September 10, 1986

MEMORANDUM FOR: Themis P. Speis, Director
Division of Safety Review and Oversight
Office of Nuclear Reactor Regulation

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

SUBJECT: SCHEDULE FOR RESOLVING GENERIC ISSUE NO. 125.II.7, "REEVALUATE PROVISION TO AUTOMATICALLY ISOLATE FEEDWATER FROM STEAM GENERATOR DURING A LINE BREAK"

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," were reviewed by the staff to identify potential generic issues and to make recommendations regarding the need for staff actions. Twenty-nine separate subtasks were identified as long-term actions for prioritization. This memorandum addresses the prioritization of one of these subtasks: Issue 125.II.7, "Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break."

The technical resolution for Generic Issue No. 125.II.7 has a HIGH priority ranking. This memorandum approves NRR staff taking appropriate actions necessary to resolve this issue. The evaluation of the subject issue is provided in Enclosure 1.

In accordance with NRR Office Letter No. 40, "Management of Proposed Generic Issue," the resolution of this issue 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 (Enclosure 2). 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 within six weeks and should be sent to the Safety Program Evaluation Branch, DSRO, NRR.

The enclosed prioritization evaluation will be incorporated into NUREG-0933, "A Prioritization of Generic Safety Issues," and is being sent to the regions, other offices, the ACRS, and the PDR by copy of this memorandum and its enclosures to allow others the opportunity to comment on the 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.

RD-8
Generic
Items

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All comments should be sent to the Safety Program Evaluation Branch, DSRO, NRR. Should you have any questions pertaining to the contents of this memorandum, please contact Ronald Emrit (X-24576).

ORIGINAL SIGNED BY:

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Prioritization Evaluation
2. GIMCS Information Sheet

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<u>ENCLOSURE 1</u>

<u>PRIORITIZATION EVALUATION</u>

Issue 125.II.7: Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break

ITEM 125.II.7: REEVALUATE PROVISION TO AUTOMATICALLY ISOLATE FEEDWATER FROM STEAM GENERATOR DURING A LINE BREAK

DESCRIPTION

Historical Background

This issue arose out of the Davis-Besse incident of June 9, 1985.[A] During the course of the investigation of the event, it was pointed out[B] that the benefits of AFW isolation are probably more than outweighed by the negative aspects of this feature.

Safety Significance

The automatic isolation of AFW from a steam generator is provided to mitigate the consequences of a steam or feedwater line break. The isolation logic, usually triggered by a low steam generator pressure signal, closes all main steam isolation valves and also isolates AFW from the depressurizing steam generator. (The AFW flow is diverted to an intact steam generator). The purposes of the AFW isolation are threefold:

(1) The break blowdown is minimized. Shutting off AFW will not prevent the initial secondary side inventory from blowing down. However, the isolation will prevent continued steaming out of the break as decay heat continues to produce thermal energy.

(2) Overcooling of the primary system is reduced. As the depressurizing steam generator blows down to atmospheric pressure, the primary system is cooled down, causing primary coolant shrinkage and (if the event occurs near the end of the fuel cycle) a return to criticality, which adds a modest amount of thermal energy to the transient. Shutting off feedwater to the faulted steam generator will reduce this effect, although once again the initial blowdown will be the dominant factor.

The significance of these first two considerations is in containment pressure. The containment is designed to accommodate a primary system blowdown followed by decay heat boiloff (the large break LOCA). A steam or feedwater line break within containment might cause the containment design pressure to be exceeded if the AFW isolation were not present.

(3) The AFW isolation is needed to divert AFW flow to the intact steam generator(s). For the case of a two-loop plant with a two-train AFW system, this is needed to meet the single failure criterion in supplying feedwater to the intact steam generator. (The situation becomes more complex for other cases, e.g. a four-loop plant with a three-train AFW system). Note that, unless the line break is in the AFW line, core cooling would still meet the single failure criterion even without the isolation, since the faulted steam generator would still be capable of heat transfer.

