnosis will not be used. The criteria developed will be symptom based.

The criteria being considered for RCP trip are:

1. RCS wide range pressure $<$ constant

2. RCS subcooling $<$ constant

3. Wide range RCS pressure $<$ function of secondary pressure

Instrument uncertainties will be accounted for. Environmental uncertainty will be included if appropriate.

No partial or staggered RCP trip schemes will be considered. Such schemes are unnecessary and increase the requirements for training, procedures, and decision making by the operator during transients and accidents.

b. The RCP trip criteria selected will be such that the operator will be instructed to trip the RCPs before voiding occurs at the RCP.

c. The criteria developed in Item 1a above is not expected to lead to RCP trip for the more likely non-LOCA and SGTR transients. However, since continued RCP operation cannot be guaranteed, the emergency response guidelines provide guidance for the use of alternate methods for depressurization.

d. The Emergency Response Guidelines contain specific guidance for detecting, managing, and removing coolant voids that result from flashing. The symptoms of such a situation are described in these guidelines and in detail in the background document for the guidelines. Additionally, explicit guidance for operating the plant with a vaporous void in the reactor vessel head is provided in certain cases where such operation is needed. Seabrook Station will utilize the Emergency Response Guidelines to develop procedures for the detection, management, and removal of Reactor Coolant System voids. Training in the use of these procedures will be provided.

e. A containment isolation signal (4.3 psig) will result in the isolation of the RCP seal water return line; however, continued operation of the RCPs is allowable because Primary Component Cooling Water will continue to be provided to the thermal barrier heat exchangers, lube oil coolers and motor coolers until the containment spray signal (18 psig) or a low PCCW surge tank level is reached. At this point in the transient, the RCPs would be tripped (if they had not been already). It should be noted that the containment spray signal does not isolate RCP seal water injection.

f. Discussed in 1a and 1c.

2. <u>Guidance for Justification of Manual RCP Trip</u>

The Westinghouse Owners Group response to this section of requirements will be reported separately at the end of 1983. PSNH will review the Westinghouse Owners Group guidance for justification of manual RCP trip and will provide a plant specific justification for manual RCP trip within three months of receipt of the Westinghouse report.

a. A significant number of analyses have been performed by Westinghouse for the Westinghouse Owners Group using the currently approved Westinghouse Appendix K Evaluation Model for small break LOCA. This Evaluation Model uses the WFLASH Code. These analyses demonstrate for small break LOCAs of concern, if the RCPs are tripped 2 minutes following the onset of reactor conditions corresponding to the RCP trip setpoint, the predicted transient is nearly identical to those presented in the Safety Analysis Reports for all Westinghouse plants. Thus, the Safety Analysis Reports for all plants demonstrate compliance with requirement 2a. The analyses performed for the Westinghouse Owners Group will be used to demonstrate the validity of this approach.

b. Better estimate analyses will be performed for a limiting Westinghouse designed plant using the WFLASH computer code with better estimate assumptions. These analyses will be used to determine the minimum time available for operator action for a range of break sizes such that the ECCS acceptance criteria of 10CFR50.46 are not exceeded. It is expected that the minimum time available for manual RCP trip will exceed the guidance contained in N660. This will justify manual RCP trip for all plants.

3. Other Considerations

a. Information regarding the quality of instrumentation which will be employed to monitor RCP trip setpoint parameters will be provided to the NRC within three months of the receipt of the Westinghouse report.

b. Seabrook Station will utilize the Emergency Response Guidelines to develop procedures for the timely restart of the reactor coolant pumps when conditions which will support safe pump startup and operations are established. Training in the use of these procedures will be provided on the Seabrook site specific simulator.

c. Seabrook Station operators will be knowledgeable/trained in their responsibility for tripping the RCPs when the trip setpoints are reached. The priority of this action and all actions following engineered safety features actuation are also considered.

