Plymouth Nuclear Matters Committee
Town of Plymouth
11 Lincoln Street
Plymouth, MA 02360

September 20, 1989

Action
Projects
for TM sign.
10/23/89 due

Mr. Thomas E. Murley
Director
Office of Nuclear Reactor Regulation
Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20814

RE: Pilgrim Nuclear Power Station
Direct Torus Vent System

Dear Mr. Murley,

Please find enclosed, copies of correspondence relating to the recently installed hardened wetwell vent at Pilgrim Station. Since several of the issues under discussion concern NRC's review and approval of the system and since you are one of the key individuals involved in the modification, we are seeking your input to help clarify this situation. It would be greatly appreciated if you could respond directly to relevant aspects of this issue in writing to the above address.

Although Generic Letter 89-16 states, "The staff found the installed system and the associated BECo analysis acceptable," we have not been able to conclude this from any of the other existing documentation. Specifically, all of the Safety Evaluations describe only the installation, not the use of the vent. Also, the logic used in Safety Evaluation 2269, dated 1/9/88, which concludes that a change to the Technical Specifications is not required, is very questionable. Do you concur with BECo's arguement there?

In addition, inadvertant or premature venting is a very serious safety question, yet, in various documentation, BECo maintains that the DTVS does not involve an unreviewed safety question. If you agree, could you explain why it does not?

Many state and local public officials, as well as numerous residents realize the close and necessary linkage between controlled venting and emergency preparedness. However, as you may well know, the adequacy of emergency planning for Pilgrim is hotly debated. The topic is even under investigation by the NRC Inspector General's office. Do you believe that the DTVS should have been allowed to be made operational without adequate emergency prepareness by the community and the licensee?

Obviously, this is a far reaching technical and politically sensitive issue within the NRC. In reviewing the documentation, we, of course, would have preferred that the NRC approach to this issue had been more straightforward: if it was a good idea, get behind it and insure that it was designed, installed, and planned for properly, and if it was a bad idea, stop it from being implemented. However, the

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existing documentation indicates official hesitancy; no one seemed to relish having their names, reputations, and careers closely tied to this plant modification. We live downwind of Pilgrim. It would reassure us if you could provide your assurances that BECo is up to the task of using this powerful new tool.

Pilgrim, as you are aware, has had a very troubled history: Some of the largest fines, longest shutdowns, most expensive capital repairs, highest O & M costs, and lowest SALPs of currently operating reactors. Now, the first DTVS in the nation is installed here and we are extremely concerned.

Also included is our report on the April 12, 1989 spill in the RCIC system at Pilgrim. There are many issues here which we feel will be of considerable interest to you. First, the AIT report contained errors. Second, the executive summary and cover letter did not reflect the conclusions from the body of the report or from the appendixes. Third, it was an interfacing systems loss of coolant accident, a topic with which you have been closely involved. Fourth, we are requesting higher level NRC review of the issue, with special emphasis on the role of NRC in the event investigation and, more broadly, in the power ascension oversight. These are serious assertions and serious requests.

Your comments on both of these matters would be greatly appreciated.

Thank you,

David C. Dixon
Vice-Chairman, Plymouth Nuclear Matters Committee

TOWN OF PLYMOUTH

11 Lincoln Street
Plymouth, Massachusetts 02360

(617) 747-1620

September 5, 1989

Mr. David F. Tarantino
District Manager
Nuclear Information Division
448 State Road, Suite 5
Plymouth Ma, 02360

Dear Mr. Tarantino,

Thank you for the information on the Direct Torus Vent System. Regretably, we already had obtained those documents, with the exception of the most recent letters between Peter Agnes and Ralph Bird, and the questions we had asked resulted from the study of those documents. We now resubmit the questions and ask you to seek direct responses to them.

The significance of this issue should not be underestimated. Prior to the DTVS, one of the final layers of defense in depth was the steel and concrete Mark 1 containment, which has a burst pressure of over 100 psi. The DTVS punches through that layer, relieving directly to the environment at only 30 psi. It is the most significant change to Mark 1 containment design in twenty years, and is the first such system in the nation. It use requires early notification and coordination with Civil Defense officials in the EPZ.

While we would like specific responses to the ten questions, the most important issues can be distilled into two main areas:

1. The NRC has indicated in several instances that they were unwilling to endorse Pilgrim's DTVS and that the installation of valve AO-5025 would require a change to the Technical Specifications. In all of the documentation available to us, the installation and the use of the DTVS were analyzed seperately. Futher, BECo states repeatedly that the system will not be made operational, that the valve AO-5025 will not be installed without formal NRC approval. The valve is now installed and operational. Can you provide this committee specific documentation indicating that NRC has now formally approved the use of the Pilgrim DTVS, that its use does not introduce unreviewed safety questions and that BECo, in proceeding with the installation, has not violated 10 CFR 50?

2. The logic behind using the DTVS is complicated.Yet, our reading of the documents indicates that there have been no changes to your EOP's incorporating the new decision trees or early notification requirements; no training on the use of the special keys, electrical jumpers, special fuses; or the other idiosyncracies of the system. No management review, no public involvement. Only the pre-existing EOP-3 relates to containment venting, and BECo did not rewrite it before implementing the new DTVS. If detailed procedures have been prepared, please issue us a copy. If not, please explain why it is not necessary to prepare to use this powerful and potentially dangerous system.

