0 8 OCT 1986

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

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

SUBJECT: SCHEDULE FOR RESOLVING GENERIC ISSUE NO. 125.II.1, "NEED FOR ADDITIONAL ACTIONS ON AFW SYSTEMS"

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 four of these subtasks: (1) Issue 125.II.1.a, "Two-Train AFW Unavailability"; (2) Issue 125.II.1.b, "Review Existing AFW Systems for Single Failures"; (3) Issue 125.II.1.c, "NUREG-0737 Reliability Improvements"; and (4) Issue 125.II.1.d, "AFW Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants."

The prioritizations of the above four issues show the following results in Enclosure 1: (1) the safety concerns of Issues 125.II.1.a and 125.II.1.d are being addressed in the resolution of Issue 124; (2) Issue 125.II.1.b has a HIGH priority ranking; and (3) Issue 125.II.1.c has a DROP priority ranking. Therefore, only the resolution of Issue 125.II.1.b will be pursued.

In accordance with NRR Office Letter No. 40, "Management of Proposed Generic Issues," the resolution of Issue 125.II.1.b 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 evaluations will be incorporated into NUREG-0933, "A Prioritization of Generic Safety Issues," and are 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 evaluations. Any changes as a result of comments will be coordinated with you. However, the

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schedule for the resolution of Issue 125.II.1.b should not be delayed to wait for these comments. 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:
Richard H. Vollmer

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Prioritization Evaluations
2. GIMCS Information Sheet

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ENCLOSURE 1

PRIORITIZATION EVALUATIONS

Issue 125.II.1.a:	Two-Train AFW Unavailability
Issue 125.II.1.b:	Review Existing AFW Systems for Single Failures
Issue 125.II.1.c:	NUREG-0737 Reliability Improvements
Issue 125.II.1.d:	AFW Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants

ITEM 125.II.1: NEED FOR ADDITIONAL ACTIONS ON AFW SYSTEMS

This issue arose during the investigation of the Davis-Besse loss of all feedwater event of June 9, 1985.[A] During the event, the main feedwater system was lost and the reactor scrammed. The AFW system should have activated and supplied feedwater to the steam generators to enable them to remove decay heat. However, during the course of the event, several failures occurred (see Issue 122) that precluded using the steam generators to remove decay heat from the primary system. The event highlighted the importance of the AFW system and also demonstrated that the AFW system might not have a reliability commensurate with its importance.

If the main feedwater system shuts down for any reason, the AFW system will supply sufficient feedwater to the steam generators to remove reactor decay heat. If the AFW system were to fail also, there would be no feedwater supply at all. The steam generators would boil off their remaining liquid water inventory and then dry out. Depending on specific plant design, core uncovery will take place roughly 30 to 90 minutes after the transient begins. After steam generator dryout, there would be no decay heat removal and the continuing thermal energy production in the core would result in primary system heatup.

In most cases, the only means of decay heat removal involve use of the AFW system, recovery of the main feedwater system, or the use of feed-and-bleed techniques. Of the three means, the use of the AFW system is subject to the highest availability. The failure of the main feedwater system has roughly a 20% probability of not being recoverable in time. Moreover, use of feed-and-bleed techniques will release primary coolant to the containment necessitating extensive (and expensive) cleanup. The use of feed-and-bleed techniques, which remove decay heat by venting hot primary coolant to the containment and replacing the lost inventory in the primary system by means of the high pressure ECCS, could still prevent core uncovery. If feed-and-bleed fails, the primary system will increase in temperature and pressure to the point where the primary system safety valves open. The pressure increase will then terminate, but the primary coolant will boil off until the core is uncovered and melts.

AFW systems are safety-grade systems. In addition, the availability of feed-and-bleed techniques provides a diverse backup. Nevertheless, AFW reliability is very important for two reasons. First, loss of main feedwater is a relatively common event, occurring roughly three orders of magnitude more often than (for example) small break LOCAs. Thus, the AFW system is challenged far more often than the high pressure ECCS and therefore has a commensurately greater need for high reliability. Second, although feed-and-bleed techniques provide a backup to AFW for removing reactor decay heat, feed-and-bleed is a means of core cooling for which the plant was not designed and may have a relatively high failure probability (see Item 125.II.9). Because of these two reasons (frequent challenges and poor backup capability), it is very important that the AFW system have very high reliability.