II. Pump Operation Criteria Which Will Not Result in RCP Trip During Transient and Accidents.

The preferred and safest method of operation following a small break LOCA is to manually trip the RCPs. Therefore, there is no need to address the criteria contained in this section.

overload devices. In addition to the 15 kV switchgear breakers, the medium voltage 15 kV penetrations are also protected by fuses inserted in the feeders outside containment. These fuses are qualified by experience and seismic testing. The 600 volt system x/R ratio used in specifying the electrical penetrations is 4. Calculations show that this value is conservatively applied because the actual ratio is considerably less than 4. Refer to Subsection 8.3.1.2

RG 1.75 (Rev 2) "Physical Independence of Electric Systems"

The design is consistent with the criteria for physical independence of electric systems established in Attachment "C" of AEC letter dated December 14, 1973, and is in general conformance with Regulatory Guide 1.75, except as follows:

<u>Battery Room Ventilation</u>. Although the four Class 1E batteries are housed in separate safety class structures, they represent only two redundant load groups (see Subsection 8.3.2). Each load group is served by a separate safety-related ventilation system. There is a cross-tie between the two ventilation systems to allow one system to serve both load groups in case the other system is inoperable. Fire dampers are provided to isolate each battery room. 47 45

For additional information on the four batteries and two redundant load groups, see Subsection 8.3.2.1.a.

Refer to Subsection 8.3.1.2.b.5 for a discussion of the onsite ac power system. 51

See Insert I

RG 1.108 (Rev 1) "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants"

The diesel generator testing is in conformance with the recommendations of Regulatory Guide 1.108 with one clarification:

The requirements of position C.2.a(5) will be met every 18 months as follows:

The functional capability at full load temperature will be demonstrated at least every 18 months by performing the test outlined in position C.2.c(1) and (2) immediately following the full load carrying capability test described in position C.2.a(3). The full load carrying

INSERT I

The requirements of position C4, as it relates to cables for the associated circuits, is clarified as follows:

> Instrumentation, control and power cables used for the associated circuits will not be covered by the Operational Quality Assurance Program (OQAP). However, programmatic controls will be applied to these items. The actual implementation of these controls will be defined by the program manuals used to control specific activities at Seabrook Station. Implementation of these programmatic controls will be verified by Quality Assurance personnel to the extent necessary to insure proper application. For further details on provisions and considerations for the associated circuits, see FSAR Chapter 8, Section 8.3.1.4.bld.

4. <u>Regulatory Guide 1.63 - Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants</u> 48

The electric penetration assemblies are designed to withstand, without loss of mechanical integrity, the maximum fault current vs. time conditions that could occur as a result of single random failures of circuit overload devices. The 600 volt system X/R ratio used in specifying the electrical penetrations is 4. Calculations show that this value is conservatively applied because the actual ratio is considerably less than 4.

To preclude damage to electric penetrations due to single failures of circuit overload protection devices, each penetration circuit, with the exception of instrumentation and low energy circuits, is provided with dual Class 1E overload protective devices. For more details refer to Subsection 8.3.1.1.c. 15 kV penetrations are protected by seismically qualified Class 1E fuses. Additional protection is provided by two non-Class 1E breakers in series. These breakers are coordinated and derive their control power from different batteries. For more details refer to Subsection 8.3.1.1.a.

5. <u>Regulatory Guide 1.75 - Physical Independence of Electric Systems</u>

The design is consistent with the criteria for physical independence of electric systems established in Attachment "C" 52
of AEC (NRC) letter dated December 14, 1973. Attachment "C" which is incorporated as Appendix 8A, is in general conformance with Regulatory Guide 1.75.

Physical separation and identification of circuits are described in detail in Subsections 8.3.1.3 and 8.3.1.4, 46 47
respectively.

c. <u>Compliance to Branch Technical Position PSB-1 - Adequacy of Station Electric Distribution System Voltages</u>

1. <u>Position B1</u>

An acceptable alternative to the second level undervoltage protection system described in Position 1 is provided. This alternative system is described in Subsection 8.3.1.1.b.4.(b).

