Pilgrim Watch
June 1, 2011
EXHIBIT 1

VOLUME 1

PILGRIM NUCLEAR POWER STATION

INDIVIDUAL PLANT EXAMINATION FOR INTERNAL EVENTS PER GL-88-20

PREPARED BY BOSTON
EDISON COMPANY
BRAINTREE, MA
SEPTEMBER 1992

Pilgrim Nuclear Power Station

Individual Plant Examination

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Revision 0 - September 1992

purposes of the Pilgrim PR.A, no credit was given for RWCU operation for heat removal.

Containment Venting

One of the options available to the operators for contr9lling containment pressure is to vent the containment. Pilgrim has a venting system which is capable of operation at pressures up to the containment design pressure of 56 psig. Containment venting is initiated in accordance with emergency procedures which require the operator to maintain the containment below the Primary Containment Pressure Limit. Keeping containment pressure below this limit permits continued functioning of equipment inside containment (such as SRVs) and maintains the structural integrity of the containment.

Direct Torus Vent

The direct torus containment vent system is a system of last resort to prevent containment pressure from rising above the 56 psig design pressure. All other forms of DHR would need to have failed or be insufficient to remove decay heat before the DTV would be required. Required support systems include DC power and nitrogen accumulators. Use of the vent is initiated by actuating a key lock switch which causes valves to align in a manner which bypasses the SBGT system vent path and utilizes a hardened vent. EOPs instruct the operator to maintain and control containment venting with either the SBGTS or DTV to limit the rate of steam release and therefore prevent net positive suction head (NPSH) problems in the suppression pool. The DTV also directs any steam· re-lease outside the reactor building, limiting the environmental conditions in the secondary containment.

performed a limited scope IPE using the IDCOR methodology to assist in SEP decision making. A number of the SEP modifications dealt directly with the DHR issue. A summary of important modifications follows:

1. Direct Torus Vent-The direct torus vent was installed as a decay heat removal system to augment the existing decay heat removal capacity. In situations where the main condenser is unavailable and decay heat cannot be removed by the RHR heat exchangers, the only venue for heat removal is by direct release of steam from the containment. Before the installation of the direct torus vent, if the containment was vented at a rate which would remove significant decay heat, the high pressure vapor released into the vent path would have ruptured the ductwork, and resulted in unfavorable conditions in the reactor building.

2. The hard piped vent was installed in the torus ventilation piping, between the inboard and outboard isolation valves. This allows the operators to release significant amounts of decay heat from the containment atmosphere over a broad spectrum of events. This system provides a diverse mode of decay heat removal which does not rely on the salt service water system as its ultimate heat sink.

3. Containment Spray Flow Reduction and Fire Water Crosstie- Another SEP modification was the blocking six of the seven spray nozzles. This modification was shown by analysis to provide better control of the depressurization process, while not compromising the effectiveness of the system. The principal benefit was to permit the use of drywell sprays over a broader range of containment temperature and pressure conditions. Another key benefit of reducing drywell spray flow was to allow the use of the fire water crosstie as a water source. Lowering the drywell spray flow permits the use of the lower capacity fire pumps, not only by providing an alternate water source, but also a source which does not rely on AC power. This allows the use of sprays during Station Blackout sequences.

In general, no additional modifications were apparent that would both be cost effective and result in a significant reduction in risk. It appears that most, if not all, of the most important event failures in Class II cutsets could be handled by operator recovery action.

function arrests an event either in vessel or ex-vessel with an intact containment. This function is not needed if containment flooding from external sources is occurring due to the extensive mass addition, i.e., if there is success using external sources at the DS or VL headings.

The system credited with this function is suppression pool cooling. Similar to drywell sprays, only one RHR pump, one RHR Heat Exchanger, and one train .of suppression pool cooiing valves are sufficient for success.

4.1.2.8 Torus Venting

The Direct Torus Vent was credited as a potential heat removal system during core damage sequences with debris either in-vessel or ex-vessel in the Class I and III sequences. The system is designed as a hard piped system, and takes advantage of the scrubbing action of the suppression pool water to reduce the amount of fission products released. Fission products liberated from the damaged core into the drywell are forced down the vent pipes, into the vent header, down the downcomers, and through the torus water.

The design of the system allows for operation at high containment pressures, but because of the hard piped design, this mode -of containment venting prevents release of primary containment atmosphere to the secondary containment, minimizing the impact of venting on the availability of systems located in the reactor building.

whether or not drywell sprays are determined to bE? available since the drywell spray operating limits· in the EOP's may instruct the operator not to use drywell sprays under certain conditions; this ·would not prevent the operation of injection.

4.3.2.4.9 Containment Heat Removal

Successful containment heat removal ensures that the containment pressure will be maintained below the containment capacity (in the absence of large quantities of non-condensible gases produced from debris/concrete attack). Following RPV failure containment heat removal.is accomplished with a RHR heat exchanger operating either in the pool cooling mode or in the drywell spray mode. Containment heat removal is branched for all sequences where containment failure has not already been determined to have occurred (the "FAILED" branch under Heading "Containment Failed Prior Core Damage" or the "NO VAP SUP" branch under Heading "Vapor Suppression").

4.3.2.4.10 Containment Venting Available

Containment venting is accomplished with· the "normalvent" and· the wetwell "direct torus·vent". Venting is initiated prior to the containment pressure exceeding 56 psig. Venting is branched for all sequences where containment failure has not already been determined to have occurred (the "FAILED" branch under Heading "Containment Failed Prior Core Damage" or the "NO VAP SUP" branchunder Heading "Vapor Suppression"). It is asked for all cases where CHR is asked. Even for situations where CHR is available, over-pressurization of the containment may occur if the debris is not cooled ex-vessel and significant quantities of non-condensible gases are produced.

between the two systems. The spool piece can be easily installed by two quick connect couplings. Once the connection between the two systems is established, the plant's diesel driven fire pump (P-140) will automatically start on low fire header pressure.

The fire water cross-tie was installed primarily for injection during an SBO event. Since its use is proceduralized, however, it can be used for low pressure injection under all accide t conditions in which the reactor has been depressurized, and the probability of loss of low pressure injection has been reduced for all accident sequences in the IPE.

Containment Pressure Control (Event Wl :

Direct torus vent: The use of the direct torus vent as a means of containment heat removal has been shown to have a major impact upon the results of the Class II accident sequences. Pilgrim installed the hard piped vent from the wetwell air space as part of the SEP program. Because the hard piped vent is designed to operate independent of AC power and instrument air sources, it is available as a containment heat removal system for a wide spectrum of events. Although not explicitly considered in the quan ification, the hard piped vent directs the steam·in the containment' atmosphere to the stack as opposed to the reactor building, extending the time for repair and recovery of failed equipment, and reducing the potential reluctanc'e to initiate venting.

Containment Spray Flow Reduction: Another modification proven to be important in the IPE models for containment heat removal purposes was the reduction of the flow capacity of the drywell spray nozzles. This allowed for a more gradual depressurization of the drywell, permitting the use of drywell sprays over a broader range of containment temperature and pressure conditions. To reduce the

Required procedure 5.3.26 is a short procedure without check off. The basic HEP from Table 20-7, item 3, is .003.

The stress level for step C is considered moderately high, step by step. Additionally, the operators are considered to be skilled due to their training in this procedure.

Therefore, from Table 20-16, the performance shaping factor is x2. The total failure probability is .003 x 2 = .006.

SUMMARY

From Figure A1-2, the probability that the operator will fail to follow FWXT procedure 'is 8.33E-3.

MXXDTVOPRY - OPERATOR FAILS TO ALIGN DIRECT TORUS VENT

STEP A - OPERATOR RECOGNIZES NEED FOR DIRECT TORUS VENT

The operators would already be in EOP-3, Primary Containment Pressure Control, as a result of elevated containment pressure. It has been calculated that it would take several hours for containment pressure to rise from 2.5 ps1g (scram / alarm setpoint) to 30 psig at which point the DTV would be manually aligned.

The basic HEP for failure to recognize the need for DTV is taken from Table 20-3, item 5 ; BHEP = .0001. This is conservative as it assumes that recognition would be only 60 minutes after the first annunciation whereas it takes several hours.

The stress level for this step is assumed to be extremely high due to the impending challenge to the primary containment. The recognition process relies on the operator knowing to read containment pressure and to react at or before 60 psig. As only pressure is necessary for successful recognition, this process is deemed step by step. Additionally, the operators are considered to be skilled due to their training in this procedure.

Therefore, from table 20-16, item 6, the performance shaping factor for step a is times 5. The total failure rate is then 5 * .0001 = .0005.

STEP B - CONTROL ROOM RECOGNIZES THE NEED TO ALIGN NORMAL VENT

The control room shift includes a Shift Technical Advisor whose training is different from that of the licensed operators. Moderate dependence is assumed between the STA and the rest of the shift. Therefore, the probability of the STA failing to recognize period was chosen to encompass this time and to correspond to AC power recovery time periods reported in the literature. Finally, with successful operator action to shed loads (which is the most likely pathway), both the A and B batteries will deplete within 15 hours - thus, 15 hours was chosen as the upper limit for the third AC power recovery time period.

Failure to recover any source of AC power within 12 hours (no load shedding) or 15 hours (successful load shedding) is assumed to result in core damage. This is regardless of whether high pressure or low pressure systems were in operation. Failure of either battery results in containment heat removal failure (the direct torus vent requires both DC batteries for operation, other systems require AC power, and firewater system operation through the containment sprays would be terminated by procedure once the containment water level reached a specified level). As the containment pressure rises it will force the SRVs closed, thus resulting in increased primary system pressure and the inability to inject with low pressure systems even if they are operable. Depletion of the batteries results in loss of control power to the RCIC and HPCI systems, with subsequent failure of these high pressure injection systems as well. Thus, unless some source of AC power is restored so that DC batteries can be charged, core damage is assumed to begin within 15 hours.

If AC power is restored, containment heat removal and continued primary system coolant inventory maintenance can cqntinue.

5. The potential for stuck open safety/relief valves were specifically addressed using event trees shown in Figures C.2-6 and C.2-7. Figure C.2-6 is used to address the situation in which at least one diesel generator is in operation while there is a SORV. This figure is structurally similar to the event tree developed for SORVs resufting from other transients in which off-site power is available. The quantification in Figure C.2-6 takes into account the source of AC power, i.e., the diesel generator. Note that if off-site power is recovered within 2 hours use of the feedwater system for high pressure injection is possible; if off-site power is not recovered within 2 hours it is assumed that the feedwater system can not be utilized for high pressure injection even if power is restored later.

   Figure C.2-7 is used for the situation in which a station. blackout exists along with a SORV. Power recovery within 12 hours is assumed to be successful for some sequences;

the operator must inhibit closure of either circuit breaker CB-501 or CB-601 (the feed breakers associated with the SBO DG or the 23KV line and the emergency buses - these are different than the breakers associated with DG A and DG B). The operator can then start the SBO diesel generator and allow one of the breakers (the "uninhibited" breaker) to automatically close when it senses current from the diesel generator. Failure to inhibit automatic closure of one of the circuit breakers will result in a diesel generator overload condition; this is assumed to result in failure of the SBO DG. By procedure, the SBO diesel generator will be used only if one or both of the other two diesel generators has failed to start or is unavailable and the 23KV line is unavailable.

125VDC Bus (Battery) "A" This bus is required to start diesel generator 1. It also is a source of power for one of the high pressure injection sources (RCIC), and the ADS valves. This bus is required for operation of the direct torus vent.

125VDC Bus (Battery) "B" - This bus is required to start diesel generator 2. It also is a source of power for one of the high pressure injection sources (HPCI), and the ADS valves. This bus is also required for operation of the direct torus vent.

Figure C.2-2: Station Blackout (SBO)

This event tree starts with input from Figure C.2-1, namely, the cut sets in which all AC power sources are unavailable (either due to mechanical/electrical faults of the diesel generators or due to failures of support systems necessary to operate the diesel generators, such as DC batteries). Events C, M, and P are exactly as defined for Figure C.2-1.

I2 — OSP Recovered 0-2 Hours: Recovery of off-site power within 2 hours is assumed to be sufficient to allow for feedwater and main condenser restoration, if necessary. For this event it is assumed that the· 345kv source must be restored. Event sequences in which off-site power is successfully restored continue using Figure C.2-4.

DG2 — Dgs Recovered 0-2 Hours: Recovery of any of the three diesel generators, or the 23kv source, will allow for should not be adversely affected to maintain the primary system pressure below the design pressure.

Reactor Coolant Inventory Makeup, High Pressure:

Loss of one bus of DC power disables either the HPCI or RCIC system. Loss of the second bus would disable the remaining system. Therefore, quantification of this event depends upon the status of the second DC bus, as well as the determination of which system (HPCI or RCIC) is affected by the initial loss.

Regardless of which DC bus is lost, at least one pump of the feedwater system should remain available for high pressure inventory maintenance. As discussed previously, the loss of DC power operating procedures instruct the operators to trip the feedwater pumps associated with the DC division lost.

There are a number of support systems which must be available for continued feedwater operation. These include: (1) Motive and control power for operation of feedwater valves, feedwater pumps, and condensate pumps (these power sources are independent of DC power), (2) Adequate water makeup to the condenser hotwell from the CSTs, and (3) SSW/RBCCW for pump cooling.

Reactor Depressurization:

The failure of one DC bus results in a higher failure probability for reactor vessel depressurization due to less redundancy in the DC power system.

Reactor Coolant Inventory Makeup, Low Pressure:

Following successful reactor depressurization, the low pressure injection systems can supply adequate makeup to the reactor. Note that the failed DC bus which initiated this sequence will not be available for control of the corresponding division of low pressure pumps. The result is that the LPCI and CS failure rate is higher than for most other initiators. The condensate system is affected by the initiators as feedwater was, with the operator instructed to trip the condensate pumps on the buses affected by the loss of DC.

Containment Pressure Control:

The main condenser is assumed to be unavailable for containment heat removal if DC power is lost. The availability of RHR is highly dependent on the availability of DC power for pump breaker control. This is especially true for the sequences in which both DC buses are unavailable. The containment direct torus venting system would be unavailable if one DC division is unavailable, and one train of the normal vent would be affected.

Continued Reactor Coolant Inventory Makeup:

This event represents the ability to maintain coolant inventory in the vessel following the occurrence of unacceptable containment conditions. Containment pressure and temperature conditions must be maintained within acceptable limits if the integrity of the containment is to be maintained. Loss of DC power affects the ability to control containment conditions, which may in turn degrade the performance of systems being used to maintain coolant inventory.

Quantification of this event is dependent upon previous system failures, since the same failures which contribute to the failure to control containment conditions may also affect coolant inventory makeup.

C.4.3 Loss of Instrument Air/Nitrogen

Compressed gas systems fulfill a variety of functions at nuclear plants. Usually, the most important of these functions is to provide the motive force for operation of valves. At Pilgrim, some of these valves are safety related and some are located inside the containment, e.g., the SRVs and the MSIVs.

Two compressed gas systems are provided at the Pilgrim plant: one which provides nitrogen for use inside containment, and one which provides air for use outside containment. The Pilgrim plant has three redundant and independent nitrogen supply systems. Therefore, loss of nitrogen sequences need not be evaluated because the loss of nitrogen initiator frequency is very low. Several independent nitrogen supplies must fail and there is no credible common cause internal event that can eliminate all sources. However, an evaluation was carried out for loss of instrument air.

The following air-operated valves are affected by loss of air:

1. Outboard MSIVs: On loss of instrument air, the outboard MSIVs are assumed to close. Although an accumulator is attached to each MSIV which keeps the valve open on loss of air unless the MSIV receives a signal to close, it is assumed that the air eventually bleeds away and the valves close. The MSIVs cannot be reopened until instrument air is recovered.

2. Feedwater Regulating Valves: These valves fail "as is" and cannot be opened or closed from the control room until air is restored.

3. Scram Valves: On loss of air, the scram valves fail open causing drive water to force the pistons upward, inserting the control rods into the core.

4. Scram Discharge Volume Vent and Drain Valves: On loss of air, the scram vent and drain valves fail closed, preventing loss of reactor water discharged from all CRDs during and after a scram.

Based on the failure positions of the air-operated valves described above, failure of the instrument air system during power operation will initiate a scram and close the outboard MSIVs. Plant response for this event would initially be similar to an MSIV closure.

C.4.3.1 Event Tree for Loss of Instrument Air

This section provides a discussion of the loss of instrument air on the frontline systems of the Loss of Instrument Air event tree.

Reactivity Control:

The reliability of the scram system subsequent to a loss of instrument air is similar to that used in other accident sequences. This independence of the scram system from the loss of air initiating event is primarily due to the redundancy in the scram system which requires a common mode failure to prevent successful scram.

Primary System Pressure Control:

The SRVs are not affected by loss of instrument air since they are Nitrogen operated.

Reactor Coolant Inventory Makeup, High Pressure:

Loss of instrument air is assumed to have no significant effect on HPCI, or RCIC. However, it fails feedwater (and condensate) because the minimum flow recirculation valves on the feed pumps open on a loss of air, diverting a significant flow away from the reactor and back to the condenser.

Reactor Depressurization:

Loss of instrument air does not affect the failure probability of depressurization. SRV actuation depends solely on Nitrogen.

Reactor Coolant Inventory Makeup. Low Pressure:

Following successful reactor depressurization, the low pressure injection systems are required. RHR, LPCI, and Fire Water Crosstie are unaffected by a loss of instrument air, but as was explained above, condensate is assumed to fail.

Containment Pressure Control:

Loss of instrument air is assumed to result in loss of the main condenser as an available heat sink due to closure of the MSIVs. Reopening of the MSIVs is not assumed to be possible until instrument air is restored. The reliabilities of the RHR system and venting system are unaffected by loss of instrument air.

Continued Reactor Coolant Inventory Makeup:

This event represents the ability to continue coolant injection following the occurrence of unacceptable containment conditions. Quantification of this event is dependent upon previous system failures, since the same failures which contribute to the failure to control containment conditions may also affect inventory control. Section B.9 of Appendix B discusses these considerations.

C.4.4 Reactor Water Level Instrumentation Reference Leg Failure

Reactor water level instrumentation failures can affect the operator's perception of the condition of the core and the automatic control of coolant makeup systems. As a result, failure of water level instrumentation can disable multiple systems and adversely affect operator response.

The potential accident initiators involving water level instrumentation which have been observed in operation are:

- o High drywell temperature causing flashing of the reference legs.
- o Leaks or breaks in one of the reference legs for the reactor water level instruments.

The Pilgrim plant has four reference legs - 2 sets of 2 legs, each coming from one of the 2 nozzles on the RPV. One leg of each set has instruments which provide signals for initiation or tripping of the HPCI and RCIC systems, low pressure injection systems, ADS system, and the MSIVs. The other leg of each set is used for feedwater control, with either leg being capable of controlling feedwater. Each leg has its own level indicator in the control room.

The reference legs associated with safety related instruments at Pilgrim are located outside of the drywell and therefore are not susceptible to high drywell temperature. Therefore, plant trip due to reference leg flashing is not considered further in this analysis. However, reactor water level instrument line failures are evaluated further below.

Previous reviews of operating experience and analytic evaluations ,have determined that loss of inventory in a reactor water level instrument reference leg could result in false indications of high reactor water level. This failure mode could initiate challenges to the plant systems required for safe shutdown. This sect.ion discusses the approach used for quantification of the core melt frequency due to a reference leg failure at the Pilgrim plant.

C.4.4.1 Initiator Frequency

The probability of a leak or break sufficient to drain one of the reference legs has been calculated for the following three cases:

- o Instrument line break
- o Instrument line leak

- o Valve misoperation causing loss of reference leg inventory

All three of these cases are assumed to have equivalent impact on the operators response and the automatic ECCS initiation logic. They may be treated in the same event tree because the level sensors connected to the reference leg are assumed to indicate high level regardless of the failure mode (based upon observed incidents). The only case that is probabilistically significant is an instrument line leak; breaks and valve misoperations are of negligible probability.

For the Pilgrim PRA it is assumed that a leak occurs in one of the two reference legs which are associated with the HPCI and RCIC systems. Thus, feedwater operation should be unaffected. If the leak occurred in a leg controlling feedwater, then the feedwater pumps are expected to ramp down due to the false high level indication. However, in this situation HPCI and RCIC would be available and this s·equence is bounded by the sequence evaluated in this section.

Because the Pilgrim plant reference legs are coupled at the nozzles, the plant is treated as a 2 leg plant for the purposes of estimating the initiating event frequency.

C.4.4.2 Event Tree for Reference Leg Leak

The event tree for a reference leg leak is provided in Figure C.4-6. Each of the headings in the event tree are discussed further below.

Continued Power Operation (i.e., Continued Feedwater Operation)(RR)

A leak in a reactor water level reference leg will not always result in a plant transient. If feedwater maintains adequate level control, then power operation will continue. At Pilgrim the operation of the feedwater system following a reference leg draindown is evaluated as follows:

The initiating event is assumed to be a leak in a leg providing HPCI/RCIC system control. Therefore, the legs associated with feedwater control should be unaffected (there is no coupling between the legs except at the nozzle; a leak in one leg is not sufficient to cause drainage in the other leg from the same nozzle). Failure to continue power operation, i.e., failure of feedwater to continue operating, is estimated to occur due to a random loss of feedwater during the 24 hour mission time. If a feedwater trip occurs, the water level will fall to the low-low level setpoint for ECCS initiation.

Maintenance Error Causes Leak in Alternate Reference Legs (ORl

The potential for a maintenance error causing failures in alternate reference legs is assessed here. Also included are errors which result during attempted repairs of the leaking leg. Loss of the alternate reference legs may occur if repairs or tests are performed on the intact legs, or if the operator inadvertently attempts repairs on an intact leg, when the leaking leg should be the one being repaired. With a failure in two of three reference legs, the high pressure injection systems (HPCI, and RCIC) would be locked out due to the false high level trip signal generated for these systems, but the feedwater system would still be available. If feedwater failed, successful coolant injection will therefore depend on the operator manually depressurizing the reactor vessel and providing coolant injection with low pressure systems while ECCS level indications are high, and the feedwater control, the shutdown and upset range instruments would indicate correctly.

Opposite Division ECCS Initiation Electronics Failure (LRl

The loss of inventory in one of the reference legs causes all level instrumentation associated with that leg to read high. If the level instruments receiving input from the other reference leg associated with the ECCS systems fails, automatic initiation of the ECCS systems would not occur, since one of the legs is reading high and one has failed. For the Pilgrim plant, one of the reference legs on each set controls the initiation of HPCI and RCIC as well as the low-low level permissive for LPCI, Core Spray, and ADS operation. With one leg's instruments reading high, the other leg's instruments must respond on low-low level following a loss of feedwater. A random failure of the instrument on the opposite reference leg will defeat the automatic ECCS initiation logic, and therefore will result in the need for manual actions. The value used for instrument failure was taken from the IPEM, and is 1E-2 per demand. This failure could occur any time following the last test of the instruments.

The instrument failure probability is coupled with the probability of the operator failing to take action in response to the level indicator in the control room that is reading high. The operator should see the high level indication on one of the level indicators and take appropriate action to identify the cause. A plant trip is unlikely, because the feedwater level indicators,will be unaffected. In this case, failure of automatic initiation of the ECCS systems due to instrument failures would require two concurrent random failures, which is very unlikely.

Reactivity Control:

The methods for initiating a scram are sufficiently redundant that the probability of successful scram is equivalent_ to that used following a general transient.

Primary System Pressure Control:

These events are unaffected by a reference leg leak.

Feedwater Available:

For thi's analysis, it was assU;med that the reference leg leak did not occur on one of the reference legs that controls feedwater. Therefore, no direct impact on the failure of feedwater is assumed. The feedwater system is assumed to trip due to random causes after the reference leg leak occurs.