While we commend BECo for going beyond the NRC requirements for mitigating severe accidents beyond the design basis, we require assurances that the system has been implemented properly and that both the utility and the state and local groups are prepared for its use. We have not obtained that assurance from the available documentation.

If you require clarification of this request, please write to our committee, care of the Town of Plymouth, or call committee member David Dixon at 508-946-1000 during the day.

Thank you,

Gerald L. Hayes Chrm

Plymouth Nuclear Matters Committee

CC: Ralph Bird, Sr. VP-Nuclear, BECo
Plymouth Selectmen
Thomas Murley, NRC-NRR
William Russell, NRC Region 1
Richard Wessman, PDI-3/NRR
Dan McDonald, NRC-NRR
Charlie Marshall, Pilgrim Resident Inspector
Members of the Nuclear Safety and Health Advisory Committee

To: Plymouth Selectmen and Plymouth Nuclear Matters Committee Members

From: David C. Dixon

Subject: Request for Information on the PNPS Direct Torus Venting System

Date: June 13, 1989

During our tour of Pilgrim last month, Mr. David Tarantino offered to have technical questions about the Direct Torus Venting System (DTVS) answered by the engineering staff. In response, our committee has developed the attached list of questions. They were reviewed and approved by committee during the May 24, 1989 meeting.

These questions have arisen from our study of the DTVS. It is an important issue which has received little public discussion, in part due to its technical nature. This vent releases pressure, and possibly fission products, from the containment during a severe accident directly into the atmosphere, thus bypassing the inherent safety offered by the steel and concrete protective containment structure. In theory, it is to be used only as a stopgap measure to keep the containment from rupturing, thereby avoiding a more serious, uncontrollable release of fission products to the environment.

There are three main issues in the analysis:
(1) Under what accident scenarios is the DTVS intended to be used, given that for some accidents it helps, some it exacerbates and others it's irrelevant?
(2) Has BECO implemented the concept properly? Has it minimized the risks of improper use of the vent, such as inadvertent or premature venting? Are their people trained to use such a powerful tool should it ever become necessary? Is the public prepared to respond?
(3) Has the NRC played its proper role in this modification? Since the modification exists to mitigate accidents beyond the design basis, the NRC has taken a hands-off approach. Also, if the NRC had maintained its initial assertions that the DTVS required a change to the Technical Specifications, public hearings could have been necessary.

We are requesting this information from BECo to enable us to issue a more complete report analyzing the DTVS. Answers to these questions will fill in some of the gaps.

DCDixon

TOWN OF PLYMOUTH

11 Lincoln Street
Plymouth, Massachusetts 02360
(617) 747-1620

June 2, 1989

Mr. David Tarantino
Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, MA 02360

Dear Mr. Tarantino,

Thank you for guiding us on the informative tour of the station last month. The time spent allowed the members of our committee to better understand the operation of the facility.

During the tour, you offered to accept questions of a technical nature about the direct torus vent. The committee has several questions for which we would like answers before proceeding with our review of the DTVS.

The members of the committee believe that the DTVS is a powerful and somewhat controversial tool which could help the plant operators mitigate the effects of a severe accident.

We need to acquire a better understanding of the system to help us evaluate the benefits and risks of this installation. Your written response would be greatly appreciated.

Should you need to discuss this request for information, please feel free to call or write to one of our committee members: David C. Dixon, 135 Gunners Exchange, Plymouth, MA. Day phone: 946-1000, ext. 2497. Eve phone: 747-0963.

Thank you again for your help in this matter. If it appears that this request might take longer than two weeks to fulfill, please let our committee know when we might expect a response.

Sincerely,

Plymouth Nuclear Matters Comm.
DCD

cc: Plymouth Selectmen
Plmouth Nuclear Matters Committee Members

QUESTION 1:
Certain actions are required to open the outboard containment valve AO-5025. Could you indicate where the fuse installation occurs to enable power to the DC solenoid? Also, who has possession of the key for the remote manual switch which opens valve AO-5025?

QUESTION 2:
Certain actions are required to open the inboard containment valve AO-5042B after the automatic containment high pressure trip point has been achieved. Could you describe the manual installation procedure for the hard wire jumper? Does this action occur behind the panel in the control room, or out in the plant?

QUESTION 3:
The earlier design for the DTVS also had an automatic reclosure of the vent if a high radiation level in the torus was achieved (1). This is now deleted from the current design (4). Could you indicate why this safety element of the design was eliminated?

QUESTION 4:
The rupture disk in the vent line is specified for 30 psi(3). Yet the containment design pressure is approximately 60 psi and ultimate rupture pressure of the containment is approximately 120 psi. Could you explain why the DTVS is intended to operate at such a low pressure?

QUESTION 5:
Are there design basis accidents for which it is calculated that the torus pressure could exceed 30 psi?

QUESTION 6:
In early correspondence with the NRC, BECO indicated that information on procedural changes associated with the physical plant modification for the DTVS would be provided(1). Later correspondence is silent on this matter. Have procedures controlling the use of the DTVS been completed, reviewed and approved by BECO? Have those procedures been reviewed or approved by the NRC? How many and who of the PNPS personnel have been trained and have formally signed off on the procedures? Can a copy of the procedures be made available to our committee?

