Central file

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CHAIRMAN

June 2, 1983

The Honorable Joseph R. Biden, Jr.
United States Senate
Washington, D.C. 20510

Dear Senator Biden:

Your March 4, 1983 letter requested information and specific responses to questions concerning the events which occurred at the Salem Nuclear Generating Station, Unit 1, on February 22 and 25, 1983. The responses to your questions are enclosed.

My fellow Commissioners and I are also concerned that these events have occurred. We have closely monitored the staff's followup of the malfunctions at the Salem plant as well as the broader implications for the nuclear power industry. The facts, data and circumstances associated with these events have been collected and documented as NUREG-0977. This information was used by the staff to determine the safety issues associated with the events. These issues were grouped into three areas: (1) equipment issues; (2) operating procedures and operator training and response; and (3) management issues. The staff evaluated each of the areas to determine the licensee's actions necessary to resolve the issues. The staff concluded, as reported in their safety evaluation NUREG-0995, that the underlying causes of the problems were identified and resolved and, as such, the Salem facility could be allowed to restart. We concurred with these findings. Concurrently, an NRC task force with representatives from three NRC offices was established to review and evaluate the generic implications.

The events can be characterized as failures of the safety system to automatically shut down the reactor. However, the operators did identify the need for plant shutdown and did manually shut down the reactor on both occasions such that the events themselves posed no serious threat to public health and safety. However, we view the failures as serious safety concerns since the automatic systems did not function as expected and if other plant conditions had existed, such as full power, considerable overpressure of the reactor system would have occurred without prompt operator action. The licensee has attributed the cause of the failures to a lack of adequate maintenance to a part of the safety system, specifically, the circuit breakers which de-energize the control rods to cause rod insertions and reactor shutdown. Additional means to trip the Salem reactors (albeit not as rapidly) are discussed in response to your Question 6.

Regarding calculation of probabilities that you mention in you letter, the industry over the past several years has provided the staff various estimates of the probability of failure to trip the reactor. The staff has recognized the substantial uncertainties in these calculations and, because of the probabilities calculated, the staff has continued its efforts to resolve the Anticipated Transient Without Scram (ATWS) issue. As indicated in the answer to your Question 11, the new proposed ATWS rule is currently being evaluated in light of the Salem events, and this reevaluation will be forthcoming.

In summary, prior to our decision to allow restart of the Salem facility, the Commission conducted a careful examination of the events and the circumstances associated with them. Based on this examination, we are satisfied that the safety implications of the short- and long-term actions have been resolved by specific commitments from the licensee. In addition, our review of the circumstances leading up to the events of February 22 and 25 led us to conclude that violations of the Salem operating license contributed to the failures that occurred. As a result, we have proposed to impose a civil penalty of $850,000 on the licensee. This is discussed in more detail in the answer to your Question 3.

Commissioner Gilinsky adds: I do not share my colleagues' confidence that the underlying causes of the problems at the Salem plants have been identified and resolved. I must add that the documents referred to by the Commission were not a sound basis for decision on Salem restart in that they left out some of the most important safety violations reflecting management deficiencies -- as can be seen by comparing these documents with those accompanying the Commission's later enforcement action, which picked up the omitted items.

Sincerely,

Original Signed By
John F. Ahearne

for Nunzio J. Palladino

Enclosure:
Responses to Questions

Cmr. Ahearne would have preferred the following:

1) In A to Q12 -- last sentence to have read, "This problem apparently occurred because there were no procedures requiring anyone to examine, evaluate or interpret the timing of events and there was inadequate or lack of, training in the use and understanding of the SOE printout."

2) In A to Q14 -- first paragraph to have additional sentence, "Additionally, maintenance conducted on the reactor trip breakers in January 1983 was conducted with guidance from a Westinghouse (Apparatus Services Division) representative who was also unaware of the existance of the bulletins"

3) In A to Q14 -- second paragraph change word "requirements" to controls"

Cleared with all Cmrs.' Offices by SECY C/R. Ref.-CR-83-74

Originating Office: EDO/NRR

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SURNAME	TCombs						
DATE	5/24/83 6/2/83	6/2/83					

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The Honorable Joseph R. Biden, Jr.
United States Senate
Washington, D. C.

Dear Senator Biden:

I am pleased to respond to your March 4, 1983 letter about the events that occurred at the Salem Nuclear Generating Station, Unit 1 on February 22nd and 25th of this year. Your letter requested information and specific response to several questions concerning the events; detailed responses are provided in the enclosure to this letter.

The other Commissioners and I were also concerned that these events occurred. We closely monitored the staff's followup of the malfunctions at the Salem plant as well as the broader implications for the nuclear power industry. The facts, data and circumstances associated with the events at Salem have been collected and documented as NUREG-0977. This information was used by the staff to determine the safety issues associated with the events. These issues were grouped into three areas: (1) equipment issues, (2) operating procedures and operator training and response, and (3) management issues. The staff evaluated each of the issues to determine the licensee's actions necessary to resolve the issues. The staff concluded, as reported in their safety evaluation NUREG-0995, that the underlying causes of the problem(s) were identified and resolved and, as such, the Salem facility could be allowed to restart. We concurred with these findings. Concurrently, an NRC task force with representatives from three NRC offices was established to review and evaluate the generic implications.

The event can be characterized as a failure of the safety system to automatically shut the reactor down. However, the operators did identify the need for plant shutdown and did manually shut the reactor down such that the event itself posed no serious threat to public health and safety. However, we view the failure as a serious safety concern since the automatic system did not function as expected and, given other plant conditions such as full power, considerable overpressure of the reactor system would have occurred without prompt operator action. The licensee has attributed the cause of failure to be a lack of adequate maintenance to a part of the safety system, specifically, the circuit breakers which de-energize the control rods to cause rod insertions and reactor shutdown. Additional means to trip the Salem reactors (albeit not as rapidly) are given in response to your Question 6.

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of the probability of failure to trip the reactor. The staff has recognized the substantial uncertainties in these calculations and, because of the probabilities calculated, the staff has continued its efforts to resolve the ATWS issue. As indicated in the answer to your Question 11, the new proposed ATWS rule is currently being evaluated in light of the Salem events, and this reevaluation is forthcoming.

In summary, we shared your concerns and consequently, prior to our decision to allow restart of the Salem facility, conducted a careful examination of the events and the circumstances associated with them and assured ourselves that the safety implications of the short and long term actions were resolved by specific commitments from the licensee.

Sincerely,

Nunzio J. Palladino
Chairman

Enclosure: Responses to Questions

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events occurred and is an example of the defense-in-depth concept utilized in the design and operation of nuclear power plants.

In your letter, you mention that this incident demonstrates the extent to which reliance is placed on fallible "human" factors. In my view, there must always be some reliance on the human factor since no design of this complexity can account for all possible occurrences. That is why nuclear plants are built using defense-in-depth concepts and why we place so much emphasis on training, procedures, and management involvement and oversight.

Regarding calculations of probabilities that you mention in your letter, the industry has provided the staff over the past several years various estimates of the probability of failure to trip the reactor. The staff has recognized the substantial uncertainties in these calculations and, in spite of the low probabilities calculated, the staff has continued its efforts to resolve the ATWS issue. As indicated in the answer to your Question 11, the new proposed ATWS rule is currently being evaluated in light of the Salem events, and this reevaluation is expected to be completed by about mid-April.

In summary, we share your concerns and are conducting a careful examination of the events and the circumstances associated with them. We are proceeding in a structured manner and intend to address your concerns, as well as others, prior to any restart decision.

Sincerely,

Nunzio J. Palladino, Chairman
U. S. Nuclear Regulatory Commission

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CHAIRMAN

The Honorable Joseph R. Biden, Jr.
United States Senate
Washington, D. C.

Dear Senator Biden:

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The other Commissioners and I are also concerned that these events occurred. We are closely monitoring the staff's followup of the malfunctions at the Salem plant as well as the broader implications for the nuclear power industry. The facts, data and circumstances associated with the events at Salem have been collected and documented as NUREG-0977. This information is being used by the staff in evaluating the licensee's actions and assessing when a restart decision for the Salem facility is warranted. Concurrently, an NRC task force with representatives from three NRC offices has been established to review and evaluate the generic implications.

A plan of action (Salem Restart Status Report) has been prepared which identifies the issues involved with the Salem events specifically, along with short and long term actions required of the utility to resolve those issues. Before recommending restart, the staff intends to obtain specific commitments from the licensee to complete the short term actions to the staff's satisfaction. The Commissioners will make the decision on the restart of the Salem facility when we are satisfied that the underlying causes of the problem(s) have been identified and resolved.

The event can be characterized as a failure of the safety system to automatically shut the reactor down. However, the operators did identify the need for plant shutdown and did manually shut the reactor down such that the event itself posed no serious threat to public health and safety. However, we view the failure as a serious safety concern since the automatic system did not function as expected and, given other plant conditions such as full power, considerable overpressure of the reactor system would have occurred without prompt operator action. The licensee has attributed the cause of failure to be a lack of adequate maintenance to a part of the safety system, specifically, the circuit breakers which de-energize the control rods to cause rod insertions and reactor shutdown. Additional means to trip the Salem reactors (albeit not as rapidly) are given in response to your Question 6.

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

FOR: The Commissioners

FROM: William J. Dircks, Executive Director for Operations

SUBJECT: RESPONSES TO SENATOR BIDEN'S QUESTIONS RELATED TO SALEM 1 EVENTS

PURPOSE: For Chairman's signature.

DISCUSSION: This letter provides a response to Senator Biden's letter dated March 4, 1983. The Senator's letter contained fifteen questions which are answered in the enclosure to the letter to be signed by the Chairman.

RECOMMENDATION: I recommend that the Chairman sign the letter.

COORDINATION: NRR, RES, Region I

SCHEDULING: Prior to Salem 1 restart.

William J. Dircks
Executive Director for Operations

Enclosure:
Letter to Senator Biden

Contact: D. Wigginton, Ext. 27354

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3/25/83

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

Responses To U. S. Senator Joseph R. Biden Jr. Questions (March 4, 1983 Letter to Chairman Palladino)

Question 1

Please provide me with a complete description of the event and its safety significance.

