JAN 28 1986

MEMORANDUM FOR: Themis P. Speis, Director
Division of Safety Review and Oversight

William T. Russell, Director
Division of Human Factors Technology

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: SCHEDULE FOR RESOLVING GENERIC ISSUE NO. 122 - LOSS OF ALL FEEDWATER

The findings of the Davis-Besse Incident Investigation Team as reported in NUREG-1154 "Loss of Main and Auxiliary Feedwater Event at Davis-Besse Plant on June 9, 1985" are being reviewed by the NRC staff to identify potential generic issues and make recommendations regarding the need for staff action. Potential generic issues that are of a long-term nature will be identified and prioritized later. However, five short-term staff actions were identified in the memo from H. Thompson to T. Speis dated August 19, 1985. These five short-term actions were prioritized as separate subtasks: (1) Generic Issue 122.1a, "Common Mode Failure of Isolation Valves in Closed Position," (2) Generic Issue 122.1b, " Recovery of Auxiliary Feedwater," (3) Generic Issue 122.1c, "Interruption of Auxiliary Feedwater Flow," (4) Generic Issue 122.2, "Initiating Feed and Bleed," and (5) Generic Issue 122.3, "Physical Security System Constraints."

The technical resolutions for Generic Issues 122.1a, 122.1c, and 122.2 are assigned a "HIGH" priority ranking and Generic Issue 122.1b is assigned a "MEDIUM " priority ranking based on the evaluations provided in Enclosure 1. DSRO is assigned the resolution of subtasks 122.1a, 122.1b and 122.1c. DHFT is assigned resolution of subtask 122.2. The resolution of Generic Issue 122.3 is assigned a "LOW" priority ranking based on the evaluation provided in Enclosure 1. Therefore, the resolution of this issue will not be pursued.

In addition, Generic Issue 122.1.c leads to more general questions and its significance may not be limited to the rather narrowly-defined present scope. Therefore, an additional generic issue on this subject will be defined and prioritized later.

In accordance with NRR Office Letter No. 40, "Management of Proposed Generic Issues," the resolution of issue 122.3 will not be tracked by GIMCS. However, Generic Issues 122.1a, 122.1b, 122.1c, and 122.2 will be monitored by the Generic Issue Management Control System (GIMCS). The information needed for this system is indicated on the enclosed GIMCS information sheet. Your schedule for resolving and completing this generic issue should be commensurate with the priority nature of the work and consistent with the NRR Operating Plan. Normally, as stated in the Office Letter, the information needed should be provided for each of the three subtask elements within six weeks.

PR-7-2B
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The enclosed prioritization evaluation will be incorporated into NUREG-0933, "A Prioritization of Generic Safety Issues," and is being sent to the regions and other offices, the ACRS, and the PDR for comments on the technical accuracy and completeness of the prioritization evaluation. Any changes as a result of comments will be coordinated with you. However, the schedule for the resolution of this issue should not be delayed to wait for these comments.

The information requested should be sent to the Safety Program Evaluation Branch, DSRO. Should you have any questions pertaining to the contents of this memorandum, please contact Harold Vander Molen (x28204).

Original Signed
H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Prioritization Evaluation
2. Generic Issue Management Control System

cc: w/o Enclosure 2:
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BWS for
DSRO
Speis
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ISSUE 122: DAVIS-BESSE LOSS OF ALL FEEDWATER EVENT

The loss of all feedwater event at Davis-Besse on June 9, 1985 resulted in the formation of an NRC project team to investigate the event. The team's findings were published in NUREG-1154[2] and were subsequently reviewed by DL. As a result of DL's review, the following items were identified as candidates for short-term staff action[1] and were forwarded to DST for prioritization:[3]

1. Potential inability to remove reactor decay heat due to questionable reliability of the auxiliary feedwater system caused by any or all of the following:

    a. Loss of all auxiliary feedwater due to common-mode failure of AFW pump discharge isolation valves in closed position.

    b. Excessive delay in recovery of auxiliary feedwater due to difficulty in restarting AFW pump steam driven turbines, if turbines are tripped.

    c. Interruption of auxiliary feedwater flow due to failures in steam and feed line break accident mitigation features (e.g., SFRCS).

2. Adequacy of emergency procedures, operator training and available plant monitoring systems for determining need to initiate feed-and-bleed cooling following loss of steam generator heat sink.

3. Physical security system constraints which could deny timely operator access to vital equipment and inhibit operator from performing local manual operations called for in emergency procedures.

The above items formed the basis for Issue 122 but were prioritized separately as shown below. The identification of each item prioritized follows the numbering system established in the DL memorandum.[1] The prioritization results are summarized in Table 3.122-1.

TABLE 3.122-1

Item	Staff Action	Priority
1.a	Common Mode Failure of AFW Pump Discharge Isolation Valves in Closed Position	HIGH
1.b	Excessive Delay in Recovery of Auxiliary Feedwater	MEDIUM
1.c	Interruption of Auxiliary Feedwater Flow	HIGH
2.	Adequacy of Emergency Procedures, Operator Training and Available Plant Monitoring Systems	HIGH
3.	Physical Security System Constraints	LOW

ITEM 122.1: POTENTIAL INABILITY TO REMOVE REACTOR DECAY HEAT

During the loss of main feedwater event, the reactor scrammed and the AFW system should have actuated and supplied feedwater to the steam generators to enable them to remove decay heat. However, in this process several failures occurred, three of which are of significance here.

(1) An operator attempted to start the two AFW trains manually, in addition to the automatic signal on low steam generator water level. Unfortunately, the operator pressed the wrong buttons, sending erroneous "low steam generator pressure" signals to both AFW trains. The AFW control systems then caused both AFW isolation valves to close. Thus, neither steam generator could receive any water. In essence, the operator caused a common mode failure.

(2) Both AFW turbines tripped on overspeed. The overspeed trips on such turbines usually have to be reset at the turbine, not from the control room.

(3) In attempting to recover the AFW system, the operators reset the erroneous signals. However, the AFW isolation valves did not open. In spite of several attempts, the plant operators were unable to open these valves from the control room, and ultimately had to open them by hand.

ITEM 122.1.A: FAILURE OF ISOLATION VALVES IN CLOSED POSITION

DESCRIPTION

Historical Background

This item addresses Findings 4, 5, 6, and 15 in Section 5.2.5 of NUREG-1154.[2] The particular issue deals with a potential inability to remove reactor decay heat because of loss of all auxiliary feedwater due to the third common mode failure discussed above. This is the failure of AFW pump discharge isolation valves to reopen on command after they had closed.

Safety Significance

With the main feedwater out of service (the transient initiator), a spurious closing of these AFW valves cannot easily be rectified, leaving only feed-and-bleed techniques available for removal of decay heat.

Westinghouse PWRs generally do not have such motor-operated isolation valves in the AFW discharge lines, but some Westinghouse plants plus roughly 16 plants (designed by B&W and CE) in addition to Davis-Besse may be susceptible to this problem.

Possible Solutions

The failure of the Davis-Besse AFW valves to reopen was ultimately traced to the torque, limit and bypass switches which control the motor operators of the valves. In essence, the high differential pressure across the closed valves necessitated a relatively large force for valve motion. The motor control switches were not adjusted to accommodate such a force. Such a

failure can happen in two ways. First, the switches can be inadvertently mis-adjusted during routine maintenance. Second, the valve may be correctly maintained but the actuation system is not designed to provide for an open command to these valves (in some PWRs), or the torque necessary to reopen these valves under some conditions may be beyond the design capacity of the valve actuators. In the case of Davis-Besse, the valves were designed to close (which is their intended safety function), but apparently less attention was paid to their ability to reopen.