High Pressure Injection - HPCI/RCIC:

As a backup to the feedwater system, HPCI and RCIC provide high pressure coolant makeup. Since the initiator is assumed to be a leak in one of the reference legs that controls HPCI and RCIC, the opposite leg's instruments must function. If.the opposite division ECCS initiation electronics do not fail, the opposite leg's instruments function successfully and HPCI and RCIC operation are assumed to be unaffected. If the opposite division ECCS initiation electronics fail, the auto start instrumentation for the HPCI and RCIC systems has failed. No credit is taken for manual start of HPCI and RCIC due to the indications of high reactor vessel level and the difficulty in manually operating these systems.

Reactor Depressurization:

Plant procedures call for reactor depressurization if water level cannot be determined. When instrument failure occurs automatic ADS operation fails and the operator must manually initiate depressurization. However, the operator may believe that level is restored since ECCS level indicators are reading high, feedwater indicators are reading normally and no high pressure injection systems are operating. The operator can vary reactor water level with the controller, and verify that both water level indicators are tracking true water level. The operator may be hesitant to depressurize the reactor vessel.

Low Pressure Injection:

This event combines the operation of three redundant low pressure injection systems: Core Spray, LPCI, and the Condensate System. The redundancy in low pressure pumps is sufficiently high that the success of adequate core cooling is governed by the ability to depressurize the reactor and establish stable cooling while contradictory level indications are present.

Containment Heat Removal. and Continued Reactor Coolant Makeup:

See the discussion in Section C.l for the general transients.

C.4.5 Internal Floods

C.4.5.1 Introduction

Generic Letter 88-20 requires an internal flooding analysis as part of the IPE process. A number of internal flooding PRAs to date have been qualitative scoping analyses which have concluded that internal flooding will not lead to core damage. However, the Oconee 3 PRA concluded flooding was a dominant contributor to the total core damage frequency and subsequently made plant modifications. Other plants have experienced maintenance events which have resulted in flooding of equipment. All of these factors provide the basis for performing the Pilgrim internal flooding analysis.

The purpose of the internal flooding analysis was to determine potential .vulnerabilities due to flooding from sources such as torus rupture and pipe ruptures. The analysis used bounding, frequently conservative assumptions while still demonstrating a low potential for core damage. Attention was focused on the major flood sources in the plant which could affect multiple systems and propagate to other areas. Low capacity systems which had limited or no impact on multiple systems and flood initiators which were bounded by other flooding events were generally not considered.

The study concludes that there is only one flooding sequence which has any impact at all on potential core damage sequences, and that sequence frequency is extremely low (less than 1E-8 per year). The flooding event analyzed involves a feedwater line break in the main steam tunnel which was assumed to fail the high pressure feedwater injection system due to flooding. All flooding initiators identified have sufficient means of providing adequate core cooling independent of equipment potentially affected by the flood.

The assumptions, methodology, mitigative factors, and results of the Pilgrim internal flooding IPE are discussed in this section.

C.4.5.2 Background

Considerable review of the Pilgrim plant design and operating procedures has been performed in the past with respect to the potential and effects of internal flooding.

Flooding occurs when mechanical components in fluid systems fail. The most serious flooding usually occurs from catastrophic failures (i.e., ruptures). Flooding analyses performed at PNPS view ruptures as initiating events rather than as events likely to occur while responding to other design basis transients or accidents. This concept was retained in determining the impact of ruptures on core damage frequency (CDF).

Fluid system ruptures usually cause the loss of the train in which they occur, and they can flood equipment in other trains or systems required to recover from the event. This two-fold effect was considered when determining the impact on CDF.

Postulated flooding events at PNPS include those which could initiate from pipe breaks outside containment (PBOC), pipe breaks inside containment (PBIC), and pipe breaks in systems with high volumetric flow rates (Sea Water, Salt Service Water, and Fire Protection

System). These events were determined to be bounding in terms of their impact on plant systems, and so initiators such as over-filling water tanks, hose ruptures, and pump seal leaks were not considered further in this analysis. Also, the effects of spray upon equipment was not evaluated.

The impact of flooding events on CDF at Pilgrim is insignificant. The following text documents the assumptions and methods used to arrive at this conclusion.

The PNPS Internal Flooding Analysis is contained in ERM 88-102 Rev. 1 "PNPS Internal Flooding Analysis". Several flooding events are discussed in that document, including pipe breaks outside containment, pipe breaks inside containment, ECCS leakage, fire protection system flooding, and seawater flooding. Section C.4.5.4 contains the methodology for determining flood levels, input a d assumptions, and a listing of flooding mitigation devices.

In addition to the evaluation discussed above, an evaluation of other potential flood sources was conducted for the PRA. The only flood source of note identified during this additional evaluation involved rupture of the torus, which might cause failure of equipment in rooms or areas which connect to the torus area. However, at· Pilgrim, all such areas (e.g., quad rooms) are located at levels above the mid-plane of the torus. Since the torus water level is normally maintained at or slightly below torus mid-plane level, a complete release of this water would not impact any other areas.

Without any other flood sources of concern, the PBOC analyses (reference SUDDS/RF 83-07 and SUDDS/RF 87-1032) were evaluated further for the PRA. Those analyses are described in FSAR Appendix 0. The worst case PBOC, in terms of flooding, is PBOC-10, feedwater break inside the main steam tunnel. This is the only event identified which could result in an initiating event (caused in this case as a direct result of the pipe rupture which causes the flood) and which disables systems which could be used to respond to the initiating event. This event results in the loss of the feedwater system because of the rupture. In addition, the CRD system is rendered inoperable because the CRD pumps would be submerged during this event. Both of these systems are potential sources of high pressure injection to the RPV.

Other potential flood sources, in addition to the torus rupture, were determined to be insignificant in terms of their impact on the plant. Flooding information contained in FSAR Appendix O focuses only on the reactor building because the turbine building was assumed not to contain any safety related equipment. Calculation S&SA 61-1 was generated to determine flood levels resulting from PBOCs occurring outside the reactor building, because safety-related cables run through the turbine building. The turbine building PBOC analysis was expanded by Calculation S&SA 62-0, radwaste building-flooding. Flooding of the radwaste building via drain lines is assumed to occur as the result of PBOC flooding in the turbine building.

The PBIC flood level in the drywell was determined by the height of the lower lip of the eight downcomers (reference S&SA 84-157). The capacity of the downcomers is assumed sufficient to mitigate the submergence effects of the DBA LOCA which bounds'all other PBICs. No safety related systems, or non-safety related'systems for that matter, are affected by this event.

ECCS leakage is assumed to accumulate for 30 days following a DBA LOCA (for the purpose of determining flood levels for the affected areas). Because open equipment drains interconnect each reactor building quadrant, except the CRD quad, flooding in one quad will eventually affect the other two. In a sense, this serves as a flooding mitigation mechanism for the quad in which the leakage occurs, but the postulated leak rate is small enough to warrant this departure from the NED flooding philosophy (i.e., take no credit for drains when evaluating flooding events).

Postulated seawater flooding occurs in three locations, the condenser bay and each RBCCW compartment. Seawater flooding in the condenser bay can occur as the result of a break in the seawater (SW)system. The RBCCW compartments could be flooded by breaks in the salt service water system. Flooding from a SW break is assumed to be contained in the condenser bay. The RBCCW compartments are protected from the effects of flooding by two dewatering lines in each area. These lines route spillage to the torus compartment. In each case, operator action is required to terminate flooding by securing the appropriate pumps (if they are still running).

Fire protection system breaks were also analyzed for their affect on safety related equipment. The only safety related areas in which FPS flooding is the predominate source are the switchgear rooms. Wire mesh panels/doors prevent any accumulation in these areas.

Insights relevant to the potential and consequences of internal flooding that are provided as a part of the IDCOR IPE Methodology were also reviewed. The focus of this review was on flooding locations in the lower turbine building and the reactor building ECCS corner rooms. It was concluded that no special flooding vulnerabilities are expected at Pilgrim, consistent with previous reviews.

C.4.5.3 Impact of Flooding on CDF

Pipe ruptures can impact CDF in two ways. They can either submerge equipment required to prevent core damage, or render inoperable the system in which the rupture occurs. As mentioned previously, the PBOC-10 event, feedwater break inside the main steam tunnel, was selected as the bounding flooding event for the PRA. This event was chosen because of its initiating event frequency (a function of the large number of components in the condensate, condensate demineralizer and feedwater systems). Also, this event results in the loss of two non-safety related high pressure injection systems.

The feedwater system would be lost because of the pipe rupture, and the CRD system would be lost because the CRD pumps will be submerged. Modifications performed for the environmental qualification project give a high confidence to the assumption that no othkr plant systems are impacted from this event. No other event has impacts which approach this in terms of initiating event or system failure probabilities.

The impact on CDF from the PBOC-10 events is similar to that evaluated for the loss of feedwater transient (TF). The major differences in quantification are the initiating event frequency (flood versus "random" loss of feedwater) and the failure of the CRD system. Even though the CRD system may be physically able to operate for the evaluation of the TF event, no credit was given for it to replace inventory lost due to decay heat. Thus, the unavailability of the CRD system due to the flood has the same impact as currently modeled for the TF sequences, i.e., no credit is given in either case.

Table C.4-1, rupture frequency, illustrates how the PBOC-10 rupture frequency was determined. The areas mentioned .in Table C.4-1 are shown on Figures C.4-1 through C.4-5. Each component on the figures was counted. The number of each type was summed and multiplied by the component rupture frequency provided in the Pilgrim PRA Internal Flooding Evaluation Methodology Revision 1 (1990) Table 2.4-2. The total for each component type was summed to arrive at the .rupture frequency for the feedwater system. This frequency (8.23E-3/yr) includes pipe breaks in the condensate system and the condensate demineralizer system as well.

The feedwater pipe rupture frequency (8.23E-3/yr) is very low in comparison with the initiating event frequency for loss of feedwater (TF), which is 0.19 per year.

Since the sequence progression is similar for the flood event and the TF event, with the same systems available (or not available) to respond, the core damage frequency can be evaluated by taking the CDF calculated for events initiated by loss of feedwater (TF) and scaling down by the factor 8.2E-3/0.19, a factor of 4.3E-2.

The core damage frequency for TF sequences is approximately 1.83E-5. Therefore, the CDF for events initiated by floods is approximately 1.83E-5 x 4.3E-2 or 7.87E-7 per year.

C.4.5.4.1 Methodology for Determining Flood Levels

The methodology used for the flood analysis is outlined in this section. The major steps are:

1. Identify potential flood locations.
2. Determine blowdown/spillage volumes.
3. Determine whic,h spaces are affected by each flooding event.
4. Determine the area of affected spaces.
5. Calculate flood levels (Flood Level=Volume/Area).

C.4.5.4.2 Input and Assumptions

1. HELB locations are listed in FSAR Appendix O (Table O.6-1).

   a. Feedwater System - 89
   b. Condensate System - 12
   c. Main Steam System - 28
   d. RWCU System - 24
   e. RCIC System (steam line) - 8
   f. HPCI System (steam line) - 4

   A high energy line is defined as piping containing fluid at a temperature above 200°F coincident with a pressure above 275 psig.

2. Frontline mitigation systems (e.g., core spray and LPCI) are not considered potential sources for flooding since they are designed to higher standards and they are not usually operating during normal conditions.

3. Circ Water/SSW pipe break locations were determined in the PNPS internal flooding analysis (ERM 88-891). The Circ Water breaks occur only in the condenser bay. The SSW breaks occur only in the RBCCW compartments.

4. No credit was taken for drains in terms of mitigating a flooding event, except in the case of ECCS leakage (postulated leak rates are small compared to the catastrophic pipe breaks and accumulation occurs over the course of 30 days). Drains were used as postulated pathways for flooding connected spaces (e.g., RHR quads/torus compartment-ECCS leakage; radwaste building-eire Water pipe break in the condenser bay). The 8 inch drains to the torus compartment from the 21 ft and 51 ft elevations in the reactor building are not assumed to mitigate the effects of flooding.

5. No credit was taken for tapering of walls, sloping floors or evaporation to minimize flood levels. The escape of flashed steam to the turbine building via a blowout panel was assumed for PBOC-10 (feedwater system break in the main steam tunnel). For all other postulated breaks, all flashing steam is condensed at atmospheric conditions within the compartment where the break occurs.

6. The volume of drains lines, sump capacity and sump pump operation were neglected when determining flood levels.

7. Blowdown times for PBOCs are based upon the maximum isolation valve closing times allowed by Tech Specs. Leak rates and duration of leakage are based upon total flows. Leakage rates represent critical flows for the line losses from the fluid reservoir up and downstream of the break location to the break location.

   The leak rate and duration for a feedwater line beak in the main steam tunnel represent actual system configuration, main steam isolation valve closure upon high steam tunnel temperature, makeup to the condenser and normal hotwell content.

8. PBOC blowdown times account for diesel starting delay, except for PBOC-10. This event would be made less severe if the condensate pumps were assumed to fail on low voltage instead of low NPSH.

9. All blowdown from PBOCs occurring in the turbine building above the 51 ft elevation falls through the grating to the turbine auxiliary room.

10. All blowdown from PBOCs occurring in the turbine building below the 51 ft elevation falls to the condenser·bay.

11. Drywell flooding is prevented by the eight downcomers to the torus.

12. ECCS leakage occurs during a LOCA at a rate of 1 gpm and accumulates for 30 days.

13. The capacity of the dewatering lines in the RBCCW compartments is 9000 gpm.

14. The maximum flood rate from a Circ Water pipe break is 200,000 gpm.

15. Gross equipment areas were derated by 2-3% to allow for equipment space when determining the net floor areas.

16. Only gravity induced flow paths are considered.

17. The flood levels for HELBs are maximized based upon quasi-static analysis of flow between relatively "still" rooms/areas. Wave fronts and reflections from walls and ,objects are deemed insignificant.

18. Floor hatches with covers are considered watertight.

19. No conduit or cable installed in the plant is qualified for submergence.

C.4.5.4.3 Flooding Mitigation Devices

1. Flood protection (from a fire main break) in the 'B' switchgear room (23 ft el.) is provided by a wire mesh door leading to the adjacent corridor.

2. Flood protection (from a fire main break) in the 'A' switchgear room (37 ft el.) is provided by a wire mesh panel leading to the turbine trucklock.

3. M0-1001-47 is enclosed by a flood barrier.

4. The actuator for M0-1001-28A is rotated into a position that is above the flood level assumed for its location.

5. Enclosures for D7, DB, D9, MCCs B17, B18, and B20 are watertight to preclude flooding of these components.

6. Significant flooding of the turbine trucklock from the turbine deck (51 ft el.) is prevented by the 4 inch curb around the perimeter of the access space.

7. Door #11 prevents flooding of the reactor auxiliary bay during PBOCs (51ft el.) in the turbine building.

8. The CRD quad contains no open equipment drains. Otherwise ECCS quads would be compromised during PBOC-10.

9. Unisolable, open equipment drains interconnect the RHR and RCIC quads within 19 inches of the floor. This prevents leakage in any one quad from compromising safety related equipment.

10. The two 14 inch dewatering lines located in each RBCCW compartment are credited with mitigating the effects of SSW pipe breaks in those spaces.

11. Door #15 will confine flooding in the condenser bay until the flood level reaches the 12.1 el.

12. A blowout panel between the main steam tunnel and the turbine building mitigates the effect of PBOC-10 by allowing flashed steam to escape from the reactor building.

C.4.5.5 Conclusions

As a result of flooding analyses performed previously for Pilgrim, augmented by additional evaluations performed for the PRA, it is concluded that the internal floods contribute insignificantly to the overall CDF. The flooding impacts are bounded by a break in the feedwater line inside the steam tunnel. The CDF calculated as a result of this flood is roughly 1/100 of the CDF calculated for events initiated by a loss of feedwater event. The CDF due to floods is approximately 7.87E-7.

C.4.6 <u>Inadvertent Open Relief Valve (IORV)/Stuck Open Relief Valve (SORV)</u>

A main steam safety/relief valve can open accidentally during. plant operation or can fail to close if it opens during a transient. In this discussion, the former is referred to as an inadvertent open safety/relief valve {IORV) and the latter as a stuck open safety/relief valve {SORV).

The focus of the IORV/SORV evaluation is an assessment of the plant response to the unique containment cooling and· coolant injection challenges presented by these types of events. Specifically, the IORV/SORV related sequences result in:

- o RPV pressure reduction.
- o Reactor decay heat being rejected to the suppression pool.

The reduction in RPV pressure may present a core cooling problem if the redundancy and reliability of the low pressure injection systems is inadequate.

Containment pressure control is the second concern. In this case there is an uncontrolled release of steam to the suppression pool through the SRV discharge line. This results a long term containment heat removal challenge.

The event tree described in this section can be used t6 model both !ORV events and SORV sequence transfers from other event trees. Because Pilgrim has two unpiped safety valves that discharge directly into the drywell, the phenomenology is similar to that described for a medium LOCA if one of these safety valves is stuck open. It is assumed that no environmentally induced problems occur within the mission time of concern for components inside containment if an unpiped safety valve is stuck open.

C.4.6.1 <u>Initiating Event Frequency for IORV/SORV</u>

The derivation of the initiating event frequency for IORV is discussed in Appendix A.

The SORV event as discussed above, is as a result of the failure of a relief valve to reclose after opening during a transient. To derive the initiating event frequency for IORV, the probability of a relief valve failing to reclose of 6.8E-3 from the IPEM was used. The transfer probabilities list of initiating events was calculated by multiplying the initiating event frequency by the.number above. Initiating events with frequencies less than 1E-3 were excluded from further analysis, as the combination of the two events results in a very low initiating event frequency. Those events which do not lead directly to an MSIV isolation and SRV challenge were excluded as well. The remaining events: loss of feedwater, MSIV closure, and loss of condenser vacuum, were quantified in an event tree similar to the IORV event tree. The quantification of SORV with Partial and Total Loss of Offsite Power is discussed in Section C.2 Loss of Offsite Power. One important difference to note between the IORV and SORV event trees and the other special initiators is that gravity feed to the condenser is insufficient to make up for the loss of inventory in the RPV due to the flow through the relief valve, and so no credit is taken for feedwater during SORV events.

C.4.6.1 Event Trees for IORV/SORV

This section provides a discussion of the frontline systems' ability to maintain core cooling and containment heat removal during IORVs and SORVs.

Because there is a high probability, based upon industry experience, that an IORV/SORV will reclose as primary system pressure falls, the event trees were constructed to distribute IORV/SORV events into two categories: (1) sequences in which the valve recloses, and (2) sequences in which the valve remains open. The information contained in Section B.2 for event P was used to determine the distribution among the two categories. The basis for initiating event frequency for IORV is contained in appendix A.

Reactivity Control:

The scram system reliability is similar to that used in other accident sequences. This independence of the scram system from the initiating event is primarily due to the redundancy in the scram system which requires a common mode failure of the scram system to prevent successful scram.

SRVs Reclose:

The major difference between the IORV and SORV event trees is this event, discussed briefly above. If a transient occurs, an SRV sticks open, but closes later at a lower pressure, this event is treated as part of the quantification for the original transient initiator. The initiating event frequency for this event is the product of the original transient initiator, times the probability that the valve will fail to reclose immediately, times the probability that the valve will fail to reclose later in the blowdown. The initiating event for SORV therefore is a relief valve that is stuck open, and will remain stuck open for the rest of the event tree.

The IORV tree is drawn with the inadvertent opening of the IORV as the initiating event. In the IORV event tree, the valve is given a chance to close by the "Valve Fails to Reclose" event. If the valve closes, the rest of the tree is quantified similar to a transient initiator tree. If not, the quantification is similar to a medium LOCA.

Reactor Coolant Inventory Makeup. High Pressure:

The IORV/SORV will eventually result in reactor depressurization. For a typical plant, RPV pressure may drop to 400 psi within 30 minutes and then continue to drop more slowly. This response makes HPCI and RCIC viable coolant injection options over the short term, (i.e., during the first hour). Note that the motor driven feedwater pumps are assumed to remain available during this event. The event tree indicates that even though high pressure injection may be successful during the short term, low pressure injection eventually will be required due to reactor depressurization.

Reactor Depressurization:

The IORV/SORV should result in a lower failure probability for depressurization because fewer valves will be required to open. However, because the failure probability for depressurization is typically dominated by common cause failures, no significant impact is expected.

Reactor Coolant Inventory Makeup. Low Pressure:

Following successful reactor depressurization, the low pressure injection systems can supply adequate makeup to the reactor. Because the condensate system is unaffected by an IORV/SORV, the probability of coolant makeup from the hotwell is similar to that evaluated in other sequences.

The IORV/SORV event tree takes credit for the use of the CRD system to maintain inventory in the reactor vessel. This is because of the fact that high pressure injection from other sources (feedwater or HPCI/RCIC) has been successful for some time, and the reactor depressuri.zes gradually through the open valve.

For the other event trees it is assumed that if the feedwater or HPCI/RCIC systems are successful, they will continue to be successful until the reactor is stabilized at hot shutdown conditions, at which time it is assumed that many options exist to continue vessel inventory control and decay heat removal. The evaluation also conservatively assumes that mechanical or electrical failures of the high pressure systems, if they occur, happen immediately upon demand of the systems. This assumption eliminates the CRD system as a short term injection source, since the flow requirements shortly after shutdown are greater than the capability of the CRD system. Thus, for most event trees, the CRD system is not considered either because the other high pressure systems are available, or the CRD system is incapable of maintaining inventory.

For the IORV/SORV event tree it is still assumed that feedwater and HPCI/RCIC, if they fail, fail immediately. Therefore, the CRD system is assumed to be inadequate in the sequences in which high pressure injection is immediately unavailable from other sources. However, on the success path for high pressure injection, the high pressure systems are available only until the vessel depressurizes through the IORV/SORV, at which time the evaluation assumes that the HPCI and RCIC systems become unavailable due to low pressure. The depressurization is slow enough that the amount of decay heat removed is sufficient to allow injection by the CRD system to be adequate for continued water inventory maintenance following reactor vessel depressurization. Because the CRD pumps, if at maximum flow, can provide adequate flow to make up to the vessel at decay heat loads after one or more hours of heat removal, the CRD system is included as a "low pressure" injection source on the path in which high pressure injection is initially available.

<u>Containment Pressure Control:</u>

An IORV or SORV is assumed to have no adverse impact on the failure rate of the main condenser or the RHR system for containment heat removal.