Response

Attachment 1 is a draft Abnormal Occurrence report which provides details of the reactor trip breaker failure events at Salem. NUREG-0977, a task force report on the facts associated with the circumstances of the events, has been issued and is Attachment 2.

With respect to the safety significance, the Salem ATWS events of February 22 and 25, 1983 posed no serious threat to public health and safety because the Salem reactor was at low power and the operators manually scrammed the reactor soon after the automatic scram signal. The event of February 22nd was a loss of one operating feedwater pump at low power. The event of the 25th was normal operation at 12 percent power with low level in one steam generator. A discussion of the safety significance of the events had they occurred at full power is given in the response to Question 8.

[redacted]

Please provide me with copies of all memoranda and internal studies that analyze this event.

Response

There are two efforts underway with respect to analysis of the Salem events. I am enclosing as Attachment 3, the Salem Restart Evaluation Report prepared by the NRC staff. This report documents the bases for a restart decision. Additionally, an NRC Task Force was established to review and evaluate the generic implications of the Salem events. A report will be forthcoming in the near future. Attachment 4 is a listing of additional internal documents which may relate to the events in question. We will provide you with copies as you request.

Question 3

Before restart, we would ask that you provide an assessment of whether NRC has reason to believe that either an operator during the incident or management action during or prior to the event acted inappropriately.

Response

An NRC fact-finding task force was at the Salem site on March 2-6, 1983, and they conducted a review of the circumstances surrounding the February 22 and 25 events. The results of this review were published as NUREG-0977, dated March 1983. This and other NRC and PSE&G efforts revealed significant deficiencies which contributed to the inoperability of the reactor trip breakers. These deficiencies involved (1) failure to adequately investigate previous failures to identify and correct conditions adverse to quality; (2) failure to correctly include the breakers on the Master Equipment List (MEL); (3) failure to properly implement procurement procedures; (4) failure to properly implement, control and distribute the MEL which contributed to inadequate quality assurance review of procurement and maintenance; (5) failure to identify and control safety related components; and (6) failure to implement surveillance testing requirements. PSE&G efforts to correct these deficiences are addressed in the Salem Restart Safety Evaluation Report. In addition, the licensee failed to promptly report, as required, certain events to the NRC.

On February 22, the reactor trip breakers failed to open automatically upon demand, apparently because of the deficiencies described in Item II of the enclosed Notice of Violation. The licensee failed to recognize, prior to restart of the reactor on February 23, that the reactor trip breakers had failed to open automatically on February 22. As a result, the reactor was operated for three additional days during which time the reactor protection system could not be considered operable.

The Commission has concluded that these contributors to the events of February 22 and 25 are the result of insufficient management involvement in establishing a safety perspective, in requiring attention to detail, and in ensuring procedural adherence. Furthermore, the Commission has determined that these contributors to the events of February 22 and 25 are as significant as the events themselves. Accordingly, the NRC has proposed imposition of civil penalties in the amount of $850,000.

With respect to operator actions, the NRC staff review has determined that the operators responses to both events were satisfactory however, the post-trip review was inadequate.

Question 4

In this event, manual shut-down was achieved some twenty-four seconds after automatic controls and back-up failed; are there incidents of this kind where 30 seconds would not have provided for adequate public health and safety?

Response

Manual shutdown of the reactor in 30 seconds following any anticipated transient will provide adequate protection for public health and safety. A more extensive discussion of the limiting anticipated transient with delayed reactor trip or failure of reactor trip is given in response to Question 8 below.

It should be noted that backup to automatic controls is a manual shutdown. Since the plant was manually shut down, the backup did not fail.

Question 5

I am informed that this type of event was calculated in the Reactor Safety Study (WASH 1400) as having an extermely low probability. What does the NRC currently calculate the probability of this type of event? How was it that the probability of occurrence was repeated in a three day time period at Salem I?

Response

In WASH-1400 a pressurized water reactor designed by Westinghouse was analyzed. The median probability of SCRAM failure was estimated at 3.6×10^{-5} per demand with an upper bound of 1×10^{-4}. The frequency of ATWS is the product of the frequency of anticipated transients requiring scram and the probability of scram failure on demand. With approximately 10 scram demands per year the median estimated ATWS frequency would be 3.6×10^{-4}/year with an upper bound of approximately 1×10^{-3}. However, only a fraction of ATWS events would result in reactor core damage. For Westinghouse designed reactors, most are expected to be relatively mild and controllable, as was the case at Salem, a plant of Westinghouse design. In addition it should be noted that the Salem event involved a failure to automatically scram, but manual scram worked as designed.

The NRC staff has been using an estimated scram failure probability of 3×10^{-5} per demand for value impact analyses being done as part of the ATWS rulemaking activities. Consideration of the Salem event would increase this estimate by about a factor of two. While this approach makes it appear that all reactors have the same likelihood of failure to scram upon demand, this is an oversimplification. There are substantial uncertainties in these calculations, and experience indicates the potential of a wider range of probability from plant to plant than might be inferred by the estimation of uncertainties in probability studies. This could be due to variances in design and operational factors (e.g., maintenance procedures and operations quality assurance which are important to the reliability of the reactor protection system).

The staff is aware of scram failure precursors which have occurred at a rate of about 1×10^{-4} per demand. This is reasonably consistent with the Salem event. However, the Salem event raises the concern that the median scram failure probability may be higher than the value currently used in the generic ATWS rulemaking value impact analyses. In light of this event, we are reassessing the ATWS rule proposals and technical bases.

The incident on February 22, 1983 was very unusual in that the operator manually scrammed the reactor within a few seconds of the automatic trip signal. A quick review of the incident on February 23, 1983 by the licensee led them to the erroneous conclusion that the reactor had scrammed automatically. Thus, the plant was restarted on the premise that the reactor protection system was completely functional, and no repairs were made. After the February 25, 1983 incident, a close examination of the plant computer records from the February 22 event showed that the automatic scram did not result in automatic opening of the reactor trip breakers. Although the sequence of events on the computer printout shows that the automatic signal was received first, additional evaluation was necessary to identify that the trip breakers in fact responded to the manually initiated signal.

Questions 6

What sequence of events would have followed the failure of a manual SCRAM, both within and outside the plant gate?

Response

a. Onsite Actions:

In accordance with Salem emergency operating procedures, the actions to be taken by plant operators in the event that the plant fails to automatically trip (scram) on demand, and the manual scram also fails, follow below:

-- open the reactor trip breakers manually by depressing either of the "open" push buttons located in the control room, for both reactor trip breakers.

-- trip the turbine by using the trip handle on the control room console. (A turbine trip also provides a signal to the reactor protection system to trip the reactor trip breakers.)

-- manually initiate a safety injection from the control room.

-- open the reactor trip breakers manually by depressing the push button physically located on either reactor trip breaker.

-- manually trip both rod drive motor-generator (MG) sets at their local control panel; these can be tripped by opening either the power supply breaker to the MG set or the output breaker from the MG set. On either case electrical power to the control rods is removed and scram occurs.

Both the MG set breakers and the reactor trip breakers are located in a switch gear room two floors below the control room. It should take about 1 minute to go from the control room to the switch gear room. In addition to the procedural steps in place at the time of the event, there is the ability to deenergize the control rod drive MG set power supplies from the control room. Revised procedures since the event also include this additional step.

These actions are taken in sequence and are progressive steps to accomplish the function of either inserting the control rods into the core by gravity, reducing rector power or injecting boron in high concentration to shut down the reactor.

b. Offsite Actions

In the event that a scram signal existed and neither an automatic nor a manual scram occurred (indicating a failure of the reactor protection system) PSE&G's Emergency Response Plan calls for declaration of an Alert Condition. The purpose for declaring this condition is to ensure that emergency personnel are readily available, in the event plant conditions degrade. This condition requires notification of Federal, State and local agencies, activation of site support centers and call-out of designated emergency response personnel.

Question 7

What other backup system would have been available if the manual SCRAM would have been ineffective and/or incomplete?

Response

The other backup system and/or operator action available in the event of a manual scram failure are delineated in response "a" to Question 6.

Question 8

At the time of the event, the reactor was reported to be operating at 12% of its rated power. Of what consequence and severity would the failure of the automatic system have been had the plant been operating at near or full power? Specifically, what other problems besides core endangerment might have occurred? What differences would have been relevant in operator reaction time?

Response

Before answering your specific questions it is important to note that in themselves, the Salem ATWS events of February 22 and 25, 1983 were not serious threats to public health and safety. The primary reasons were that the reactor was at low power in both events and the operators alertly scrammed it manually without depending on the automatic scram. However, the events are important in that they are indicators or early warnings of more serious ATWS accidents that could occur. The Salem events would have been more severe if (1) they had occurred when the reactor was at full power; (2) the initiating event had been more severe but within the range of anticipated events for this station; and (3) there were either human errors or additional equipment failures.

For the four-loop Westinghouse PWR design, of which Salem is typical, there have been many previous analyses of severe ATWS accidents in connection with the ATWS rulemaking (Unresolved Safety Issue A-9). We will rely on these analyses performed for the composite four-loop Westinghouse PWR design in answering your specific questions.

To begin with, assume that the reactor were operating at full power soon after startup following a refueling (in the case of Salem 1 in February, the reactor was ascending to full power after a refueling outage). If the reactor were to then experience a complete loss of main feedwater (this would be comparable to the event on February 22, which involved the loss of the one feedwater train that was operating) and if the automatic reactor scram fails entirely (this was the case at Salem), then the automatic turbine trip would also fail (as it did at Salem). Loss of feedwater under these conditions results in a large mismatch between the heat generation and the heat removal rates for the reactor coolant system because the secondary system can no longer remove all the heat generated in the reactor core. If the operator manually scrams the reactor within 30 seconds (as was the case at Salem), there would be no appreciable heatup or pressurization of the primary system and no serious threat to public safety.

