UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

F. Cherny

MATT TAYLOR.

JUN 11 1985

MEMORANDUM FOR: Robert M. Bernero, Director
Division of Systems Integration

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

SUBJECT: SCHEDULE FOR RESOLVING AND COMPLETING GENERIC ISSUE NO. 105 - INTERFACING SYSTEMS LOCA AT BWRS

The technical resolution for Generic Issue No. 105, "Interfacing Systems LOCA at BWRs" is assigned a "HIGH" priority ranking. This memorandum approves NRR staff taking appropriate actions necessary to complete this issue. The evaluation of the subject issue is provided in Enclosure 1.

This evaluation assumes that there is no leak testing of the reactor coolant system pressure isolation valves (PIVs) in BWRs. This is consistent with the SRP and the ASME code. However, the Standard Technical Specification, Section 3.4.3.2 of plants licensed since 1980 requires leak testing of all pressure isolation valves every 18 months and after maintenance on the valves. MEB has also been requiring operating plants to test all PIVs as part of the ASME Inservice Testing program review. Thus, some plants have requirements for testing PIVs. The validity of these STS and IST requirements are currently being reviewed. This review may result in some changes to these requirements in the interim while this issue is being resolved. Therefore, the resolution of this issue should consider the results of this review.

Because the overpressure events that are the subject of this issue were caused by personnel errors during maintenance and surveillance, this issue should be closely coordinated with issue HF-02, "Maintenance and Surveillance Program Plan," which is assigned to DHFS. However, the resolution of this issue should consider both the equipment and human-factors-related changes that may be needed to limit the risk. If the human-factors-related changes related to this issue have general applicability, they can be included in the MSPP. Resolution of this issue should not be delayed until the MSPP, for which a schedule has not been determined, is completed.

17pp. 8506200433 X A

This issue is also related to Generic Issue 96, "RHR Suction Valve Testing," which considers the failure of the pressure isolation valves between the reactor coolant system and the residual heat removal system in PWRs. The resolution of this issue should also be coordinated with Generic Issue 96.

In accordance with NRR Office Letter No. 40, "Management of Proposed Generic Issues," 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.

The attached prioritization evaluation will be incorporated into NUREG-0933, "Prioritization of Generic Safety Issues," and is being sent to other NRC 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, DST. Should you have any questions pertaining to the contents of this memorandum, please contact Louis Riani (24563).

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

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

cc: See next page

cc w/o Enclosure 2:
V. Stello
J. Funches
R. Minogue, RES
J. Taylor, IE
C. Heltemes, Jr., AEOD
J. Davis, NMSS
W. Russell, DHFS
F. Rowsome
W. Minners
R. Baer, IE
ACRS
PDR

cc w/Enclosure 2:
K. Pulsipher
B. Sheron
R. Bosnak
F. Cherny
R. Emrit
L. Riani

ENCLOSURE 1

PRIORITIZATION EVALUATION

GENERIC ISSUE NO. 105

"INTERFACING SYSTEMS LOCA AT BWRS"

ISSUE 105: INTERFACING SYSTEMS LOCA AT BOILING WATER REACTORS

DESCRIPTION

Historical Background

Generic Issue B-63, "Isolation of Low Pressure Systems Connected to the Reactor Coolant Boundary," which was resolved and implemented as Multiplant Action B-45, required leak testing of the check valves that isolate those low pressure systems that are connected to the reactor coolant system and outside the containment. However, except for Oyster Creek and Nine Mile Point, these low pressure systems in BWRs are isolated with check valves that have actuators. These actuators are used to test the operability of these valves. This operability test was considered sufficient to assure the integrity of the pressure isolation function and leak testing of pressure isolation valves in BWRs was not required. However, beginning in 1980, the BWR Standard Technical Specifications, Section 3.4.6.2 required the leak testing of all reactor coolant system pressure isolation valves at least once every 18 months and after any work on a valve. This STS requirement was also applied to operating plants as they submitted their inservice testing program for review.