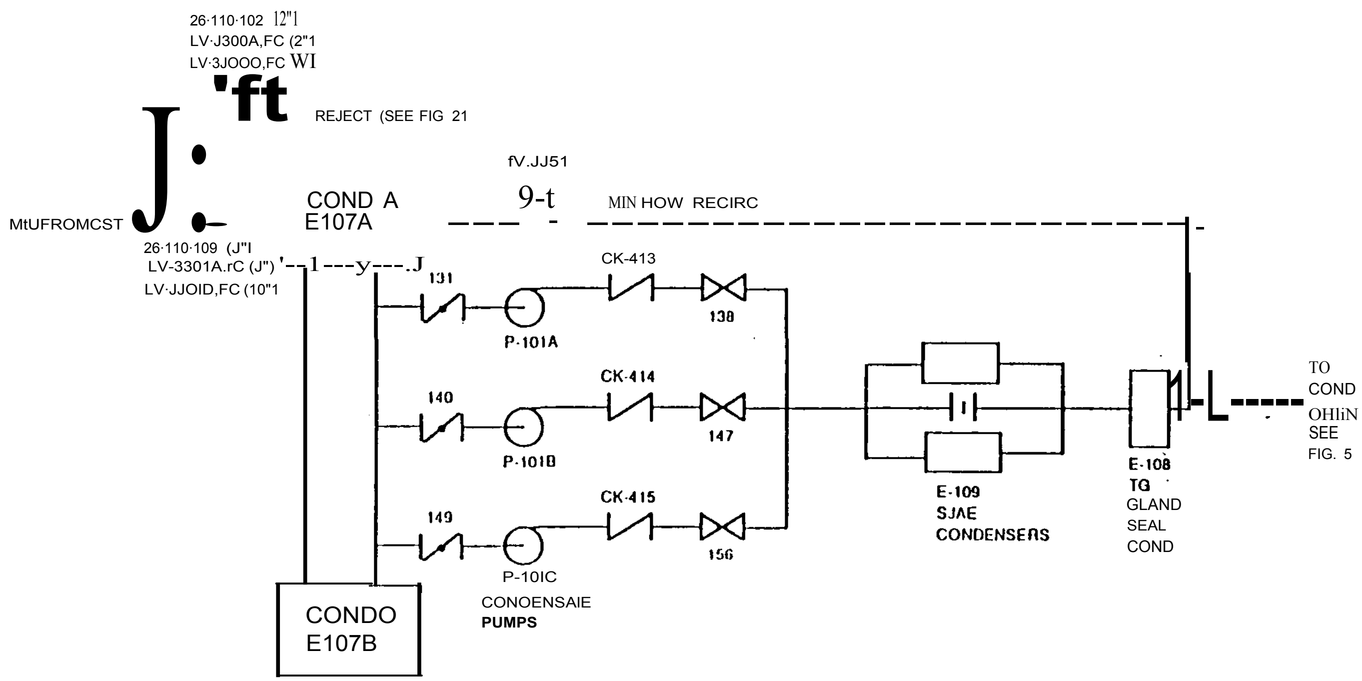
Continued Reactor Coolant Inventory Makeup: This event represents the ability to continue coolant injection following ,the occurrence of unacceptable containment conditions. Quantification of this event is dependent upon previous system failures, since the same failures which contribute to the failure to control containment conditions may also affect inventory control. Section B.9 of Appendix B discusses these considerations.

Table C.4.-1

RUPTURE FREQUENCY

Component	Area 1	Area 2	Area 3	Area 4	Area 5	Total	Component Rupture Frequency	Total Frequency
Check Valve	3	0	3	4	0	10	4.2E-9	4.2E-8
AO Valve	1	0	6	0	14	21	1.1E-8	2.31E-7
MO Valve	0	5	5	0	14	24	4.4E-9	1.06E-7
Manual Valve	3	3	7	2	14	29	1.5E-9	4.35E-8
Piping >3"	20	20	35	8	60	143	8.5E-10	1.22E-7
Piping <3"					40	40	8.5E-9	3.4E-7
Heat Exchanger	3	8	4	0	0	15	8.5E-10	1.28E-8
Restriction Orifice	1	1	0	0	15	17	1.5E-9	2.55E-8
Pump	3	0	3	0	0	6	1.5E-9	9.0E-9
Tank	0	0	0	0	7	7	8.5E-10	5.95E-9
								9.4E-7/yr

1. Rupture Frequency: 9.4E-7/hr (8760 hrs/yr) = 8.23E-3/yr.
2. Heat Exchangers were assigned the same frequency as tanks.
3. Pumps and restriction orifices were assigned the same frequency as manual valves.
4. Basket strainers (Figure C.4-5) were counted as restriction orifices.



lilli£
I. THE SUCTION SIDE OF TilE PUMPS IS NOT CONSIDERED SUSCEPTIBLE TO RUPTURE.