If the manual scram were delayed, then there would be an increase in reactor coolant temperature and a decrease in coolant density, producing a surge of coolant to the pressurizer which increases the pressurizer level and the system pressure. The pressurizer relief and safety valves would open to limit pressure buildup. Steam generator inventory would decrease, as the result of boiloff with no replenishment by feedwater flow, and further reduce heat transfer from the primary to the secondary system. Operator actions to cause reactor scram within about 1 to 1-1/4 minutes after loss of feedwater would keep the system pressure near the normal operating pressure and there would be no damage to the reactor. If a manual scram were delayed slightly more, but accomplished within about 1 and 1/2 minutes, Service Level C (about 3200 psi) would not be exceeded and it would be expected that emergency core cooling could be established. After that time, scram would do little to reduce the peak pressure, but it would assist reactor recovery. For the typical four-loop Westinghouse plant, a peak pressure of 3650 psi is estimated if there is no manual scram. For conditions specific to the Salem plant, Westinghouse recently calculated that the peak pressure is only 3200 psi. Pressures above Service Level C increase the likelihood of permanent deformation of valves needed to actuate emergency core cooling needed for recovery of the plant.

Some key assumptions in this severe accident were as follows: (1) a moderator temperature coefficient is minus 8 x 10^{-5} $\Delta K/K^{o}F$, a value that is not exceeded 95% of the time, (2) no credit for turbine trip, and (3) credit for the normal capacity of the pressurizer power operated relief valves (i.e., they are assumed to be unblocked). If the turbine were to be tripped by the operators (in accordance with procedures at Salem) at 30 seconds into the event, the maximum system pressure would decrease about 950 psi even if there were no manual scram. The result would be a peak pressure of about 2700 psi, well within the capability to establish emergency boration and core cooling. There should be little or no fuel damage in this case. This is the most likely event if there were to be a complete loss of scram capability (both manual and automatic).

If both power operated relief valves (PORV) were blocked, there would be an increase of about 300 psi in the maximum pressure of 3650 psi for the severe loss of feedwater ATWS with no turbine trip. If the turbine is tripped, the blockage of both PORVs would increase the peak pressure from 2700 psi to 2950 psi. Blockage of only one PORV (as was the case at Salem on February 25) would increase the pressure about half as much as blockage of both PORVs.

If only one of the two normal feedwater trains were lost, a complete failure to scram from full power with no operator action should result in a mild pressure transient and no fuel damage (Westinghouse calculates a peak pressure of 2330 psia for Salem).

If auxiliary feedwater is initiated by the operator earlier than the 60 seconds assumed in the severe ATWS analysis described above, there would be little decrease in the peak pressure (AFW from two of two trains is equivalent to about 8% of full power). Verification of initiation is called for in the Salem procedures.

Design changes are being considered in connection with an NRC rulemaking that would reduce the likelihood of ATWS events in the Westinghouse design by the decreasing the reliance on the manual scram addressed by your questions. They involve the diversification of the present breaker design for interrupting power to the control rods. Such a change, in the case of the Salem event, would have eliminated the need for manual scram.

The rulemaking also considers practical changes that should reduce the consequences of ATWS events in the Westinghouse design. These are the provision of diverse, automatic initiation of both turbine trip and auxiliary feedwater. It can be seen from the discussion above that the automatic turbine trip removes the need for the operator to manually trip in the first two minutes of an extreme ATWS event to avoid exceeding Service Level C (about 3200 psi).

The Westinghouse analysis for Salem is attached (Attachment 5). Except for small differences in the pressure values due to the design details for that plant, the Westinghouse analysis generally agrees with the staff description provided above (which derived from earlier generic analyses by Westinghouse).

Question 9

Are there other initiating events (i.e., besides low steam generator water level) in which operators would have had less time to respond?

Response

The most limiting anticipated transient combined with delayed reactor trip or failure of reactor trip is the loss of feedwater transient which leads to low steam generator water level. This transient is discussed in the answer to Question 8.

Question 10

The Preliminary Notification of Event or Occurrence notes that an alert was "belatedly declared." What was the cause and effect of this delay? Given the relative urban proximity of the plant, of what consequence would this delay have been?

Response

Following the manual reactor trip on February 25, 1983, the operators' attention was first devoted to placing the plant in a stable condition, which was achieved within a few minutes. At this point in time, there was uncertainty in the minds of the operators as to whether the reactor trip alarm was a valid signal. Personnel from the Instrumentation and Control (I&C) Department were dispatched to examine the annunciators, instrumentation, and protection system circuitry. The shift supervisors waited until they were sure that there had been a failure of the automatic reactor trip system to function properly, as determined by the I&C testing, before declaring an Alert and making the associated notification. This delay in classifying the event as an Alert had no consequences for the surrounding population. Per the Station Emergency Procedures and Federal Regulations, there are four classes of emergency action levels (EAL). These are: Unusual Event, Alert, Site Area Emergency, and General Emergency. The rationale for the notification associated with Unusual Event and Alert classifications is to provide early and prompt notification of minor events which could lead to more serious consequences given potential operator error or equipment failure, or which might be indicative of more serious conditions which are not yet fully realized. A gradation is provided to assure fuller response preparations for more serious indicators. Events involving more serious plant degradation would include other control room indications of reactor and plant parameters or radiation levels that should have enabled the operators to promptly classify the event in accordance with the predetermined EALs appropriate for the situation. These would also require proper notification upon reclassification.

The failure of the reactor trip breakers represented an actual, substantial degradation of the level of safety of the plant in that an important safety system had failed to operate as designed. By definition, this event is classified as an Alert even though the reactor was in a stable, safe condition. By definition, for an event classified at this level it is unlikely that an offsite hazard would evolve and a necessary prerequisite for such classification is a determination that the situation can be corrected and controlled by the plant staff.

However, as a precautionary step for Alert-type events, advisory level notifications to the emergency response organizations of Federal, State, and local authorities are made. Thus, the only offsite effect of the delay in classifying this event as an Alert was a delay in making such offsite advisory level notifications.

Question 11

Please detail the history of resolution of the unresolved safety issue of Anticipated Transient Without SCRAM and state the significance of this event in its eventual resolution?

Response

The possibility of a transient with the inability of the reactor protection system to function was first raised in the late 1960s. The reactor manufacturers performed studies and submitted them to the Atomic Energy Commission, Regulatory Staff in 1970-1971. The Regulatory Staff issued WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors" in September 1973, that contained a Licensing Position on ATWS.

The reactor manufacturers felt that the costs to implement the Licensing Position were too expensive for what was considered to be a very low probability event. The AEC Staff and then the NRC Staff continued to evaluate the probability and consequences of ATWS with more studies supplied by the reactor manufacturers and by the Electric Power Research Institute (EPRI). In 1978-1980, a four volume Technical Report, NUREG-0460, "Anticipated Transients Without Scram For Light Water Reactors," was published with recommendations for resolution of ATWS.

A proposed rule was presented to the Commission in October 1980 (SECY-80-409) to have each applicant perform an evaluation of their plant with respect to prevention and mitigation from ATWS. A Utility Group of 20 electric utilities was formed in the summer of 1980, because they felt that the NRC requirements would be prohibitively expensive. The Utility Group submitted a proposed rule on September 16, 1980 (PRM-50-29) to install certain hardware on plants by vendor type as a resolution to this issue.

At about this same time Commissioner (and later Chairman) Joseph Hendrie was searching for a new approach to resolve the logjam between the NRC staff and the industry. A plan to develop and implement a program for reliability assurance, plus prescribe certain hardware fixes, was presented to the Commission on July 16, 1981. The Commission voted unanimously to publish the "Hendrie" rule and the "Staff Rule" (based on SECY-80-409) for public comment in the Federal Register and to include the Utility Rule as a third alternative. Thirty-nine public comments were received, with the majority of the comments recommending no rule or preferring the Utility Rule.

The NRC Staff presented a revised plan in SECY-82-275 to the Commission on July 13, 1982 to resolve ATWS by forming a Task Force and Steering Group. The proposed Task Force recommendations were presented to the ACRS on October 22, 1982 and to the Committee to Review Generic Requirements on November 3, 1982 and again on January 26, 1983. At the time of the Salem 1 incident the new proposed rule was about 95 percent complete. Currently, the rule is being evaluated in light of the Salem event to determine whether any changes are warranted.

Question 12

The report of this event indicates really two separate events; was the restart of the plant after the first event adequately and properly justified?

Response

In retrospect, it is clear that restart after the February 22 event was not adequately nor properly justified.

The February 22 event, the first event, involved several problems including the deenergization of a 4-KV bus, a loss of control power and indication for the operating main feed pump, loss of a reactor coolant pump, and normal lighting. As a result of these factors, control over the steam generator feedwater system was lost. The control room supervisor apparently recognized the situation and ordered a manual reactor trip at about the same time that the reactor protection system called for a trip. Based on interviews with plant operators, it was not apparent to the operators that the reactor had failed to trip upon receipt of a low-low steam generator water level signal. Based on available information it appears that the operators preoccupation with the numerous other alarms in the control room and a problem with the manual scram switch may explain why they failed to notice the automatic trip failure. The manual reactor trip was in fact inserted 3.6 seconds after the reactor protection system called for a trip.

A post event review of the February 22 event was conducted by plant staff in an attempt to determine what had occurred and to resolve any equipment problems detected. The sequence of events (SOE) computer printout was the best evidence available which could have revealed that the reactor trip breakers had failed to open when the reactor protection system called for a reactor trip. The SOE printout was examined by plant staff members but no attempt was made to sort out the precise timing of each recorded event and therefore, it was apparently not recognized that the reactor trip breakers had failed to open from the low-low steam generator water level signal, a reactor protection system (RPS) signal. The individuals reviewing the event concentrated on the other problems identified above. Specifically, the information provided by the plant computer was apparently used only to verify the sequence of events and not the time intervals between events. Later in the afternoon of February 23, the Assistant General Manager of the station, convinced that the problems of the previous day were understood and corrected, gave approval to restart the plant.