Because loss of feedwater events are relatively frequent, the AFW system is subject to frequent challenges. Therefore, the AFW system must be characterized by very high availability. This issue consists of four parts, each of which seeks to ensure adequate AFW reliability:

(a) Two-Train AFW Unavailability

This issue is concerned that AFW systems consisting of only two-trains may not have adequate reliability.

(b) Review Existing AFW Systems for Single Failures

This issue seeks confirmatory deterministic reviews of AFW systems at operating plants to ensure that they meet the single failure criterion.

(c) NUREG-0737 Reliability Improvements

This issue proposes that PRA analyses (i.e. fault trees) be performed on AFW systems at operating plants to ensure adequate reliability.

(d) AFW Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants

This issue is concerned explicitly with a possible design problem at B&W plants.

These four parts of the issue are prioritized separately below:

ITEM 125.II.1.A: TWO-TRAIN AFW UNAVAILABILITY

DESCRIPTION

This issue is centered upon the subject of the reliability of the AFW system.[A] There are seven older PWRs that have two-train AFW systems. (Originally, there were more but some plants have since added a third train or made other equivalent upgrades). These AFW systems generally consist of one motor-driven train and one turbine-driven train and thus possess some diversity as well as redundancy. However, the turbine-driven trains have not proven to be as reliable as the motor-driven trains (except, of course, for the case where all AC power is lost). The more modern practice has been to use a three-train system where two trains are motor-driven and one is driven by a steam turbine. Such a system will, in principle, be more reliable than the two-train systems described above, both because of the greater redundancy of the three vs. two trains and because of the lower reliance on the steam turbine.

CONCLUSION

This issue is the same as Issue 124, "AFW System Reliability." Issue 124 will consider whether AFW system unavailability needs to be improved for plants with two-train designs.[C] Therefore, this issue should be DROPPED as a separate issue.

ITEM 125.II.1.B: REVIEW EXISTING AFW SYSTEMS FOR SINGLE FAILURE

DESCRIPTION

Historical Background

This issue is centered upon the subject of the reliability of the AFW system.[A] The AFW system is considered an engineered safety feature and thus is required to meet the single failure criterion which can be considered a very primitive reliability requirement. An unsuspected single failure susceptibility could increase the AFW system failure probability by two orders of magnitude or more.

Safety Significance

The issue addresses the concern that there may be some unsuspected single failures which were not detected during the licensing process. Therefore, this issue proposes to re-review the AFW systems of all operating PWRs to make doubly sure that no single failures exist which by themselves could cause all AFW trains to fail.

Proposed Solution

The systems to be examined have already been subjected to licensing review. Therefore, any single failures are not going to be obvious, but instead are likely to be quite subtle. Very thorough reviews will be required. It must also be remembered that AFW trains are intentionally designed to be independent. Any single failure found is most likely to be a subtle design anomaly which the designer (as well as all subsequent reviewers) failed to notice.

Several AFW systems have been examined by the Office of Inspection and Enforcement in the course of the Safety System Functional Inspection (SSFI) program. Conversations with the SSFI team have indicated that some single failure problems as well as other potential common mode failures have been found by this program. However, these problems were not discovered by examining system design, but instead arose in the course of very thorough investigations involving extended site visits, equipment inspection, and interviews as well as design reviews. Therefore, the proposed solution is not a simple design review, but instead is a more thorough investigation along the lines of the SSFI program.

Frequency Estimate

The sequence of interest is straightforward. It is initiated by a non-recoverable loss of main feedwater. If the AFW system fails, the SUFP is not re-enabled in time, and feed-and-bleed techniques fail, core-melt will ensue.

For the initiating event frequency (non-recoverable loss of main feedwater), we will use 0.64 event per reactor-year, based upon the Oconee PRA done by Duke Power Co.[C] This figure is based upon fault tree analysis and should be reasonably representative of most main feedwater system designs.

For a three-train AFW system, Reference E gives a "typical" unavailability of 1.8E-5 per demand. The presence of a single failure susceptibility will greatly increase this figure to perhaps the square root of the original figures because half the redundancy would be removed. The change in AFW unavailability would then be about 4.2E-3 failures per demand. We will assume a typical value of 0.20 for the failure probability of feed-and-bleed cooling, based upon the calculations presented under Issue 125.II.9, "Enhanced Feed-and-Bleed Capability." Multiplying these figures out, the change in core-melt frequency is:

$$(0.64/\text{year})(4.2\text{E-}3)(0.20) = 5.4\text{E-}4/\text{year}$$

Consequence Estimate

The core-melt sequence under consideration here involves 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 basemat 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 SSFI program has required about 1000 staff-hours per plant and system. This is about $50,000 of salary and overhead. In addition, hardware changes are likely to cost on the order of $100,000 per plant (i.e. more than $10,000 but less than $1,000,000) plus another $50,000 in paperwork. Thus, we will assume a per-plant cost on the order of $200,000.