Figure C.1-/ -1 Condensate System Area 1

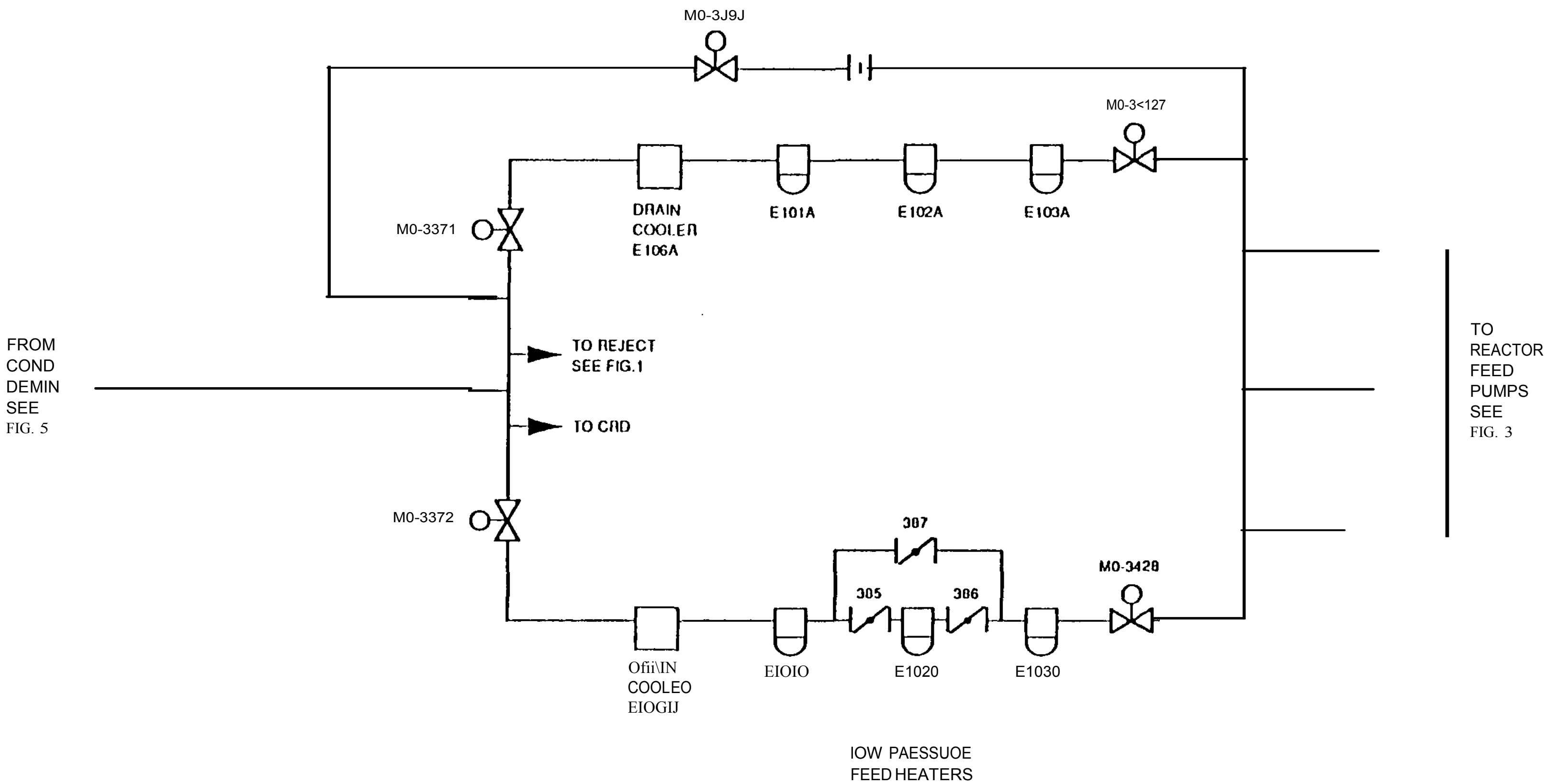


Figure C..ll -2 Low Pressure Feed Heaters Area 2

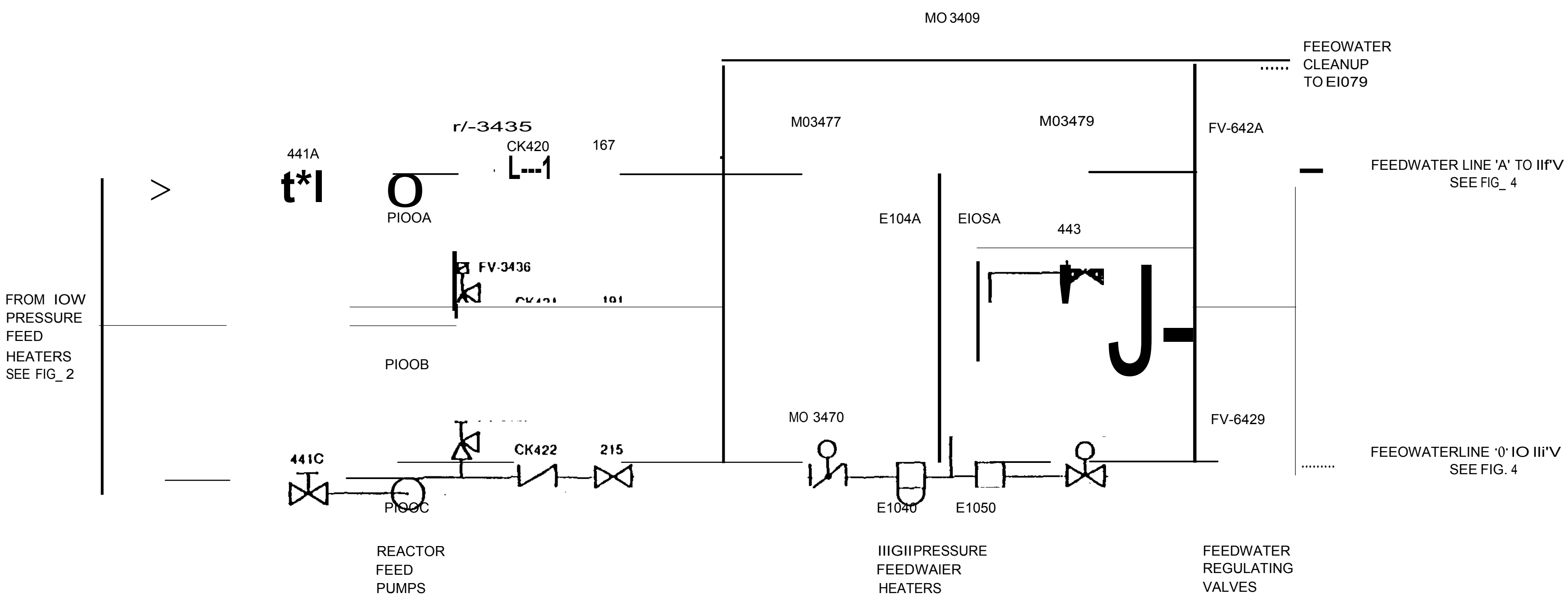


Figure 4 -3 Feedwater System Area 3

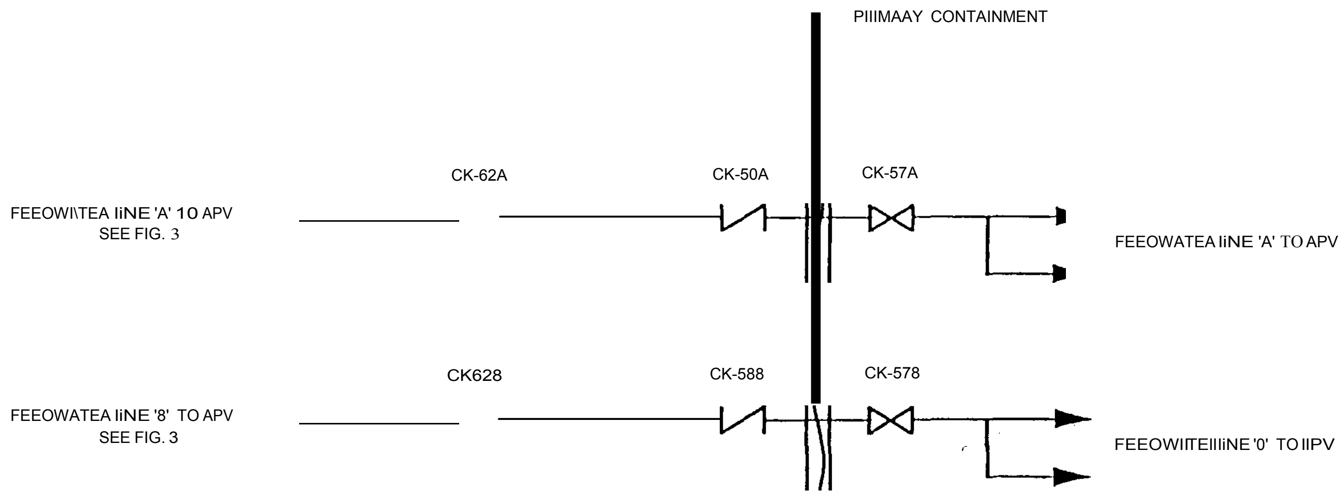


Figure C. 't ·-4 Feedwater System Area 4

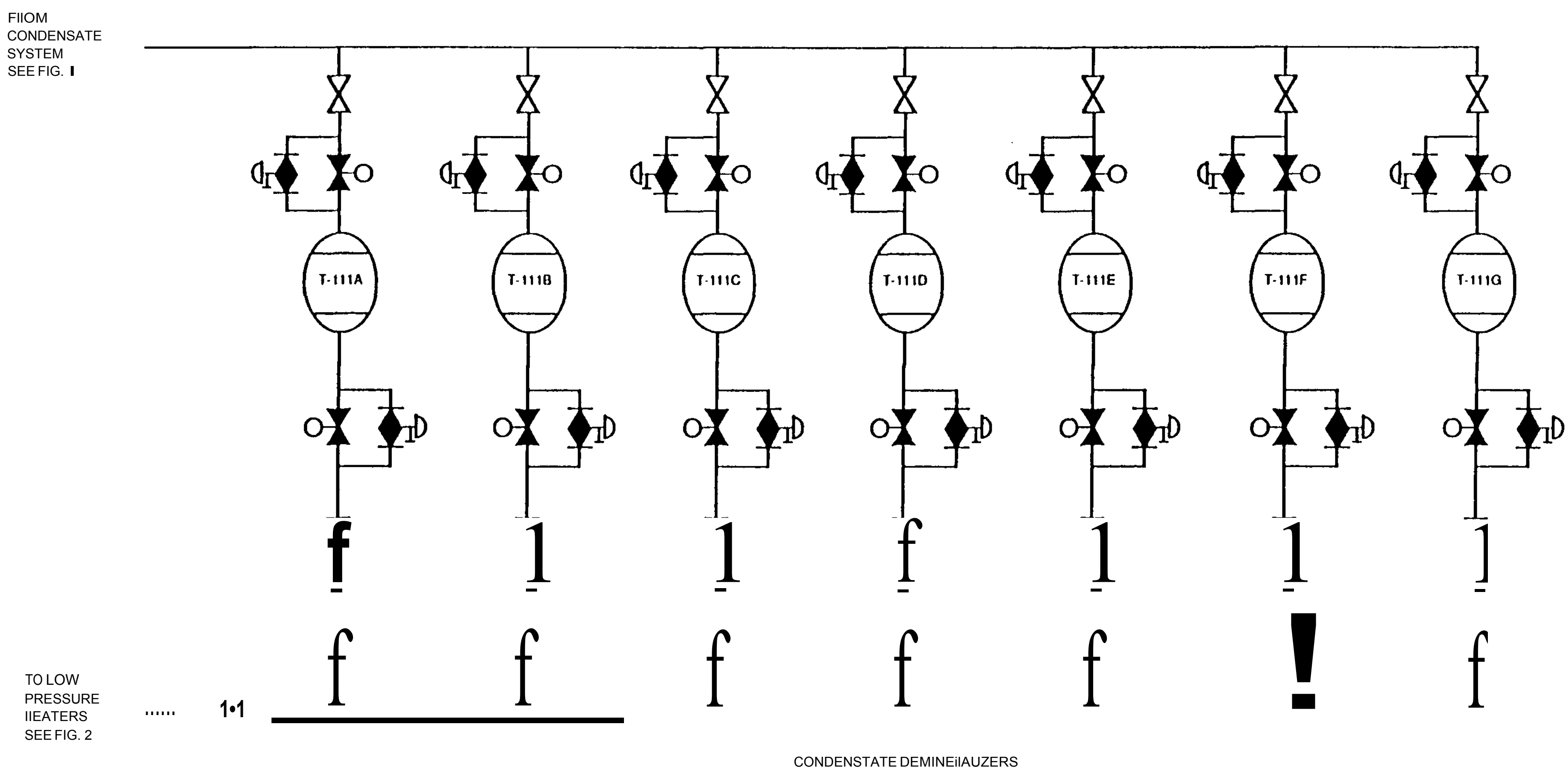


Figure c.. t.f -5 Condensate Demineralizers Area 5

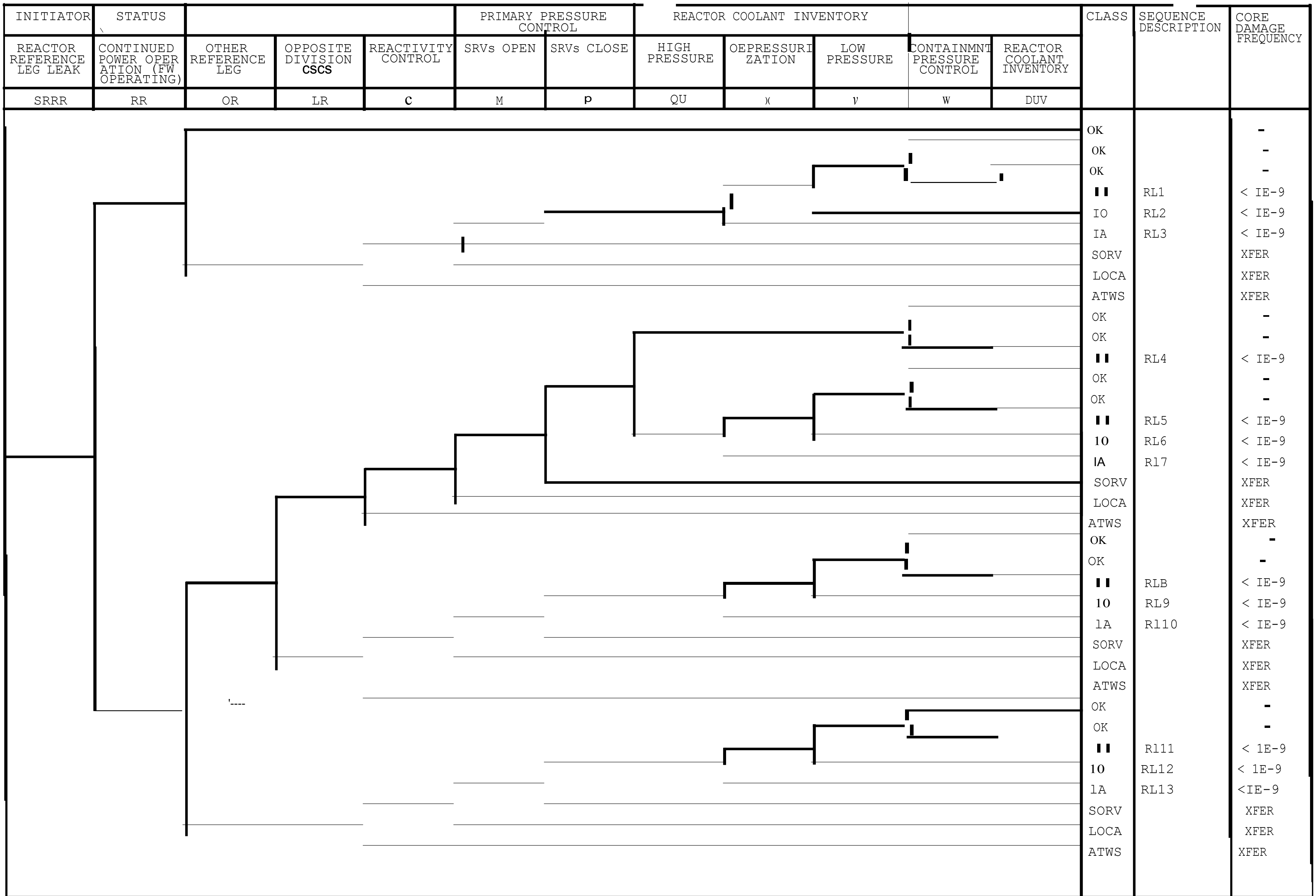


FIGURE C.4-6 REFERENCE LINE BREAK

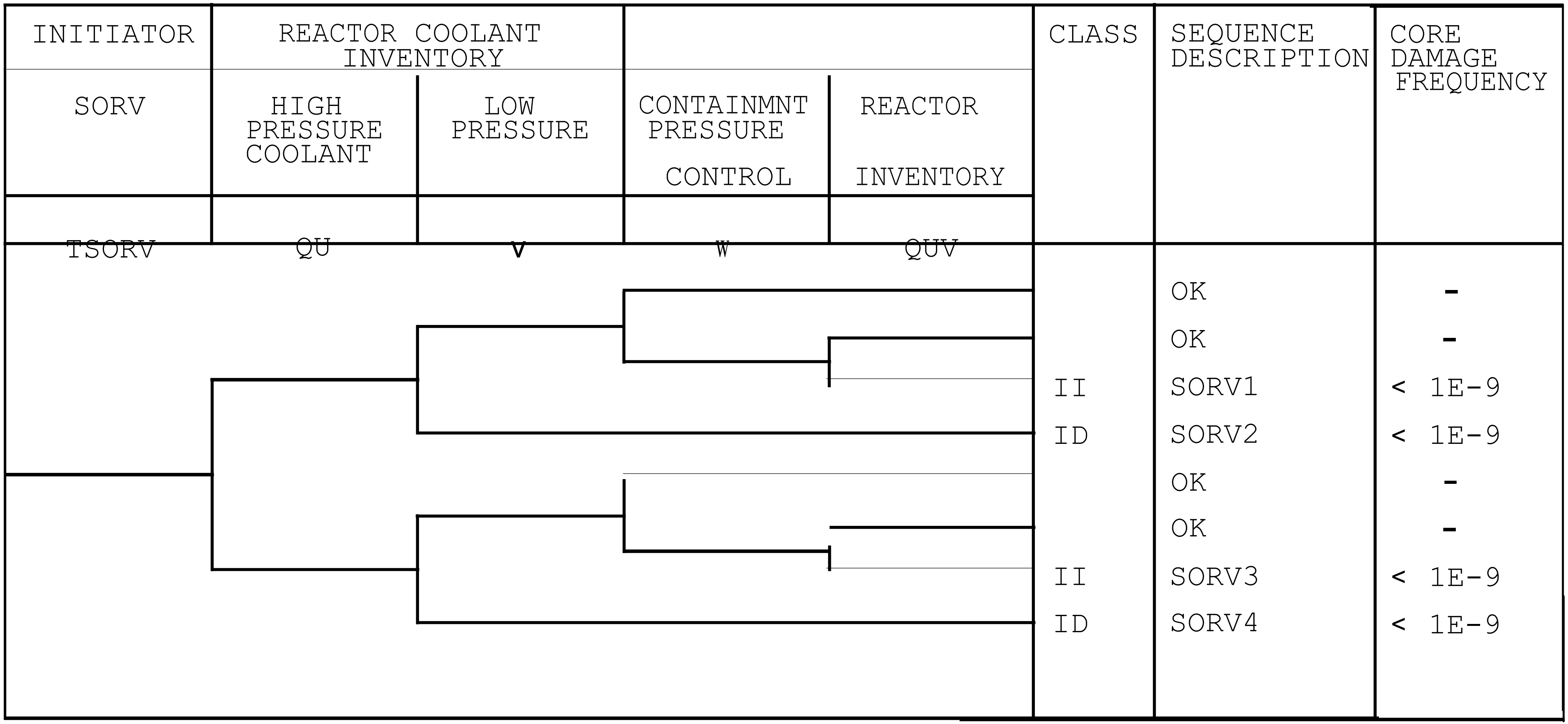


FIGURE C.4-7 NON LOOP STUCK OPEN RELIEF VALVE

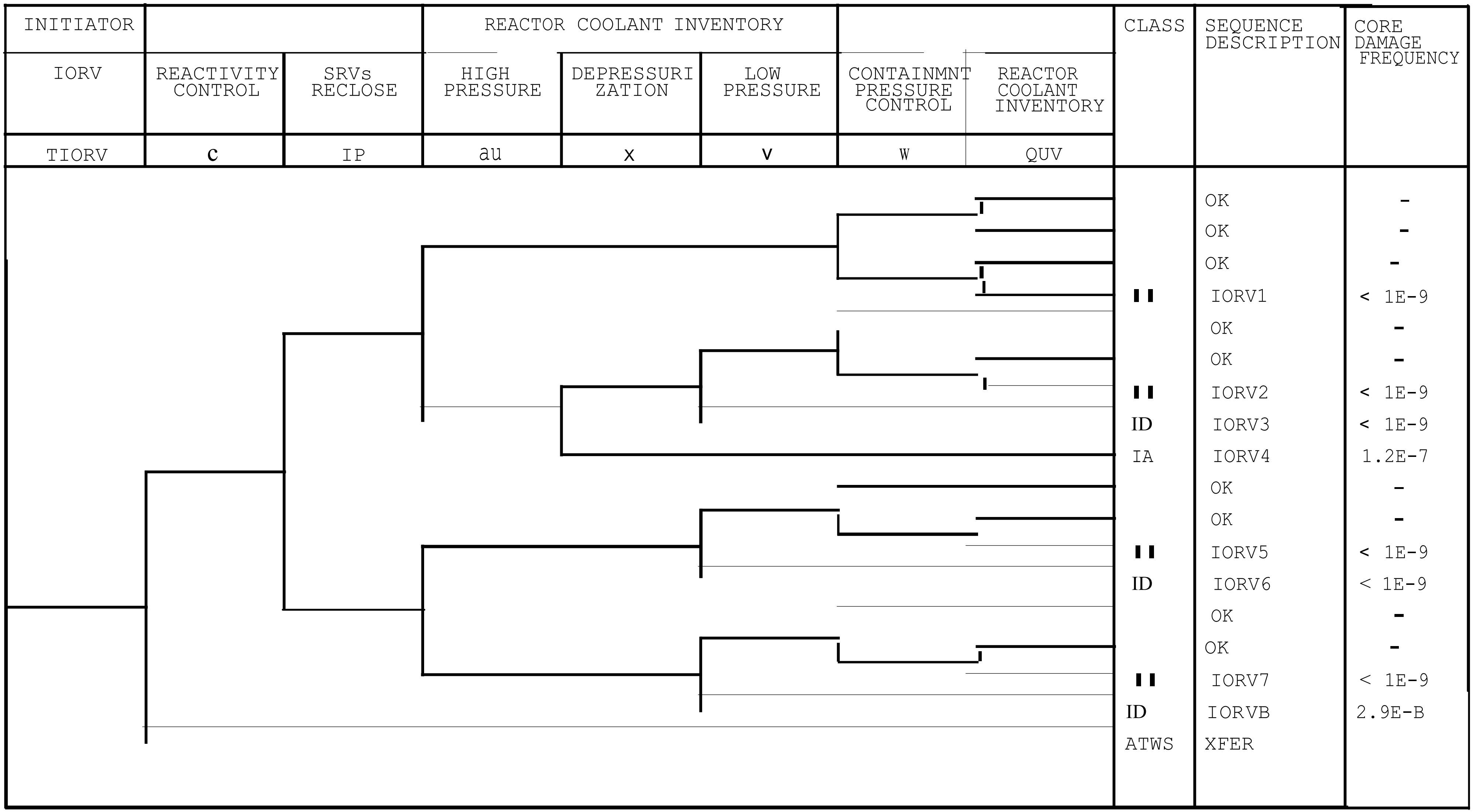


FIGURE C.4-B INADVERTENTLY OPENED RELIEF VALVE

C.S ATWS SEQUENCES

A portion of the spectrum of low frequency accident sequences postulated in PRAs are associated with a transient with a coincident failure to scram. This section evaluates potential risk contributors from these sequences at the Pilgrim plant. Principal contributors to the core damage frequency associated with a failure to scram can involve sequences affecting accident classes IC and IV.

The specific topics to be discussed relative to ATWS and the operation of the Pilgrim plant include the following:

- Background: Section C.S.l
- Response of Pilgrim during ATWS: Section C.5.2
- Scram System Reliability: Section C.5.3
- Criteria for Acceptable Safe Shutdown: Section C.5.4
- Operator Error Probabilities: Section C.5.5
- Transient Initiator Frequency: Section C.5.6
- ATWS Event Trees: Section C.5.7

C.S.l Background

One of the functional requirements for successful accident mitigation is the ability to insert sufficient negative reactivity into the core to bring the reactor subcritical. In preceding sections, sequences investigated are those in which successful control rod insertion has been accomplished and the focus of the evaluation is on subsequent functional requirements such as coolant injection and containment heat removal. This section focuses on those event sequences in which an initiator, principally an anticipated transient, occurs coupled with a failure to insert the control rods. Other initiators, such as LOCAs, when coupled with the conditional probability for failure of reactivity insertion may also be included in the contributors to Class IV, but are generally of low frequency compared to the dominant ATWS sequences discussed in this section.

Substantial effort has been made in recent years to the reduction in risk at BWRs associated with ATWS events. Modifications to the control rod drive scram discharge systems in the form of increasing their volumes and providing diverse instrumentation for level monitoring have been performed reducing the potential for common mode reactor trip failure due to mechanical causes. Modifications associated with the ATWS rulemaking (10CFR50.62) similarly reduce the potential for control rod insertion failure due to electrical reasons through the installation of Alternate Rod Insertion (ARI) solenoid valves. Mitigation of ATWS events also has been improved through the installation of automatic trip of the recirculation pumps, enrichment of the boron associated with Standby Liquid Control, and through the improvement of emergency operating procedures. Each of these improvements maximizes the amount of time available to the operator to take effective action in terminating the event.

This analysis incorporates the effects of these safety enhancements in evaluating the residual risks associated with ATWS at the Pilgrim Plant.

C.5.2 Response During ATWS

Pilgrim's response to a postulated failure to insert the control rods following an anticipated transient potentially involves the operation of a number of both normally operating and standby safety systems. The basic functions that satisfy the requirement for a safe and successful shutdown are:

- Primary system pressure control;

- Reactivity control;
- Coolant injection and primary system inventory control; and
- Containment heat removal.

For postulated ATWS sequences, Table C.S-1 summarizes the required functions and systems available to mitigate the potential adverse plant conditions.

Table C.5-2 summarizes the nomenclature used to describe these functions in the event tree evaluation.

Response of the Pilgrim plant to an ATWS event has been extensively analyzed to determine the potential success paths which are available at the various stages of the event. The following is a condensed discussion of these analyses and is presented to provide a basis for selection of the various headings of the-event tree and their success 'criteria which follow in later sections of this report.

The discussion is divided into the response of the reactor nd the containment to an ATWS.

For the purpose of discussion the event under consideration is an MSIV closure with a failure to insert control rods due to mechanical causes. This particular event is typical of that historically analyzed from a regulatory perspective and is generally considered to be bounding in its effects on primary system and containment response and on the time available for the operator to take appropriate actions to terminate or mitigate the event. Where there are aspects of the Pilgrim plant response to ATWS that are important during scenarios other than for this particular initiator, these features will be identified and their effect on plant response during these other scenarios discussed at related stages of the MSIV event.

Primary System Response

It is assumed conservatively that the transient begins with the reactor at full power and that the cause of the failure to trip prevents all control rod drive mechanisms from insertion. Initial reactor power less than full power or partial rod insertion either automatically or due to manual actions in response to the event will result in less limiting plant conditions and more time available for operator action than presented in this discussion.

Analysis of a reactor trip from full power for the Pilgrim plant has been performed with TRACG [Ref C.5-2]. The code was developed by General Electric Company and features three-dimensional reactor vessel thermal hydraulics coupled with one-dimensional core neutron kinetics.

During the first few seconds of the event reactor pressure rises to the point that the four safety relief valves open (1090 psig), the alternate rod injection and recirculation pump trip (RPT) setpoints are reached (1175 psig) and the two spring loaded safety valves actuate (1275 psig). The four safety relief valves are piped directly to the torus and heatup of the suppression pool begins at this point. Pressurization of the containment by way of steam flow to the drywell through the unpiped safeties also begins.

The rate at which power continues to be generated is limited, however, by the void increase and reactivity reduction which results from the recirculation pump trip. Reducing reactor flow to natural circulation results in a drop of nearly 40% in reactor power.

If the failure to scram were due to electrical causes, this reduction in power and pressure would be sufficient to provide time for ARI solenoid valves to bleed the pressure from the air headers to the scram valves, gradually causing rod insertion (-15 sec) and termination of the event. Also, if the main condenser were available, the pressure rise resulting from closure of the admission valves would result in actuation of the turbine bypass valves (940 psig) which can relieve as much as 25% rated steam flow [Ref C.5-3]. If either ARI or the turbine bypass· valves are available and effective in performing their intended functions, then in conjunction with RPT, the pressure rise in the primary system will be limited to well below primary system pressure limits.

For MSIV closure events, steady state reactor pressure is greater than that which would occur during events with the main condenser available. Vessel steam flow for these events is analyzed to be just below the capacity of the safety relief valves, thus preventing reactor over-pressure conditions [Ref C.5-5]. To provide margin on reactor pressure design limits, an automatic trip of the feedwater pumps occurs when the initial pressure rise reaches 1400 psig. This trip provides additional reductipn in reactor power in two ways. First it reduces core inlet sub-cooling which increases the voiding within the core. Also, the subsequent drop in reactor level reduces the natural circulation flow rate through the core resulting in a decrease in reactor power. During MSIV closure events, the combinat·ion of RPT and feedwater pump trip (FWT) reduces reactor power to the point that the pressure rise which occurs. in the primary system is again, well below design limits. Feedwater pump trip is not necessary for events in which the main condenser is available and will not occur because the additional capacity provided by the bypass valves prevents the initial pressure rise from reaching the FWT setpoint. Without further operator intervention, then, events with the main condenser and feedwater pumps available will continue with reactor pressure elevated above the safety relief valve setpoint, reactor water level near normal due to the availability of feedwater, and steam flowing both to the suppression pool through several safety relief valves and to the main condenser through the turbine bypass valves. The rate of energy relief will be split about evenly between the bypass and safety valves at an approximate total steam flow between 50% and 60% rated [Ref C.5-1].

Continuing with the MSIV closure event description, a reduction in power level and reactor pressure will begin to occur following trip of both the recirculation and feedwater pumps. This is because of the combined effect of the loss of forced circulation, core inlet sub-cooling, and the lowering of reactor water level. On reaching the reactor low-low water level setpoint (-49") only two to three safeties will be open. At this level the HPCI and RCIC systems will receive a signal to actuate and begin injection of cold water to the vessel. A temporary additional reduction in reactor pressure will occur as a result of this cold water addition to the point that as few as one safety valve will remain open. Once the cold water reaches the reactor core, however, a power increase will occur and reopening of additional safeties will begin. The approximate power that ultimately will result will be equal to that required to heat up and boil the water being injected to the vessel. The steam flow associated with the injection of water from the HPCI and RCIC systems to the reactor under these conditions is approximately 40% of the steam flow associated with normal power operation. This requires three to four safety valves to maintain control of reactor pressure.

The reactor water level calculated to be reached through the use of HPCI at rated flow is above the top of the fuel assemblies but·below the reactor low-low level setpoint. As a result of maintaining reactor inventory at this level, the ADS timer will begin operation. Because of the actuation of the unpiped safeties early in the event, containment pressure is expected to be in excess of 2.5 psig at this time and automatic operation of the ADS is expected within two minutes of reactor water level falling to below reactor low-low water level.

If operation of the ADS occurs automatically, it could result in depressurization of the reactor to below the shutoff head of low pressure injection systems such as LPCI and core spray. It is desirable to avoid uncontrolled injection from these systems in order to prevent power spikes associated with the rapid insertion of cold water from these high volume sources. If HPCI is in operation, reactor power will return to a sufficiently high level that the reactor pressure should be higher than the shutoff head of low pressure pumps. Without HPCI, depressurization and injection with low pressure systems ill occur.

However, TRACG analysis [Ref· C.S-6] has shown that uncontrolled injection with low pressure systems during an ATWS will not result in substantial fuel damage or threaten the integrity of the reactor vessel. This is because reactor power and pressure rise as a result of cold water insertion to the point that the shutoff head of the low pressure systems will be exceeded. This terminates low pressure injection flow to the vessel until reactor power and pressure once again drop back to levels at which low pressure systems can resume injection. In this regard, automatic low pressure injection during an ATWS is self-limiting under these conditions.

It is still desirable to avoid automatic and uncontrolled low pressure injection (to allow the operator easier means of controlling reactor level) and ultimately to reduce the potential for boron washout. During the MSIV closure event the means of preventing ADS operation and depressurization of the reactor is to inhibit ADS. Either or both methods are acceptable in accordance with Emergency Operating Procedures (EOPs). Care should be taken, however, if the 'latter method is used in preventing automatic operation of the ADS. It should be remembered that during events beginning with the isolation of the primary system the feedwater pumps are tripped initially for the purpose of assuring that reactor power is sufficiently low that safety valves are adequate to control reactor pressure. Returning reactor water level to near normal requires returning multiple feed pumps to service while the reactor is still at power. However, the potential for raising the water level to near normal such that there is little margin on safety valve capacity is judged to be low for a number of reasons. Emergency operating procedures warn the operator to slowly raise water level during ATWS situations in anticipation of rises in reactor power and pressure such as this (post boron injection), and the operators are trained to anticipate the need to lower level during such an event as opposed to raising level in an uncontrolled manner. Even in the unlikely event the operator were to assume that a normal transient were in progress as OP,posed to an ATWS event, it would be unusual for actions to be taken to return multiple trains of feedwater to service in an attempt to restore reactor level to above normal levels.

Should the operator not take action to inhibit ADS operation in either manner, depressurization of the reactor to the shutoff head of low pressure systems will occur. Approximately three minutes occurs between the time that the ADS actuates until the shutoff pressure of LPCI and core spray is reached. At this point operator action to control level with low pressure systems is desired. Again, even if uncontrolled injection is allowed, primary system response is such that substantial core damage is not expected nor will the integrity of the primary system be in jeopardy [Ref C.S-6].

Through al] of this, the operator should be attempting to shut down the reactor by way of control rod insertion or actuation of Standby Liquid Control (SLC). The design of SLC at Pilgrim is such that it is capable of injecting the equivalent of 86 gpm of 13% sodium pentaborate solution. Analysis indicates that reactor shutdown can be achieved in 12 minutes following initiation of this system [Ref C.S-7]. This analysis assumes that adequate mixing of the boron solution is occurring as it enters the vessel. The adequacy of mixing is dependent on flow through the reactor. Sufficient flow and adequate mixing are predicted to occur as long as the operator maintains the water level at or above the top of the fuel. For most scenarios, as a result, the reactor water level will be sufficiently high and adequate mixing will occur during the injection phase.

In addition to actuation of SLC, the operator may be attempting to limit steam flow to the containment by minimizing power through reactor level control. The power level with the reactor water level near normal level, the main condenser in service, and natural circulation occurring, is expected to be between 50% to 60% full power. As stated above, under these conditions steam flow is split between the main condenser and safety valves. For the MSIV closure event, again under natural circulation conditions and with HPCI and RCIC maintaining reactor inventory, reactor water level is lower and the reactor is limited to near 40% initial power [Ref C.S-1]. Operator action to limit injection flow to the reactor and lower level to the top of the fuel can reduce power still further to as low as 10% to 20% rated power [Ref C.S-8]. The advantage of taking this action is to decrease the amount of steam being directed to the containment and maximize the amount of time available for operator action and SLC injection to take effect in shutdown of the-reactor.

Containment Response

As noted above, containment response to ATWS events depends on the rate at which reactor power is directed to the containment. The amount of energy released to the containment early in the event is governed by automatic response of plant systems and equipment to the ATWS. Early during the MSIV closure event, for example, five to six safety valves are required to prevent reactor over-pressure. As a result, steam is being released to both the suppression pool through the safety relief valves and directly to the drywell through the unpiped safeties. Containment pressure rises quickly during the event to above the containment high pressure setpoint of 2.5 psig. Shortly after the recirculation and feedwater pumps have tripped, however, power drops to the point that only the piped relief valves are required and all energy release is to the suppression pool. From this point of the transient, containment pressure slowly rises as the suppression pool heats up and becomes saturated.

Containment response during the latter stages of an ATWS event is principally governed by operator response to the transient. Several containment parameters are important in determining the most appropriate operator actions during the event. These parameters include- the suppression pool temperature, which indicates when the Residual Heat Removal (RHR) system should be initiated, the boron injection initiation temperature (BIIT) by which boron initiation should have commenced and reactor level/power control is required, the heat capacity temperature limit (HCTL) at which point reactor depressurization is required, and the containment temperature and pressure limits for which action is required to preserve containment integrity.

For transients occurring at decay heat loads, the latter limit would be represented in the emergency procedures by the primary containment pressure limit (PCPL). The PCPL has a value of 56 psig at normal suppression pool levels and a corresponding suppression pool temperature (assuming all heat is being directed to the suppression pool) near 300F. A lower value is used during ATWS events where steam flow rates to the suppression pool are substantially greater than decay heat levels. The IPE methodology suggests a suppression pool temperature limit of 260F at which time items such as suppression pool loads greater than normal and the potential for inadequate vapor suppression at steam flow rates greater than those associated with decay heat loads may become considerations.

Conservatively assuming little action by the operator to terminate the event or limit the rate at which energy is entering the containment, the following are the approximate time frames to reach these various limits:

	BIIT	HCTL	260F	
Time (min) power	5	12	28	(These values assume reactor near 30% to 40% rated is being direct to the suppression p'ool following RPT)

As stated above, the operator plays a significant role in terminating or otherwise affecting the course of an ATWS event. Perhaps the most significant of these actions is the initiation of SLC. To be effective in shutting the reactor down in time to avoid exceeding some of these containment limitations, SLC must be actuated sufficiently ear.ly to permit injection of a sufficient amount of boron to achieve reactor shutdown prior to exceeding recommended limits associated with these parameters. As noted earlier, Pilgrim specific analysis indicates that the existing SLC system configuration and the enriched concentration of boron will permit reactor shutdown in approximately 12 minutes. Assuming this rate of injection with little other operator action, the time available to the operator to initiate SLC and prevent exceeding 260F for the suppression pool in an MSIV closure transient is 20 minutes. As before, this value assumes that the steam flow to the pool is 30% to 40% rated power and drops off linearly as boron concentration increases in the vessel).

In fact, other operator actions are expected which increase the amount of time available for SLC to become effective and reduce the energy addition to the containment. Such actions include lowering level in accordance with emergency procedures to reduce reactor power level to the maximum extent practical. For sequences in which the main condenser is available, this action can result in termination of steam flow to the suppression pool all together by reducing power to below the capacity of the turbine bypass valves regardless of whether or not SLC has been actuated or is effective. For MSIV closure sequences, however, it is assumed that the operator action to limit the power level in this manner is highly coupled with SLC initiation. That is, the operator will be performing this action only if it is also recognized that SLC should be initiated. In this rega,rd it is expected that level control will be effective only during the period in which SLC injection to the vessel is occurring and will result in extending the time for effecting reactor shutdown ·by only a few minutes. Time available for the operator to take these actions including actuation of SLC for.the MSIV isolation transient before pool temperature exceeds 260F is 25 minutes. This value assumes that SLC and level control are initiated simultaneously, and that for main condenser events this action effectively terminates all steam flow to the suppression pool and for MSIV closure events power is reduced to <20%.

Still more time is available for operator action to inject SLC given the guidance provided in the emergency procedures. For MSIV closure events, for example, in excess of an hour can be made available to initiate SLC if level control is initiated at the boron injection initiation temperature as directed by the EOPs. However, because these events are assumed to oe highly coupled (i.e., the operators are highly likely to have injected SLC given that they are attempting level control) the time frames listed above are used in determining the likelihood of operator action to shutdown the reactor during ATWS events.

Besides operator actions in controlling reactor level and actuating SLC, containment heat removal equipment plays a role in determining containment response to an ATWS. Normal heat removal equipment includes the main condenser, RHR, sprays from external sources, and containment venting. For the purpose of removing heat at reactor power levels, only the main condenser is considered to have sufficient capacity to prevent the containment pressure and temperature from rising. Even if reactor level control is used to reduce reactor power to as low as 10% to 20% rated, the combined capacity of RHR and the vent are insufficient to prevent the temperature of the suppression pool and containment pressure from rising (each loop of the RHR heat exchanger is capable of approximately 2.5% rated power at a suppression pool temperature of 260F).

If sufficient venting paths are initiated to manage this steam generation rate, a substantially greater steam flow addition to the reactor building will occur than under decay heat conditions, possibly degrading the environment in the reactor building more than expected for containment decay heat removal failure events. As a result little credit for heat removal systems other than the main condenser is taken while the reactor is at power.

Once the reactor is shutdown, however, these other means of heat removal become more viable. Less credit for repair of these systems is taken, however, because of the short duration over which containment heatup occurs during ATWS. Detailed assumptions associated with the adequacy of system and operator actions described above and used in the quantification of the ATWS event tree are presented below in the success criteria section of this discussion.

C.5.3 Scram System Reliability

The single system in the ATWS sequences which has a dominating effect on the probabilistic quantification of ATWS quantification is the scram system consisting of the reactor protection system logic, the control rods, the control rod hydraulic system, and the control rod drive mechanisms.

The common mode failure to scram estimate of 3E-5 per demand is taken from NUREG-0460 for this evaluation. This failure to scram estimate is allocated between mechanical and electrical failures based upon observed precursors at BWRs. The allocation of the BWR scram system failure rate for this evaluation is based upon observed precursors and operating experience with BWR scram systems. The allocation is 2.25E-5 per demand for electrical failures, 7.5E-6 per demand for mechanical failures. Reference C.5-10 contains details on the precursors used to calculate the allocation. Of the events listed in Table 3.3-1 of Reference C.S-10, one event has been eliminated for all plants, and one mechanical precursor, namely the July 1980 SDV event at Dresden, was eliminated as a potential event at Pilgrim because of modifications made to the plant in response to the ATWS rule. This leaves eight total events, of which two are potential common cause mechanical failures. Thus, the allocation between mechanical events and electrical failures becomes 1/4 mechanical, 3/4 electrical for the Pilgrim plant. The overall value of 3E-5 per demand is highly uncertain (log normal distribution with an error factor of 20 to 30). It is recognized that the value used in this probabilistic evaluation, i.e., the mean point estimate from NUREG-0460, may be conservative.

C.5.4 Success Criteria

Success criteria for the functional events that must be accomplished to achieve shutdown of the reactor during ATWS events are discussed below:

Control Rods (RPS)
Success at this heading implies insertion of control rods over the first several seconds of a transient as a result of a signal from the Reactor Protection System (RPS). Successful reactor shutdown in this time frame reduces power to decay heat levels and results in the use of equipment important to cooling the core and removing heat from the containment as outlined in the transient event trees. Following failure of the RPS heading, the systems necessary to achieve successful shutdown depend to some extent on whether the cause of the failure to insert rods is mechanical or electrical in origin. Electrical RPS failures can be successfully mitigated by tripping the recirculation pumps (RPT) and subsequently causing control rod insertion by actuation of alternate rod injection solenoids (ARI). Mechanical control rod insertion failure or electrical failures with coincident loss of ARI require actuation of standby liquid control (SLC) to terminate the event.

Alternate Rod Insertion (ARil
In conjunction with recirculation pump trip (RPT), alternate rod injection (ARI) equipment can successfully provide rod insertion by

bleeding the pressure from the pneumatic supply to the scram valves, effectively terminating those ATWS sequences initiated by an electrical failure to scram. Actuation signals for ARI include high reactor pressure (1175 psig) and/or low reactor water level (-46"). It is necessary to trip the recirculation pumps in addition to actuating ARI in order to reduce power below safety valve capacity while the air headers to the scram valves depressurize.