In retrospect, it is apparent that the post trip review of the February 22 event was not conducted in sufficient detail to disclose the malfuntion incurred by the reactor trip breakers since the information (i.e., the computer SOE printout) was available and if properly reviewed, would have revealed that the reactor trip system had malfunctioned. This problem apparently occured becaused the existing procedures for post-trip review did not explicitly require anyone to examine, evaluate or interpret the timing of events.

Question 13

What circumstances explain how the undervoltage trip breakers which are considered "Safety Grade Components" could have been mislabeled during recent maintenance?

Response

The mislabelling, or incorrect classifications, of the maintenance activity as "non safety-related" was due to personnel error coupled with inadequate administrative reviews. In this case, the classifier, by instruction, should have contacted the Engineering Department for the classification; he did not. As a result, there is a need to better understand the management controls that allowed this situation to develop. As can be seen in the staff's Restart Evaluation Report, this specific item was considered in resolving the Management Issues.

This issue demonstrates the need to examine more carefully the management control systems (procedures, audits, etc.) associated with maintenance activities for other than the reactor trip breakers.

Question 14

How is it possible that company officials were unaware of a 1974 safety circular from the vendor explaining special maintenance procedures?

Response

The licensee has indicated that they were unaware of the existence of the vendor's (Westinghouse) 1974 technical service bulletins that provided preventive maintenance recommendations for the reactor-trip circuit breakers.

There were no administrative requirements to ensure proper distribution and adherence to vendor technical bulletins. Therefore, even if Salem had received the circular, they might not have followed it.

Westinghouse has established an interdivisional task force to review current methods for distribution of technical information within the Corporation and methods for distribution of this information to utilities. Additionally, Westinghouse recently has provided (after the February 25 event) the Salem Station with all technical information for equipment supplied for Salem. Apparently, communication problems exist between the Nuclear Services Division and other Westinghouse divisions.

Finally, it should be noted that Salem had in their possession vendor manuals for breakers which recommended a preventive maintenance program. The recommended maintenance was never implemented, however, from the time the breakers were installed in 1976 until January 1983.

Question 15

What remedial steps can the Commission outline that would prevent present and future notices from vendors going unimplemented?

Response

The NRC has established a Task Force to review and evaluate the broader implications of the Salem event. The report from this task force is forthcoming. The issue of vendor notifications to nuclear utility customers is being addressed.

DRAFT ABNORMAL OCCURRENCE

Reactor Trip Breakers Failed To Open On RPS Trip Signal

Date and Place:

On February 25, 1983, Public Service Electric and Gas Company reported an event at Unit 1 of the Salem Nuclear Generating Station, a Westinghouse designed, pressurized water nuclear power plant located in Salem County, New Jersey.

Nature and Probable Consequences:

At 12:21 am on February 25, 1983, a low-low water level condition in one of the four steam generators initiated a reactor trip signal in the Reactor Protection System (RPS). The reactor was at 12% rated thermal power at the time preparatory to power escalation after a recently completed refueling outage. Upon receipt of a valid reactor trip signal, the reactor trip circuit breakers which supply power to the reactor control rods failed to open (opening of either circuit breaker would have caused the reactor to trip). About 25 seconds later, operators manually initiated a reactor trip from the Control Room. The reactor trip circuit breakers opened as a result of the manual trip signal and this resulted in insertion of all control rods and shutdown of the reactor. Following the manual trip, the plant was stabilized in the hot standby condition. All other systems functioned as designed. Later that morning when the cause of the failure had been determined by the licensee, the plant was placed in cold shutdown at the request of the NRC.

Investigation of this incident on February 26, 1983 by the NRC revealed that a similar failure occurred on February 22, 1983, at Salem Unit 1. At 9:55 pm on February 22, with the reactor at 20% power, operators were attempting to transfer the 4160 volt group electrical busses from the station power transformers to the auxiliary power transformers, a routine evolution during power escalation. During the transfer attempt, one of the 4160 busses deenergized resulting in the loss of one reactor coolant pump and power for the operating main feed pump control and indication. At 9:56 pm, a low-low level condition occurred in one steam generator (due to the loss of the main feed pump), initiating a reactor trip signal. Due to the abnormal conditions created by the loss of the 4160 volt bus and in anticipation of loss of steam generator water levels, the operator was directed at about the same time to manually initiate a reactor trip. It was understood by plant personnel and was reported to the NRC that the automatic reactor trip signal due to the low-low water level in one steam generator had, in fact, caused the reactor to trip. On February 26, 1983, as a result of NRC queries, the sequence of events computer printout for February 22 was again reviewed and it revealed that the reactor trip breakers actually opened in response to the operator's manual trip signal. Consequently, it is now evident that on February 22 (as on February 25) the two reactor trip breakers failed to open upon receipt of an automatic trip signal from the reactor protection system.

Since the operators initiated a manual reactor trip shortly after receipt of the automatic trip signals on both February 22 and February 25, no adverse consequences occurred and the reactor was in a safe condition.

Cause or Causes:

On February 25, approximately 2 hours after the event, the cause of the failure to trip was determined by licensee instrumentation technicians to be failure of the undervoltage (UV) trip mechanism associated with each of the two reactor trip circuit breakers to function as designed. The UV trip mechanism consists of a relay and attached mechanical latches; upon receipt of a trip signal from the Reactor Protection System (RPS) the UV coil is deenergized and the mechanical latches cause the trip breaker to open. Opening of either circuit breaker causes a reactor trip. (A manual trip signal operates both the UV trip relay and a separate shunt trip relay within each breaker. The shunt trip relay is energized upon a manual trip signal. Either relay is designed to cause the circuit breakers to trip; and in the February 22 and 25 events, it was the shunt trip relay which actually caused the reactor trip breakers to open.) The failure of the UV trip mechanism was determined by the licensee and the vendor, Westinghouse, to be excessive friction on a mechanical latch lever in the UV trip mechanism. The cause of the excessive friction is still under investigation. The circuit breakers are Westinghouse Type DB-50.

Previous failures of a reactor trip breaker have occurred. Following a DB-50 reactor trip circuit breaker malfunction at the H. B. Robinson Nuclear Power Station in 1973, Westinghouse issued Technical Bulletin NSD-TB-74-1 in January 1974 recommending certain periodic maintenance measures, including lubrication, to improve the reliability of DB-50 breakers. In February 1974, Westinghouse issued a letter (NSD DATA LETTER 74-2) which, among other things, specified that a dry or near dry molybdenum disulfide lubricant should be used in the UV trip mechanism.

It appears that no preventative maintenance was conducted on the Salem Unit 1 DB-50 circuit breakers until January 1983. Additionally, the lubrication recommendations of the Westinghouse 1974 Technical Bulletin and Data Letter were not implemented during the January 1983 maintenance, since personnel performing the maintenance (including a Westinghouse service representative) were not aware of this information.

There have been two previous events at Salem Unit 2 involving a failure of one reactor trip circuit breaker to trip. On January 6, 1983, a reactor trip occurred due to a low-low water level condition in one steam generator and only one reactor trip breaker operated. The second trip breaker finally opened 25 minutes later, although the reactor had already tripped from opening of the other reactor trip circuit breaker. The failure of this trip breaker was concluded by the licensee to be due to dirt and corrosion interfering with proper operation of the UV trip mechanism. As a result of this event, maintenance was conducted on all

Unit 1 reactor trip circuit breakers in January 1983, under the supervision of the circuit breaker vendor, Westinghouse. All breakers were satisfactorily tested after maintenance. Licensee Event Report (LER) 83-001/03L dated January 27, 1983, provides further details of the January 6 event.

On August 20, 1982, during surveillance testing of the Reactor Trip System on Salem Unit 2, one reactor trip breaker would not trip. The cause of the breaker malfunction was concluded by the licensee to be failure of the UV relay coil. The affected coil was replaced, and the breaker was satisfactorily tested. LER 82-072/03L, dated September 8, 1982, provides further details of the August 2 event.

Actions Taken To Prevent Recurrence

Because of the generic implications of this issue, the NRC issued IE Bulletin No. 83-01 on February 25, 1983 to all pressurized water nuclear power plants to inform them of this event. For all pressurized water reactors having DB type reactor trip circuit breakers using UV trip attachments, certain actions were required. These actions included prompt surveillance testing of the breakers, ensuring that preventive maintenance programs on the breakers include the recommended Westinghouse program, and reviewing with operators procedures to be followed in the event of a failure of the reactor to trip on receipt of an automatic trip signal.

With respect to Salem, the NRC staff met with the licensee at the site on February 26 and in Bethesda on February 28. The licensee has proposed certain actions with respect to these breakers including implementing quality assurance requirements, augmenting surveillance test requirements, developing a maintenance program, incorporating the Westinghouse recommendations, and revising procedures to require the operator to employ a manual trip whenever an automatic trip signal is received. The NRC is reviewing these actions to determine whether they are sufficient to correct the deficiencies.

An NRC task force has been assigned to review and evaluate the implications of this event. A Region I task force was assigned to collect facts and data on-site to provide the bases for the generic review. Additional corrective actions may be required at Salem and at other power reactors as a result of the task force review.

Attachment 2

NUREG-0977

NRC Fact-Finding Task Force Report on the ATWS Events at Salem Nuclear Generating Station, Unit 1, on February 22 and 25, 1983*

U.S. Nuclear Regulatory Commission

Region I Task Force

*This title supersedes the title of earlier drafts, which refers only to the February 25, 1983 event. References to the earlier title in the text and bibliographic data sheet have not been corrected in this printing.