Value/Impact Assessment

There are 80 PWRs operating or under construction. As of March 1988 (the earliest that any hardware changes are likely to be made), these 80 plants will have a combined remaining license lifetime of 2508.4 calendar-years. At a 75% capacity factor, this is about 23.5 years of operation per plant. Priority parameters can now be calculated:

Remaining years of operation, per plant	23.5
Cost per plant, millions of dollars	0.2
Person-rem per core-melt	1.5E5
Change in core-melt frequency, per reactor-year	5.4E-4
Change in person-rem per reactor	1900
Change in person-rem per million dollars	9500

Other Considerations

(1) The AFW system and its support systems do not contain contaminated fluids and are located outside of containment. Thus, there is no occupational exposure associated with the fix for this issue.

(2) Averted accident costs and averted cleanup exposure are considerations, but will only drive the priority figures still higher. Thus, they will change no conclusions and will not be treated here.

(3) The high values of the parameters are predicated on finding at least one plant that needs upgrading. The SSFI personnel emphasized that this is not likely to happen without an approach similar to that of the SSFI, but such an approach is likely to bear fruit. It may be feasible to incorporate this issue into the SSFI program.

CONCLUSION

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

ITEM 125.II.1.C: NUREG-0737 RELIABILITY IMPROVEMENTS

DESCRIPTION

Historical Background

After the TMI accident, all PWR licensees were asked to perform an unavailability analysis of their AFW systems. This information is now somewhat out of date partly because the AFW systems were subject to some (NUREG-0737) modifications after the analyses were made[B] and partly because the analyses themselves are rather primitive by modern standards.

Safety Significance

This item seeks to upgrade the AFW unavailability analyses to reflect the NUREG-0737 modifications and improvements and to ensure that the AFW system reliability is commensurate with the system's safety importance.

Proposed Solution

The proposed solution for this issue is to perform a PRA of all AFW systems and require modification of any systems which have an unacceptably high failure probability.

PRIORITY DETERMINATION

Issue 124, "AFW System Reliability," will consider whether seven PWRs with two-train AFW systems have AFW system unavailabilities that need to be improved. Therefore, this issue need cover only the three-train AFW systems.

To prioritize this issue, several questions need to be answered. First, how reliable must the AFW system be to have reliability commensurate with its safety importance? Generic Issue 124 has selected an unavailability of 1.0E-4 failures per demand as the upper limit of acceptability.[C] We will use this same figure. The second question is, how many plants are likely to be found which cannot meet the 10^{-4} cutoff? Reference E summarizes analyses of ten three-train AFW designs:

Design	Failures/Demand		log(failures/demand)
Summer 1	1.2E-5		-4.92
McGuire	2E-5		-4.70
Comanche Peak	2E-5		-4.70
Diablo Canyon	3.7E-5		-4.43
San Onofre 2&3	2.2E-5		-4.66
SNUPPS	2.0E-5		-4.70
Waterford	1.4E-5		-4.85
Midland	1.0E-5		-5.00
Seabrook	2E-5		-4.70
Catawba	0.7E-5		-5.15
Arithmetic mean:	1.8E-5	Logarithmic mean:	-4.78
Arithmetic std dev:	8.4E-6	Logarithmic std dev:	0.22

These 10 analyses can be considered a statistical sample. The cutoff of 10^{-4} failures per demand is 9.76 standard deviations above the mean on a linear scale and 3.55 standard deviations above the mean on a logarithmic scale. The shape of the distribution is unknown, of course, but we will examine both a normal and a log normal distribution and use the worst case. Based upon these distributions and in the absence of any other information, if another three-train AFW design were evaluated, the probability of this new design being above the cutoff is:

Normal Distribution	essentially zero
Log Normal Distribution	2.0E-4

What this means is that 10 sample designs are all well below the cutoff. Had the sample average been close to just below 10^{-4}, one would be confident of finding a plant or two over the limit. However, the mean is far below the limit (where "far" is defined in terms of the width of the distribution) and the per-plant probability of being over the limit is small.

