



Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Nuclear Department

April 22, 1983

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20014

Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch 1
Division of Licensing

Dear Mr. Varga:

RESOLUTION OF TMI ACTION ITEM II.K.3.5
AUTOMATIC TRIP OF REACTOR COOLANT PUMPS
SALEM GENERATING STATION
UNITS NO. 1 AND 2
DOCKET NOS. 50-272 AND 50-311

PSE&G hereby transmits its plan for resolution of the TMI
Action Item II.K.3.5, as requested by Generic Letter 83-10d.

This plan was developed by the Westinghouse Owner's Group.
Some Salem-specific information and implementation commitment
dates were added to the Owner's Group generic resolution, but
no changes have been made from the generic resolution.

Briefly, the first portion of the plan is to implement
Revision 1 to the Emergency Response Guidelines (ERG) by June
30, 1984. This revision is scheduled to be completed by
Westinghouse on July 31, 1983. It will include specific
guidelines for the operator to manually trip the Reactor
Coolant Pumps (RCP) on indication of a small break Loss of
Coolant Accident (LOCA). The specific RCP trip criteria will
use existing Reactor Coolant System instru-
mentation. No physical plant changes are involved.

The second portion of the plan is to provide detailed
justification that a manual RCP trip is acceptable. This
involves our review and endorsement of analyses being
performed by Westinghouse for the Owner's Group. The new
analyses demonstrate that the consequences of a small break
LOCA with manual RCP trip (as required by Revision 1 to the
ERG) are nearly identical to the existing FSAR analyses which
assume a loss of RCP at the start of the accident. The
Owner's Group analyses are scheduled for completion by the end

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The Energy People

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Mr. Steven A. Varga, Chief
U.S. Nuclear Regulatory Commission

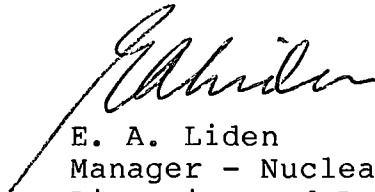
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of 1983. The commitment for PSE&G's review and endorsement of the new analyses is June 30, 1984, the same date as for implementation of Revision 1 to the ERG.

Should you have any questions in this regard, we will be pleased to discuss them with you.

Very truly yours,



E. A. Liden
Manager - Nuclear
Licensing and Regulation

Enclosure

CC: Mr. Donald C. Fischer
Licensing Project Manager

Mr. Leif Norrholm
Senior Resident Inspector

STATE OF NEW JERSEY)
) SS. COUNTY OF SALEM
COUNTY OF SALEM

RICHARD A. UDERITZ, being duly sworn according to law deposes
and says:

I am a Vice President of Public Service Electric and Gas
Company, and as such, I find the matters set forth in our
response dated April 22, 1983, to the NRC's Generic Letter
83-10d, concerning Resolution of TMI Item II.K.3.5, Automatic
Trip of Reactor Coolant Pumps, are true to the best of my
knowledge, information and belief.


RICHARD A. UDERITZ

Subscribed and sworn to before me
this 24TH day of APRIL, 1983


Notary Public of New Jersey

RUDOLPH L. von FISCHER JR.
Notary Public of New Jersey
My Commission Expires Sept. 10, 1986

My Commission expires on _____

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. Darrel G. Eisenhut of the Nuclear Regulatory Commission to all Applicants and Licensees with Westinghouse designed Nuclear Steam Supply Systems (83-10 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 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 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-O Basic Revision, Appendix A). Relevant information regarding the expected results of using this 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 (SGTR) at Prairie Island in 1980 and the SGTR 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 with a SGTR 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 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 SGTRs 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 ERGs developed by Westinghouse for the Westinghouse Owners Group. The Westinghouse scheduled date for completion of Revision 1 is July 31, 1983. Salem will implement Revision 1 to the ERGs to its plant specific procedures by June 30, 1984.

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 Westinghouse schedule for completion of the second part of the program is the end of 1983. Salem will review and endorse this study by June 30, 1984.