In summary, the automatic isolation is needed only to help mitigate a relatively rare event (steam or feedwater line break) and even then is only remotely connected with sequences leading to core-melt.

In contrast, this isolation has definite disadvantages. If both channels of the controlling system were to spontaneously actuate during normal operation, all AFW would be lost and the MSIVs would close. Most newer plants use turbine-driven main feedwater pumps. Thus, main feedwater would be lost also. If the plant operators fail to correctly diagnose and correct the problem, only feed-and-bleed cooling would be available to prevent core-melt. Similarly, if spurious AFW isolation were to occur during the course of another transient, once again only feed-and-bleed cooling would be available to prevent core-melt.

Reference B also notes that the long term success of AFW for main feedwater transients, steam generator tube ruptures, and small LOCAs may also be compromised. During controlled cooldown, the thresholds for automatic AFW isolation are crossed. Procedures call for operators to lock out the isolation logic as the steam generator pressure approaches the isolation setpoint. Under the circumstances, the accompanying distractions make it possible that the operators will forget to override the AFW isolation logic in the permissive window. Thus, AFW reliability in these scenarios may be significantly degraded.

The safety significance of this issue arises from the fact that the negative aspects involve accident sequences which have more frequent initiators, and more significant consequences, than those of the positive aspects.

Possible Solutions

Reference B proposes a very straightforward solution: simply disconnect the AFW isolation valve actuators from the automatic logic and depend on plant procedures, i.e. have the operators close the AFW isolation valves (by remote manual operation from the control room) in the event of a line break. These procedures would require careful verification of the existence of a line break before isolating a steam generator from AFW.

PRIORITY DETERMINATION

Frequency Estimate

It is necessary to calculate estimates of both the positive and negative aspects of disabling the automatic AFW isolation. The positive aspects are due to a decrease in the frequency of loss of all feedwater events. There are three accident sequences of interest.

(1) The first sequence is initiated by a spontaneous actuation of both channels of the isolation logic. (We will assume a two-loop plant design for prioritization purposes). There is no data readily available for such actuations. However, it is possible to make an educated guess. Reference C provides some perspective, based upon actual experience with other systems:

Inadvertent Safety Injection Signal, PWR	0.06/RY
MSIV Closure, PWR	0.03/RY
Steam Relief Valve Open, PWR	0.04/RY
Inadvertent Startup of BWR HPCI	0.01/RY

Based upon these figures, it is expected that spontaneous actuations will occur with a frequency on the order of 0.03/RY. Of course, this would isolate only one steam generator. However, such systems generally have a common mode failure probability on the order of 5%. (In addition, the second train of AFW has an unavailability due to other causes of roughly 1%. However, the main feedwater system would still be available in this case). Thus, the frequency of both steam generators isolating is (0.03/RY) (0.05), or 1.5E-3/RY. Of course, the plant operators are likely to reset the logic and turn the transient around. We will assume a 1% (minimum) failure probability for recovery by operator action. This leaves feed-and-bleed cooling for which we will assign a typical failure probability value of 0.20 and a maximum failure probability of 0.60, based on the calculations presented under Item 125.II.9, "Enhanced Feed-and-Bleed Capability." Multiplying these figures together gives a core-melt frequency of 3E-6/RY typical, 9E-6/RY maximum.

(2) The second sequence is initiated by another, independent transient. During the course of this transient, and the consequent perturbation of a great many plant systems, the AFW isolation logic is triggered. The MSIVs close, causing a loss of main feedwater (if main feedwater has not previously been lost), and the AFW isolates. Again, unless the AFW isolation valves are reopened, only feed-and-bleed is available as a means of core cooling.

The AFW isolation logic can be triggered during a transient in two ways. The first is by some type of inadvertent systems interaction, e.g. electromagnetic coupling. The proper fix for this problem is to eliminate the systems interaction which may well have other consequences in addition to AFW isolation. Therefore, this effect will not be considered here.