There are 80 PWRs operating or under construction. Seven of these have two-train AFW systems and are covered by Generic Issue 124. This leaves 73 plants. The probability of detecting one or more of these plants with an AFW unavailability greater than 10^{-4} per demand is:

$$1 - (1 - 2E\text{-}4)^{73} \approx (73)(2E\text{-}4) = 0.014$$

That is, based upon the available knowledge regarding three-train AFW designs and in the absence of other information, a PRA of all three-train AFW systems has only a few percent chance of finding a system that needs upgrading. (This does not mean that these AFW systems are problem free. It does mean that the problems probably will not be found by means of PRA, unless considerably more information is available.)

Frequency Estimate

The sequence of interest is straightforward. It is initiated by a non-recoverable loss of main feedwater. If the AFW system fails and feed-and-bleed techniques fail, core-melt will ensue.

For the initiating event frequency (non-recoverable loss of main feedwater), we will use 0.64 event per reactor-year, based upon the Oconee PRA done by Duke Power Co.[C] This figure is based upon fault tree analysis and should be reasonably representative of most main feedwater system designs.

Next, the change in AFW failure probability must be estimated. We will assume that the AFW system "as is" has an unavailability equal to that of a "typical" two-train AFW system which would be about 6.7E-4 per demand, the average of the seven plants listed in Reference D. The AFW system after upgrading would be at most 1.0E-4. Therefore, the change would be about 5.7E-4.

We will assume a typical value of 0.20 for the failure probability of feed-and-bleed cooling, based upon the calculations presented under Issue 125.II.9, "Enhanced Feed and Bleed Capability." Multiplying these figures out, the change in core-melt frequency is:

$$(0.64/\text{year})(5.7\text{E-}4)(0.20) = 7.3\text{E-}5$$

Consequence Estimate

The core-melt sequence under consideration here involves 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 basemat 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 involved would include administrative charges, the costs of the PRAs, and possibly costs of hardware changes, should they be required. It is not clear at this point whether the PRAs would be done by the licensees or the NRC. In any case, the cost of the PRA of one AFW system is likely to be on the order of $50,000 or more (half a staff-year). For 73 plants, this is $3.65 million. We will not calculate the administrative and hardware costs, but instead will use the $3.65 million as a minimum figure.

Value/Impact Assessment

Because this issue deals with only an expectation value for the number of plants, but does not necessarily expect to affect any specific plant, the per-plant parameters (core-melt per reactor-year and person-rem per reactor) are not meaningful. Instead, the "aggregate" parameters (core-melt per year and total person-rem) are appropriate.

As of March 1988 (the earliest that any changes are likely to be made), the 73 subject plants will have a combined remaining life of 2317.8 calendar-years. At a 75% capacity factor, this works out to an average of 23.8 years of operation remaining per plant. Priority parameters can now be calculated:

Plants to be examined	73
Number of hypothetical plants needing modification (expectation value)	0.014
Remaining lifetime per plant, operational years	23.8
Change in core-melt/year for hypothetical plant	7.3E-5
Change in person-rem/year, for hypothetical plant	11.0
Total cost, all reactors (in millions of dollars)	≥ 3.65

Change in:

core-melt per year, total all reactors	1.0E-6
person-rem, total all reactors	3.7
person-rem/million dollars	≦ 1

Other Considerations

(1) The statistical logic presented above does not rule out specific systems needing attention. The proper conclusion is that, unless more information is forthcoming (for example, specific design or performance problems), a non-specific general search such as this is difficult to justify because there is no specific reason to believe a problem will be found this way, based on past experience.

Also, the continuous distribution assumption implies that design anomalies, such as the single failures of Item 125.II.1.B, have been fixed. This item must not be viewed in isolation.

(2) Issue 124, "AFW System Reliability," in addition to its attention to plants with two-train AFW systems, also is considering whether to require confirmation that the remaining PWRs have AFW system reliabilities that are less than 1.0E-4 per demand. However, Issue 124 has not produced a decision at this time, nor does a decision appear to be forthcoming in the near future. Therefore, this issue cannot be subsumed within Issue 124.

(3) In most cases, the fix will not involve work within radiation fields and thus will not involve occupational radiation exposure.