Recent BWR operating experience indicates that the isolation valves between the reactor coolant system and low pressure interfacing systems (including related test and maintenance requirements) may not adequately protect against overpressurization of low pressure systems. There have been three reported failures of the boundary between the reactor coolant system and low pressure injection systems in approximately 200 BWR-years of operation.[A] Two of the events (Vermont Yankee - 12/12/75 and Browns Ferry 1 - 8/14/84) were the result of maintenance errors which left the testable isolation check valve in the open position. The third (Pilgrim - 9/29/83) was the result of personnel errors (improper combination of surveillance tests) and a stuck open failure of an isolation check valve. In all three of these cases, there was a degradation of the pressure isolation valves due to personnel errors. None of these plants were required to leak test pressure isolation valves.

This issue, which is limited to pressure isolation valves in BWRs, is related to Generic Issue 96, "RHR Suction Valve Testing," which considers the failure of the pressure isolation valves between the reactor coolant system and the residual heat removal system in PWRs.

Safety Significance

Overpressurization of low pressure piping systems due to RCS boundary isolation failure could result in rupture of the low pressure piping. This, if combined with failures in the emergency coolant injection (ECI) and/or the decay heat removal (DHR) systems, would result in a core-melt accident with an energetic release outside the containment building, causing significant off-site radiation release. The Standard Technical Specifications require leak testing of pressure isolation valves at least after every refueling and in some cases more frequently. Therefore, this issue applies to BWRs licensed before 1980.

Possible Solution

For the purpose of this evaluation, it is assumed that the frequency of low pressure system overpressurization events will be reduced by instigating a more rigorous revised inspection program (follow specific test and post-maintenance procedures, conduct surveillance tests one at a time, performing leak tests after operability demonstrations or flow tests) and making minor hardware modifications such as modifications to testable check valve air supply lines to precluding interchanging the lines (different threads, different size connectors, color coding, labeling). Major system hardware changes are not anticipated.

Affected Plants

Operating BWRs which have RCS/RHR system interface configurations similar to Hatch Unit 2 have been identified and include: Duane Arnold, Brunswick 1

and 2, Cooper, Dresden 2 and 3, Hatch 1, Fitzpatrick, Monticello, Peach Bottom 2 and 3 Pilgrim, and Quad Cities 1 and 2.[B] Browns Ferry 1 also experienced a similar isolation boundary problem. Therefore, the list of affected plants utilized in this analysis also includes BWR 3 and 4 class operating plants (i.e., Millstone, Browns Ferry 1, 2 and 3 Vermont Yankee). Therefore, the total number of potentially affected operating BWRs considered in this analysis is 20 with an average remaining life of 26 years.

PRIORITY DETERMINATION

The prioritization of this issue is based on analysis performed by PNL.[C]

Frequency/Consequence Estimate

Since this generic issue applies only to BWR plants, the Browns Ferry, Unit 1, IREP probabilistic risk assessment (PRA)[D] was used in the estimation of public risk reduction. The general approach was to use available historical data for failure of the high pressure/low pressure isolation boundary and a probability estimate for piping failure due to overpressurization to modify the appropriate LOCA sequences from the Browns Ferry PRA. These modified appropriate (affected) LOCA sequences are then assumed to represent the current (base case) level of plant risk associated with this issue. Specifically, the event Ls, large break LOCA, from the Browns Ferry PRA is redefined as the product of the probability of failure of the high pressure/low pressure isolation boundary and the probability of failure of the low pressure piping as a result of overpressurization. From the historical

---

*In NUREG-0677[F], a probability of BWR intersystem LOCA of 6.2E-4/ry was calculated. No contribution from maintenance and operator errors was included in deriving the above frequency of BWR intersystem LOCA. The BWR intersystem LOCA frequency derived for this prioritization analysis (1.5E-3/ry) which is based on recent LERs is predominated by operator and maintenance errors and appears to be an expected value when compared to the value derived in NUREG-0677.