QUESTION 7:
During the March 7, 1988 tour of PNPS by Mr. Russell, Dr. Murley, and Dr. Thadani, BECO responded to the questions posed by the NRC in their "Initial Assessment of Pilgrim SEP." In that presentation, BECO stressed that the declaration of General Emergency and recommendations for protective actions will be issued by BECO early in events which may lead to containment venting(3). Does BECO have

approved guidelines and procedures in effect for recommending evacuation of the EPZ in events which may lead to containment venting? Have the people in the EPZ who are charged to draft emergency action plans been briefed on the DTVS and the impact of such early notification and potential evocuation?

QUESTION 8:
Has any similar venting system been installed and made operational at any other GE Mark I, II, or III facility in the U.S. or elsewhere? Did Vermont Yankee proceed with a DTVS? Are there DTVS outside the U.S. which vent through carbon or gravel beds, resulting in a ground level release on utility property? Are there any DTVS operational which vent through a stack, resulting in an aerial dispersion with potentially greater geographic contamination? What are the pros and cons of either arrangement?

QUESTION 9:
We request clarification of BECO's actions in light of the NRC's stated positions on the DTVS. In the NRC's initial assessment of the Pilgrim Safety Enhancement Program, the NRC was not prepared to endorse the use of the DTVS (2). Further, the NRC stated that the installation of an additional branch line and containment isolation valve would require a change to the plant Technical Specifications(2). Thus the NRC concluded that the installation of the DTVS could not be implemented under the provisions of 10 CFR 50.59 (2). However, the additional branch line and the new outboard containment isolation valve AO-5025 have been installed. BECO claims that NRC approval is not required because, first, containment venting has been previously approved in the Boiling Water Reactor Owners Group Emergency Operating Guidelines, and second, valve AO-5025 meets the NRC requirements for a sealed closed isolation valve as defined in NUREG 0800 SRP 6.2.4 (3). Could you provide documentation from the NRC which indicates their concurrance that the DTVS can be implemented and that such action does not require a change to the Plant Technical Specifications?

QUESTION 10:
One of the arguements for the DTVS, that the system offers "significant improvements relative to existing vent capability (3)", comes as a surprise to observers who were not aware that plans for contaiment venting during severe accidents had been previously developed. Could you describe the containment venting procedures which existed before the implementation of the DTVS, and how the DTVS offers a significant improvement to that system? Had these prior plans ever been approved by the NRC?

Supporting backup documentation which exists would be helpful to our committee, such as:

- Logic Diagrams
- P & ID's
- Elec. One-Line Diagrams
- System Descriptions
- UFSAR/Tech Specs
- Relevant Procedures

Thank you for your consideration of this request.

Notes:

1) R.G. Bird, Senior Vice President--Nuclear, BECO, Letter dated July 8,1987 to S.A. Varga, Director Div. of Reactor Projects I/II, NRC, "Information Regarding Pilgrim Station Safety Enhancement Program"

2) S.A. Varga, Letter dated August 21, 1987 to R.G. Bird "Initial Assessment of Pilgrim Safety Enhancement Program"

3) S.J. Collins, Deputy Director, Division of Reactor Projects, NRC, Letter dated May 31, 1988 to R.G. Bird "NRC Region I Inspection Report #50-293/88-12"

4) R.G. Bird, Letter dated August 18,1988 to US NRC, Document Control Desk, "Revised Information Regarding Pilgrim Station Safety Enhancement Program"

TOWN OF PLYMOUTH

11 Lincoln Street
Plymouth, Massachusetts 02360
(508) 747-1620

September 8, 1989

To: Plymouth Selectmen
cc: Members of the Nuclear Safety and
Health Technical Support Group
Thomas Murley, NRC
Charlie Marshall, Pilgrim Resident Inspector

From: Plymouth Nuclear Matters Committee

This report is a translation, summary, and critique of the 100+ page Augmented Inspection Team report of the April 12, 1989 accident in the Reactor Core Coolant (RCIC) system at Pilgrim. We hope that these pages elicit a wider public discussion of the accident and provide access to technical information for those unable to study the larger report.

We conclude that this accident was more significant than indicated by the AIT report. Further, that certain aspects of the AIT conclusions were incorrect, the technical analysis was faulty, and the cover letter and executive summary did not reflect the serious nature of the accident as described in the body of the report.

In our review of the available documentation describing recent problems at Pilgrim, the April 12 accident is by far the most serious. Indeed, the number, variety, and degree of errors and malfunctions which occurred could, under probable alternative situations, have caused a far more serious accident, endangering the health and safety of the public.

At a minimum, we are requesting that those authorities who are responsible for protecting public safety and regulating the nuclear industry at the town, state and national level study this accident and strongly request that the NRC convene an Incident Investigation Team. This higher level inspection team will not only review the details of the accident, but also, from a broader perspective, assess the influence of the regulatory process on the cause or the course of the event.

One of our concerns has already been realized when NRC commissioner Zech responded to Alba Thompson's letter of July 13, 1989, stating, "(the event) was evaluated by the AIT to be of minor safety significance with minimal effect on plant equipment". These conclusions by the NRC must be challenged, for they are not supported by the facts of their own investigation.

Our additional concern is that now that the enforcement action has been issued, the matter will be shelved; the scrutiny of both specific and generic concerns will cease and necessary corrective actions will not occur.

An annotated version of our report is available for those who wish to study in greater depth the full AIT report. Should further clarification be desired, please write the committee care of the Town, or contact committee member David Dixon and 508-946-1000, ext. 2497, during the day.

Thank you,

Gerald S. Hayes Chm.