The second way to trigger AFW isolation is by the actual existence of low pressure in the secondary system, caused by the initiating transient. In this case, the isolation is working as designed (but not as intended). Low pressure transients are relatively rare, since the steam space in question is usually right on top of a significant quantity of water at saturation temperature. Low pressure will occur only if steam is vented at a rapid rate in sufficient quantity to cool the water inventory via boiloff to the point where saturation pressure drops below the AFW isolation setpoint. The other possibility is a dryout of the steam generator.

This is possible for B&W plants because of the relatively low water inventory in the steam generators. However, such an event in a Westinghouse or CE plant would probably imply that the main feedwater and AFW had already failed.

There is no readily available way of estimating the probability of a pressure drop, given a transient. However, Reference C gives a frequency of 0.04/RY for events where PWR steam relief valves open. Thus, we can assume that depressurization events occur with at least this frequency. If we further assume that perhaps 10% of these pressure drops are deep enough to trigger AFW isolation, and again assume a 1% probability of failure of the operators to recover AFW, the resulting core-melt frequencies are 8E-6/RY typical, 2.4E-5/RY maximum.

(3) The third sequence involves the long term success of AFW for main feedwater transients. During controlled cooldown, the thresholds for automatic AFW isolation are crossed. Procedures call for the operators to lock out the isolation logic as the steam generator pressure approaches the setpoint. If the operators fail to do so, both trains of AFW will isolate. Main feedwater is also unavailable, since its loss initiated the transient. Again, only feed-and-bleed would be available for core cooling.

Non-recoverable loss of main feedwater events are estimated to occur with a frequency of 0.64/RY.[D] We will assume a 1% minimum probability of operator failure to bypass the isolation logic and another 1% minimum probability of failure of the operators to recover the AFW system. In addition, there is still feed-and-bleed cooling which, because the plant is already partially cooled down, should have a better than usual chance of succeeding. We will therefore assume 10% instead of 20% or 60% for feed-and-bleed failure probability. The result is a core-melt frequency of 6.4E-6/RY.

The three sequences above add up to a "typical" core-melt frequency of 1.7E-5/RY and as much as 3.9E-5/RY for a plant with marginal feed-and-bleed capability. Now we must estimate the negative aspects of the proposed fix.

The first negative scenario is the feedwater line break. Here, a break in the feedwater line to one steam generator initiates the sequence. With the proposed fix, the line is not isolated and one train of AFW simply pumps water out of the break. If the operator fails to manually isolate the break, the remaining AFW train fails, and feed-and-bleed techniques fail, core-melt will result.

Steam and feedwater line breaks are estimated to occur at a combined rate of 1.0E-3/RY (see Issue A-22). Because steam lines are larger and not as subject to water hammer phenomena, the feedwater lines are expected to be more likely to break than the steam lines. We will therefore assume that feedwater lines will break with a frequency of 9E-4/RY, i.e. 90% of the total line break frequency.

The unaffected single train of AFW should have a failure probability on the order of 0.01 or less. Consistent with the positive scenario calculations, we will assume a 1% probability of operator failure to manually isolate the affected steam generator and a 20% typical, 60% maximum feed-and-bleed failure probability. The product is a core-melt frequency of 1.8E-8/RY typical and 5.4E-8/RY maximum.

The remaining scenario is a steam line break. This scenario may involve the theoretical possibility of containment failure by overpressure, but does not lead to core-melt. We will assume a 1.0E-3/RY frequency of line break as before and a 10% probability that the line break is in the steam lines as opposed to the feedwater line breaks of the previous scenario. Once again, the probability of the operator to fail to manually isolate is assumed to be 1%. The frequency of higher than expected containment pressure due to long term steaming in the faulted steam generator is then 1.0E-6/RY.

Consequence Estimate

The core-melt sequences under consideration here 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(s). These are conditions likely to produce significant hydrogen generation and combustion.

The Zion and Indian Point PRA studies used a 3% probability of containment failure due to hydrogen burn (the "gamma" failure). We will follow this example and use 3%, bearing in mind 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 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 per event.