The solutions are implicit in the causes. For this prioritization we assume that the actuation system is equipped to issue open commands so the solution is to verify that the valves, as designed, are capable of reopening in the presence of a differential pressure, and upgrade the calibration and maintenance procedures.

PRIORITY DETERMINATION

Frequency Estimate

To estimate accident frequencies, the example of Reference 4 will be followed, in which the relatively simple transient classifications of the Oconee RSSMAP study[5] were used, but frequency and probability estimates were taken from the more modern sources such as the more detailed probabilistic analysis of Oconee 3 done by EPRI and Duke Power Co.[6]

The affected sequences in the RSSMAP study are T_1M(LOPNRE)LU, T_2MLU and T_3M(PCSNR)LU, where

T_1 is a loss of offsite power (LOOP) transient with an assumed frequency of 0.05 transients per plant year (or more) based on Reference 7.

T_2 is a non-recoverable loss of the Power Conversion System caused by other than a loss of offsite power, with an assumed frequency of 0.64 transients per plant year based on the Oconee PRA. [6]

T_3 is a transient with the Power Conversion System initially available, with an assumed frequency of 5.7 transients per plant year also based on the Oconee PRA.

M This is a failure of the power conversion system. The probability is unity for T_1 and T_2 sequences. For T_3 sequences, we will use 3.7E-3, obtained by summing the failure modes listed in Section A8.3.8 of the Oconee PRA.[6]

LOPNRE This is the probability of non-recovery of offsite power in 40 minutes after a LOOP event. We estimate this to be roughly 0.25, based on the generic curves given in Reference 7.

PCSNR This is the probability of non-recovery of the Power Conversion System (really, main feedwater) in 30 minutes. The Oconee PRA[6] uses 0.3 for a similar event (event REFDW2). It must be remembered that this figure is somewhat optimistic because of the ability to cross connect at the Oconee site.

L is failure of the AFW system, and

U is a failure to cool the core via feed and bleed. For Oconee, and most other plants, this is essentially a failure of the high pressure ECCS. The assumed probability is 0.015 based on the Oconee PRA.[6]

The unquantified parameter is ΔL, the change in the AFW failure probability to be attributed to this issue. It is composed of three factors: the probability of spurious isolation, the probability of failure to reopen on demand, and the probability of failure of reopening (in time to prevent core damage) by manual action.

Davis-Besse has been in operation for eight years. The licensee reports a frequency of loss-of-feedwater events of 0.67 per year.[8] Thus, the AFW system has had about five real challenges. One of these was the June 9 event where an operator inadvertently pushed the wrong button and caused a spurious isolation. One would therefore expect the spurious isolation rate to be roughly one in five AFW demands, or 20%, and dominated by human error. However, it would be naive to assume that this event (and its associated multi-month shutdown) has gone unnoticed in the control rooms of other plants. Nor can it be assumed that all other plants have an AFW control panel like that of Davis-Besse. On the other hand, the AFW discharge isolation valves may be initially closed at the time of the demand, as they were at the outset of the accident at Three Mile Island Unit 2. We will assume a 5% minimum likelihood of spurious or inadvertent AFW isolation, and assume further that plants with a high (e.g., 20%) likelihood will be addressed by Issue 122.1.C.

Next is the question of failure of the isolation valves to open on demand. As was mentioned before, this can happen either by errors in maintenance or by a lack of foresight in design. For the case of errors in maintenance, we turn to the valve failure data tabulated in Reference 9. Of the 393 motor-operated valve (MOV) failures listed, 75 involved torque, limit or bypass switches, and 34 of these (about 8.7% of all the failures) appeared to be adjustment or calibration errors. Since the same crews and procedures are used on all AFW trains, these failures are very likely to be present on all

trains. Given a failure on one train, we will assume an 8.7% probability (as in Reference 9) that the failure was due to improper torque or limit switch adjustment and that the analogous valves on the redundant trains will also fail. The RSSMAP study used an MOV control failure rate of 6.4E-3 per demand. The probability of failure to reopen due to maintenance error is the product of these two figures, or 5.6E-4.

For the case of lack of foresight in design, there is no extensive tabular data. This particular scenario, by its very nature, will affect both valves. However, this does not mean that both valves necessarily will fail to open. Reference 2 describes tests of the actual valves at Davis-Besse, five of which were at a full differential pressure of 1050 psid. One valve failed to open twice. The other valve failed once, but opened successfully two times. Thus, for a two-train AFW system, the probability of neither valve opening would be expected to be on the order of 1.00 x 0.33, or 33%, based on this admittedly sparse data.

Finally, the probability of the operator failing to reopen the valves manually must be estimated. In the case of the Davis-Besse event, the spurious closure occurred about six minutes into the event. Reference 2 mentions a 30 minute interval before core damage would be expected. Thus, the operators had about 24 minutes in which to reopen at least one valve. In actual fact, it took an average of 7.5 minutes (about a third of the available time) to open these two valves. This is plenty of margin, and would normally imply a failure rate (due to timeout) of a percent or two. However, it should be noted that, except for one button pushing error (which is understandable in the light of hindsight), this operating crew performed very well. The shift supervisor and his assistant were astute in diagnosing the AFWS misalignment (while being faced with a barrage of other information)

and took the correct action to manually open the auxiliary feedwater block valves. We will assign a 10% probability of failure to manually reopen the valves, based purely on judgment of the human factors aspects.

Putting these factors together, the AFW failure probability is the product of a 5% probability of inadvertent AFW isolation, a 33% probability that neither valve will reopen on demand and a 10% probability that manual opening will not be attempted or will fail to be accomplished in time.

The product is 1.7E-3 per demand. In addition, no solution is perfect. We will assume that any resolutions adopted will be at least 90% effective. Thus, the change in AFW failure probability will be on the order of 1.5E-3.

The change in core melt frequencies can now be estimated. The cut sets are:

T_1M*LOPNRE*ΔL*U	3.0E-7/py
T_2M*ΔL*U	1.5E-5/py
T_3*M*PCSNR*ΔL*U	1.5E-7/py
Total ΔF =	1.5E-5/py

Consequence Estimate

Normally, accident sequences such as the ones discussed in the previous section would be distributed across a spectrum of containment failure modes in a variety of ways. However, because the sequences of interest here are similar in their final stages prior to core melt, all three sequences will be distributed across the containment failure modes in the same manner.

All three principal accident sequences involve a core melt with no large breaks initially in the reactor coolant pressure boundary. The reactor is likely to be at high pressure (until the core melts through the lower vessel head) with a steady discharge of steam and gases through the PORV. These are conditions likely to produce significant hydrogen generation and combustion.

The Zion and Indian Point studies used a 3% probability of containment failure due to hydrogen burn (the "gamma" failure). We will follow this example and use 3%, remembering that specific containment designs may differ significantly from this figure.

In addition, the containment can fail to isolate (the "beta" failure). Here, the Oconee PRA[6] figure of 0.0053 will be used. If the containment does not fail by isolation failure or hydrogen burn, it will be assumed to fail by base mat melt-through (the "epsilon" failure).