Plymouth Nuclear Matters Committee

SUMMARY AND COMMENTS ON THE APRIL 12, 1989 ACCIDENT AT THE PILGRIM NUCLEAR POWER STATION PLYMOUTH, MASSACHUSETTS

Introduction: On April 12, 1989, during a test of the Reactor Core Isolation Cooling (RCIC) system, radioactive high pressure water backed up into low pressure piping systems, causing damage to equipment and the release of radioactive water and steam into the RCIC Area and the Residual Heat Recovery Area B (RHR-B). The accident was caused by an unanticipated combination of errors by several different people, errors in approved procedures, and by faulty valve maintenance.

It was an event which could have caused an "Interfacing Systems Loss of Coolant Accident (LOCA)" a scenario where the cooling water leaks out of the reactor. This type of LOCA is particularly dangerous because the containment is bypassed. Boston Edison reacted very well to this event and the NRC took keen interest, dispatching an Augmented Inspection Team (AIT) to study the accident. The types of problems which occurred could have, under credible alternative conditions, caused far more serious consequences.

Yet BECO concluded, and the NRC concurred, that the accident was not even an "Unusual Event," a classification which indicates merely that the level of safety at the plant had been degraded. More disturbing, the AIT concluded the accident was not a significant precursor to an Interfacing Systems LOCA.

There are many disturbing aspects to the April 12, 1989 accident and the subsequent NRC report. The purpose of this summary is to evaluate the accident, and translate the AIT report.

II. What is the Logic System Functional Test (LSFT) for the Reactor Core Isolation Cooling (RCIC) system, which was being performed when the accident occurred?

The RCIC is a safety system which provides another means of supplying cooling water to the core during certain accidents. It backs-up the High Pressure Coolant Injection (HPCI) system, serving a similar function. However, the RCIC is not taken credit for in the safety analysis of design basis accidents, so it is not considered an Engineered Safety Feature. The purpose of the LSFT is to demonstrate that the RCIC pump shuts off if the reactor water level gets too high, but automatically restarts when the reactor water level drops to a preset low level. The RCIC LSFT (procedure 8.M.2-2.10.11.1) is done every six months, as per the Technical Specifications (TS).

III. What happened before and during the accident on April 12, 1989?

Prior to the accident, this was supposed to happen:	Prior to the accident, this happened:
All involved personnel were to have been briefed	This was not done.

on this infrequently done procedure.	
The control room operator sets the position of eight valves, actually changing two of them from closed to open.	Evidently done correctly The report does not indicate problems.
An operator was to position the circuit breakers for the D.C. power to seven motor operated RCIC valves, either on or off, and put tags on the circuit breaker handles.	Of the seven, six were positioned wrong, and one valve, not even part of the test, was turned off because of a typo in the procedure(1301-17).
A second operator is to check the work of the first operator.	The two operators had done the tagging together, and apparently did not check each others work.
The Instrument and Control technician, who was running the test, was to review/inspect and accept the tagging, and sign the tagout sheet.	The I&C technician signed the sheet.
The control room operator and the I&C technician should have seen from the graphics panels in the control room that the valves were not set up correctly for the test.	They did not observe the problems indicated on the graphics panels.

The logic test was then begun, involving the lead I&C technician, the control room operator and two other I&C technicians at a control panel in another part of the plant. During the test, a restart of the RCIC is simulated, but with power to the RCIC pump blocked off. But since the RCIC pump discharge valves 1301-49 and 1301-50 still had power to their actuators (incorrectly), they opened. Since the upstream side of those valves was not pressurized, water backed up into the RCIC pump and the RCIC pump low pressure suction piping. Check valve 1301-50 is supposed to close, prohibiting flow in the upstream direction, but it could not, because it had been previously temporarily repaired with an injected material (Furmanite), and when the valve had been subsequently later overhauled, some of the Furmanite was left on the valve stem, prohibiting its closure. Hot, high pressure water thus backed up in the system damaging some instrumentation, opening a relief valve, and causing thermal and pressure shock to the RCIC system. The relief valve spewed radioactive water and steam into the RCIC area, and since the floor drains are connected, the Residual Heat Recovery (RHR-B) area was also contaminated with radioactive water and steam.

IV. What went wrong?

The personnel did not follow procedures for the RCIC LSFT and did not follow procedures for positioning the valves or for tagging circuit breakers.

The control room operators did not recognize nor correct the problems shown on the system graphics panels. Ultimate responsiblity lies with these senior individuals, who failed, in this instance, to perform their duties.

The knowledge available from a similar 1983 accident in the High Pressure Coolant Injection system was not incorporated into plant documentation.

Even though the written, approved LSFT procedure contained a critical error, somehow the error had gone undetected during previous, supposedly uneventful LSFT's.

The control of the Furmaniting procedure was poor, as was the subsequent check valve overhaul.

When valve 1301-17 was tagged out in the open position, the plant lost redundant containment isolation, in violation of the Technical Specifications. This in itself requires notification under 10CFR 50.73 and possibly the declaration of an Unusual Event.

It is unclear to both BECO and NRC whether or not the check valve 1301-50 is a containment isolation valve, and if so, that it should be leak tested as such. The AIT tabled this issue to "future FSAR revisions".

It may have been discovered that the leak testing procedures for the check valve 1301-50, and perhaps for other check valves at Pilgrim and elsewhere, do not indicate the valves actual leakage when installed. Further study is pending.