Recirculation Pump Trip (RPT)

Automatic recirculation pump trip occurs on the same high reactor pressure and low reactor level signals as initiates ARI. Tripping the recirculation pumps eliminates forced circulation, reducing the flow through the core. This causes additional voiding in the core and a corresponding reduction in power. This reduction in power assists in the mitigation of ATWS in two ways. First, reactor power is reduced to below safety valve capacity quickly during the event providing protection of the reactor from over-pressure. Second, it minimizes the amount of steam directed to the suppression pool, increasing the time available for the operator to take actions to initiate SLC. Reducing the power to near safety valve capacity permits time for ARI to bleed down the pressure in the air headers to the scram solenoids and insert control rods, 'if the failure to scram is due to electrical causes. If the failure to trip is due to mechanical causes, however, the effectiveness of tripping the recirculation pumps is dependent on the status of the main condenser. Safety valves, in conjunction with the turbine bypass valves, are sufficient to relieve all the power being generated by the reactor after the recirculation pumps are tripped (80% capacity as opposed to -55% power). For sequences in which the main condenser is not available, however, an additional reduction in power is desirable to assure that reactor power remains below the capacity of the safety valves. This additional power reduction is provided by tripping the feedwater pumps. Feedwater pump trip (FWT) occurs on high reactor pressure (1400 psig). The effect of tripping the feedwater system is a loss of sub-cooling at the core inlet and a lowering of level and hence flow through the reactor. These conditions result in additional void generation and further power reduction.

Given these Pilgrim specific design features, success at the recirculation pump trip heading implies the following: for sequences in which the main condenser is available, trip of either of the recirculation pumps is required (to trip the recirc lation pumps, either the field breaker or the drive motor breaker to each recirculation pump motor generation set must open); for sequences in which the main condenser is not available, it is assumed that both recirculation pumps and all three feedwater pumps must be tripped to attain the necessary reduction in power. Failure of the appropriate combination of recirculation and feedwater pumps to trip is assumed to lead to power levels above the capacity of the safety valves and subsequent failure of .the reactor vessel on over-pressure. In fact, analysis indicates that failure of feedwater pump trip will result in a steam flow rate which is near but slightly below the safety valvcapacity. Tripping the feedwater pumps results in additional margin on the safety valve capacity providing further assurance that reactor over-pressure conditions do not occur.

RPV Pressure Control (SRVs)

RPV pressure control success criteria with safety relief valves vary depending upon the initiating event. Events in which the main condenser remains available and steam flow is occurring through the turbine bypass valves require only three to four safety relief valves to relieve the steam not being directed to the main condenser. Events in which the main condenser is unavailable and feedwater is in operation require five to six safety relief valves to open in addition to the trip of both recirculation pumps. All three feedwater pumps should trip to provide additional margin on safety relief valve capacity, as noted above. Failure of a safety relief valve to open during a reactor trip failure without the main condenser, or failure of multiple safety relief valves to open with main condenser availability is assumed to lead to reactor vessel over-pressure failure.

Coolant Inventory Make-up (QU)

The HPCI and feedwater systems are each as umed to be adequate high volume high pressure means of coolant inventory makeup during the initial stages of an ATWS event from 100% power with no control rod insertion. EOP 02 requires the operator to stop and prevent all RPV injection from all sources except CRD and SLC if:

a) reactor power is above 3%, and

b) torus water temperature is above the Boron Injection Initiation Temperature Curve, and

c) drywell pressure is above 2.5 psig or any SRV is open, and

d) water level is above top of active fuel.

He is directed to lower water level until one of these values recovers below the level listed above, and then maintain water level between that level and -155 inches. He is not allowed to raise level above that point until the hot shutdown boron weight of SLC solution has been injected.

By itself, HPCI is sufficient to keep the reactor level above the top of the active fuel following a failure to trip from full power. Operation of the feedwater system can have a number of competing effects on the outcome of the transient depending on the initiating event. For events in which the feedwater system remains in operation or is returned to service early in the event, feedwater operation is capable of maintaining reactor inventory.

If the feedwater system is returned to service following a high pressure reactor trip, care must be taken by the operator not to raise the water level back to the point that reactopower causes the plant to exceed the values listed above. The operators are trained to anticipate the need to lower level during such an event as o posed to raising level in an uncontrolled manner, making this situation unlikely.

The HPCI system is a lower volume system than feedwater, capable of providing on the order of 4000 gpm at elevated reactor pressure (approximately 1/4 normal feedwater flow during full power operation). The lower flow capacity is such that during an ATWS, HPCI is not capable of returning reactor level and power to the point that safety relief valve capacity is approached. A heading for HPCI level control is therefore not included in the ATWS event tree as it is for feedwater. TRACG analyses indicate that the equilibrium level reached with HPCI during an ATWS from full power is less than the reactor low-low watelevel setpoint. As a result, it is assumed that HPCI is capable of maintaining adequate core cooling during an ATWS without feedwater but will not prevent automatic ADS actuation.

SLC Injection (C2,& C4)

Whether inventory makeup is being accomplished by high or low pressure injection systems, reactor shutdown following a failure to insert control rods requires initiation of Standby Liquid Control. Successful SLC injection requires the operation of either of two SLC pumps and may also require additional power level reduction by way of reduction in reactor water level if there is a significant delay in SLC initiation. Each pump is capable of injecting the equivalent of 86 gpm of 13% sodium pentaborate solution into the vessel.

The effectiveness of SLC injection is represented by several headings in the ATWS event tree.

The first heading examines the potential for mechanical or electrical failure of SLC (heading C2). Failure of SLC due to equipment problems is assumed to lead to the need for alternate methods of injecting boron as outlined in plant emergency operating procedures.

The remaining heading for SLC actuation represents the effects of injecting sufficient boron to shutdown the reactor at successively later periods of time during the transient. The points in time considered important for the purpose of this event tree include SLC injection in time to take action to prevent challenging the containment. Depending on how late in the transient that SLC is initiated, its success in preventing containment over-pressurization may also depend on taking actions to lower reactor level in order to reduce power.

SLC initiation is needed to achieve reactor shutdown in order to prevent challenges to containment integrity. The second SLC initiation heading reflects action to inject SLC in this time frame (C4). Operator action early during this period requires only that SLC be initiated. For events in which high pressure injection systems are in operation, calculations indicate that approximately 20 minutes are available to initiate SLC if the reactor remains at high pressure while 20 to 30 minutes exist if depressurization to low pressure systems occurs.

As SLC is delayed later in the period, operator action to lower reactor water level to limit reactor power and the amount of energy being directed to the suppression pool will be required in addition to initiation of SLC. With water level/reactor power control, operator action may be delayed an additional several minutes and still be successful in shutting down the reactor during sequences in which the primary system is isolated and all the energy is being directed to the suppression pool. Since the same operators will be injecting SLC as will be controlling water level, no credit is given in the PRA for water level control if the operator fails to inject boron in event C_4.

Safety Valves Reclose (P)

For sequences in which SLC is injected sufficiently early that the reactor is shutdown prior to the occurrence of containment conditions requiring reactor depressurization, it is important to examine the potential for inadvertent depressurization of the reactor through a stuck open safety relief valve. An open SRV will necessarily result in reactor depressurization following shutdown and the need for low pressure injection systems.

Low Pressure Injection (V)

The low pressure heading of the ATWS event trees has two success criteria depending on the status of reactor shutdown. For those events in which SLC was actuated early and the reactor is effectively shutdown prior to the need for low pressure systems, the success criterion for this heading is the same as for transients in which reactor trip was successful. Systems available to provide low pressure inventory makeup include condensate, LPCI, core spray and the fire system.

For sequences in which reactor depressurization occurs prior to reactor shutdown it is assumed that water level control with low pressure systems must occur with the reactor still at power. Analysis of this situation has been performed by GE using TRACG [Ref C.S-6]. From these analyses it has been determined that uncontrolled injection from low pressure injection systems is a self limiting event that will not result in large power excursions or pressure spikes to the point that core or vessel integrity is threatened. However, it is highly likely given that the reactor is not yet shutdown that the operator will be attempting to control low pressure injection rates to reduce reactor power in accordance with the power/level control contingency of the emergency procedures. Control of reactor level in this manner with systems such as LPCI and core spray is assumed to be more difficult than at decay heat loads. Because level control may be at or near the top of the active fuel during these scenarios, a slightly greater possibility of core damage is assumed under these conditions as compared to transients in which decay heat loads govern the need for injection.

Boron Dilution (UH)

Even following successful reactor shutdown, ATWS scenarios entail operating considerations in addition to those which would occur during more routine transient events. Among them is the need for the operator to limit vessel water injection to prevent uncontrolled injection from the low pressure systems into the reactor. Two possible failure modes are considered at this node:

- o SLC has been initiated and the reactor is subcritical. Injection in an uncontrolled manner with the low pressure systems can result in washing boron from the core. A rapid reactivity excursion may result.
- o A slower transient in which the boron is washed from the reactor vessel due to extended operation of low pressure injection systems without level control.

The purpose of this heading is to examine the potential for this event and the need for operator action to preclude its occurrence.

Containment Heat Removal (W). Containment heat removal in the ATWS event trees takes several forms depending on the status of SLC and its success in attaining reactor shutdown.

The first version of containment heat removal is similar to the containment pressure control heading of the transient event trees. Systems such as the main condenser, RHR, and containment sprays have the ability to control containment pressure. This definition of the containment heat removal heading applies to those sequences in which shutdown was effected prior to exceeding containment limits. The need for these systems will occur much earlier following an ATWS event due to the fact that early in the event the energy being directed to the suppression pool is that associated with reactor power as opposed to decay heat. As a result, little time is assumed to be available for repair of heat removal systems following an ATWS.

The other version of this heading involves using containment heat removal systems to remove energy at rates associated with reactor power. For sequences in which the main condenser is available, success at this heading requires that the reactor water level be lowered to the point that all of the energy being produced in the reactor is being directed through the turbine bypass valve. This action can provide an effectively unlimited amount of time for operator action to initiate SLC. However, since the same operators will be injecting SLC as will be controlling water level, no credit is given in the PRA for water level control if the operator fails to inject boron in event C_4.

For sequences in which the main condenser is not available, success at this heading would imply that injection to the vessel be terminated such that reactor water level would be lowered to the point that reactor power would drop to below the capacity of other available heat removal systems (such as RHR). Assuming the ATWS began with the reactor at full power, the reactor water level associated with this heat generation rate is less than the minimum steam cooling level. Little chance of success is given to this heading as current emergency operating procedures do not suggest this mode of ATWS mitigation.

C.S.S Operator Error Probabilities

There may be a wide spectrum of operator actions which contribute to the operator success in the implementation of the reactivity control procedures. The evaluation of the operator error probabilities for failure to perform these actions is dependent upon the following:

- o Time available for action to be performed

- Indication of the need for action
- Stress on the operator
- Successful performance of the associated actions
- Number of members of the crew involved in the decision making and dependencies among them
- Degree of difficulty of the operation
- Hesitancy in performing the action.

Each of these can be referred to as a performance shaping factor duririg the overall assessment of the potential for operator error.

The values used in the Pilgrim ATWS event trees are based on guidelines presented in Appendix A, and in Appendix A of the IPEM. The dominating factor in estimating operator error probabilities in this evaluation is the time available to the operator to successfully accomplish the actions required by the emergency procedures. Examples of the operator actions that are important during ATWS, and their probabilities, are listed below:

Action	HEP	Comment
Operator fails to depressurize	.09	Governed by the time frame
SLC injection prior to 260F w/o level control	.04	No main condenser action must occur: within 13 min
	.04	Main condenser available, action must occur within 12 min

C.5.6 Transient Initiator Frequency

The anticipated transient initiators are the same types as considered in the other sections of the IPE. These transient types include turbine trips (TT), loss of feedwater (Tp), MSIV Closure (TM), loss of condenser vacuum (Tc), loss of offsite power (TE), and IORV (T_1).

The frequencies of these initiators are presented in Appendix A.

The majority of transient initiators at Pilgrim result from turbine trips or lead to turbine trips. However, two principal distinctions made in this analysis are turbine trips which proceed with normal systems available (i.e., the condenser available as a heat sink) and those turbine trips in which the condenser is unavailable due to MSIV closure or the closure of the turbine bypass valve. These initiating events were the initiating events for the ATWS event trees. The various modes of failure to scram, and the subsequent events are captured in the event trees. The difference between isolation and non-isolation ATWS events are accounted for in the success criteria for the various safety functions.

C.5.7 ATWS Event Trees

This subsection takes the qualitative and quantitative information presented in Sections C.5.1 through C.5.6 and uses it to quantify the ATWS event trees which are used for the Pilgrim IPE.

C.5.7.1 MSIV Closure Initiator ATWS Event Tree

General Discussion

The MSIV closure class of transient initiators are an important class of accident initiator because they adversely affects the normal heat sink. Figure C.5-1 is the event tree for the MSIV isolation type initiating events for ATWS accident sequences. The initiator frequencies are determined from operating experience includes those postulated turbine trip with failure to scram sequences which may become isolation events. These initiating events are included in the MSIV closure ATWS initiator event tree.

The operator response to an ATWS initiated by an MSIV closure must be relatively rapid as all of the energy being produced in the reactor is being directed to the suppression pool. A need to inhibit ADS operation in some way is required during this event as the feedwater pumps will trip on high reactor pressure causing reactor water level to drop below the low-low setpoint.

Even given the operator's action to reduce power by lowering the reactor water level, substantial amounts of heat will still be transferred to the suppression pool until the boron is injected and sufficient mixing occurs. to reduce the heat load to decay heat levels. Therefore, the operator has less time for action in the MS!V closure initiated transient than in the turbine trip with bypass case, for example.

System heading success criteria for the MSIV closure initiator include:

Heading	Success Criteria
RPT	Both recirculation pumps and all feedwater pumps must trip
SRVs	Five to six safety relief valves are required to open
HPCI	Must operate to maintain reactor water level
SLC	Must be initiated within 20 minutes (20-30 minutes with depressurization to low pressure systems)

C.5.7.2 Non MSIV Closure ATWS Event Tree

General Description

The postulated effects of ATWS on the turbine trip are evaluated in Figure C.S-2 for cases with the bypass valves available. Implicit in the construction of the event tree for a turbine trip with bypass is the fact that feedwater is initially supplying coolant injection to the reactor.

For those events which continue as turbine trip events with feedwater available, the ability of the plant to cope with such events is good because use of the normal heat sink can potentially provide substantial time for the operator to take appropriate action to initiate SLC without challenging containment, while also maintaining adequate coolant injection.

System heading success criteria dependent on the turbine trip initiator include:

Heading	Success Criteria
RPT	Only one recirculation pump is required to trip to reduce power to below bypass valve and SRV capabilities
SRVs	Three to four safety relief valves are required
Feedwater	Normal operation precludes automatic actuation of the ADS. Feedwater operation will not result in power levels in excess of the safety relief valve capacity
SLC	Must be initiated within 18 minutes (but can be postponed indefinitely with reactor level control within first 25 minutes)

References

C.5-1 General Electric Company, NEDC-31425 (Draft) "Evaluation of ATWS at the Pilgrim Nuclear Power Station," 1988.

C.5-2 TRACG Analysis Results (Draft) presented by General Electric for Pilgrim ATWS events, September and October 1987.

C.5-3 Pilgrim Nuclear Power Station System Lesson Notes, Main Turbine.

C.5-4 R.E. Henry to C.S. Brennion, June 22, 1987, Comparison of TRACG Analysis with MAAP (modified Chexal-Layman).

C.5-5 Pilgrim ODYN Analysis, SEP Meeting of March 27, 1987.

C.5-6 General Electric Company, TRACG Analysis Assuming Uncontrolled Low Pressure Injection (unpublished).

C.5-7 S. Mintz to J.E. Torbeck, September 17, 1987, Time to Hot Shutdown for ATWS Events in Pilgrim.

C.5-8 BWR IPE Methodology, IDCOR Technical Report, T86.3B1, Vol. 1 & 2, March 1987.

C.5-9 Andersen, V.T., and Burns, E.T. "Human Error Probabilities in the BWR Individual Plant Evaluation Methodology," Proceedings, 1988 IEEE Conference on Human Factors, June 1988.

C.5-10 Burns, E.T., "Reassessment of the Scram Failure Probability", TENERA, L.P., document S-470058-037, September 1988.

Table C.S-1

SUMMARY OF REQUIRED FUNCTIONS AND SYSTEMS AVAILABLE FOR POSTULATED ATWS SEQUENCES

FUNCTION	SYSTEMS USED TO FULFILL NECESSARY FUNCTIONS	ASSUMED RESULT IF FUNCTION FAILS
Insert Adequate Negative Reactivity	ARI (electrical RPS) or SLC	High Containment Pressure
Hi Pressure Coolant Injection Make-up	FW or HPCI	Demand on Low Pressure Systems
Containment Heat Removal	PCS (at Power) and PCS or RHR (Reactor Shutdown)	High Containment Pressure
Short Term Pressure Control	Safety Relief & Turbine Bypass Valves and RPT (and FWT for MSIV events)	LOCA, Possible Degraded Core, High Containment Pressure
Low Pressure Coolant Injection Makeup	1 LPCI or 1 Core Spray or 1 Condensate	Inadequate Core Cooling

Table C.S-2

DEFINITIONS OF FUNCTIONS OF EACH SYSTEM APPLIED
IN THE ATWS EVENT TREE DEVELOPMENT

DESIGNATOR	SYSTEM	FUNCTION
C_M, C_E	Reactor Protection System	The RPS has been divided into electrical and mechanical functions for the study. The mechanical function includes the operation of the CRD hydraulic system, the physical insertion of a sufficient number of control rods to bring the reactor subcritical, and other mechanical components as required. The electrical portion of the RPS includes generation of a scram signal through the logic, and the de-energizing of the scram solenoid valves.
C_2, C_4	Poison Injection	Termination of reactor poweris required to assure containment and core integrity. Following failure to scram this function is accomplished through initiation of SLC. The C2 heading is used to evaluate the potential for failure of SLC due to mechanical or electrical cause, C4 is used to evaluate the potential for failure to inject SLC prior to the pool temperature exceeding 260F (level control is not credited if SLC injection failed)..

Table C.S-2 (Continued)

DEFINITIONS OF FUNCTIONS OF EACH SYSTEM APPLIED IN THE ATWS EVENT TREE DEVELOPMENT

DESIGNATOR	SYSTEM	FUNCTION
R	Recirculation Pump Trip (plus Feedwater pump trip when required)	This system is designed to be completely diverse from the RPS (both electrically and mechanically). The RPT is intended to trip the recirculation pumps which will reduce the flow through the core and lead to reduced moderation in the core and lower core power level. Feedwater pump trip can be used to further reduce core power during isolation transients by reducing core inlet sub-cooling and lowering reactor water level.
K	Alternate Rod Insertion (ARI)	This system is completely diverse to the RPS. ARI is considered effective in terminating events in which the failure to scram is due to electrical causes. This is accomplished by depressurizing the pneumatic supply to the scram values.
M	SRVs Open	Successful pressure control requires sufficient SRVs open to maintain reactor pressure below reactor vessel pressure limits. This requires five to six safety values during events initiated by isolation of the primary system and only 3 to 4 SRVs for events in which the main condenser is available.
p	SRVs Close	This event includes effect of a single relief valve remaining stuck open (SORV) during the transient. Depressurization to low pressure systems is assumed to occur once reactor power is reduced.

Table C.S-2 (Continued)

DEFINITIONS OF FUNCTIONS OF EACH SYSTEM APPLIED IN THE ATWS EVENT TREE DEVELOPMENT

DESIGNATOR	SYSTEM	FUNCTION
QU	Coolant Injection	The coolant injection function requires sufficient water in the reactor vessel to maintain the core covered. The methods available to perform this function vary with the transient. Feedwater or HPCI are sufficient to prevent core uncovery during power generation. If the RPV is isolated, only HPCI is available.
UH	Operator Controls Level	The operator is required to control level and prevent overflow of the reactor to the suppression pool following SLC injection. This prevents dilution of boron in the primary system and any subsequent return to power. Also, inadvertent actuation of ECCS is included in this heading which might result in washing boron from the core and a prompt critical situation.
W	Heat Removal	Heat is removed from the reactor through either the main condenser or steam relief through the safety relief valves to the suppression pool. However, the heat must also be removed from the suppression pool or a failure of the containment could result from over-pressurization. With the reactor at power, only the main condenser is assumed to be capable of relieving the energy being produced in the core. After reactor shutdown, the main condenser, RHR, spray from external sources and the vent are capable of maintaining the containment below design pressure.

(HIGHLIGHTING ADDED)

Pilgrim Watch
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EXHIBIT 2

U.S. Reactor Owners See Retrofits to Avoid Blasts

By Mehul Srivastava, Jim Polson and Rachel Layne - May 19, 2011 -Bloomberg

Entergy Corp. (ETR), the second-largest U.S. nuclear operator, and Duke Energy Corp. (DUK) said the industry may need to retrofit reactors or bolster safety systems after a pressure-relief system failed in Japan, contributing to the worst nuclear disaster since Chernobyl.

Venting systems at Tokyo Electric Power Co.'s Fukushima Dai-Ichi reactors were designed to allow engineers to release pressurized gas into the atmosphere to avoid dangerous hydrogen explosions. The systems were installed in the U.S. and in Japan after the partial core meltdown at Three Mile Island Unit 2 in 1979.

The vents were built into General Electric Co. (GE) boiling- water reactors, including the stricken Japanese plant that was rocked by at least two blasts blamed on trapped, exploding hydrogen. A conclusion that the vents were at fault may add costs for nuclear-power generators as politicians from Germany to India question the safety of atomic energy.

The hydrogen explosions in Fukushima "call the modification into question," said Tony Roulstone, who directs the University of Cambridge's masters program in nuclear technology in England. "If these vents don't work, then the design looks wrong. Fixing it will take some design work, but won't be wildly expensive."

The U.S. Nuclear Regulatory Commission is "looking at effectiveness of containment venting strategies," Charlie Miller, head of the post-Fukushima safety review, said at a May 12 agency meeting. The vent system is "worthy of a look" after the disaster, he said.

'Fully Expects'

Entergy "fully expects" the U.S. Nuclear Regulatory Commission to order new equipment installed and new procedures to be adopted as a result of the accident in Japan, said Jim Steets, a spokesman for the New Orleans-based company that owns 11 reactors. Exelon Corp. owns the largest number of U.S. reactors.

Operator error or lack of power at the facility may explain why venting systems didn't work at Fukushima. "There are multiple explanations for failure of venting systems in Japan to prevent hydrogen explosions," Michael Burns, another Entergy spokesman, said in an e-mailed statement yesterday.

The meltdown at the Dai-Ichi plant in Japan began after it was damaged by an earthquake March 11. A subsequent tsunami took out the primary power supply, and diesel generators either worked only briefly or also were flooded, according to a May 18 interim review of the incident by the U.K.'s Office for Nuclear Regulation.

Adequate Power?

Because loss of power needed to open valves may have contributed to the hydrogen explosions, "one of the things we expect to be testing with our own units is, do we have adequate auxiliary power, could it withstand fire or flood?" Jim Rogers, chairman and chief executive officer of Charlotte, North Carolina-based Duke Energy, told Kathleen Hays on Bloomberg Radio "The Hays Advantage" yesterday.

It's still unclear whether Japanese engineers opened the vents to release pressure in the containment building, according to the U.K. report.

"It is certainly possible that inadequacies in the venting routes may have featured in the devastating explosions that were seen in Reactor Units 1 and 3," the review found. "This may indicate that more attention should have been given in the design and safety assessment to the robustness of the venting routes."