data (3 isolation boundary failures in about 200 BWR plant years) a probability of failure of the isolation barrier of 1.5E-2/ry is estimated. Analysis of the low pressure piping reveals that the hoop stress in the low pressure piping would not be expected to exceed the yield value for the piping. Thus, failure of the low pressure piping was assumed to be likely only in the presence of a significant crack in the piping. Using data available on intergranular stress corrosion cracking (IGSCC), estimates of the number of piping welds in the low pressure piping systems, and estimates of the distribution of depth of cracks (percent of thru wall) from existing pipe crack data, PNL estimates the conditional probability of an intersystem LOCA, via the pipe cracking scenerio, of 1.0E-1/event given an overpressurization of the low pressure piping. This gives a new estimate of Ls of 1.5E-3/ry*, as opposed to the value of Ls derived in the Browns Ferry PRA (3E-8/ry). When this new value of Ls (1.5E-3ry) is input to the affected core melt minimal cutsets in the Browns Ferry PRA, a base case core melt frequency due to isolation boundary failures is calculated to be 6.31E-06. The effect of accidents resulting in direct core melt releases outside containment is assumed to be best estimated by the BWR release Category 2. When the dose conversion factor for BWR Category 2 events (7.1E+6 MR/event) is multiplied by the base case core melt frequency, a public risk of 44.7 man-rem/reactor year is calculated.

Implementation of the assumed resolution for this issue is assumed to reduce the core melt frequency and public risk due to overpressurization and failure of low pressure systems connecting to the RCS to those values calculated from the Browns Ferry PRA (i.e., 1.22E-10 events/ry and 8.66E-4 man-rem/ry, respectively). Therefore, implementation of the assumed resolution of this issue is estimated to result in a reduction in core melt frequency of 6.3E-6 and a reduction of public risk of 44.7 man-rem/ry. The total public risk reduction for the 20 affected plants over their 26 year average remaining lifetime is calculated to be 2.3E+4 man-rem.

It should be noted that the probability of intersystem LOCA may well be greater than that calculated above based on piping failure. Other components in low pressure systems, such as pump seals, heat exchanger tubes, thermocouple wells, etc., would also be subject to overpressure failures. Also, while not explicitly considered in calculating the estimated core melt frequency and risk, the failure of all low pressure systems due to overpressure resulting from failure of pressure isolation valves contributes further to the risk. Although the risk from other interfaces has not been calculated, the evaluation of Generic Issue 96 shows that the risk from failures of the valves isolating the RHR system in a PWR is at least an order of magnitude less than the risk calculated for this issue. The failure of the pressure isolation valves in a BWR RHR system would affect only part of the ECCS system, rather than all as in a PWR. Therefore, the risk in a BWR would be even less than in a PWR.

In addition, LOCA releases in the auxiliary building would also be expected to present an additional common mode failure mechanism for failure of redundant safety systems located in the auxiliary building. These considerations could not be included within the scope of the limited efforts performed for a prioritization analyses. However, were they to be included, we would expect the estimate of frequency for intersystem LOCA and resultant core melt to be greater. For that reason, we believe that the priority conclusion reached on the basis of the simplified analysis performed for this generic issue is conservative.

Cost Estimate

Resolution of the issue is assumed to result in improved surveillance, maintenance and test procedures, and minor modifications to make the air actuation system for testable check valves "fool proof."

NRC Cost: It is assumed that resolution of this issue will require five staff months of technical effort and technical contract support for a more precise probabilistic risk assessment, for a total resolution cost of about \$100,000. It was assumed that NRC staff (NRR and I&E) review of licensee implementation of the assumed resolution of the issue would require 5 staff weeks/plant for a cost of about \$230,000. Resident inspector surveillance of site actions emanating from the resolution of this issue are estimated to require 0.5 staff weeks/ry for a present worth of about \$325,000 over the remaining lifetime of the 20 affected BWRs. The total present worth of NRC cost for this issue is thus estimated to be about \$650,000.