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Attachment 3

NUREG-0995

NRC SAFETY EVALUATION
RELATED TO PLANT RESTART
PUBLIC SERVICE ELECTRIC & GAS COMPANY
SALEM NUCLEAR GENERATION STATION
UNIT NOS. 1 AND 2
DOCKET NOS. 50-272 AND 50-311

Date: April 28, 1983

ATTACHMENT 4

SALEM DOCUMENTS

1. 12/17/71 Westinghouse publication NCD-ELEC-18 - Replacement of Undervoltage Attachments on Breakers in Reactor Trip Switchgear.
2. 12/2/77 Document from PSE&G, re Project Directives.
3. 12/16/81 Operating Reactor Events Briefing.
4. 2/22/83 Regional Duty Officer Log
5. 2/22/83 Letter, Midura (PSE&G) to Haynes, re Reportable Occurrence 83-005/03L.
6. 2/23/83 U.S. NRC Region I Morning Report.
7. 2/24/83 Region I Morning Report.
8. 2/24/83 Regional Duty Officer Log.
9. 2/25/83 NRC Duty Officer Log, re initial notification of NRC of Salem Event.
10. 2/25/83 Region I Morning Report
11. 2/25/83 Operations Center Computer Log of 2/25/83 Event.
12. Preliminary Notications of Unusual Occurrence, PNO-I-83-10, PNO-V-83-08, PNO-IV-83-07.
13. 2/25/83 Letter, Midura (PSE&G) to Haynes re Reportable Occurrence 83-011/01P.
14. 2/25/83 Preliminary Notification PNO-I-83-11, re Reactor Trip Breakers Failed to Open on Trip Signal (ATWS).
15. 2/25/83 Preliminary Notification PNO-I-83-11A, re Reactor Trip Breakers Failed to Open on Trip Signal (ATWS).
16. 2/25/83 U.S. NRC IE Bulletin No. 83-01, re Failure of Reactor Trip Breakers (Westinghouse DB-50) To Open on Automatic Trip Signal.
17. Slides presented by licensee (Public Service Electric and Gas Company) during 2/28/83 meeting with NRC staff.
18. 2/28/83 Letter, Hal B. Tucker (Vice-President, Nuclear Production, Duke Power COmpnay) to Denton.

19. 2/28/83 U.S. NRC Regional Daily Report, re Region II and III reported plants that are affected by IE Bulletin No. 83-01.

20. 2/28/83 Memo, Dircks to Denton, re Evaluation of the Implications of the Salem Unit 1 Event.

21. 2/28/83 Letter, Midura (PSG&E) to Haynes, re Reportable Occurrence 83-012/01P.

22. 3/1/83 Letter, R. Uderitz (Vice President, Public Service Electric and Gas Company) to Eisenhut, re Reactor Trip Breaker FAilure No. 1 Unit Salem Generating Station Docket No. 50-272.

23. 3/1/83 Letter, E. P Rahe (Manager, Nuclear Safety Department, Westinghouse) to Denton.

24. 3/1/83 Letter, J. J. Shephard (Chairman, Westinghouse Owners Group) to Denton, re Salem RT Breaker Incident.

25. 3/1/83 Note, J. T. Beard to Holahan/Ippolito, re Breaker Failures at Robinson.

26. 3/1/83 Note J. T. Beard to Holahan/Ippolito, re LER Search/Review.

27. 3/1/83 Letter, H. B. Tucker (McGuire) to Denton, re Additional Information on reactor trip circuit breakers.

28. 3/1/83 Memo, Dircks to Commissioners, re Salem Unit Event.

29. 3/2/83 Commission briefing slides on Salem Event of 2/25/83, presented by Eisenhut 3/2/83.

30. 3/3/83 Board Notification 83-26, re Failure of Reactor Trip Breakers to Open in Trip Signal.

31. 3/3/83 Task Interface Agreement No. 83-20, indicates certain NRC responsibilities regarding evaluation of Salem 1 restart.

32. 3/3/83 Questions provided to PSE&G thru Fisher (P.M. Salem 1/2) concerning Salem 1&2 Q-List.

33. 3/3/83 Memo, Stello to Eisenhut, et. al., re IE Bulletin No. 83-01: Failure of Reactor Trip Breakers (Westinghouse DB-50) to Open on Automatic Trip Signal (Dated February 25, 1983).

34. 3/4/83 Note, Capra to Mattson, re Comments Regarding Items Included on Salem 1/2 Q-List.

35. 3/4/83 Memo, Houston to Knight, re Propbsed Scram Breaker Test Frequencies at Salem Unit 1.

36. 3/4/83, Page from "The Energy Daily".

37. 3/4/83 Memo, Houston to Lainas, re Salem 1 Restart SER, Definition of Safety-Related DSI (ICSB) Input.

38. 3/4/83 Memo, Houston to Knight, re Proposed SCRAM Breaker Test Frequencies at Salem Unit 1.

39. 3/4/83 Memo, Silver to Mattson, re RRG meeting on Salem Event.

40. 3/5/83 Telex from Lee Catalfomo.

41. 3/5/83 Starostecki's proposed outline of "Findings Report".

42. 3/7/83 Letter, Starostecki to Uderitz.

43. 3/7/83 Input to Region I Salem 1 Restart Report from Kennedy

44. 3/8/83 draft of Salem Restart Action Plan, prepared by NRC Region I (Dircks to Commissioners, unsigned).

45. 3/8/83 Memo, Denton to Dircks, re Evaluation of the Implications of the Salem Unit 1 Event.

46. 3/8/83 Memo, Haynes to Heltemes, re Possible Abnormal Occurrence - Salem Unit 1 Failure of Reactor Trip Breakers to Open on Trip Signal.

47. 3/8/83 Letter, Uderitz to Starostecki, re Reactor Trip Breaker Failure.

48. 3/9/83 Letter, Gary Toman to Noonan.

49. 3/9/83 Letter, Uderitz to Starostecki, re Confirmatory Action Letter CAL 83-02.

50. 3/9/83 Memo, Miraglia to Eisenhut, re Use of DB-50 Breakers in RPS at Ginna and Haddem Neck.

51. 3/9/83 Memo, Fisher to Varga, re Interim Draft Salem Restart Report.

52. 3/9/83 Memo, Knight to Lainas, re Salem Unit 1 Restart Report.

53. 3/10/83 Memo, Fischer to Denton, et al, re Daily Highlight.

54. 3/10/83 Memo, Eisenhut to Vollmer, et al, re Congressional Subcommittee Request.

55. 3/10/83 Memo, Mattson to Management Oversight Members of the Salem Generic Implications Task Force.

    Enclosures: (a) 3/8/83 Memo, Denton to Dircks, re Evaluation of the Implications of the Salem Unit 1 Event.

    (b) 3/9/83 Memo, Starostecki to Mattson, re Report of the Region I Task Force on the ATWS Events at Salem Nuclear Generating Station, Unit 1.

    (c) Draft Outline for the 4/18/83 Task Force Report.

56. 3/10/83 Memo, H. Silver to Mattson, re Meeting Notice of INPO Meeting on Salem Generic Implications.

57. 3/10/83 Memo, Heltemes to Denton, re Potential Design Deficiency in Westinghouse Reactor Protection System.

58. Westinghouse publication I. B. 33-850-3D, effective May 1970, re Instructions for Types DB-50, DBF-16 and DBL-50 Air Circuit Breakers.

59. Plant status summary of Salem Units 1 and 2 from 1/3/83 to 2/25/83.

60. Itemized list of Westinghouse domestic plants using DS Breakers.

61. Unconfirmed list of Westinghouse domestic plants using DB Breakers.

62. Itemized list of Westinghouse domestic plants and Reactor Trip Breakers being used.

63. Salem Unit 1 plant computer printout during event of 2/25/83.

64. Inventory of control room instrument recorder strip charts for 2/25/83 Salem Unit 1 event. (Obtained during plant visit.)

65. Control room instument recorder strip charts for 2/22/83 Salem Unit 1 event. (Obtained during plant visit.)

66. Salem electrical drawing #240148 B 9654-0.

67. Salem plant procedure IPD-18.1.004 Solid State Protection System Reactor Trip Breaker and Permissive P-4 Test - Train A.

68. Salem plant procedure IPD-18.1.008 Solid State Protection System Functional Test - Train A.

69. Salem Emergency Procedure EPI-1 Notification of Unusual Event/Significant Event.

70. Salem Unit 1 Alarm Procedures.

71. Certificate of Conformance accompanying UV trip attachment 23A9019G61.

72. Copy of Work Order No. 925774 for Reactor Trip Breakers.

73. Salem plant procedure IPD-18.1.009 Solid State Protection System Functional Test - Train B.

74. Salem plant procedure IPD-18.1.005 Solid State Protection System Reactor Trip Breaker and Permissive P-4 Test - Train B.

75. Salem Emergency Instruction I-4.3 Reactor Trip.

76. Salem 1 Restart Report.

77. Salem 1 Restart Report with Cases comments dated 3/7/83.

78. Salem Restart Report. (3/9/83 markup)

79. Telex from FRC regarding Maintenance Procedures.

80. Instructions for Itemizing Equipment for MEL.

81. One page to Fisher regarding Salem Nuclear Generating Station Reactor Trip Switchgear Operational Verification Program.