V. How did BECO respond?

The Augmented Inspection Team's report indicates that the BECO immediate response was appropriate and timely. Specifically, the radiological protection organization's response to the event was prompt, efficient, and thorough. Eleven people were slightly contaminated.

After the event, BECO formed three investigative teams, led by an oversight group: a team to evaluate the effects on the RCIC system, a team to detail the accident, and a peer review.

VI. How did the NRC respond?

The NRC's William Russell, Region 1 administrator, initiated the Augmented Inspection Team on April 13, 1989. Their report was published May 8, 1989. The AIT is NRC's second level events investigation, the first level being an Incident Investigation Team (IIT). It should be noted that the convening of an AIT for an event deemed by the licensee to be less significant than the lowest level Emergency Action Level --"Unusual Event" -- possibly indicates that the NRC felt that the event might have been more serious.

Perhaps the reason the NRC took great interest in the event, was the possibility that this accident involved an

Interfacing Systems Loss of Coolant, or was a significant precursor event to an Interfacing Systems Loss of Coolant Accident (Intersystems LOCA)

Criteria which exist within the NRC for the determination of a significant event are:

1. Event sequence not previously analyzed or could be far more serious with credible alternative conditions.

2. System interaction resulting from a previously unrecognized interdependence of systems and components.

3. Improper operation, maintenance, or design that has or could cause common cause/common mode failure of a safety system.

4. Unexpected system or component performance with serious safety implications or radiation release.

5. Multiple failures (including personnel errors) occurred in the event.

6. Equipment failures (particularly non-safety equipment) that caused serious transients and challenges to safety system.

A case can be made that all of these conditions were met. The April 12, 1989 accident at Pilgrim was very significant.

If the AIT report is studied closely, other problems are uncovered which are not discussed in the cover letters, executive summaries, the Licensee Event Report, or the news summaries. First, it is not known for certain when the event terminated, or when the the release stopped. Second, it is not known how much water backed up past the check valve 1301-50. Third, it is not known what pressure was seen by the RCIC pump or suction piping. Specifically, the logic used to arrive at a figure of 400 psi was incorrect. The fact that the pressure switch 1360-21 was not ruptured does not indicate that the pressure remained below 500 psi. Rupture of the switch can occur in a range of 900 to 2000 psi, and is a very unreliable indicator of what pressure actually occurred. Fourth, since the duration of the release is unknown and the pressure of the piping is unknown, the amount of water released by the relief valve is also unknown. The "approximately 100 gallons" referred to in the AIT report is optimistic guessing.

VII. What is an Interfacing System LOCA?

In NRC's words, "Recent BWR operating experience indicates that the pressure isolation valves may not adequately protect against overpressurization of low pressure systems. This overpressurization may result in the rupture of low pressure piping. This event, if combined with failures in the emergency core cooling systems (ECCS) and other systems (eg. feedwater) that may be used to provide makeup to the reactor coolant system, could result in a core melt accident with the possible release of fission products outside the primary containment. Some ECCS failures may be the direct result of the initial rupture and/or its environmental effects."

This type of accident, it should be emphasized, is very critical because it bypasses the containment and it bypasses emergency preparedness. It is a "hot" topic in NRC circles (see attachments). Two recent Interfacing Systems LOCA precursors have been scrutinized: a January 20, 1989 accident at Arkansas Nuclear One Unit 1, and a March 8, 1989 accident at Vogtle Unit 2.

Further, the NRC recently issued, "Interfacing Systems LOCA: Boiling Water Reactor" as NUREG 5124 (see attachment). This report is mentioned in the AIT report, but it is unclear whether the AIT report is accurate. The AIT report indicates that BECO complies with the recommendations of NUREG 5124. However, BECO's Technical Specifications require an RCIC LSFT every six months and one of NUREG 5124's main conclusions is to perform this test only at shutdown, when the reactor is depressurized, in order to reduce the chances of an Interfacing Systems LOCA. For BECo to comply with NUREG 5124, a change to the Technical Specifications would be required.

VIII. Was the April 12, 1989 accident a potential precursor to an Interfacing Systems LOCA ?

The AIT report argues that there were several isolable barriers in place to avoid an intersystems loss of coolant: the check valve 6-58A, the check valve 1301-50, and the two block valves 1301-49 and 1301-48. First, check valve 6-58A is a "feedwater check valve" which is known for frequent failures. In particular, leakage test results for Valve 6-58A are very poor. And based on past history, if a leakage test were done today, it is likely that it would fail. Next, relying on 1301-50 is questionable because it is not certain that the valve ever closed during the accident. Finally, valves 1301-48 and 1301-49 were involved in multiple personnel and administrative errors: they were incorrectly described in the LSFT procedure, they were incorrectly tagged, improperly verified, and not observed properly in the reactor control room. To base the analysis on the adequacy of these valves, is overly optimistic.

The AIT used a variety of narrow criteria to avoid concluding that this was a Interfacing Systems LOCA. Yet the NRC has recently said that that type of analysis is not proper and does not help achieve the goal of reducing the vulnerability of nuclear power plants to Interfacing System Loss of Coolant Accidents.

IX. Could it have been worse?

There are many credible alternative conditions which would have made this event much, much worse:

--The plant could have been operating at a higher power level.

--Check valve 6-58A might not have been able to prevent backflow.