The hardened-vent systems were designed by a consortium of reactor owners advised by GE. U.S. regulators recommended installation of the systems in a September 1989 letter to owners.

Meeting Requirements

"At this point, we still believe that under design-basis conditions, the hardened-vent system would operate as designed and meets the current regulatory requirements," said Jim Klapproth, chief consulting engineer for GE Hitachi, the nuclear-power joint venture of GE and Hitachi Ltd. "We are evaluating the situation in Japan to determine if there's any difference in the design or operator actions."

Scott Burnell, a spokesman for the U.S. Nuclear Regulatory Commission, declined to comment on any similarities between the venting and suppression systems at the Japanese plant and U.S. reactors.

It's "still too early to be drawing conclusions on either events at Fukushima or possible recommendations from our task force" that's leading a safety review of U.S. reactors, Burnell said in an e-mail yesterday.

U.S. reactor operators are able to decide for themselves whether to vent radioactive gas when reactor pressure is high, said Carrie Phillips, a spokeswoman for Atlanta-based Southern Co. (SO), which operates two GE reactors with containment systems similar to the Japanese plant.

Hatch Tests

Southern and U.S. regulators tested the venting system at its GE-designed Hatch reactors in Georgia in March and they worked as intended, Phillips said.

Explosions at the Japanese reactors destroyed the buildings, satellite photos show, making it more complicated for engineers to restore power and bring the units under control. Thousands of people were evacuated after radiation levels near the plant soared.

More information is needed about the explosions at Dai- Ichi, including a still unexplained blast at Unit 2, said David Lochbaum, the director of the nuclear safety project at the Cambridge, Massachusetts-based Union of Concerned Scientists, a nonprofit.

“We haven’t yet heard what the most likely scenario is,” said Lochbaum. “There are some signs that the vents may not have worked.”

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EXHIBIT 3

(Attached separately, June 1, 2011)

Pilgrim Watch
June 1, 2011
EXHIBIT 4

BELLONA http://www.bellona.org/articles/articles_2011/fautly_hydrogen_vents

Faulty hydrogen vents at Fukushima Daiichi spell trouble for US nuclear plants as well

Part of: Nuclear meltdown in Japan

The remnants of reactor No 1 (left) and No2's buildings following the explosions.
TEPCO

Logs from the hours following the 9.0 magnitude earthquake and tsunami that struck the Fukushima Daiichi nuclear plant and interviews with workers show that US-designed hydrogen vents malfunctioned, leading to devastating pressure buildups and explosions at three of the plant's six reactors, causing massive released of radiation. Charles Digges, 18/05-2011

The failure of the vents calls into question the safety of similar nuclear power plants in the United States and Japan.

After the venting failed at the Fukushima Daiichi plant, the hydrogen gas fueled explosions spewed radioactive materials into the atmosphere, reaching levels about 10 percent of estimated emissions at Chernobyl – or 5 million Curies – according to Japan's nuclear regulatory agency.

Hydrogen venting – though it releases radiation – was critical to reliving intense pressure building up in reactor Nos 1, 2 and 3 at the plant after the March 11 tsunami knocked out all power to the cooling systems to the reactors. As cooling water stopped flowing through the reactors, they began to dangerously overheat.

Bellona nuclear physicist Nils Bøhmer said that in such circumstances, venting is the lesser of two evils.

"If [the venting] had worked it would have minimized the accident," he said. "But since it failed, the reactors exploded, releasing far more radioactivity into the environment."

Tokyo Electric Power Co (TEPCO) has now finally confirmed that meltdowns occurred in reactor Nos 1, 2 and 3 all of which suffered hydrogen explosions that blew away their containment buildings – the first line of defence against radiation leaks – thanks to the failed General Electric hydrogen vents.

"These disclosures mean that TEPCO has known since the beginning that the situation was far more sever than they told the public and even the government," said Bøhmer.

"There also seems to have been a tremendous amount of human error involved, particularly delayed decisions on behalf for the government, and they will be in the hot sea for this," he said

Vents in already flawed reactors found faulty by Japan

American officials have said reactors in the United States would be safe from such disasters because they were equipped with new, stronger venting systems.

But TEPCO, which runs the Fukushima Daiichi, now says that it had installed the same vents years ago.

Over the course of the ongoing crisis, several people involved in designing the GE Mark 1 reactor in the 1960 – which accounts for five of the six at Fukushima Daiichi, several more throughout Japan, and 28 in the United States – have come forward to say that as early as the 1970s the reactor design had flaws related to the small size of its containment chamber, which would lead to catastrophic pressure buildups.

Several of the initial engineers who reviewed the Mark 1 design resigned from GE when the technology giant decided to press ahead with the project despite known defects.

Japanese government officials have also suggested that one of the primary causes of the explosions was a several-hour delay in a decision to use the vents, as Tokyo Electric managers agonized over whether to resort to emergency measures that would allow a substantial amount of radioactive materials to escape into the air.

But when the decision to use the vents finally came nearly a day after the tsunami had overwhelmed the plant, it was found they did not work, according to TEPCO records and documents.

Record release catalogs dramatic failures on day of quake

A release of documents that catalog the actions of workers at the plant following the disaster obtained by the New York Times, as well as Bellona Web email interviews with experts in Japan, show the most comprehensive evidence to date that mechanical failures and design flaws in the venting system contributed to delays in the venting procedure that might have saved the reactors and blocked massive releases of radiation into the environment.

The documents paint a dramatic picture of increasing desperation at the plant in the early hours of the disaster, as workers who had finally gotten the go-ahead to vent realized that the system would not respond to their commands.

Venting would have prevented worse disaster

While venting would have allowed some radioactive materials to escape, one official in Japan told Bellona Web in an email interview that those releases would have been far smaller than those that followed the explosions. Bellona's Bøhmer, in the early days of the crisis, confirmed this to be the case, saying venting was a necessary evil to avert more dire catastrophe.

The blasts were also responsible for breaches in containment vessels – also a theory forwarded by Bellona's Bøhmer early on – which have complicated efforts to cool the fuel rods, contain radioactive leaks from the site, and which have led to fuel melts through the reactor cores at No 1, 2 and 3.

TEPCO has said that one reason the GE designed vents did not work is because they relied on the same source of power as the rest of the plant – the aged backup generators in basements that were disabled by an inundation of water. TEPCO also reported that the earthquake could have damaged valves that are a part of the hydrogen venting system.

Though the valves are operated from the control room, they are also designed to be operated manually. But by the time the government green-lighted the hydrogen release more than 17 hours after the tsunami hit, radiation levels made the manual release too dangerous to approach, said the TEPCO logs, according to the New York Times.

Fukushima must be lesson one for the nuclear industry

"The Japanese experience should be a learning experience for the world," said Bøhmer. "That lesson should be that nuclear industries worldwide must undertake enhanced emergency training – where the industry in step with the kinds of disasters that can occur, then the impact of this disaster might have been lessened”

David Lochbaum of the Washington DC-based Union of Concerned Scientists agreed.

“Japan is going to teach us lessons,” he said. “If we’re in a situation where we can’t vent where we need to, we need to fix that.”

Whatever causes the malfunction of the venting system will necessitate a review in both the United States and Japan to determine whether the venting systems need retrofitting.

Officials from General Electric would not comment on the vent situation when reached by Bellona Web for comment today.

According to TEPCO logs as cited by the New York Times, the gravity of the crisis was evident as soon as the tsunami breached the plant’s sea walls, which many experts have said were too low in the first place.

Building pressure not released until day after quake

Within 12 hours of the quake, pressure inside the reactor had reached twice what it was designed to withstand, raising fears that the vessels that house fuel rods would rupture, setting a possible meltdown in motion. Such high pressures, as Bellona nuclear physicist Nils Bøhmer suggested, also made pumping additional cooling water into the reactor – something that was noted on the TEPCO log.

According to the account of an anonymous source who spoke with the New York Times, Japanese government officials ordered TEPCO to ventilate, but TEPCO wished to continue weighing other options. Fukushima director Masao Yoshida wanted to vent as soon as possible, and engaged in a heated shouting match with TEPCO Vice President Sakae Muto.

Venting did not begin until 17 hours after the government ordered it – a whole day after the quake hit – and six hours after TEPCO management gave the go ahead, according to TEPCO logs.

As efforts to manually open the vents at reactor No 1 failed because of spiraling radiation level, workers at reactor No 2 also tried to manually open its venting system. Pressure in the reactor did not fall though, so it was unclear whether the venting had succeeded the records show. At reactor No 3, manual attempts to open the vents failed, said the records.

Because of the venting failures, the explosions began. The first reactor to explode was reactor No 1, on Saturday, March 12, followed by reactor No 3 on Monday. Next was reactor No 2 early on Tuesday morning.

With each explosion, radioactive materials soared into the air, forcing the evacuation of tens of thousands of earthquake survivors living near the plant, contaminating crops and sending a plume of radioactive isotopes as far as the United States and Europe within days.

US says its vents safe, but Fukushima proves otherwise

TEPCO only in recent days has acknowledged that damage at the plant was worse than previously thought, with fuel rods most likely melting completely at reactors Nos 1, 2 and 3 in the early hours of the crisis, escalating the danger of more catastrophic releases of radioactive materials. The company also said new evidence confirms that at Reactor No 1, the pressure vessel, the last layer of protection, was broken and leaking radioactive water.

The improved venting system at the Fukushima plant was first mandated for use in the United States in the late 1980s as part of a "safety enhancement program" for boiling-water reactors that used GE's Mark I containment system, which had been designed by General Electric in the 1960s. Between 1998 and 2001, Tokyo Electric followed suit at Fukushima Daiichi, the records indicate.

The GE Mark 1 design has been cited by Japanese and American engineers as being flawed from the its first designs on the drawing board, and is fast becoming in industry opinion the Western version of the fatally flawed Russian RBMK - 1000, the reactor that exploded at Chernobyl.

Whistleblowers ignored for four decades

Mitsuhiko Tanaka, an engineer who was responsible for implementing the GE Mark 1 design at the Fukushima Daiichi plant in the early 1970s, said in a recent interview that for 40 years, he and others knew the reactor design to be faulty because its containment vessel is too small to adequately withstand the kinds of pressure buildups that bore out at reactor Nos 1, 2, and 3.

And 35 years ago, Dale Bridenbaugh and two of his colleagues at General Electric resigned from their jobs after becoming increasingly convinced that the Mark 1 reactor, which they were reviewing, was so flawed it could lead to a devastating accident.

"The problems we identified in 1975 were that, in doing the design of the containment, they did not take into account the dynamic loads that could be experienced with a loss of coolant," Bridenbaugh told ABC News in March.

"The impact loads the containment would receive by this very rapid release of energy could tear the containment apart and create an uncontrolled release," said Bridenbaugh in televised remarks.

Tanaka, who says he turned his back on nuclear energy after Chernobyl in 1986, brought several specific concerns about Fukushima Daiichi’s reactor No 1 to the Japanese government in 1988 with no results. Two years later he made his concerns public in a book entitled “Why Nuclear Power is Dangerous.”

The red flags raised by the book remained ignored by the government, Tanaka said.

“Critics of the reactor design were totally ingored,” said Bellona’s Bøhmer.

“This means that regulations are not strong enough, and that regulators are far too close to their industry couterparts, which leads to obvious problems being ignored,” he said.

Bøhmer urged the nuclear industry worldwide to undertake stress tests of their reactors.

“It is vital that all nuclear regulators review their reactor safety, especially those reactors that are similar in design to the ones at Fukushima Daiichi,” he said.

<u>Exhibit 5</u>

(Attached Separately)

Pilgrim Watch
June 1, 2011
EXHIBIT 6

From the Bulletin Archives: Containment of a reactor meltdown

By Frank von Hippel | 16 March 2011

Article Highlights

- A filtered vent system could relieve the pressures inside a dangerously pressurized containment building by releasing some of its radioactive gases through a large filter system.
- The industry is concerned that accident mitigation techniques could be interpreted as tacit admissions that serious accidents can happen.

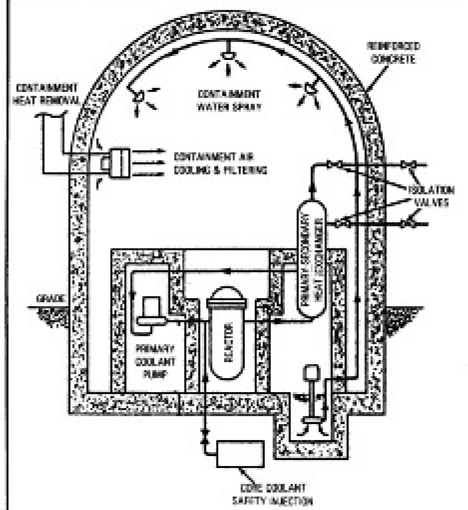
Editor's Note: Authored by Jan Beyea and Frank von Hippel, this article originally appeared in the September 1982 issue of the Bulletin.

Any good scientist or engineer believes implicitly in Murphy's law: "If something can go wrong, sooner or later it will go wrong." The US Atomic Energy Commission, which until 1975 had the responsibility for ensuring the safety of US civilian power reactors, had many good scientists and engineers involved in its work. And during its history it repeatedly considered the consequences of all the safety systems in a nuclear reactor failing, the fuel melting and the volatile radioactive isotopes in the fuel being released to the atmosphere.

The answer which came back from major studies in 1957 [1], 1965 [2], and 1975 [3] was always that the consequences could be very serious indeed. This finding underlined the importance of preventing nuclear reactor meltdown accidents. As a result, the Atomic Energy Commission and the Nuclear Regulatory Commission (NRC), its successor in the area of nuclear safety regulation since 1975, required so many redundant safety systems on nuclear power plants that both nuclear regulators and the nuclear industry became convinced that the likelihood of a reactor meltdown accident had been reduced to a negligible level.

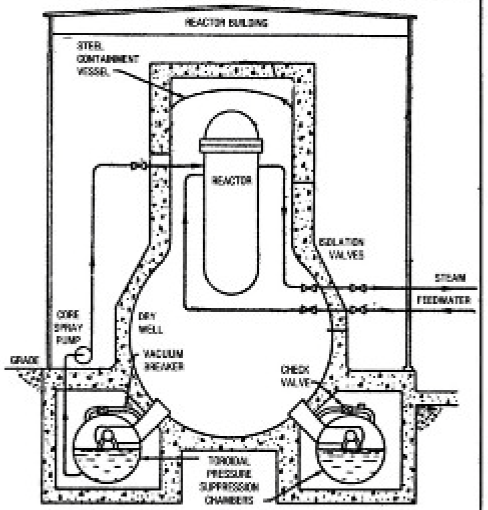
The massive failure of safety systems and the associated confusion which has occurred repeatedly at nuclear power plants since 1975 -- with serious damage resulting at Brown's Ferry in 1975 [4] and Three Mile Island in 1979 [5] -- have, however, thrown this confidence into question. Our purpose here, therefore, is to draw wider attention to the possibilities for increased public protection offered by the last barrier between the radioactivity released from a molten core and the outside world: the reactor containment building.

Figure 1
LARGE VOLUME PRESSURIZED WATER CONTAINMENT



Because of its large volume (about 60,000 cubic meters), this containment can hold all of the steam released in the first minutes of a loss of coolant accident. Subsequently steam pressure should be reduced by the containment water sprays.

Figure 2
SMALL VOLUME BOILING WATER CONTAINMENT



The combined volume of the dry well and the connected free space over the pressure suppression pool is only one eigth that of the containment shown in Figure 1. Steam from the dry well bubbles through the water in the pressure suppression chamber and is condensed. This could prevent overpressurization by steam but not by other non-condensable gases such as hydrogen and carbon dioxide.

Source: T.J. Thompson and J.G. Beckerly, *The Technology of Nuclear Reactor Safety*, vol. 2, chap. 21 (Cambridge, Mass.: MIT Press, 1973).

The containment. Reactor containment buildings are both massive and well-equipped (Figures 1 and 2). Most are designed to withstand internal pressures of three to four atmospheres and may maintain their integrity at more than six atmospheres internal pressure. They also have water sprays, water pools or compartments full of ice-whose purpose is to reduce pressures by removing steam from the containment atmosphere.

Reactor containment buildings today are not designed to contain a reactor core meltdown accident, however. Their "design basis accident" is a loss-of-coolant accident in which large amounts of volatile radioisotopes are released from a temporarily over-heated core, but in which the uncontrolled release of energy from the core into the containment atmosphere is terminated by a flood of emergency core cooling water before an actual meltdown occurs. This is essentially what happened during the accident at Three Mile Island although, due to various errors, the core remained only partially cooled for a period of hours.

The threat of overpressurization. If for any reason the emergency core cooling system were not effective and a core meltdown occurred, the build-up of internal pressure in a sealed reactor containment building could rupture it within a matter of hours. The threat would come from steam, hydrogen and other gases.

For an extended period of time after a reactor shutdown, the radioactive fission products in a reactor core generate heat at a rate great enough to turn hundreds of metric tons of water into steam per day (Figure 3). It would take only about 300 metric tons of steam to increase the pressure inside even a large (60,000 cubic meter volume) Three Mile Island type of containment building by about ten atmospheres. It is apparent, therefore, that unless the containment cooling system operates reliably and effectively to keep this steam pressure from building up, the containment will quickly be overpressured by steam alone [6].

Hydrogen is another potential contributor to the pressurization of the containment. It is produced when water or steam comes into contact with a metal which binds oxygen so strongly that the metal can take oxygen away from water molecules. Because it absorbs relatively few neutrons, one such metal, zirconium, is the structural material of choice used in the cores of water cooled reactors. Zirconium starts reacting rapidly with steam at temperatures above 1,100 degrees C. About one half the zirconium in the core of Three Mile Island Unit No. 1 was oxidized during the accident there [7].

For a small volume (boiling water reactor type) containment, the mere pressure developed by the amount of hydrogen generated at Three Mile Island would have been enough to raise the containment pressure by one to three atmospheres.

For a large volume containment, the principal hazard associated with the hydrogen would be fire or explosion, and in fact the hydrogen did burn at Three Mile Island. Fortunately, however, the initial pressure in the containment building was such that the containment was able to withstand the resulting pressure increase of about two atmospheres. Some existing reactor containments would not have withstood the pressure rise associated with the burning of this much hydrogen - even given an initially low pressure.

In small boiling water reactor containments the probability of a hydrogen fire is eliminated by "inerting" the containment with an atmosphere of pure nitrogen. This is not done, however, in ice condenser containments, which are designed to withstand much lower internal pressures than most other containments. On September 8, 1980, during a final review of the design of Sequoyah Nuclear Power Plants, Units 1 and 2 (which are equipped with ice condenser containments) the NRC'S watchdog, the Advisory Committee on Reactor Safeguards, pointed out in a letter to the Commission that: "For events involving more than 30 percent oxidation of the zirconium, hydrogen control measures may be necessary to avoid containment failure."

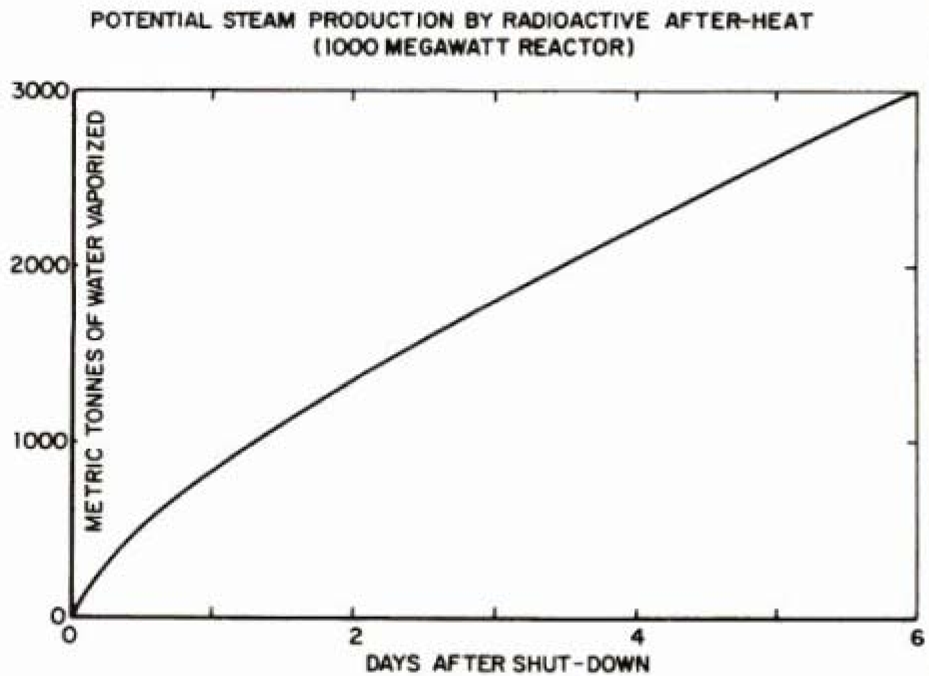
The remaining threat to containment integrity from overpressurization during a core meltdown accident would arise from the carbon dioxide and carbon monoxide liberated as the molten core melted its way down through the concrete basemat of the reactor building [8; 9].

This listing is sufficient to suggest why one of today's small volume reactor containment buildings would probably rupture during a core meltdown accident and why there is a significant, although less certain, probability of failure for a large volume pressurized water reactor type containment [3].

The regulatory response. The situation we have just described was first explored by an Atomic Energy Commission advisory committee in 1966 when the AEC was just beginning to license the construction of today's large commercial power reactors. The advisory committee recommended in its report, however, that the Commission should undertake only "a small-scale, tempered effort on [the] problems . . . associated with systems whose objective is to cope with the consequences of core meltdown. . . ." The committee did not recommend a crash program on the development of better containments because it felt that "to produce effective designs, if indeed feasible, might require both considerable fundamental research and practical engineering application." Instead, the committee advised the Commission that "for the time being, assurance can be placed on existing types of reactor safeguards, principally emergency core-cooling"[10].

The Commission accepted this advice and went ahead with the licensing of containment buildings whose integrity depended upon the successful functioning of emergency core cooling systems. A small amount of research was conducted for a time into the possibility of improved containment concepts. As the Commission certified time after time that existing containment designs were adequately safe, however, this research was phased out.

Figure 3



The figure shows the cumulative amount of water which would be evaporated by the radioactive after-heat generated after shut-down by the core of a typical modern 1,000-megawatt light water reactor. In the absence of heat removal from the containment, the steam pressure so generated would threaten the containment integrity within hours.

Periodically, the issue of improved containment designs was brought up by outsiders. For example, in 1975 the American Physical Society Study Group on Light Water Reactor Safety recommended that "more emphasis should be placed on seeking improvement in containment methods and technology" [11]. By that time, however, so many tens of billions of dollars had been invested in nuclear power plants which were already operating or in an advanced stage of construction, that the nuclear safety authorities were unwilling to question the basic safety design features of nuclear power plants.

This attitude was expressed in a memorandum written on September 25, 1972 by Joseph Hendrie, then Deputy Director for Technical Review of the Atomic Energy Commission. Hendrie was responding to the suggestion by a senior member of the Commission staff, Steven Hanauer, that because of the safety disadvantages of small volume containment buildings such as the General Electric boiling water reactor pressure suppression containment shown in Figure 2 and the ice condensor pressure suppression containment design being proposed at the time by Westinghouse, "I recommend that the AEC [Atomic Energy Commission] adopt a policy of discouraging further use of pressure suppression containments." Hendrie's response is reproduced in full below:

"With regard to the attached, Steve's idea to ban pressure suppression containment schemes is an attractive one in some ways. Dry containments have the notable advantage of brute simplicity in dealing with a primary blowdown, and are thereby free of the perils of bypass leakage.

However, the acceptance of pressure suppression containment concepts by all elements of the nuclear field, including Regulatory and the ACRS [Advisory Committee on Reactor Safeguards], is firmly imbedded in the conventional wisdom. Reversal of this hallowed policy, particularly at this time, could well be the end of nuclear power. It would throw into question the operation of licensed plants, would make unlicensable the GE and Westinghouse ice condensor plants now in review, and would generally create more turmoil than I can stand."