82. Master File No. 255 from PSE&G (2 page excerpt.)

83. General Conclusions of 3/4/83 Visit to Salem Site (3/4/83 draft).

84. Conclusions on Operator Training and Procedures (3/7/83).

85. Input to Region I Salem Restart Report (3/7/83).

86. Information Paper on Salem Restart Action Plan (3/8/83 draft markup).

87. Summary of Licensees' Responses to IEB 83-01.

88. Staffs Comments on Salem Restart (3/8/83).

89. Information Paper on Salem Restart Status Report (3/10/83 draft).

90. Equipment Specification Cover Sheet for Reactor Trip Switchgear (3/3/83).

91. SSPS Train "B", Reactor Trip Breaker UV Coil Functional Test.

92. Maintenance Procedure A-11 (Rev. 0).

93. Maintenance Procedure A-11 (Rev. 1 draft).

94. Maintenance Procedure M3Q-2 (Rev. 1).

95. QA list for SER Salem 2.

96. Record of Maintenance on Breakers.

97. Reactor Trip and Safety Injection of 2/22/83 (dated 2/23/83)

98. Salem Generic Implications, Agenda for RRG Meetings, March 8-11, 1983.

99. Draft NRC Region I Task Force Charter.

100. Region I Summary of Actions Taken as Result of Salem 1 ATWS Event.

101. Salem Restart Report - SECY-53-98A, March 14, 1983

102. Salem Restart Status Report - SECY-83-98C, March 29, 1983

103. Salem Restart Evaluation - SECY-83-98D, April 8, 1983

104. Salem Restart Evaluation - SECY-83-98E, April 11, 1983

105. Letter PSE&G to R. Starostecki - Additional Information on Corrective Actions, March 18, 1983

106. Letter F. P. Rahe to R. Vollmer, Informkation on Field Service on Breakers, March 24, 1983

107. Letter from E. P. Rahe to R. C. DeYoung, Information on Trip Breakers, March 31, 1983

108. Letter PSE&G to D. Eisenhut, Supplement to Corrective Actions, April 3, 1983

109. Letter E. P. Rahe to H. Denton, Information on Westinghouse investigation of malfunctions, March 22, 1983

110. Letter PSE&G to D. Eisenhut, Additional details of independent management diagnostic, April 11, 1983

111. Letter V. Gilinsky to W. Dircks, Salem Breaker Testing, April 11, 1983

112. Letter to D. Eisenhut from PSE&G, Completion of Short Term Action Items, April 13, 1983

113. Summary of March 14, 1983 meeting with PSE&G & Staff, Restart Status, April 18, 1983

114. Letter PSE&G to D. Eisenhut, Commitment to independent management diagnostic, April 4, 1983

115. Letter PSE&G to R. Starosteci, Vendor Manual Program, March 23, 1983

116. Letter W. Carrington to S. Pandry, Contract Work Scope (FRC), April 22, 1983

117. Letter W. Dircks to Commissioners, Verification of Actions Performed by PSE&G, April 22, 1983

118. Letter PSE&G to R. Starostecki, Additional Information and Comments on NUREG-0977, April 22, 1983

119. Letter PSE&G to D. Eisenhut, Responses to 10 CFR 50.54(f) letter, April 22, 1983

120. Letter PSE&G to D. Eisenhut, Beta Corporation Reports, April 28, 1983

121. Letter PSE&G to D. Eisenhut, Clarification of Response to 10 CFR 50.54(b) letter, April 27, 1983

122. Letter PSE&G to D. Eisenhut, Corrective Action Summary, April 28, 1983

123. Salem Restart Evaluation, April 28, 1983

Analyses of the Salem Nuclear Plant for Postulated Feedwater Malfunction without Automatic Reactor Trip

WESTINGHOUSE ELECTRIC CORPORATION

M. P. Osborne
Transient Analysis
Nuclear Safety Department

Approved: 2/3/83
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3999Q

SCOPE

In light of the recent failures of the reactor trip breakers to automatically function at the Salem plant, the purpose of this study is to realistically predict the consequences of a failure to trip for limiting plant transients while the plant is at full reactor power. The transients analyzed, specifically for the Salem plant, are a partial loss of steam generator main feedwater flow due to the trip of a single main feedwater pump and also a complete loss of main feedwater flow due to the loss of both main feedwater pumps. The latter, less probable, event is that presented in the Salem plant FSAR. As stated previously, the purpose of this study is to realistically predict the response of the plant to these events and, as such, the plant systems are assumed to function normally with the sole exception being the common mode failure of the reactor breakers to automatically function as was experienced on February 22 and 25, 1983. It should be noted that the spurious steam generator level trip generated on 2/25/83 was as a result of normal expected feedwater control system difficulties experienced at low (11%) power levels. It also should be noted that the loss of a feedwater pump on 2/22/83 was due to a normal manuevering of an electrical bus while configuring the plant in preparation for a power escalation. Both of these events are not normally expected at full power and thus one should consider more credible events such as a feedwater heater dropout rather than the more limiting and much less frequent feedwater pump malfunctions.

The study considers a thirty second operator response time for a manual reactor trip following the automatic protection system demand signal, a simulation of the actual response time of the February 25, 1983 event. The study also considers a more conservative operator response of five minutes in order to determine the sensitivity of the plant response to operator action.

DESCRIPTION OF TRANSIENT EFFECTS

Generic studies (WCAP 8330 Westinghouse Anticipated Transients Without Trip Analysis) of failure to trip events previously submitted to the NRC have identified the limiting full power events to be malfunctions affecting steam generator main feedwater flow. The reduction in main feedwater flow affects the overall heat removal capability of the steam generators and, as a result of the mismatch between the primary side heat generation and the secondary side heat removal produces a heatup of the primary system coolant. If the reactor is tripped promptly, the auxiliary feedwater system provides sufficient heat removal capability to remove decay heat. However if feedwater flow to the steam generators is reduced or terminated without subsequent reactor trip the secondary system will be unable to remove all of the heat that is generated in the core. This heat buildup in the primary system is a function of the amount of the feedwater reduction and is indicated by rising reactor coolant system temperature and pressure, and by increasing pressurizer water level due to the insurge of the expanding reactor coolant. Water level in the steam generators drops as the remaining inventory in the steam generators is boiled off due to inadequate spply of feedwater. When the steam generator water level falls to the point where the steam generator tube bundle is uncovered and primary to secondary heat transfer is reduced, reactor coolant system pressure and temperature increase at a greater rate. This greater rate of temperature and pressure increase is maintained as the pressurizer fills completely and water is discharged through the pressurizer relief and safety valves. Reactivity feedback, due to the high primary system temperature, reduces core power. As a result the system presure begins to decrease and a steam space is again formed in the pressurizer.

The limiting criteria for the postulated transients is that reactor coolant pressure be maintained sufficiently below the pressure corresponding to the ASME Code Service Level C (Emergency) stress limits. For the reactor coolant system, the corresponding pressure is 3200 psia.

CONTROL ROOM INDICATIONS AND MITIGATING ACTIONS

Although the reactor is prevented from tripping automatically by the common mode failure of the reactor trip breakers, there are many control room indications and alarms which are generated during the transient which would serve to alert the operator that the event has taken place. These indications in addition to emergency procedures, which require the verification of a successful reactor trip before all other actions, would support the mitigation of the consequences of the transient.

For a loss of normal feedwater event, in addition to normal process control alarms (pump trip, temperature, pressure, level and flow deviation alarms for both primary and secondary systems), the following audible alarms would be generated:

1. Steam/feedwater flow mismatch and low level (each steam generator)
2. Overtemperature Delta-T turbine runback
3. Overtemperature Delta-T reactor trip demand
4. Overpower Delta-T turbine runback
5. Overpower Delta-T reactor trip demand
6. High pressurizer pressure reactor trip demand
7. High pressurizer level reactor trip demand
8. Steam generator low-low level reactor trip demand
9. Low steam pressure safety injection (in coincidence with high flow)
10. Low reactor coolant loop flow reactor trip demand

Tables 1 and 2 show the time sequences for these alarms.

As part of the procedures the operator is required to exercise following any reactor trip demand, the operator is required to first verify the successful accomplishment of the reactor trip by observing rod position indicators, rod bottom lights, neutron flux, or reactor trip breaker position indications. The following actions are available to the operator in the main control room if an unsuccessful reactor trip occurs:

1. Manual reactor trip (with subsequent automatic turbine trip)
2. Manual turbine trip
3. Manual turbine runback (200%/min.)
4. Manual safety injection
5. Manual control rod insertion.

Outside the obvious benefit of an immediate reactor trip, the turbine trip or turbine runback action is the most important, if a reactor trip cannot be obtained manually, to terminate the steam flow demand from the steam generators to preserve steam generator inventory. Steam pressure and hence primary system temperature will be controlled by means of the steam dump control system, steam generator relief and/or safety valves. Other means outside the main control room are available:

1. Local manual trip of any reactor trip breaker
2. Local manual trip of the rod control system motor-generator sets
3. Local manual trip of the turbine

TRANSIENT SIMULATION

Analyses were performed to simulate both a partial and complete loss of main feedwater. These analyses are based upon previous models consistent with previous submittals to the NRC by Westinghouse on ATWS (NS-TMA-2182, T. M. Anderson to Dr. S. Hanauer, 12/30/79) but also are modified to more accurately model the Salem Plant.

The following conditions were assumed for both analyses:

1. Initial normal full power operation at beginning of core life. This corresponds to the current condition of the Salem Plant and is also the limiting condition since the moderator temperature coefficient is at its least negative value. A value of -8 pcm/°F, which is valid for 95% of core life, was assumed.

2. Both the pressurizer relief and safety valves are assumed to function. There are two relief and three safety valves. Pressurizer heaters and spray also function automatically.

3. The automatic turbine runback on either Overtemperature or Overpower Delta-T signals is operable. The runback setpoint is 3% below the trip setpoint. The turbine runback operates on a 30 second cycle. Turbine load is first reduced 5% in 1.5 seconds. If at the end of the 30 seconds the runback signal still exists, the load is further reduced another 5% and so on. The load reduction has a mitigating effect on the transient and helps reduce peak primary system pressure.

4. The rod control system is assumed to be in the manual mode consistent with actual practice. Automatic action of the rod control system would cause rod insertion when primary temperature increases and would be less conservative.

5. The steam dump control system is available. The capacity of the steam dump is 50% of nominal steam flow at full power.

6. Auxiliary feedwater flow (1760 gpm) begins at 10 seconds following receipt of the low-low steam generator level signal. This response time is based upon actual test data from the Salem Plant.

7. Operator action is assumed to initiate a successful manual trip. Turbine trip is initiated via the reactor trip breaker opening.

8. For the complete loss of feedwater transient, the main feedwater pumps are assumed to coastdown to zero flow in five seconds. For the loss of a single pump, one pump is assumed to coastdown to zero flow in five seconds; however, the remaining pump has rated flow capacity of 70% of nominal full power feedwater flow. Therefore, the second pump (the Salem Plant has two pumps) will increase its flow to 70% flow. The response time for the second pump is 20 seconds.

9. Nominal control and protection system setpoints were assumed.

TRANSIENT RESULTS

1. Loss of a Main Feedwater Pump

The sequence of events for both a 30 second and 300 second delay of manual reactor trip are shown in Table 1. The transient primary pressure calculations are shown in Figure 1. The low-low steam generator level setpoint is reached at 99 seconds; auxiliary feedwater is automatically initiated. Ten seconds later, auxiliary feedwater begins to be delivered to the steam generators.