--Check valve 1301-50 might have stuck 40 or 60 degrees

off its seat, rather than the assumed 15 degrees.
--the operators might not have concluded that the correct action to take was to close valves 48 and 49. After all, these valves were supposed to have been closed, tagged, with power removed from the motor operators, making them inoperable from the reactor control room.
--the low pressure piping could have ruptured.
--the steam release could have degraded the environment at both the RCIC and the RHR-B to the point where these systems would not be available to help maintain adequate coolant level in the reactor core.

The AIT report did not include an evaluation of the potential consequences of credible alternative conditions, an important step in a well executed analysis of this potentially disastrous event. It is not known why. The analysis by the AIT did not even share the concern evidenced by BECO's conclusion that, "...the errors and programmatic deficiencies noted could have caused significantly greater problems under other circumstances. Therefore, this event should continue to be treated as significant."

X. Several things need to happen to resolve the issues raised by this accident:

The check valves 1301-50 and 6-58A should be leak tested.

The RCIC LSFT should be redone (procedure 8.M.2-2.10.11.1)

The RCIC damage evaluation should be closely reviewed by independent technical experts.

The design problem concerning the placement of the check valves and block valves should be resolved.

Analyze the NRC enforcement actions for appropriateness.

Resolve the classification problems, and associated testing requirements for the check valve 1301-50.

Review BECO's compliance with NUREG 5124, and change the Technical Specifications as required.

Convene the higher level NRC events investigation, the Incident Investigation Team (IIT). The difference from this and the AIT is that the IIT is broader in scope, and includes an evaluation of the influence of the regulatory process on the accident. This serious request is made necessary by the type and degree of error in the AIT analysis, the continual problems which are occurring at Pilgrim during the ongoing power ascension program, the wider implications of the root causes of this accident for management of the facility, and the closer scrutiny required by a facility which is one of the worst in the nation, by several objective measures.

XI. Conclusions

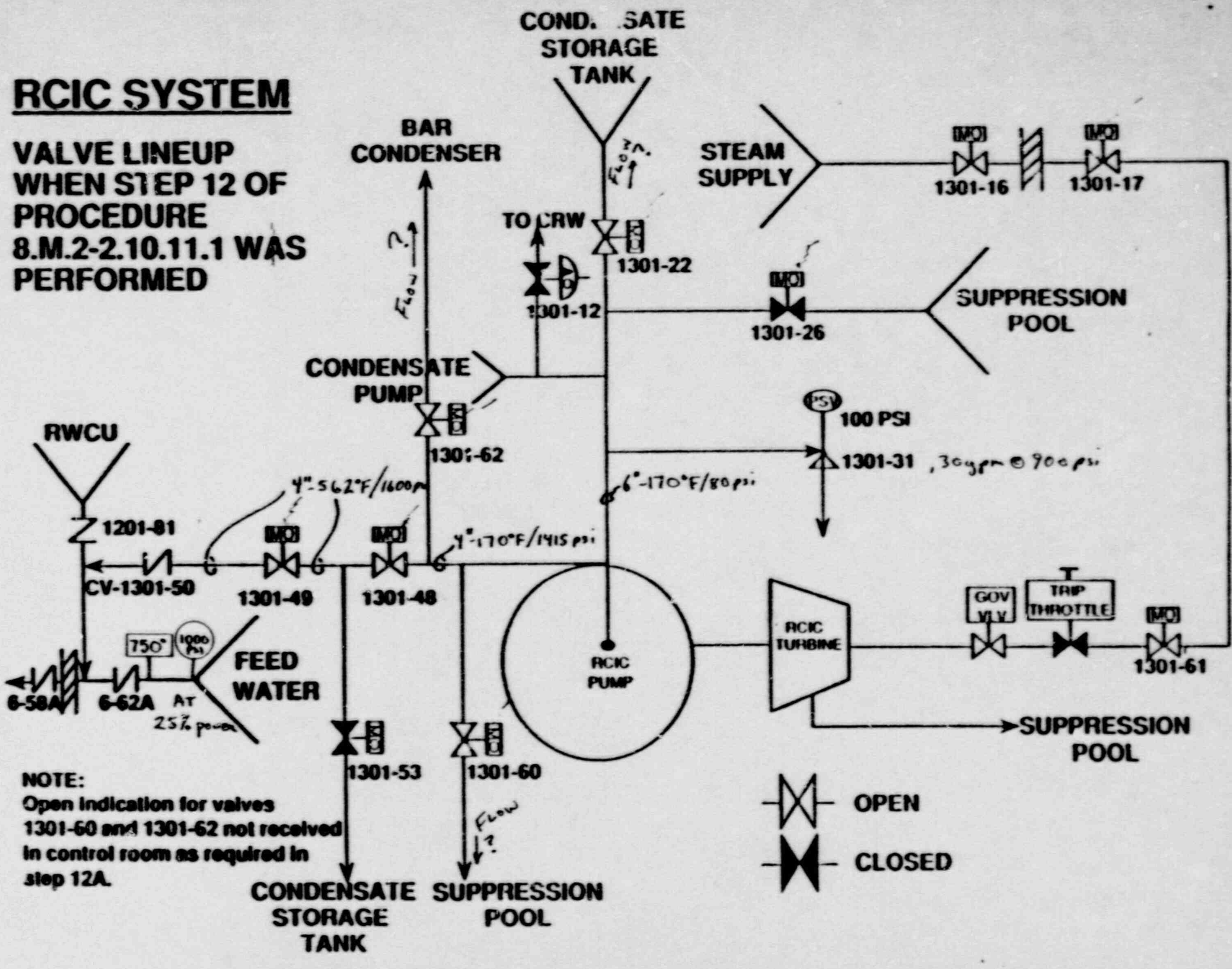
The April 12, 1989 accident at Pilgrim has serious implications which were not thoroughly evaluated nor objectively reported in the NRC's Augmented Inspection Team report. It could have been much worse.