These figures should cover all PWRs with large dry containments. They do not apply to ice condenser containments. Because of the low free volume in such a containment, failures due to overpressure are more likely and the averaged consequences may be significantly greater. However, we are not aware of any ice condenser plant which has an automatic AFW isolation affected by this issue.

The steam line break - containment rupture scenario is different. The containment pressure is unlikely to exceed the design pressure by more than a few percent, if at all. In most cases, the containment is calculated to fail at 2 to 2 1/2 times its design pressure. Therefore, containment failure by overpressure is at most a very remote theoretical possibility. We will assume that the overpressure failure probability cannot be greater than 3%, the hydrogen burn figure (a highly conservative assumption). The only radioactive release comes from the containment atmosphere and any primary coolant leakage or discharge from the PORV(s). We have no consequence estimates for such an event. However, the consequences can be conservatively bounded by those of a PWR-8 event, which is a successfully mitigated LOCA with failure of the containment to isolate. The PWR-8 consequences are 7.5E4 person-rem. Thus, the steam line break event will have "average" consequences of at most (0.03)(7.5E4) or 2250 person-rem, and probably much less.

Cost Estimate

The proposed fix for this issue is simply to remove some leads from some equipment, an action which is likely to be more than paid for by decreased maintenance and testing. Nevertheless, even a relaxation of requirements such as this will require review of each affected plant's isolation logic, to be certain that the net effect is an increase in plant safety. In addition, technical specification and procedural changes, with their associated paperwork, will be necessary. We will assume per plant costs of $32,000 to the industry and $25,000 to the NRC, which are typical for a complicated and controversial technical specification change.

Value/Impact Assessment

It is not known how many plants are affected by this issue. In many plants, the AFW isolation logic has provisions to prevent isolation of feedwater to more than one steam generator. Others may not even have this isolation logic. We will assume that about 25% of the PWRs will be affected by this issue.

There are 83 PWRs. As of spring 1987 (the earliest that this issue is likely to result in changes), the remaining collective calendar life will be 2571 reactor-years. At a 75% utilization factor, this is 1928 reactor-years, or about 23 operational years per reactor.

The change in core-melt frequency is the algebraic sum of the various scenarios:

	Core-melt averted per reactor-year	
	Typical	Maximum
Spontaneous actuation	3.0E-6	9.0E-6
Transient initiated	8.0E-6	2.4E-5
Cooldown initiated	6.4E-6	6.4E-6
Feedwater line break	-1.8E-8	-5.4E-8
Net change in core-melt frequency	1.7E-5	3.9E-5

The net change in person-rem per reactor-year is obtained by multiplying the change in core-melt frequency by 1.5E5 person-rem (average) per core-melt. Then, the steam line break scenario must be subtracted. The consequences of the steam line break scenario (upper bound) are simply (1.0E-6 overpressures/RY) [2250 (average) person-rem/overpressure], or 2.3E-3 person-rem per reactor-year.

	Change in person-rem/RY	
	typical	maximum
core melt scenarios	2.6	5.9
steam line break	≤0.0023	≤0.0023
Net change:	2.6	5.9

Priority parameters can now be calculated:

Person-rem/reactor (max)	140
Person-rem, total all reactors	1300
Core-melt/reactor-year (max)	3.9E-5
Core-melt/year, total all reactors	3.5E-4
Person-rem per million dollars	1100

Other Considerations

(1) It should be noted that the maximum values are based upon a plant with marginal feed-and-bleed capability. The subset of PWRs which are affected by this issue may not include such a plant. Thus, the "maximum" plant may not exist.