(4) The occupational exposure averted due to post-feed-and-bleed cleanup and post-core-melt cleanup is a minor consideration. Reference H estimates the occupational exposure associated with cleanup to be 1800 person-rem after a primary coolant spill and 20,000 person-rem after a core-melt accident. If the frequency of feed-and-bleed events is 5E-6/year total, the actuarial cleanup occupational exposure averted is only 0.2 person-rem. Similarly, a total core-melt frequency of 1.0E-6/year corresponds to an actuarial averted cleanup occupational exposure of only 0.5

person-rem. If averted occupational exposure were added to the person-rem per reactor and person-rem per million dollars figures above, no conclusions would change.

(5) The proposed fix would reduce core-melt frequency and the frequency of feed-and-bleed events and therefore would avert 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 5E-6/year and the average remaining lifetime is 31.7 calendar-years at 75% utilization, then 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 $3300. Similarly, the actuarial savings of averted core-melt cleanup (which is assumed to cost one billion dollars if it happens) are about $12,000. The actuarial savings from replacement power after a core-melt up to the end of the plant life are also about $12,000. (This last figure represents the lost capital investment in the plant.) If these theoretical cost savings were subtracted from the expense of the fix, the person-rem per million dollars would not change significantly.

CONCLUSION

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

ITEM 125.II.1.D: AFW STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM/ICS INTERACTIONS IN B&W PLANTS

DESCRIPTION

This issue is centered upon the subject of the reliability of the AFW system[A] which is safety-grade. This item is targeted specifically at B&W plants[A] and would require a re-examination of the AFW system reliability.[D] The reasons given are two-fold. First, assessments made shortly after the TMI accident indicated that the AFW system in B&W plants had (at that time) an unavailability approximately an order of magnitude higher than those in most other PWRs.[D] (This does not account for the subsequent modifications to these AFW systems.) Second, this item calls for explicit attention to the interactions between the

AFW system and the Steam and Feedwater Rupture Control System (SFRCS) and between the AFW system and the Integrated Control System (ICS). Such interactions are important because the initiating transient may well be caused by a problem with the ICS and any possible interactions between the ICS and AFW or SFRCS would be a potential source of a common mode failure, defeating the system needed to mitigate the transient.

PRIORITY DETERMINATION

On the general question of AFW unavailability, the B&W plants have already updated their reliability analyses to reflect the post-TMI modifications.[B] These updates have satisfied the original concern.[F]

The specific issue of the ICS-SFRCS-AFW interactions deserves more discussion. The function of an SFRCS is to control the AFW system. The name (Steam and Feedwater Rupture Control System) is somewhat misleading in that the SFRCS also initiates AFW for loss of main feedwater events. Those plants with an SFRCS should have no interactions between the ICS and the SFRCS or AFW systems.

There are some B&W plants that have used the ICS to control the AFW system. Of these, two plants (Crystal River and ANO-1) have installed an "Emergency Feedwater Initiation and Control (EFIC) System" to replace the ICS as the control system for AFW. (The EFIC system is an improvement over SFRCS in that the EFIC system will not allow both steam generators to be isolated simultaneously. The SFRCS at Davis-Besse has also been modified such that it will no longer allow both steam generators to be isolated simultaneously.) Two plants remain. Of these, Rancho Seco will install an EFIC system at the next refueling outage and Three Mile Island 1 will install a system similar to EFIC, but designed by the licensee, at its next refueling outage.

Under these circumstances, the concern is not with SFRCS-AFW interactions, but instead reduces to ensuring that there is no interaction between the ICS and the AFW or its control system that can cause a common mode failure. For plants with two-train AFW systems, this will be covered by the analyses of Issue 124.[C,F] The remaining plants will be examined under the B&W Reassessment Program which places considerable emphasis on the ICS.[G]

CONCLUSION

This item is covered in Issue 124 and the B&W Reassessment Program and should be DROPPED as a separate issue.

REFERENCES

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

B. Memorandum for H. Thompson from R. Bernero, "Auxiliary Feedwater Systems," August 23, 1985.

C. Memorandum for H. Thompson and T. Speis from R. Bernero, "Request for Comments on Draft CRGR Package with Requirements for Upgrading Auxiliary Feedwater Systems in Certain Operating Plants," October 3, 1985.

D. Memorandum for H. Thompson from G. Edison, "Recommendation for Longer term Generic Action as a Result of Davis-Besse Event of June 9, 1985," September 11, 1985.

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

F. Memorandum for F. Miraglia from G. Edison, "Prioritization of Generic Issue 125.II.1.D," April 25, 1986.

G. BAW-1919, "B&W Owners Group Trip Reduction and Transient Response Improvement Program," May 1986.

H. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission.

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."