The character and number of causes of this accident may be unprecedented and are deeply disturbing. Further NRC investigation is warranted. Independant assessment of certain technical aspects is also warranted.

Furthermore, when this accident is viewed in light of the other problems which are occurring during the power ascension, the SCRAMS, the maintenance and design problems, the unresolved valve actuations, the equipment failures, the personnel errors, etc., it seems prudent to question whether the intense pressure to get Pilgrim back on line is contributing to an unsafe situation with potentially disastrous consequences.

RCIC SYSTEM

VALVE LINEUP WHEN STEP 12 OF PROCEDURE 8.M.2-2.10.11.1 WAS PERFORMED



NOTE:
Open indication for valves 1301-60 and 1301-62 not received in control room as required in step 12A.

NUREG/CR-5124
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Interfacing Systems LOCA: Boiling Water Reactors

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MURLEY LAUNCHES STUDY OF RISK OF INTERFACING SYSTEMS LOCAs

Thomas Murley, director of NRC's Office of Nuclear Reactor Regulation, has launched a program to confirm that probabilistic risk assessments (PRAs) accurately reflect the low probability that an interfacing systems loss-of-coolant accident (LOCA) will lead to severe core melt accidents with significant off-site releases.

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Murley initiated the effort because he is skeptical of PRA data that uniformly shows the chances of severe core melts prompted by interfacing systems LOCAs are low. "I have to be frank," Murley told the Advisory Committee on Reactor Safeguards (ACRS) April 6. "I am not believing the numbers. The numbers are telling us that...intersystem LOCA is not a problem. I shouldn't say I don't believe it. I say I'm skeptical, so we're going to start taking some actions."

The concept of an interfacing system LOCA—an accident sequence beyond design basis—was first identified in the 1975 Wash-1400 reactor safety study, and was labeled PWR sequence V.

Murley said precursor events to the intersystem LOCA scenario—including the 1987 event at the West German Biblis-A PWR—concerned him and prompted him to initiate the NRC review program (INRC, 5 Dec. '88, 1). Under the sequence, failure of the check valves separating the primary circuit from the low-pressure injection system portion of the emergency core cooling system could result in a LOCA that suddenly discharges into the low-pressure system and bypasses containment.

The NRC initiative was only about "a week old" when Murley addressed the ACRS, and he said he doesn't anticipate requiring any specific industry initiatives at this time.

"The goal is to have high confidence—and I stress that high confidence—that the probability of an interfacing systems LOCA, which could lead to an unisolable LOCA outside containment, is less than ten-to-the-minus-six per year for each plant in the U.S.," Murley said. Murley added that NRC hopes to wrap up the review in about a year, and that the agency may, depending on the outcome of the review, recommend changes to industry training programs to see whether they can be improved so that operators will be "sensitized" to the significance of the long-postulated accident sequence.

At reactors that have experienced precursors to the sequence V event, "the operators...didn't know how close they had come or what the ramifications were of the situation, so we think the whole industry as well as NRC has to be sensitized," Murley said.

"This sequence is important in my judgment because it bypasses the containment and it bypasses emergency preparedness," Murley said in defending his decision to move forward with the initiative. "It effectively bypasses two levels of our defense-in-depth safety philosophy under the worst circumstances," Murley said. "The worst circumstances (are) that you have a break out in the RHR (residual heat removal) system which then causes you to not only lose coolant but to lose all your safety injection capability, and which ultimately then leads to core damage and core meltdown to an open containment.

"That goes straight to the atmosphere and it can happen in a short time," he added. "The worst time calculations that I've seen can lead to core uncoverage in a half hour, core damage in 45 minutes, and off-site doses in the 100 rem range in an hour or hour-and-a-half. So it's the importance of that sequence that caused me to consider taking another look at it. I have no evidence that the probability of it happening is higher than what is said in the PRAs, (but) I'm starting to see these precursors, so rather than take the PRA results at face value, I'm going to be a little skeptical, just because of this sequence and its consequences."

Murley rejected suggestions by ACRS members that the sequence V scenario be considered as part of the Individual Plant Examinations (IPEs) that NRC has required of all U.S. nuclear facilities.

"I think it's just going to overburden IPE," he said. "IPE was never meant to be the vehicle to resolve all issues associated with severe accidents. If we were to ask licensees to look at event V as part of their IPEs, three years from now we would get back something that I almost guarantee wouldn't be worth anything. I don't think they have the methodology (that) would be good enough (so) that I would be satisfied and I also don't want to wait for three to five years."

Last year, when details of the 1987 Biblis event surfaced, Murley said the agency was considering the need for further guidance on the issue.—*Dave Airozo, Washington*

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EXECUTIVE SUMMARY

This study was performed by the Risk Evaluation Group, Department of Nuclear Energy, Brookhaven National Laboratory for the Office of Nuclear Regulatory Research, Reactor and Plant Safety Issues Branch, Division of Reactor and Plant Systems, U.S. Nuclear Regulatory Commission. The objectives of this study are to investigate the vulnerability of current boiling water reactor (BWR) designs to an interfacing systems LOCA (ISL), identify any improvements that would significantly reduce the frequency of ISLs, determine the cost-benefit considerations thereof, and determine the effects and the cost benefit relationship of instituting leak testing programs of the pressure isolation valves for those plants that do not currently have such a requirement.