Using the usual prioritization assumptions of a central Midwest plains meteorology, a uniform population density of 340 persons per square mile, a 50 mile radius and no ingestion pathways, the consequences are:

Failure Mode	Percent Probability	Release Category	Consequences (person-rem)
gamma	3%	PWR-2	4.8E6
beta	0.5%	PWR-5	1.0E6
epsilon	96.5%	PWR-7	2.3E3

The "weighted-average" core melt will have consequences of 1.5E5 person-rem.

Cost Estimate

The costs associated with resolving this item depend on the nature of the solution. A check of the valve operator design is relatively inexpensive. A test to ensure the valves will open will cost significantly more. Finally, if valve operators are found to be insufficiently sized, the cost of replacement will be higher still. In addition, improvements in maintenance may also be required.

For prioritization purposes, we will assume that a check of design (rather than extensive testing) will be done, and that one plant will be found where the valves would not re-open with a significant differential pressure present. We will assume further that the motor is strong enough to open the valve, and that the problem can be fixed by changing torque, limit and bypass switch setpoints. Because maintenance error is a relatively minor contributor, we will (for now) not address the issue of improved maintenance.

For each plant affected, two staff weeks should be sufficient to check the valve design. For the (hypothetical) plant where a problem is found, six staff-months should suffice to find a solution. Finally, six staff-months plus two staff-weeks per plant of NRC time will probably be necessary to impose the requirement.

Thus, for 17 plants, the total cost will be roughly $240,000, assuming that a staff year costs $100,000.

Value/Impact Assessment

On average, the B&W and CE plants have about 31 calendar-years of licensed lifetime remaining per plant. This is roughly 24 years of operational life.

Priority parameters can now be calculated, under the assumption that one plant will find and correct a problem.

Person-rem/reactor	50
Person-rem, total	50
Core melts/reactor-year	1.5E-5
Core melts/year	1.5E-5
Priority score, person-rem/million dollars	250

Other Considerations

1. There is no significant occupational exposure associated with the fix for this issue. The valves in question are not exposed to contaminated fluids, since they are in the secondary system.

2. There are offsetting savings which could be credited against the expenditures above. The cost of a core melt would be about a billion dollars, plus replacement power for the rest of the plant lifetime. In an actuarial sense, using the accident frequencies estimated above and assuming a 5% annual discount rate, this corresponds to about 430,000 present-day dollars per plant.

   Also, even if core melt is avoided, if the plant is ever placed in a situation where feed-and-bleed techniques are used, major cleanup will be necessary because of rupture of the quench tank. If cleanup lasts six months, the actuarial cost is about 770,000 present-day dollars per plant.

   Finally, it should be noted that the Davis-Besse event has kept the plant shut down for over three months. The frequency of this situation is about 1.2E-2 per reactor-year, which corresponds to an actuarial cost of roughly 4.6 million dollars per plant.

Obviously, if any of these three considerations were included, the cost-benefit ratio would be favorable indeed. It would be very much in the licensee's financial interest to fix this problem.

3. The figures assume that the feed-and-bleed failure probability is 0.015. In actual fact, Reference 2 gives the impression that the Davis-Besse operators were rather reluctant to initiate feed-and-bleed. Thus, this figure may be somewhat optimistic. Also, some (CE) plants do not have power-operated relief valves on the primary system, and thus cannot use feed-and-bleed techniques.

4. Some plants operate with the AFW isolation valves in the closed position. Thus, these plants will not need an inadvertent isolation to encounter a problem. On the other hand, these plants are more likely to be designed to open under differential pressure or to find the problem by normal testing.

5. The discussion has addressed only PWRs. Boiling water reactors (BWRs) have analogous systems (HPCI and RCIC) for mitigating loss-of-feedwater events. Moreover, these systems have normally-closed motor-operated isolation valves in the discharge line. But these valves are tested during normal system tests. In addition, BWRs can rapidly depressurize via the ADS and use low pressure systems for decay heat removal.

6. An I&E Bulletin on the subject of valve operability is being considered. This may well be sufficient to resolve the issue for most plants. However, some followup action may be appropriate, particularly for plants where the viability of feed-and-bleed is doubtful. If such a plant were also susceptible to the common-mode valve problem described here, the core-melt frequency could approach 1.0E-3.

7. This issue is related to Item II.E.6, "In Situ Testing of Valves." Although II.E.6 is also concerned with valve operability, this new item differs in that the potential for commonality is a primary concern. II.E.6 is geared toward the single failure rate per valve, not the potential for common-mode failures, but is not specific as to which valves or which failure mode.

8. This issue is also similar to GI 87, which concerns the failure of the HPCI steam line isolation valves to close following a break in the line downstream of the valves. These failures are also due to a design problem in which the valve may not have been designed to operate under some overlooked conditions. There may be other systems with valves that are not designed to operate under all likely conditions and therefore a widening of the scope of this issue may be in order.

9. It was assumed that the probability of both AFW isolation valves failing to reopen was 33%. In some cases (e.g., undersized actuators), this figure may be nearly 100%, which would triple the priority parameters. However, this would change no conclusions.

CONCLUSION

Based on the change in core-melt frequency, this issue should be placed in the HIGH priority category.

ITEM 122.1.B: RECOVERY OF AUXILIARY FEEDWATER

DESCRIPTION

Historical Background

This item addresses Findings 4, 8, and 15 in Sections 5.2.4 and 6.2.4 of NUREG-1154.[2] The particular issue deals with a potential inability to remove reactor decay heat due to the second common mode failure discussed above. This is the excessive delay in recovery of auxiliary feedwater due to difficulty in restarting AFW pump steam turbines, if the turbines are tripped.

Safety Significance

Some method of decay heat removal is necessary within 30 minutes after the start of this type of transient in order to prevent core uncovery. The turbines tripped about 7 minutes into the event. Thus, 23 minutes were available. Although it only took 4 1/2 minutes for a pair of equipment operators to go to the AFW pump rooms and start work, considerable difficulty was experienced in resetting and restarting the turbines. Thus, it might well have taken longer than 23 minutes to get the AFW pumps in operation. Had other decay heat removal techniques (i.e., startup feed pump and primary side feed and bleed) also failed, core damage would have resulted.

This issue is applicable to any PWR. However, it is of greatest importance to a plant with only steam-driven AFW trains (such as Davis-Besse), of less importance for plants with one steam-driven train plus one or two motor-driven trains. In addition, non-B&W plants are less susceptible because of their greater water inventory in the steam generators, which provides more time before active means of decay heat removal are essential.

Davis-Besse is the only remaining plant with only steam-driven auxiliary feedwater. Thus, this analysis will be geared to the next-most-susceptible plant class: a B&W plant with one steam-driven and one motor-driven AFW train.

Possible Solutions

The Davis-Besse event exhibited two problems that led to delay in AFW restart. The first problem was that the turbine overspeed trips had to be manually reset, requiring plant personnel to be dispatched to the AFW pump rooms. A possible solution is to make the trip resettable from the control room. The trip mechanism is usually a latch hook device on the trip-and-throttle valve. A mechanical device will unlatch the hook and trip the turbine at a preset speed (usually 125% of rated). Other signals can be used to trip the latch hook by means of an electrical solenoid. In either case, the hook must be reset manually. The solution, which has been done on some BWR RCIC turbines, is to wire the protective circuits into the throttle mechanism rather than the trip solenoid. The mechanical overspeed trip remains active, but is supplemented by an electrical overspeed trip (set at 110%) which can be remotely reset.