(2) The proposed fix does not involve work within radiation fields and thus does not involve occupational radiation exposure (ORE). However, the ORE averted due to post feed-and-bleed cleanup and post-core-melt cleanup is a consideration. Reference E estimates the ORE associated with cleanup to be about 1800 person-rem after a primary coolant spill and about 20,000 person-rem after a core-melt accident. The "typical" frequency of feed-and-bleed events is simply the "typical" core-melt frequency (1.8E-5/RY) divided by the feed-and-bleed failure probability (0.20). The actuarial figures are:

Averted per plant feed-and-bleed cleanup ORE	3.6
Averted per plant core-melt cleanup ORE	7.9
Per plant total:	11.5
Total, all plants:	240.0

Thus, the averted ORE is not dominant, but is still a significant fraction of the averted public risk.

(3) The proposed fix reduces core-melt frequency and the frequency of feed-and-bleed events and therefore averts cleanup costs and replacement power costs. The cost of a feed-and-bleed usage is dominated by roughly six months of replacement power while the cleanup is in progress. If the average frequency of such events is 1.7E-5/0.20 or 8.5E-5/RY and the average remaining lifetime is 23 operational years at 75% utilization, and making the usual assumptions of a 5% annual discount rate and a replacement power cost of $300,000 per day, the actuarial savings for feed-and-bleed cleanup works out to be $55,000. Similarly, the actuarial savings of averted core-melt cleanup (which is assumed to cost one $1 billion if it happens) are about $200,000. The actuarial savings from

replacement power after a core-melt up to the end of the plant life are about $260,000. (This last figure represents the lost capital investment in the plant). Obviously, these savings would more than offset the cost of the fix if they were included.

(4) The analysis of the first negative scenario, the feedwater line break, assumed that non-isolation of the ruptured line would cause one AFW train to fail. A special situation can arise for plants with a limited AFW water supply (e.g. saltwater plants). In such a case, the continued loss of clean water out of the feedwater line break can in theory cause failure of the second AFW train by exhausting the water supply, provided that the loss is not terminated either by the operator or by protective trips (for runout protection) on the first AFW train. In such a case, the scenario's negative contribution to the proposed fix's averted core-melt frequency rises from -1.8E-8 to -1.8E-6 (typical). The net change in core-melt frequency would then drop from 1.7E-5 to 1.6E-5, which would not change the conclusion.

CONCLUSION

Based upon the figures above, particularly the core-melt frequencies, this issue should be placed in the HIGH priority category.

REFERENCES

A. Memorandum for T. Speis from H. L. Thompson, "Longer-Term Generic Actions as a Result of the Davis-Besse Event of June 9, 1985," November 6, 1985.

B. Memorandum for H. L. Thompson and W. Minners from F. H. Rowsome, "Another Generic Safety Issue Suggested by the Davis-Besse Incident of June 9, 1985," September 9, 1985.

C. EPRI NP-2230, "ATWS: Frequency of Anticipated Transient," January 1982

D. Memorandum for W. Minners from K. Kniel, "Value/Impact Assessment for Draft CRGR Package Requiring Upgrading of Auxiliary Feedwater Systems in Certain Operating Plants," January 16, 1986.

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

ENCLOSURE 2

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, MEDIUM, or Nearly-Resolved (Note 1 or Note 2 from NUREG-0933)
3.	Action Level	- Degree of management attention needed to process generic issues in accordance with established schedules: L1 - No management action necessary L2 - Division Director action necessary L3 - NRR Director action necessary L4 - EDO Director action necessary L5 - Commission action necessary
4.	Office/Div/Br	- First-listed has lead responsibility for resolving issue; others listed have input to resolution.
5.	Task Manager	- Name of assigned individual responsible for resolution
6.	TAC Number	- TAC number assigned to the issue.
7.	Title	- Generic Issue Title
8.	Work Authorization	- Who or what authorized work to be done on the issue.
9.	Contract Title	- Contract Title (if contract issued).
10.	Contractor Name/FIN	- Contractor Name and FIN. (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 into which the technical resolution will be incorporated.
13.	Status	- Describes current status of work.
14.	Problem/Resolution	- Identifies 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 SRP revisions or other changes that document requirements.
	Milestones	- Selected significant milestones. The "original" scheduled dates remains unchanged. Changes in scheduled dates are listed under "Current." Actual completion dates are listed under "Actual."