Industry Cost: Implementation of the assumed resolution of this issue is estimated to require about 4 man-weeks per plant for revision of surveillance, maintenance and test procedures, and installation of "fool proof" features on the testable check valve actuation system, plus about \$2,500 per plant for materials (connectors, tags, etc.). Thus, an implementation cost of \$220,000 is estimated. Increased surveillance testing, reduction of allowable concurrent testing and improved post-maintenance inspection procedures are estimated to increase plant maintenance and surveillance efforts by 40 man-hours/ry. Thus, the present worth of the increase in plant operation and maintenance costs for the 20 affected plants over their remaining lifetime is calculated to be about \$650,000. Total industry cost for resolution (and implementation) of this issue is therefore estimated to be about \$875,000.

Total NRC and industry costs for resolution and implementation of this issue are thus estimated to be \$1,525,000.

Value/Impact Assessment

Based on a total public risk of 2.3×10^4 man-rem and a total cost of \$1.5 million, the value/impact score is given by:

$$S = \frac{2.3E+4 \text{ man-rem}}{\$1.5M}$$

$$S = 15{,}000 \text{ man-rem/\$M}$$

Other Considerations

A relatively small total increase in operator exposure (ORE) (530 man-rem) is calculated due to assumed increases in surveillance and post maintenance inspections. The calculation assumes 40 man-hours/RY for increased maintenance in a 25 millirem/hr field at the 20 affected BWRs for their remaining lifetime. Reduction in the estimated frequency of core melt and non-core melt intersystem LOCA which might be attained is calculated to result in a total averted operator exposure of 215 man-rem: 65 man-rem due to clean up of core melt events and 150 man-rem due to clean up of non-core melt intersystem LOCAs. Both the increased ORE and the averted operator exposure are insignificant in comparison to the calculated public risk reduction of 2.3E+4 man-rem, and would not alter the recommendations indicated by the value/impact assessment.

At an estimated industry clean up and replacement power costs of $1.65 Billion for a core melt accident and $720 million for a successfully mitigated LOCA, the frequency reduction of core melt and non-core melt intersystem LOCA estimated for resolution of this issue would result in an averted accident cost savings with a present worth of about $2,700,000. This exceeds the total expected NRC and industry cost for resolution of the issue, and would therefore lend support for a decision to pursue resolution of the issue.

CONCLUSION

Significant reduction in public risk reduction and reduction in the frequency of core melt accidents is calculated for the resolution of Generic Issue 105. The analysis indicates that this reduction may be achieved at a relatively

small cost to the NRC and the industry, resulting in a very favorable value impact ratio (15,000 man rem/$M). If the averted cost of the cleanup of intersystem LOCAs is included, the net impact is a cost saving. Therefore, we recommend that the resolution of Generic Issue 105 be pursued with a HIGH priority.

REFERENCES

A. Memorandum for W. Minners from G. Holohan, "Prioritization of Interfacing System LOCA at Boiling Water Reactors," October 25, 1984.

B. AEOD Engineering Evaluation Report No. AEOD/E414, "Stuck Open Check Valve on Residual Heat Removal System at Hatch Unit 2," U.S. Nuclear Regulatory Commission, May 31, 1984.

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

D. NUREG/CR-2802, "Interim Reliability Evaluation Program: Analysis of the Browns Ferry Nuclear Plant, Unit 1," Appendices B and C.

E. NUREG-0933, "A Prioritization of Generic Safety Issues," U.S. Nuclear Regulatory Commission, July 1984.

F. NUREG-0677, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," M. R. Rubin, U.S. Nuclear Regulatory Commission, May 1980.

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 needed 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							