This memorandum became public as a result of a Freedom of Information Act suit by the Union of Concerned Scientists reinforced by Congressional pressure following Hendrie's appointment to the chairmanship of the Nuclear Regulatory Commission in 1977.

Filtered vents. As more and more nuclear power plants went into operation, the attention of those who wished to improve reactor containment designs turned to safety systems which could be "retrofitted" onto existing plants and to one specific idea in particular. This was a "filtered vent" system which could relieve the pressures inside a dangerously pressurized containment building by releasing some of its radioactive gases to the atmosphere through a large filter system. There the most dangerous radioactive species would be trapped before the filtered containment gases were allowed to escape. It would be relatively easy to add such a system onto an already completed containment building because the filter system could be installed in a separate building outside the existing containment building and connected to it through a large valve and underground pipe (Figure 4 [12]).

The installed cost of one of these systems has been estimated to be between $1 million and $20 million per reactor, an amount which is small in comparison with the more than $1 billion total cost of a modern nuclear power plant [13].

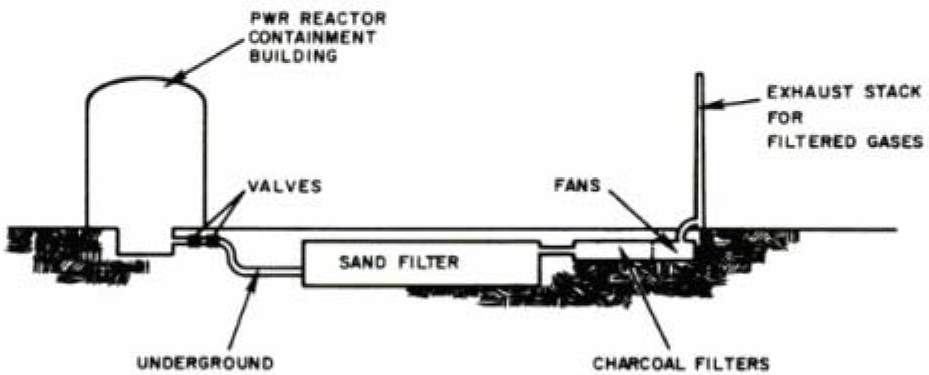
Despite these attractive aspects of the vented containment concept, the Nuclear Regulatory Commission proceeded to investigate it extremely slowly and cautiously. While the Commission's slowness can only be deplored, its caution is appropriate: prescriptions for nuclear safety, like those for drugs, should be both safe and effective and the staff has concerns in both areas.

In the area of effectiveness the staff's concerns focus on the possibility that in certain accident sequences the pressure buildup inside the containment might be so rapid that no exhaust system of realistic size could release gas fast enough to save it. The pressure rise associated with a hydrogen fire could, for example, be very rapid. Rapid increases in steam pressure could also occur within the containment of a pressurized water reactor as a result of sudden contacts between large amounts of molten core and large amounts of water.

According to current ideas, a melting reactor core would not drip away. Instead, it is believed more likely that a large fraction of the core would suddenly collapse and fall into the water remaining at the bottom of the reactor pressure vessel. In the past there has been concern in the reactor safety community about such an event resulting in a "steam explosion" violent enough to propel the top of the reactor vessel through the shell of a containment building. This concern has been downgraded in most recent studies but inside even a large containment building a rapid increase in pressure of about one atmosphere could occur.

Figure 4[12]

GENERAL ARRANGEMENT OF A PWR FILTERED VENT SYSTEM



If the pressure inside the containment climbed to dangerous levels, the isolation valves could be opened and some of the containment gas released through sand and activated charcoal filters.

In some scenarios, where the primary pressure system around the reactor core and its attached piping remain intact until the core actually melts through the pressure vessel, the melt-through would relieve the steam pressure in the primary system, with the result that certain water in the system would be mobilized and pour into the pressure vessel on top of the molten core. This could cause a rapid pressure rise of one to three atmospheres. And finally, after melting through the pressure vessel, the molten core could, once again, fall into a pool of water collected in the cavity below the vessel. Another rapid increase in pressure could then result [9,I].

There appear to be strategies that can reduce the threat of containment failures resulting from such pressure increases if in fact further analysis should establish this threat as a serious one: Indeed, the Nuclear Regulatory Commission is already beginning to require hydrogen "igniters" capable of burning any accumulating hydrogen in stages before concentrations can build to levels where a single fire will be intense enough to endanger the containment. The magnitude of some of the steam pressure rises associated with core meltdowns in pressurized water reactors could also be reduced by relieving the pressure in the primary system and flooding the containment building with water to a level which covers the pressure vessel when a meltdown appears inevitable. And, as we have seen, a filtered vent would make possible still another strategy: early venting so as to reduce the pressure base on which any subsequent sudden pressure increases would build.

The possibility of early venting is two-edged, however, because it requires a judgment that nothing else can be done to prevent a major release of radioactivity. That judgment might be wrong or the filtered venting system might even operate accidentally. The resulting releases would be dominated by the non-filterable radioactive noble gases which would contribute about one-thousandth of the cumulative radiation dose from an uncontained meltdown accident. The Commission's safety concern about filtered venting, therefore, focuses on the fact that a filtered vent system while offering some protection against large releases of radioactivity to the atmosphere would also increase by an uncertain amount the frequency of public exposure to very much smaller releases.

This concern is akin to the one about automobile seat belts -- that by slowing a passenger's escape from a vehicle in some accident situations, a seat belt could contribute to rather than prevent a death. But seat belts, as we know from statistics, save vastly more lives than they endanger. In the case of reactor core meltdown accidents we (fortunately) have no statistics yet. The Commission will, therefore, have to make a careful judgment. It seems likely that the final conclusion will be that, for a well-designed system, the reduction in the risks of large releases will greatly exceed the increased risk of small releases. At the current level of effort, however, it will take many years before thorough safety analyses have been concluded on each major type of reactor containment; and then more years may be taken up in conducting specific safety analyses on each plant chosen as a candidate for retrofit.

The industry response. In response to the Three Mile Island accident, the US nuclear industry could have put its own resources into investigating the possibilities for the reduction of radioactive releases following core melt accidents. Unfortunately, it did not. Instead, the industry mounted a concerted campaign to convince both the public and government that, even in case of containment failure, the resulting release of radioactivity to the atmosphere would be much less than has always been thought. In particular, the electrical utilities' Electric Power Research Institute published a study which concluded, in effect, that improved containments were not necessary [14].

The Institute report claimed that, even in the event of a core meltdown accident and a containment failure, "due to the solubility of the volatile fission product compounds and the aerosol behavior mechanisms, the off-site dispersion of radioactive materials (other than gases) following a major LWR [light water reactor] accident will be small." The electric utilities' public relations departments and the nuclear industry press sprang into action and advertised these claims with great fanfare, noting that "If findings like these are verified . . . it would go far toward deflating the doomsday predictions of anti-nuclear groups" [15]. The Nuclear Regulatory Commission, aside from a few staff comments in the trade press, expressed no public reservations concerning the significance of these claims, which tended to give them further credibility.

The Commission did, however, authorize an effort to examine the Institute's claims as a collaborative enterprise between Commission staff members and technical experts at three major national laboratories. In March 1981 this team stated in a draft report:

"The results of this study do not support the contention that the predicted consequences of the risk dominant accidents have been overpredicted by orders of magnitude in past studies. For example, the analysis in this report indicates that . . . 10% to 50% of the core inventory of iodine could be released to the environment" [16].

Under pressure from the industry, the Commission subsequently rewrote the summary language so that it no longer appeared to be a rebuttal to the Electrical Power Research Institute report. Nevertheless, the technical conclusions remained the same.

The role of public pressure. There are by now many examples of public pressure being required to offset the paralyzing effect of industry opposition to nuclear safety initiatives - especially when the purpose of the initiatives is to mitigate the consequences of nuclear reactor accidents. The industry is apparently concerned that the adoption of accident mitigation techniques, such as off-site preparations for emergencies and retrofitting containment buildings with filtered venting systems, could be interpreted by the public as tacit admissions that serious accidents can happen.

It was only after Congressional pressure developed for improved emergency planning in the aftermath of Three Mile Island, for example, that the Commission converted the recommendations of a Nuclear Regulatory Commission/Environmental Protection Agency task force report into Commission policy and extended the emergency planning zone for accidents out to 16 kilometers from reactors.

In Sweden, it appears that the political pressure of that country's debate over nuclear power may have already forced a decision in the case of filtered venting. Prior to that country's March 1980 referendum on the future of nuclear power the pro-nuclear side was eager to support every safety measure proposed by a special Swedish government committee of enquiry, created after the Three Mile Island accident. Filtered venting was one measure recommended by this committee. After the referendum, the Swedish government, noting that subsequent studies had failed to uncover any basis for a reconsideration of this decision, indicated in a parliamentary bill that it would move forward to implement filtered venting starting with the Barsebäck reactor located just 20 kilometers across the sound from Copenhagen [17].

Without the pressure of a political referendum, it is doubtful that progress on filtered venting would have been any faster in Sweden than it has been in the United States.

Unfortunately, there are no comparable political events on the horizon in the United States. It is possible, therefore, that it will take an accident more serious than Three Mile Island to overcome the inertia that is holding back further development of containment improvements in this country. If a large release of radioactivity occurs in such an accident, the U.S. nuclear industry may well follow the example of its Swedish counterpart and endorse containment improvements in an attempt to salvage a future for nuclear power in the United States.

The prognosis for our society will be bleak, however, if we protect ourselves only after experiencing every variety of disaster. It is, therefore, to be hoped that the Commission and its watchdogs will press ahead with work on accident consequence mitigation strategies from the "study" stage to the decision stage.

The Commission received exactly this recommendation from its Three Mile Island "Lessons Learned Task Force" in October 1979:

"The Task Force recommends . . . that a notice of intent to conduct rulemaking be issued to solicit comments on the issues and specific facts relating to the consideration of controlled, filtered venting for core-melt accidents in nuclear power plant design and that a decision on whether and how to proceed with this specific requirement be made within one year of the notice" [18].

The Commission, however, did not commit the necessary resources. Now, almost three years later, it is further away from such a decision than it was then.

The Commission could also be pressured into adopting the recommendation made to it in a September 10, 1980 letter from its Advisory Committee on Reactor Safeguards: that it proceed without further delay to require utilities to do design and risk reduction studies with regard to the installation of filtered vent systems on their nuclear power plants [19].

Of course the filtered vent strategy should not be pursued to the exclusion of other containment improvement strategies which may also prove useful. We have focused on the vented containment concept here because it is specific evidence for our more general contention that there is a great potential for enhancing the capabilities of reactor containment buildings to retain the radioactivity from accidents which might otherwise contaminate an area"the size of Connecticut." [See box.]

An area the size of Connecticut

Among nuclear power opponents one of the most widely used characterizations of the hazard from reactor accidents is based on a quote from the files of the long-suppressed 1965 Atomic Energy Commission study on reactor accident consequences: "The possible size of such a disaster might be equal to that of the state of Pennsylvania"[2].

What exactly would happen over this area?

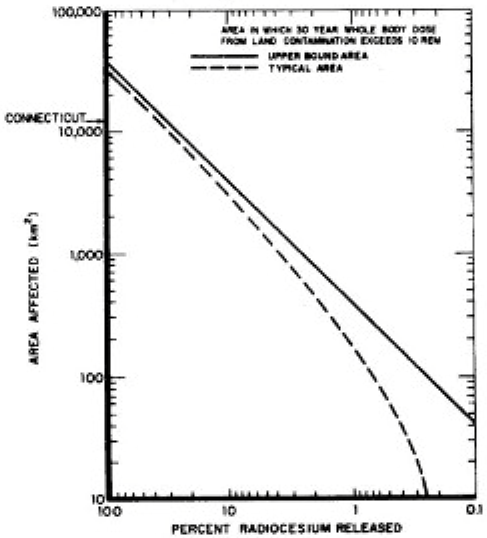
The study found — as have many studies since [3, 11, 20] — that the most widespread danger from a reactor accident would be thyroid damage from the ingestion of radioactive iodine. Milk might be contaminated with radioiodine above the protective action limits specified by the Federal Radiation Council over "areas which would range from 10,000 to 100,000 square kilometers" [2]. The area of Pennsylvania is 115,000 square kilometers; hence the comparison.

The problem of milk contamination by radioiodines appears to us to be a relatively manageable one [21], so we focus instead on two potential consequences of reactor core meltdown accidents which are less manageable than milk contamination and could also affect huge areas. These are the hazards of long-term contamination of land and property by radioactive cesium; and thyroid damage resulting from the inhalation of radioactive iodine-131.

For land contamination we have set the threshold at a standard level corresponding, in the absence of decontamination, to a cumulative whole-body dose from penetrating external gamma radiation of 10 rem to any resident population over the first 30 years following the accident. (The duration of land contamination will be dominated by 30-year half-life cesium-137.) This 10-rem dose would be approximately three times higher than the average whole-body dose from natural background radiation over the same period and might cause on the order of one extra cancer death among every 1,000 people exposed at that level [22].

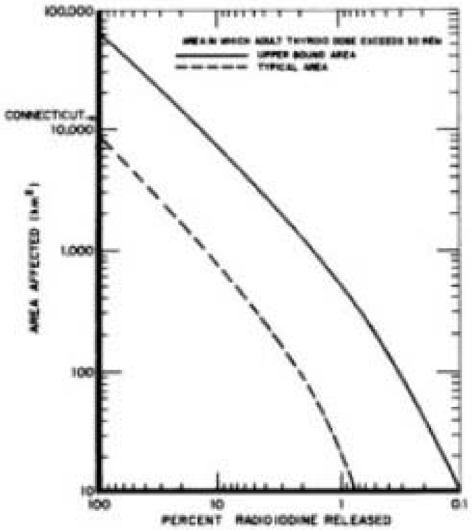
In the case of thyroid irradiation we have chosen a

Figure 5



threshold dose from inhalation of 30 rem for adults. The Environmental Protection Agency's guideline threshold dose to the thyroid for mandatory evacuation is 25 rem [23]. The dose to the thyroids of exposed children in the same area might exceed 150 rem [24]. For an X-ray dose of

Figure 6



150 rem to a child's thyroid, the probability of subsequent thyroid surgery has been found to be on the order of a few percent [25].

There has been less follow-up on the consequences per rem to the thyroid of internal beta-radiation emitted by iodine-131. The U.S. Food and Drug Administration, therefore, assumes that iodine-131 irradiation is as damaging to the thyroid per rem as X-ray radiation [26].

Figures 5 and 6 show, as a function of the percentage released into the atmosphere of the inventories of radioactive cesium and iodine from the core of a modern commercial power reactor, "typical" and realistic upper bound areas over which the long-term doses from ground contamination and the thyroid inhalation doses would exceed the thresholds specified above [27]. The upper bound curves in the figures are about the highest which can be obtained for reasonable choices of parameters using the standard simplified model for atmospheric dispersion. We show no lower limit for the area which could be affected because it could be essentially zero. A heavy rain could, for example, scrub the radioactive aerosols from the air soon after they were released from the containment.

For an uncontained meltdown, most studies predict that from 10 to 90 percent of the radioactive iodines and cesiums in the core could be released [3, 16]. It is apparent from figures 5 and 6 that the area affected by such releases with doses above the specified thresholds could be on the order of 10,000 square kilometers. Even if this is closer to the area of the state of Connecticut than Pennsylvania, it is still a very substantial area. It is also apparent that the areas at risk could, for example, be decreased by about one hundredfold if reactor containment systems could be made effective enough to reduce any releases to less than one percent of the core inventories [28]. □

1. U.S. Atomic Energy Commission, Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants, WASH-740 (1975).
2. U.S. Atomic Energy Commission, Documents Relating to the Re-examination of WASH-740. Approximately 200 unpublished documents, dating from 1964 to 1966, were made available to the public in the Commission's public document room in 1973 as a result of suits and threats of suits under the Freedom of Information Act. See also David Burnham, "A.E.C. Files Show Effort to Conceal Safety Perils," N.Y. Times (Nov. 11, 1974).
3. US Nuclear Regulatory Commission, Reactor Safety Study (Washington, D.C., WASH-1400 Or NUREG-75/014, 1975). Initiated by the Atomic Energy Commission, this study was published in its final form by the Nuclear Regulatory Commission.
4. U.S. Congress, Joint Committee on Atomic Energy Hearings, "Brown's Ferry Nu- clear Plant Fire" (Sept. 16, 1975); Daniel F. Ford, Henry W. Kendall, Lawrence S. Tye, Brown's Ferry: The Regulatory Failure (Cam- bridge, Mass.: Union of Concerned Scientists, 1976).
5. Report of the President's Commission on the Accident at Three Mile Island (1979).
6. We have assumed containment atmosphere temperatures of about 150 degrees Centigrade in these calculations.
The free volume in a containment typical of those used in most operating U.S. boiling water reactors is about 7,900 cubic meters. The effective free volume of boiling water reactor containments may be less than half of their nominal volumes, however, since the volume over the pressure suppression pool is connected to that of the "dry well" by what amounts to a one-way valve. Therefore, it would be possible, in principle, for steam to drive the "noncondensable" gases into the 40 percent of the total free volume over the pressure suppression pool, leaving the pressure in that chamber at a much higher level than in the dry well surrounding the reactor vessel after the steam condensed. (See Figure 2 for a representation of these chambers in a boiling water containment. The range of pressures cited in the text allows for this possibility.)
7. In the Three Mile Island accident an estimated 44 to 63 percent of the 22,600 kilograms of zirconium in the core were oxidized. See Report of the

President's Commission on the Accident at Three Mile Island, (staff reports), 11, p. 14.
8. For a "high-carbonate concrete" having 80 weight percent CaCOs and an initial radius of the core debris on the reactor cavity floor of 3.05 meters the "WECHSL" code predicts that the core will have penetrated 80 centimeters into the concrete 10 hours after it has landed on the surface and will have thereby released 27 metric tonnes of C02, 13 of CO, 9 of HzO, and 140 kilograms of HZ.The carbon monoxide and hydrogen result from reactions between COZ and H 2 0 and hot metals (steel and zirconium) in the melt. The oxides of carbon would add about two-thirds of an atmosphere to the pressurization of a small containment. A "medium-carbonate" concrete is characterized as having 46 weight percent $CaCO_2$ and therefore presumably would release about half as much CO_2 plus CO. Another code, "INTER" predicts about twice as much gas evolved as WECHSL. See also W.B. Murfin, I., p. 5. 18.
9. W.B. Murfin, Report of the Zion/Indian Point Study (U.S.NRC NUREG-CR-1409-1413, 1980),Summary, p. 49.

10. Report of the Task Force on Power Emergency Cooling, "Emergency Core Cooling," U.S. AEC, ~ 1 ~ 2 4 2(219666), p. 9.
11."Report to the APs by the Study Group on Light Water Reactor Safety," Review of Modern Physics 47, Sup. No. 1 (1975), p. S7.
12. B. Gosset, H.M. Simpson, L. Cave, C.K. Chan, D. Okrent and I. Catton, Post-Accident Filtration as a Means of Improving Containment Effectiveness (University of California at Los Angeles, UCLA-ENG 7775, 1977). The principal radioisotopes which would not be removed by such a filtered vent system would be the noble gases: radioactive krypton and xenon.
13. D. Carlson and J. Hickman, A Value Impact Assessment of Alternate Containment Concepts (Washington, D.C.: Nuclear Regulatory Commission, NUREGKR-0165, 1978). The Murfin Report estimates a $20 million price tag. More elaborate versions would cost more.
14. M. Levenson and F. Rahn (Electric Power Research Institute), "Realistic Estimates of the Consequences of Nuclear Accidents," paper presented at the International Meeting of the American Nuclear Society, Washington, D.C. (Nov. 20, 1980).
15. John O'Neill, "Scientists Say NRC Greatly Overestimates Accident Risks," Nuclear Industry (Dec. 1980), p. 27.
16. U.S. Nuclear Regulatory Commission, Technical Bases for Estimating Fission Product Behavior During LWR Accidents, NUREG-07 draft (March 6, 1981; final, June 1981). The basic points in the NRC experts' review were immediately apparent to knowledgeable readers of the Institute report. See Frank von Hippel, an invited briefing to the NRC as recorded in the transcript, "NRC Meeting on Iodine Release from Accidents and Estimates of Consequences," (Nov. 18, 1980), pp. 38-61. For accidents in which the damage is sufficient to open large pathways from the core to the containment, there will not be sufficient water available to trap the radioactive materials of concern, nor will the pathway be so tortuous that a significant amount will stick to surfaces before reaching the containment atmosphere. Similarly, if the containment fails early enough, there will be insufficient time for aerosols to settle to the reactor building floor. These three mechanisms are the basis for the claims made in the Electric Power Research Institute report.
17. Government bill to Swedish Parliament, 1980/81:90. It is expected that theBarseback reactor would be equipped with a filtered vent system by 1985.
18. Nuclear Regulatory Commission, TMI-2 Lessons Learned Task Force Final Report (Washington, D.C.; NUREG-0585, 1979), PP. 3-5.
19. Advisory Committee for Reactor Safeguards letter to the NRC on "Additional ACRS comments on Hydrogen Control and Improvement of Containment Capability" (Sept. 8, 1980). The point was reiterated in a Feb. 10, 1981 ACRS letter on "ACRS Report on Requirements for Near Term Construction Permits and Manufacturing Licenses."
20. Jan Beyea, "Some Long-Term Consequences of Hypothetical Major Releases of Radioactivity to the Atmosphere from Three Mile Island," a report to the President's Council on Environmental Quality (Princeton University, Center for Energy and Environmental Studies Report #109, 1980).
21. The longest lived radioiodine of concern for reactor accidents is eight-day half-life iodine-131, of which only one-thousandth the original will remain after eight weeks. The area of land contamination will, therefore, have decreased after eight weeks by orders of magnitude from its original size. During

the period of contamination it would be quite straightforward to arrange where necessary that dairy cattle be shifted from pasture to relatively uncontaminated stored feed, and to divert any contaminated milk to the production of powdered milk, cheese, etc., which could be stored until its radioactive contamination had decayed to negligible levels.

22. U.N. Scientific Committee on the Effects of Atomic Radiation, Sources and Effects of Ionizing Radiation (New York: United Nations, 1977), p. 414; U.S. National Academy of Sci- ences, Committee on the Biological Effects of Ionizing Radiation, The Effects on Population of Exposure to Low Levels of Ionizing Radiation (Washington,D.C., 1980); Eliot Marshall, "New A-Bomb Data Shown to Radiation Experts," Science 212 (1981), p. 1,364 and the related letters in Science 213 (1981), pp. 6, 8, 392, 602, 604.

23. U.S. Environmental Protection Agency, Manual of Protection Action Guides and Protective Actions for Nuclear Incidents Washington, D.C.: EPA-520/1-75-001197, 5), Table 5.2.

24. U.S. Environmental Protection Agency, Environmental Analysis of the Uranium Fuel Cycle 11: Nuclear Power Reactors (Washington, D.C.: EPA-520/9-73-003-C, 1973), Table 40.

25. L.H. Hempelmann and others, Journal of the National Cancer Institute, 55 (1975), p. 519.

26. U.S. FDA, Proposed Recommendations on Use of Potassium Iodide as a Thyroid Blocking Agent in a Radiation Emergency (April 1981). For an early release, the thyroid dose from the 21-hour half-life isotope iodine-133 would be approximately one-third that of iodine-131. In March 1954, 22 Marshallese children on Rongelap atoll received an estimated 700 to 1,200 rem thyroid dose from drinking water contaminated with such short-lived radioiodines from the "Bravo" H-bomb test. Almost all subsequently required thyroid surgery and were put on lifetime thyroid hormone medication. (Robert A. Conard and others, Review of the Medical Findings in a Marshallese Population Twenty-Six Years After Accidental Exposure to Radioactive Fallout [Brookhaven National Laboratory, BNL 51261, 1980].)