The second problem was that the two equipment operators were unsuccessful in their attempts to get the turbines running, and were saved by the arrival of an experienced operator. The most obvious solution to this problem would be to require the plant operators to practice going through the procedures of resetting and starting the turbines, assuming a remote reset is not provided. Hands-on practice of this task is not now part of operator training.

PRIORITY DETERMINATION

Frequency Estimate

Problem 1

The affected sequences and cut sets are the same as those for Item 122.1.A except parameter ΔL, the change in AFW failure probability to be attributed to this item. This is governed by three factors: the probability of a resettable turbine trip, the probability of failure to manually reset and restart the turbine, and the probability of failure (in this study) of the one motor-driven AFW train.

First, we must estimate the probability of a turbine trip, either during the auto-start or while running. PRA fault trees model individual components and their failures, but do not normally model the trips of spurious and/or readily resettable trips of concern here. Thus, PRA fault-tree-based estimates are really estimates of the failure rate assuming that the manual reset problem has been fixed. (Also, the turbine-train-only failure rate is remarkably difficult to separate out of most PRA studies.) We will use a value of 3E-2 failures per demand, based on the station blackout calculations for a two-train AFW system in Reference 11.

In Reference 10, 112 of the 170 AFW events tabulated were failures of turbine rather than motor-driven pumps. Of the 112 turbine events, 40 were trips, usually on overspeed. Thus, given a failure of a turbine-driven AFW pump to operate, there is a 35% chance that a (manual) reset might recover the pump. Therefore, the before-fix failure rate is 3E-2 times 1.35 or 4.1E-2/demand.

We must now estimate the change in turbine failure rate due to eliminating the need for manual reset. In the Davis-Besse event, the operators were

able to reset the two turbines in 4.32 and 4.77 minutes (but not get them running), which was about one fifth of the 23.4 minutes available before core uncovery.[2]

One would expect that, for a straightforward task such as resetting and restarting a turbine, the time needed would be described by a reasonably symmetrical distribution centered about an average time. Here, the 4.5 minute average time of the two unsucessful resets at Davis-Besse is probably a reasonable estimate of a general mean time for an experienced operator to successfully complete the task. This number is also consistent with oral communications we have had with operations staff at two other plants, and with a walk-through of the procedure at Davis-Besse by NRC staff. However, we have no direct information about the width of the distribution--the minimum and maximum time needed for completion. Thus, we will use a pragmatic approach. We will keep the peak of the distribution at 4.5 minutes and fix it at zero at time equals zero. Further, we will use the single-event Poisson distribution, which will extend out to infinity in the positive direction. The formula is:

$$P(t) = \lambda t \exp(-\lambda t)$$

The peak of the distribution is at $t = 1/\lambda$. So we will use $\lambda = 1/4.5$ minutes $= 0.220$. The probability of not resetting the turbine before 23.4 minutes is obtained by integrating this formula from 23.4 minutes to infinity. The integral is:

$$P(t > t_o) = (1 + \lambda\ t_o) \exp\ (-\lambda t_o)$$

$$= 0.036 = 3.6\%$$

Again, this approach is pragmatic rather than rigorous--the formula is appropriate for randomly distributed events, which this really is not. In the actual event at Davis-Besse, it is evident that the operating crew worked as fast as they could. It is also evident that the task of resetting and restarting the turbines was far from smooth. Many things went wrong. Moreover, things might well not be easy and straightforward in another, similar event. Nevertheless, a factor of five margin in the time actually taken is significant. Thus, 3.6% does not seem unreasonable in spite of the rather sparse mathematical basis.

In addition, there is a finite probability that plant operators will encounter difficulty in moving through the plant and entering the AFW pump rooms due to locked doors, etc. To account for this, we will add a 1% probability of an insurmountable difficulty in reaching the turbines, based on the calculations in Issue 122.3, to get a total probability of failure to reset of 0.046.

ΔL can now be estimated. First, the change in the turbine-driven train's failure rate is:

$$4.1\text{E-}2 \quad \frac{\text{failures}}{\text{demand}} \times 0.35 \quad \frac{\text{Turbine Trips}}{\text{Total failures}} \times 0.046 \quad \frac{\text{Failure to reset}}{\text{turbine trip}}$$

$$= 6.6\text{E-}4$$

In addition, we must estimate the unavailability of the motor driven train. Reference 11 gives a "typical" AFW system unavailability of 1E-3 per demand for a two train system. Such a figure includes common-mode failures and common component failures in addition to the individual train failures. For our purposes, we will assume that the common-mode and common-component

contributions are small, and thus that the turbine train contribution enters as a multiplicative factor. The non-turbine failure rate is then 1E-3/3E-2, or 0.033.

Giving credit for the motor-driven train, if AC power is available,

$\Delta L = (6.6E\text{-}4)(0.033) = 2.2E\text{-}5.$

If AC power is not available,

$\Delta L' = 6.6E\text{-}4.$

One more figure is needed. Since the turbine-driven AFW pump is especially significant for loss of all AC power (station blackout), a diesel unavailability is needed. Reference 7 gives a range of 1.1E-3 to 6.8E-3 for a one-out-of-two diesel configuration. We will use 2.7E-3, the middle of this range.

Cut sets can now be calculated:

$T_1M*LOPNRE*\Delta L*U$	4.1E-9/py
$T_1M*LOPNRE*DIESELS*\Delta L'$	2.0E-8/py
$T_2M*\Delta L*U$	2.1E-7/py
$T_3*M*PCSNR*\Delta L*U$	2.1E-9/py

Total ΔF = 2.4E-7 core melts/plant-year.

Problem 2

In the first problem, it was assumed that the only question was the time available for a qualified operator to locally reset a tripped AFW turbine. The fact that neither of two equipment operators were able to get the

turbines running at Davis-Besse strongly suggests that the probability of failure is nearly unity over the course of a half hour, if the individuals involved have never performed this task before. (This task is generally not part of an operator's training.) In general, during off shifts, experienced personnel are present in very limited numbers. In a future event, the more experienced personnel are likely to be busy with other tasks (e.g., getting diesels started), and a less experienced operator may once again be faced with the task of resetting and re-starting AFW turbines.

This second problem is not amenable to the exponential time calculations of Problem 1, since the average time needed for inexperienced personnel is likely to be far in excess of 30 minutes. Thus, we will arbitrarily assume that, should an event occur during the evening, night, or weekend shifts (76% of the time), there is a 50% probability that an AFW turbine trip reset will be assigned to an inexperienced operator: who is at most 10% likely to succeed in getting the turbine running in the required time. Thus, the change in the probability of failure to re-start the turbine becomes (0.76) x (0.50) x (0.90) = 0.342.