30 Second Delay

For the case where there is only a 30 second delay, there are no subsequent reactor trip signals generated. There is no large heatup of the reactor coolant because the steam generator tube bundle does not uncover. Thus there is always adequate secondary side heat removal. The peak pressure of 2286 psia which occurs at 30 seconds, is only slightly above the pressure at which the presurizer sprays are actuated.

For this transient, the reactor coolant system integrity is not challenged.

5 Minute Delay

For the case where operator action is delayed 300 seconds (5 minutes), the reactor coolant system temperature increases, reaching the Overpower Delta-T setpoint for turbine runback at 190 seconds. This signal is maintained and thus turbine power continues to reduce 5% every 30 seconds until the turbine load is at 75%. At this point, the sum of the main feedwater flow from one pump plus the auxiliary feedwater flow is equal to the turbine steam flow. Therefore, steam generator level does not continue decreasing and stabilizes. The operator initiated reactor and turbine trip at 399 seconds occurs after the steam and feedwater flow have matched. The peak primary system pressure of 2330 psia at 267 seconds occurs before the steam and feed flow are matched. This pressure is below the relief valve setpoint (2350 psia). The pressurizer sprays, combined with the effect of reduced turbine load prevent any significant overpressurization. Again, reactor coolant pressure stays below service Level C limits of 3200 psia.

2. Loss of All Main Feedwater

The sequence of events for this transient are presented in Table 2. The transient pressure calculations are depicted in Figure 2.

The low-low steam generator level setpoint is reached at 33 seconds; 10 seconds later, auxiliary feedwater is delivered to the steam generators.

30 Second Delay

An automatic turbine runback due to an Overpower Delta-T is initiated at 43 seconds and turbine load is reduced 5%. The pressurizer relief valves open and maintain pressure at the setpoint value (2350) until the operator trips the plant at 63 seconds. Steam dump is initiated and reduces the primary temperature to the no load value of 547°F. For this transient the reactor coolant system pressure is well below 3200 psia.

5 Minute Delay

As in the previous case, the heatup of the primary coolant caused a turbine runback initiated by an Overpower Delta-T signal. The turbine load is reduced twice in 5% increments until the load is 90% of nominal load. Steam pressure starts to drop due to the boil off of water in the steam generators, generating a low steam pressure alarm. At this time primary pressure starts to increase and there is an insurge into the pressurizer, causing both pressurizer high level and pressure trip alarms to be actuated. The steam generator tube bundle begins to uncover, causing a larger rate of increase in primary pressure and temperature. The pressurizer fills and the peak pressure reached is 3491 psia. Nuclear power has decreased at this point to about 30% of nominal due to the negative moderator temperature reactivity feedback. As the relief rate of water through the relief and safety valves increases, the primary system pressure starts to decrease and the safety and relief valves close about 30 seconds after the time of peak pressure. The operator trips the reactor manually at 333 seconds.

CONCLUSIONS

Loss of a Main Feedwater Pump

The results presented here demonstrate that for the loss of one main feedwater pump, there are at least six major alarms in addition to others generated to alert the operator to the fact that a malfunction has occured. Furthermore, even for the event with a five minute delay in reactor trip automatic turbine runback reduces steam flow to match the capability of the auxiliary feedwater. For this event there is no threat of overpressurization in that the pressurizer relief valve setpoint is not even reached.

Complete Loss of Main Feedwater

For the complete loss of feedwater, operator action consistent with the action time taken at the plant on the February 25, 1983 event is sufficient to prevent overpressurization of the reactor coolant system. Peak primary system pressure results only in pressurizer relief valve actuation without the actuation of pressurizer safety valves. Furthermore, there are 3 major alarms which are actuated in addition to the steam generated low-low level alarm to alert the operator to take action.

As discussed earlier, it is a major reduction in primary to secondary heat transfer capability which causes the primary system heatup and pressure increase. A turbine trip reduces the amount of steam flow and the rate at which the level in the steam generator drops. If the turbine is tripped before there is a significant loss of steam generator inventory, the tubes will not uncover and the primary system will not overpressurize. Based upon the results discussed in the previous section, operator action to trip the turbine at or before one to one and a half minutes following the low-low level trip and alarm would prevent overpressurization of the reactor coolant system beyond 3200 psia.

It should be noted that the core nuclear characteristics (a moderator reactivity coefficient of -8 pcm/°F) used are not representative of the actual core design for the Salem Plant. Previous ATWS analyses have shown the peak pressure to be a strong function of the coefficient and there is a 100 psi reduction for every 1 pcm decrease in the coefficient. The Salem core is designed to operate such that by the time the plant reached full power it would have a coefficient of -10.5 pcm/°F or 2.5 pcm less than the coefficient in the study. This coefficient would be reduced even further by approximately 2 pcm/°F per month of operation (see Figure 3). The 10.5 pcm coefficient results in a peak pressure for the limiting case of five minute operator action of 3241 psia (a 250 psia reduction from 3491 psia) which is within the calculational band of the ASME Stress Level C limit. Therefore, the case represented in Figure 2 would not exceed the acceptance criteria.

Summary

In conclusion, this study has demonstrated the ability of the Salem Nuclear plant to withstand the effects of postulated gross feedwater malfunctions without reactor trip at full power with an artificially long delay for operator action. The results show acceptable response which is within calculational uncertainties of the ASME Stress Level C limits. These results are further affected by the low probability of these events occuring at full power in addition to the expected increasingly beneficial nuclear characteristics of the plant over core life.

TABLE 1

Sequence of Events

Loss of One Feedwater Pump

Event	Time[1]	Time[2]
Loss of one pump (alarm)	0	0
Remaining pump delivers maximum flow	20	20
Low-low SG level setpoint (alarm); auxiliary feedwater signal (alarm)	99	99
Auxiliary feedwater begins	109	109
Operator trips reactor and turbine	129	---
OP Δ T runback setpoint (alarm) turbine load reduced 5%	---	190
Turbine load reduced 5%	---	233
OP Δ T trip setpoint (alarm)	---	220
Turbine load reduced 5%	---	250
Peak Pressure Occurs	---	267 {2330 psia}
Turbine load reduced 5%	---	280
Turbine load reduced 5%	---	310
High pressurizer level setpoint (alarm)	---	333
Operator trips reactor and turbine	---	399
	3 alarms prior to trip	6 alarms prior to trip

(1) 30 second delay before manual trip

(2) 300 second delay before manual trip

TABLE 2

Sequence of Events
Complete Loss of Main Feedwater

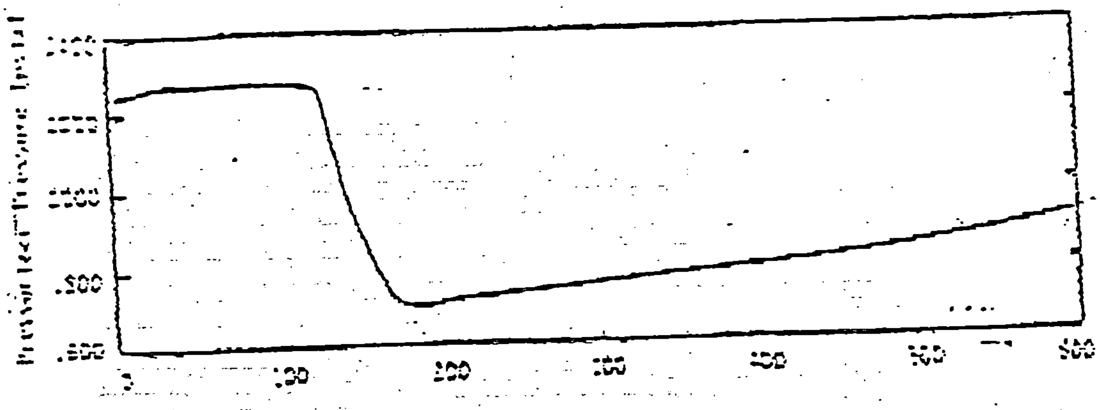
Event	Time[1]	Time[2]
Loss of main feedwater pumps (alarm)	0	0
Low-low SG level setpoint (alarm); auxiliary feedwater signal generation	33	33
OPΔT runback setpoint (alarm) turbine load reduced 5%	34	34
OPΔT trip setpoint (alarm)	43	43
Auxiliary feedwater begins	43	43
Pressurizer relief valves open	55	55
Operator trips reactor/turbine	63	---
Turbine load reduced 5%	---	64
High pressurizer level trip setpoint (alarm)	---	85
Low steam pressure SI (alarm)	---	85
High pressurizer pressure setpoint (alarm)	---	88
SG tubes begin to uncover; steam flow drops pressurizer safety valves open	---	92
Pressurizer fills	---	95
Peak pressure	---	134 (3491 psia)
Pressurizer safety valves close	---	142
Pressurizer relief valves close	---	155
Low RC flow setpoint (alarm)	---	159
Operator trips reactor/turbine	---	333
	4 alarms prior to trip	7 alarms prior to trip