27. A detailed discussion of the derivation of Figures 5 and 6 may be found in, Jan Beyea and Frank von Hippel, Nuclear Reactor Accidents: The Value of Improved Containment (Princeton University, Center for Energy and Environmental Studies Report #94, 1980).

28. Although it is not possible to filter out the noble gases, doses in excess of 10 rem would be received from the noble gases over an area which would be smaller than 1 percent of 10,000 square kilometers.

Pilgrim Watch
June 1, 2011
EXHIBIT 7

May 16, 2011 NYT

Japanese Officials Ignored or Concealed Dangers

By NORIMITSU ONISHI and MARTIN FACKLER

OMAEZAKI, Japan — The nuclear power plant, lawyers argued, could not withstand the kind of major earthquake that new seismic research now suggested was likely.

If such a quake struck, electrical power could fail, along with backup generators, crippling the cooling system, the lawyers predicted. The reactors would then suffer a meltdown and start spewing radiation into the air and sea. Tens of thousands in the area would be forced to flee.

Although the predictions sound eerily like the sequence of events at the Fukushima Daiichi plant following the March 11 earthquake and tsunami, the lawsuit was filed nearly a decade ago to shut down another plant, long considered the most dangerous in Japan — the Hamaoka station.

It was one of several quixotic legal battles waged — and lost — in a long attempt to improve nuclear safety and force Japan's power companies, nuclear regulators, and courts to confront the dangers posed by earthquakes and tsunamis on some of the world's most seismically active ground.

The lawsuits reveal a disturbing pattern in which operators underestimated or hid seismic dangers to avoid costly upgrades and keep operating. And the fact that virtually all these suits were unsuccessful reinforces the widespread belief in Japan that a culture of collusion supporting nuclear power, including the government, nuclear regulators and plant operators, extends to the courts as well.

Yuichi Kaido, who represented the plaintiffs in the Hamaoka suit, which they lost in a district court in 2007, said that victory could have led to stricter earthquake, tsunami and backup generator standards at plants nationwide.

"This accident could have been prevented," Mr. Kaido, also the secretary general of the Japan Federation of Bar Associations, said of Fukushima Daiichi. The operator of the plant, Chubu Electric Power Company, temporarily shut down Hamaoka's two active reactors over the weekend, following an extraordinary request by Prime Minister Naoto Kan.

After strengthening the plant's defenses against earthquakes and tsunamis, a process that could take a couple of years, the utility is expected to restart the plant.

Japan's plants are all located in coastal areas, making them vulnerable to both quakes and tsunamis. The tsunami is believed to have caused the worst damage at the Fukushima plant, though evidence has begun emerging that the quake may have damaged critical equipment before the waves struck.

The disaster at Fukushima Daiichi, the worst nuclear accident since Chernobyl, directly led to the suspension of Hamaoka here in Omaezaki, a city about 120 miles southwest of Tokyo. But Mr. Kan's decision was also clearly influenced by a campaign, over decades, by small groups of protesters, lawyers and scientists, who sued the government or operators here and elsewhere.

They were largely ignored by the public. Harassment by neighbors, warnings by employers, and the reluctance of young Japanese to join antinuclear groups have diminished their numbers.

But since the disaster at Fukushima and especially the suspension of Hamaoka, the aging protesters are now heralded as truth-tellers, while members of the nuclear establishment are being demonized.

On Friday, as Chubu Electric began shutting down a reactor at 10 a.m., Eiichi Nagano, 90, and Yoshika Shiratori, 78, were battling strong winds on the shoreline leading to the plant here. Mr. Shiratori, a leader of the lawsuit, led the way as Mr. Nagano followed with a sprightly gait despite a bent back. The two men scrambled up a dune, stopping only before a "No Trespassing" sign.

"Of course, we're pleased about the suspension," Mr. Nagano said, as the strong wind seemed to threaten to topple him. "But if we had done more, if our voices had been louder, we could have prevented the disaster at Fukushima Daiichi. Fukushima was sacrificed so that Hamaoka could be suspended."

Unheeded Warnings

In 1976, a resource-poor Japan still reeling from the shocks of the oil crisis was committed fully to nuclear power to achieve greater energy independence, a path from which it never strayed despite growing doubts in the United States and Europe.

That year, as Hamaoka's No. 1 reactor started operating and No. 2 was under construction, Katsuhiko Ishibashi, a seismologist and now professor emeritus at Kobe University, publicized research showing that the plant lay directly above an active earthquake zone where two tectonic plates met. Over the years, further research would back up Mr. Ishibashi's assessment, culminating in a prediction last year by the government's own experts that there was a nearly 90 percent chance that a magnitude 8.0 quake would hit this area within the next 30 years.

After the 1995 Kobe earthquake, residents in this area began organizing protests against Chubu Electric. They eventually sued the utility in 2003 to stop the plant's reactors, which had increased to four by then, arguing that the facility's quake-resistance standards were simply inadequate in light of the new seismic predictions.

In 2007, a district court ruled against the plaintiffs, finding no problems with the safety assessments and measures at Hamaoka. The court appeared to rely greatly on the testimony of Haruki Madarame, a University of Tokyo professor and promoter of nuclear energy, who since April 2010 has been the chairman of the Nuclear Safety Commission of Japan, one of the nation's two main nuclear regulators.

Testifying for Chubu Electric, Mr. Madarame brushed away the possibility that two backup generators would fail simultaneously. He said that worrying about such possibilities would "make it impossible to ever build anything." After the Fukushima Daiichi disaster, Mr. Madarame apologized for this earlier comment under questioning in Parliament. "As someone who promoted nuclear power, I am willing to apologize personally," he said.

In the early days of nuclear power generation in Japan, the government and nuclear plant operators assured the public of the safety of plants by promising that they would not be located on top of active fault lines, Mr. Ishibashi, the seismologist, said in an interview.

But he said that advances in seismology have led to the gradual discovery of active fault lines under or near plants, creating an inherent problem for the operators and the government and leading to an inevitable conclusion for critics of nuclear power.

"The Japanese archipelago is a place where you shouldn't build nuclear plants," Mr. Ishibashi said.

Advances in seismology also led to lawsuits elsewhere. Only two courts have issued rulings in favor of plaintiffs, but those were later overturned by higher courts. Since the late 1970s, 14 major lawsuits have been filed against the government or plant operators in Japan, which until March 11 had 54 reactors at 18 plants..

In one of the two cases, residents near the Shika nuclear plant in Ishikawa, a prefecture facing the Sea of Japan, sued to shut down a new reactor there in 1999. They argued that the reactor, built near a fault line, had been designed according to outdated quake-resistance standards.

A district court ordered the shutdown of the plant in 2006, ruling that the operator, Hokuriku Electric Power Company, had not proved that its new reactor met adequate quake-resistance standards, given new knowledge about the area's earthquake activity.

Kenichi Ido, the chief judge at the district court who is now a lawyer in private practice, said that, in general, it was difficult for plaintiffs to prove that a plant was dangerous. What is more, because of the technical complexities surrounding nuclear plants, judges effectively tended to side with a national strategy of promoting nuclear power, he said.

“I think it can’t be denied that a psychology favoring the safer path comes into play,” Mr. Ido said. “Judges are less likely to invite criticism by siding and erring with the government than by sympathizing and erring with a small group of experts.”

That appears to have happened when a higher court reversed the decision in 2009 and allowed Hokuriku Electric to keep operating the reactor. In that decision, the court ruled that the plant was safe because it met new standards for Japan’s nuclear plants issued in 2006.

Critics say that this exposed the main weakness in Japan’s nuclear power industry: weak oversight.

The 2006 guidelines had been set by a government panel composed of many experts with ties to nuclear operators. Instead of setting stringent industrywide standards, the guidelines effectively left it to operators to check whether their plants met new standards.

In 2008, the Nuclear and Industrial Safety Agency, Japan’s main nuclear regulator, said that all the country’s reactors met the new quake standards and did not order any upgrades.

Concealing Facts

Other lawsuits reveal how operators have dealt with the discovery of active fault lines by underestimating their importance or concealing them outright, even as nuclear regulators remained passive.

For 12 years, Yasue Ashihara has led a group of local residents in a long and lonely court battle to halt operations of the Shimane nuclear plant, which sits less than five miles from Matsue, a city of 200,000 people in western Japan.

Ms. Ashihara’s fight against the plant’s operator, Chugoku Electric Power, revolves around not only the discovery of a previously unknown active fault line, but an odd tug of war between her group and the company about the fault’s length, and thus the strength of the earthquakes it is capable of producing.

The utility has slowly accepted the contention of Ms. Ashihara’s group by repeatedly increasing its estimate of the size of the fault. Yet a district court last year ruled in favor of Chugoku Electric Power, accepting its argument that its estimates were based on the better scientific analysis.

“We jokingly refer to it as the ever-growing fault line,” said Ms. Ashihara, 58, who works as a caregiver for the elderly. “But what it really means is that Chugoku Electric does not know how strong an earthquake could strike here.”

Her group filed the lawsuit in 1999, a year after the operator suddenly announced that it had detected a five-mile-long fault near the plant, reversing decades of claims that the plant’s vicinity was free of active faults.

Chugoku Electric said the fault was too small to produce an earthquake strong enough to threaten the plant, but Ms. Ashihara’s suit cited new research showing the fault line could in fact be much longer, and produce a much stronger earthquake. It got a boost in 2006, when a seismologist announced that a test trench that he had dug showed the fault line to be at least 12 miles long, capable of causing an earthquake of magnitude 7.1.

After initially resisting, the company reversed its position three years ago to accept the finding. But a spokesman for the Chugoku Electric said the plant was strong enough to withstand an earthquake of this size without retrofitting.

“This plant sits on solid bedrock,” said Hiroyuki Fukada, assistant director of the visitor center for the Shimane plant, adding that it had a 20-foot, ferro-concrete foundation. “It is safe enough for at least a 7.1 earthquake.”

However, researchers now say the fault line may extend undersea at least 18 miles, long enough to produce a magnitude 7.4 earthquake. This prompted Ms. Ashihara’s group to appeal last year's ruling.

Ms. Ashihara said she has waged her long fight because she believes the company is understating the danger to her city. But she says she has at times felt ostracized from this tightly bound community, with relatives frowning upon her drawing attention to herself.

Still, she said she hoped the shutdown of Hamaoka would help boost her case. She said local residents had already been growing skeptical of the Shimane plant's safety after revelations last year that the operator falsified inspection records, forcing it to shut down one of the plant's three reactors.

In Ms. Ashihara's case, the nuclear operator acknowledged the existence of the active fault line in court. In the case of Kashiwazaki-Kariwa nuclear plant in Niigata, a prefecture facing the Sea of Japan, Tokyo Electric Power Company, or Tepco, the utility that also operates Fukushima Daiichi, did not disclose the existence of an active fault line until an earthquake forced it to.

In 1979, residents sued the government to try overturn its decision granting Tepco a license to build a plant there. They argued that nuclear regulators had not performed proper inspections of the area's geology — an accusation that the government would acknowledge years later — and that an active fault line nearby made the plant dangerous. In 2005, the Tokyo High Court ruled against the plaintiffs, concluding that no such fault line existed.

But in 2007, after a 6.8-magnitude earthquake damaged the plant, causing a fire and radiation leaks, Tepco admitted that, in 2003, it had determined the existence of a 12-mile-long active fault line in the sea nearby.

Weighing the Chances

The decision to suspend Hamaoka has immediately raised doubts about whether other plants should be allowed to continue operating. The government based its request on the prediction that there is a nearly 90 percent chance that a magnitude 8.0 earthquake will hit this area within the next 30 years. But critics have said that such predictions may even underestimate the case, pointing to the case of Fukushima Daiichi, where the risk of a similar quake occurring had been considered nearly zero.

"This is ridiculous," said Hiroaki Koide, an assistant professor at the Research Reactor Institute at Kyoto University. "If anything, Fukushima shows us how unforeseen disasters keep happening. There are still too many things about earthquakes that we don't understand."

Until March 11, Mr. Koide had been relegated to the fringes as someone whose ideas were considered just too out of step with the mainstream. Today, he has become an accepted voice of conscience in a nation re-examining its nuclear program.

For the ordinary Japanese who waged lonely battles against the nuclear establishment for decades — mostly graying men like Mr. Nagano and Mr. Shiratori — the Hamaoka plant's suspension has also given them their moment in the sun.

The two worried, however, that the government will allow Hamaoka to reopen once Chubu Electric has strengthened defenses against tsunamis. Chubu Electric announced that it would erect a 49-foot high seawall in front of the plant, which is protected only by a sand dune.

"Building a flimsy seawall isn't enough," Mr. Shiratori said. "We have to keep going after Chubu Electric in court and shut down the plant permanently."

"That's right," Mr. Nagano said, the smallness of his bent frame emphasized by the enormous plant behind him. "This is only the beginning."

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EXHIBIT 8

NYT
GREEN-A BLOG ABOUT ENERGY & ENVIRONMENT
MAY 19, 2011, 3:57 PM

The Importance of Venting, When a Reactor Threatens to Blow Its Stack

By MATTHEW L. WALD

Center for Strategic and International StudiesHow the failure to vent containments at Fukushima allowed explosions to occur.

As I wrote in Thursday's paper, the Fukushima Daiichi accident is renewing a debate over whether the emergency venting systems that were added to boiling water reactors 20 years ago should be opened only directly by operators or function automatically.

At Fukushima and at plants in the United States, they variously require button-pushing in the control room, electricity or compressed gas to operate the valves, and/or muscle power on a crank. After the March 11 earthquake in Japan, operators there couldn't make the valves work through any of these methods.

This may have made the Fukushima accident more serious, because in several of the reactors, pressures rose well above design limits and the primary containments, the steel-and-concrete enclosures that are a crucial barrier against the release of radioactive materials, sprung leaks. It is not clear now precisely where those leaks are, but the containments are penetrated by pipes in numerous places, and a seal around one or more of those pipes may have failed.

The leaks have made stabilization more difficult because the Tokyo Electric Power Company has been pouring in water by the ton to submerge the reactor cores and prevent melting. Instead, the water is flowing out through the holes into the secondary containments, and from there, in some cases into the soil and the sea, picking up radioactive contamination on the way.

When the reactors were designed in the 1960s, the idea was that in the event of an accident, all of the radioactive materials would be bottled up in the primary containment. This was itself a philosophical reversal, in the sense that it was an acknowledgment that it might be impossible to hold everything in.

The notion was that early in the accident, the steam would present a risk of overpressure but would not hold very much radioactive material; letting out a burp, the thinking went, would keep other structures intact so that more contaminated materials would be held in. The part that was vented, the torus, a tank holding about one million gallons of water, would "scrub" the dirty stuff out of the steam and leave the fragments of fuel and other radioactive materials behind in the water. Thus the radioactive releases would in theory be quite small.

Yet since 1989, when the Nuclear Regulatory Commission told American plant operators that it liked the venting idea, the thinking has changed further on the operation of boiling water reactors. For one thing, most American reactors have been allowed to bolster their steam output so they can make more electricity. To get permission to do this, a reactor owner must arrive at a calculation that the emergency core cooling system could still work in case of excess heat.

Some plants now anticipate high pressure, and, in fact, require it for safe operation.

In an accident in which a reactor vessel dumped water on the floor, emergency pumps would suck that water up and put it back into the reactor. But if those pumps sucked too hard, they would pull in steam bubbles or air, which could disable the pump.

Reactor operators have argued, though, that steam bubbles will not develop, even at relatively high temperatures. The reason is that in an accident, the containment will be at a higher pressure, and water will not boil at the standard 212 degrees at high pressure; temperatures have to be much higher.

The pumps can continue suctioning water in those conditions. But this requires keeping control of venting, because if the pressure falls too far, the pumps will stop working. This phenomenon is known as net positive suction head. or N.P.S.H.

"Vents and the N.P.S.H. problem go hand in hand," said Arnie Gundersen, a nuclear consultant and critic of the Vermont Yankee plant, which was one of the first to use the argument to win permission to boost its output.

"If a vent sticks open or if the containment breaches, the emergency core cooling system pumps will fail to cool the core if they rely on the overpressure credit," he said.

If the vent is operated with an electrically driven valve, as in the current design, operators can control how much steam they let out and how much pressure they keep in. The alternative is probably a rupture disk, a thin piece of steel that breaks at a pre-designed level, just below the pressure that is likely to rupture the containment.

In a system that relies on a rupture disk, a valve left in the open position would also be installed. If the disk ruptured, the only way to re-close the system would be to close the valve. But just as the Fukushima operators discovered that in an accident there may be no way to open a valve, some experts fear that in another accident, with a rupture disk having performed its function, there would not be a way to close the valve.

Anthony G. Sarrack, an engineer who warned the Nuclear Regulatory Commission five years ago that the vents would work better with a passive design, suggested in an e-mail that the debate over passive venting is essentially over.

“Although multiple people claim that they aren’t clear about whether it is better to have passive venting or valves that require action to allow containment venting,” he said, the next generation of boiling reactor waters include a passive design (nearly identical to the one I recommended).”

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EXHIBIT 9

http://www.joewein.net/blog/2011/05/18/fukushima-had-new-vents-but-system-failed/

Fukushima had new vents, but system failed

May 18th, 2011 ·

The New York Times reports that the reactors at the Fukushima 1 nuclear power plant were equipped with an improved system for venting steam in an emergency, but it failed to work. Originally it had been reported that TEPCO did not retrofit the units that had been online since 1970s with the new designs introduced in the US in the 1980s. However, it appears to have done so between 1998 and 2001.

The problem was, the improved system relied on the same sources of electricity to operate valves as the cooling system, so when the cooling system stopped working as diesel generators failed and dangerous levels of steam pressures built up, the venting system designed to protect the containment wasn't working properly either. Instead of one system protecting the public if the other system failed, both had a single point of failure, their dependence on diesel generators in the flooded turbine hall basements.

The executives did not give the order to begin venting until Saturday — more than 17 hours after the tsunami struck and six hours after the government order to vent.

As workers scrambled to comply with their new directive, they faced a cascading series of complications.

The venting system is designed to be operated from the control room, but operators' attempts to turn it on failed, most likely because the power to open a critical valve was out. The valves are designed so they can also be opened manually, but by that time, workers found radiation levels near the venting system at Reactor No. 1 were already too high to approach, according to Tokyo Electric's records from the accident's early days.

At Reactor No. 2, workers tried to manually open the safety valves, but pressure did not fall inside the reactor, making it unclear whether venting was successful, the records show. At Reactor No. 3, workers tried seven times to manually open the valve, but it kept closing, the records say.

The results of the failed venting were disastrous.

Reactor No. 1 exploded first, on Saturday, the day after the earthquake. Reactor No. 3 came next, on Monday. And No. 2 exploded early Tuesday morning.

The venting system could also have been damaged by the earthquake.

According to the NYT, the new venting system bypasses filters that hold back much of the radioactivity.

When TEPCO was talking about venting the reactors, before the spectacular hydrogen explosions, they reassured the public that the release of gas would be “filtered”. Either they were misleading the public, or they were talking about the old venting system, which was suspected of not being able to cope with the pressure of an emergency release, which is the very reason the new system had been introduced.

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NYT May 18, 2011

http://www.nytimes.com/2011/05/19/science/earth/19nuke.html?_r=1&ref=matthewlwald

U.S. Was Warned on Vents Before Failure at Japan's Plant

By **MATTHEW L. WALD**

WASHINGTON — Five years before the crucial emergency vents at the Fukushima Daiichi nuclear plant were disabled by an accident they were supposed to help handle, engineers at a reactor in Minnesota warned American regulators about that very problem.

Anthony Sarrack, one of the two engineers, notified staff members at the Nuclear Regulatory Commission that the design of venting systems was seriously flawed at his reactor and others in the United States similar to the ones in Japan. He later left the industry in frustration because managers and regulators did not agree.

Mr. Sarrack said that the vents, which are supposed to relieve pressure at crippled plants and keep containment structures intact, should not be dependent on electric power and workers' ability to operate critical valves because power might be cut in an emergency and workers might be incapacitated. Part of the reason the venting system in Japan failed — allowing disastrous hydrogen explosions — is that power to the plant was knocked out by a tsunami that followed a major earthquake.

Copies of Mr. Sarrack's correspondence with the N.R.C. were supplied by David Lochbaum, a boiling-water-reactor expert who works for the Union of Concerned Scientists, a nonprofit group based in Cambridge, Mass., that is generally hostile to nuclear power.

"The Nuclear Regulatory Commission cannot claim ignorance about this one," he said.

Plant managers and nuclear regulators are warned about far more problems each year than actually occur, but in this case, the cautionary note was eerily prescient and could rekindle debate over whether automatic venting systems are safer alternatives.

While staff members at the Nuclear Regulatory Commission considered Mr. Sarrack's warning, they decided against changes.

On Wednesday, a commission spokesman, Scott Burnell, said the commission still believed that existing venting systems were a "reasonable and appropriate means" of dealing with a rise in pressure after an accident. But he has also said that the commission's staff members are studying the events at Fukushima Daiichi for "lessons learned," and that they had identified means of "reducing risk even further" by making the vents "more passive." He said the staff had not yet chosen a way to do that.

One way would be using rupture disks, relatively thin sheets of steel that break and allow venting without any operator command or moving parts when the pressure reaches a specified level. But many in the industry argue that using such a disk requires that there be a way to close the vent once pressure is relieved in order to hold in radioactive materials.

The accident at the Fukushima Daiichi plant was the first time the venting systems were put to the test.

Pressure began to build in three reactors soon after the tsunami hit because the plant's cooling system stopped operating when the electricity went off. Without an adequate flow of cool water in the reactors, the fuel rods began to overheat and produce explosive hydrogen gas.

Managers were worried about venting because it would release significant amounts of radioactive materials, but when they finally gave the order to do so — after being told to by the government — the workers found the venting system inoperable. With the power out, their commands from the control room did not open the valves. They then discovered that that radiation levels at one reactor were so high they could not attempt to manually open the valve. And at two other reactors, their attempts to open the valves failed, possibly because the equipment itself was damaged in the earthquake.

In Units No. 1 and No. 3, the gas leaked from primary containment structures and fueled explosions that ripped apart the reactor buildings, spewing radioactive material into the air. Unit No. 2 suffered a hydrogen explosion inside the primary containment.

Mr. Sarrack, reached by telephone, said that his proposal was opposed by the operations department officials at his company, who wanted direct control over the reactor rather than employing automatic systems. He was working at the time at the nuclear plant in Monticello, on the Mississippi River near Minneapolis.

He said he continued to believe that a passive system, like one using a rupture disk, would work better and could be set to rupture at a pressure just slightly less than the pressure at which the containment would rupture. In those cases, he said, venting is always preferable; the releases of radioactive materials during deliberate venting are expected to be lower than those resulting from explosions.

But the consensus in the nuclear industry supports the existing systems. Douglas E. True, the president of ERIN Engineering and Research of Walnut Creek, Calif., said: "In some cases you can argue it might be better to have a rupture disk. In other cases, it would certainly be better to have a manually controlled system." For example, he said, the disk is backed up by a valve that is normally in the open position. If the disk ruptured and there was no electricity, it might be impossible to close the valve, and the venting would be permanent.

The Fukushima plant was designed by General Electric, and the venting systems that failed in Japan exists at similar plants designed by G.E. in the United States.

In a statement, James Klapproth, the nuclear energy chief consulting engineer at GE Hitachi, said that his company believed that the venting system would have operated in an accident within the "design basis" of the plant," but that the Fukushima disaster was worse than what the plant was designed for. He said that the industry in this country had considered passive systems "at one time."

Exhibits 11& 12

(Attached Separately)