For this problem, the change in the turbine-driven train's unavailability is:

$$4.1\text{E-}2 \ \frac{\text{failure}}{\text{demand}} \times 0.35 \ \frac{\text{turbine trips}}{\text{failure}} \times 0.342 \ \frac{\text{failures to re-start}}{\text{turbine trip}}$$

$$= 4.9\text{E-}3$$

Giving credit for the motor-driven train as before:

ΔL = (4.9E-3) (0.033) = 1.6E-4 (AC power available)

$\Delta L'$ = 4.9E-3 (AC power not available)

Cut sets can now be calculated:

$T_1M^*LOPNRE^*\Delta L^*U$	3.1E-8
$T_1M^*LOPNRE^*DIESELS^*\Delta L'$	1.7E-7
$T^2M^*\Delta L^*U$	1.6E-6
$T^3{}^*M^*PCSNR^*\Delta L^*U$	1.6E-8

$$\text{Total } \Delta F = 1.8\text{E-}6 \ \frac{\text{core melts}}{\text{plant year}}$$

Consequence Estimate

The consequence estimate is the same as that for Item 122.1.A. The "weighted-average" core melt will have consequences of 1.5E5 person-rem.

Cost Estimate

Problem 1

Changing the turbine trip logic on a safety-related system is likely to require 6 staff months of effort per plant, even if no major procurement is needed. In addition, at least 3 staff months of generic work plus a week of effort on each plant will be required of the NRC staff. The total cost for the 9 PWRs with two AFW trains (excluding Davis-Besse) is thus at least a half million dollars.

Problem 2

Having operators practice the task of resetting and manually starting AFW turbines is relatively inexpensive. (If, after the first time, more than half an hour of the operator's time is needed, there is little point in the

exercise.) However, this is a continuing expense. We will assume one staff month of administrative effort per plant to set the program up, plus two staff weeks every year thereafter of actual practice. Assuming a 5% discount rate and an average remaining life of 28 calendar years (see below), this is about $620,000 total for nine plants. NRC costs are again likely to be one staff month of generic work plus a staff week per plant, or about $26,000. The total cost is roughly $650,000.

Value/Impact Assessment

The nine PWRs with two-train AFW systems have about 252 calendar years of collective license lifetime remaining. This is roughly 189 years of operational life.

Priority parameter can now be calculated.

	Problem 1	Problem 2
Person-rem/reactor	0.7	6
Person-rem, Total	7	50
Core Melts/reactor-year	2.4E-7	1.8E-6
Core Melts/year	2.1E-6	1.6E-5
Priority score, person-rem/million dollars	13	80

Other Considerations

1. There is no significant occupational exposure associated with the fix for this issue. The valves in question are not exposed to contaminated fluids, since they are in the secondary system.

2. There are offsetting savings which could be credited against the expenditures above. The cost of a core melt would be about a billion dollars, plus replacement power for the rest of the plant lifetime. In

an actuarial sense, using the accident frequencies estimated above and assuming a 5% annual discount rate, this corresponds to about 6000 present-day dollars/per plant.

Also, even if core melt is avoided, if the plant is ever placed in a situation where feed-and-bleed techniques are used, major cleanup will be necessary because of rupture of the quench tank. If cleanup lasts six months, the actuarial cost is about 10,000 present-day dollars per plant.

3. The figures assume that the feed-and-bleed failure probability is 0.015. In actual fact, Reference 2 gives the impression that the Davis-Besse operators were rather reluctant to initiate feed and bleed. Thus, this figure may be somewhat optimistic. Also, some (CE) plants do not have power-operated relief valves on the primary system, and thus cannot use feed-and-bleed techniques. Raising the feed-and-bleed failure probability to 0.1 would put this issue into the high priority range.

4. Some plants may have still other means of decay heat removal (e.g. the high head service water system at Oconee). For these plants, the figures would have to be adjusted downward.

5. These figures should not be used for BWR HPCI and RCIC systems. The BWR systems generally have a greater number of trips and an elaborate isolation system.

6. The calculations above are based on an AFW system with one motor and one turbine-driven train. A plant such as Davis-Besse, with only two turbine-driven trains, will be significantly more susceptible to this issue because whatever tripped the first turbine may well trip the

second also. Other plants which originally were equipped with only turbine-driven trains include Turkey Point 3 and 4 and Haddam Neck. The Turkey Point Units share three turbine-driven AFW trains and also have now installed a motor driven train apiece. Haddam Neck has two turbine-driven trains and has installed one (manual start) motor-driven train. The availability and surveillance requirements for the new motor driven trains on these plants have not been added to the plant's technical specifications and they are as yet not capable of being powered from onsite emergency power. Nevertheless, given the presence of these diversely powered trains, these plants are not likely to need special treatment for this issue.

CONCLUSION

This issue is of high priority for those plants which cannot remove decay heat by feed-and-bleed or other alternative means. Thus, it should be subsumed into Issue 122.2 for such plants. For the remainder, based on the figures above, this issue should be placed in the MEDIUM priority category.

ITEM 122.1.C: INTERRUPTION OF AUXILIARY FEEDWATER FLOW

DESCRIPTION

Historical Background

This item addresses Finding 6 in Section 5.2.2 of NUREG-1154.[2] The particular issue deals with a potential inability to remove reactor decay heat because of the interruption of all auxiliary feedwater flow due to the first common mode failure discussed above. This is the closing of the AFW pump discharge isolation valves. This is related to Issue 122.1.A, which deals with another problem that prevented the isolation valves from reopening.

Safety Significance

The definition of this issue in Reference 1 is ambiguous in that the full title, "Interruption of Auxiliary Feedwater Flow due to Failures in Steam and Feed Line Break Accident Mitigation Features (e.g., SFRCS)" refers to the second failure described under 122.1., but the bases presented are Section 5.2.2 and Finding 6 of NUREG-1154,[2] which refer to the first failure (i.e., of main, not auxiliary, feedwater). We will address both in this analysis.

The first sub-issue is the spurious closure of the MSIVs, in this case as a result of a turbine trip. Most plants of recent design are equipped with turbine-driven main feedwater pumps. Closure of the MSIVs will shut off all feedwater flow. Moreover, once MSIVs are closed, the reopening of these valves is a rather elaborate procedure. The loss of main feedwater is not easily recoverable.

The second subissue is the isolation of auxiliary feedwater. This is done in the event of a steam line break within containment, to prevent exceeding the containment design pressure. The containment is designed to accommodate the initial blowdown of a steam generator. If feedwater to the affected steam generator is not shut off, the boil off due to decay heat will continue to dump steam to the containment. However, in a transient involving loss of main feedwater but no steam line break, shutting off AFW flow is very undesirable. It must also be remembered that loss-of-feedwater events are far more frequent than steam line breaks.

Possible Solutions

Inadvertent MSIV closure has in the past been considered a relatively rare transient. In the particular case of the Davis-Besse transient, the steam generator level sensors had been replaced by a new type of transmitter.[2] The

rapid closure of the turbine stop valves sent a pressure wave up the steam lines back to the steam generators. This phenomenon is not new; it is routinely allowed for in the analysis of BWR transients, where the reactor core is directly sensitive to the pressure pulse. However, the new transmitters were of a design that did not dampen out the pressure pulse, which cased them to trip. A possible solution would be to add some damping to the level signal, at those plants where this has proven to be a problem.

The inadvertent isolation of AFW flow appears to be primarily a human factors problem associated with the controls layout. This could be solved by a redesign of this portion of the control panel. If on further study it appears that spurious isolations are occurring because of hardware problems, other actions (e.g., possibly using high containment pressure in a logical "and" with low steam generator pressure) might be necessary. In addition, the question of whether an operator should anticipate automatic actuations or simply observe and confirm them should be addressed in the long term.