(1) 30 second delay before manual trip

(2) 300 second delay before manual trip

Pressure Transient - Loss of One Feedwater Pump

30 Seconds Delay Before Trip



300 Seconds Delay Before Trip

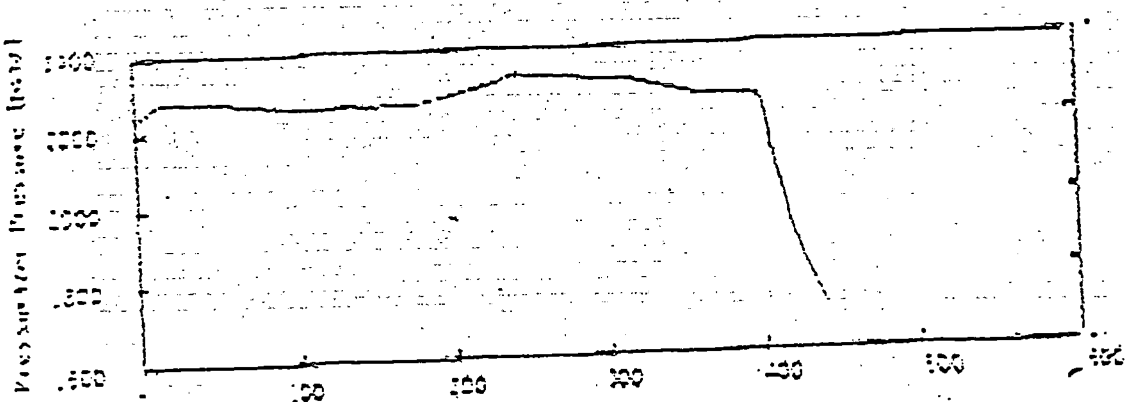
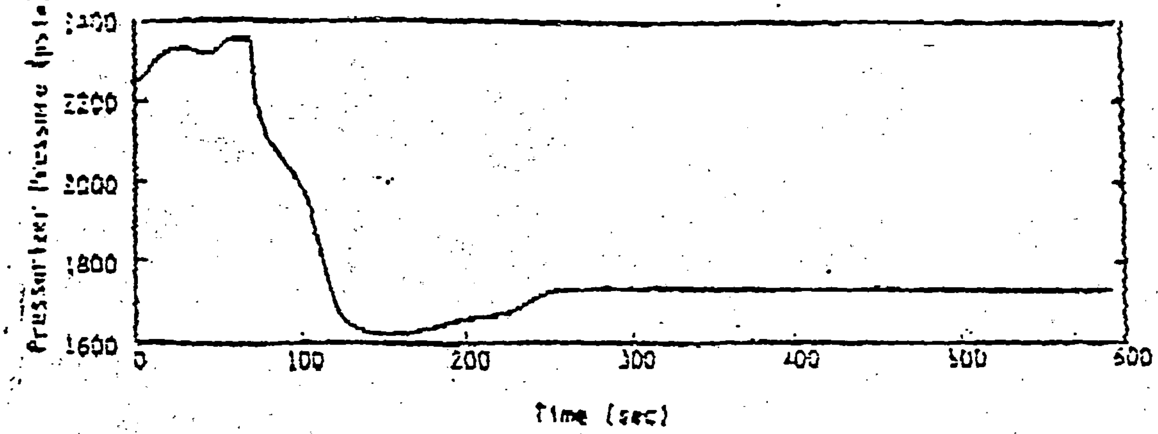


FIGURE 2

Pressure Transient - Loss of All Main Feedwater

30 Seconds Delay Before Trip



300 Seconds Delay Before Trip

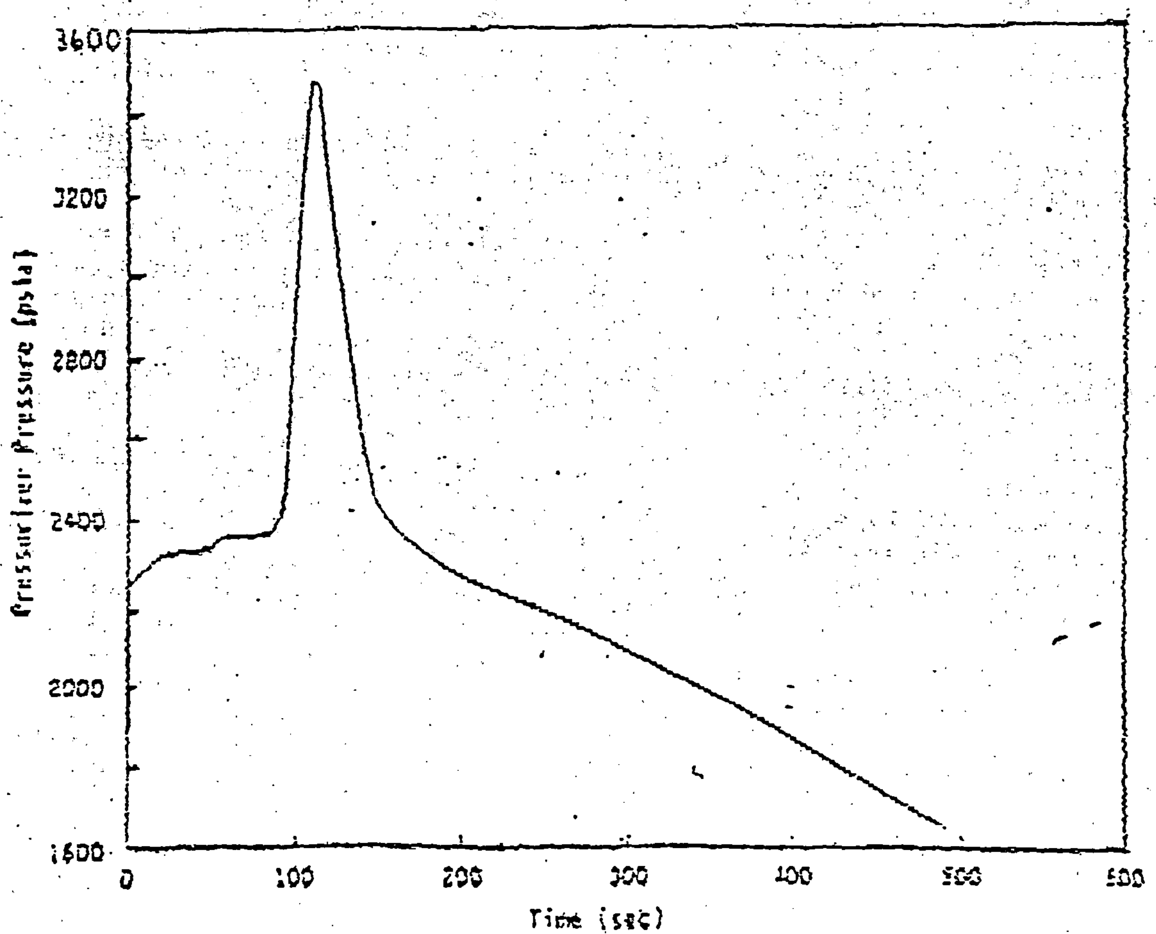
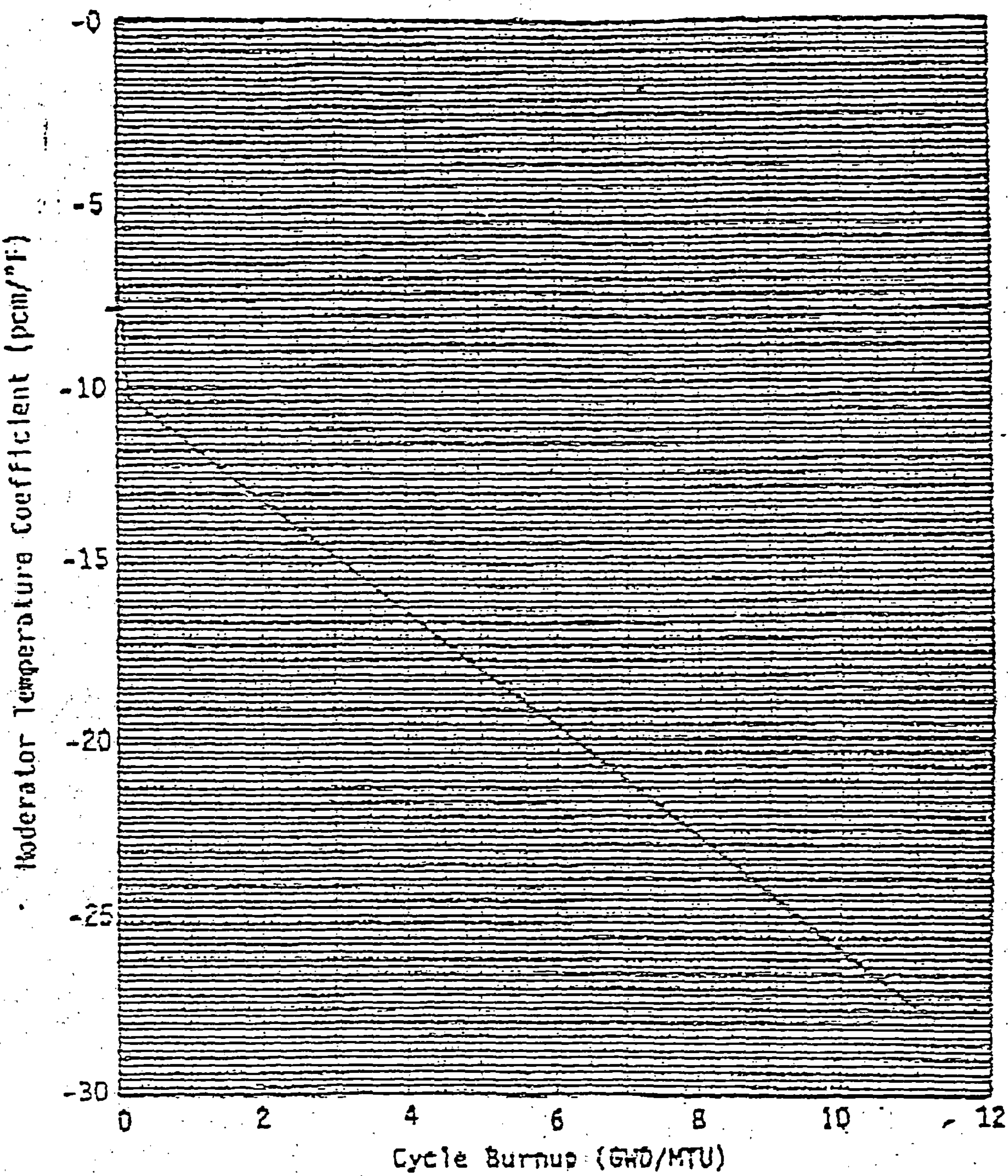


FIGURE 3

MODERATOR TEMPERATURE COEFFICIENT DURING CYCLE 5
AT HFP, ARO, EQUILIBRIUM XENON CONDITIONS *



*WCAP 10242, "The Nuclear Design of Salem Unit One Power Plant Cycle 5"

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