2. <u>Position B2</u>

The Seabrook Station design meets Position 2 of Branch Technical Position PSB-1. The bypass of the load shedding feature during sequencing, and its restoration in the event of a subsequent diesel generator breaker trip, is discussed in

For clarification of position C4 as it relates to associated circuits, refer to FSAR Section 8.1.5.3b. 52

(c) All Non-Class 1E protective circuit breakers will be periodically inspected approximately once every five years according to a program developed for the inspection of Non-Class 1E equipment. This program will be in accordance with manufacturer's recommendations for maintenance and inspections.

Since Class 1E and Non-Class 1E protective devices are identical, any generic degradation such as setpoint drift, manufacturing deficiencies, and material defects will be detected and corrected as a result of the rigorous program performed on the Class 1E protective devices to satisfy the requirements of ANSI N-18.7-1976 and Regulatory Guide 1.63; therefore, credit can be taken for this equipment to function under DBE conditions.

(d) The probability of an ensuing fire is minimized because all cables utilized for these associated circuits are specified, designed, manufactured, and installed to the same criteria as Class 1E cables. Factors that have been taken into consideration include flame retardancy, non-propagating and self-extinguishing properties, splicing restrictions, appropriate limitations on raceway fill, cable pulling and termination requirements, appropriate cable derating, and environmental qualifications.

(e) Degradation of an associated circuit because of a raceway failure during a DBE, has been eliminated because all electrical raceway systems within the Nuclear Island are seismically analyzed.

(f) Other design considerations that contribute to the integrity of these associated circuits are:

1) Cables associated with one train are never routed in raceways containing Class 1E or associated cable of another train or channel.

2) All cables for instrumentation circuits utilize shielded construction which minimizes any unacceptable interaction between Class 1E and associated circuits.

3) All circuits entering the reactor containment are provided with protective devices complying with Regulatory Guide 1.63. For exceptions see Subsection 8.3.1.1.C.7(a). 53

Based on the above design features and analysis, we do not consider these associated circuits to pose any challenges to any Class 1E circuits. Therefore, the ability for safe plant shutdown under DBE conditions has not been jeopardized.

52

The above provisions and considerations used for the associated circuits during the construction phase of the plant will also be used during the operations phase.

to normal operation as practical, the full operational sequence that brings the system into operation, including portions of the protection system, is tested.

b. Compliance with Regulatory Guides

1. Regulatory Guide 1.6 - Independence Between Redundant Standby Power Sources and Between their Distribution Systems 52

The safety-related portion of the station dc system for each unit includes four batteries. The redundant safety-related load groups are each fed by a separate battery and battery charger. There is no provision for automatically connecting one battery-charger combination to any other redundant load group, nor is there any provision for interconnecting batteries either manually or automatically. To further enhance safety and reliability, two dc supply buses of the same train may be connected together manually, but circuit breaker interlocks prevent an operator error which would parallel two batteries. (See Figure 8.3-37).

2. Regulatory Guide 1.32 - Criteria for Safety Related Electric Power Systems for Nuclear Power Plants 52

The design is consistent with the requirements of this regulatory guide. For details, refer to Subsections 8.3.2.1.c and 8.3.2.1.e.

3. Regulatory Guide 1.75 - Physical Independence of Electric Systems 52

The design is consistent with the criteria for physical independence of electric systems established in Attachment "C" of AEC letter dated December 14, 1973. Attachment "C" is incorporated as FSAR Appendix 8A and is considered similar to Regulatory Guide 1.75.

For clarification of position C4 as it relates to associated circuits, refer to FSAR Section 8.1.5.3 b.

4. Regulatory Guide 1.129 - Maintenance, Testing and Replacement of Large Lead Acid Storage Batteries for Nuclear Power Plants 52

For compliance to this regulatory guide, refer to Subsection 8.3.2.1.e.

c. Compliance with IEEE-308, Class 1E Electric Systems

The station dc system conforms to the requirements of IEEE-308. The power supplies, distribution system, and load groups (see Subsection 8.3.2.1) are arranged to provide direct current electric