This item appears to be associated with B&W plants. The isolation logic and AFW control is quite different for the other PWR vendors. (CE-designed plants may be susceptible to the first subissue.)

PRIORITY DETERMINATION

Frequency Estimate

The affected sequences and cut sets are the same as those for Item 122.1.A with the exception of the parameter L which is redefined as follows:

L - This is the failure rate of the auxiliary feedwater system. Reference 11 gives 1E-3 per demand as "typical" for a two-train system (offsite power available) and 1.8E-5 per demand as "typical" for a three-train system.

The first subissue, inadvertent MSIV closure, has the effect of turning the T_3-initiated transients into T_2-initiated transients. (T_1 transients are unaffected). If every transient led to MSIV closure (as Reference 2 Section 5.11 seems to imply), the parameters and sequences are straightforward:

$\Delta T_2 = 5.7 - 0.64 = 5.06$

$\Delta T_3 = -5.7$

For plants with a two-train AFW system:

$\Delta T_2 M^*L^*U$	7.6E-5
$\Delta T_3{}^*M^*PCSNR^*L^*U$	-9.5E-8

Net change, ΔF = 7.6E-5 per plant-year

For plants with a three-train AFW system:

$\Delta T_2 M^*L^*U$	1.4E-6
$\Delta T_3{}^*M^*PCSNR^*L^*U$	-1.7E-9

Net change, ΔF = 1.4E-6 per plant-year

The second subissue, AFW isolation, affects parameter L. The change in L is composed of two factors: the change in the probability of spurious isolation and the probability of failure to reopen on demand.

As discussed in issue 122.1.A we will assume a 5% minimum likelihood of spurious AFW isolation, and assume further that another plant with a high (e.g., 20%) likelihood exists.

The second factor is the failure of the isolation valves to reopen on demand. We will assume that Item 122.1.A has been addressed independently, and that this failure probability is now governed by the failure of a human operator to diagnose and correct the problem. The operator failure rate for such a situation is not independent of the spurious actuation error described above. We will assume, based on judgment, that 95% of the time, the operator will correct the error by resetting the inadvertent isolation and reopening the isolation valves.

For the more realistic (5% inadvertent isolation probability situation, the cut sets become:

$T_1M*LOPNRE*\Delta L*U$	4.7E-7
$T_2M*\Delta L*U$	2.4E-5
$T_3*M*PCSNR*\Delta L*U$	2.4E-7

Total ΔF = 2.5E-5 per plant-year

For the more extreme (20%) case, this change in core melt frequency would be four times this, or 9.9E-5.

Consequence Estimate

The consequence estimate is the same as that for Item 122.1.A. The "weighted-average" core melt will have consequences of 1.5E5 person-rem.

Cost Estimate

The core melt frequencies are in a range where costs that are within reason will not affect priority assignments. Consequently, no cost analysis has been made.

Value/Impact Assessment

Without a detailed design examination, it is not possible to determine exactly how many plants are affected. The B&W plants have an average of 29.5 calendar years (22 operational years) of lifetime left. Priority parameters are:

	Subissue 1	Subissue 2
Person-rem per reactor	250	80
Core melts per reactor-year	7.6E-5	2.5E-5

Other Considerations

1. There is no significant occupational exposure associated with the fix for this issue. The valves in question are not exposed to contaminated fluids, since they are in the secondary system.

2. The figures assume that the feed-and-bleed failure probability is 0.015. In actual fact, Reference 2 gives the impression that the Davis-Besse operators were rather reluctant to initiate feed-and-bleed. Thus, this figure may be somewhat optimistic, which would raise the priority scores still higher.

3. The two subissues were evaluated separately above because they involved two separate failures in the Davis-Besse event. Nevertheless, it should be noted that both involved the SFRCS. In essence, one control system apparently has the capability to shut off both main feedwater (by MSIV closure) and auxiliary feedwater. Although two distinct failures were involved at Davis-Besse, there may well be a single failure within the SFRCS which could do both. Deterministic evaluations of this system should recognize of the seriousness of such a failure mode.

CONCLUSION

Based on the core melt frequency figures above, this issue should be placed in the HIGH priority category.

ITEM 122.2: INITIATING FEED AND BLEED

DESCRIPTION

Historical Background

This issue deals with the adequacy of emergency procedures, operator training, and available plant monitoring systems for determining the need to initiate feed-and-bleed cooling following loss of the steam generator heat sink. It is based upon Findings 10, 17 and 18 in Sections 6.1.1 and 6.1.2 of NUREG-1154.[2] Essentially, the operators were reluctant to take the rather drastic step of initiating feed-and-bleed cooling, probably because they believed restoration of the AFW system was imminent. The fact that feed-and-bleed cooling releases primary coolant to the containment (implying an extensive shutdown for the purpose of decontamination) may also have influenced their actions. Finally, the normal control room instrumentation was inadequate to clearly inform the operators that feed and bleed was called for. The SPDS which would have displayed the necessary information was not operable.

The reactor vendors have provided their customers with feed-and-bleed procedures. Feed-and-bleed capability is not currently specifically required by the NRC, although the techniques, benefits, and costs are being evaluated as part of USI A-45. Basically, feed-and-bleed cooling is a method of last resort which can avert core damage if main and auxiliary feedwater is lost and other methods of decay heat removal are unavailable. For plants licensed without a PORV the lack of feed and bleed capability was a significant issue and the need for a highly reliable auxiliary feedwater system was emphasized.

Safety Significance

Probabilistic risk analyses give considerable credit for feed-and-bleed cooling. A failure rate of one or two percent is a typical assumption. However, the Davis-Besse event chronology leaves an impression that this failure probability may be overly optimistic.

In addition, it should be noted that, depending on specific plant design, there may be a fairly short time period in which feed-and-bleed cooling will be successful. If the plant operators delay too long before initiating feed-and-bleed cooling, their error may not be retrievable by later action.

This issue applies to all plants which can use feed-and-bleed techniques. This is all PWRs except for a few CE-designed plants which have no pressurizer PORVs.

Possible Solutions

The solution is a matter of emphasis on safety vs. operation, training in existing procedures, and possibly an upgrading of instrumentation at certain sites. In addition, the procedures themselves could be upgraded to make the

criteria for initiation of feed-and-bleed cooling more direct and unambiguous, leaving less room for operator reluctance. (For example, in the case of Davis-Besse, basing the initiation of feed and bleed on hot leg temperature rather than on steam generator parameters has been suggested.) Here, we will concentrate on ensuring that existing procedures are followed. The general technical aspects of feed-and-bleed decay heat removal will be addressed under USI A-45.

PRIORITY DETERMINATION

Frequency Estimate

The question of interest is, what is the change in core-melt frequency if the failure probability of feed-and-bleed cooling (U) is changed? References 5 and 6 assume a failure probability of 0.015 for non-ATWS sequences (RSSMAP parameter "HPMAN") and 0.10 for the (higher stress) ATWS sequences ("HPMAN1"). The operators' performance during the Davis-Besse event[2] leaves a strong impression that these figures are too low. We will assume, based purely on judgment, that failure probabilities of 0.10 for non-ATWS sequences and 0.50 for ATWS sequences are more reasonable estimates.