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 D

Each of the requirements stated in the attachment to NRC letters 83-10 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 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 ERGs scheduled for July 31, 1983. Salem will implement Revision 1 to the ERGs to its plant specific procedures by June 30, 1984. If Revision 1 to the ERGs contain significant deviations from the existing RCP trip criteria, the criteria will be evaluated for immediate inclusion in the Salem Emergency Instructions.

- 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) SGTR up to and including the design bases, 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 ERGs 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.

The Salem procedures for natural circulation and natural circulation cooldown specifically address voiding recognition and treatment. The LOCA Emergency Instructions do not specifically address voids since it is assumed that the actions provided by the procedure will provide for adequate core cooling without treatment of the voids. A void in the head is not detrimental to core decay heat removal provided safety injection flow is maintained. In addition, the LOCA procedures adequately address inadequate core cooling (ICC) mitigation should an ICC condition develop.

- e) Isolation of the RCP auxiliary water services are based on containment isolation requirements. The Component Cooling Water (CCW) isolation valves close automatically on containment hi-hi pressure and the seal injection line isolation check valves close on reverse flow. Accordingly, CCW and seal injection isolation are not anticipated during SGTR's and other non-LOCA transients for which RCP operation may be desirable. The No. 1 seal leakoff line isolates on Phase A containment isolation which occurs concurrently with safety injection actuation. Although phase A isolation could occur during SGTR and other non-LOCA transients, seal bleedoff flow would continue via the relief valve on the seal bleedoff line. The relief valve is set at 150 psig, and it discharges to the Pressurizer Relief Tank which is located inside the containment.
- f) Discussed in 1a and 1c.

2. Guidance for Justification of Manual RCP Trip

The Westinghouse Owners Group response to this section of requirements will be reported separately at the end of 1983. Salem will review this report to ensure its applicability to the Salem units. The Salem endorsement of the report is scheduled to be completed prior to Salem implementation of Revision 1 to the ERG.

- 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 estimates 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) The specific instrumentation employed for the RCP trip setpoint will, of course, depend on the criteria selected for RCP trip. The Reactor Coolant System (RCS) instrumentation being considered is listed in Table 1. All of the instrumentation listed in Table 1 is presently Class 1E except for the RCS Wide Range Temperatures. The RCS Wide Range Temperature instrumentation is scheduled to be upgraded to Class 1E at the next refueling outage.
- b) The ERGs contain guidance for the timely restart of the RCPs when conditions which will support safe pump start-up and operation are established.

The Emergency Operating Procedures (EOPs) being developed at Salem in response to NRC Generic Letter 82-33 will utilize the appropriate RCP restart considerations and criteria contained in

the ERGs. Any changes to that criteria in Revision 1 to the ERGs will be evaluated and incorporated as appropriate.

- c) The Salem Emergency Instructions (EIs), which are in place, contain the RCP trip criteria developed in accordance with the ERGs. All operators have been trained on the EI's and on the basis behind the RCP trip criteria. In addition, as part of the requirements of Generic Letter 82-33, all operators will receive in-depth training on the new EOPs before they are implemented. All changes to the RCP trip criteria in Revision 1 to the ERGs will be evaluated and incorporated as appropriate.

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.

TABLE I

<u>Function</u>	<u>Range</u>	<u>Number of Channels</u>
RCS Wide Range Pressure	0-3,000 psig	one
Pressurizer Pressure	1,700-2,500 psig	Three
Steam Generator Pressure	0-1,200 psig	Three per steam generator
Pressurizer Vapor Temperature	100-700° F	one
Pressurizer Liquid Temperature	100-700° F	one
RCS Wide Range Temperature		
Hot Leg	0-700° F	one per loop
Cold Leg	0-700° F	one per loop
RCS Narrow Range Temperature		
Hot Leg	530-650° F	one per loop
Cold Leg	510-630° F	one per loop
Average	530-630° F	one per loop