This study is based upon the detailed examination of three plants (Peach Bottom, Nine Mile Point 2, and Quad Cities) with the goal of taking the plant-specific findings and extrapolating the results to aid in the resolution of NRC Generic Issue 105.

Recent BWR operating experience indicates that the pressure isolation valves may not adequately protect against overpressurization of low pressure systems. This overpressurization may result in the rupture of low pressure piping. This event, if combined with failures in the emergency core cooling systems (ECCS) and other systems (e.g. feedwater) that may be used to provide makeup to the reactor coolant system, could result in a core melt accident with the possible release of fission products outside the primary containment. Some ECCS failures may be a direct result of the initial rupture and/or its environmental effects.

One of the primary goals of this study was to determine the cost-benefit relationship associated with requiring plants that do not currently have leak testing requirements on their pressure isolation valves (PIVs) to institute such a program. However, all of the reference plants already have various requirements related to leak testing. Therefore it was decided that since none of the reference plants represented a true "base case" model in this area an additional base case model would have to be created. The base case model was taken to be the Peach Bottom model with the PIV leak testing aspects removed. Removing the leak testing benefits from the Peach Bottom model resulted in a large increase in predicted core damage frequency due to ISL. Based upon the results of a separate sensitivity study, it appears sufficient for the leak testing program to include provisions such that leak testing be performed at each refueling as well as after individual valve maintenance. The risk-based benefits calculated for this leak testing program show that such testing schemes are cost effective.

In addition, the offsite risk-based cost-benefit considerations for the suggested testing program were calculated to be fully cost effective whether or not the break in the low pressure system was assumed to be submerged under water. A submerged break would result in trapping of some of the aerosol fission products in the water and thus lower the predicted offsite consequences. The results indicate that in spite of uncertainty in predicting fission product release the benefits in risk reduction outweigh the cost of implementing such a leak testing program.

The insights from this study fall into two basic categories. The first category deals with assuring that the pressure boundaries are intact prior to increasing reactor pressure and the second category deals with how to avoid placing the plant unnecessarily into a more vulnerable mode of plant operation. Table 1 provides a convenient collection of the pertinent core damage frequencies (CDFs) presented throughout this report. The table will be used to facilitate comparisons and derive insights.

The first category above is addressed by PIV leak testing provisions. From Table 1, "Peach Bottom (no leak testing)" represents an analysis wherein the Peach Bottom model was stripped of all credit for its current leak testing practices. "Peach Bottom (current)" refers to the Peach Bottom plant as found and modelled. "Peach Bottom (with leak testing)" reflects the minimum leak testing provisions derived from this study (i.e. leak testing all air-operated check valves at each refueling and individually after maintenance). Comparing the "no-testing case" to "Peach Bottom (current)" shows that the existing level of leak testing has already reduced the Peach Bottom CDF due to ISLs by an order of magnitude. Comparing "Peach Bottom (current)" to "Peach Bottom (with leak testing)" shows another order of magnitude reduction is still available. A significant benefit (similar to that derived for Peach Bottom) for such a leak testing program is expected to hold across the BWR population.

The second category of insights is addressed by changing current testing practices. These testing practices can be almost as significant as implementation of a leak testing program, however, they are quite plant-specific. The dominant example from this study is found at Nine Mile Point 2 (NMP). By comparing the two NMP-2 entries in Table 1, there is apparently more than a two order of magnitude decrease in the CDF for ISL available by prohibiting the currently allowed practice of stroke testing the valves in the steam condensing lines to the RHR heat exchangers (with the reactor pressurized) and allowing the stroke testing to await a convenient shutdown (with the reactor depressurized).

A second example of significant testing-induced risk can be seen by comparing "Peach Bottom (current)" with "Peach Bottom (logic test at shutdown)" from Table 1. This is the single most effective corrective action identified for the Peach Bottom plant in reducing core damage frequency. Current Peach Bottom testing requirements include the provision to test the ECCS logic every six months independent of whether or not the reactor is pressurized. By holding off on the ECCS logic system functional test until a reactor shutdown comes along, (i.e., the reactor is depressurized), the ISL CDF can be reduced by almost an order of magnitude.

In summary, the results of this study show that institution of a minimum leak testing program for the air-operated pressure isolation check valves represents a significant reduction in the estimated ISL CDF for the three plants studied, which should apply across the entire BWR population. In addition, it has been shown that some of the current BWR testing practices can also represent a large contribution to ISL CDF and that this testing-induced risk is easily removed by rather simple and cost-effective changes to existing testing procedures (as discussed directly above).

Table 1
Summary of Estimated ISL CDF vs. Plant States

Plant State	CDF/Year
Peach Bottom (No leak testing)	1.86E-5
Peach Bottom (Current)	1.02E-6
Peach Bottom (With leak testing)	1.97E-7
Nine Mile Point 2 (Current)	8.81E-6
Nine Mile Point 2 (With all fixes)	3.22E-8
Peach Bottom (Logic test at shutdown)	1.21E-7