In making the calculations, the parameters were the same as in Issue 122.1.A, except:

a. The frequency of loss of main feedwater transients T_2,(momentary and sustained) was set at 2.13 per year, based on Reference 6.

b. The AFW failure probability (L) was set as follows, based on Reference 11:

	Offsite Power Available	No Offsite Power
3-train AFW	1.8E-5	5.1E-5
2-train AFW	1.0E-3	1.7E-3

In addition, the computerized RSSMAP analysis was changed as follows:

a. The probability of loss of onsite power (B_3) was changed to 1.3E-3, a figure more representative of a twin diesel system. (Oconee uses hydroelectric generators for emergency power.)

b. Oconee's capability of feeding the steam generators with the High Head Service Water System was disabled (HHMAN = 1.0).

A series of computer calculations was performed, in an attempt to obtain both the "best" answer and some information as to the sensitivity of the answer to a variety of conditions.

Calculation	ΔF, core melts per reactor-year
3-train AFW system HPMAN raised to 0.1 HPMAN1 raised to 0.5	3.3E-5
3-train AFW system, HPMAN raised to 0.1 ATWS sequences unchanged	9.2E-6
2-train AFW system HPMAN raised to 0.1 HPMAN1 raised to 0.5	1.0E-4
2-train AFW system HPMAN raised to 0.1 ATWS sequences unchanged	8.1E-5

Test case, original RSSMAP parameters. HPMAN raised to 0.1 ATWS sequences unchanged	2.0E-5

Clearly, the change in the feed-and-bleed failure probability has a strong effect on core-melt frequency. The figures span the decade from 1E-5 to 1E-4. We will use the first calculation (3.3E-5), remembering that the figure for a plant with a two-train AFW system will probably be greater. In addition, it should be noted that even a partial solution will make a significant reduction in core-melt frequency.

Consequence Estimate

The consequence estimate is the same as that for Item 122.1.A. The "weighted-average" core melt will have consequences of 1.5E5 person-rem.

Cost Estimate

The fix for this issue is likely to be procedural in nature, with upgrades in equipment more likely to be done under USI A-45. We will assume that six staff months per plant will suffice for refresher training on these procedures. NRR costs are likely to be on the order of six staff-months of generic effort plus two staff-weeks per licensee. For 55 operating PWRs, this is roughly three million dollars.

Value/Impact Assessment

There are 55 operating PWRs, with an aggregate of about 1700 calendar-years or 1300 operational years of lifetime remaining. Priority parameters can now be calculated:

Person-rem/reactor	100
Person-rem, Total	6500
Core melts/reactor-year	3.3E-5
Core melts/year	1.8E-3
Priority score, person-rem/million dollars	2000

Other Considerations

1. For a plant with a two-train AFW system, the per-reactor and per-reactor-year figures will be roughly three times as large.

2. This issue does not involve occupational exposure.

3. There is an offsetting saving which could be credited against the expenditures above. The cost of a core melt would be about a billion dollars, plus replacement power for the rest of the plant lifetime. In an actuarial sense, using the accident frequencies estimated above, assuming a 5% annual discount rate and subtracting off the feed-and-bleed cleanup costs which would reduce the core-melt costs, this corresponds to about a present worth of \$1.2M per plant.

4. In contrast to the sa[illegible] associated with averting a core-melt, an unnecessary use of feed and bleed will result in major cleanup costs. If half the uses of feed and bleed are unnecessary and a cleanup lasts six months, the actuarial cost is roughly 400,000 present-day dollars per plant (based on a residual frequency of unnecessary use of feed and bleed of 5E-4 per reactor-year).

CONCLUSION

Based on the figures above, this issue should be placed in the HIGH priority category.

ITEM 122.3: PHYSICAL SECURITY SYSTEM CONSTRAINTS

DESCRIPTION

Historical Background

This particular issue arose out of Finding 9 in Section 3.6 of NUREG-1154, which states:

> "The locked doors and valves in the plant had the potential for significantly hampering operator actions taken to compensate for equipment malfunctions during the event and were a significant concern to the equipment operators."

In the Davis-Besse event, the operators were able to reach the AFW pump room with no reported difficulty. There were difficulties in resetting and restarting the turbines and in opening the isolation valves, but these were not related to locking devices.

Safety Significance

Barriers and locks are present for purposes of physical security, as the title of this issue implies. In addition, barriers are provided for other purposes, such as personnel protection, fire zone isolation and flood protection. Valves are locked not only for security reasons, but also because inadvertent opening of these valves may have economic or safety consequences. The presence of the locking devices and barriers must strike a balance between these purposes and the fact that these devices may impede free movement in the plant and some local operations during an emergency. It should be noted that the control boards in the control room are also liberally supplied with keylock switches. This issue applies to all reactors.

Possible Solutions

The possible solution for this issue is to completely evaluate the net effect of a given barrier on plant safety, and either remove it or (in extreme cases) provide an alternate means of entrance (with its own locks), should the analysis so indicate.

PRIORITY DETERMINATION

This issue is not new. Issue 81 evaluated the impact of locked doors and barriers on safety, considering the frequency of a need for entry into the plant, the likelihood of procedural error (e.g., wrong key) and probability of successful forcible entry in a timely fashion. The conclusion was a "drop" priority assignment.

Item 81 considered only non-security barriers. A barrier that was installed for security reasons is not as likely to be forcibly penetrated in a few minutes. Moreover, the scenario here is slightly different than that of Issue 81. It should be noted, however, that the Davis-Besse experience confirms some of the assumptions of the Issue 81 evaluation, since there were in fact no problems with locked doors or valves.

Frequency Estimate

We will estimate frequency based on a loss of main feedwater event consistent with Issue 122.1.A. The frequencies and probabilities are: non-recoverable loss of main feedwater (0.67 per reactor-year), failure of auxiliary feedwater (use 1.0E-3 for a "typical" two-train system and 1.8E-5 for a "typical" three-train system), and failure of feed-and-bleed cooling (0.015).

We will further assume that a locked barrier may prevent entry into the auxiliary feedwater pump room(s) and that such entry could recover the AFW system. This is a high stress situation. Thus, we will assume that there is a 10% chance of human error (e.g., wrong key) and a 10% chance of non-recovery. (The chance of mechanical lock failure estimated in Issue 81 is 0.001.) We will not assume credit for forcible penetration.

We will not consider the padlocks and chains on the valve wheels, in view of the existence of bolt cutters and the fact that there will be two or three redundant trains.

The result is a change in core-melt frequency of 1.0E-7 for plants with two AFW trains, and 1.8E-9 for plants with three AFW trains.

Consequence Estimate

The consequence estimate is the same as that for Item 122.1.A. The "weighted-average" core melt will have consequences of 1.5E5 person-rem.

Cost Estimate

Issue 81 estimated a one-time evaluation of existing locked doors to cost $200,000. We will use this as a minimum per-plant cost, recognizing that an adverse finding will incur labor and equipment costs that may be much larger.

Value/Impact Assessment

We will assume a 30 year operational lifetime. The priority parameters become:

	3 AFW trains	2 AFW trains
Person-rem/reactor	0.01	0.5
Core melts/reactor-year	1.8E-9	1.0E-7
Priority score, person-rem/million dollars	≤0.04	≤2.3

Other Considerations

The analysis is based on the PWR design. It is not expected that a BWR design would be greatly different from that of a three AFW-train PWR, given the ability of HPCI, RCIC and the ADS low-pressure ECCS to mitigate transients.

CONCLUSION

Based upon the figures above, this issue should be placed in the LOW priority category.

References

1. Memorandum for H. L. Thompson, Jr., from D. M. Crutchfield, "Potential Immediate Generic Actions as a Result of the Davis-Besse Event of June 9, 1985," August 5, 1985.

2. NUREG-1154, "Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985," July 1985.

3. Memorandum for T. Speis from H. L. Thompson, Jr., "Short Term Generic Actions as a Result of the Davis-Besse Event of June 9, 1985," August 19, 1985.

4. Memorandum for H. R. Denton from T. P. Speis, "Adequacy of the Auxiliary Feedwater System at Davis-Besse," July 23, 1985.

5. NUREG/CR-1659, "Reactor Safety Study Methodology Application Program: Oconee No. 3 PWR Power Plant," May 1981.

6. NSAC-60, "A Probabilistic Risk Assessment of Oconee Unit 3," June 1984.

7. NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants," draft report for comment, May 1985.

8. Letter to T. Novak (NRC) from R. P. Crouse (Toledo Edison) enclosing Davis-Besse AFW analysis done by EDS Nuclear, Inc., December 31, 1981.

9. NUREG/CR-2770, "Common Cause Fault Rates for Valves: Estimates Based on Licensee Event Reports at U. S. Commercial Nuclear Power Plants, 1976-1980," February 1983.

10. NUREG/CR-2098, "Common Cause Fault Rates for Pumps," February 1983.

11. Memorandum for O. Parr from A. Thadani, "Auxiliary Feedwater System - CRGR Package," November 9, 1984.

12. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," February 1983.

GENERIC ISSUE MANAGEMENT CONTROL SYSTEM

The Generic Issues Management Control System (GIMCS) provides appropriate information necessary to manage safety related and environmental generic issues through technical resolution and completion. For the purpose of this management control system technically resolved is defined as the point where the staff's technical resolution has been issued. Generally, speaking, this occurs when the technical resolution has been incorporated into one or more of the following:

(a) Commission policy statement/orders
(b) NRC Regulations
(c) Standard Review Plan
(d) Regulatory Guide
(e) Generic Letter

GIMCS is part of an integraded system of reports and procedures that would manage generic safety issues, TMI-related issues, and proposed new generic issues through the stages of prioritization, technical resolution, development of new criteria, review and approval, public comments, and incorporation into the Standard Review Plan (SRP), as appropriate. NUREG-0933 provides an evaluation for a recommended priority listing based on the potential safety significance and cost of implementation for each issue; NRR Office Letter Number 40 provided procedures and criteria for adding new generic issues to the system; and GIMCS provides proposed scheduling for resolving and completing issues on the prioritized listing. GIMCS will provide information to manage and control issues that are ranked High-priority generic issues, Medium-priority generic issues, issues for which possible resolution has been identified for evaluation, issues for which a technical resolution is available (as documented by memorandum, analysis, NUREG, etc.), and issues designated by the Director of NRR as issues for which resources have been made available for resolution and completion. Issues ranked as either "Low" or "Dropped" are not allocated resources, therefore, there is no resolution to be tracked by GIMCS.

Some new generic issues prioritized and processed in accordance with NRR Office Letter No. 40 may not have resources allocated for resolution and completion. These issues will be listed in GIMCS as inactive issues. These will generally be Medium priority issues that have no safety deficiency demanding high-priority attention, but there is a potential for safety improvements or reduction in uncertainty of analysis that may be substantial and worthwhile. Efforts for resolution of these issues will be planned, over the next several years, but on a basis that will not interfere with the resolution of High-priority generic issue work or other high priority work. Thus, some (Medium) generic issues will be inactive until such time as resources become available to resolve the various issues. As resource allocations are directed at issue resolution, they will become active. The detailed schedule for resolving and completing the generic issue will be developed and monitored by the management control system.

Management and control indicators used in GIMCS are defined as follows:

1. Item No. - Generic Issue Number.

2. Issue Type - Safety, Environmental or Regulatory Impact High, Note 1 or Note 2 (From NUREG-0933), Medium.

3. Action Level - Degree of management attention need to process generic issues in accordance with established schedules
   - L1 - No management action is necessary
   - L2 - Division Director action is necessary
   - L3 - Director NRR action is necessary

4. Office/Div/Br - 1st listed has lead responsibility for resolving issue, others listed have input to resolution.

5. Task Manager - Name of assigned individual responsible for schedule updating.

6. Tac Number - Each issue should be assigned a TAC #.

7. Title - Generic Issue Title.

8. Work Authorization - Who or what authorized work to be done on generic issue.

9. Contract Title - Provide Contract Title (if contract issued).

10. Contractor Name/FIN No. - Identify Contractor Name and FIN Number (as appropriate). If contract is not yet issued, indicate whether the contract is included in the FIN plan.

11. Work Scope - Describes briefly the work necessary to technically resolve and complete the generic issue.

12. Affected Documents - Identifies documents that the technical resolution will be incorporated into to identify new criteria.

13. Status - Describes current status of work.

14. Problem/Resolution - Identifies potential problem areas and describes what actions are necessary to resolve them.

15. Technical Resolution - Identifies detailed schedule of milestone dates that are required for completing the issue through the issuance of the SRP revision or other change that documents requirements.

    Milestones - Selected significant milestones. The "original" schedule remains unchanged. Changes in schedule are listed under "Current". Actual completion are listed under "Actual".

TYPICAL MILESTONES

Other Division Involvement	Original	Current	Actual
o Date information requested from Division			
o Date received from Division			
Contractor Information			
o Proposal Solicited			
o Proposal Evaluated and Accepted			
o Contract Schedule, if applicable			
o Testing Schedule, if applicable			
o Draft NUREG/CR report from contractor/consultant			
Staff review of draft NUREG/CR report			
Value Impact Statement prepared (coordinated with SPEB and RRAB as applicable)			
Final report prepared by Division (include SPEB preliminary comments and SRP revision)			
---------- 2 wks			
Final report forwarded to DST for processing			
----------- 2 wks			
CRGR Package to NRR Director for Review			
----------- 1 mo			
OMB Clearance obtained concurrently if applicable			

Review Package to CRGR

----------- 1 mo

CRGR review and EDO approval
completed

----------- 1 mo

Federal Register Notice of
Issuance of SRP for
Public Comment

----------- 3 mo

Division review of public
comments completed

----------- 2 wks

Comments incorporated and
transmitted to DST for
processing

----------- 2 wks

Final CRGR package to
NRR Director for review

----------- 1 mo

Review Package to CRGR

----------- 1 mo

CRGR review and EDO approval
completed

----------- 1 mo

Federal Register Notice of
Issuance of SRP

GENERIC ISSUE MANAGEMENT CONTROL SYSTEM

Issue Number	Issue Type	Action Level	Office/Div/Br	Task Manager	Tac No
		Active-L1	NRR/	TBP	TBP

Title ----------------

Work Authorization --- Memorandum to from H. R. Denton dated

Contract Title ------- To Be Provided.

Contractor Name/
FIN No. ------------ To Be Provided.

Work Scope ----------- To Be Provided.

Affected Documents --- To Be Provided.

Status --------------- To Be Provided.

Problem/Resolution --- To Be Provided.

Technical Resolution - To Be Provided.

Milestones	Original	Current	Actual
New Issues - Schedule To Be Developed			

As of First Quarter FY-84

OFFICE							
SURNAME							
DATE							